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FUSION TECHNOLOGY DEVELOPMENT

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by
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Section 1

FUSION TECHNOLOGY DEVELOPMENT OVERVIEW

1. FUSION TECHNOLOGY DEVELOPMENT OVERVIEW

In FY97, 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, International 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), Plasma Interactive Materials (Section 3), Magnetic Diagnostic Probes (Section 4) and RF Technology (Section 5). 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. BOOTSTRAP CURRENT OVERDRIVE

Bootstrap current overdrive has been determined to be an attractive method for current ramp-up to its full value in a transformerless operation of the ARIES-ST power plant. This current ramp scheme will require high β_p and low collisionality, ν^* . When operating at a given fraction, n_G , of Greenwald's density limit, we find $\beta_p \propto n_G T I_p^{-1}$ and $\nu^* \propto n_G B_T T^{-2}$. We demonstrated that stable high β_p equilibria ($\epsilon \beta_p \geq 3$) exist at low aspect ratios ($A \leq 2.8$). Based on these equilibria, we examined collisionality effects on the bootstrap current using the modified Hinton-Hazeltine formula, appropriate for general geometry, developed by CRPP (Lausanne) and GA. The results of the examination indicate that at $A = 1.4$, with $\beta_p = 3.5$ and $n_G = 2.8$, a target plasma of $B_T = 2.0$ T and $I_p = 0.3$ MA would be required for the ramp-up operation. Our work on this concept for rampup was made available to the ARIES-ST study for inclusion in the IEEE paper by T.K. Mau.

2.2. FINITE p' AT THE PLASMA BOUNDARY

Finite pressure gradients at the edge of the plasma, which imply finite current density as well, are routinely observed in the DIII-D experiments. However, numerical MHD stability studies routinely assume that p' , and the current density, are zero at the edge of the plasma. To remedy this, several sequences of equilibria were generated with $p'_{\text{edge}} = (0, 50\%, 75\%, 85\%)$ of p'_{max} at $A = 1.4$ using the TOQ equilibrium code. Finite p'_{edge} was found to improve stability to ballooning modes but to reduce stability to kink modes. Increasing triangularity improves kink stability so higher β_N can probably be achieved by raising p'_{edge} and increasing δ to ~ 0.7 . Reducing δ to allow a tapered center post does not look promising because of the kink stability.

2.3. LIMIT FOR RANGE OF ASPECT RATIO

We have previously considered the dependence of β and β_N upon aspect ratio. An attempt to study other values of A in as much detail as we have devoted to $A = 1.4$ would require extensive searches to determine an optimal $p'(\psi)$ at each A . We have taken the far more modest course of examining only ballooning stability and using only the p' profile found optimal at $A=1.4$ (strongly peaked off-axis but going to 0 at edge) for A

ranging from 1.2 to 2.8. We looked at a range of triangularities from 0.2–0.6 and at elongations of 2, 2.5 and 3. The results show both β and β_N improving with increasing κ and decreasing A. Because we did not modify the profiles as we varied A, the bootstrap fraction falls off somewhat for some of the higher A cases, but is still always in excess of 80%. The triangularity yielding the highest β does not vary much with A. Even at A = 2.8 it is approximately 0.44 for $\kappa = 2$ and approximately 0.52 at $\kappa = 3$. The data for β_N versus A at $\kappa = 3$ can be roughly fit to $\beta_N \propto 1/A^{1/2}$.

2.4. PLASMA START-UP WITH OUTER PF-COILS

To implement the bootstrap current overdrive scheme, an initial target plasma with a low level of plasma current must first be formed. We have explored the possibility of using the flux available from the divertor and outer poloidal field coils for plasma initiation. The time-dependent MHD code, DINA, was used to simulate the start-up scenario in a DIII-D geometry. The modeling indicates formation of the target plasma with a current of about 200 kA may be feasible. Results from these studies were made available to the ARIES-ST study for the IEEE paper by T.K. Mau.

2.5. MEETINGS

1. ARIES-Low Aspect Ratio Power Plant Design Project meeting, December 1996, PPPL, attended by Dr. Ron Stambaugh.
2. ARIES-Low Aspect Ratio Power Plant Design Project meeting, March 1997, UCSD, attended by Dr. Ron Stambaugh and Dr. Clement Wong.

2.6. PUBLICATIONS

T.K. Mau, *et al.*, "Plasma Physics Basis and Operations of the ARIES-ST Tokamak Power Plant," IEEE/NPSS Symposium on Fusion Engineering, San Diego, California, October 1997.

Section 3

PLASMA INTERACTIVE MATERIALS

3. PLASMA INTERACTIVE MATERIALS

3.1. CARBON EROSION AND DEUTERIUM UPTAKE

Carbon samples have been exposed to the steady-state outer strike point under ELMing and ELM-free H-mode discharges. Maximum net erosion rate as a function of incident heat flux was determined. Over the power range studied, the net erosion at the outer strike point is seen to scale linearly with the incident power flux. The measured net erosion rate for C at a heat flux of 2 MW/m^2 is substantial ($16 \text{ nm/s} = 50 \text{ cm/burn-y.}$) during ELMing H-mode. This measured rate of erosion is about an order of magnitude higher than models used for the ITER divertor design. This difference is under investigation, but is believed to be caused by differences in redeposition efficiency, self sputtering and the effect of oblique angles of incidence on sputtering yields.

Measurement of carbon net erosion and deuterium retention of long-term exposure tiles in DIII-D indicates a more serious problem in regions of net redeposition. The net change in the surface profile of the divertor floor from measurements before and after nine months of plasma operation is shown in Fig. 3-1. Accumulation of deuterium along the same area was also mapped by nuclear reaction analysis (NRA) as shown in Fig. 3-1. NRA measurements showed peak deuterium areal density of about $8 \times 10^{18} \text{ D/cm}^2$ in a co-deposited layer about $6 \text{ }\mu\text{m}$ deep, mainly at the usually detached inboard divertor leg, indicating a deuterium inventory in the divertor of about one gram. That layer of carbon near the inner divertor strike point has an atomic saturation concentration of $\text{D/C} \approx 0.25$, which is not significantly lower than the laboratory-measured saturated retention of 0.4, despite the He glow discharge cleaning between every discharge and regular 350°C bakes of the DIII-D vessel. DiMES short-term exposure results are overlaid in Fig. 3-1 showing the location of the DiMES sample along the major radius of DIII-D lower divertor. Both carbon erosion rate and deuterium areal density match closely the long term exposure results at the outer strikepoint which is an area of net erosion and low re-deposition. Based on these results modeled for ITER dimensions and DiMES run conditions, at an estimated accumulation rate for tritium of the order of 0.01 g/s , it is likely that before a few hundred DT plasma (1000s) discharges the 1-kg in-vessel tritium inventory limit for ITER will be reached due to the co-deposition of tritium with carbon. Therefore, for the ITER design, knowledge and understanding of the distribution and rates of carbon redeposition become very important. An intense search for effective methods of *in-situ* tritium or carbon coating removal has been initiated by the ITER program.

Fig. 3-1. Erosion and deuterium retention results from long-term exposure of DIII-D tiles and from DiMES experiments.

3.2. DISRUPTION MELT LAYER STUDIES

We determined that the best type of disruptions for studying melt layer movements are double-null divertor negative central shear (NCS) high power disruptions. IR thermography of previous disruptions shows a very broad footprint for the heat load to the lower divertor floor and shows temperature increases well in excess of the 650°C melting point of aluminum. We planned to use a high power NCS beta disruption to

provide a high energy, broad footprint of heat flux to the lower divertor floor to melt a 10 μm coating of aluminum placed on the DiMES sample. We would then determine the movements of this melted layer. However, the experiment was unsuccessful because the desired NCS disruption could not be obtained in the seven discharges in which it was attempted. The cause of this MHD recovery from disruption is under investigation. Disruption melt layer experiments will continue.

3.3. CARBON DUST STUDIES

A dust collection experiment was carried out during the lower single null portion of two plasma discharges. The DiMES sample was designed so that parallel heat flux was incident onto a small graphite face (about 1 mm $\times$ 10 mm region). The face was arranged so that carbon flowing downstream from the face would redeposit onto the DiMES sample surface, and carbon vapor emission from the face would deposit onto a stainless steel collector disk positioned beneath the face, out of view of the incident plasma flowing parallel to the field lines. The estimated heat flux on the small graphite face was 50 MW/m². The sample was removed following the run, and visual inspection clearly indicated a significant carbon deposition onto the stainless collection disk, as well as a downstream re-deposited film.

Arrangements were made for J. Carmack of INEEL to collect DIII-D dust during the FY98 removal of 20 upper and lower DIII-D divertor tiles.

3.4. PUBLICATIONS

C.P.C. Wong, D.G. Whyte, R.J. Bastasz, J.N. Brooks, W.P. West, and W.R. Wampler, "Divertor Materials Evaluation System (DiMES)," will be presented at, and published in, the Proceedings of, the 8th International Conference on Fusion Reactor Materials, Sendai, Japan, October, 1997.

D.G. Whyte, J.N. Brooks, C.P.C. Wong, R.J. Bastasz, W.P. West, and W.R. Wampler, "DiMES Divertor Erosion Experiments on DIII-D," J. Nucl. Mater. **241-243** (1997) 660.

Section 4

MAGNETIC DIAGNOSTIC PROBES

4. MAGNETIC DIAGNOSTIC PROBES

4.1. FAST WAVE REFLECTOMETER DEMONSTRATION ON DIII-D

The electronics for the DIII-D fast wave launch antenna were installed. Mixers and data acquisition electronics for nine receiver loops were installed and tested. Initial data was collected during the July DIII-D campaign. While the inteferometer data from the new system looked good, no reflectometer signal was seen. Poor coupling between the plasma and the receiving loops is thought responsible for lack of signal. Replacing the receiving loops with a design similar to the launch antenna is expected to result in enough improvement in the coupling to give a good reflectometer signal.

4.2. RADIATION TESTING OF AN EQUILIBRIUM TEST COIL

An equilibrium test coil fabricated in FY96 (Fig. 4-1) was irradiated in HFBR. This type of coil will be used in ITER primarily for reconstructing the magnetic equilibrium. The coil was instrumented with a long pulse integrator in order to determine if radiation effects would cause problems with a 1000 s integration. An unacceptably high potential of 46 mV across the coil leads was observed with the coil in the radiation field. The requirement for ITER is < 6 mV, although a safety factor of ten below the 6 mV would be prudent.

Fig. 4-1. Coil/connector assembly inserted into HFBR.

A radiation-induced EMF (RIEMF) as high as 3.5 V was also observed between the center conductor and the outer shield of the mineral-insulated cable that made up the coil. The radiation induced potential across the coil center conductors is consistent with the following model:

1. the RIEMF between the center conductor and the shield drives a current through each of the leads of the coil to ground;
2. an imbalance in the electrical resistance in the two sides of coil leads (perhaps including the coil itself) to ground results in a potential difference across the integrator.

The integrator front end circuit has to be designed to handle this large potential between the shield and the center conductor. The resistance of the coil itself has to be considered since the RIEMF may not be uniform across the coil and the several ohms of resistance in the coil itself can be enough to cause an unacceptable integrator drift.

Coil types and integrator modifications have been defined for future radiation tests. Unfortunately, the HFBR reactor has been shut down and will not be available for the foreseeable future. Because test pieces and thermal calculations for the tests are specific to the particular reactor, test coil design work was suspended pending identification of a neutron source.

After exploring several options for another neutron source, we concluded that the JMTR reactor in Japan is the most suitable. The neutron levels in JMTR (8×10^{13} n/cm²/s) are appropriate and access is adequate although more limited than in HFBR. Test coils for JMTR will require a new mechanical design, new thermal calculations, new parts and U.S. and well as Japanese approvals. Dr. Tatsuo Shikama of JAERI has expressed interest in collaborating with us to facilitate testing on the JMTR reactor.

4.3. PUBLICATIONS

H. Ikezi, J.S. deGrassie, R.I. Pinsky, and R.T. Snider, "Plasma Mass Density, Species Mix and Fluctuation Diagnostics Using Fast Alfvén Wave," *Rev. Sci. Instrum.* **68**(1), (1997) 478.

E.J. Strait, J.D. Broesch, R.T. Snider, and M.L. Walker, "A Hybrid Digital-Analog Long Pulse Integrator," *Rev. Sci. Instrum.* **68** (1997) 381.

Section 5

RF TECHNOLOGY

5. RF TECHNOLOGY

Preparations were made for FY97 rf technology exchange meetings between the U.S. and Japan and between the U.S. and Korea. Support was provided to DOE OFES representative T.V. George in selecting and organizing U.S. team members. Technical agendas were established in coordination with Japanese and Korean hosts.

The Japanese meeting, held in October, was the continuing U.S./Japan Radio Frequency Technology exchange. The meeting in Korea will be the first U.S./Korean rf technology and will include discussions exploring future mutually beneficial collaborations.