CONF-830406--10

Invited Paper to be Submitted at the 5th Topical Meeting on the Technology of Fusion Energy, April 26-28, 1983, Knoxville, Tennessee

CONF-830406--10

DE83 009595

KEY ISSUES OF FED/INTOR IMPURITY CONTROL SYSTEM*

By

Mohamed A. Abdou
Argonne National Laboratory
Fusion Power Program
Argonne, IL 60439

Submitted October 1982

* Work supported by the U.S. Department of Energy.

KEY ISSUES OF FED/INTOR IMPURTY CONTROL SYSTEM

Mohamed A. Abdou
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

A key part of the FED/INTOR activity over the past year has focused on examining the critical issues and developing credible physics and engineering solutions for the impurity control system(1-3). The primary emphasis of the work was on the edge-region physics, plasma-wall interaction, materials, engineering and magnetic considerations of the poloidal divertor and pump limiter.

The key materials issues involve the selection of a plasma-side material for the limiter or divertor plate and the methods for fabricating and attaching the material to a structural heat sink. The choice of material for the surface of a limiter or divertor plate depends upon the pre-sheath temperature, T_s, in front of the surface. The pre-sheath temperature is approximately the same as the plasma edge temperature (T_g) except that $T_s < T_g$ in the case of the divertor when T_g is low (< 100 eV) because of cooling in the divertor channel. Detailed erosion/redeposition calculations show that: 1) most of the materials sputtered from the limiter or divertor plate surfaces are redeposited near the origin; 2) medium- and high-Z materials result in attractive engineering solutions if T_s < 50 eV; 3) only low-Z materials are viable for T_s > 50 eV because of unacceptable self sputtering for medium- and high-Z materials; 4) low-Z materials result in acceptable design solutions at T_s > 700 eV; 5) acceptable design solutions in the range 100 eV < T_s < 400 eV require delicate balance, as predicted, between erosion and redeposition.

Plasma transport calculations have been performed to evaluate the most probable values of T_g and T_s. The most probable edge temperature range is 100 eV < T_g < 300 eV. Low edge temperatures (T_g < 50 eV) require high-edge radiation while high-edge temperatures (T_g > 700 eV) require pellet fueling and high vacuum pumping.

For T_s < 50 eV, tantalum and tungsten are the preferred plasma-side materials. Erosion by sputtering is small and these high-Z materials are most resistant to plasma disruptions. Tantalum is preferred over tungsten because of superior fabrication properties.

Beryllium is the preferred plasma-side material for T_s > 50 eV because all other materials appear to have a serious flaw in at least one area. Recent results on chemical sputtering of graphite require that its maximum temperature be limited to < 500°C. This temperature limit combined with the rapid decrease in the thermal conductivity of graphite under irradiation imply a small tile thickness and a short lifetime (~ 0.6 y) under FED/INTOR conditions. Boron is rejected because of extremely poor fabrication and thermophysical properties. Boron carbide suffers from poor thermal shock resistance. A key issue for silicon carbide is whether its self sputtering yield will exceed unity at particle energies > 500 eV. In addition, SiC has poor thermophysical properties. The newly developed SiC with high thermal conductivity appears to lose its advantage under irradiation. The thermal conductivity is predicted to decrease within ~ 1-2 month of irradiation to a final low value that seems to be independent of the initial value.

The key problem for beryllium is melting under plasma disruptions. Under FED/INTOR conditions, the lifetime of beryllium tiles on the surfaces of limiter and divertor plate is ~ 3.8 years if the melt layer does not erode. This lifetime is reduced to ~ 1.8 years if all the melt layer erodes and the thermal-quench time constant for plasma disruptions is 20 ms. A much shorter time constant results in substantial reduction in the lifetime.

The impurity control system appears to present some of the most difficult design issues for tokamaks. The two leading candidates, a poloidal divertor and pump limiter, each have considerable uncertainties. The divertor appears to have advantages in impurity control and helium pumping but adds considerable mechanical and magnetic complexity to the reactor. Uncertainties in the scrapeoff conditions, erosion by physical sputtering, disruptions and arcing, and in redeposition of eroded materials make prediction of the lifetime of the limiter and divertor plates very difficult.

References

1. W. M. Stacey, et al., USA Final Phase 2A Report.

2. J. Schmidt, et al., Chapter VI, "Impurity Control Physics," ibid.

3. M. A. Abdou, et al., Chapter VII, "Impurity Control and First Wall Engineering," ibid.