A Proposal to the United States Energy
Research and Development Administration
For Continuation of Fusion Reactor Technology Studies

Progress Report
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I. Introduction

Significant progress was made in the last year within both conceptual design and experimental research areas. Major achievements were:

- . Completion of the "Balance of Plant" design for the UWMAK-I reactor, including economic and environmental assessments.
- . Completion of the design of a second UWMAK reactor and publication of a detailed report (UWMAK-II, UWEDM-112).
- . Proposal and subsequent investigation of "carbon curtain" approach to protect the plasma from impurity contamination and the wall from plasma ion bombardment.
- . Proposal and continued investigation of the ISSEC concept to protect the CTR first walls from radiation damage.
- . A complete delineation of the effect of injected power on the optimum Q of driven tokamaks and the discovery that driven, unignited plasmas can withstand impurity levels of 2-4 times those in ignited systems.
- . Initiation of plasma-wall interaction experiments aimed at measuring effects of sputtered atoms on the plasma and plasma ions on carbon cloths.
- . Initiation of 14 MeV neutron cross section measurements to determine long level radioisotope production and the production rates of H and He atoms.
- . The design of an experiment to determine the integrity of the UW magnet conductor proposed in the UWMAK-I study.
- . Proposal of a driven Tokamak as an engineering test reactor. Initial work shows this device to be modest in size ($I = 2.4$ MA) worthy of further detailed study.
- . Publication of 17 open literature papers and internal reports on various aspects of fusion reactor design.
- . Presentation of papers on Fusion Technology at 5 International Meetings.
- . Presentation of 17 papers at National Meetings.
- . Participation in 16 ERDA sponsored meetings in Washington, D.C. or elsewhere.

II. Progress

A. Completion of UWMAK-I "Balance of Plant" Design

The details of the power cycle, plant layout, environmental impact, resource requirements, and economics of the UWMAK-I reactor have been published (UWFDM-68, Vol. II). This report has been sent to nearly 300 requestors of the original UWMAK-I document and only a few highlights of the study are quoted below.

UWMAK-I is a lithium cooled stainless steel structure which produces 5000 MW_t for a 90 minute burn time and requires approximately 6 1/2 minutes to recharge the transformer coils. The fusion energy is deposited in a primary lithium coolant ($T_{\text{max}} = 489^{\circ}\text{C}$) and then transferred to the steam generator (pressure = 2000 psia, $T = 404^{\circ}\text{C}$) via a secondary sodium loop ($T_{\text{max}} = 456^{\circ}\text{C}$). The design of intermediate Li-Na heat exchangers and Croloy-2 1/4 steam generators has been completed along with a complete turbine generator cycle design. The gross electrical output of the plant is maintained at a constant level of 1681 MWe by diverting ~7% of the secondary sodium into storage tanks during the 90 minute burn. This sodium is then used during the recharge cycle to generate steam while the lithium flow in the primary circuit removes the decay heat from the blanket. It has been determined that ~208 MWe of the plant output is required to maintain the plant components which means that 1473 MWe are available for sale. This amounts to an overall efficiency of 29%, or stated another way, 31% during the actual burn cycle.

The design of a building which provides a vacuum of 1 torr around the reactor has been completed along with the plant layout for lithium loops, sodium loops, steam generating equipment, tritium extraction facilities, turbine-generator units, cooling towers, etc. Other special building considerations

such as superconducting stored energy facility, harmonic filter building, switch yards, gas storage and cooling towers have also been given preliminary design consideration.

The potential problems of siting a reactor like UWMAK-I have been considered. Such problems as plant size, stray magnetic fields, leakage of radioactive isotopes and the potential for large "conventional" accidents have been given preliminary consideration and the conclusion is drawn that the release of tritium is still the most serious problem to be faced. Various scenarios show that the equilibrium release rates of tritium to the environment should be within the current regulatory limits but that such limits may be exceeded following a release of 1% of the total tritium inventories at ground level under the worst climatic conditions. No realistic accident which would actually result in such a release has been found. The use of 100 m high stacks to disperse tritium from the containment building can considerably reduce the magnitude of this problem.

It was shown that the materials resource demands of a UWMAK-I type reactor may be quite severe, especially for Cr. Approximately 132 metric tonnes of metal are required per electrical megawatt. Roughly 62% of this demand is for Fe, 10% for Pb, 8% for Cr, 5% for Ni, 5% for Cu, 5% for Na, and 1% each for Mn, Li and B.

The cost of the UWMAK-I power plant was estimated following standard procedures for economic evaluation of nuclear power plants. It was found that the capital cost of the present plant would be ~\$1000 per MWe and the operating costs were ~22 mills per kW-hr.

It was concluded that since the UWMAK-I reactor represents a first attempt at a detailed and self consistent reactor, the operating conditions, power output

and economic costs were well within reason. Several areas of further work were clearly outlined including such items as a superconducting stored energy facility design, methods for reducing tritium leakage, need for evacuated buildings, shielding requirements for personnel, need for reduced reliance on Cr containing alloys, and research into finding blanket materials which will have longer reactor lifetimes.

B. UWMAK-II Reactor Design

The major design features of the D-T Tokamak Power Reactor, UWMAK-II were presented at the Fifth IAEA Conference on Plasma Physics and Controlled Nuclear Fusion Research held in Tokyo, Japan, November 11-15, 1974. The details of reactor design were given in UWFD-112 published in May 1975. A brief summary of UWMAK-II and major conclusions drawn are given below.

UWMAK-II is a 5000 MW(th) (1710 MW(e)) helium cooled tokamak fusion reactor. The fuel is deuterium and tritium and the structural material is 316 stainless steel. The design has a low aspect ratio (2.6) and low toroidal field on axis (35.7) kG with 24 extended "D" shaped superconducting TF coils. A superconducting air core transformer is designed with windings outside TF coils while the vertical field magnets, also superconducting, are inside the extended "D's". The plasma operating conditions have been determined using pessimistic scaling laws resulting from the trapped ion instability. Such scaling can be beneficial in the reactor regime. A double null poloidal field divertor is included for impurity control and a two-dimensional woven carbon cloth mounted on the vacuum chamber wall is included to protect the plasma from high Z impurities and the first structural wall from erosion by charged particle bombardment. The helium cooled blanket contains solid Be in canned rods for neutron multiplication and solid lithium aluminate, also in canned rods,

for tritium breeding. The breeding ratio is 1.06, the energy per fusion is 21.56 MeV, and the tritium inventory in the blanket is low, 42 g. Radiation damage problems are similar and as serious as those found in our previous study. Periodic replacement of the blanket modules every two years will probably be required because of radiation damage. A detailed mechanism for blanket removal that does not require removal of the shield or the VF and TF coils has been given.

Some of the major conclusions from the UWMAK-II study are:

- . Pessimistic plasma transport, such as trapped particle modes, is nevertheless adequate for reactor systems and the short particle confinement time (~ 8 s) allows for removal of alphas. The startup of the plasma requires a minimum average density of $\sim 4 \times 10^{13}/\text{cm}^3$ in order to reach ignition regardless of injected power. This is related to the sharp increase in diffusion as density decreases for trapped particle modes.
- . The scrape off plasma is very unstable to loss cone instabilities. The expected growth rates and turbulent scattering will destroy mirror confinement and lead to a rapid rate of transport to the particle collectors.
- . Even with Bohm cross-field diffusion, the divertor efficiency is $\geq 99\%$.
- . The divertor is very poor at ionizing incoming neutrals in the scrape off zone. The "screening" efficiency is $\leq 10\%$.
- . The remaining problem, not yet sufficiently analyzed, which may cause the divertor to have a poor efficiency is charge exchange in the scrape off zone. This requires a better treatment of the boundary.
- . The placement of the divertor coils inside the TF coils results in a large reduction in stored energy requirements and allows a faster current rise time.

- . Trapped ion mode scaling increases the energy loss to the divertor and results in a very difficult heat removal problem.
- . High pressure helium cooling results in a high structural fraction in the blanket. This reduces the breeding ratio and increases the amount of material required. (5% in UWMAK-I vs 20% of the blanket volume in UWMAK-II.)
- . The use of solid breeding material with a low thermal conductivity requires a very small diameter, fuel rod type geometry for containment.
- . In order to maintain a low tritium inventory, the diffusion paths must be short which requires that the solid be composed of small (10's of micron) diameter particles. It is feared that these particles will sinter in the high temperature and neutron flux environment.
- . The use of a solid breeder material in a helium cooled system requires the use of large amounts of Be.
- . Because of the low reserves and high cost of Be, it will be necessary to recycle the Be which is removed every 2 years because of radiation damage to the first wall.
- . There are very few problems associated with the transport of radioactive corrosion products.
- . Monte Carlo calculations show that the breeding ratio in solid lithium compound blankets will be considerably less than calculated by 1-D homogeneous calculations.
- . The radiation damage problem in 316 SS will be as severe in UWMAK-II as in UWMAK-I. The walls will have to be changed at least every 2 years because of loss of ductility.
- . The carbon curtain can protect the plasma from impurity contamination and from first wall erosion by plasma ion sputtering and blistering.
- . The lifetime of the carbon curtain will probably be limited to ~2 years due to swelling problems.

- . A severe swelling problem exists in metallic Be because of the high, ~3000 appm of helium gas generated per year.
- . Swelling will also be a problem in the LiAlO_2 because of the ~22,000 appm per year of He generated during the breeding of tritium.
- . The radioactivity and afterheat levels of II are within a factor of 2 of UWMAK-I.
- . The tritium inventory in the blanket of UWMAK-II is ~40 grams, but when all of the tritium in the system is tabulated, solid breeding blankets only reduce the total tritium inventory by a factor of ~2.
- . The tritium leakage to the environment is ~1 curie per day.
- . A secondary sodium loop is required because of load leveling requirements and to reduce tritium leakage.
- . The idea of multiple redundancy and stationary TF magnets is much more compatible with a conservative reactor design.
- . A plausible method for first wall removal has been developed. It only involves moving ~300 tonne units as opposed to ~3500 tonne units in UWMAK-I.
- . The secondary vacuum wall around the TF coils reduced the need to provide an evacuated reactor building.

In conclusion, our general feeling is that we have made definite progress in many areas of reactor technology through the UWMAK-II design. However, if these improvements were applied to UWMAK-I, we think the latter may be a better reactor system. We do not see a great deal of advantage in a solid breeding blanket configuration, and such a scheme may be more difficult from an engineering standpoint, produce lower breeding ratios, and certainly is more expensive. Further analysis of UWMAK-II will be continued in FY-76.

C. The Carbon Curtain Concept

It is clear that low Z impurities (e.g. C, O, Si, etc.) have a much less serious impact on plasma performance and larger concentrations of such elements can be tolerated. We describe here a method by which the benefits of low Z fusion reactor first walls can be achieved while continuing to take advantage of the high strength and ductility of a metallic vacuum wall. This metallic vacuum wall will also be protected from charged particle and neutral atom sputtering thus lowering first wall erosion rates.

The proposed solution is to place a flexible, 2-dimensionally woven carbon cloth between the plasma and the metallic vacuum wall. This cloth will act as a curtain and intercept practically all ions and neutral atoms leaking from the plasma. It will similarly collect all metallic ions emanating from the vacuum wall and prevent their introduction into the plasma.

The carbon cloth suggested for protecting the plasma consists of continuous carbon filaments, normally 5-10 microns in diameter, collected into bundles of 700-1000. These bundles may be twisted together to form threads and the threads twisted together to form yarns of any diameter. The yarns are then woven, by conventional techniques, into cloth which is flexible and porous. Such cloth typically has a surface area of 1m^2 per gram of material.

It is proposed that such a cloth be hung between the plasma and first wall to collect the ions and neutral atoms leaking from the plasma. The curtain will also collect any metallic atoms emanating from the vacuum walls. The curtain will be cooled by radiation only and has no structural role in a reactor.

The first question that occurs when contemplating the use of a carbon curtain is its compatibility with the vacuum requirements for fusion reactors. This problem can be considered from two positions; first, what is the vapor pressure of the graphite at operating temperature and second, how much

dissolved and adsorbed gas will be evolved at the operating temperature of the curtain.

The vapor pressure of carbon has been measured and reported in the literature. If we assume that fusion reactor plasmas require vacuums of 10^{-5} torr or better, then we find that we are limited to operating temperatures of $\sim 2000^{\circ}\text{C}$ or less. The equilibrium temperature of the carbon curtain in a plasma chamber can be calculated given the heat flux and appropriate sink temperatures and emissivities. Assuming a 10 watt/cm^2 heat load, an effective emissivity coefficient of ~ 0.5 , and a first wall temperature of 600°C , we find that the curtain will heat up to 960°C . The vapor pressure of carbon at this temperature is less than 10^{-19} torr.

Graphite is commonly thought to adsorb gases readily because of experience with carbon and charcoal at high pressures and low temperatures. However, most gases adsorb on graphite. The rapid advance in graphite technology has led to new forms of graphite and carbon-carbon composites which have demonstrated extremely good properties in high vacuums. It has been shown that at low pressures, graphite is not as efficient a gas adsorber as was commonly thought. For example, the sticking probability for molecular hydrogen on graphites is $< 3 \times 10^{-6}$. The sticking probability of atomic hydrogen (the most likely form of hydrogen isotope in a fusion reactor) has been reported as 0.015 and 0.038 at room temperature and $p < 10^{-5}$ torr. Others have found that the maximum oxygen coverage on graphite at 300°K and 10^{-9} torr to be 0.02 monolayers. The sticking probability of CO on graphite is 5×10^{-5} and decreases rapidly as surface coverage approaches 0.01 monolayer. If we assume that in an extreme case, 0.3 of a monolayer of gas were to adsorb on graphite, then a curtain necessary to cover some 3000 m^2 of fusion reactor first wall would contain $\sim 500,000$ torr-liters of gas. However, most fusion reactors will have vacuum pumping systems capable of several thousand torr-liters per second

so that even this pessimistic assumption about gas adsorption will not present too great a problem to large reactor systems. In other words, this amount of gas could be pumped from the reactor and the pressure returned to $\sim 10^{-5}$ torr in less than one hour. Such a large pumping load would only be encountered after the chamber has been opened to air. (Presumably less than once a year.)

The amount of gas evolution in graphite after baking at 300°C for 24 hours has also been studied. It is found that the evolution varies from 5.7×10^{-3} torr-liters/gram for pyrolytic carbon to 6.3×10^{-2} torr-liters/gram for fine grain, extruded graphite. Again if we assume that worst case, we find that a maximum of 350,000 torr-liters would be evolved in a typical UWMAK-II type carbon curtain.

It is expected that prior baking at higher temperatures or in situ heating before plasma operation can significantly reduce the level of gas evolution compared to the cases considered above. The remaining gas can easily be handled by typical reactor vacuum systems.

Another area for concern is the effect of neutron bombardment on carbon cloth at $\sim 1000^{\circ}\text{C}$. Using a typical neutron spectra from UWMAK-II, we find that ~ 2500 atomic parts per million of helium is produced per year per MW/m^2 neutron wall loading. It is also calculated that the displacement damage amounts to $10.8 \text{ dpa}/\text{yr}/\text{MW}/\text{m}^2$. This is roughly equivalent to the damage produced per year by $1.5 \times 10^{22} \text{ n}/\text{cm}^2$ in a fission spectrum. Analysis of isotropic, near isotropic and anisotropic graphite irradiated from $1-2 \times 10^{22} \text{ n}/\text{cm}^2$ (fission) shows that there may be some shrinkage (a few percent) or even a few percent expansion. However, one cannot extend this information linearly with time as it is possible that "run away" swelling may occur in less than 5 years of $1 \text{ MW}/\text{m}^2$ wall loadings. (Direct comparison of current irradiation data on graphite to cloth material is difficult because radiation damage in carbon is so dependent on structure.

The induced radioactivity due to C^{14} , and resulting afterheat of such a curtain is also found to be negligible. For example, the radioactivity after one year of exposure amounts to 2×10^4 dps/g and the afterheat resultant from this decay is only 1.5×10^{-10} watts/g.

Some of the carbon will be removed by sputtering due to charged particle and neutron bombardment. We have calculated that the erosion rate will be approximately 11 microns/year of operation in a reactor like UWMAK-II. Some of this carbon will be transported into the plasma and eventually to vacuum pumps or divertor collector systems, while some of the carbon will deposit on colder sections of the plasma chamber. Since the carbon curtains would probably be greater than 2000 microns thick, there is no particular problem envisioned with respect to curtain thinning.

The compatibility of carbon and a hydrogenic environment also appears to be reasonable. It is known that at pressures of $<10^{-2}$ torr there is essentially no "methane" formation with molecular hydrogen above 400°C . Since the curtain will operate at $\sim 1000^\circ\text{C}$ and drop to the reactor ambient temperature ($\sim 500^\circ\text{C}$) during plasma recycling, there is little problem anticipated here.

From these considerations, it appears that the placement of a carbon curtain between the plasma and first wall of typical fusion reactors can have the following advantages:

- . It will intercept all plasma ions and neutrals from plasma, thus preventing blistering and charged particle sputtering of the metal walls.
- . It will prevent all neutron sputtered atoms from re-entering the plasma.
- . The curtain will effectively exchange low Z impurities for high Z impurities, thus reducing the power radiated from the plasma and the current required in tokamaks for ignition.
- . There will be a reduction in hot spots due to inhomogeneous particle fluxes (both in normal operation and in a failure mode).

. The use of a carbon curtain may allow removal of divertor on tokamaks. This concept could also be useful in near term plasma experiments where impurity levels will be critical.

Potential problems with using carbon in high neutron fluxes include large radiation induced growth and helium generation. Both of these problems could eventually require replacement of the curtain after a few years of operation. However, such a procedure is easily performed at relatively little cost.

D. The ISSEC Blanket Design Concept as a Method to Minimize the Radiation Damage to CTR First Walls from D-T Neutrons

The idea of the carbon curtain has been expanded to a 3-dimensional passive carbon shield which shifts, or tailors the neutron spectrum to lower energies. The spectral shifter is a non-structural element which can be made of graphite, silicon carbide, or three dimensionally woven carbon fibers (and containing other materials as appropriate) placed between the neutron source and the first structural wall. The softened neutron spectrum incident on the structural components leads to lower gas production and atom displacement rates than in more standard fusion blanket designs. In turn, this results in longer anticipated lifetimes for the structural materials and can significantly reduce radioactivity and afterheat levels. In addition, the neutron spectrum in the first structural wall can be made to approach the flux shape in fast breeder reactors. Such spectral softening means that existing radiation facilities may be more profitably used to provide relevant materials radiation damage data for the structural materials in these fusion blanket designs. This general class of blanket concepts is referred to as Internal Spectral Shifter and Energy Converter, or ISSEC concepts. Specific design concepts have been presented which fall into three main categories: ISSEC/EB concepts based on utilizing existing designs which breed tritium behind the first structural wall; ISSEC/IB concepts based on breeding tritium inside the first

vacuum wall; and ISSEC/Bu concepts based on using boron, carbon and, perhaps, beryllium to obtain an energy multiplier and converter design that does not attempt to breed tritium or utilize lithium.

A detailed analysis of the nuclear performance of the ISSEC system is given in a paper published in the June 1975 issue of Nuclear Technology. Only a few details of that work will be quoted here.

It was found that the two most sensitive materials parameters during neutron irradiation, displacements per atom (dpa) and atomic parts per million (appm) of helium gas generated, were reduced by a factor of 3 and 7 respectively by 12.5 cm of a carbon ISSEC. Increasing the ISSEC thickness to 25 cm reduced the dpa values by a factor of 10 and the helium production by a factor of 30 over the unprotected wall case. The breeding ratio in liquid Li (enriched to 90% Li-6) behind a vanadium wall was reduced below 1 due to the parasitic absorption of neutrons in the ISSEC and structure. However, this could be remedied by including Be_2C pellets in the ISSEC for neutron multiplication. It was also shown that such a Be_2C addition could produce a breeding ratio of >1 in solid breeding materials like LiAlO_2 .

Several concepts using tritium breeding compounds inside the vacuum vessel were explored and it was determined that a combination of C, Be_2C , and LiAlO_2 could produce a breeding ratio of >1 while reducing the helium gas production rates by a factor of ~ 12 and displacement rates by a factor of ~ 5 .

Finally, the use of ISSEC to ameliorate the damage in the first walls of a D-T "Burner" reactor was investigated. It was shown that the inclusion of B_4C in a normal 12.5 cm carbon ISSEC could reduce the helium gas production by a factor of 7 and the displacement rate by a factor of ~ 4 .

Analysis of the damage in the graphite reveals that dpa values per year of operation at 1 MW/m^2 vary from ~ 10 at the front surface to ~ 2 after 25 cm of carbon ISSEC thickness. The helium generation rates vary from 2700 appm/yr

to ~100 appm/year over the same thickness. The effects of such high helium contents at high temperatures still need to be investigated.

The most severe limitation of this concept is related to the heat removal rate. Currently, a 2000°C temperature limit means that the ISSEC thickness is constrained <12.5 cm. Methods of increasing this thickness are being pursued in the next two year's program.

E. Beam Driven Tokamak Studies

Our studies of driven Tokamaks continued this year with an examination of the potential of unignited, driven Tokamaks as thermonuclear reactors. We also examined the relationship between the TCT concept, driven reactors in general, and ignited plasmas.

In summary, it was found that the effect of injected power on the optimum Q is opposite in low current machines as compared to high current tokamaks. For I less than approximately 3 MA, it is best to inject the maximum power permissible of approximately 200 keV deuteron beams into a TCT having a 100% tritium target plasma. By contrast, machines with I greater than about 4 MA are IDT's and achieve the optimum Q by injecting beams into a near 50-50 D-T target plasma at the minimum power necessary to maintain a thermal equilibrium. The calculations indicate the plasma will ignite for sufficiently large plasma current and in this way one can see a natural transition from a TCT to an impurity dominated driven Tokamak (IDT) and then to an ignition machine as the plasma current is increased.

It was also found that the presence of a small amount of high Z impurity (typically, 0.1 to 0.2%) can prevent ignition of a tokamak plasma, as has been discussed previously, but that such systems could nevertheless be operated as high gain energy amplifiers. In such impurity dominated tokamak power reactors (IDT), impurity levels that are two to four times higher than in ignited systems can be tolerated without a large reduction in net power or in plant

efficiency. Driving an IDT with external injection power can also be useful as a method for extending the burn time if impurity levels increase as a function of time.

Optimum power amplification in an IDT requires energetic deuteron beams with energies between 150 and 300 keV in order to maximize the beam-plasma fusion rate and to minimize the beta component associated with the energetic beam itself. In addition, it was shown that such IDT systems are a natural extension of the TCT concept, which is important in understanding the TFTR as an integral part of the United States CTR program.

It was also found that one can tolerate roughly an order of magnitude more low Z impurity, such as carbon, and still have an ignited tokamak plasma. However, for carbon levels greater than about 5 to 7% of the electron density, the power output drops sharply. The reason is that for a fixed β , the fuel ion density is reduced to allow such large low Z concentrations. This is relevant in view of our recent discussions regarding low Z liners in tokamaks.

Importantly, beams are not a unique heating source for the IDT concept and the injected power levels required are fairly modest (between 25 and 150 MW). These are not beyond the technical potential of RF sources and the main open question is the coupling of RF power to the bulk plasma ions. If relatively uniform heating is possible at low frequencies, RF power would offer a potentially higher energy source efficiency and the possibility of added amplification due to beam-plasma fusions from an energetic ion-runaway distribution tail.

F. Plasma-Wall Experiments

It has become quite apparent from our conceptual studies that critical experiments need to be done on carbon liners and carbon cloths to verify the concept. We have initiated such studies and a brief summary of the work performed this year is given below.

Two types of plasma-wall experiments were performed this year. The first involved a measurement of the charge state of carbon atoms which were sputtered and then stripped by a hot plasma. The second experiment measured the radiation damage to bulk graphite and graphite cloth target, due to bombardment with He^+ and H^+ ions.

The carbon stripping experiment used a coaxial E x B gun as a plasma source. We measured a number of plasma parameters prior to looking at the stripping. Using a curved plate energy analyzer, we were able to determine that the ion temperature was 500 eV. This measurement was confirmed by time of flight measurements. The ion density was measured as 5×10^{12} particles/cm² by using a faraday cup. This gives a total of 10^{14} ions/cm² and a current density of 4 AMAS/cm². Using previous results, the electron temperature was estimated at 15 eV.

The E x B gun produces a pulse of hot H^+ plasma for a duration of about 10 μsec . About 5 μsec after the arrival of the H^+ pulse at the target a C^{++} plasma, produced from electrode material, reaches the target. We use a 1/3 meter monochromator to look at the spectral lines produced during the 10 μsec pulse of the hot H^+ plasma. The spectral lines corresponding to CII, CIII, and CIV were observed. A larger sputtering yield from the target was observed to the arrival of the C^{++} ions.

A cold H^+ plasma arrives at the target about 10 μ sec after the hot plasma. This delay insures that the carbon spectra are due to physical sputtering and not chemical sputtering, since there is no cold plasma present during the measurement.

The target (carbon cloth) surface temperature rise in this experiment is about 20°K. At this temperature, one would expect very little chemical sputtering, as will be seen in the results of the second experiment.

A computer code was written to give us time dependent stripping results. This code calculates the intensity of the spectral lines for a given electron temperature and density, as a function of the sputtering yield.

The formation of methane and acetylene was observed using a residual gas analyzer. Upon puffing hydrogen into the system, the partial pressure of methane rose by a factor of 2. When the plasma was generated, the partial pressure of methane rose by about a factor of 20. Since the E x B gun plasma is about 2% carbon, the introduction of the target had little effect on the production of methane. We observed a factor of 5 increase in acetylene upon puffing hydrogen into the system, and a factor of 20 increase upon firing the plasma.

The radiation damage experiment used a Hall accelerator, which was developed under NSF sponsorship for the production of negative ions. This source produces 800 eV ions of helium or hydrogen in a pulse of 2 milliseconds long, with a flux of 5×10^{18} ions/cm²sec. This allows a dose of 5×10^{15} ions/cm² in one pulse. With a repetition rate of one pulse every 80 seconds, we obtained total doses of 5×10^{18} ion/cm². After bombarding carbon cloth with helium SEM showed no qualitative evidence of erosion, but did reveal blister formation on the surface. The blisters were ellipitical in shape, with the major axis

of the blister transverse to the axis of the graphite fiber. There were about 5 bubbles per micron of fiber length.

The examination of the graphite cloth exposed to H^+ revealed significant erosion. The fiber diameter was reduced from about $8\ \mu$ to about $6\ \mu$ by the exposure. No evidence of blistering was observed. The temperature rise of the fibers due to bombardment was $530^\circ K$.

The lack of significant erosion upon exposure of the cloth to He^+ indicated that the large erosion upon exposure to H^+ is due to chemical sputtering. It has been observed that chemical sputtering with H^+ is a strongly temperature dependent phenomenon. Bulk graphite samples were bombarded by hydrogen. The temperature rise due to the bombardment was only $50^\circ K$, and no qualitative erosion was observed. Upon heating the samples to $600^\circ C$, we found significant erosion of the bulk graphite sample, in accordance with the expected temperature dependence of chemical sputtering. This observation of the surfaces was only qualitative, using a scanning electron microscope.

G. Neutron Cross Section Measurements for the Production of Long Lived Radioisotopes, H and He Atoms

The present investigation concerns measurements of cross sections for the production of isotopes of hydrogen and helium by 14 MeV neutrons in materials of interest to the development of fusion reactors. A knowledge of these cross sections is essential to the choice of materials for the first wall of a fusion reactor since the generation of hydrogen and helium affects the wall's mechanical properties.

The plan of the present study is to observe the charged particles produced in the reactions of interest to radiation damage problems with a detector placed far enough from the neutron source that it can be effectively shielded from the source. This requires the design of a transport system for

the charged particles from the sample where they are produced by the neutrons, to the detector. The first method considered was an electrostatic particle guide, a cylinder with a high voltage center wire. The charged particles drift down the cylinder describing helixes about the wire and are counted by a detector at the end of the cylinder. This method was abandoned because the voltage on the wire needed for the present work was too high.

The second approach involves the use of magnetic focusing. A computer program was obtained from the University of Wisconsin Physical Sciences Laboratory and run on the University of Wisconsin Univac 1110 computer to investigate the use of quadrupole magnet doublets and triplets.

The acceptance angles and particle trajectories for quadrupole doublets and triplets of various configurations were calculated as a function of particle momentum. These magnets act as low resolution momentum spectrometers transporting to the detector only part of the total momentum distribution. The calculations indicated that a triplet would have a larger momentum pass band and the triplet was chosen for further detailed analysis. As the detector count-rate is a function of target size, detector size and momentum, the change of the count rate was calculated for many values of each of these variables. The results indicated that the target dimensions should be less than those of the detector to give a momentum pass band of acceptable shape.

Neutron shielding of the detector has been considered and there will be adequate room for shielding materials. Because charged particles from the sample are transported to the detector inside an evacuated pipe, the inner pipe wall will be a source of background charged particles. Calculations of aperture sizes to shield the walls have been made.

The maximum and minimum reaction angles obtained for various source-to-target distances were calculated when the target, triplet and detector are moved as a whole back and forth past the neutron source. Linear translation

rather than rotation of the triplet about the target axis makes the support of the triplet much simpler.

Construction of the magnets for the triplet has started and the detector system has been designed.

This work is being carried out in close contact with the experimental physics division of Lawrence Livermore Laboratory (LLL). Preliminary experiments on backgrounds and charged-particle transport through magnetic lenses have been carried out at the LLL intense neutron source (RTNS), and these measurements confirm the feasibility of the proposed method.

These experiments utilized a magnetic quadrupole doublet that was available at LLL and two types of detector systems. In one system a scintillator was used with electronic particle-gamma ray discrimination to eliminate pulses from neutron-induced gamma rays. The measurements showed that while such discrimination reduced the gamma ray background, there was a charged-particle background larger than the signal. Thus the scintillator can not be used alone. The other system consisted of two silicon detectors 8 mm in diameter, one a thin dE/dx detector, the other a thick energy detector. They were used as a fast coincidence telescope with electronic charged particle identification. With this system protons from the $Al(n,p)$ reaction could be observed, but protons from $Nb(n,p)$ could not be distinguished from the background.

A program has been started to explore whether 14-MeV neutrons induce long-lived (half-lives of years) activities in components of fusion reactors. High purity metal foils of elements of interest to reactor design have been assembled and are placed near the 14-MeV RTNS source at LLL wherever this source is operated at high intensity for extended periods. In order to emphasize long half lives the irradiation will be continued for about a year, then the short lived activities will be allowed to decay before an attempt will be made to identify previously unobserved long activities. A determination of the

half life of ^{92}Nb produced in the reaction $^{93}\text{Nb}(n,2n)^{92}\text{Nb}$ is being carried out by the LLL radiochemistry division. Preliminary results differ by a factor of about three from previous measurements.

H. Experimental Testing of Magnet Conductor Design

The design and initial fabrication has been completed on a device to experimentally determine the integrity of the magnet conductor proposed last year. The primary intent of this device is to determine the areas of possible degradation in a stainless steel test ring containing an embedded copper conductor bonded into the ring with fiberglass reinforced epoxy and subjected to cyclic loading at cryogenic temperatures.

Figure 1 illustrates the overall design of the test fixture and associated dewar. The conical test fixture will be subjected to cyclic compressive loads which will generate a radial expansion of the test ring and its imbedded conductor. The radial expansion of the conductor ring will produce tensile circumferential strains similar to those expected in an inservice situation.

Strain gages fixed to the conductor will supply the necessary feedback to control the hydraulic pressure applying the load to the conical test fixture. Other gages will serve as a means to monitor the structural response of the test ring and conductor and the progression of any failure which might occur.

An applied radial displacement on the test ring will produce a nonuniform stress distribution and potentially high localized shear stresses at the interfaces of the several materials. To determine how the test ring and the conical test fixture should be fabricated, a finite element stress analysis of several ring designs has been initiated. The variation of circumferential normal stress in two possible cross-sections of the ring serves to illustrate the influence of the conductor position in the ring. From this analysis, the position of the imbedded copper conductor can be determined in order to produce

the desired stress and strain distributions. The distribution desired is the one which most nearly approximates the predicted stresses in the actual toroidal field magnets. This finite element analysis also allows the simulation of various boundary conditions which the conical test fixture may apply to the conductor ring. The effects of varying the distribution of pressure on the inner edge or of varying the radial displacements of the inner edge have been investigated.

EXPERIMENTAL ASSEMBLY

III. Industrial Participation

Fiscal year 1975 was very productive in bringing in industrial scientists to the fusion technology area. This year we have had cooperative efforts with 11 different organizations.

. Monsanto Corporation	. United Aircraft Research Laboratories
. Babcock and Wilcox	. McDonald Douglas Laboratories
. Sargent and Lundy	. Aerojet General
. Atomics International	. Hanford Engineering Development Laboratory
. Battelle Northwest	. Fiber Materials Inc.
. Bechtel	

The number of individuals in these firms which have played a substantial role in the design studies is 18. We expect to be working with many of these companies in the coming year as well as several organizations which are now laying plans for cooperation in FY 76.

IV. Other Points of Interest

We have continued our past efforts to provide tutorial lectures in the area of fusion technology for professional, governmental, industrial and social organizations. A list of the activities along these lines is attached and we expect to continue such work in the future.