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ANNUAL REPORT

**OCTOBER 1, 1995
THROUGH
SEPTEMBER 30, 1996**

FUSION TECHNOLOGY DEVELOPMENT

GENERAL ATOMICS

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FUSION TECHNOLOGY DEVELOPMENT
ANNUAL REPORT TO THE
U.S. DEPARTMENT OF ENERGY

OCTOBER 1, 1995 THROUGH SEPTEMBER 30, 1996

by
PROJECT STAFF

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Cover Photos: Twelve-element combine fast wave current drive antenna.
Large photo: antenna operating in the presence of a plasma.
Inset: antenna as installed.

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1. FUSION TECHNOLOGY DEVELOPMENT OVERVIEW

In FY96, the General Atomics (GA) Fusion Group made significant contributions to the technology needs of the magnetic fusion program. The work was supported by the Office of Fusion Energy Sciences, ITER and Technology Division, of the U.S. Department of Energy. The work is reported in the following sections on Fusion Power Plant Design Studies (Section 2), Plasma Interactive Materials (Section 3), SiC/SiC Composite Material Development (Section 4), Magnetic Diagnostic Probes (Section 5) and RF Technology (Section 6). Meetings attended and publications are listed in their respective sections.

The overall objective of GA's fusion technology research is to develop the technologies necessary for fusion to move successfully from present-day physics experiments to ITER and other next-generation fusion experiments, and ultimately to fusion power plants. To achieve this overall objective, we carry out fusion systems design studies to evaluate the technologies needed for next-step experiments and power plants, and we conduct research to develop basic knowledge about these technologies, including plasma technologies, fusion nuclear technologies, and fusion materials. We continue to be committed to the development of fusion power and its commercialization by U.S. industry.

2. FUSION POWER PLANT DESIGN STUDIES

For FY96 the primary technical responsibility of General Atomics was to coordinate the ARIES-reversed shear (ARIES-RS) divertor design. In the following we focus on the reporting of the ARIES-RS work.

2.1. INTRODUCTION

Power exhaust and particle handling have been two of the most difficult design problems for magnetic toroidal devices. The high peak surface heat flux and target plate erosion rate are primary concerns, followed by the control of impurity migration and helium ash removal. For the ARIES-RS design, the double null divertor configuration creates a practical means to enhance ash removal and vacuum pumping while focusing the flow of transport power to the divertor plates. From an energy conservation point of view, the problem is to account for the distribution of radiative and particle power generated in the fusion reaction with additional power contribution from current drive. We have to account for the subsequent migration of particles from the core through the plasma mantle and the scrape-off layer (SOL) and into the divertor region and targets. Along the way particles can lose their power by atomic processes and radiate additional power to the first wall and therefore reduce the maximum heat flux to the divertor.

Based on the experimental results of DIII-D, we identified the distribution of power from the core to various regions and surfaces of the first wall and divertor. The most difficult power handling problem involves the outer divertor, since 80% of the power flows out of the main plasma is into the outer divertor scrapeoff layer. Without help from radiation, we estimated that with a total transport power of about 500 MW, the peak heat flux to the divertor plate can be as high as 18 MW/m² which is not acceptable from an engineering design point of view due to excessive temperature and thermal stresses. Therefore, we investigated the possibility of impurity addition in the divertor region in order to increase the radiation contribution and reduce maximum heat flux at the divertor plate. The physics calculation of this approach is presented in the next section, followed by the selection of divertor configuration, power balance, coolant routing and thermal-hydraulics design, thermal stress analyses of the divertor plate and a summary on particle exhaust and vacuum system design.

2.2. DIVERTOR PHYSICS

To avoid overheating the divertor surfaces, the power flow into the scrapeoff can be dispersed over a wider area of the plasma facing surfaces. One way this can be done is by enhancing the radiated power from the divertor and main plasmas, often referred to as "radiating divertor" and "radiating mantle" methods, respectively. In the former, excessive power flowing into the divertors is dissipated largely by radiating it away to adjacent divertor surfaces. In the latter, this power flow is radiatively dissipated at the edge of the main plasma *before* it passes into the SOL. While both methods had their attractive elements, we settled on the "radiating divertor" approach, since it was more consistent with the ARIES-RS design.

A relatively simple accounting of the conducted power of particles from the burning core into the scrapeoff layer and onto the divertor floor, consistent with a peak conducted power flux to the floor of 4 MW/m^2 , allows a determination of the required radiated power which will be added on to form the total heat flux. A simple model of the SOL and divertor plasma, based on Barr and Logan [1], was used to determine the plasma density and temperature distributions. Radiation was increased to meet the divertor radiation requirements by adding neon to the plasma and using the calculated density and temperature profiles and coronal radiation rates. Heat flux on the plate under the outboard divertor leg is described by $Q_{\text{div}} = g_T C_s n_{\text{div}} T_{\text{div}}$, where g_T is the power transmission coefficient, C_s is the poloidal component of the flow velocity at the sheath, n_{div} and T_{div} are the electron density and temperature at the divertor strike points. Plasma Z_{eff} is characterized by considering both the helium and impurity concentrations in the main plasma.

We considered four candidate gases in this study: neon, argon, krypton, xenon. Although neon is a relatively poor radiator inside the main plasma, it radiates well in the cooler scrape off and divertor regions. While the higher-Z impurities (i.e., argon, krypton, and xenon) also can radiate well *per ion*, adding too much of these impurities to the plasma would either unacceptably dilute the main plasma or quench it entirely. The neon-injection case was chosen on the basis of it having the least impact on the density, temperature, and current density of the main plasma in the present ARIES-RS design. The resulting radiated power and heat flux distributions were used in evaluating the cooling requirements for the plasma facing components. Our study shows that adding neon to the plasma system, under conditions consistent with present ARIES-RS design parameters, would radiate a sufficient amount of power in the SOL and divertor regions. The helium concentration is taken as $\sim 18\%$ of the plasma ion density. To be consistent

with the core boundary conditions of $Z_{\text{eff}} = 1.7$ and an edge density of $0.3 \times 10^{20} \text{ m}^{-3}$, it was found that a divertor impurity enrichment factor (ratio of core-to-divertor impurity concentration) of 13 was required. Such a high enrichment factor is far beyond the present database. This enrichment factor can be reduced by increasing the edge density and concomitantly increasing the core radiation.

2.3. DIVERTOR CONFIGURATION AND STRUCTURAL DESIGN

The divertor has to be designed to meet all the functional requirements of the plasma facing components design. The divertor surface configuration is defined from the following criteria: the outboard divertor plates intersect the magnetic field line at an angle equal to or less than 15° and the inboard divertor plates intersect the magnetic field line at an angle of less than 56° to reduce the local heat flux. Vanadium alloy is selected as the divertor coolant channel and back plate material. A 2 mm layer of tungsten was attached to the integral channels which are 1 mm thick to withstand particle bombardment with acceptable erosion rate and to satisfy material temperature limits. Fig. 2-1 and 2-2 show the double null divertor cross-sectional view and details including the first-wall/blanket and structure ring. The divertor module illustrated consists of two principal parts: the target plates and structure. Each divertor is covered by three target plates: inboard, outboard and dome. The target plates are mechanically connected to the structure ring through strong adjustable screw-type attachments. These attachments can be designed to offer lateral flexibility for thermal expansion and to react the full force of disruptions. The vacuum pumping channels are located behind the dome, near the strike points on the inboard and outboard target plates.

2.4. ENERGY BALANCE

Based on the results of the distributed power at the divertor and the selected divertor geometry, we estimated details of the radiation distribution from the upper and lower divertor radiating zones. Inboard and outboard radiating zones are approximated by elliptical cross-section rings for both divertors. A 3-dimensional toroidal geometry radiation model was used to account for all of the radiation from the divertors, core, and the corresponding heat fluxes to all the surfaces in the plasma chamber. The outboard divertor plates are receiving most of the power at a total of 227 MW. Two maximum heat flux peaks of 5.71 MW/m^2 (at the strike point) and 5.8 MW/m^2 (30 cm away and closest

Fig. 2-1. Cross-section of the ARIES-RS double-null divertor and corresponding inboard and outboard first walls.

Fig. 2-2. Details of the ARIES-RS divertor design with indication of design functions.

to the outboard radiating ring) were identified. The rest of the radiation distributed to the inboard divertor and first wall was also calculated and used as inputs to the following heat transfer calculations.

2.5. THERMAL HYDRAULIC DESIGN

The divertor plates have to be designed for a maximum heat flux of 5.8 MW/m^2 and an average value of 1.92 MW/m^2 . An additional requirement is to maintain a high outlet coolant temperature suitable for a high efficiency power conversion system because the power from the divertor region (surface heat flux+ nuclear heating) amounts to roughly 15% of the total thermal power. About 70% of the divertor power is coming as surface heat flux and 30% as volumetric heating. In order to maximize the heat removal capacity of the divertor plates, the total liquid metal coolant flow is routed first to cool the outboard plate, then the other divertor plates, and finally the back structure. In this way, the coolant continues to receive heat from the structure to reach the desired outlet temperature of 610°C , and the coolant exit temperature from the outboard plate is lowered to about 460°C , allowing a higher surface heat flux for a given structural material temperature limit. There is one inlet tube and one outlet tube for each divertor region. The inlet and outlet coolant manifolds are welded to the plates and connected to the divertor structure. By this routing scheme, the divertor maximum temperature will be lower than the design limit of 750°C . With a coolant inlet and outlet temperature of $330^\circ\text{C}/650^\circ\text{C}$, respectively, we are able to achieve the same coolant conditions as the lithium coolant blanket.

There are three critical locations at the divertor plates in terms of temperatures and stresses. One of them is the point where the maximum surface heat flux strikes the plate. The inlet manifold has to be arranged as close as possible to this point in order to minimize the temperatures here. The second critical location is the point 30 cm away from strike point where the heat flux peak of 5.8 MW/m^2 was identified. The other critical location is the coolant exit where the coolant temperature is increased to 610°C . This limits the tolerable length of the divertor plates in the poloidal/radial direction.

Thermal hydraulic parameters for one of 32 divertor modules are summarized in Table 2-1.

Table 2-1
Thermal Hydraulic Parameters For 1/32 Divertor Module

Coolant	Li
Structural material	V-4Cr-4Ti
Armor	W
Thermal power from divertor plate surfaces, MW	10.88
Thermal power from nuclear heating, MW	4.75
Total thermal power of 1/32 divertor, MW	15.63
Total area of divertor surface, m ²	5.67
Average surface heat flux, MW/m ²	1.92
Coolant inlet/outlet temperature, °C	330/610
Total coolant volumetric flow rate, m ³ /s	2.73x10 ⁻²
Coolant velocity in inlet/outlet tubes, m/s	1.0
Coolant velocity in manifolds, m/s	<0.5
Outboard plate surface area, m ²	2.10
Average surface heat flux, MW/m ²	3.39
Thermal power from heat flux, MW	7.12
Thermal power from nuclear heating, MW	0.65
Total thermal power of outboard divertor plate, MW	7.77
Coolant channel dimension, mm	4.0x8.0
Divertor plate total thickness, cm	5.0
Length of outboard divertor plate, m	1.12
Total number of coolant channels	190
Average coolant channel velocity, m/s	4.74
Temperature at Three Critical Locations:	
1. At the strike point	
Surface heat flux, MW/m ²	5.71
Maximum temperature in V-alloy tube wall, °C	527
Maximum temperature in W armor, °C	625
Maximum temperature in V-alloy local shield, °C	516
2. 30 cm from strike point	
Surface heat flux, MW/m ²	5.80
Maximum temperature in V-alloy tube wall, °C	681
Maximum temperature in W armor, °C	771
Maximum temperature in V-alloy local shield, °C	538
3. At the coolant exit	
Surface Heat flux, MW/m ²	0.62
Maximum temperature in V-alloy tube wall, °C	517
Maximum temperature in W armor, °C	528
Maximum temperature in V-alloy local shield, °C	632

2.6. THERMAL STRESS ANALYSES

Thermal stress analyses of the ARIES-RS divertor concept were completed using 2D and 3D finite element structural models input to the COSMOS and ANSYS codes, respectively. The 2D models were used primarily to provide design guidelines for reducing the maximum thermal stress intensities. The design recommendations resulting from the 2D models include cutting vertical slits through the tungsten and vanadium between coolant channels and castellating the 2 mm tungsten layer in the poloidal and toroidal directions of the divertor. Also, it was determined that the divertor should be supported in such a way to allow for thermal growth and rotation of the cross-sections while maintaining capability to react halo current loads. These recommendations were input as boundary conditions to a 3D model of a single coolant channel of the outboard divertor. A steady state heat transfer analysis was performed to calculate the thermal gradients for input to the 3D structural model which included the effects of a cracked and uncracked tungsten layer. From both 2D and 3D results, the maximum stress intensities exceeded the elastic stress allowable for the vanadium of 330 MPa. However, the results for the castellated tungsten case indicate that the design can satisfy the inelastic criteria specified in code case N-47-29 of the ASME boiler and pressure vessel code. This code case provides an alternate criteria when the criteria for elastically calculated primary plus secondary stresses exceed the 3 Sm limit. Other outstanding structural concerns about the divertor design are: (1) the residual stresses developed during the process of bonding tungsten to vanadium, (2) stress singularities at free edges of the tungsten/vanadium interface which may result in debonding failures, and (3) design of the divertor support to allow unrestrained thermal expansion of the divertor and to adequately transmit halo current loads to the vessel.

2.7. PARTICLE EXHAUST AND VACUUM SYSTEM DESIGN

The ARIES-RS design has a fusion power of 2167 MW which means that $7.7 \times 10^{20}/s$ of T and D are burned. Given the burn fraction of 0.304, with no recycling, ion fluxes for T, D, He and Ne in the exhaust system are:

$$T = 1.8 \times 10^{21}/s, D = 1.8 \times 10^{21}/s, He = 7.75 \times 10^{20}/s \text{ and } Ne = 1.36 \times 10^{20}/s$$

The corresponding molecules after neutralization becomes the vacuum pumping design inputs. Helium exhaust and transport studies on TEXTOR-94 [2] have shown that a pumping rate amplification of a factor of 5 is needed to maintain the He ash

concentration at the desired level. The ITER vacuum system has also adopted this technique. A factor of 5 increment means that the pumping system must be capable of evacuating 4.405×10^{21} molecules/s for each D_2 and T_2 , 3.8×10^{21} molecules/s of He and 6.8×10^{20} molecules/s of Ne. Using the average divertor plate surface temperature of 650°C , the total throughputs of 1268 Torrℓ/s was obtained.

These neutral particles, which collect in the divertor throat, are pumped through the divertor slots provided on the inboard and the outboard sides of the divertor plates. Fig. 2-2, shows an inboard slot located at $R = 3.89$ m and an outboard slot located at $R=4.74$ m. Similar slots are located on the top and bottom divertors. The neutrals, after passing through the slots, collect in a plenum located behind the central divertor plate. The plenum is toroidally continuous and has exhaust ducts located in each blanket/shield sector, 16 on the top and 16 on the bottom. Since the pumping station is located in the reactor basement area, the neutrals which are exhausted above the top divertor must be ducted to the bottom, which is an additional impedance over that of the lower divertor. To compensate for this, the lower divertor duct impedance will be made higher than its complementary upper divertor duct. In this way, the total impedance for both upper and lower divertor will be equal.

Finally, the gases must be directed to the pumping station through vacuum vessel ducts. The vacuum system selected for the ARIES-RS power plant is relatively conventional. The primary pumps used are compound cryopumps. The system is flexible to accommodate a lower burnup fraction which would increase the DT exhaust rate and decrease the He exhaust rate. This can be accomplished by minor increases in the cryopump surface area and increases in the conductances of the various ducts, giving a higher effective pumping speed.

2.8. REFERENCES

- [1] L. Barr and B.G. Logan, *Fusion Technology*, **18**, 251 (1990) 3.
- [2] P. Zwicker *et al.*, "Helium Exhaust and Transport Studies in the TEXTOR-94 Tokamak," Oak Ridge National Laboratory, 3rd International IEA Workshop on Helium Transport and Exhaust, Charleston, South Carolina (1995).

2.9. MEETINGS

- 1. 12th Topical Meeting on the Technology of Fusion Energy, Reno, Nevada, June 1996.

2.10. PUBLICATIONS

C.P.C. Wong, *et al.*, "Erosion Results from the DiMES and Design Implications to ITER and Starlite," 12th Topical Meeting on Technology of Fusion Energy, Reno, Nevada, June 1996.

3. PLASMA INTERACTIVE MATERIALS

3.1. INTRODUCTION

The Divertor Material Evaluations Studies (DiMES) hydraulic mechanism allows insertion and retraction of graphite samples into the divertor floor of DIII-D. The samples are implanted with a Si depth marker in order to measure the net erosion or redeposition of the graphite [1,2] Thin (100 nm) metal films of beryllium, vanadium, molybdenum and tungsten were deposited on the samples to study the erosion, transport and redeposition properties of these trace metals in the all carbon plasma-facing environment of DIII-D. The shaping control and well diagnosed divertor of DIII-D allows the sample to be exposed to well characterized steady-state outer strike point plasmas over several discharges. In FY96, we continued to perform erosion and redeposition experiments under normal and disruptive operations. We have also coordinated measurement and modeling results. Our measured carbon physical erosion rate, which is about a factor of 10 higher than ITER had assumed, can have strong impact on the selection of ITER PFC materials.

3.2. EROSION RESULTS AND MODELING

Recent results of measured graphite erosion from DiMES were put together. Figure 3-1 shows that the net loss of the carbon increases with increasing incident heat flux. These results were obtained during ELM-free and ELMing plasmas using both depth-marking [3] and colorimetry [4] techniques. The REDEP code [3,5,6] has calculated a gross erosion of carbon which was 5 times the measured net erosion rate for the ELM-free case with an incident heat flux of 0.7 MW/m^2 , corresponding to a redeposition rate of 80%. At higher heat flux the net erosion is larger than the extrapolated value using this redeposition rate, indicating that the redeposition rate could be decreasing. This is despite the expected increase in local redeposition due to the increased divertor plasma density at the higher heat flux. The effect of ELMs on net carbon erosion appears to be small compared to the erosion from the quiescent plasma. Modeling indicates that although the instantaneous erosion rate is about three times higher than during the quiescent phase, the small duty factor (3%) of the ELMs means that they have a 10% effect of net erosion. This has been verified by the similarity of ELM-free and ELMing plasma's erosion measurements at the same heat flux. A comparison between the DiMES measured net erosion rates for DIII-D and the predicted

carbon erosion rates currently used for ITER have shown that the experimental measurements are a factor of 10 higher than the calculation. However, the agreement between the DiMEs results and REDEP is quite good. These differences are under investigation, but are believed to be caused by differences in redeposition efficiency, self-sputtering, and the effect of oblique angles of incidence on sputtering yields. These differences are important for estimating the lifetimes of the ITER divertor plates.

Fig. 3-1. Measured net carbon loss rate at DIII-D outer strike point versus incident heat flux measured by infrared thermography.

3.3. METAL EROSION

Metallic erosion was examined after the exposure of Be and W coated samples. Significant amounts of arc tracks are present on tungsten films after plasma exposures and these are contributing to the measured loss rate of the tungsten. The exact cause and phenomenology of this arcing is under investigation.

The toroidal redeposited pattern of the metals has shown that the e-folding length of the metals decreases with increasing atomic number, and also increases ionization rate. This behavior is in agreement with predicted sputtering yields for these materials. The result for W is shown in Fig. 3-2. The WBC code [5] Monte-Carlo simulation obtains good agreement with the experimental data indicating that an adequate understanding exists of the sputtering geometry, the ionization processes and the ion trajectories. The sputtering rate of Be into the divertor plasma has been measured via visible line

spectroscopy and supported by divertor plasma modeling. This also indicates that the diverted plasma has been well characterized during the steady-state erosion processes. The loss rates of the metal films are difficult to interpret due to the highly perturbing effect of the 1%–2% carbon background plasma eroding and depositing onto the metals. Surface analysis by Rutherford Backscattering (RBS) has indicated that a significant amount of material mixing could be taking place during the exposure (e.g. the Be “soaks” into the graphite), making the problem even more difficult.

Fig. 3-2. Comparison of experimental (dots) and WBC code calculations (line) of toroidal distribution of redeposited Tungsten from ELM-free H-mode exposure.

3.4. ANALYSIS FROM SANDIA NATIONAL LABORATORY

Post-exposure ion-beam analysis was done on another Be/W coated sample to determine erosion and deposition of carbon, beryllium and tungsten. This sample was exposed to the outer strike point for 18 s during steady state ELMing H-mode with 2.5 MW of neutral beam heating. The sample had stripes of tungsten and beryllium spanning the strike point, allowing erosion of these metals to be observed as a function of

position across the strike point. A 100 nm thick tungsten metal film was 10% to 20% removed by the plasma exposure. Eroded tungsten was found to be redeposited on neighboring carbon surfaces very close to its point of origin in agreement with earlier results. Erosion of beryllium varied across the strike point from about 42% to 76% of the film, which was initially 90 nm thick. After exposure, the beryllium was found to be mixed with carbon to depths much greater than the initial thickness of the film. The reason for this is uncertain. The spatial distribution of redeposited tungsten and beryllium was mapped. Peak carbon erosion was 50 nm. These results show that metal erosion rates are much higher than previously measured for ELM-free H-mode plasmas. Coupled with the plasma parameters of this experiment, modeling effort at ANL will try to further understand these results. We will continue to examine the chemical composition of this Be and C mixture.

3.5. DISRUPTION CHARACTERIZATION

During the FY96 DIII-D disruption experimental campaign, we attempted to perform the first phase melt layer studies during tokamak disruptions. This entailed exposing a thin (10 μm) aluminum layer on a graphite sample to the outer strike point of a radiative disruption. A previous DIII-D disruption experiment had measured a sufficient temperature ($> 1000^\circ\text{C}$) to melt the Al near the DiMES probe location. Following is a brief summary of the three attempts at performing this experiment in the past fiscal year.

In June four dedicated shots using an argon gas puff to disrupt the plasma were attempted. However, the patch panel and/or plasma shape did not allow us to move the outboard strike point (OSP) to the DiMES location. Therefore, the disruption's heat flux "footprint" missed DiMES. The edges of the Al film showed some slight effects of the limited heat flux which it received.

In August, disruptions were "piggybacked" at the end of the current flattop of another experiment (puff and pump). Problems were encountered in getting the plasma to disrupt promptly using the Ar puff, and again, the IR camera showed no sign of significant heat flux and/or temperature increase to the DiMES location. The sample was therefore not inserted.

Again in August, disruptions were again piggybacked onto the end of the current flattop. This experiment was a q_{95} scan to study transport. This meant that plasma conditions and OSP control were varied between most shots. Once again, we were unsuccessful at moving the OSP to the DiMES location (coil current limits) and no significant temperature increase to the DiMES probe was measured.

In summary, we were not able to melt the Al layer with disruptions due to difficulty in controlling the location of disruptive strike point. These experiments helped us to plan for the next series of experiments.

1. We MUST have dedicated machine time to tailor the discharge shape and the disruption to maximize our chances of steering the disruption footprint towards the DiMES location.
2. We recognize that producing reproducible disruption heat loads is very difficult. Only repeated attempts can achieve our goal.

A depth-marked graphite coupon was exposed to several VDE (Vertical Displacement Event) disruptions. A net redeposition of 60–90 nm was found on the sample.

3.6. CHARGE-EXCHANGE NEUTRAL MEASUREMENTS

The DiMES probe head equipped with a Si-wafer collector behind a slotted mask is used to collect charge-exchange (CX) neutrals at the divertor floor. Previous interpretations of this data during disruptions led us to believe that there was a substantial amount of shielding due to the cold neutral layer in front of the plate. However, more recent analysis of a non-disruptive discharge has shown that the CX flux collected during the disruption experiment was probably from the background quiescent plasma, rather than the disruption. This leads us to the possibility of using this diagnostic during detached plasma operation to characterize the flux and energy of CX neutrals impinging on the wall. The effect of the neutrals in high-power-flux reactor divertors will be very important.

3.7. REFERENCES

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2. 12th Topical Meeting on the Technology of Fusion Energy, Reno Nevada, June 1996.

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4. SiC/SiC COMPOSITE MATERIAL DEVELOPMENT

4.1. ADVANCED COMPOSITE FABRICATION AND EVALUATION

A fabrication method was devised to allow for rapid production of small composite disks. A number of composite coupons were fabricated under the same conditions while varying the type of reinforcing fibers and the interfacial coating. The samples were circular with a diameter of ~5 cm and a thickness of ~2 mm. Absolute properties were not expected to be as good as those achieved using the more complex large chemical vapor infiltration (CVI) reactor, but a relative comparison between samples would allow for an evaluation of the reinforcing fibers.

Three types of Nicalon fiber were received from Dr. Gerald Youngblood of Battelle Pacific Northwest Laboratories (PNL). The three types were as follows:

1. Ceramic Grade Nicalon
2. High Grade Nicalon
3. Nicalon Type S (super)

About 20 g of each type of fiber was received. Under direction from PNL and Oak Ridge National Laboratory (ORNL), a unidirectional lay-up for the samples was employed.

A graphite tool was designed and fabricated that provided a unidirectional fiber preform that was densified with SiC matrix in the CVI reactor as shown in Fig. 4-1.

Fig. 4-1. Schematic of chemical vapor infiltration reactor.

Six samples, as summarized in Table 4-1, were fabricated and sent to PNL for evaluation.

Table 4-1
Samples Evaluated at PNL

Sample	Reinforcement	Interfacial Coating
10841-83-1	CG Nicalon Cloth	~0.2 μm C
10841-83-4	High Nicalon	~0.2 μm C
10841-83-7	Super Nicalon	~0.2 μm C
10841-86-8	Super Nicalon	~0.2 μm C
10841-87	High Nicalon	~0.4 μm C

Sample 10841-83-1 was made with 2-D cloth to establish reference infiltration conditions. Sample 10841-83-3 was the first unidirectional fiber reinforced coupon and had poor fiber architecture. Sample 10841-83-7 had a surface delamination and was fabricated again as sample 10841-86-8. Sample 10841-87 was made with high nicalon and a ~0.4 μm thick carbon interfacial coating.

4.2. CHARACTERIZATION

Microstructural evaluations of samples 10841-83-4, 10841-83-7, and 10841-83-8 were performed at PNL. Samples were mounted in epoxy and polished down to 1 μm diamond polishing compound.

A scanning electron micrograph of sample 10841-87 is shown in Fig. 4-2. At 4500 $\times$, a $\sim 1\ \mu\text{m}$ "gap" is apparent between the high Nicalon fiber and the matrix. Also, a ring of matrix material about 1.8 μm thick has different contrast than the rest of the matrix material. Some of the polishing grit appears to be lodged in the gap. This gap is very unusual and is only observed in the high Nicalon samples. It is too large to be due to the small difference in coefficient of thermal expansion (CTE) between the fiber and matrix. The CTE for SiC matrix is between 4.0 and $4.9 \times 10^{-6}/^{\circ}\text{C}$ while the CTE for the high Nicalon is between 4.0 and $3.5 \times 10^{-6}/^{\circ}\text{C}$. The CTE for the matrix is greater than that of the fiber and should shrink more than the fiber, applying a compressive force on the fiber upon cooling to room temperature. One possibility is that the particular fiber lot was unstable and consolidated under the CVI conditions, resulting in a volume shrinkage. The gap phenomenon was not evaluated further.

Fig. 4-2. Scanning electron micrograph (4500 $\times$) of polished cross-section sample 10841-87.

An optical micrograph (500 $\times$) of sample 10841-87 is shown in Fig. 4-3. The edge of the matrix material near the fibers and the fibers themselves are not well resolved. There appears to be the same gap as observed in Fig. 4-2, but it is difficult to quantify.

Hi Nic (400 μ mc) /CVI-GA, Qpt $\times$ 500, Hand Polish 6 μ m

Fig. 4-3. Optical micrograph (500x) of polished cross-section of sample 10841-87.

Figures 4-4 and 4-5 are scanning electron micrographs of a cleaved surface of sample 10841-87. The fibers are only loosely held by the matrix and can actually move or slide through the thickness of the sample. Chemical analysis by electron dispersion of X-rays (EDX) was performed in the three areas indicated in Fig. 4-4. These are shown in Figs. 4-6-4-8. All three spectra look similar. Carbon is not resolved under the conditions employed.

Figure 4-9 shows an optical micrograph at 50x of sample 10841-83-4. Some large interfiber pores are apparent that may be responsible for the low density of the sample.

A scanning electron micrograph of sample 10841-86-7 is shown in Fig. 4-10. This sample is reinforced with high Nicalon fibers and may have a small gap between the fibers and matrix. The matrix appears to be dense and uniform.

4.3. INTERFACIAL COATINGS

Silicon carbide interfacial coatings were fabricated from methyl silane. The goal was to deposit porous coatings to achieve a toughening of the composite. Initial coatings were dense with little porosity. Subsequent coatings yielded a porous structure.

A scanning electron micrograph of the porous coating on Nicalon type S is shown in Figs. 4-10-4-12. In Fig. 4-12 part of the coating appears to have been removed, perhaps by adhering to an adjacent fiber. This behavior is indicative of a good interfacial coating of porous silicon carbide.

Fig. 4-4. Scanning electron micrograph of cut surface of sample 10841-87. Numbers indicate areas analyzed by EDX.

Fig. 4-5. Scanning electron micrograph of cut surface of sample 10841-87.

E2734 GA Hi Nic (400 nmC) #1

Fig. 4-6. EDX of area 1 in Fig. 4.

E2734 GA Hi Nic (400 nmC) #2

Fig. 4-7. EDX of area 2 in Fig. 4.

E2734 GA Hi Nic (400 nmC) #3

Fig. 4-8. EDX of area 3 in Fig. 4.

10841-83-4 (Hi Nic / Cul GA 200nmc) 50 ×

Fig. 4-9. Optical micrograph (50x) of polished cross-section of sample 10841-83-4.

Fig. 4-10. Scanning electron micrograph (3000x) of polished cross-section of sample 10841-83-4.

Fig. 4-12. Scanning electron micrograph of Nicalon fiber with porous silicon carbide coating.

Fig. 4-11. Scanning electron micrograph of Nicalon fiber with porous silicon carbide coating.

One 20 g spool of Nicalon type S fiber was coated with porous silicon carbide and sent to Dr. Gerald Youngblood at PNL.

4.4. COMPOSITE PANEL FABRICATION

We had a meeting with Professor Kohyama of Kyoto University. He discussed his thoughts on the Japanese/DOE fusion materials collaboration, and indicated that they are more in favor of fundamental studies at this time than DOE. In particular he indicated it was premature to proceed with the proposed pressurized creep tubes. He would like to receive more SiC composite material fabricated at GA. A quote for fabrication of a SiC panel was provided to Kohyama. After discussions between Russ Jones and Kohyama, Russ proposed a collaborative program where GA and ORNL would provide materials for testing. Initially, the fiber architecture consisted of 2D high Nicalon fabric with a 0°, 30°, 60° fiber lay-up. This architecture requires a multiple of three layers of fabric. For thickness requirements it was decided to proceed with a 9 layer structure. Discussions on fiber architecture and number of layers are in progress. GA received a sample of high Nicalon fabric, 0.5 m × 1 m. The sample was coated with ~0.3 μm of carbon in a chemical vapor deposition reactor from a propylene precursor. Once the fiber architecture has been resolved, the fabric will be cut and a SiC composite panel fabricated.

A small sample of SiC composite was sent to Dr. Sato of the University of Tokyo for internal friction (IF) experiments. This material was taken from GA inventory. It was fabricated from commercially carbon coated CG Nicalon and consequently had poor toughness properties but would be suitable for IF testing. The sample was improperly packaged and broke during transit. A second sample was packaged more carefully and sent.

5. MAGNETIC DIAGNOSTIC PROBES

5.1. INTRODUCTION

ITER and other future tokamaks require magnetic diagnostics for the control and operation of the tokamak and for understanding plasma behavior. The magnetic probes must be located relatively close to the plasma and survive the neutron flux and fluence with acceptable noise characteristics. To address this issue two prototype magnetic probes, an equilibrium coil and a fast response coil, and complete instrumentation based on the baseline ITER magnetic probe design have been built and tested in the HFBR reactor.

A prototype of the coils that will be used for equilibrium reconstruction in ITER along with a long pulse integrator underwent a flux test in HFBR. A radiation induced EMF across the sensor leads was observed that is larger than acceptable for ITER. We have proposed several possible solutions for ITER to this problem.

A prototype fast response coil underwent a fluence test in HFBR. A neutron flux-related, turn-to-turn resistive short was observed that will affect the design of the ITER fast response coils.

5.2. EQUILIBRIUM COIL TEST

A prototype magnetic sensor was built based on the ITER magnetic diagnostic baseline design but scaled to fit into the HFBR reactor (Fig. 5-1). The magnetic sensor, instrumented with a long pulse integrator (LPI) developed under this contract, was inserted into HFBR. A unacceptably large radiation induced EMF was observed across the sensor leads. Further work is required to demonstrate a solution to the radiation induced EMF.

The magnetic sensor that was developed consists of mineral insulated (MI) coax formed around an alumina ceramic spool as shown in Fig. 5-1. This sensor is similar in construction to the inductive coils in the ITER baseline design that will be used for magnetic equilibrium reconstructions (we will refer to the prototype magnetic sensor shown in Fig. 5-1 as the equilibrium coil). The requirements for ITER for the equilibrium coils are that the radiation induced drift, by any mechanism, in the integration of the signal across the leads be less than 1 mV-s for 1000 s in a radiation field that is between 0.1 and 1 times the first wall flux. In HFBR the neutron flux of 4.25×10^{14} n/cm²-s is

roughly the same as the expected ITER first wall flux. The gamma flux in HFBR is roughly 6 W/gm.

Fig. 5-1. Magnetic probe assembly drawing.

5.2.1. REACTOR TEST RESULTS

The measured drift across the equilibrium coil during the reactor test is shown in Fig. 5-2 both before the insertion into the reactor and with the coil in the radiation field. The triangles in Fig. 5-2 represent the baseline corrected data integrated for 500 s. As can be seen in Fig. 5-2, with one of the leads shorted to ground (common mode measurement), the drift is less than 1 mV-s with a resolution better than 1 μ V. However a differential measurement reveals that there is a radiation induced EMF across the leads that introduces a drift in the integration of about 46 mV-s and limits the resolution to about 46 μ V. This drift is well above the acceptable limit for ITER. The common mode measurement eliminates the EMF problems but is also unacceptable for ITER because of noise and the ground loop that is introduced. There are several possible paths to recover a working diagnostic for ITER. They include: 1) a partial bleed of the EMF coupled with increased conductivity of the central conductor; 2) compensation of the drift by using a dummy coil in the radiation field and; 3) materials and geometry selection to reduce the EMF. The first path is the most straight forward with the smallest impact on the ITER design. The second path has a major impact on the cost and complexity of the in-vessel

sensor package. The third path is somewhat uncertain and is being pursued by the Russians.

Fig. 5-2. Acceptable change in integrated signal in radiation field with EMF shorted to shield

The existence of the EMF across the sensor leads is a new unexpected result and is probably due to an imbalance in the geometry of the coil in relation to the radiation field. Figures 5-3, 5-4, and 5-5 show that this EMF is not a constant current source and that the EMF depends both on flux and fluence.

5.3. FAST MAGNETIC COIL TEST

A second prototype magnetic coil was built and tested in the HFBR reactor. The design of this second coil was based on the ITER magnetic diagnostic baseline design of the high frequency (or fast) coils. These coils are made of bare ceramic coated wire wound on a ceramic spool. These coils will not be used for equilibrium reconstructions. However, they will be embedded in the blanket modules close to the plasma in order to have reasonable frequency response and will need to survive at least as long as the blankets. Also, these probes will require electrical connectors between the MI cable and the bare wires that are close to the coils.

The prototype fast coil that was tested in HFBR is shown in Fig. 5-6 and is made from ceramic-coated, copper-clad nickel wire wound on a ceramic spool with electrical connectors between the MI feed cables and the spool.

Fig. 5-3. EMF not from a constant current source.

A fluence test of the fast coil could be made in HFBR since the expected fluence of the fast coil for the life of ITER is roughly 10^{20} n/cm² (a reduction of 100 over the first wall due to the blanket shielding) and 30 days in HFBR gives about 2.5×10^{20} n/cm².

5.3.1. REACTOR TEST RESULTS

The prototype fast coil was made up of two coaxial coils wound in two layers (see Fig. 5-6). This design allowed one coil to be used as an exciter for the second coil to verify the integrity of the coils *in-situ*. A radiation-induced short between the two coils was immediately observed that was flux dependent. Removal of the coils from the radiation field eliminated the apparent short, verifying that the radiation field was responsible. We believe that the radiation-induced short is due to radiation-induced charge carriers that are also responsible for the radiation-induced EMF between the conductors. The impact on the ITER fast coil design is that a larger spacing between the turns in the coils is required. The coil and the connectors suffered little degradation during the 30 day HFBR test.

Fig. 5-4. EMF depends on fluence.

Fig. 5-5. The radiation-induced EMF depends on flux.

Fig. 5-6. Coil assembly.

5.4. REFERENCES

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5.5. MEETINGS

1. 16th IEEE/NPSS Symposium on Fusion Engineering, Champaign, Illinois, September, 1995.
2. 11th Topical Conference on High Temperature Plasma Diagnostics, Monterey, California, May, 1996.

5.6. PUBLICATIONS

E.J. Strait, et al., "A Hybrid Digital-Analog Long Pulse Integrator," Proc. of the 11th Topical Conference on High Temperature Plasma Diagnostics, Monterey, California, May, 1996.

6. RF TECHNOLOGY

6.1. U.S./JAPAN BILATERAL EXCHANGE WORKSHOP

The U.S./Japan rf technology exchange originally scheduled to be held in Japan in December 1996 was canceled because U.S. funding limitations precluded sending any U.S. representatives to the meeting.

6.2 THIRD INTERNATIONAL WORKSHOP ON ELECTRON CYCLOTRON HEATING TRANSMISSION SYSTEMS

The Third International Workshop on Electron Cyclotron Heating Transmission Systems was held in December 1995. Thirty-three scientists and engineers from nine countries attended, made 20 presentations and engaged in spirited discussions on all aspects of ECH transmission systems. The proceedings were published and distributed.

6.3. ECH WINDOW DESIGN AND FABRICATION

6.3.1 110 GHz WINDOW

General Atomics fabricated a 4" x 4" 110 GHz distributed window which was delivered in September 1995 to Communications and Power Industries (CPI). Hot tests at CPI confirmed the power handling capability of the window. Tests were conducted with a reduced beam size at 200 kW with 0.7 s pulses without any arcing or excessive window temperatures. The power density and pulse length were equivalent to that in a full size 1.2 MW CW beam with a peak-to-average power ratio of 2.7. This window was assembled using a gold braze material to bond the sapphire strips to the niobium frame. The braze was successful except for small leaks at two locations, and re-braze efforts were unsuccessful.

Another window was made for the DIII-D program using thinner sapphire strips to reduce microwave reflection. The thinner sapphire was selected as a result of computer modeling at LLNL and analytic modeling at MIT which showed that the effective thickness of the sapphire strips is larger than the actual thickness. The window was not tested because there were gaps in the gold braze metal at the ends of each sapphire strip due to inadequate wetting with the sapphire.

The sapphire strips from the above window were removed, and the frame was used again in another attempt to make a window for mounting on a DIII-D gyrotron. This

window used copper braze material rather than gold. The copper braze technique was developed under the 170 GHz window task described below. A photo of this window mounted in the interface hardware is shown in Fig. 6-1. This window was tested at CPI, and the predicted losses of 4%–5% were confirmed. However, the window was damaged at the end of the test run. Testing with a reduced beam spot size demonstrated the equivalent of 500 kW for 350 ms. The sapphire strips were removed from the frame, and the window was reassembled and brazed again. For a variety of reasons, this window proved not to be leak tight. Based on what has been learned from the window fabrication experience, there is high confidence that a good leak-tight window could be made by starting with a new frame and by using copper braze material. Due to funding limitations and schedule requirements for gyrotron installation at DIII-D, no further work is presently planned on the 110 GHz distributed window.

Fig. 6-1. 110 GHz distributed window.

6.3.2 170 GHz WINDOW

Because of the difficulties encountered in brazing the 110 GHz windows using gold, an improved technique was developed using copper as the braze material. Some of the difficulties with the 110 GHz windows were due to the relatively high solubility of

niobium in the gold and the consequent inability to re-braze. Metallography and SEM analysis on niobium/copper braze interfaces show that there is very little dissolution of niobium into the copper and little penetration of the copper into the niobium. As a result of this work, copper is now considered to be the preferred braze material for distributed window fabrication.

During FY95, under GA funds a new modular window design for low-loss, wide-band performance was developed. In this approach, niobium vanes are made individually and brazed into a niobium frame at the same time as the sapphire strips are brazed into the frame. The vanes are designed to match the impedance as the microwaves pass through the window. This results in a traveling wave in the sapphire rather than a standing wave and consequently reduces the microwave losses. The new configuration also broadens the frequency range over which the window reflection is acceptably low. During FY96 the niobium vanes, frame and sapphire strips for a 1.25" x 1.25" 170 GHz prototype window were procured, but the window has not yet been assembled. Tapers needed for testing the prototype window in Japan were also procured.

Sapphire strips for a full-size 170 GHz distributed window were ordered and received during FY96. Two vendors successfully fabricated the 55 niobium vanes for the full-size window, and the niobium frame was designed and ordered. Additional funds are needed to coat the parts and to assemble, braze and test the window.

6.4. JAERI CRYO WINDOW TESTS

A JAERI-provided cryogenically-cooled sapphire window designed for 110 GHz transmission was tested at General Atomics using the 1 MW 110 GHz Russian gyrotron. Two JAERI members, Dr. Atsushi Kasugai and Mr. Mikio Saigusa, participated in the testing. The objective of this testing was to investigate the performance of C-surface cut sapphire at cryogenic temperatures. Tests were performed up to the present limits of the gyrotron of 500 kW and 500 ms. Test runs were made with a constant power level of 500 kW for pulse lengths from 100 ms to 500 ms, and for constant pulse length of 500 ms with rf power levels increasing from 200 kW to 500 kW in 100 kW increments.

The preliminary results are that the dielectric loss tangent is approximately 3×10^{-5} over the temperature range of 10 to 20°K. This is a factor of 2 better than an A-surface cut sapphire disc measured in Japan.

6.5. JFT-2M COMBLINE ANTENNA

The comblime antenna designed in FY95 was fabricated and installed on the JFT-2M tokamak early in 1996. A photograph of the completed twelve-element array and the pair of associated limiters as installed in the JFT-2M vacuum vessel is shown in Fig. 6-2. Tests at JFT-2M in April 1996 demonstrated that the comblime antenna system requires no adjustments, *i.e.*, no tuners are required. The tests also confirmed that the concept minimizes the feedline voltage and current, so that transmission reliability is maximized and transmission losses are minimized.

Fig. 6-2. The completed twelve-element comblime antenna array and the pair of associated limiters installed in JFT-2M.

The antenna was designed to couple up to 0.8 MW at 200 MHz (the existing transmitter at JFT-2M consists of four 0.2 MW modules). For the initial set of

experiments at JFT-2M, two of the 0.2 MW modules were connected to the antenna, one at each end. A directional spectrum could be produced by operating one transmitter only. A non-directional spectrum could be produced by feeding both ends simultaneously, so 0.4 MW could be applied to the antenna for testing its voltage limits.

The input impedance of the combline was well-matched to the transmission line during all conditions studied, including ohmic, neutral beam heated L-mode, and ELMing H-mode plasmas. This is illustrated in Fig. 6-3, where 1.8 MW of neutral beam heating sustains an ELMing H-mode plasma.

Fig. 6-3. Power into combline, reflected power from the input (zero), and the power not coupled to the plasma after one pass through the combline, P_{trans} , for a JFT-2M discharge with ohmic (600–650 ms), L-mode (650–665 ms), ELM-free H-mode (665–720 ms), and ELMing H-mode (720–850 ms) conditions.

Although the power levels injected into JFT-2M in these experiments were small compared to the multi-MW levels used on large tokamaks (because of the small size of the combline), the rf electric fields sustained without breakdown in these experiments actually exceeded the ITER electric field design criteria. Future experiments using the

JFT-2M combine will use hybrid junctions to combine the four transmitter modules to work towards the demonstration of 0.8 MW operation.

6.6. HYBRID TUNING SYSTEM

The Hybrid Tuning System (HTS) is a high power ferrite-based rf matching device which will allow real-time matching of an rf generator to a dynamic load. The DIII-D HTS is scheduled to be completed in FY97. Upon completion at the vendor in Germany, the unit will undergo high power acceptance tests at IPP Garching. The IPP is also quite interested in this new technology. These acceptance tests will take place in early 1997.

Because of the limited run time available on DIII-D, initial plasma testing of the HTS will also take place at IPP Garching on the ASDEX-U tokamak, so that a timely assessment of the capability of the unit can be made.

6.7. MEETINGS

1. ECRF Working Group Meeting, Moscow, March 1996.
2. 19th Symposium on Fusion Technology, Lisbon, Portugal, September 1996.

6.8. PUBLICATIONS

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