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DRIVEN-CURRENT TOKAMAK (DCT) SCOPING STUDY*

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ABSTRACT

The present Department of Energy (DOE) plan calls for the construction of an Engineering Test Reactor (ETR) that is to be the last major experimental fusion device prior to the commercialization of fusion power. The plasma driver of the ETR is to be either a long-pulse tokamak or a tandem mirror machine. The possibility of using the Tokamak Fusion Test Reactor (TFTR) facility to consolidate the physics and technology database for the tokamak version of the ETR has been considered. This paper addresses two of the options being considered: (1) a superconducting toroidal field (TF) coil-hydrogen plasma alternative and (2) a superconducting or hybrid TF coil-high Q alternative. Both options assume essentially steady-state operation through the application of rf current drive. The options are evaluated on the basis of performance and cost determined by application of the Fusion Engineering Design Center (FEDC) Tokamak System Code.

INTRODUCTION

In conjunction with the Princeton Plasma Physics Laboratory (PPPL), several tokamak configurations were evaluated using the FEDC System Code. Self-consistent tokamak parameters and the associated direct capital costs were determined for Driven Current Tokamaks (DCT) as candidate plasma drivers for the ETR. The study was conducted for devices that considered hydrogen plasma operation only; for devices sized for D-T ignited plasmas employing superconducting TF coils; and for devices sized for D-T plasmas utilizing 10-T superconducting TF coils for long-pulse operation, augmented by copper inserts for ignited short-pulse operation.

All DCT configurations considered in this study assume steady-state operation achieved by RF current drive. A reactor-relevant tokamak configuration (FED/INTOR type) was maintained throughout these studies that included provisions for (1) modular torus components and (2) a single vacuum boundary for the superconducting TF coils and the plasma. Parameters and costs of the three classes of DCT devices considered in this study are discussed.

HYDROGEN PLASMA TOKAMAKS

The parameters and cost for each of the first five cases considered are presented in Tables 1 and 2. These devices are all sized for a maximum TF field of 8 T and are considered for hydrogen plasma operation only. Sixteen TF coils are used and are sized to be consistent with a 1.5% peak-to-average ripple at the plasma edge. The five cases range in size from a Princeton Large Torus (PLT) size plasma in Case 1 up to a device with a plasma sized to give ignition if operated in D-T in Case 5. Case 2 parameters are consistent with the Large Coil Program (LCP) size coils. Case 3 is the same as Case 2 except that a poloidal divertor is used instead of a pumped limiter impurity control system. The difference in cost of Cases 2 and 3 is minimal. Case 4 is sized for 120% of LCP size coils. The major radius (and aspect ratio) of each device is selected to provide a poloidal field (PF) system capable of supplying adequate flux linkage to the plasma for inductive current startup. The values of plasma temperature and beta were selected as a function of plasma size in order to obtain values of plasma auxiliary heating power which (1) produce essentially a constant value of average thermal loading (MW/m^2) and (2) does not exceed the available space for placing injectors around the torus. Note from Table 1 that as the minor radius increases from 0.45 to 1.2 m the plasma temperature is increased from 6 to 10 keV and beta is increased from 3 to 5%. The plasma heating power increases from 18 to 63 MW, but the

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average thermal wall loading is maintained at about 0.3 MW/m^2. The capital cost of these five devices, tabulated in Table 1, indicates that cost is approximately a linear function of size for hydrogen plasma tokamaks.

The DCT-2 is designated as the tokamak consistent with LCP size TF coils and would be a strong candidate if the option to develop the nonnuclear technology and physics required for a minimum cost ETR machine were pursued. The direct captial cost of DCT-2 is estimated to be $317 million; however, this could be reduced by taking into account the existing equipment and buildings at the TFTR site to $208 million. See Table 3 for a list of credits for using existing equipment at the TFTR site. This cost is in 1981 dollars and does not include indirect costs (engineering, installation, and contingency) or escalation.

D-T IGNITION SIZE TOKAMAKS USING 10 TESLA SUPERCONDUCTING TF COILS

DCT-6 was sized to provide ignition and to accomodate sufficient shielding (0.6 m inboard, 0.9 m outboard) for D-T operation. A minor radius of 1.07 m was required as shown in the parameters presented in Table 4. Twelve 10 T TF coils are employed, sized for a 1.5% peak-to-average ripple at the plasma edge, to enhance torus sector removal. Again, the major radius is chosen to satisfy inductive plasma current startup requirements. The capital cost for DCT-6 is shown in Table 5 and is estimated to be $677 million.

A Hydrogen plasma version of DCT-6 was also considered, and the resulting cost information is provided in Table 5. Hydrogen plasma operation requires an increase in rf heating power to 74 MW in order to sustain the plasma at 10 keV in the absence of alpha heating. The costs of the hydrogen plasma versions reflect the increased costs of heating power and ac power and the decreased costs for shielding (bulk, rf shielding, vacuum duct), tritium processing equipment, hot cells, maintenance equipment, and heat transport equipment. The relative cost of a device such as DCT-6 sized for hydrogen as compared with the same device sized for D-T results in approximately a 20% reduction or a difference in direct capital cost of approximately $150 million.

D-T IGNITION SIZE TOKAMAKS USING HYBRID TF COILS

The next four cases (DCT-7A, 7B, 7C, and 8) were sized for ignition conditions and employed 10 T superconducting TF coils augmented by water-cooled copper inserts. The copper serves a dual function: it supplies magnetic field on axis and it acts as shielding to attenuate neutrons. This magnetic field arrangement allows a reduction in plasma minor radius since the TF coil is effectively moved closer to the plasma and therefore delivers a higher field on-axis for the same maximum TF coil field. Two basic configurations were considered: (1) the copper insert inside the TF coil cryostat adjacent to the superconducting coil and (2) a copper insert outside the TF coil cryostat as part of the stainless steel shield. Two operating scenarios are envisioned: a long-pulse (approximately steady-state) driven mode operation, utilizing only the superconducting TF coil, and a shorter pulse (~100-s) ignited operation using field augmentation by the copper insert coils. A 20-s current ramp is assumed for the copper insert for all configurations considered in this study. These devices combine the long-pulse objective of a superconducting tokamak with the nuclear technology and physics objectives of a D-T burning experiment. Several tokamak configurations were evaluated for this option using the FEDC system code; among these are (1) the location of the copper coil (in the cryostat or in the shield), (2) the inclusion or omission of a bucking cylinder, and (3) the plasma current startup scenario (by purely inductive means or by rf assist).

Hybrid TF Coil With Copper Insert In The Cryostat

DCT-7A and DCT-7B are TF coil configurations with the copper insert coils inside the TF cryostat adjacent to the TF coil. Case DCT-7A parameters and cost summaries are also presented in Tables 4 and 5, respectively. Note that a 0.73 minor radius and an on-axis field of 7.3 T (5.5 T on-axis supplied by the 10 T superconducting coil and 1.8 T on-axis from the copper insert coils) are consistent with ignition condition and 290 MW of fusion power. A major radius of 3.6 m is required. The copper insert has a thickness of ~20 cm. A total of ~140 MW of resistive losses are encountered during the 100-s burn pulse when the water-cooled copper insert coils are energized. For the superconducting mode of operation only, 95 MW of fusion power are

generated and a value for Q of 3.0 is obtained. The capital cost for DCT-7A was estimated based on the following optimistic assumptions: (1) no bucking cylinder is required—the TF forces are restrained by wedging action plus the use of support rings; (2) a forced-flow solenoid is used, as opposed to a pool boiled design, which allows a saving of 0.10 m in the radial build. The copper insert will provide added protection to the outboard leg of the TF coil, but additional shielding may be needed between the TF coils to provide biological protection, perhaps through the use of an igloo.

Case DCT-7B is similar to DCT-7A except assumptions 1 and 2 above have not been made; that is, a bucking cylinder is used and a pool-boil solenoid is used. The parameters and cost for DCT-7B are included in Tables 4 and 5. The effect of retaining the bucking cylinder and pool-boil solenoid is to increase the major radius from 3.6 to 4.2 m, the minor radius from 73 to 78 cm, and the capital cost from $597 to $646 million. The TF coils for both DCT-7A and DCT-7B are access limited, with 20 cm of space allowed between the outboard shield and TF coil outer leg.

Hybrid TF Coil With The Copper Insert In The Bulk Shield

Cases DCT-7C and DCT-8 are for a configuration with the copper insert coil located coincident with the stainless steel shield. In addition to the changes associated with the copper insert location, it was decided (in collaboration with PPPL) to make some modifications to the physics parameters relative to those of case 7B based on a different functional relationship between plasma current, the beta limit, and confinement time. The new functional relations are (1) $\tau_e \propto a \times I_p$ and (2) $\beta \propto (1+\kappa^2)/(q_\psi \times A)$ where τ_e is energy confinement time, a is the plasma minor radius, I_p is the plasma current, β is the beta limit, κ is the plasma elongation, q_ψ is the safety factor at the plasma edge, and A is the aspect ratio. Physics parameters changes included (1) an increase in the plasma current to 6 MA, (2) a reduction of the safety factor q_ψ to a value of 2.0, (3) a value of beta of 3.3%, and (4) a round up of the plasma minor radius from 0.78 to 0.8 m. For case 7C, these values result in derived parameters of $\epsilon\beta_p$ of only 0.3 (as opposed to previously used values of 0.5) and an ignition margin of 15% (as opposed to zero margin for previous cases), based on the FEDC Physics System Code model. The magnetic field on-axis is maintained at 7.5 T (5.7 T contribution from the superconducting TF coil and 1.8 T contribution from the copper insert coil). Moving the copper insert closer to the plasma results in a relatively high value of ripple, an approximately 2.5% peak-to-average ripple at the plasma edge, for 16 TF coils and copper inserts. The capital cost of DCT-7C is essentially the same as for Case 7B: $654 million compared with $646 million for Case 7B, with the slight increase of Case 7C attributed to the rf and PF electrical system associated with the higher plasma current (6.0 MA as opposed to 4.2 MA). The higher plasma current requires additional flux linkage for inductive startup. However, the major radius was not increased from the 4.2 m value of Case 7B, although rf-assisted startup was assumed to provide the equivalent shortfall of 15 Wb. Performance and cost summaries for Case 7C are included in Tables 4 and 5.

The following changes were implemented for DCT-8, relative to DCT-7C, in an effort to reduce cost: (1) to reduce the major radius to 3.6 m, (2) to reduce the field on-axis to 7.0 T, and (3) to assume additional rf current drive during startup (27 Wb). In addition, it was decided to substitute 10 MW of Lower Hybrid Resonance Heating (LHRH) for 10 MW of Ion Cyclotron Resonance Heating (ICRH) to provide for the current drive requirement, which had the effect of increasing plasma heating cost by approximately $25 million. Derived parameters for DCT-8 shows a 20% ignition margin, 360 MW of fusion power, a value of Q = 3 during long-pulse superconducting operation, and a maximum ripple at the plasma edge of 1.7%. Resistive losses in the copper insert coils are 137 MW during the 100-s high field pulse. The capital cost of DCT-8 is $631 million at a new site. If this device were to be located at the TFTR site, a credit of $109 million estimated for existing buildings and equipment, reducing the direct capital investment to $522 million.

The costs of the copper insert device (Cases DCT-7A — DCT-8) are compared with DCT-6, a superconducting TF coil tokamak at 10 T magnetic field strength, in Table 6. Changes in major assumptions as one progresses from Case 6 to Case 8 are also noted.

CONCLUSIONS

- The cost of DCT devices sized for hydrogen plasma operation only varies almost linearly with major radius (DCT-1 — DCT-5). DCT-2 (with LCP size TF coils) would be a strong candidate if the option to develop the nonnuclear technology and physics option of ETR were pursued.

- The cost reduction in direct capital cost of a tokamak sized for hydrogen operation only is approximately $150 million or approximately 20%, relative to a device sized for D-T operation (DCT-6-H_2 versus DCT-6-D-T).

- Employing a copper insert coil (which also serves as a neutron shield) in conjunction with a 10 T superconducting coil (to boost the peak TF field to 12 T) allows a smaller device and a cost reduction of approximately 5% relative to an all superconducting TF tokamak sized for a 10 T peak field. Marginal ignition and the capability for a full inductive current startup are common to both machines. The hybrid TF coil configuration would demonstrate an even greater cost advantage (about 15-20%) if the all superconducting machine were also required to produce a 12 T peak field.

- DCT-8 was selected as the reference copper insert device for the high Q option of the ETR. The copper insert is located in the shield and rf-assisted startup is assumed.

- Of the options considered in this study, the superconducting coil-high Q option appears to be the most cost-efficient approach and the best suited to supply the necessary physics and technology database for the tokamak ETR.

Table 1. Superconducting long-pulse hydrogen devices parameters[a]

PARAMETER	Case 1 (PLT Size Plasma)	Case 2 (LCP Size Coils)	Case 3 (Poloidal Divertor)	Case 4 (1.2 x LCP)	Case 5 (Ignition Size Plasma)
Major radius, R_o (m)	2.27	2.92	2.92	3.28	3.89
Plasma radius, a (m)	0.65	0.77	0.77	0.92	1.20
Aspect ratio, A	3	3.8	3.8	3.6	3.2
Ion temperature, T (keV)	6	8	8	10	10
Average beta, β (%)	2	3	3	5	5
Poloidal beta divided by aspect ratio, $\epsilon\beta_p$	0.6	0.3	0.3	0.4	0.4
Safety factor, q_ψ	2.7	2.7	2.7	2.7	2.7
Plasma current, I (MA)	3.8	6	6	5	7.2
Field swing, B_τ (T)	3	6.7	4.7	4.6	4.4
Heating power, P_{heat} (MW)	10	33	33	51	63
Thermal load (MW/m^2)	0.33	0.20	0.28	0.32	0.26
Volt-second needed to achieve full plasma current (V · s)	13	24	34	46	76

[a]Common parameters include the following:

Elongation, κ = 1.6
Maximum field, B_{max} = 8 T
Peak/average ripple = 1.5%
Number of TF coils = 16

Table 2. Cost summary for Cases 1-5 (millions of dollars)[a]

	Case 1 (PLT Size Plasma)	Case 2 (LCP Size Coils)	Case 3 (Poloidal Divertor)	Case 4 (1.2 x LCP)	Case 5 (Ignition Size Plasma)
Magnets	28.4	57.0	60.4	75.4	116.0
Torus	3.0	4.7	4.7	5.8	7.9
Cooling system	9.2	11.0	11.0	12.8	14.4
Tritium & fueling	2.0	2.0	2.0	2.0	2.0
Plasma heating	33.0	36.1	36.1	80.2	89.2
Electrical	25.0	37.0	41.1	47.5	74.8
Vacuum pumping	1.0	1.0	1.0	1.0	1.0
I & C	67.0	67.0	67.0	67.0	67.0
Maintenance equipment	19.0	23.0	23.0	25.0	26.0
Facilities	50.0	58.0	58.0	62.8	71.9
TOTAL	237.6	316.8	324.3	379.5	470.2

[a]Direct cost in 1982 dollars; does not include indirect costs (engineering, installation, contingency) nor escalation.

Table 3. Estimated direct capital cost credits from siting at TFTR

	Credit TFTR site (Row cell)
Magnets	0
Torus	0
Cooling	0
Heating	10
Fueling	5
Electrical	20
Vacuum pumping	0
Diagnostics & I&C	25
Maintenance	2
Facilities	30
	109

Table 4. Tokamak parameters (DCT-6 – DCT-8)

Parameter	DCT-6	DCT-7A	DCT-7B	DCT-7C	DCT-8
Geometry					
Major radius, R (m)	4.64	3.53	4.21	4.2	3.6
Minor radius, a (m)	1.07	0.73	0.78	0.8	0.8
Scrape-off layer (m)	0.10	0.10	0.10	0.1	0.1
Plasma elongation, κ	1.6	1.6	1.6	1.6	1.6
Aspect ratio, A	4.26	4.81	5.4	5.25	4.5
Plasma					
Plasma temperature, T (keV)	10	10	10	10	10
Electron density, $n_e = 10^{20}$	1.5	2.2	2.1	2.2	2.2
Safety factor, q_ψ	2.7	2.7	2.7	2.0	2.25
Plasma current (MA)	5.5	4.3	4.2	6.0	6.0
Ripple (peak to average), edge (%)	1.5	0.5	1.0	2.5	1.75
Total beta, β (%)	4.1	3.6	3.1	3.3	3.9
Beta poloidal/aspect ratio, $\epsilon\beta_p$	0.5	0.5	0.5	0.28	0.33
Field on-axis, B_T (T)	5.0	7.3	7.5	7.5	7.0
Auxiliary heating (MW)	0	0	0	0	0
Q	∞	∞	∞	∞	∞
Fusion power (MW)	370	390	345	395	360
Neutron wall loading (MW/m²)	1.1	1.7	1.6	1.8	1.9
Injected power @ 10 T startup (MW)	37	24	37	35	33
Shield					
Inboard thickness (m)	0.6	0.3	0.4	0.6	0.6
Outboard thickness (m)	0.9	0.9	0.9	0.9	0.9
TF Coils					
Number of TF coils	16	16	16	16	16
Maximum TF field (T)	10	12	12	11.3	11.3
TF field at superconductor (T)	10	10	10	10	10
Current density, J_{TF}, S. C. (a/cm²)	2200	2200	2200	2200	2200
Ampere turns, S. C. (MAT)	130	99	122	120	92
Ampere turns, copper (MAT)	-	11	26	37	34
Current density, copper (a/cm²)	-	1400	1400	1400	1400
Copper insert resistive loss (MW)	-	140	170	140	140
PF Coils					
Total Volt-Second from PF (VS)	85	49.3	59.5	69	40
Volt-Second required for startup (VS)	70	49.3	59.5	84	67
Total PF amp turns (MAT)	32	40.0	40.2	44	50

Table 5. Capital cost summary (in millions of dollars)[a]
(DCT-6 – DCT-8)

	DCT-6 D2	DCT-6 D-T	DCT-7A D-T	DCT-7B D-T	DCT-7C D-T	DCT-8 D-T
Magnets	152.2	152.2	110.0	129.5	132.7	113.6
Torus	0.3	66.3	43.5	50.0	49.8	45.1
Cooling system	16.4	34.5	26.7	29.2	31.8	28.2
Tritium & fueling	2.0	54.0	54.0	54.0	54.0	54.0
Plasma heating	111.0	58.0	65.0	72.5	69.5	88.0
Electrical	65.9	58.6	55.7	63.3	68.7	58.3
Vacuum pumping	1.0	1.0	1.0	1.0	1.0	1.0
I&C	67.0	67.0	67.0	67.0	67.0	67.0
Maintenance equipment	26.0	60.0	60.0	60.0	60.0	60.0
Facilities	74.9	126.0	113.8	119.4	119.2	115.7
Total	524.7	677.4	596.7	645.9	653.7	630.9

[a]Direct capital cost only. Engineering, installation, contingency, escalation not included.

Table 6. Evolution of high Q DCT scoping study

DCT-6	DCT-7A	DCT-7B	DCT-7C	DCT-8
16 superconducting TF coils	16 superconducting TF coils 16 copper-in-cryostat inserts	→	16 copper-in-shield inserts	→
0.37 m access between shield & TF coil	0.2 m access between shield & TF coil	→	→	→
1.5% ripple at plasma edge	0.5% ripple	1% ripple	2.5% ripple	1.7% ripple
Full inductive current start-up	→	→	RF assisted start-up (15 webers)	RF assisted start-up (27 webers)
0.9-m outboard bulk shielding	→	→	→	→
Bucking cylinder	No bucking cylinder	Bucking cylinder	→	→
Pool boil OH solenoid	Forced flow OH solenoid	Pool boil OH solenoid	→	Forced flow solenoid
No LHRH	→	→	→	10 MW LHRH
R_0 = 4.65 m	R_0 = 3.6 m	R_0 = 4.2 m	→	R_0 = 3.6
B_T = 5.6 T	B_T = 7.5 T	B_T = 7.5 T	→	B_T = 7.0 T
I_p = 5.5 MA	I_p = 6.5 MA	I_p = 6.2 MA	I_p = 6.0 MA	→
TF_{max} = 10	TF_{max} = 12	TF_{max} = 12	TF_{max} = 11.5	TF_{max} = 11.5
Ignition margin = 1	Ignition margin = 1.0	Ignition margin = 1	Ignition margin = 1.15	Ignition margin = 1.2
EPRI scaling	EPRI scaling	EPRI scaling	New scaling	New scaling
$577 million	$597 million	$546 million	$554 million	$631 million