GEORGIA INSTITUTE OF TECHNOLOGY   OFFICE OF CONTRACT ADMINISTRATION

PROJECT ADMINISTRATION DATA SHEET

[x] ORIGINAL   [ ] REVISION NO. ______

Project No. E-26-611 (Follow-on to E-26-686)   ~~GTRI~~/GIT   **DATE** 12/10/82

Project Director: Dr. R. G. Bateman & Dr. W. M. Stacey   **School/~~Lab~~** Nuclear Engineering

Sponsor: Dept. of Energy, Oak Ridge Operations, TN

Type Agreement: Contract DE-AS05-78ET52025, Mod. A005

Award Period: From 12/1/82 To 11/30/83 (Performance) --- (Reports)

Sponsor Amount: Total Estimated: $ 100,000   Funded: $ 100,000

Cost Sharing Amount: $ None   **Cost Sharing No:** N/A

Title: Fusion Studies Program

ADMINISTRATIVE DATA

OCA Contact William F. Brown   Ext. 4820

1) Sponsor Technical Contact:

Mr. Albert L. Openaker
Reactor Sys. & Applications Branch
Div. of Dev. & Technology
Office of Fusion Energy
Dept. of Energy
Washington, DC 20545

2) Sponsor Admin/Contractual Matters:

Ms. Joyce Carringer
Procurement & Contracts Division
Dept. of Energy
Oak Ridge Operations
P. O. Box E
Oak Ridge, TN 37830
(615) 576-7564

Defense Priority Rating: None

Military Security Classification: None
(or) Company/Industrial Proprietary: ______

RESTRICTIONS

See Attached Gov't Supplemental Information Sheet for Additional Requirements.

Travel: Foreign travel must have prior approval – Contact OCA in each case. Domestic travel requires sponsor approval where total will exceed greater of $500 or 125% of approved proposal budget category.

Equipment: Title vests with GIT, if acquired by us and listed Appendix "A".

COMMENTS:

Mod. A005 adds $100,000 as follow-on to E-26-686. New project number is because of separate financial reporting requirement.

Total contract value (including previous project numbers) is $565,000

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GEORGIA INSTITUTE OF TECHNOLOGY OFFICE OF CONTRACT ADMINISTRATION

SPONSORED PROJECT TERMINATION/CLOSEOUT SHEET

Date 9/13/85

Project No. E-26-611 School/~~Lab~~ Nuclear Eng.

Includes Subproject No.(s)

Project Director(s) Dr. R. G. Bateman & Dr. W. M. Stacey ~~GTRC~~ / GIT

Sponsor Department of Energy, Oak Ridge Operations, TN

Title Fusion Studies Program

Effective Completion Date: 11/30/83 (Performance) 11/30/83 (Reports)

Grant/Contract Closeout Actions Remaining:

- [X] None
- [ ] Final Invoice or Final Fiscal Report
- [ ] Closing Documents
- [ ] Final Report of Inventions
- [ ] Govt. Property Inventory & Related Certificate
- [ ] Classified Material Certificate
- [ ] Other

Continues Project No. E-26-686 Continued by Project No. E-26-639

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FORM OCA 69.285

GEORGIA TECH FUSION STUDIES PROGRAM

Contract DE-AS05-78ET-52025
Quarterly Progress Report
For the Period
October 1, 1982 - April 30, 1983

1. RIPPLE REDUCTION FOR A COMMERCIAL TOKAMAK REACTOR

We have used superconducting Ripple Reduction Poloidal Field (RRPF) coils [1-4] together with blocks of ferro-magnetic iron shielding to design a commercial tokamak reactor similar in size to STARFIRE [5] with only eight, rather than twelve, toroidal field coils. The RRPF coils consist of overlapping dipole coils placed just above and below the neutron shielding, where they share the TF coil cryostat and support structure. They produce most of the poloidal field needed to hold a plasma equilibrium with 6% average beta, K = 1.7 elongation, and a pair of separatricies suitable for a poloidal divertor. The RRPF coils also reduce magnetic ripple near the top and bottom of the plasma. They have been carefully designed to carry a maximum field of little more than 8 tesla, so that they can be pulsed as needed during plasma start-up.

Blocks of iron magnetic shielding [6] under each TF coil are used to reduce the magnetic ripple at the midplane from 5.5% to less than 1% throughout the plasma [1]. Since there is clearance for at least 0.9m of blanket and shielding between the laminated iron blocks and the first wall there should be little neutron damage or heating of the iron. Although the crescent shaped iron blocks are rather large (averaging 3.5 m wide, 2.m deep, 6.8 m high, 370 metric tons each), they are simple to construct and should cost much less than the TF coils themselves. The GFUN3D computer code [7,8] was used to determine the effect of the iron on the magnetic field. This code was carefully checked against results of the EFFI code [9] without iron and against results cited by Turner et al [6] with iron. We found that the iron does not noticeably affect the poloidal field from the plasma current and applied coils. The axisymmetric poloidal field, however, is large enough to reduce the ripple directly (by 0.88%) by adding to the strength of the field. This increased field strength also slightly diminishes the ability of the iron to reduce ripple (by 0.35%). The net result of these offsetting effects is that somewhat less iron is needed for ripple reduction when the poloidal field is included in the computation. Further work is in progress to minimize the iron needed.

In the commercial reactor design we developed [1], each of the eight sectors of neutron shielding and blanket can be pulled straight out, while in the STARFIRE design [5] 12 of the 24 sectors had to be shifted over before being pulled out. Hence, in our design there are fewer cuts to be made and there is much more access. We were not able to further reduce the number of TF coils to 6, which would have improved access even more.

Work is in progress to refine the calculation of the separatrix for a poloidal divertor. There appears to be enough room for the divertor chambers and ample access for the vacuum ducts. The close proximity of the RRPF coils seems to hold the X point of the divertor separatrix fixed in space better than conventional PF coils placed outside the TF coil set.

2. RIPPLE REDUCTION FOR A FED/INTOR TOKAMAK REACTOR

We have completed the conceptual design of a FED/INTOR-sized tokamak reactor consisting of 8 toroidal field (TF) coils with 7.0m by 9.4m active bore [2]. Superconducting RRPF coils placed between the TF coils and the neutron shielding produce part of the shaping field needed for a D-shaped plasma equilibrium and they work to reduce magnetic ripple near the top and bottom of the plasma. The RRPF coils are designed to share the TF coil cryostat and support structure, without obstructing access to the shielding, blanket and first wall. Normal (e.g. water cooled copper) saddle coils are imbedded in the shielding one meter away from the first wall in order to reduce magnetic ripple near the midplane from more than 4% to less than 0.3% throughout the plasma. They carry less than 0.6 MA turns of current and consume less than 16 MW of power. The RRPF coils together with a modest amount of current in two pairs of poloidal field coils outside the TF coil set hold a D-shaped equilibrium with more than 5% volume averaged beta, elongations exceeding 1.6 bounded by a separatrix suitable for a poloidal divertor. A field line following code has been used to show that field lines are well confined. Each of the eight sectors of neutron shielding and blanket can be pulled straight out after the saddle coils which straddle the cuts have been removed.

3. NEUTRAL BEAM DRIVEN IMPURITY FLOW REVERSAL IN TFTR AND FED/INTOR

The NBI driven impurity flow reversal experiment in PLT [10] has been used to benchmark [11] a predictive model for impurity transport based upon Pfirsch-Schluter theory and the Stacey-Sigmar [12] theory for effects of beam momentum input and drags. These effects include modification of flows in the flux surface and of the radial potential gradient, as well as direct momentum input. The "benchmarked" model has now been applied to predict NBI driven impurity flow reversal effects in TFTR and in larger tokamak plasmas of the type suggested by the FED and INTOR conceptual designs.

When only the co-injected beams are used in TFTR, limiter-sputtered impurities are excluded from the inner half of the plasma, and the inward impurity flow in the outer half of the plasma is significantly reduced. The same model predicts a significant concentration of impurities in the center of the plasma when the co-injected and counter-injected beams are operated to produce no net toroidal momentum input. Operation of only the counter-injected beams results in the prediction of a rapid accumulation of limiter-sputtered impurities in the center of the plasma.

Application of the model to a FED/INTOR type plasma leads to the prediction that $\sim$20-30 MW of 150-200 KeV NBI may suffice to significantly inhibit, although not prevent, limiter-sputtered impurities from entering the center of the plasma. This suggests that NBI (or rf) flow reversal might be used to achieve a "cold-radiating edge" and thereby limit sputtering erosion.

4. TECHNOLOGY DEVELOPMENT NEEDS REVIEW

The technology development needs for magnetic fusion have been identified from an assessment of the conceptual design studies which have been performed, in collaboration with investigators at other institutions. A summary of worldwide conceptual design effort was prepared. The relative maturity of the various confinement concepts and the intensity and continuity of the design efforts were taken into account in identifying technology development needs. This work was published [13] and distributed.

A large body of conceptual design work exists from which the technology development needs can be specified for the reactor embodiments of the various magnetic fusion confinement concepts. These conceptual designs have evolved as the understanding of the physics of the respective confinement concept has evolved and as the technological constraints have become better defined.

There is a remarkable commonality among the technology development needs specified from the conceptual design studies of the "conventional" reactor embodiments of the different confinement concepts. The tritium processing and storage requirements are essentially identical; the nuclear, materials, vacuum and fueling requirements are quite similar; the parameters required of the different modes of plasma heating are quite similar, although the requirement for specific heating modes varies among confinement concepts; the general character of the superconducting magnet technology requirements are comparable; and the general electrical and remote maintenace systems requirements are similar. The technology development needs for the compact, high-density embodiment overlap in many areas with the needs for the conventional embodiment for a given confinement concept. However, the first-wall and blanket technology development needs tend to be more demanding for the compact embodiments of a refinement concept, and the magnetic development needs are different (i.e. normal conductors).

This commonality implies that a common development program could address the technology development needs that have been defined for the two most developed concepts - the conventional tokamak and tandem-mirror. Moreover, this commonality further suggests that such a development program would have a rather good chance of encompassing within its envelope of objectives any future evolution of technology needs specified for the conventional reactor embodiments of either the mainline (tokamak and tandem mirror) or alternative confinement concepts. Many of the development needs of the compact systems would also be included within such a program.

REFERENCES

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12. W. M. STACEY, Jr. and D. J. SIGMAR, Nucl. Fusion, 19, 1665 (1979).

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Pages missing from report.

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Annual Tech Progress Report

two theories with the PLT results for the impurity transport flux at a single radial location indicated that the "rotation" effect was the right order of magnitude, but that the "momentum input and drag" effect was too small by more than an order of magnitude. A more detailed comparison[13] found that the "momentum input and drag" effect could account for the coinjected results to within a factor of 2 and for the counter-injected results to within a factor 4, if the temperature screening effect was suppressed. This latter comparison also confirmed that the "rotational" effect was the right order of magnitude at the radial location considered in the initial comparison, but found that the radial dependence of the "rotation" effect was strongly at odds with the experimental results, even reversing sign at some locations. These results would seem to indicate that both the "momentum input and drag" effect and the "rotation" effect may be important in accounting for the effect of intense momentum (neutral beam or rf) injection upon impurity transport in tokamaks, in which case a consistent formulation that included both effects is necessary.

Work is presently in progress[14] to incorporate the "direct," momentum input and drag", and "rotation" effects into a self-consistent formalism.

B. Previous Work Under Contract

1. Preliminary estimates were made (GTFR-25) of the possible effectiveness of neutral beam (NB) driven impurity flow reversal in tokamak plasmas of the FED/INTOR/ETR size. Based on an approximate analysis, it was estimated that 25-50 MW of co-injected NB power would prevent edge-generated impurities from entering the plasma.

2. The original formalism[3] for the "momentum input and drag" effect on impurity transport was extended to include temperature gradient effects (GTFR-21), reduced to a computationally tractable form and incorporated into an 1D transport code.

3. A careful analysis of the PLT flow-reversal experiment[10] was performed[13] in order to "benchmark" a calculational model which could then be used to predict the effect of NB impurity

flow reversal in future tokamak plasmas. The model included Pfirsch-Schluter transport and the "direct" and "momentum input plus drag" transport effects due to NB injection. When the impurity transport fluxes predicted by this model are multiplied factor of two, they are in reasonable agreement with the impurity transport fluxes inferred from the experimental data for both the case of no injection (inward impurity flux) and co-injected beams (outward impurity flux). This model (with the factor of 2 multiplier) was adopted for subsequent analyses.

4. The benchmarked calculational model was applied to a plasma with FED/INTOR/ETR parameters[13]. Coinjection of as little as 10MW of NB power (at 35^{o} to the normal) was predicted to significantly inhibit the penetration of edge-sputtered iron impurities to the center of the plasma, resulting in an accumulation of impurities in the outer region of the plasma. This suggests the possibility of using NB impurity flow reversal in ETR to produce a "cold, radiating edge", which would result in low sputtering rates and make a pumped-limiter feasible.

5. The benchmarked calculational model was applied to the TFTR plasma. The predicted effect of NB injection of inhibiting (co-injection) or enhancing (counter-injection) the penetration of an edge-sputtered impurity to the center of the plasma was dramatic. With the maximum coinjected power of 16MW, 30-50% less tungsten or iron impurities are predicted to penetrate to the central half of the plasma, which increases the radiated power from the outer half of the plasma by a similar amount. This result implies that NB flow reversal will be an effective impurity control mechanism for TFTR which can be implemented by preferentially coinjecting the beams.

6. Calculations are in progress to estimate the effect of NB impurity flow reversal in a "commercial" tokamak with parameters similar to those of STARFIRE.