TOKAMAK FUSION NEUTRON SOURCE FOR A FAST TRANSMUTATION REACTOR

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A conceptual design has been developed for a tokamak D-T fusion neutron source, based on ITER physics and technology, for a sub-critical fast reactor that would transmute the fissionable transuranic isotopes in spent nuclear fuel.

I. Introduction

The concept of fast-spectrum, sub-critical nuclear reactors driven by tokamak D-T fusion neutron sources based on ITER physics and technology[1] is being developed in a series of studies at Georgia Tech[2]. The transuranics-fueled reactors produce 3000 MWth, which enables them to fission the annual TRU discharge from three 1000 MWe LWRs. Nuclear fuel cycle studies indicate that it is practical to achieve > 90% transuranics (TRU) burnup with a neutron source strength $P_{fus} < 200 MW$ by repeatedly recycling and reprocessing the TRU fuel to remove neutron absorbing fission products and add fresh TRU[3-5], but that to achieve such high burnups without reprocessing the TRU fuel would require a neutron source strength[5].

Previous conceptual design studies for a He-cooled reactor[6,7] evolved to a (R=3.74 m, I=8.2 MA, B_{TFC}=11.8 T, P_{fus}<200 MW) tokamak neutron source based on ITER physics[1], as shown in Fig. 1. This paper reports physics and magnet structural studies of extending the source strength to P_{fus} = 400-500 MW within this same geometric/magnetic configuration.

II. Neutron Source Strength Capability

A fusion performance code, representing the applicable engineering and physics constraints[8], was used for systems analysis of plasma performance. The parameters (R= 3.7 m, a= 1.08 m, κ=1.7, δ=0.4, B_{TFC}=11.8 T, $\Delta\varphi$=26.3 V-s, n/n_{GW} = 0.75) were fixed based on the previous GCFTR design[7] and contours of P_{fus} and P_{aux} in (β_N-H_{98}) space were calculated for I = 9 and 10 MA; the latter is shown in Fig. 2. As can be seen, there is a wide range of (β_N-H_{98}-P_{aux}) over which P_{fus} = 400-500 MW can be achieved.

Table I shows the plasma performance parameters calculated for the 10 MA and 9 MA

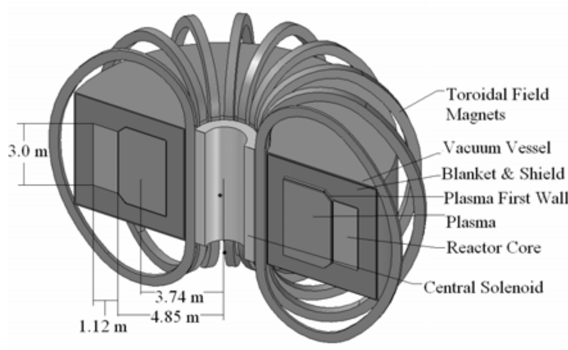

Fig. 1 Gas Cooled Fast Transmutation Reactor

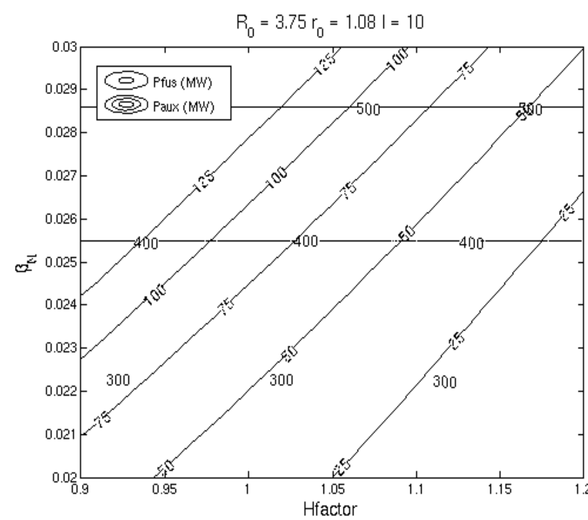

Fig. 2. Neutron Source Strength Operating Space of I = 10 MA Plasma

plasma current cases, and the ITER parameters. As a result of realistic parameter constraints, a current of 10 MA is necessary to meet the design objective of 500 MW fusion power at a reasonable level of auxiliary heating.

III. Adaptation of ITER Divertor to Helium Coolant

The heat flux on the divertor is of order 5 MW/m^2 for ITER and 1-2 MW/m^2 for GCFTR at P_{fus} = 200 MW, during normal operation. To handle this heat

TABLE I Neutron Source Parameters

Parameter	9 MA	10 MA	ITER 15 MA
P_{fus} (MW)	403	498	410
S (10^{19}/s)	14.2	17.5	14.4
P_{aux} (MW)	100	100	
Q_p	4.1	5.1	10.0
β_N(%)	2.85	2.85	1.8
H-factor	1.13	1.06	1.00
q_{95}	5.4	4.0	
V-s, Startup	98.5	107.3	
I_{bs} (MA)	2.63	2.55	
γ_{cd} (10^{-20} A/Wm^2)	0.456	0.58	
n_e (10^{20}/m^3)	1.84	2.05	
B_φ (T)	5.9	5.9	5.3
Γ_n (MW/m^2)	1.45	1.78	0.5
q_{FW} (MW/m^2)	0.32	0.35	0.15

flux, the ITER divertor (Fig. 3) employs either carbon fiber, carbon, or tungsten tiles, joined to copper blocks. The copper blocks are hollow with a smooth tube or a swirl tape along the tube and are assembled on the inner and outer vertical targets, as well as the divertor dome.

Water is the coolant for ITER, but because helium is the primary coolant for GCFTR, a thermal analysis to determine whether helium can cool the ITER divertor was made. In order to facilitate the adaptation to helium, the coolant flow for the GCFTR will not be in series, as is the case for ITER, but will have individual coolant loops for the inner vertical target, outer vertical target, and dome.

Divertor heat removal was modeled analytically based on straight pipe flow that is a certain distance below a uniform heat flux. Heat removal was also analyzed in three dimensions using Fluent[9] which solves the energy equation coupled with the Navier-Stokes equations. A three-dimensional model and mesh for one cooling channel of the outer vertical target, created using Gambit[9] consists of a copper block with a smooth tube. The analytical calculations of the maximum surface temperature of the block and average coolant exit temperature, when cooling with water, agreed with those of ITER[10], giving confidence in using the model to analyze helium. However, operating conditions for helium differ somewhat from water, in that the operating pressure increases from about 4 MPa with water to 6.5 MPa with He. The helium inlet temperature is 300 K, and helium mass flow rates vary from 0.4 to 1.2 kg/s.

The maximum surface temperature of the copper block was a major design constraint in the analysis of the coolant channel. Figure 4 shows the Fluent results of peak surface temperature on the copper block for the mass flow and heat flux region analyzed.

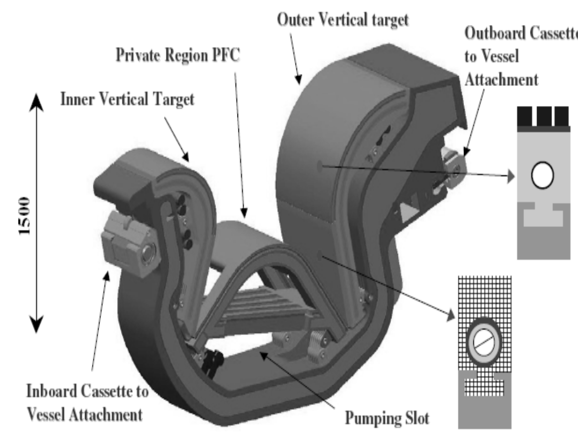

Fig. 3: Divertor Cassette[10]

The heat flux was modeled as a uniform heat flux on the entire plasma-facing surface with the initial conditions mentioned previously. Each mass flow rate case was run for seven different heat flux values ranging from 0.5 to 2 MW/m^2. The maximum allowable surface temperature was 773 K[10]; 696 K and 579 K represent 90% and 75% of the limit, respectively. The Fluent results agreed well with parallel analytical calculations. As Fig. 4 shows, using a uniform heat flux approximation results in an achievable heat removal for the 0.5 to 2 MW/m^2 divertor heat flux range anticipated for normal operation of GCFTR, without enhanced heat removal.

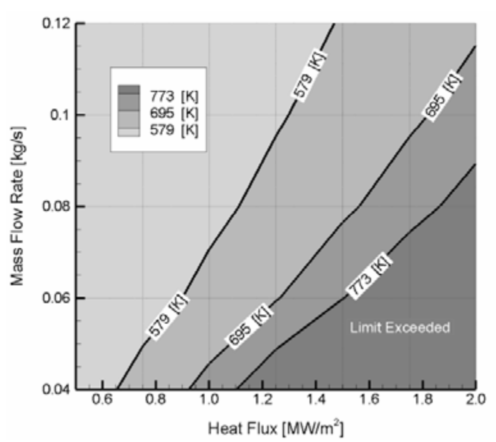

Fig. 4: Maximum Surface Temperature.

The mass flow rate range analyzed corresponds to an inlet velocity range of 60 m/s to 190 m/s. This velocity could be reduced by increasing the cross-sectional area of the channel. For this study, the flow tube is 10 mm in diameter, maintaining ITER's dimensions. The Fluent results show that a channel could sustain 1 MW/m^2 and avoid failure at the 60 m/s inlet velocity range, while an inlet velocity of 143 m/s would be required for 2 MW/m^2.

Based on the pressure drop across the channel that Fluent converged upon, the pumping power per channel was calculated. This calculation indicates that a 1 MW/m^2 heat flux at 0.04 kg/s would require approximately 250 W, and a 2 MW/m^2 heat flux at 0.1 kg/s would require 4.8 kW, per channel. The ratio of the pumping power to the total heat removed per channel (assuming an 85% pumping efficiency) results in 0.5% and 6.1% for the 1 and 2 MW/m^2 cases, respectively. To estimate the pumping power for GCFTR, these values were scaled up based on the total number of channels. This yields a total pumping power of approximately 511 kW and mass flow rate of 89 kg/s for the 1 MW/m^2 heat flux and 10.6 MW with 222 kg/s for the 2 MW/m^2 heat flux. This may be somewhat low since the center dome will require a larger pumping power than the outer targets.

Using results from both the analytical calculations and the Fluent model of the outer vertical target, approximate values were found for the effect of heat transfer enhancement by swirl tape. The enhancement multiplication factors used were 2 for the friction factor and 4 for the convective heat transfer coefficient[10]. The solution procedure involved iteration on the friction factor using the Colebrook formula and the Petukhov correlation for Nusselt number. This created a reduction in the required mass flow rate by approximately 45%, and thus a reduction in the pumping power.

The center dome region of the divertor is approximately 0.8 meters longer than the outer vertical target and thus presents a more limiting case. The analytical model indicates that an approximately 25% increase in mass flow rate would be required to sustain the same heat flux relative to the outer vertical target. The use of swirl tape or other heat transfer enhancement methods would improve the operating limits for the center dome.

Based on this analysis, it seems feasible to adapt the ITER divertor design for use in GCFTR using helium as the coolant.

III. Heating and Current Drive Systems

An important element in determining the GCFTR current drive requirement is the bootstrap current. Since GCFTR is to operate in steady state, it is mandatory that non-inductively driven current plus bootstrap current be equal to the total required plasma current of 9 to 10 MA. The required non-inductively driven current is $I_{cd} = (1 - f_{bs}) I_p$. With all the auxiliary power available for current drive and allowing the current drive to be described by the current drive figure of merit

$$\gamma_{cd} = \overline{n}_{e20} R_0 (1 - f_{bs}) I_p Q_p \frac{1}{P_{fus}}$$

we can determine a necessary value of this parameter for GCFTR. An empirical formula for the bootstrap current fraction (of the plasma current) is given by

$$f_{bs} = (1.32 - 0.235(\frac{q_{95}}{q_0}) + 0.0185(\frac{q_{95}}{q_0})^2)(\sqrt{\frac{1}{A}}\beta_p)^{1.3}$$

The required bootstrap current fraction in GCFTR at steady-state, 500 MW$_{th}$ operation is 0.25.

With the goal of increasing the tokamak fusion power to 500 MW$_{th}$, a H&CD system was designed to increase H&CD capabilities while meeting the operational requirements and design constraints of the higher-output tokamak. Due to geometric constraints on plasma access through the annular reactor core and cost, Neutral Beam Injection was precluded. Because of limited plasma access, it was determined that a lower hybrid (LH) H&CD system provided the correct size and power, and it has the highest achieved current drive efficiency.

The required current drive figure-of-merit for the LH H&CD system at 500 MW$_{th}$ fusion power, 10 MA of current, and a bootstrap current fraction of 25% is 0.577. With anticipated near-term advances, this value should be achieved soon. Additionally, higher bootstrap current should be achieved in the future, easing the RF current drive requirements.

TABLE II: GCFTR H&CD properties

Property	GCFTR	ITER
I_{bs} (MA)	2.50	~7.5
f_{bs} (%)	25	~50
I_p (MA)	10	15
P_{aux}(MW)	100	110
P_{tot}(MW)	120	130
# Port Plugs	6	10*
PD (MW/m^2)	33	9.2 **

* 4 equatorial, 3 upper, 3 NBI, ** ICRH power density

The LH port designs are based on the port plugs used in ITER. Each port has a power of 20 MW. In GCFTR, there are two sections of the annular fission reactor between the magnets absent on opposite sides of the reactor. Because of the highly constrained geometric options, the six 20 MW LH port plugs are centered vertically and toroidally in the outer plasma chamber wall in an arrangement shown schematically in Fig. 5, with three ports in each of the two absent reactor section. Based on the ITER design, it is expected that each LH H&CD port would provide 20 MW of heating and ~1.5 MA of current drive. Table 2 shows a comparison for the GCFTR and ITER H&CD systems

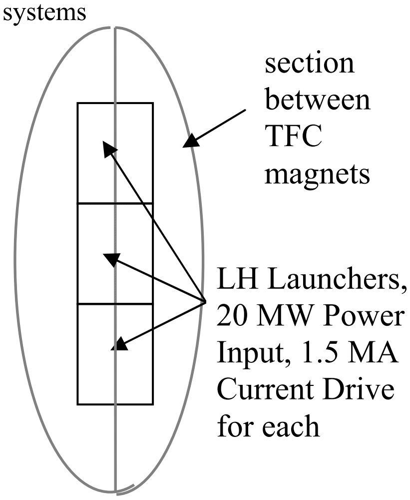

Fig. 5: LH Port Geometry

IV. Superconducting Magnet System

The central solenoid (CS) for the GCFTR is directly adapted from ITER and uses a cable-in-conduit Nb_3Sn cooled superconductor surrounded by an Incoloy 908 jacket. The conducting strands have an internal diameter of 38 mm of conducting material and coolant chamber, surrounded by a 51 mm square of structural Incoloy 908. The plasma current of 10 MA determined the requirement for the inductive start Volt-seconds, VS_{start}, to increase from the previous design value to a VS_{start} of a minimum of 107.3 Volt-seconds. A configuration of a flux core radius of 0.88 m and a CS thickness of Δ_{OH} =0.48 m created a VS_{start} of 108 V-s, satisfying the minimum needed, as well as leaving the total radius at 1.36 m, the same as the previous design. The maximum stress allowable in the CS solenoid, with a maximum magnetic field of 13.5 T, by ITER standards and Incoloy 908 limitations, is 430 MPa. When using a flux radius of 0.88 m and thickness of 0.48 m, the stress created is about 399.9 MPa. This is very close to the limit; therefore any increase in plasma current above 10 MA would require a thicker CS to reduce the stress.

The toroidal field coils (TFCs) are designed using ITER[12,13] as a basis. The GCFTR uses a Nb_3Sn cable-in-conduit superconductor with an Incoloy 908 jacket for support, as ITER does. The GCFTR will have 16 TFCs. The thickness of the TFC for GCFTR is determined by conserving the tensile stress calculated in the same manner for the ITER TFC. The tensile stress is approximately equal to the magnetic force/cross section area or $s=F/A=(C^* I_{TF}^{2}/A)$. Keeping the stress constant, the area of the GCFTR TFCs is found when the ITER parameters of a coil current of 9.13 MA and area of 0.3 m^2 are used. The TF coil current needed in the GCFTR is calculated using Ampere's Law. The magnetic field on the major axis of the plasma is calculated so that the current inside the conductor can be calculated. From the required current in the TFC to create B_{TFC} we are able to calculate the required area by keeping the stress constant from ITER. Using these equations the area of the TFCs for the GCFTR comes out to 0.1567 m^2. Keeping the radial thickness of the TFCs the same as the previous design of 0.43 m requires the TFCs to have a new width of 0.3645 m in the toroidal direction. The new width of the TFCs was checked using a CAD model to make sure that 16 TFCs would fit around the CS without overlap.

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