A SUPERCONDUCTING TOKAMAK FUSION TRANSMUTATION OF WASTE REACTOR

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1. INTRODUCTION

We are developing a Fusion Transmutation of Waste Reactor (FTWR) concept—a sub-critical, metal fuel, liquid metal cooled fast reactor driven by a tokamak DT fusion neutron source. An emphasis is placed on using nuclear, separation/processing and fusion technologies that either exist or are at an advanced state of development and on using plasma physics parameters that are supported by the existing database.

We have previously discussed the general capabilities of DT tokamak neutron sources for driving transmutation reactors [1] and developed a design concept for a FTWR [2] based on normal conducting magnets. The concept has been further developed in papers dealing with nuclear design and safety [3] and with the evaluation of the potential impact on radioactive waste management [4].

The purpose of this paper is to examine how the FTWR design concept would change if superconducting magnets were used.

2. SUPERCONDUCTING MAGNETS

The primary FTWR design objective was to minimize the overall size of the reactor. Therefore regular conducting Oxygen Free High Conductivity Copper (OFHC) magnets were selected in the initial design. However, it became apparent that the ohmic heating losses associated with these magnets were too large, even when operating at liquid nitrogen temperature. Therefore, superconducting magnets are now being investigated to avoid these ohmic heating problems. The new objective is to minimize the overall size of the reactor while employing superconducting magnets, satisfying all other physics requirements and remaining within the current physics database.

The magnet design parameters of the ITER-FEAT design [5,6] were adapted for the new FTWR-SC design concept (Table 1). The central solenoid (CS) has a flux core of 1.1 m, a radial thickness of 0.77 m and a maximum field of 13.5 T. The (18) toroidal field coils have a radial thickness of 0.91 m, a bore of 3.8 m and a maximum field of 11.8 T. The superconductor is Nb_3Sn and the insulator is C / SiO_2. The poloidal coils employ NbTi as the superconducting material. Prototypes of both coil systems have been tested in the ITERR R&D program. Detailed stress analyses were performed for these magnet systems in the ITER design, and we have checked with simple calculations [2] that tensile stresses are within the ASME limits for each of the magnet systems.

Table 1: Superconducing Magnet Parameters

	TF coil	CS coil
Conductor	Nb_3Sn	Nb_3Sn
Coolant	**Supercritcal Helium**	**Supercritcal Helium**
Structure	**Stainless Steel 316**	**Incoloy 908**
Insulators	C / SiO_2	C / SiO_2
Cross-Sectional Area	$0.83\ m^2$	$2.45\ m^2$
Coolant Temperature	**5 K**	**4.7 K**
Field @ Conductor	**11.8 T**	**13.5 T**
ASME Allowable S_m	**193.33 MPa**	**193.33 MPa**
Tensile Stress	**110 MPa**	**149 MPa**

3. SHIELD

The other major dimensional change introduced by the use of superconducting magnets is the increased shielding required to protect the magnets from neutron damage. The magnets must be shielded to protect against radiation damage and heating effects of the fusion neutrons, fission neutrons, and secondary gammas. Several different shield compositions were investigated (Table 2), and many of them satisfied the overall dose requirements at similar thicknesses (50-85 cm). The selected shield, is composed of $W/ZrD_2/B_4C/Pb$ with 10% coolant ($Li_{17}Pb_{83}$) and 10% steel in all regions except for Pb. This combination of materials has been investigated in similar applications [7]. The dose requirement can be satisfied with a shield thickness as small as 54 cm for the reference composition, however a thickness of 65 cm was selected to provide margin for uncertainty. A thickness of 0.65 m was found to provide adequate shielding for a 40 FPY lifetime with a fast neutron dose limit of 10^9 rads. The overall dose at 65 cm is 1.37×10^8 rads.

Additionally, we allowed an extra 10 cm for gaps or additional shielding on the inboard. Since the plasma is shifted outward, we did not otherwise allow for a gap between the plasma in our model and the wall on the inboard side.

Table 2: Shielding Tradeoff / Study

Shield composition	Coolant	Coolant / Steel Percent	Minimum length to satisfy 10^9 rad dose requirement (cm)
$HT9 / B_4C$ (FTWR design)	$Li_{17}Pb_{83}$	**10% / 10%**	**82.5**
W / Pb	$Li_{17}Pb_{83}$	**10% / 10% (in all regions except Pb)**	**59.8**
$W / ZrD_2 / B_4C / Pb$	$Li_{17}Pb_{83}$	**10% / 10% (in all regions except Pb)**	***54***
$W / B_4C / Pb$	$Li_{17}Pb_{83}$	**10% / 10% (in all regions except Pb)**	**63**

4. RADIAL BUILD

Allowing 0.9 m for the plasma radius and 0.17 m for inboard scrape-off layer plus vacuum vessel plus gaps, 0.65 m for the reflector-shield and 2.85 m for the magnet system results in an increase of major radius from 3.1 m in FTWR to 4.5 m in FTWR-SC. The increased aspect ratio of 5.0 is similar to that of the ARIES-I Tokamak Reactor [8]. The overall radial build (Figure 1) of the reactor has changed significantly between the FTWR and FTWR-SC designs.

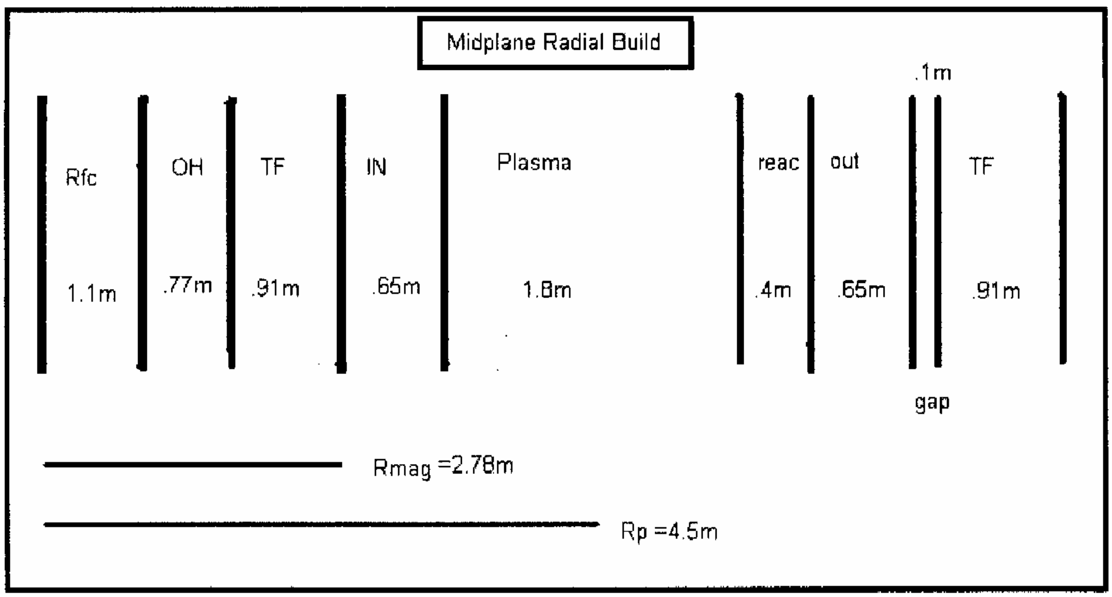

Figure 1: Radial Build of FTWR-SC
(R_{fc}= flux core, OH=OH coil, TF=Toridal Field coil, IN=Inner shield, Reac=Reactor, Out=Outer shield)

5. PERFORMANCE PARAMETERS

The power output P_{th} and transmutation rate (TR) of the FTWR-SC can be scaled from the FTWR values since $P_{th} \sim TR \sim P_{fus}/(1-k)$, where $k = k_\infty (1-L(1-R))$ is the neutron source multiplication factor. The same composition and height and width of the annular core are specified for the FTWR-SC and FTWR, so k_∞ and the leakage (L) are the same for both. For the purposes of this paper we assume that the reflection probability (R) is also the same; hence $P_{th} \sim TR \sim P_{fus}$. If the FTWR-SC and FTWR plasmas operate at the same temperature and density, the power in FTWR-SC is about 50% greater (4500 MW_{th}) than in FTWR. Thus, the FTWR-SC and FTWR have the same core power density of 124 kW/liter.

The fusion power (225 MW) and neutron source (8.0×10^{19} #/s) are also 50% greater. The FTWR-SC would have a 50% greater actinide loading (40.5 MT) and would operate on the same 4-batch fuel cycle as the FTWR [2], destroying the actinide content of spent nuclear fuel at the rate 153 MTU/FPY.

The LWR support ratio of the FTWR-SC would be 50% greater than for the FTWR, or 4.5 GW_e-LWR/FTWR-SC.

With η_{th} = 40%, the FTWR-SC would produce 1800 MW_e. The power required to operate the FTWR-SC is 365 MW_e, which leaves a net electrical power production of 1435 MW_e (Q_e = 4.9). The FTWR was designed to have Q_e = 1.0.

The BOC inventory is a function of the fusion rate and the operating parameters of the tritium system. We used a simple estimate of the beginning of cycle tritium inventory--a tritium inventory equivalent to the total number of fusions occurring in the first 30 full power days of operation must be available at the beginning of each cycle. The BOC tritium inventory for the reference fuel cycle of the liquid breeder is 0.27 kg, which is 50% larger than that of FTWR. Additionally, the maximum tritium inventory is 1.64 kg. As for the solid breeder the BOC inventory is 16.53 kg and peak inventory is 18.60 kg.

6. PLASMA PARAMETERS

The reference operating parameters at the maximum fusion power of 225 MW and H(y,2) = 1.0 are I = 6 MA, β_N = 2.5%, $\langle T \rangle$ = 8.25 keV, q_{95} = 3.09, n/n_{GW} = 0.8 and Q_p = 2.0, all of which (except for Qp) are within the existing tokamak database. Table 3 provides explanation of the various plasma parameters. The Plasma Operating Contour Plot (POPCON) of figure 2 displays the various plasma parameters of the reference design as a function of plasma density and temperature.

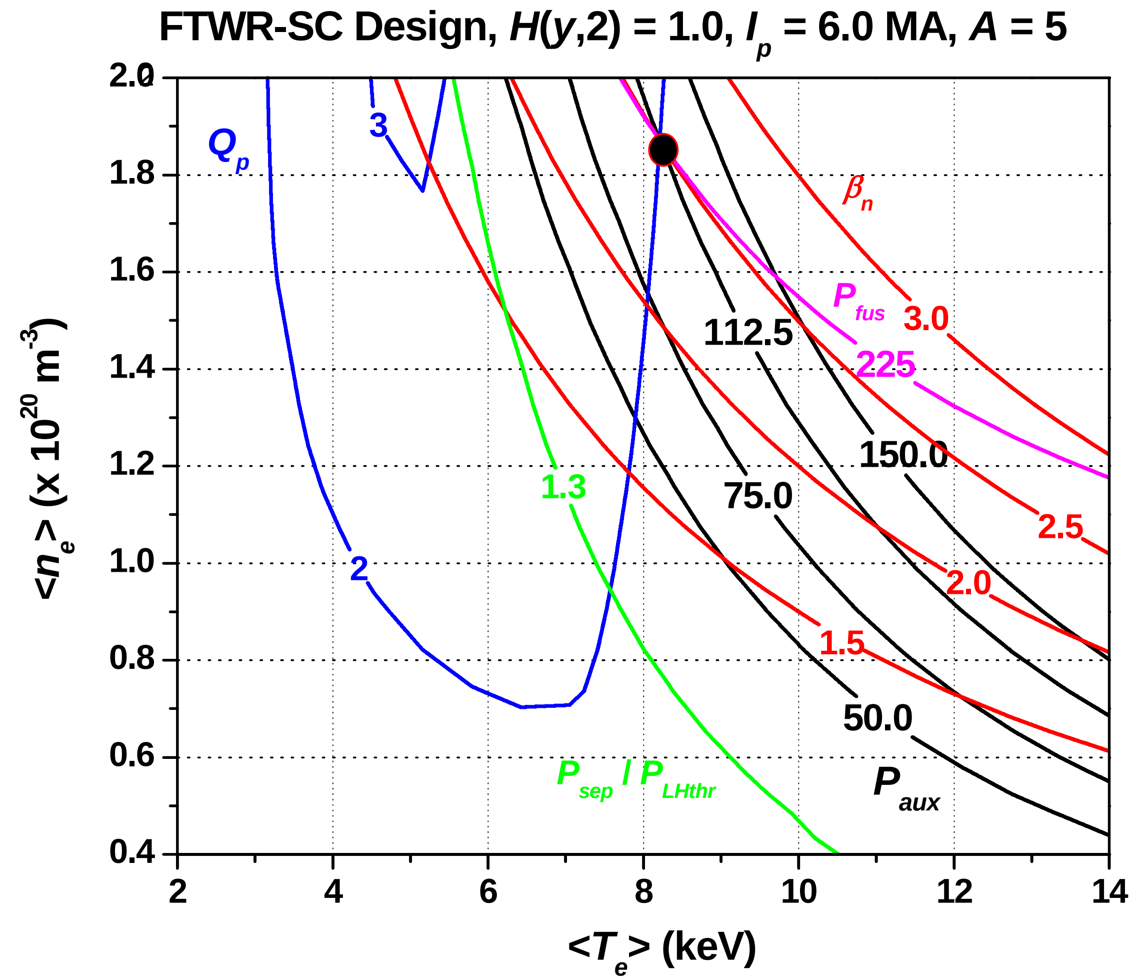

Figure 2: Plasma Operating Contour Plot (P_{aux} is the auxiliary heating power; P_{sep}/P_{LHthr} is the ratio of the power crossing the separatrix to the L-H power threshold; P_{fus} is the fusion power)

7. CURRENT DRIVE

The central solenoid produces a flux swing of 177 Vs, taking into account a field inversion in the CS coil from 13.5 T to -12 T, but not including any contributions from the poloidal field coils. For the 6 MA reference case, this flux variation is sufficient for plasma start-up and a 13 min current flat top.

Steady-state current drive operation is the preferred operating mode for the FTWR-SC. Assuming that all the auxiliary power is available for current drive, and using a simple scaling for the fast wave current drive efficiency [2], we estimate 4.1 MA of non-inductively driven current. A conservative estimate [2] of the bootstrap current is 1.5 MA. We can drive the full 6 MA by optimizing the plasma profiles to increase the bootstrap current fraction and/or by operating at a slightly lower Q and higher temperature to increase current drive power and efficiency.

Table 3: FTWR-SC / FTWR Parameter Comparison

Symbol	Parameter	FTWR-SC	FTWR
A (R_o/a)	Aspect Ratio	5.00	3.48
I_p (MA)	Plasma Current	6.00	7.00
κ	Elongation	1.77	1.70
δ	Triangularity	0.40	0.40
B_o (T)	Magnetic Field @ Plasma Center	7.48	6.10
H-Factor	Confinement Enhancement	1.00	1.10
P_{fus} (MW)	Fusion Power	225	150
P_{nw} (MW/m^2)	Neutron Wall Load	0.79 (225MW / 284.8m^2)	0.79 (150MW /189.9m^2)
P_{thw} (MW/m^2)	Thermal Power to first Wall	0.291	0.270
f_{BS}	Bootstrap Fraction	0.24	0.38
n_{avg} ($10^{20}m^{-3}$)	Average Neutron Density	1.9	2.0
S (#/sec)	Neutron Source Strength	8.00 x 10^{19}	5.32 x 10^{19}
β_n (%)	Normalized Beta	2.5	2.5
n/n_{GW}	Greenwald Density Ratio	0.80	0.75
q_{95}	Safety Factor @ 95% Flux Surface	3.09	3.00
Q_p (P_{fus} / P_{aux})	Plasma Q	2.0	2.0
Q_e	Electric Power Amplification Factor	5.0	1.0

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