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FUSION TECHNOLOGY DEVELOPMENT
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U.S. DEPARTMENT OF ENERGY

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by
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Section 1

FUSION TECHNOLOGY DEVELOPMENT OVERVIEW

1. FUSION TECHNOLOGY DEVELOPMENT OVERVIEW

In FY95, 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 Studies (Section 2), DiMES (Section 3), SiC Composite Studies (Section 4), Magnetic Probe (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.

Section 2

FUSION POWER PLANT DESIGN STUDIES

2. FUSION POWER PLANT DESIGN STUDIES

2.1. OVERVIEW

In 1995, the goal of fusion power plant design studies was to provide critical assessment of the projected performance of the demonstration power plant (Demo), of the commercial power plant, and of systems and components; to develop potential solutions to critical issues; and to offer improvements. This activity helps to define the goals and R&D needs for the fusion program and thus adds focus to near-term activities.

GA continued to play a lead role in the Starlite project. We led the evaluation of the Demo helium-cooled blanket options and the divertor physics and engineering design. We provided support in the areas of critical issues and design requirements identification, safety and licensing, and power conversion system evaluation for the Demo design.

2.2. DEMO-DESIGN

2.2.1. WORK BREAKDOWN STRUCTURE

The work began in FY94 and continued. We completed the preparation of the Demo Work Breakdown Structure (WBS).

2.2.2. CRITICAL ISSUES

We coordinated the identification of critical issues in the areas of blanket/first wall, current drive, plasma facing components and superconducting magnet designs.

2.2.3. BLANKET EVALUATION AND DESIGN

A design evaluation was performed on 18 MPa helium-cooled, V-alloy blanket options. The configuration shown in Fig. 2-1 was developed to allow a coolant outlet temperature of 650°C. The coolant can then be fed directly to a direct cycle gas turbine power conversion system, resulting in a gross efficiency of about 47%. Among the breeders, Li_2O , Li and a mixture of Li_2O and Li were evaluated. The Li breeder was selected for further study to avoid the problems related to the use of Li_2O : radiation damage, contact resistance for the solid form and mass transfer for the mixture form.

Fig. 2-1. 1st wall and blanket tubing arrangement.

These design options all have the capability of breeding adequate tritium without the use of a neutron multiplier while meeting all the material design temperature limits. The safety advantage of this helium-cooled, vanadium alloy, lithium breeder (He-V-Li) design over the lithium self-cooled design remains, due to the fact that no liquid metal primary or secondary loop will be needed. Critical issues of materials compatibility and reliability of a high pressure gas design were identified and addressed.

Based on the same configuration, we found that this He-V-Li blanket concept can handle neutron wall loading in the range of 9 to 11 MW/m². We also assessed the use of ferritic steel for this blanket concept, and found that this combination of material can handle neutron wall loading in the range of 6 to 8 MW/m².

2.2.3.1. Materials Compatibility

Material compatibility between helium impurities and V-alloy was identified as a critical issue for the He-V-Li blanket concept. We proposed to address this issue by using a highly efficient impurity extraction system, by limiting the ingress of impurities, by using V-alloy surface modification and by developing an oxidation resistant alloy.

For surface modification, V4-Cr4-Ti surface with a 10 µm thick layer of either silicon, aluminum or yttrium is believed to have a high chance for success in preventing structural material weakening at low concentrations of O₂, N₂, H₂O, and H₂ impurities at temperatures to 650°C. This observation is based on preliminary experimental work carried out at ANL and ORNL. Our Japanese colleagues have already carried out scoping studies on oxidation resistant V4-Cr4-Ti alloy containing Si, Al, and Y additives to enhance chemical compatibility.

2.2.3.2. Reliability Of High Pressure Helium System Design

Another critical issue identified is the reliability of the high pressure helium system. We analyzed and proposed the leak-before-break assessment method to make sure that the probability of bursting is low. In the unlikely event of tube bursting accident, we proposed a multi-barrier structural design approach and fast response shut-off valves to limit the propagation of blanket component damage.

2.2.4. PLASMA FACING COMPONENTS DESIGN

2.2.4.1. Divertor Physics

We performed scoping level analysis of heat exhaust for the Demo divertor strawman design to reduce the maximum surface heat flux from 80 MW/m² to about 5 MW/m². It was concluded that enhanced impurities radiation in the plasma core mantle, the scrape-off-layer and the divertor region may be a possible approach. Suitable impurities for this application are Ne, Ar or Kr.

2.2.4.2. Divertor Engineering

Evaluation was performed on the helium-cooled and lithium-cooled divertor designs to handle a maximum surface heat flux of 5 MW/m². Both designs used V-alloy as the structural material. Results show that, by assuming the integrity of the MHD insulation layer and by using a tube size of 10 mm in diameter, the lithium-cooled design can handle this heat flux with a coolant outlet temperature of 450°C, which corresponds to a gross efficiency of about 39%. For similar thermal performance, in order to meet the temperature and thermal stress limits, the helium design will have to use smaller diameter tubes (6 mm).

2.3. SAFETY AND LICENSING

Top level Demo safety requirements were derived from qualitative policy statements for the public and for the workers. These requirements were categorized as direct measure of consequence or risk, generic events and those that are quantitative. The categories were subdivided into normal and off normal events. For defining design limits, a two-tier approach was used: a rigid required-by-law limit and a more stringent project specific goal or target limit that can dictate design configurations and materials selections.

In response to the task of describing a logic for designing a fusion facility such that public evacuation drills are not required, the historical and current perspectives were reviewed. Based on NRC regulations, a logic based on the iteration between design analysis and safety evaluation was derived. We then proposed a hazard assessment approach for the evaluation of the ARIES-IV and ARIES-II designs. Several initiating

events that could result in radioactivity releases were identified, including Loss of Coolant Accidents and Loss of Flow Accidents.

In order to demonstrate a method for assuring ARIES-IV compliance with the requirement that Protective Action Guides (PAG) not be exceeded above a certain frequency, we proceeded to develop an uncertainty distribution for offsite doses following a helium coolant line break. MACCS code was used to account for weather variability in performing probabilistic offsite dose analysis. Tritium is expected to be the dominant contributor to dose. A dose estimate was made for the reference EPRI Advanced Light Water Reactor (ALWR) site which encompasses the weather of 80% to 85% of all U.S. sites. The results showed that ARIES IV may meet the PAG requirement depending upon the fraction of HT that converts to HTO as it is transported through the environment. Potential design changes were recommended that include ways to reduce the accident source terms, to mitigate accident releases, and to reduce likelihood of the accident.

2.4. POWER CONVERSION SYSTEM EVALUATION

An evaluation of the power conversion system was performed. Various coolants (Li, helium, water, molten salt and organic) and structural materials (ferritic steel, vanadium alloy and SiC-composite) were considered in combination with various basic system arrangements (direct, indirect and indirect + intermediate loop cycles). The conclusion was that helium-cooled/SiC-composite/Brayton cycle system has the highest thermal efficiency potential of 52%, the lithium/V-alloy/rankine cycle system can achieve 45%, the Ferritic-steel/lithium system can achieve 37%, and the ferritic-steel/water system cannot achieve the design goals of removing high surface heat flux at the divertor while allowing a high enough coolant outlet temperature for efficient power conversion.

A projection was made on the future cost of electricity and the required gross thermal efficiency for fusion power to be competitive in the year 2025. It was found that in order to reach an acceptable cost range of 60 to 65 mills/kWh, the power plant gross thermal efficiency will need to be in the range of 50% to 55%. At present, only the helium-cooled, SiC-composite Brayton cycle system can meet this requirement.

2.5. INTERNATIONAL EXCHANGE

2.5.1. RUSSIAN COLLABORATION

The Russian team at the Kurchatov Institute (Moscow) finished the helium-cooled divertor task, and a report was delivered. They completed the second task on assessment of vanadium-based alloys development in Russia and a report was completed and delivered. These reports were distributed to the Starlite team.

2.5.2. PRC EXCHANGE

Prof. G. Sheng of Southwestern Institute of Physics (SWIP), People's Republic of China, completed a six months visit at GA. During the visit he contributed to first wall design, initiated the Supercode activity at GA and supported the further development of the NEWT-1D one-dimensional scrape-off-layer transport code.

2.6. MEETINGS

We participated in and contributed papers to the relevant fusion power plant meetings.

2.7. PUBLICATIONS

C. Wong, B. McQuillan, R. Schleicher and E. T. Cheng, "Evaluation of U.S. Demo-1 Helium-cooled Blanket options," 16th IEEE Symposium on Fusion Engineering, Champaign, Illinois, September 30–October 1, 1995.

Section 3

PLASMA INTERACTIVE MATERIALS

3. PLASMA INTERACTIVE MATERIALS

3.1. OVERVIEW

Surface material erosion due to plasma-material interaction during normal operation and disruptive events will limit the performance and lifetime of tokamak plasma facing components, particularly the divertor, where large incident particle and heat fluxes are expected. The erosion phenomenon is a complex process involving gross sputtering due to physical and chemical sputtering, sublimation and evaporation, and redeposition due to plasma transport of ions back to the surface. High power flux due to transient events such as ELMs and disruptions may cause surface melting and bulk material transport. The primary goal of the Divertor Material Evaluation Studies (DiMES) program on DIII-D has been to provide data required for bench marking the codes presently being developed in a real tokamak geometry. In FY95 we provided basic materials erosion data resulting from exposure of well characterized samples to tokamak divertor plasma. These data were obtained during both normal, transient (*e.g.*, ELMs) and off-normal (*e.g.*, disruption) events. In addition, the DiMES system was used as a test vehicle of different advanced plasma facing materials (*e.g.*, V-alloys and coatings) and as a carriage for unique plasma diagnostics that cannot be installed permanently inside a tokamak.

3.2. EROSION STUDIES UNDER NORMAL OPERATION

In FY95, various DiMES samples with or without coatings of V, Mo, Be, and W were been exposed to various plasma discharges. These samples were prepared by SNL and implanted with a Si depth-marker. The net erosion and redeposition of the exposed coating and graphite were determined using either Rutherford back scattering (RBS) or nuclear reaction analysis (NRA), performed by SNL, before and after the exposure experiments. Numerous DIII-D diagnostics were used to quantify the divertor plasma parameters of temperature and density, particle and heat fluxes, sputtered particle distribution, and magnetic field configuration.

For normal operation, a Be and W coated sample was exposed to ELM-free H-mode discharges with 0.7 MW/m^2 incident heat flux, and a peak carbon erosion rate of $\sim 4 \text{ nm/s}$ was measured. This matches quite well with the REDEP and WBC modeling code estimates developed at ANL. Later analysis indicated that the carbon redeposition

fraction is about 80% which is lower than the 90% value used by ITER. Another Be and W coated sample was exposed to higher power ($\sim 2 \text{ MW/m}^2$) ELM-ing H-mode plasma shots. When comparing earlier exposure of a V and Mo coated sample, results show that the erosion rates for the different metal films decreased with atomic number. The observed redeposition profiles of a given material, and in particular the longer transport distances of Be compared to W or Mo, are well explained by previous and ongoing modeling at ANL. The net erosion rate of Be for this sample was found to be about a factor of 5 lower than predicted for a pure Be surface. This will have significant impacts on the resulting lifetime of Be-coated ITER divertor. A total of eight different samples were exposed to DIII-D plasma. These were performed under dedicated and piggyback experiments. In addition to C, Be, and W materials, $\text{B}_4\text{C}/\text{SiC}$ and V-alloy were also exposed. Two were disruption experiments.

Selected results for samples 70, 71 and 74 are presented in Tables 1 and 2. Typically, the positioning control of the separatrix on the sample is accurate to within 5 mm. Other

Table 3-1
Exposure Conditions for Samples 70, 71 and 74 Exposed to H-mode Plasma

OSP Parameter	ELMing	Heat Flux (MW/m^2)	Ion Flux ($\text{s}^{-1}\text{m}^{-2}$)	n_e (m^{-3})	T_e (eV)	Exposed time (s)
DiMES 70 and 71	No	0.7	1.2×10^{24}	4×10^{19}	70	4
DiMES 74	Yes	2.0	2.5×10^{24}	1.0×10^{20}	45	13

Table 3-2
Erosion Results Summary for Samples 70, 71 and 74

Erosion Parameters		Peak Erosion Rate (nm/s)		Toroidal Redeposition e-Folding Length (cm)	
Material	Z	DiMES 70 & 71	DiMES 74	DiMES 70 & 71	DiMES 74
Be	4	0.9	1.4	1.2	>1.5
C	6	3.6	16	—	—
V	23	0.5	—	0.7	—
Mo	42	0.3	—	0.5	—
W	74	0.1	0.45	0.25	0.4

global plasma parameters are: gas = D₂, I_p = 1.35–1.4 MA, B_T = –1.9 T, magnetic pitch angle at outboard strike point (OSP) = 1.5–2.5, Z_{eff} = 1.5, carbon fraction in core = 1%–2%, and oxygen fraction in core = 0.1%–0.2%. The sample surface temperature is in the range of 60° to 300°C.

3.3. DISRUPTION STUDIES

For disruption studies, a DiMES sample with a slotted cap on top of a silicon wafer was designed. Slots at different angles reveal the spatial distribution of neutrals while the implantation depth in Si gives their energy distribution. Results of a radiative disruption indicated that the directional flux of deuterium neutrals is nearly isotopic and that the deuterium strikes the divertor floor at an average energy of 100 eV. Modeling efforts, using A*THERMAL and SPLASH codes, and physics analysis will continue in FY96 to determine the effect of the cold neutral layer in front of the target during a disruption. The cold neutral layer should lead to a reduction of ion flux to the divertor plate due to charge-exchange processes.

3.4. POLOIDAL COUPON EXPERIMENT

Final design approval was granted for the installation of demountable coupons during the February 1995 vent. Demountable coupons were installed around the DIII–D vessel. These coupons will provide information on the distribution of erosion around the torus.

3.5. DIAGNOSTICS

A triple Langmuir probe was designed and installed on the DiMES mechanism to provide high time-resolution electron temperature and density measurements during Vertical Displacement Event (VDE) disruptions. J. Cuthbertson of UCSD collaborated on this project. The probe is equipped with a recessed holder which can contain a Si29 depth-marked coupon, allowing erosion/redeposition measurements simultaneously with high time resolution temperature and density measurements during disruptions and ELMs. The probe was assembled and tested successfully in DIII–D.

3.6. V-ALLOY EXPOSURE

A screw-top DiMES sample containing a Vanadium disk was exposed to DIII-D pump down and bake-out conditions to study its performance under typical tokamak conditions. The exposed sample was sent to ANL for testing of Vanadium alloy's tensile properties after exposure to typical tokamak bakeout temperatures and gases.

3.7. DEUTERIUM IMPLANTATION

Deuterium areal density measurements were made by W. Wampler of SNL on Be/W and Mo/V coated DiMES samples. High uptake of about 2×10^{17} atoms/cm² was recorded for V. Uptake for Be and C were lower by a factor of two, and Mo and W were lowest by about a factor of 10.

3.8. MEETINGS

A successful U.S./Japan workshop on helium-cooled high heat flux components was held and organized at GA. There were 32 participants from Japan, EC and various U.S. institutions. Experimental results on removing 34 MW/m² of surface loading by helium at a pressure of 4 MPa was presented by C. Baxi of GA and D. Youchison of SNL.

3.9. PUBLICATIONS

C. Wong, D.G. Whyte, W.P. West, R. Bastasz, W.R. Wampler, J.N. Brooks, and P.B. Parks, "Material Erosion Measurements and Disruption Studies on DIII-D using DiMES," 1995 APS meeting.

P.B. Parks, R. Bastasz, C.P.C. Wong, D.G. Whyte, W.P. West, W.R. Wampler, "First Measurements of the Ion Energy Distribution at the Divertor Strike Point," 16th IEEE Symposium on Fusion Engineering, Champaign, Illinois, September 30–October 1, 1995.

Section 4

SiC/SiC COMPOSITE MATERIAL DEVELOPMENT

4. SiC/SiC COMPOSITE MATERIAL DEVELOPMENT

4.1. OVERVIEW

For the development of low activation SiC/SiC composite material for fusion power plant applications, the following tasks that can be performed at General Atomics were identified and proposed to DOE/OFE.

- Advanced composite material fabrication, improvement and evaluation.
- Leak tight composite material development and evaluation.
- Development of joining techniques.
- Assess low activation materials development needs for fusion energy application including their use for improved concepts and alternate concepts.

4.2. TASKS FOR 1995 AND BEYOND

The GA SiC/SiC composite program funded by DOE/OFE was initiated. A program plan to fabricate SiC/SiC composite materials with alternate fibers and coatings was established. Samples are to be fabricated with the indicated reinforcement coating as follows:

1. CG Nicalon fabric with 0.2 mm C coating.
2. CG Nicalon fabric with 0.2 mm porous SiC coating.
3. CG Nicalon , unidirectional fiber, with 0.2 mm C coating.
4. Hi Nicalon, unidirectional fiber, with 0.2 mm C coating.
5. Hi Nicalon, unidirectional fiber, with 0.2 mm porous SiC coating.
6. Hi Nicalon, unidirectional fiber, with 0.4 mm porous SiC coating.
7. Super Nicalon, unidirectional fiber, with 0.2 mm C coating.

8. Super Nicalon, unidirectional fiber, with optimized coating from Hi Nicalon study (*e.g.* 0.2 mm porous SiC coating).
9. Super Nicalon, unidirectional fiber, with optimized coating from Hi Nicalon study (*e.g.* 0.4 mm porous SiC coating).

The tooling for the SiC/SiC composite infiltration was designed and fabricated to fit the experimental Chemical Vapor Infiltration (CVI) reactor at GA. Upon receiving High-Nicalon and Super-Nicalon fibers from PNL, samples 1, 4 and 7 were completed. Preparation of remaining samples will continue in FY96.

Section 5

DIAGNOSTICS

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. These diagnostics measure the plasma position, equilibrium shape, loop voltage, plasma current, diamagnetism, and other transient electromagnetic phenomena.

Studies of equilibria reconstructions from magnetic measurements by the JCT and the U.S. home team have shown that the magnetic probes for ITER must be located relatively close to the plasma. This requirement determines that the magnetic probes must be able to survive the neutron flux and fluence near the first wall with acceptable noise characteristics. To address this issue a prototype magnetic probe and complete instrumentation based on the baseline ITER magnetic probe design has been built to be tested in the HFBR reactor at BNL. The test is to be conducted at neutron flux levels comparable to what the ITER magnetic coils are expected to experience.

5.2. COIL DESIGN

A magnetic sensor was developed consisting of mineral insulated (MI) coax formed into a coil around an alumina ceramic spool (Fig. 5-1). The design allows for active cooling of the sensor using an interlaced cooling coil. Surface temperatures for the configuration of Fig. 5-2 range from 315–450°C with the cooling coil, to 600–845°C without the cooling coil.

MI coax with its protective coax construction is chosen rather than bare copper wire to improve survivability and reliability. Both ends of the MI coax can pass through and seal to a vacuum flange. Unlike bare wire, this avoids in-vessel electrical connections and avoids metal-to-ceramic vacuum seals.

Fig. 5-1. Concept design for a high frequency inductive coil magnetic sensor shown with and without cooling coil.

Fig. 5-2. Schematic of ITER first wall showing location of high frequency inductive coil magnetic sensor (not to scale).

A drawback of MI coax is the reduced frequency response due to the metal sheath that forms both the outer jacket of the coax and the metal cooling coil tube. However, as demonstrated on DIII-D, the frequency response of the sensor can be improved by tuning of the coil/coax circuit to a resonance by adjusting the inductance of the coil and capacitance of the coax. The solid curve in Fig. 5-3 is an example of the improvement in the frequency response of a sensor with a tuned resonance over the response of the sensor without a tuned resonance. Another potential drawback of MI coax is the possibility of nuclear-induced emf between the outer sheath and center conductor.

Fig. 5-3. The dashed curve is the normalized field strength at the sensor. The attenuation of the magnetic field strength is due to shielding by the MI cable metal sheath. The solid curve is the frequency response of the sensor including the shielding by the MI cable metal sheath and a damped resonance involving the inductance of the coil and the capacitance of the cable.

A prototype sensor was made and inserted into the HFBR fission reactor at BNL (Fig. 5-4). The sensor included an excitor coil and a pickup coil. The sensor did not include a cooling coil due to the limited space available in the HFBR reactor. An amplifier connected to the excitor coil generated a magnetic field and an integrator connected to the pickup coil measured the output signal over 1000 s. Results showed the design concept for the MI coax sensor and integrator to be sound, although some changes to the integrator electronics are needed.

Fig. 5-4. Simplified schematic for testing in a nuclear environment the response of an inductive coil magnetic sensor made from mineral insulated (MI) cable.

5.3. LONG PULSE INTEGRATOR

The instrumentation package that was developed includes an innovative long pulse integrator (see Fig. 5-5) with drift less than 2 mv-s/1000 s and resolution of 2 μ v-s over 1000 s, as shown in Fig. 5-6. The resolution of the long pulse integrator will detect a continuous drift in the plasma which, over the 1000 s pulse, would result in a 1% error in the total integrated signal expected from a coil in ITER. These achieved specifications on the integrator are required for plasma shape reconstructions and control on ITER for the planned 1000 s pulses.

Fig. 5-5. Long pulse integrator (LPI) architecture.

Fig. 5-6. Example of analysis of data acquired on DIII-D: (a) raw input signal from magnetic probe, (b) integrated baseline signal, (c) integrated input signal, (d) integrated signal minus integrated baseline signal.

5.4. MEETINGS

1. IAEA Technical Committee Meeting on Magnetic Diagnostics, Kharkov, October, 1994 (travel funded by USIIC contract).
2. ITER Diagnostics Expert Group Meeting, Naka, February, 1995 (travel funded by USIIC contract).
3. ITER Diagnostics Workshop, Varenna, August, 1995 (travel funded by USIIC contract).
4. ITER Diagnostics Expert Group Meeting, Varenna, September, 1995 (travel funded by USIIC contract).
5. 16th IEEE/NPSS Symposium on Fusion Engineering, Champaign, September, 1995.

5.5. PUBLICATIONS

“Frequency Response of Metal-clad Inductive Magnetic Field Probes,” General Atomics Report GA-A22049.

“Magnetic Diagnostics for Future Tokamaks,” to be published in the Proceedings of the 16th IEEE/NPSS Symposium on Fusion Engineering.

“A Digital Long Pulse Integrator,” to be published in the Proceedings of the 16th IEEE/NPSS Symposium on Fusion Engineering.

Section 6

RF TECHNOLOGY

6. RF TECHNOLOGY

6.1. U.S./JAPAN BILATERAL EXCHANGE WORKSHOP

The purpose of the U.S./Japan Bilateral Exchange Workshops is to exchange fusion-related rf technology information and to identify possible future collaborative efforts. This exchange benefits both countries through the sharing of experiences, solutions, and unique equipment for joint experiments. This allows both parties to develop rf heating options in the most efficient, cost effective manner.

In December 1995, this exchange was held in the U.S. There were nine attendees from Japan and nine from the U.S. Twenty-three presentations were made covering ICRF, ECH, and LH technology. The contributions included talks on mode conversion, cryo window testing, recent developments on 170 GHz gyrotrons and transmission systems, folded waveguide concept, combine collaboration, and other subjects. The exchange was held at the same time as the International Infrared and Millimeter Wave Conference and the Third International Workshop on Electron Cyclotron Heating (ECH) Transmission Systems. Copies of the presentations from the U.S./Japan Workshop were prepared and distributed.

There will be a U.S./Japan rf technology exchange in Kyoto, Japan, in November 1996. It is expected to be attended by members of the Ministry of Education, Science and Technology Agency, JAERI and selected members from the U.S. rf community.

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. Copies of the presentations were prepared and distributed.

The next ECH Transmission Systems Workshop will be held in 1997, again in conjunction with the U.S./Japan and Infrared and Millimeter Wave Conference.

6.3. ECH WINDOW DESIGN AND FABRICATION

6.3.1. 110 GHz WINDOW

General Atomics fabricated a 4" × 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. The window was tested at 200 kW for up to 0.7 s with a reduced beam diameter. 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.

Making this first full-size 110 GHz window considerably advanced the distributed window fabrication technology. Two vendors learned how to make the closely-spaced, long pilot holes in the niobium frame with the necessary high precision. Vendors also successfully machined the water channels and sapphire slots. The sapphire strips were successfully brazed into the niobium frame except for two locations where there were vacuum leaks because of braze problems. These problems will be remedied on the next window. The window tests at CPI were with vacuum on each side of the window, so the leaks did not affect the testing.

In addition to confirming the power handling ability of the window, the tests at CPI verified the model for calculating the sapphire temperature versus power level. The peak sapphire temperature at maximum microwave power was 150°C. The measured power losses in the window were 7% rather than the calculated 4% because of the braze difficulties mentioned previously.

Based on what was learned in making this first full-size window, a new 110 GHz window is being fabricated for testing and installation on a CPI gyrotron for subsequent use on DIII-D. Computer modeling at LLNL, which was confirmed by analytic modeling at MIT, of the first window showed that the effective thickness of the sapphire strips is larger than the true thickness. The new window uses thinner sapphire to account for this effect and also to account for the temperature-dependent dielectric constant of the sapphire. The new window is being made with improved brazing techniques.

6.3.2. 170 GHz WINDOW

Considerable progress was made during FY95 on 170 GHz distributed window development. The GA-funded 1.25" × 1.25" 170 GHz prototype window project developed a new modular window design for low-loss, wide-band-pass performance. The niobium vanes are made individually and brazed into a niobium frame at the same time as the sapphire strips are brazed into the frame. With the integral frame design, the fabrication tolerances for pilot holes for 170 GHz windows would have been even more demanding than for the 110 GHz windows. The modular approach simplifies the fabrication by relaxing the tolerances on the pilot holes. The 170 GHz vanes match 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.

The prototype window parts will be fabricated early in FY96. The window will be assembled and brazed by mid-FY96. Discussions are underway with JAERI on testing the prototype window. It appears that the best time for testing the window will be in August 1996 after JAERI has made modifications to their 170 GHz gyrotron. Interface hardware for these tests was fabricated in FY95 under the 170 GHz window task.

Design and fabrication activities were also started for the full-size 1 MW CW 170 GHz distributed window. This window will also use the modular design. Two vendors have been identified that have the requisite capabilities for fabricating the vanes for this window. The window is scheduled to be completed by the end of September 1996 and to be hot-tested on a 170 GHz gyrotron by the end of CY96.

6.4. JAERI CRYO WINDOW TESTS

The JAERI cryogenically cooled gyrotron window was tested at GA using the 500 kW 110 GHz Varian gyrotron during December, 1994. Pulses on the order of 100 kW for 400 ms at a 1 Hz rate were passed through the window. The initial analysis indicates that the sapphire in this window, procured from Crystal Systems, is of better quality than the sapphire procured from Kyocera for the window used in Japan. However, neither window is suitable for 1 MW operations. JAERI would like to confirm this

conclusion by retesting the window late in CY96 when GA has a 1 MW 110 GHz gyrotron operational.

6.5. JFT-2M COMBLINE ANTENNA

Negotiations with the JFT-2M team led to an agreement that a nine strap or longer comblin antenna could be mounted in JFT-2M if the installation is performed during the fall 1995 vent, and if the soft x-ray detector is not blocked. Drawings and layouts of the internal hardware for JFT-2M were received and a conceptual design of a nine-strap and a fifteen-strap comblin antenna were developed. These concepts were sent to Japan for review and comment. The review of the conceptual design resulted in a 12 strap antenna that would straddle one port and extend between the midplane and the lower divertor shelf.

The design, illustrated in Fig. 6-1, consists of twelve modules with each module containing a current strap, a double-layer front Faraday shield and a single-layer back Faraday shield. The plasma-facing parts of the front Faraday shield were coated with TiC to reduce sputtering. Two of the modules were modified to incorporate an attachment to a 50 ohm stripline feed. The stripline was routed behind the antenna modules and connected to the vacuum coaxial input feedthroughs. The design was completed in July, 1995 and the parts will be completed by November, 1995. The plan is to install the antenna at the end of the divertor installation vent, now scheduled for January 1996. Testing of the antenna will be performed during April or May 1996.

6.6. HYBRID TUNING SYSTEM

A representation of the hybrid tuner is shown in Fig. 6-2. Most work on the hybrid tuning system was necessarily deferred until delivery of ceramic stand-offs from Spinner. Delivery is expected by August of 1996, in which case testing on ASDEX is planned to begin in September. Upon successful demonstration on ASDEX, the unit will be shipped to GA for incorporation into the DIII-D experiments during the spring of 1997.

Fig. 6-1. Combine antenna module with double Faraday shield and side limiters.

Weight of 2 FFTs including
support structure: 3000kg

Name Tunergruppe UFA-Diagnostik		113		0040100010A1	
Date 01.04.96		Date 01.04.96		Erstellt für SK1330B	
Author H. Martin		Date 01.04.96		Hybrid-Tuner.kompl.	
Drawing Person		Date 01.04.96		0060.100 00	
Name		Name		Ers. d. Zeichn. für	
AFT		AFT		Best. Sheet	
Date		Date		B	

Fig. 6-2. Representation of the hybrid tuner.