IMPURITY CONTROL AND VACUUM PUMPING SYSTEM DESIGN AND ANALYSIS FOR NEXT-GENERATION TOKAMAKS*

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ABSTRACT

Impurity control system design and performance studies were performed in support of the Tokamak Fusion Core Experiment (TFCX) preconceptual design. Efforts concentrated on the pumped limiter and vacuum pumping system design configuration, thermal/mechanical and erosion lifetime performance of the limiter protective surface, and helium ash removal performance. Analysis results indicate that the limiter/vacuum pumping system design provides marginally adequate helium ash removal. Difficulties in providing adequate helium ash removal for more compact or higher fusion-power-density devices are addressed. Erosion, primarily by disruption-induced vaporization and/or melting, limits the protective surface lifetime to about one calendar year or only about 60 full-power hours of operation. In addition to evaluating impurity control system performance for nominal TFCX conditions, these studies attempt to focus on the key plasma physics and engineering design issues that should be addressed in future research and development programs.

INTRODUCTION

Next-generation tokamaks will require impurity-removal systems to exhaust the helium "ash" formed during relatively long-duration, deuterium-tritium (D-T) fusion burn periods. Design and analysis of impurity control systems for several next-generation tokamak design concepts were performed as part of the Tokamak Fusion Core Experiment (TFCX) effort. Because of its attractive configurational aspects, TFCX impurity control system studies concentrated on the pumped limiter. Although some aspects of the engineering design and analysis of magnetic divertors and pumped limiters differ, there are many similarities. For example, both approaches require a protective surface

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capable of withstanding high fluxes of energetic particles; and both require a high-speed, vacuum-pumping system to exhaust the helium ash.

The purpose of this study was to establish feasible engineering design solutions and performance evaluations for these two aspects of the TFCX impurity control system. In addition to presenting the design configuration, requirements, operating conditions, and performance evaluation for the TFCX pumped limiter and vacuum pumping system, issues and concerns for impurity removal in next-generation, long-burn duration, tokamak devices are discussed.

Impurity control systems were examined for each of the four TFCX design options, characterized as (1) the nominal superconducting (S/C), toroidal field (TF) coil option; (2) the high-performance S/C TF coil option; (3) the nominal copper TF coil option; and (4) the high-performance copper TF coil option. Major parameters of the four design options are listed in Table 1.

Although details of the design and performance for pumped limiter and vacuum pumping system components differ for the four options, the overall design features are the same. Impurity control components are identified in Fig. 1, which is the elevation view for the nominal S/C TF coil option. The limiter design forms a continuous toroidal belt at the

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bottom of the device and features a flat surface with a single leading edge. The rationale for selecting this set of design features is discussed in ref. 1. The vacuum pumping system features large vacuum ducts (approximately 1-m diam.) and high-speed, compound cryopumps.

Fig. 1. TFCX nominal S/C TF coil option.

Pumped Limiter

The pumped limiter is designed to remove or "scrape off" a fraction of the particles leaving the plasma and thus provide a means to exhaust the D-T fusion-reaction-produced helium and other impurities. In performing this task, the limiter is subjected to intense heat loads due to impingement by large fluxes of energetic particles. Heat loads and particle loads on the limiter were defined consistent with the TFCX Preconceptual Design Specification.²

It is assumed that 64% of the total alpha power reaches the limiter via charged-particle impingement. The total ion current to the limiter is estimated, assuming a particle confinement time of 0.1 s. The distribution of the heat and particle fluxes along the limiter surface was determined using the Schmidt model,³ which predicts an exponential falloff of the heat flux, the particle flux, and the particle energy in the scrapeoff layer.

Using the approach of ref. 1 and the local limiter/field line geometry, flux distributions were calculated for the four TFCX options. Peak values are listed in Table 2 for all four options. The peak heat flux, including the contribution from both charged particles and charge-exchange neutrals, varies from about 2.6 MW/m² for the nominal S/C and nominal copper options to 5.5 MW/m² for the high-performance copper option. The leading edge is located to allow 10% of the particles leaving the plasma to pass underneath the limiter surface. The heat flux at the leading edge peaks at a value significantly lower than

that on the flat surface and therefore should present no additional design problem.

Intense heating of the limiter during plasma disruptions is expected. Most (80%) of the plasma internal energy is released during the thermal quench phase of the disruption. During the subsequent current quench phase of the disruption, the remaining plasma internal energy (20%) and 50% of the plasma magnetic energy (assumed to be converted to plasma internal energy) are deposited.

The resulting disruption heating conditions, which were found to be essentially the same for all four options, consist of a thermal quench duration of 1 ms with an energy density of 5 MJ/m² and a current quench duration of 10 ms with an energy density of 2 MJ/m². About 4000 disruptions are expected during the lifetime of TFCX.

The limiter blade is constructed of protective surface tiles attached to an actively cooled copper substrate. Several critical issues, such as thermal shock damage and tritium retention and permeation, remain to be investigated in R&D studies before selecting a protective surface material. Beryllium and graphite are the leading protective surface tile material candidates. Both materials have a low atomic number and are therefore more compatible with the plasma than higher-Z candidates such as tungsten, molybdenum, or tantalum.

Since tile erosion rates are large (discussed later), tile thickness must be maximized to maximize limiter lifetime. Temperature and stress distributions were calculated as a function of tile thickness using the two-dimensional, finite-element thermal analysis code TACO2D and the structural analysis code NIKE2D. The duplex-limiter, protective surface configuration is shown in Fig. 2. The substrate was assumed to be a double-walled, copper panel with water coolant.

reducing tile size would have no impact on the allowable graphite tile thicknesses determined in this study.

Fig. 2. Configuration used for thermal/mechanical analysis.

Fatigue failure data⁴ were used to define design limits on the cyclic thermal stresses. The design limits apply a factor of 2 reduction on the failure stress or a factor of 10 reduction on the number of cycles to failure, whichever is more limiting. For two years of TFCX operation (2.2×10^4 cycles), the elastically calculated fatigue stress range limit is 210 MPa.

The tile-to-substrate thermal contact conductance required to adequately cool the tiles was examined. The maximum tile thickness, which was defined in this case as the thickness that results in the maximum allowable tile temperature at the plasma side surface, is shown in Fig. 3 as a function of thermal contact conductance. Maximum allowable temperatures are assumed to be 850°C for beryllium, based on sublimation, and 1000°C for graphite, based on radiation-enhanced sublimation. For either tile material, the thermal contact conductance must be above about 10 kW/m²-K to adequately cool the tiles and should be above about 10³ kW/m²-K to maximize the tile thickness.

It is judged that a bonded attachment method such as brazing or a plasma-sprayed coating is required to achieve and maintain these relatively high values of thermal contact conductance. The effects of the constraint of the bonded tile on the substrate stresses are included in this analysis. The width of the tile is assumed to be large, compared with its thickness. The entire blade is free to expand but is constrained from bending. As demonstrated in ref. 5, for small enough tiles (less than 10 mm x 10 mm), the tiles provide essentially no constraint. If tiles of this size were feasible, stresses in the substrate would be lower than those computed in this study. For beryllium tiles, the allowable thickness could be significantly increased. Since graphite tile thicknesses are limited by peak surface temperature,

Fig. 3. Thermal contact conductance requirements.

Peak graphite tile temperatures are presented in Fig. 4. Two upper temperature limits were considered for graphite. If it is necessary to maintain the graphite temperature below the threshold for chemical sputtering erosion (approximately 400°C), the maximum tile thickness ranges from 5 mm to 14 mm, depending on the TFCX design option. Based on a review of chemical erosion data,⁶ it is likely that chemical erosion at the particle fluxes experienced by the limiter will be negligible compared to physical sputtering erosion. At the radiation-enhanced sublimation limit of 1000°C, the maximum allowable tile thickness ranges from 14 mm to 36 mm, depending on the TFCX design option. Thermal stress limits in the graphite tiles and copper substrate were found to be less restrictive than either graphite temperature limit.

Results presented in Fig. 5 demonstrate that the copper substrate thermal stresses exceed the fatigue limit, with a beryllium tile thickness ranging from 2 mm to 7 mm, depending on the TFCX design option. Thickness limits based on sublimation of beryllium are less restrictive.

Physical sputtering erosion rates were found to be about 180 mm per full-power year for both protective surface tile materials. Redeposition of sputtered material and self-sputtering are not included in this analysis. Although these physical sputtering erosion rates are significant when specified in terms

of full-power operating time, the erosion rate in calendar time is quite low (about 0.5 mm per calendar year), due to the low availability of TFCX.

entire ten-year TFCX lifetime, is 150 mm for graphite and ranges from 180 mm to 650 mm for beryllium, depending on the portion of melt layer that is lost.

These disruption erosion rates are much larger than the sputtering erosion rates. This is in direct contrast to the results obtained in the Fusion Engineering Device⁷ (FED) and the International Tokamak Reactor⁵ (INTOR) studies. This can be explained by examining the disruption frequency in terms of disruptions per full-power operating period. With a disruption frequency of $1.9 \times 10^{-3} \text{ s}^{-1}$, TFCX has approximately a factor of 30 more disruptions per burn time than FED and a factor of 230 more than INTOR.

Erosion lifetimes for the protective surfaces were calculated by combining the tile thickness sizing results and erosion rates. Depending upon the TFCX design option, lifetimes with graphite tiles allowed to operate at 1000°C vary from 70 full-power hours (1.2 calendar years) to 170 full-power hours (2.8 calendar years). If the graphite tiles are limited to temperatures below 400°C, lifetimes are reduced by approximately a factor of 2.5. Lifetimes for a limiter with beryllium tiles range from 10 full-power hours (0.15 calendar years) to 30 full-power hours (0.5 calendar years) if the melt layer remains intact. Lifetimes are reduced by a factor of 3 if the entire melt layer is lost.

VACUUM PUMPING SYSTEM

The TFCX vacuum pumping system is designed to pump down the vacuum chamber initially and between burns and to exhaust the helium ash during the burn period. During the burn cycle, the neutral gas pressure in the limiter slot is estimated² to be $2.0 \times 10^{-2} \text{ Pa}$ ($1.5 \times 10^{-4} \text{ torr}$). The vacuum pumping system should be adequately sized to remove helium as it is produced by fusion with only a 5% helium concentration. Since the fusion power differs for each of the four TFCX options, the pump speed required for helium ash removal varies.

Major vacuum pumping system components are identified in Fig. 1. During the burn cycle, the plasma particles enter the toroidally continuous entry slot underneath the limiter blade. The entry slot is connected to 16 passages called limiter slots, one in each torus sector. The limiter slots are connected to ducts that lead to 0.8-m-diam. valves and compound cryopumps. The duct area varies as it passes through the region between the TF coils. The 0.8-m-diam. valves are required to be all metal for tritium compatibility.

Fig. 4. Graphite tile thickness limits.

Fig. 5. Beryllium tile thickness limits.

Disruption erosion was calculated using the model described in ref. 7. Graphite is expected to lose 36 μm per disruption by sublimation. Beryllium loses 44 μm per disruption by vaporization; in addition, a 160- μm -thick melt layer is formed. The stability of this melt layer is a key issue in the comparison and selection of a reference tile material. The total disruption-induced erosion, over the

Each of the 16 compound cryopumps has cryocondensation panels for pumping hydrogen and cryosorption panels for pumping helium. The pump speed for helium is $50 \text{ m}^3/\text{s}$, and the hydrogen pumping speed must be approximately the same.

The net helium pumping speed from the limiter slot to the vacuum pumps ranges from $170 \text{ m}^3/\text{s}$ for the high-performance copper option to $280 \text{ m}^3/\text{s}$ for the nominal S/C option. As shown in Table 3, these pump speeds are slightly lower than those required to remove the helium ash at a 5% helium concentration. The helium concentration of the plasma will therefore increase slightly, and the fusion power will decrease slightly. An estimate of the self-consistent helium content and fusion power is shown in Table 3 for all four options. These relatively slight changes to the nominal TFCX parameters are judged to be acceptable.

Since helium ash removal is only marginally acceptable for the TFCX design options, long-burn-duration devices with higher fusion power densities are expected to have helium exhaust difficulties. Assuming the configuration approach used for the TFCX design (see Fig. 1) and using a limiter slot pressure of 0.02 Pa with 5% helium concentration,² the limiting fusion power density estimated as a function of major radius is plotted in Fig. 6. These estimates were performed by assuming that the net vacuum pumping system speed is limited only by the conductance of the ducts leading from the limiter to the high-speed vacuum pumps; that is, the speed of the vacuum pumps was assumed to be high compared to the conductance of the ducts. Conductances were scaled from the major and minor radii according to relations developed from the calculations for the four TFCX design options. Devices with fusion power densities falling significantly above this line are expected to experience helium exhaust difficulties if a TFCX-type pumped limiter or single-null poloidal divertor is used.

Fig. 6. He exhaust performance limits.

For high-fusion-power-density devices, significant design changes such as the use of a double-null, poloidal divertor, or top and bottom pumped limiters with two sets (upper and lower) of vacuum ducts and pumps are required to significantly improve helium exhaust performance. These design changes would have a significant impact on device access, maintenance, and support structure arrangement, which could render the overall TFCX device configuration approach infeasible.

Using the vacuum pumping system optimized for helium ash removal, the base pressure during chamber pumpdown was estimated to range from about $1.3 \times 10^{-7} \text{ Pa}$ for the high-performance copper device to $2.3 \times 10^{-7} \text{ Pa}$ for the nominal S/C device. These base pressures satisfy the guideline of $1.3 \times 10^{-6} \text{ Pa}$ (10^{-8} torr).

Key Issues

Several key issues remain to be resolved before selecting a specific design. Experimental verification of pumped limiter performance before the next-generation device is built should be a high priority. Without this verification, the flexibility to include a poloidal divertor must be factored into the design and built into the device, thus negating much of the cost advantage of using pumped limiters.

Large uncertainties in plasma edge and helium ash removal conditions lead to conservative design approaches and highly uncertain engineering performance characteristics. For example, if the limiter slot pressure were assumed to be 1 Pa (as it was in FED studies) instead of the 0.02 Pa specified for TFCX

studies, then the vacuum pumping speed requirement would be reduced by about two orders of magnitude, leading to dramatic changes in the vacuum pumping system design.

Materials and fabrication R&D programs are required to address tile material selection and tile-to-substrate attachment issues. A data base for sputtering erosion of graphite at relevant particle fluxes, particle energies, and surface temperatures is required to establish chemical erosion characteristics. The stability of the beryllium melt layer formed during disruption-type heating must be resolved. The data base for tritium permeation and retention for these materials is poor. Techniques for reliably achieving the good tile-to-substrate thermal contact required to keep the tiles below upper-temperature limits must be developed.

CONCLUSIONS

Results of this study establish feasible engineering design and performance for the TFCX impurity control system. Significant findings from this study include (1) design constraints on the tile-to-substrate attachment method and the impact of these constraints on limiter lifetime, (2) the dominance of disruption-induced erosion compared to sputtering erosion, and (3) the identification of impurity control difficulties for small, high-fusion-power-density tokamak devices.

A thermal contact conductance between the tile and substrate greater than about $10 \text{ kW/m}^2\text{-K}$ was found to be necessary to adequately cool the tiles. It is judged that this relatively high thermal contact conductance can be achieved and maintained only with a bonded attachment scheme such as a brazed joint or diffusion bond.

Analysis results indicate that surface temperature limits the graphite tile thickness to 36 mm (for the nominal S/C TF coil option). Beryllium was limited to much thinner tiles (only 7 mm) by thermal stress fatigue of the rigidly attached copper substrate.

Erosion of the limiter surface by both sputtering and disruption-induced melting and vaporization was evaluated. Disruption erosion is the dominant erosion mechanism for TFCX. This result is in direct contrast to results from the FED and INTOR studies. Erosion lifetimes are 2.8 years of operation for graphite with a temperature limit of 1000°C and 0.5 years of operation for beryllium, assuming that the entire melt layer formed during disruptions remains intact.

Helium ash removal during the burn, as opposed to plasma chamber pumpdown, was found to size the vacuum pumping system. To satisfy

the helium ash removal requirement, a high net vacuum pumping speed (approx. $300 \text{ m}^3/\text{s}$) is required at the reference helium concentration of 5%. With duct sizes maximized and high-speed cryopumps provided at each torus sector, the design still could not achieve the desired pump speed. Helium concentrations, calculated for the four TFCX design options, ranged from 6.0% to 8.5%, and the corresponding decreases in fusion power ranged from 4% to 9%. These relatively slight changes in the reference parameters are judged to be acceptable. However, based on these studies it is concluded that adequate helium ash removal performance may be difficult or impossible for devices which are smaller in size (i.e., major radius) and/or which have higher fusion power density than those examined for TFCX.

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Table 1. TFCX parameters

Parameter	Design option			
	Nom. S/C	Hi-perf. S/C	Nom. Cu	Hi-perf. Cu
Major radius, m	4.08	3.61	3.55	2.35
Minor radius, m	1.52	1.30	1.37	0.83
Plasma current, MA	11.2	10.4	11.0	8.9
Field-on-axis, T	3.73	4.23	3.88	5.83
Fusion power, MW	264	269	238	244
Pulse length, s	647	470	509	184

Table 2. Limiter conditions during normal operation

Condition	Design option			
	Nom. Cu	Hi-perf. Cu	Nom. S/C	Hi-perf. S/C
Heat flux, MW/m ²	2.6	5.5	2.6	3.3
Ion flux, 10 ²² m ⁻² .s ⁻¹	1.0	0.9	1.0	1.0
Ion energy, keV	0.6	1.2	0.5	0.6
Neutral flux, 10 ²¹ m ² .s ⁻¹	2.2	2.6	2.2	2.2
Energy of neutrals, 0.32 keV		0.65	0.28	0.36

Device option	Pumping speed (m ³ /s)		Fusion power (MW)	
	Desired	Actual	Desired	Actual

Table 3. Helium ash removal performance

Device option	Pumping speed (m ³ /s)		Helium content %	Fusion power (MW)	
	Desired ^a	Actual	Actual	Desired	Actual
Nom. Cu	310	220	6.7	238	226
Hi-perf. Cu	320	170	8.5	244	220
Nom. S/C	350	280	6.0	264	256
Hi-perf. S/C	360	260	6.6	269	256

^aPump speed required to achieve the desired 5% helium content in the plasma.

TFCA

TF Coil

Fig. 1 NOMINAL SC OPTION - ELEVATION VIEW

ANXX-8408-044-TB-FEDC

Fig. 2

CONFIGURATION USED FOR THERMAL/MECHANICAL ANALYSIS

ANN-6406-176-JH-FE06

Fig. 3 Thermal Contact Conductance requirements

TILE THICKNESS LIMITS:

THERMAL STRESS FATIGUE

ASME-8400-111-JN-7800

TILE THICKNESS LIMITS:

MAXIMUM TEMPERATURE

ASME-8400-111-JN-7800

Fig. 4 Graphite Tile thickness limits

Being done
for JRT Tm

Fig. 5. Ber. II. μ - thickness limiter

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Fig. 6. Helium exhaust performance limits