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Proceedings of the
INTERNATIONAL WORKING SESSIONS ON

Fusion Reactor Technology

JUNE 28-JULY 2, 1971

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee, U.S.A.

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PREFACE

The field of fusion reactor technology has as its objective the identification and investigation of the technological requirements of power by nuclear fusion. Recent advances in plasma confinement coupled with a cautioned optimism in the scaling behavior of fusion devices has generated a growing interest in fusion reactor technology. As evidence of this interest I note that in the past two years two conferences were devoted to fusion reactor technology (the International Conference on Nuclear Fusion Reactors, Culham Laboratory, Culham, England, September 1969 and the Symposium on Thermonuclear Fusion Reactor Design, Texas Tech University, Lubbock, Texas, June 1970), prior to this period only a few isolated papers concerning fusion reactor technology could be found in conference proceedings. Furthermore, in June of this year the IAEA for the first time included a session on fusion reactors in a plasma physics conference (IAEA Fourth Conference on Plasma Physics and Controlled Fusion Research, Madison, Wisconsin, June 1971).

The International Working Sessions on Fusion Reactor Technology had a threefold purpose: (1) to review the state-of-the-art in fusion reactor technology, (2) to assess the work done since the Culham Conference, and (3) to identify areas for future work. Interest in the Working Sessions was most enthusiastic. There were about 150 attendees, including representatives from Canada, France, Germany, Italy, Japan, Russia, the United Kingdom, and the United States.

The program consisted of eight topical sessions, two panel discussions, and a summary session on the final day. Two concurrent sessions were held on each day (unfortunately, this did present some conflicts of interests). Each session started with a state-of-the-art review by the session chairman. The session agenda was organized by the chairman in cooperation with the session attendees. Althougn participants did not present formal papers at the sessions, they did report on their work, and there was a substantial exchange of ideas and information. Because of the large number of attendees, the sessions evolved into seminars rather than workshops, however, an informal atmosphere still prevailed.

The Proceedings of the Working Sessions include the State-of-the-Art Presentation and the Summary of Session reported by the session chairman. In most cases the Presentation and the Suumary appear as two separate contributions, however, the chairmen of the sessions on Blanket Design, Plasma Fueling and Recovery, and Energy Conversion Systems have combined the Presentation and the Summary into one contribution. The Proceedings also include some delightful sketches which David J. Rose presented at the closing of the Summary session.

The contents of the Proceedings require no further explanation, however, I feel that some general observations about the field of fusion reactor technology are in order.

First, it must be realized that the scope and emphasis of fusion reactor technology studies reflect our current understanding of plasma physics. Since unexpected results in plasma physics may emerge, we must be prepared to alter some of our basic views with regard to the technological requirements of power by fusion.

Second, the field is only in its infancy, which manifests itself in several ways:

(a) There has been very little experimental work to date. The studies are primarily paper studies based on conceptual fusion reactor designs.

(b) It is a rapidly expanding field. Ideas are being generated at a rapid pace and it is difficult to assess which ideas merit pursuit.

(c) There is a wide range of background experience in various areas of the technology studies. For example, in the area of neutronics we can draw upon two decades of experience in the design of fission reactors. On the other hand, in the area of fueling and recovery of unburned fuel we have essentially no technological experience upon which to draw. A collarary to this situation is that many investigators have considered neutronics, but few have considered fueling and recovery of unburned fuel.

(d) It is premature to project figures for the price of power produced by fusion since fusion power will involve several as yet undeveloped technologies.

Finally, let me offer an answer to a question which is often asked of people investigating the technology of fusion reactors, and that is, "Is it not premature to seriously consider the technology of fusion reactors?" Anyone working in fusion reactor technology must have a basic optimism with regard to the potential of fusion as a source of energy although various workers in the field may suggest different timetables for the realization of fusion power. On the basis of this optimism technology studies are necessary in order to insure that the technology of fusion reactors is consistent with the physics of fusion reactors. Moreover, if if on the one hand we promote fusion as a desirable source of power, on the other hand we are obligated to assess the impact which fusion power may have on our society. Such an assessment requires an understanding of fusion reactor technology.

At this point I would like to acknowledge those groups and individuals who contributed to the success of the Working Sessions. First, I would like to thank the sponsors, the Oak Ridge National Laboratory and the American Nuclear Society Technical Group on Controlled Nuclear Fusion, for their cooperation. In particular, I wish to acknowledge the valuable assistance of Charles E. Normand, ORNL Conference Coordinator. My deep appreciation is extended to the session chairmen, the panel moderators, and to Floyd L. Culler, Deputy Director of ORNL, for his very instructive welcoming address. I happily acknowledge the valuable comments of many colleagues, especially Herman Postma and David J. Rose. Bettye Pope has undertaken the secretarial work of the Working Sessions and the Proceedings in addition to her regular duties. To Bettye go my heartfelt thanks for a job well done.

Don Steiner
Organizing Chairman
Editor of Proceedings

TABLE OF CONTENTS

SESSION 1

ENGINEERING DESIGN OF BLANKETS

Chairman

A. P. Fraas
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SUMMARY OF SESSION

ENGINEERING DESIGN OF BLANKETS

A. P. Fraas
Oak Ridge National Laboratory

Developments Since Culham

The Culham Conference highlighted the seriousness of the radiation damage problem at the vacuum wall. As a consequence of this, an appreciable effort in several quarters has been directed toward the possibility of reducing the vacuum wall power density. This is primarily a question of costs and most particularly a question of magnet cost. If these costs can be kept sufficiently low, it becomes quite practicable to employ power densities of the order of 1 MW/m^2 instead of the 10 MW/m^2 contemplated in the designs considered at Culham.

Data presented at Culham indicated that fusion reactors should give enormously reduced inventories of volatile radioactive materials relative to fission reactors and thus greatly ease reactor safety problems. This in turn has led to a mucn closer look at the problems of 14 MeV neutron activation of the structure, particularly the vacuum wall. The initial estimates of niobium activity made at ORNL early in 1969 took account only of the activity of ^{94}Nb. Subsequent work yielded higher values, and R. S. Pease, while visiting from Culham Laboratory, called our attention to the much higher levels of activity resulting from fast neutron interactions with niobium as found by Stephen Blow at Culham early in 1970. This led to a more comprehensive analysis of both fast and thermal neutron effects and still higher values for the total activity and the afterheat of niobium. These estimates were made in July 1970 by Don Steiner at ORNL. The calculations of Blow and Steiner showed that there is a strong incentive to reduce the amount of structure in the region near the vacuum wall. Further, this turned attention to possible materials other than niobium that might give a much reduced activity. Work by Don Steiner at ORNL indicates that the use of vanadium in place of niobium should give a much lower amount of activation of the structure.

So far as hazards to the general public are concerned, the tritium inventory represents the most serious consideration because the tritium is volatile. This in turn has led to further work on tritium recovery systems designed to minimize the tritium inventory and to minimize the amount of tritium percolation into the atmosphere or the steam system.

Presentations at the Blanket Session

John Mitchell presented the latest notions as to what the blanket and shield region might look like in a full-scale fusion reactor. This design concept was based on use of a 75 cm-thick shield region containing a large amount of steel to attenuate the fast neutron flux as rapidly as possible. The mass of steel would also be used as the basic structure for the reactor, and the segments of the blanket region would be mounted on it as large studs projecting radially inward from what John Mitchell referred to as the "submarine hull" structure of the shield. These studs would be approximately rectangular prismatic cans filled with lithium. The lithium would be cooled by boiling potassium in a system of passages inside the can. Capillary surfaces would be employed to distribute the liquid potassium feed over the heat transfer surfaces. This in turn raises the problem of methods for controlling the feed flow rate in such a way as to assure that all surfaces would be supplied with liquid potassium, but that none of them would be flooded with an excessive supply.

Discussion of this design led to questions regarding the extent to which activation of the potassium close to the vacuum wall might prove to be a problem, thermal stresses and possible cracking of the structure enclosing the lithium, and the extent to which the iron-water shield would be adequate to protect the magnet from excessive heating by secondary gamma rays. These problems have been recognized at Culham but work has not progressed to the point where the questions can be answered.

Fred Ribe presented the advanced thinking at Los Alamos on their Theta- and Z-Pinch reactors. Their current concept entails the use of a large superconducting magnet as an energy storage device. The superconducting coil serves as the primary winding of a transformer. Interrupting the flow of current in the superconducting winding gives a powerful surge of current in the secondary winding which would represent a single turn around the plasma region of the pinch machine.

The large toroidal transformer is surrounded by a second toroid with a larger major diameter and a much smaller minor diameter lying in the same plane. With this energy storage device it is estimated that they could cut their recirculating power from 25% to 10%.

Gus Carlson of Livermore presented a nice study of the EM pumping power required for a number of idealized blanket geometries. This study disclosed that the EM pumping power required for a mirror machine in which the flow channels follow the magnetic field lines fairly closely would be quite acceptable, but, if the flow passages deviate widely from the magnetic field lines, the pumping power will be excessive. He extended his parametric studies to Tokamak configurations, varying the conductivity of the wall as a major parameter. Results of these estimates indicate that the electromagnetic pumping power for the Tokamak-type machines would be excessive for even the lowest electrical conductivity that it seems possible to obtain in practice if the lithium must be pumped in and out of the magnetic field. However, it does not preclude circulation of the lithium within the magnetic field and the use of a suitable coolant circulated through a heat exchanger in the blanket.

Roth of NASA presented some interesting pictures indicating how a ferromagnetic fluid might be used to cool the blanket and at the same time actually produce some net power output. This would be done by allowing the fluid to be drawn into the magnetic field and then heating it above the curie point so that it would flow out of the magnetic field without a need to overcome an electromagnetic force. The difficulty arises in finding a ferromagnetic fluid with properties suitable for use in a fusion reactor.

Roger Hancox of Culham presented some studies they had made on the magnetohydrodynamic pumping problems. In this work the wall power density and the magnetic field strength were related to the ratio of the coolant flow passage to the total vacuum wall surface area, the temperature rise in the coolant, a representative dimension of the coolant passage, the specific heat of the coolant, the electrical conductivities of the coolant and the wall, and the coolant density. The results of this study indicate that the maximum power density obtainable with a 50 kG field would be

1200 W/cm^2 of vacuum wall, and that for a 100 kG field the maximum of power density would be only 300 W/cm^2 of vacuum wall. Higher power densities would require higher coolant flow velocities and these would entail unacceptable pumping power losses.

S. Förster of Jülich presented a summary of their helium-cooled blanket which would be coupled to a gas turbine. The studies at Jülich indicate that the volume fraction of niobium plus helium in the blanket could be held to 4% for operation at 1000°C and 60 atm in the helium system. If the molybdenum alloy TZM were employed, the molybdenum and helium volume fraction could be reduced to only 3%. The heat transfer matrix employed in the blanket would consist of 1-cm diameter tubes on 3-cm centerlines. The helium would flow radially outward through a large passage and then would branch into a set of about ten 1-cm diameter tubes radiating from the large supply tube. The 1-cm diameter tubes would pass back in sinesodial fashion to the outer perimeter of the blanket. Förster estimates that the overall thermal efficiency obtainable with this system would be 46%. Fraas pointed out that his discussions with Curt Keller at Escher Wyss in Switzerland had indicated that this high thermal efficiency probably entails an excessive investment in hear exchanger equipment both in the interstage compressor coolers and in the recouperator so that an economic optimum would probably be obtained at a thermal efficiency of perhaps 38%.

One of the problems associated with helium cooling is the presence of high pressure helium in the blanket and the associated hazard potential. Another problem depends on the rate at which heat is generated in the vacuum wall. If about 15% of the total power is deposited in the vacuum wall by x-rays, ion conduction, etc., the local variations in vacuum wall temperature will run around 150°F for a total wall loading of 1 MW/m^2 and helium cooling passages on 2-inch centerlines. The consequent thermal stresses are probably acceptable in niobium, but lower values would be highly desirable.

Estimates of the fraction of the total power that will appear as heat in the vacuum wall varied from 1% to 31%. In view of the enormous difference in the difficulties of cooling the vacuum wall for these two extreme values, it seems highly desirable to establish the vacuum wall heating rate for each of the various types of fusion reactor.

There was recognition of the large reduction in heat transfer coefficient for forced convection of liquid metal as a consequence of suppression of eddy diffusivity in a strong magnetic field, but no analyses have been made to show the consequences of this. A. P. Fraas reported that he has initiated a parametric study of the temperature distribution in typical geometries assuming no eddy diffusivity, just thermal conduction, but no results have been obtained as yet. Rough preliminary calculations indicate that it will be possible to get an acceptable temperature distribution with a feasible geometry.

The last portion of the session on blanket design was devoted to a discussion of key problems much in need of attention. These are as follows:

1. Eddy diffusion of heat in magnetic field.
2. Permeation of H, D, and T through structural metals (Nb, V, TZM, SS, superalloy).
3. Effects of radiation on permeation.
4. Effects of T decay on metals.
5. Combined effects of radiation damage, fluid environment, temperature on structural walls.
6. Studies of blanket conceptual designs with a view to relative cost, neutron economy, induced activity, serviceability, fabricability, EM pumping power, blanket thickness, wall power loading, etc.
7. Relate surface radiation effects, especially heat generation in the vacuum wall, to reactor parameters (plasma confinement, injection, divertors, etc.)
8. T removal system.
9. Refractory metal fabrication.
10. Divertor geometries, divertor wall heat loadings, etc.
11. Boiling heat transfer might be affected by a strong magnetic field (nucleation sites are crucial).

DISCUSSION

A. GIBSON (Culham): I would like to add a comment that in the stellarator the field lines outside the separatrix come out through the blanket and around the windings so there is in principal a possibility of having lithium flow along the field lines. Furthermore by the time a Tokamak has been equipped with an axisymmetric divertor a similar situation will exist for that configuration.

FRAAS: I'm anxious to see a drawing of what these things would look like. I've pled for one and still haven't seen one. I agree, that in principal, it ought to be that way.

GIBSON: I think you'll find a drawing published in the papers of Gourdon for the Torsatron configuration.

G. MILEY (U. Ill.): Could you comment on the severity of the tritium problem for a helium-cooled system in these cans versus the other systems?

FRAAS: These present a different set of problems. This was not discussed at our session but the tritium permeation rates are very much dependent on the presence of oxide films. Either on the water-side of a heat exchanger between the liquid metal and the steam systems, or on the helium-side of a heat exchanger, one could manage to generate an oxide film or perhaps coat it with a layer of ceramic and in this way inhibit the permeation across that boundary. In my own opinion, the really key question is the degree to which we can do this. It will make an enormous difference in the practicality of keeping that leakage down. So I think it can be handled. I think that the helium system presents a different set of problems. I don't see that they are less tractable than those in other systems, except for one thing, and that is, remember what one is trying to do is get a very low concentration of tritium in a fluid. We happen to have experimental data which shows that you can get an exceedingly low concentration of hydrogen (actually the experiments that we carried out here with deuterium) in a potassium system. If you put in a couple of percent of lithium into NaK (our work was done with NaK not potassium) it is soluble and it will grab any hydrogen there and form a lithium hydride which is soluble at high temperatures but is exceedingly insoluble at temperatures near the melting point of the NaK and it can be removed with a cold-trap. We have run NaK systems, not recently, but some seven years ago, and obtained concentrations of deuterium in NaK containing a couple of percent of lithium, which are adequate to assure that the permeation of tritium into a steam system or into a helium system would be trivial.

G. CARLSON (LRL): As Art indicated, the example calculations that I made for lithium coolant pressure drop in example fusion reactor blankets, did seem to show that it could be done in open mirror systems with a sufficient amount of care in the design but I was rather pessimistic about the possibilities for closed systems. Alan Gibson mentioned the possibility of using a divertor in the toroidal systems to gain an exit path for the fluid flow, and that is a possibility that I did not consider. I didn't feel qualified to specify the magnetic field configuration of the divertor because I don't know anything about divertors. The other point Art made about the possibility of circumventing the problem by an internal circulation of lithium, not bringing the lithium out of the blanket but taking the heat out of the blanket with another fluid is a possibility, but I would like to caution you that the pressure drop of the internal circulation of lithium has to be calculated. You can't just assume that it's small, and that's a calculation that I have not done. I probably will look into it now. And finally, Art said you could take the heat out through a heat exchanger with potassium. Some of you that were not at the session may wonder what the difference is between potassium and lithium. In the particular system or idea that Art's talking about you don't take the potassium out as a liquid, you bring it in as a liquid and out as a vapor so the flow velocities are very much reduced because you are taking your energy out as latent heat. That is the possibility which would result in low magnetic pressure drops.

R. ROTH (NASA): I think your remark about a one foot ahead due to capillary forces might be unduly pessimistic, although it's true of technology at present. There are trees that manage to lift a head of water over a hundred meters through capillary and osmotic forces, so perhaps the design of a fusion blanket ought to start with the study of trees.

J. MITCHELL (Culham): I had hoped to show in my talk our beliefs that you can put enough steel in the shield region to support the whole blanket system in space unlike Art's system which is floating in water.

FRAAS: John proposes a massive steel and water shield, like a submarine hull, so that it will be a sturdy structure from which you could hang this relatively flimsy blanket.

MITCHELL: I only say it can be massive, if you need it.

FÖRSTER (KFA): I would like to make a very short statement about the tritium inventory in the plant. This inventory depends strongly upon the electrical efficiency of the plant. If, for instance, one goes up in efficiency from 40 to 60%, the total inventory could be lowered by a factor of about 35%. I believe this would reduce the problem of environmental pollution and also the contamination of power site.

G. GRAVES (LASL): I have just one comment. When you speak of flowing these fluids along the magnetic field lines, I think we should keep in mind that we have to look at the integral of the energy uptake that's involved in the flow through the reactor. In some cases, I don't see this going one single pass through the system without taking up an excessive amount of energy. You have to get around that problem somehow.

FRAAS: I agree. I think it's going to be very tricky to come out with a satisfactory arrangement. I hope I didn't give the impression that it's easy; it's a question of whether it seems possible, or not.

D. STEINER (ORNL): I have one question about the helium cooling system. Is the idea there to recover the tritium by allowing it to pass through the metal walls into the helium system and come out with the helium? Is Ernie Johnson here by any chance? Do you have any comments about the difficulty of tritium removal using helium sparging in a lithium system relative to its difficulty in a flibe system?

E. JOHNSON (PPPL): The question of taking tritium from helium is not a serious problem at all.

STEINER: But getting it out of the lithium system. Would that be any problem?

JOHNSON: I think not, but we need two kinds of information there. Now Art Fraas talked about cold trapping from NaK using lithium in a sense as a getter. The situation in using cold trapping from liquid lithium is a little more difficult because the solubility is quite a bit higher. You could stick mercury in the system and knock the solubility out of sight. But I think if we had (1) some good information on the rate of chemical combination of lithium and tritium over a high temperature range and (2) good information about the solubility of tritium in lithium at very low concentrations, and in particular the equilibrium pressure exerted by the tritium at low concentrations, we could be pretty definitive in coming up with a scheme for removing the tritium to what seems to me extremely low concentration levels. One comforting thing is the likelihood that in the vapor phase there would be insignificant recombination of lithium in tritium. If that's the fact of the matter, then it seems to me that you have not too great difficulty in arriving at schemes which would reduce the tritium content in lithium to whatever level you consider necessary. The question then is one of economics, i.e., how much money are you willing to spend to achieve a given tritium concentration.

SESSION 2

PLASMA HEATING AND IGNITION

Chairman

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STATE-OF-THE-ART
PRESENTATION

HEATING AND IGNITION MECHANISMS FOR FUSION REACTOR PLASMAS

H. K. Forsen
University of Wisconsin

Before trying to outline the various schemes that may be used to heat reactors, it is necessary to consider just how far one has to carry the external heating to bring about ignition.

Generally the ignition temperature of a particular fuel cycle is that temperature at which bremsstrahlung losses are just balanced by fusion power production. That is

$$4.8 \times 10^{-37}\, n_i^2\, Z^3\, T_e^{1/2} = n_1 n_2 \langle\sigma v\rangle Q.$$

This is usually solved for $T_e = T_i$ and is density independent.

The Lawson criterion[1] also gives the requirement on density and confinement ($n\tau$), requiring that their product exceed some temperature dependent value which centers around 10^{14} cm^{-3} sec for D-T.

A better calculation of the ignition requirements is to write down an equation for the power density as a function of temperature. This equation must balance losses due to bremsstrahlung and energy or particle transport against gains. The gains are from the release of fusion reaction products, which we assume to rapidly thermalize, plus whatever external power one adds from the outside. When no external power is required to balance the system,it can be said to be ignited. Sweetman and others[2] have calculated this for a D-T system and using a pessimistic Bohm loss rate they find a maximum of 0.25 watt/cm^3 is required.

Turning now topossible methods of producing this power, Fig. 1 shows most of the heating methods and on which systems they might be applied. It is pointless to suppose to review in detail all these methods in this introductory review and therefore let us merely point out some aspects of each method. There are experts on most of these topics; they will have important points to make and can perhaps tell us whether the question marks should be "yes" or "no" and whether the "yeses" should be question marks.

OHMIC OR RESISTIVE HEATING - TURBULENCE

Ohmic or resistive heating, where the plasma currents are used for stabilization, may be limited in the case of tokamaks because of the limiting stability factor and the disappearance of anomalous resistivity at higher densities. We also have the problem in classical heating that $P = I^2R \propto I^2/T^{3/2}$. While this is an open question at this time, it is well known that if the current flow density $j > nev$ corresponds to a speed which is in excess of that given by $v_s = \sqrt{T_e/m_i}$ or the ion acoustic speed, anomalous resistance is encountered due to ion acoustic turbulence so long as $T_e > T_i$. Current tokamaks do not operate at this high a current density but they have been proposed. On the other hand, present tokamaks do operate in the range where $j > nev_{th}$ or where the drift speed is greater than the electron thermal speed. In this case the plasma is unstable to current driven electrostatic ion cyclotron waves.

In general the name turbulence is related to almost anything in which anomalous small-scale plasma behavior occurs. For the specific application we are interested in here, we find that the process of setting up electric fields in a plasma which are greater than that required for electron runaway will cause strong turbulence.

In linear devices two stages of heating appear to occur. The first is connected with the development of an ion-acoustic instability and the second with the generation of a fast electron beam and the resulting development of a two stream instability.

In closed systems, large driving electric fields have not led to significant runaway electrons because large amplitude fluctuating potentials tend to give rise to anomalously large collision frequencies (given approximately by the Buneman[3] formula, $\nu \approx (m_e/m_i)^{1/3}\ \omega_{pe}$). This causes greatly enhanced resistivity with increased electron and ion heating. Some measurements indicate efficiencies approaching 60%.

R-F HEATING

Here we consider several kinds of R-F heating: ECH, ICH, TTMP, and other proposed electromagnetic heating schemes.

Electron Cyclotron Heating

In ECH, an EM wave at the cyclotron frequency of electrons is caused to illuminate the plasma. This requires high frequencies (2.8×10^{10}/kG) where it is difficult to get significant amounts of RF power for long periods of time.

If the plasma density is sufficiently low such that $\omega_{pe} < \omega_{ce}$, the plasma dielectric properties may be ignored and the RF waves generated by external sources will interact directly with the plasma electrons. At higher densities the plasma particles interact collectively to reflect the waves and other coupling mechanisms must be invoked. However, in this case the waves still penetrate and heat the plasma to a skin depth.

In either case the wave can take energy from the particles or give energy to the particles depending on the phase unless some nonreversible conversion through collisions occur. Piliya and Frenkel[4] give us the heating rate in a uniform field as

$$\frac{d\omega}{dt} = \frac{e^2 E_\perp^2 \nu (\omega^2 + \omega_{ce}^2)}{m[(\omega^2 - \omega_{ce}^2)^2 + 4\omega^2\nu^2]} + \frac{e^2 E_\parallel^2 \nu}{m\omega^2}$$

where $\nu \ll \omega$ and is the effective collision rate whether due to particle-particle collisions, turbulent or stochastic collisions and/or wave-particle collisions.

Ion Cyclotron Heating

Effects similar to ECH can occur when illuminating a plasma with RF radiation at the ion cyclotron frequency. Here the frequency is considerably lower (1.5×10^6/kG for protons) where higher powers are possible but the coupling is considerably more difficult.

Collective coupling has been shown to be effective by Stix[5] using slow cyclotron waves where proper conditions of density, wavelength and frequency are met. For good coupling of wave energy to particle motion the RF structure is wound to match the optimum wavelength and should be located in a high magnetic field region. As the particles move into a weaker field, wave dispersion occurs and field energy is transferred to ions when $\omega_{rf} = \omega_{ci}$.

Several experiments employ such heating but the principal use has been on stellarator configurations where at densities of 10^{12}-10^{13} with several hundred volt ions but colder electrons were obtained.

For reactors certain problems are posed by this mechanism. These include the requirement of a magnetic beach, possibly the need to propagate around a toroidal geometry, the losses in the surrounding metallic walls, and prevention of damage to coil structures.

Transit Time Magnetic Pumping

This method of energy coupling to a plasma also requires coils around the plasma but the spacing and number is more advantageous. Wall absorption will still be a problem.

The principle is to perturb a closed magnetic surface with a frequency $\omega \ll \omega_{ci}$ and with a coil spacing to provide a wavelength such that the wave phase velocity (v_ϕ) is about equal to the ion thermal speed ($v_{\theta i}$) but that $\lambda \gg \rho_i$,the ion gyroradius.

In this case the direct conversion of EM energy into ion energy occurs rather uniformily (20-30% radial variation) because the general effect is due to the ions magnetic moment in the ∇B produced by the wave.

If the plasma is cold the coupling will be poor because of the need to match the phase velocity of the wave at low and high temperatures with the ion thermal velocity.

Wort[6] gives the energy input per unit volume as

$$P_p = \pi^{1/2}\omega \quad nkT_i \frac{v_\phi}{v_\theta} \quad b^2 \left[1 + \frac{T_e^2}{T_i^2}\right] \quad \exp\left(- \frac{v_\phi^2}{v_\theta^2}\right)$$

where $b = \Delta B/B_o$.

For the case where $T_e \stackrel{\sim}{\sim} T_i$ and $v_\phi \stackrel{\sim}{\sim} v_\theta$

$$\frac{dT_i}{dt} = 0.43\ \omega\ T_i b^2.$$

However, since spacing determines λ and thus ω, ω is determined by v_θ or $T_i^{1/2}$ and therefore

$$\frac{dT_i}{dt} \sim T_i^{3/2}$$

which also shows why coupling to a cold plasma is poor.

Canobbio[7] has shown that for TTMP in tokamaks that relatively high current flow can inhibit heating by overstability. This is a consequence of the effect of collision. He shows the heating rate is maximum in the intermediate collision frequency range $\nu_c < \nu_{ii} < \omega$ where it does not depend on collisions. It is ν_{ii}/ν_c smaller at low collision frequencies because of nonlinear distortion of the distribution function of resonance particles. A critical collision frequency (ν_c) has been calculated from neoclassical theory for the various regimes.

Other schemes for wave heating have been proposed and these include off resonance ECH heating which has been observed. At first glance it offers no clear advantage over direct ECH but perhaps something is yet to be developed.

For ions, fast magneto-acoustic waves with $\omega << \omega_{ci}$ or Alfven waves with $\omega \gtrsim \omega_{ci}$ are porposed with various absorption mechanisms. In the latter it could occur on a magnetic beach like regular ICH. In the former proposed by the Karkov group, absorption can occur under conditions of Cherenkov absorption by electrons or if the wave is perpendicular to the magnetic field, absorbed by resonance ions under the resonance condition $\omega \stackrel{\sim}{\sim} 2\omega_{ci}$.

ADIABATIC COMPRESSION

Because the magnetic moment is an adiabatic invariant for charged particles in a magnetic field, one can heat particles by increasing their perpendicular energy simply by increasing the magnetic field slowly compared to the gyrofrequency.

In actual practice compressional heating can be accomplished such that it is not adiabatic and as a consequence it can appear to be 1, 2 or 3 dimensional compression. Rose and Clark[8] give us the following summary:

For an adiabatic compression an ideal gas has PV^{γ} = const. where γ is the ratio of specific heats and is related to the number of degrees of freedom δ by $\gamma=(2+\delta)/\delta$. Since PV = RT we also have

$$\frac{T_2}{T_1} = \left(\frac{V_1}{V_2}\right)^{\gamma-1} \quad \text{for an equilibrium situation.}$$

For an initially isotropic distribution we find for an initial total energy U_1 that[8]

$$U_2 = \frac{1}{3} U_1 \left[3 - \delta + \delta(V_1/V_2)^{\gamma-1}\right]$$

For a one-demensional compression $\delta=1$ and $\gamma=3$ and the heating is greater than a two-dimensional compression which is in turn greater than that in three dimensions.

Such heating is proposed for the PPPL Tokamak.

SHOCK HEATING

Both collisional and collisionless shocks have been studied and are possible heating mechanisms. Generally shocks are driven by some form of piston such as a rapidly changing electromagnetic or magnetic field and they represent the propagation of energy into the surrounding medium. If the medium is immersed in a magnetic field, the propagation of the shock is strongly influenced by the angle θ between the field and shock propagation direction. Collisionless shocks are shocks which have a rather sharp boundary where the transition zone thickness is less than a collisional mean free path. These shocks exist only for Alfven Mach numbers ($M_A = u_s/v_a$) greater than $[2(1+\sin\theta)]^{1/2}$, where u_s is the shock speed and v_a is the Alfven speed.

It has been predicted and found that, for Alfven Mach numbers greater than a certain value (2.5-3), the shock is not laminar and various electron-ion two stream instabilities, ion acoustic waves and a general turbulence sets in which results in dissipation in the shock front. In this case the shock thickness becomes equal to the ion Larmor radius.

For $M < M_{crit}$ the ions are heated by the boundary ohmic currents as if the plasma resistivity were anomalously high by a factor of about two. This is beginning to be understood as weak turbulence due possibly to an electron cyclotron drift instability.[9] Such heating is important in many shock tube experiments.

For $M > M_{crit}$ nonadiabatic ion heating takes place again due to turbulence and such heating is important in fast rising Z- and θ-pinches as well as some fast plasma guns.

LASERS & RELATIVISTIC ELECTRON BEAMS

During the past few years considerable interest has been focused on extremely high power, short pulse devices for the production and heating of plasmas. For this, both laser heating and relativistic electron beams are proposed. Let us take lasers first.

In shining photons at solid D-T pellets the simplest absorption mechamism is inverse bremsstrahlung. This is true only so long as the plasma is underdense to the photon wavelength because again $\varepsilon = 1 - (\omega_p/\omega)^2$. The absorption length for this process is given by[10]

$$\lambda_{ab} = 5 \times 10^{27} \frac{T_e^{3/2}}{n_e^2 Z \lambda^2} \left(1 - \frac{\lambda^2}{\lambda_p^2}\right)$$

where here the λ's are in cm and λ_p is the vacuum wavelength at the plasma frequency, T is in eV and n_e is in cm^{-3}. From this we see that the radiation penetrates the plasma only if the term in brackets is positive definite or that $n_e\lambda^2 < 10^{13}$. If the wavelength is short compared to that at the plasma frequency or more simply if the plasma high frequency dielectric constant is close to unity, then the absorption length varies as $1/n^2\lambda^2$ for fixed T_e. Optimization of this process is obviously possible.

Other absorption processes are, of course, inverse cyclotron absorption or electron cyclotron absorption. For these mechanisms to be important requires extremely large magnetic fields or long wavelength lasers and, of course, both may become possible.

High power 10.6 μm N_2-CO_2 lasers can effectively heat plasmas in the density range 10^{17}-10^{19} particles/cm^3. The absorption length of this radiation is a few 10's of cm to km depending on the density and temperature.

The French[11] calculate the Lawson criterion for photon energy flux for ignition and find

$$\phi = 5 \times 10^{14} \text{ W/cm}^2 \ (T = 10^8 {}^\circ K), \tau = 10^{-6} \qquad \text{Nd}$$

$$\phi = 5 \times 10^{12} \text{ W/cm}^2 \ (T = 10^8 {}^\circ K), \tau = 10^{-4} \qquad CO_2$$

They have used fluxes within an order of magnitude of this to produce electron temperatures in excess of 1 keV and 10^4 neutrons/shot. This was with a Nd glass laser of 100 J and 3 nsec focused on a solid target.

The Germans[12] have used a single pulse from a mode locked Nd laser in the 0.1 to 10 joule range at 5 psec and have measured temperatures up to a few keV.

Relativistic electron beams can be used to do similar things but, of course, the wavelength of the electrons is much different. Electron beams have considerable advantages over laser beams because they can deliver over two orders of magnitude more energy than Q switched lasers and the efficiency of production of beams approaches 35% compared to 0.1% for nsec lasers, according to Rudakov.[13] On the other hand, lasers can be more easily focused and the pulse duration reduced to psec. Rudakov proposes to use heavy metals surrounding the D-T pellet to reduce rapid expansion. In doing this he calculates that the critical energy can be reduced to 10^5 J and the critical power to 10^{14} W. He suggests this reduction because of the high penetrating power of relativistic electrons and magnetic isolation of the plasma due to the beams magnetic field. Self-focusing may be possible by using the dissipation of reverse current in the plasma. He proposes beams of 1-10 MeV at 10^7-10^8 A.

Rudakov also gives the linear stopping distance for electrons as[13]

$$\lambda_1 = \lambda_c (\varepsilon)\bar{\theta}^2 \frac{\varepsilon}{mc^2} \left(\frac{T_e}{mc^2}\right)^2$$

where $\lambda_c(\varepsilon)$ is the Coulomb path length of a relativistic electron of energy ε, $\bar{\theta}$ is the mean scattering angle and T_e is the electron temperature.

Whether energy fluxes of the required magnitude can be developed remains to be seen. If they are developed they may not be economic and yet for start up sources they may provide the extremely large energies that are required.

GUN HEATING

Plasmas produced by a fast theta pinch, conical Z-pinch, or coaxial gun may also be considered as heating sources for start up or injection. While not much has been done to scale up guns to match the size of proposed experiments, it seems possible that this area could be expanded considerably.

A problem arises when one tries to make pulsed plasma guns capable of generating sufficient, high temperature plasmas necessary for ignition and still be able to trap this plasma in a stable configuration. In the LRL BB experiment several guns are used to generate fairly large quantities of plasma; however, for closed line devices, the plasma must polarize to cross the confining field or else it must enter by separating field lines because of high β. For etiher case one wonders how the plasma knows to stop at the correct radial position for trapping. It may be possible to use sources spread around the devices and have trapping equivalent to that which takes place in present experiments.[14] On the other hand, conversion to neutrals and collisional trapping as with energetic beams is possible but presents problems because of the large angular dispersion.

This form of ignition is probably only possible when no background cold plasma exists within the trapping zone to establish field lines as equipotentials. This is a problem which would eliminate their consideration for tokamaks.

The only reason these devices are better for injectors than neutral beams is that the method of plasma acceleration is through macroscopic plasma forces and not through electric field acceleration. This means that plasma currents of from thousands to hundreds of thousands of times greater than that available in beams are possible but probably not in a steady state manner. The disadvantage is in the lower energy per particle that.is found with gun plasmas. For any high flux pulsed gun source, it is difficult to get energies above the kilovolt range.

NEUTRAL BEAM HEATING

This method of plasma heating looks especially attractive because one can separately control the particle energy, composition and flux from the outside which makes it also useful for reactor control.

Because single charges cannot be injected and trapped in a magnetic field without a collision (Liouville's Theorem), particles entering a plasma immersed in a magnetic field can only be trapped if they undergo a change in e/m while in the confining region. Usually this is accomplished by collisional ionization of neutral atomic beams at energies up to about 100 keV, this is reasonably straightforward but above this energy, the atomic cross sections for almost all gases suggest that the incoming ion beams are more likely to emerge from the gas cells as ions rather than neutrals. If the ion beams are molecular ions and in particular, negative ions, the equilibrium fractions converted to the neutral state can be significantly greater at the higher energies than with protons.

Ion, or neutral beams to be more exact, can carry large amounts of energy into a plasma region because, within the limitations posed by the previous problem, they can have almost any desired energy. Because the power input goes as the energy times the beam current, it is highly desirable to inject large amounts of current. To increase the beam current one runs into the problem of space charge limiting. However, extraction from large plasma surfaces will make current modules of 10 amperes possible.

The comment on arbitrary energy of injection needs a bit further explanation because the problem is the same one found in the thermalization of reaction α particles. That is, if the charged particles (trapped, injected ions or α particles) have an energy which is too large compared to plasma ions, then the ions are heated mostly through collisions with plasma electrons which are directly heated by the beam. That is, most of the fast particle energy goes to electrons. One very simple way to see this is by the interaction cross section for Coulomb collisions which goes as $\sigma \sim 1/(m_r v_r^2)^2$. This says electrons and α's have the lowest relative velocity and reduced mass, whereas plasma ions have a velocity more nearly equal to that of medium energy beam ions. Such approximations merely say the collision frequency between these particles is greatest. The energy transfer per collision is also important and this is mass dependent in favor of ion-ion collisions.

Independent of the details of the collision, thermalization of beam particles in a plasma can produce substantial heat input in closed systems and, of course, in open systems. Whether the thermalization will be fast enough to prevent a destructive perturbation on the distribution function remains to be seen. Certainly the beam should be less of a perturbation than the reaction products.

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HEATING METHOD	SYSTEM		
	Low β Open	Low β Closed	High β
OHMIC	?	Yes	Yes
TURBULENT	Yes	Yes	Yes
R.F.	Yes	Yes	?
ADIABATIC COMPRESSION	Yes	Yes	Yes
SHOCK	?	?	Yes
LASER & RELATIVISTIC BEAM	Yes	Yes	Yes
PLASMA GUN	Yes	?	?
NEUTRAL BEAM	Yes	Yes	?
OTHER	?	?	?

Fig. 1

SUMMARY OF SESSION

PLASMA HEATING AND IGNITION

H. K. Forsen
University of Wisconsin

In the heating and start-up session there were 13 contributions. Before reviewing them I would like to make some comments about the sessions in general. It seems that there are at least two points of view represented in the conference and perhaps they reflect the extremes of the spectrum of those attending these sessions. One point of view seems to be that an individual's own problem is so difficult that he cannot imagine anybody working on any other problem. The other view is that another's individual's problem looks so difficult that one really ought to spend full time on it. I am sure that there are many tough problems or we would already have controlled fusion but thus far no one problem appears insurmountable. This kind of conference, where plasma physicists and reactor engineers get together, can contribute significantly to the understanding of how various problems interrelate and I think that this will be very important in the future.

In the ignition part of this session, Sweetman of Culham gave the only talk directly related to what is required for ignition. In his analysis he took a steady state reactor at a power balance taking into account heating from fusion reactions and loss processes which are due to the energy transport and radiation. For a reactor producing about 30 watts/cm^3 of fusion power, two cases of energy loss rates were considered. One is the very pessimistic Bohm confinement scaling which suggests that significant external heating is required. In this worse case, .25 watt/cm^3 is required to reach ignition and it is achieved at about 9 kV. In the case of classical confinement, which is the most optimistic case, ignition appears to require only 0.1 watt/cm^3 and occurs at about 4 kV. Without preempting anything that will be said later, one can say that energy systems which are presently available can deliver these kinds of powers to present sized experiments. It turns out that the problem is one of coupling the energy to the plasma.

Sweetman also discussed the problem of getting energetic particles across the diameter of a hot plasma if one does beam heating.

It appears that if you try to distribute the energy or the particles across the diameter of a plasma operating at temperatures of between 10 and 20 kV in a toroidal system very energetic particles are required. Generating these energetic particles may be a problem, and we will return to this later.

Figure 1 was made up for the three major confinement approaches with the heating methods that appear applicable. During the session it was hoped that some certainty could be given to the "yes's" and the question marks clarified. As it turned out this was possible only to a limited extent because direct theoretical or experimental evidence was not available for all cases.

If we follow Fig. 1 across as applied to the various systems, it can be used as an outline of what was discussed by the participants. Taking ohmic heating first, we heard from Mills of Princeton and Strelkov of Kurchatov on problems relating to tokamaks or low-β closed systems.

Both made calculations on the energy loss time, including the radiation processes from plasma, to try and determine whether ohmic heating could carry these systems to the ignition point. According to Mills one might just be able to get there because the diffusional losses decrease as you go up in temperature. Strelkov also pointed this out, but he suggested that one needs very high currents, very large magnetic fields and reduced synchrotron radiation, at least for the kinds of tokamaks and tokamak scaling which are now envisioned. Strelkov's calculations included four models of the energy confinement time scaling and he considered two cases. One had no additional external heating beyond ohmic and the other case included some form of additional external heating. Like Mills, he found that without additional heating, ignition looks marginal. In Strelkov's model, if one adds external heating, such as from beams, rf or adiabatic compression, you find that all the difficulties which came about before are reduced. In the Princeton experiments, Mills suggests that by slightly compressing the plasma in one of two ways, ignition looks promising. The two methods include increasing the vertical stabilization field which moves the plasma in towards the major axis or by increasing the poloidal field to compress the plasma in the minor diameter. Both methods are to be tried in the not too distant future.

Moving now to turbulent heating we must remember that this category includes almost anything that is nonclassical. To be sure, the anomalous resistivity observed in tokamaks may be due to weak turbulence and as such this topic was not specifically covered. Other nonlinear heating schemes are separated out as special topics and will be discussed later.

Roth of NASA described a method of heating where energy is primarily coupled to the ions. It can be thought of as ohmic or turbulent or perhaps even rf heating. In his experiments he applied an electric field perpendicular to the confining magnetic induction as in a Penning discharge. He reported measurements on a mirror system where the ion temperature showed a linear relationship to the applied voltage up to the limits of the power supply. Problems of low density due to plasma loss out the ends may be overcome in a bumpy torus system, now under construction, which will employ the same heating scheme.

In the area of rf heating there were talks by Kristiansen of Texas Tech, and Sprott of ORNL. Kristiansen discussed problems in cooling and insulating support and coupling structures for any rf system. Because of the size of reasonable feed lines and the need to handle large powers, something like 2000 watts/cm^2 is the required heat conduction. High voltage insulation in the presence of large magnetic fields makes the arcing problem especially difficult. When the effects of neutrons on insulating materials are added, the difficulties become compounded.

Sprott discussed problems in electron cyclotron heating but pointed out two areas where advantages occur. One is to use ECRH to provide a hot electron target plasma for neutral beam trapping and the other is in a high-β bumpy torus. While the rf energy is predominantly coupled to perpendicular motion of electrons, it is still possible to get around any instability problems by heating at an upper off resonance. This heating process is not so well understood as the resonance heating but progress is being made in this area.

Shock heating as applied to high-β theta pinches was discussed by Ribe of LASL. His real concern was on the magnetic energy required to shock heat a reactor system. If one balances joule losses and field energy, then a larger plasma radius is highly desirable. Superconductivity magnets can be useful to provide both the slow bias field and perhaps the energy storage for the fast imploding magnetic sheath. However, high voltage Blumleins are actually envisioned at this time and this technology needs to be developed for shock heating. Ribe combined several equations relating the dynamics, applied voltage, compression and the like to predict the final plasma temperature and it is clear that a smaller compression ratio is helpful. It would appear that this approach has considerable merit for high-β systems.

Haught of United Aircraft described work on laser produced plasmas and it was generally concluded that because of the low density of conventional fusion plasmas, steady state laser heating is not practical. On the other hand, lasers may be attractive for reactor ignition where a target plasma is created from a solid pellet which could produce fusion or where it is used simply as a target to collisionally trap a neutral beam. Theory and experiment are starting to come closer together in terms of scaling laws to produce either the highest temperature plasma or the most ionized particles as a function of target size.

Problems of coupling the energy from photons to plasma particles are reduced as the square of the wavelength of the photons. Therefore, electron beams may have something to offer over laser beams and this was the position of Rebut of CEA in France. He suggested accelerating counter streaming electrons down the axis of a linear theta pinch device of about 5×10^{16} cm^{-3} density. Heating comes about through Coulomb interactions and his calculations suggest that about 18 MW/m length per cm^2 cross section is required to carry the system to reactor conditions.

This could be provided by beams of around 200 kA at 400 kV energy. Since the efficiency of production of these beams can be considerably above N_2CO_2 lasers, not to mention Nd, perhaps there is some merit in the approach.

Before moving on to systems where the energy is produced in particles outside the container, Husseiny of Wisconsin had some interesting suggestions about thermalization rates of beams and reaction produced ions. That is, the slowing down and thermalization of a fast neutral or electron beam, or reaction particles in a background plasma may not have been computed correctly in the past. He claims that by considering the $\ell n\Lambda$ term to be energy independent in relaxation calculations, one gets answers which are too optimistic. Similarly, neglected quantum mechanical effects in the slowing down range can lead to results which suggest test particles slow down faster than, in fact, they may.

By far the largest interest in the session was in energetic neutral beams. Morgan of ORNL, Sweetman and Thompson of Culham and Post of LRL reported on progress in the development of high current, medium energy neutral beams. At the first conference of this kind held in Culham in 1969, investigators were talking about beam currents of 10's to 100's of milliamps. Here everyone was working on ampere beam systems with 10 amps being the not-too-distant design goal. This has been possible by the expansion of extraction surfaces and the development of larger plasma sources. In order to maintain the plasma boundary layer as the first electrode, a structure has been used over the extraction aperture. By collimating the open areas in this aperture with the accel and decel electrodes, the beam quality has been maintained. Morgan pointed out that by using these techniques 1 amp/cm^2 looks possible but structure cooling will be required. He further suggested that as one gets to energies above 100 kV at these currents, the hard practical problems of beam handling,pegging and breakdown start to become severe.

Thompson reported scaling laws for these large ampere systems and showed curves on the results of their work that indicate post acceleration of lower energy systems (40-50 kV) may be the optimum way to go. Systems like this would use many smaller extractors operating at 10-20 ma/cm^2 current density. Problems of optimizing the production of H_2^+ or H_3^+ for conversion to neutrals in a gas cell were also discussed.

Post reviewed the work of the Livermore and Berkeley groups in the beam area. They are particularly interested in mirrors and direct conversion so their beam energies are optimized for these systems. He recognized and reported on the difficulties of converting ion beams to neutral beams at energies above 100 kV as did both Morgan and Thompson. Because of this problem, negative ions are the best ions to work with and they hope to undertake studies on this soon. Post acceleration of high current negative ions is a problem that has not been attacked and the practical aspects of it look formidable. The Berkeley group has been able to use computer codes to help optimize their extraction apertures and this may lead to even better multi-apertured systems and may help in the post acceleration problem. Right now, however, the ability to produce large surface area, uniform density plasmas is marginal.

Essentially nothing was described in the way of plasma guns that would lead one to believe that they would be useful in either ignition or heating. We do know, however, that interest in this area still exists for some systems; the problems outlined in the initial discussion of this session seem extremely difficult to overcome.

Before concluding this summary I would like to especially thank Dr. Larry D. Stewart of ORNL for helping take notes on what was a rather informative session.

DISCUSSION

H. POSTMA (ORNL): It seems that a number of the experiments described were really patterned toward a next generation of experiments rather than toward reactor problems. Would you care to go through the list that you just showed and eliminate those methods that really are not pertinent to reactor consideration. This could be either because of electrode destruction through radiation damage or melting due to the high flux of particles or because as in ohmic heating where things just run out at a kilovolt or so. Such a discussion would give some indication of what methods are realistic for reactor considerations?

FORSEN: Yes, I will do that but as soon as I say this doesn't look good someone will challenge me, but let us try it anyway. If we look at Fig. 1 we can start down the list. Ohmic heating varies as the resistivity of the plasma and we know this varies as $1/T^{3/2}$. As one goes to high temperatures in any system the heating rate decreases. If we take toroidal low beta systems, we know that you can compensate for this decrease by raising the current because the heating rate also goes like the current squared. In general, I think it was felt by both people who talked on this subject that ohmic heating is not going to be enough and yet by carrying ohmic heating as far as one can, it really says that this reduces the burden on neutral beam heating that you will have to use if that is the way you choose to go. Just following low beta closed systems down the list, one has to conclude ohmic heating is an important consideration. Whether the anomalous resistance that is so helpful and that is seen now will be important when we really get into reactor regimes I think is questionable.

In high β systems ohmic heating is not such an important effect. The ions are preferentially heated in these systems by shock heating. The electrons are heated ohmically by the current in the boundary layer but it's the ions we need to get hot. Turbulent heating, of course did not come out at this conference, but crudely all that turbulence means is the lack of understanding of the classical ohmic heating. We know how to generate these anomalies, that is by running the current density somewhat higher than given by the acoustic speed of the particles or even at the thermal speed as has been seen in rf systems. I feel that when we get to reactor systems, turbulence will still be there but whether it will be anomalous to the point that carries one very far is questionalbe. Therefore, I don't look at this as a vital heating mechanism in low beta systems. Again, in high beta systems turbulence takes place when the shock speed exceeds the Alfven Mach number. Exceeding this is highly desired and generally can be achieved in these systems. Effects which take place in the boundary layer or from the boundary layer will be important in preferential ion heating.

In terms of rf heating, I am somewhat pessimistic about trying to couple sufficient energy into a plasma to carry it to ignition. The problem again is that it is easiest to couple the energy to the electrons and as soon as one gets to the densities that are needed in thermonuclear reactors, the coupling efficiency drops significantly. We do know that the waves penetrate at least a skin depth into the plasma and you might end up heating the surface--by electron cyclotron heating at least. If this is where you want to heat then you can but generally that is like the fueling problem where the high density ends up on the boundary and I think it is an undesirable situation. If you can do some of the things which are being tried with ion cyclotron heating, or off-resonance heating then the real problem is with the rf structure. We know this kind of ion cyclotron heating works for the stellarator using a Stix coil but a reactor is different. It depends upon whether the waves can penetrate into the plasma and whether you are going to allow yourself to have a magnetic beach to thermalize the energy. Tokamak systems and other low beta toroidal systems require beaches or some region where the wave energy is coupled to the ions and this may not be easy.

Adiabatic compression on the other hand, always works if you can handle the plasma and program the fields. Whether one is going to be able to change the magnetic fields in reactor systems with superconducting coils is still a question. The high beta people suggest getting around this problem by returning the flux inside the superconducting coils. I think this is the way to go but it may be very expensive.

Shock heating is very important for high beta systems. I do not see it doing anything for the other two systems and as far as we could tell from this conference, we would have to put noes in there.

For lasers and relativistic beams there appeared no one here except Rebut who believed either mechanism was effective as a heating system. I think probably one would have to say no and yet if one is looking for a very energetic system to light off your reactor and were able to get to the extremely high power lasers, or maybe even relativistic beams and couple those beams into the magnetic fields, then these might be important for startup sources. That is, to start up the reactor and now hope that you can confine whatever density that you have such that it carries on. Additional fuel and energy can be injected by lower energy, high density neutral beams.

In terms of plasma guns, I am afraid that there is very little hope for them unless we go to pulsed systems. Here again we have the problem of the magnet and how to rapidly change a superconducting field. Neutral beams look like the one area that is rather independent of the system that we talk about. Of course, it is not sufficient in high beta pulsed systems because we can't get the currents in the time scales that we are talking about, to be effective. Certainly anything which is steady state and even slow, the kinds of neutral beams that appear possible now are going to be very important. That's a long answer, but perhaps your question calls for even more.

R. ROTH (NASA Lewis): I want to set the record straight on the Penning discharge that I reported. The best Lawson parameters that I reported in the talk were an ion kinetic temperature of 5 kV, a density of a few time 10^{10}, and a confinement time of about between 20 and 30 microseconds with an electron energy of about 200 eV. Other parameters that you mentioned were measured under different conditions and not simultaneously in this experiment. Under those Lawson parameters we are getting steady state neutron production at a rate of about 10 microwatts.

P. HUBERT (CEA, France): I should like to comment about Penning or PIG discharge heating in toroidal systems. It seems that in general PIG discharges are accompanied by lots of instabilities so they do not give much hope to attain something like ignition. However, I should like to suggest work in very low density PIG discharges initiated at very low pressures. Some years ago I succeeded in achieving PIG discharges at pressure as low as 10^{-10} Torr, and it seems to me that doing this in toroidal geometry would be a very interesting way to do plasma physics or diffusion studies in the collisionless regime.

R. MILLS (PPPL): Strelkov was unable to be here this morning because he is on his way home, but I think I am correct in saying that we are in agreement on the question of ohmic heating with no anomalous resistance and it does not look quite as bad as the impression that Forsen may have given. If we readily accept the idea of ~ 1 second confinement time in the reactors that we have been talking about, and if we really believe that we are dealing with a diffusion process then we get about 10 seconds of confinement time for something that is about three times larger in radius. If you can confine a plasma for the order of 10 seconds, ohmic heating with no anomalous resistance is sufficient to get up to the ignition condition. If you cannot do that, you can also do it by dropping your initial density for a while because then you require less energy and the temperature comes up faster. After you achieve initial ignition conditions, one could then allow gas to go in and build up the operating densities to the desired point. If you are a little bit short by the ohmic heating or if you are still in trouble, then just a little bit of adiabatic compression goes a long way for the last bit. So both Strelkov and I are somewhat optimistic on the possibility of ohmic heating ignition. Perhaps with only a little bit of assistance from the other methods.

FORSEN: I would agree with you in that Strelkov indicates that he has not even thought about the other kinds of heating mechanisms that might be required. However, I would disagree with you in that I do not think you can start at low densities and then build it up by injecting gas because I think you end up putting plasma on the wall when you do it. Charge exchange is going to be a very difficult problem.

M. GOTTLIEB (PPPL): I, too, am not very optimistic about the methods of rf heating but they really have not been tested adequately. When you look at such schemes as adiabatic compression you find that you can almost make it sufficiently hot from ohmic heating alone. Using a small adiabatic compression carries you to ignition conditions so it looks as though that we are not really looking for a very large heating factor and rf might actually provide that, although there are numerous difficulties. Two of the particular methods that look attractive at the moment or at least look possible are lower hybrid heating, which does seem to penertrate and does seem to couple well. The other possibility a small probability at the present time, is that of ICRH. There are several problems: there is a coupling problem to the plasma and of course, there is the problem of how the wave does indeed dissipate its energy. There is a beach automatically in a tokamak just because of the fact that it is a fat device and magnetic field changes vary appreciably over the volume of the system, but there is a serious question now as to whether the energy will actually go to ions. It looks as though most of it will go into electrons according to theoretical estimates that have been made. And, of course, as you mentioned the launching mechanisms are very difficult to protect when you consider the plasma bombards these wave launching mechanisms. They have to be very simple schemes, essentially they have to be holes in the wall in order to get this rf in, but as I say, we only need a small amount and it is quite possible that this might do it. We just do not know enough at the present time.

FORSEN: Let me ask Dr. Gottlieb and Dr. Mills a question. Do you think we will be able to do ohmic heating ignition on small scale devices or will we have to go to some other system of heating? Ohmic heating depends on long confinement time and we cannot get that in small systems.

GOTTLIEB: What we need to do at the present time is to find out enough about the heating mechanisms so that we can predict how large a system will operate. Ignition does require a large system, ignition that is if you are going to both ignite and meet the Lawson criterion--in other words not have it go out right away.

J. CLARKE (ORNL): I would like to second Dr. Gottlieb's remarks. I tend to agree with your assessment of the various ignition techniques with regard to their applicability, but I would like to just state that there are other uses for some of these techniques other than ignition. For instance, you may know that there is a thing called a thermal instability, calculated by Furth at PPPL, which causes the heating due to the ohmic current in a tokamak under some conditions to concentrate in a region of high temperature and cause a thermal runaway phenomena. There is also the problem of cooling at the surface of a plasma due to wall interactions which tends to push the plasma in the direction of this thermal instability. A technique like rf heating, as you pointed out, would tend to interact more with the surface of the plasma rather than the interior and therefore may have a potential use in counteracting this type of potential problem. I am sure there are other examples with the other techniques, so that even though we can not see how each particular one of these techniques would contribute to reactor ignition, they are all important subjects for research.

FORSEN: I think that's a good point. For anybody to say at this time what will or will not work is like saying that fusion will or will not work, we do not know yet.

HEATING METHOD	SYSTEM		
	Low β Open	Low β Closed	High β
OHMIC	?	Yes	Yes
TURBULENT	Yes	Yes	Yes
R.F.	Yes	Yes	?
ADIABATIC COMPRESSION	Yes	Yes	Yes
SHOCK	?	?	Yes
LASER & RELATIVISTIC BEAM	Yes	Yes	Yes
PLASMA GUN	Yes	?	?
NEUTRAL BEAM	Yes	Yes	?
OTHER	?	?	?

Fig. 1

SESSION 3

NEUTRONICS

Chairman

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STATE-OF-THE-ART PRESENTATION

THE ROLE OF NEUTRONICS CALCULATIONS IN FUSION REACTOR TECHNOLOGY

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1. INTRODUCTION

Assuming the use of a D-T fuel cycle (which produces 14 MeV neutrons) in fusion reactors then the first function of neutronics is to show that tritium can be bred adequately in any proposed blanket model. This is done by calculating neutron spectra in the various regions of the blanket and integrating these with tritium producing cross sections to work out reaction rates. The basic nuclear data used must be carefully scrutinized so that a realistic error may be assigned to calculated reaction rates.

The next important function of neutronics calculations is to determine heating rates, both through the breeding blanket and in the very cold region of the superconducting coil. These heating rates are normally found in two stages. Firstly, the "local" energy deposition via charged particle production and recoil nuclei is estimated. Secondly, the amount of energy released as γ-rays must be determined, and the way this energy is transported through the blanket and deposited in it, be calculated.

Because of the presence of highly energetic fluxes of neutrons in the blanket, both radiation damage and induced activity will occur in materials. In particular the middle-weight nuclides used for structure will be at risk. It is an important secondary function of neutronics work to show from reaction rates what the effect of the following phenomena will be: helium gas bubble formation from (n α) reactions; atomic displacement rates; transmutation of materials; induced activity; and the afterheat associated with induced activity.

The prime objective of this paper is to present a summary of neutronics work in fusion reactor technology which may serve as a useful base-point for newcomers to the field. There will be numerous references to original work, though the author is not so sanguine as to imagine that all important contributions will be note. For the sake of succinctness, elaboration of points relevant to the fields of engineering, economics, metallurgy, etc. will be kept to a minimum.

2. FUSION REACTOR DESIGN

The reactor designs which have been produced so far fall into about six groups, each based upon a different thermonuclear plasma confinement concept. A useful source of information which includes reports on mirror systems, theta pinches, tokamaks and stellarators, is the book of the Proceedings of the Conference on Nuclear Fusion Reactors held at Culham, U. K. in September, 1969. Another fairly recent publication due to Carruthers et al[1] makes an economic survey of a conceptual toroidal system and identifies many technological problems requiring investigation.

There are a number of features common to many current reactor systems, and for our present purpose we can identify them by referring to Fig. 1, due to Homeyer.[2] There is an inner core of confined plasma at a temperature of about 10 keV, fuelled by a 50:50 mixture of deuterium and tritium. This is contained in a vacuum vessel made of some refractory material with a dimension of 3 - 4 meter diameter. The fusion reaction is:

$$D + T \rightarrow (\alpha + 3.52 \text{ MeV}) + (n + 14.06 \text{ MeV}) \qquad (1)$$

and thus some 80% of the reaction energy is carried away by the 14 MeV neutron. The two main functions of a blanket region surrounding the central plasma column are to allow the kinetic energy of the neutron to be absorbed at heat, and also to breed tritium for further fusion reactions. In order to do this the blanket must be about 1 meter thick. A superconducting coil placed round the outside of the blanket will provide the large magnetic field ($\sim$ 100 kG) required to contain the plasma. Energy deposited in this coil from neutron and γ-radiation can represent a substantial power loss because of the difficulty of removing heat from a source operating at liquid helium temperature, 4.2°K. A shielding region of thickness between 0.5 and 1.0 meter is required.

3. CHOICE OF MATERIALS

Homeyer[2] discusses selection of materials in Chapter III of his thesis. He considers various coolant fluids, vacuum wall and other structural materials, neutron moderators and multipliers, all with regard to their nuclear, thermal, cost, compatibility, and general physical characteristics.

Adequate tritium breeding must be achieved in the blanket, and therefore the presence of copious amounts of lithium in some form or other is mandatory. Tritium is bred via the two reactions:

$$^{6}Li + n \rightarrow \alpha + t + 4.8\ \text{MeV (at low energies)} \qquad (2)$$

$$^{7}Li + n \rightarrow n' + \alpha + t \ -2.47\ \text{MeV (at high energies)} \qquad (3)$$

Natural lithium contains 7.42 at % ^{6}Li and 92.58 at % ^{7}Li. The two most suitable chemical forms appear to be lithium metal (M.P. = 180°C) and the fused salt 66% LiF + 34% BeF_2 known as Flibe. The melting point of Flibe is rather high at 455°C. The advantages of lithium over Flibe include better themal properites, lower melting point, and better breeding characteristics (Steiner,[3] and Blow et al[4]). The overriding drawback with liquid lithium is the electromagnetic pumping loss caused by its high electrical conductivity in the presence of a large magnetic field. Both lithium and Flibe appear to be compatible with molybdenum and niobium, even at temperatures approaching 1000°C.

The choice of structural materials is still a very open question since the blanket may operate anywhere in the regime 500 - 1200°C. At the upper end of the scale strength considerations alone would limit the choice to the refractories niobium and molybdenum with melting points at 2470°C and 2610°C. Tungsten is not considered since its nuclear properties are less favorable, and it is more brittle and costly than either of the other two. Vanadium (M.P. = 1890°C) is another possibility because of its induced activity properties (Postma[5] and see Section 11). The lower end of the scale would allow more traditional materials such as steels and nickel based alloys (M.P. $\sim$ 1400°C). A particular alloy known as INOR-8 (16% Mo, 7% Cr, 5% Fe, balance Ni) was developed for containing the fuel salt in the ORNL molten salt breeder reactor project, the fuel salt being Flibe with 1% molar UF_4. Nickel based alloys and liquid lithium are not compatible because of the high solubility of nickel in lithium.

Graphite is an attractive material because of its low neutron absorption, and its ability to withstand high temperatures. It is resistant to corrosion by fused fluorides, but carbon does dissolve in liquid lithium and appropriate cladding would have to be used in a graphite-lithium system.

γ

4. NEUTRONICS CODES

The actual calculation techniques employed fall into two camps, viz. transport methods and Monte Carlo techniques. Both are discussed in Greenspan et al.[6]

Transport codes start from the fundamental Boltzmann transport equation describing neutron conservation at a generalized point in phase space. A series of difference equations appropriate to neutrons in different energy bands is set up and solved by iterative methods. The numercial techniques, often based on physical intuition, which are necessary to ensure fairly rapid convergence of solutions have been developed over a number of years. They are primarily associated with the name of B. G. Carlson of LASL (Carlson[7]). A program in common use is the ORNL code ANISN (Engle[8])

The Monte Carlo method is well described as a "computer experiment" in that it consists of tracking individual neturons until they are absorbed or escape. Each time a decision has to be made, e.g., what the neutron energy will be after a collision, data tables are consulted to see what the possibilities are, and a random number generated to pick one of these possibilities. After a pre-selected number of neutrons has been tracked, the track lengths in a region are added up and divided by the region volume to give a value for the neutron flux. A variance estimate is made to derive the error incurred in tracking a limited number of neutrons. A Monte Carlo code in common use in the U.K. is SPECIFIC II (Holbrough and Lipscombe[9]).

5. TRITIUM BREEDING

Breeding blankets have been studied since fusion reactors were first mooted, and a study of requirements complemented with a short critical review of suggested systems was made by Barton and Strehlow.[10] Some subsequent studies (Impink,[11] Bell[12,13,14])looked at breeding in Flibe and lithium blankets, possibly moderated with graphite, with beryllium for neutron multiplication and molybdenum or copper between source and blanket. Lontai[15] studied the enhancement in breeding and power production on introducing fissile nuclides in the system.

A fusion feasibility study at ORNL (Steiner[3]) led to:

(i) A preference for niobium instead of molybdenum on the basis of superior welding characteristics,

(ii) the rejection of beryllium on the grounds of its high cost.

A simple model based on Steiner's ideas is shown in Fig. 2. In Table 1 we show what happens to neutrons in this blanket in terms of various reaction rates (Blow et al[4]).

Table 1

REACTION RATES IN A FUSION REACTOR BLANKET
(normalized to one 14 MeV source neutron)

Reaction	(n,2n)	Parasitic Absorption	Escaped	T_6	T_7	T
Reaction Rate	0.11	0.20	0.04	0.88	0.52	1.40

T_6 = Tritium production via ^{6}Li (n,t) reaction.

T_7 = Tritium production via ^{7}Li (n, tn') reaction.

$T = T_6 + T_7$

The total breeding figure of 1.4 offers a comfortable excess, but it must be remembered that this result is for a representative and somewhat idealized system. Allowing 10-15% for access regions suggests that a realistic minimum figure to aim for is 1.15.

Lee[16] has looked at tritium breeding in spherical geometry. He deduced that for pure lithium the breeding ratio was 2.0, and it dropped to 1.0 for lithium diluted with 20 % niobium for structure.

Most of the calculations so far mentioned have not made allowance for resonance self-shielding or thermalization effects. Resonance self-shielding in niobium has been shown by Bell et al[14] and by Steiner[17] to increase the tritium breeding by about 5%.

6. NUCLEAR DATA

All calculations are limited by the accuracy of the raw data they use. In attempting to analyze fusion systems we use data inherited from work on fission systems, and this is indadequate in two main respects:

(i) in providing accurate non-elastic cross-sections in the range 1 - 14 MeV (e.g., (n,2n) in niobium and molybdenum),

(ii) in providing γ-ray spectra from non-elastic events (mainly (n,n'), (n,γ), and (n,2n)).

A survey has been made of materials typically found in a fusion reactor blanket to see if their cross-sections were known accurately enough to allow a tritium breeding ratio to be calculated to 1% (Crocker et al[18]). Cross-sections which were not accurate enough for breeding or other calculations are shown in Table 2. The distribution of experimental points around the preferred curve for the important ^{7}Li (n,tn') breeding reaction is shown in Fig. 3, the data being taken from Pendlebury.[19]

7. NORMALIZATION OF FLUX VALUES

So far we have been content to calculate reaction rate ratios for blanket assemblies in terms of one source 14 MeV neutron. But to work out heating rates, induced activities etc., we need to associate neutron flux values with the reactor power rating. Homeyer[2] made such a correlation, and he deduced that the maximum power rating would be 5 MW m^{-2} of vacuum wall, this limit being determined by the heat removal capability of the Flibe coolant. Both Carruthers et al[1] and Rose[20] have made more recent estimates and conclude that a heat load in the first wall of 9 MW m^{-2} of 14 MeV neutrons could be tolerated. It is to be noted that this figure implies a wall loading from <u>all</u> energy sources of 13 MW m^{-2}, since the total energy release per fusion event is reckoned to be 21 MeV (see Carruthers et al[1]).

A wall rating of 9 MW m^{-2} is equivalent to 4.0 x 10^{18} n m^{-2} sec^{-1} of 14.06 MeV neutrons. Thus the source intensity of neutrons is determined and absolute values of the flux through the blanket may be deduced. For the particular model shown in Fig. 2 the flux in the first wall region is calculated to be 2.8 x 10^{15} n cm^{-2} sec^{-1}. This is very similar to the core

TABLE 2

NUCLEAR REACTIONS HAVING INADEQUATE DATA FOR NEUTRONICS, HEATING, DAMAGE, AND ACTIVITY CALCULATIONS

Reaction	Sphere of Interest	State of Data Accuracy
1. ^{7}Li (n,tn')	(i) Tritium Breeding	15% Accurate
2. Nb (n,2n)	(i) Tritium Breeding (ii) Recoil Heating (iii) Displacement Damage	Nb-was 450 mb at 14 MeV, now from 1100-1500 mb
3. Mo (n,2n)	(iv) Transmutation	Mo-no published measurement
4. F (n,abs)	(i) Neutron Absorption (ii) Gamma-Ray Heating	Several reactions contributing Poor experimental agreement
5. Gamma-Ray Spectra from (n,γ) and (n,n') in Nb	(i) Gamma-Ray Heating	No complete spectral measurement. Poor absolute values
6. Nb (n,γ)	(i) Damage	20% Accurate
7. Excitation of first state in ^{93}Nb	(ii) Radioactivity	Not measured
8. ^{94}Nb (n,γ)	(iii) Radioactivity	Only one measurement 30% accurate?

value of the flux in the Dounreay Fast Reactor (DFR), which is 2.5 x 10^{15}n cm^{-2} sec^{-1}, and is about a third of the value in the core of the Prototype Fast Reactor (PFR) which is 8.5 x 10^{15} n cm^{-2} sec^{-1}.

An integral form of the spectrum in the first wall region is shown in Fig. 4. It is harder than the spectrum in the core of DFR which is shown for comparison. Nevertheless, some 75% of the neutron flux lies below the 14 MeV peak as a result of back scattering in the lithium coolant channels.

8. HEATING RATES IN BREEDER AND SUPERCONDUCTING COIL

Heating rate calculations on standard blanket configurations have been published by Homeyer[2], Steiner[21], and Werner et al.[22] Bell et al[14] give heating rates in the copper coil peculiar to the Los Alamos pulsed θ-pinch reactor design.

Steiner[21] analyzed the configuration of Fig. 2 assuming a 14 MeV neutron wall loading of 10 MW m^{-2}. The heat rating of the first wall is over 100 watts cm^{-3}, and the value drops to $\sim$ 1 watt cm^{-3} at the edge of the 1 meter thick breeder region.

Heating rate values are of prime importance in the thermal design of a blanket system. The calculation is a lengthy one consisting of four distinct steps:

(i) Calculate neutron spectra in the blanket regions.

(ii) Integrate these spectra with local energy deposition cross-sections (also known as "kerma" factors, from kinetic energy released in materials).

(iii) Integrate the neutron spectra with secondary γ-ray production factors to find total energy released as γ-radiation, and the spectral description of this radiation.

(iv) Track the γ-rays to find where their energy is deposited in the blanket.

Nuclear heating calculations are hampered at the moment by a lack of data. A start has been made in accummulating both kerma factors and γ-ray yield data at the Radiation Shielding Information Center at Oak Ridge (see, e.g., Ritts et al[23]).

The interest in heat deposition rate in the cold superconducting coil region arises, as mentioned in Section 2, because of the power loss involved in removing heat from a region at 4.2°K. Homeyer[2] used as operating criterion that the loss from this cause should not be more than 2%, which meant that heat deposited must be less than 2×10^{-5} of the energy recoverable from the blanket. The shield configuration necessary to achieve this was 30 cm of an 80% lead, 20% borated water mix, followed by 20 cm of LiH and 6 cm of lead. The total shield thickness was thus 56 cm.

For a 5000 MW(th) reactor the Homeyer shield model would allow 100 kW to be deposited in the coil. Homeyer[2] suggested that, and Carruthers et al[1] assumed that a further 10 cm of shielding would reduce the heat load to 10 kW, and lead to a negligible contribution of refrigeration equipment to the capital cost of a reactor. Fraas[24] has recently analyzed the situation more closely by using cryogenic data from the large space simulation chamber at the NASA Manned Space Flight Center at Houston. The shield thickness was increased until the increase of material cost equalled the reduction in cost of the cryogenic system. The nuclear heat deposition rate in the coil was calculated to be 1.15 kW for a 5000 MW(th) system. It was estimated that this would take a lead and borated water thickness approaching 120 cm.

Recent calculations by the author would indicate that an iron (as non-magnetic stainless steel) and borated water is much to be preferred to lithium hydride as a moderator and low energy neutron absorber. It is at least as efficient and is cheaper and easier to handle than LiH.

9. RADIATION DAMAGE

Radiation damage effects are to be dealt with elsewhere but there are some points which follow on directly from neutronics work and these will be mentioned under two separate sub-headings.

(a) Helium Bubble Formation

The (n,α) reaction rate for niobium in a fusion reactor first wall is about 2×10^{-4} events per atom per year. Helium atoms formed in the niobium are relatively immobile (diffusion coefficient of He in Nb goes from 10^{-19} to 10^{-14} cm^2 sec^{-1} between 600°C and 1200°C). They will diffuse slowly to form aggregates which subsequently glow into bubbles. It may be noted that after 5 years there will be 10^{-3} atomic fraction of helium in the first wall. What this means in terms of fractional swelling depends on the pressure of gas in the bubbles. If it were atmospheric then 100% swelling would result since gaseous densities are typically 10^{-3} times solid densities.

In fact, as Martin[25] has shown, the swelling may be only 1% even at 1200°C. The point is that the average bubble radius is small ($\sim 10^{-6}$ cm) and the contained gas is at high pressure. However, under non-equilibrium conditions bubbles may migrate, coalesce and enhance the swelling. Martin[25] shows that under realistic temperature gradients at an average irradiation temperature of 1200°C the swelling may rise to the unacceptably high level of 30%.

(b) Displacement Damage

Materials exposed to the very energetic neutrons emitted from thermonuclear reactions will suffer substantial radiation damage from both elastic and non-elastic reactions. The primary recoil atoms slow down to rest by collisions with other atoms in the metal lattice. The energy of the recoil atoms may be dissipated in three ways, either by exciting lattice vibrations, by excitation of electrons, or by knocking other atoms from lattice sites. In this way a cascade of displacements and excitations is produced. For fusion neutron irradiation of niobium the most energetic primary knock-ons will have an energy of $\sim$300 keV, and will produce about 6000 displacements in a highly excited cascade region with a demension of typically 100 Å.

The observation of the formation of voids in materials irradiated in a fast fission flux has highlighted the need for improved radiation damage calculations. The problem is to try and relate observed swelling with calculaged displacement rates, but this is not an easy matter. The initial difficulty of finding the fraction of primary recoil energy which goes in displacing further atoms (and this"damage fraction" is a function of recoil energy) was met by models proposed by Kinchin and Pease[26] and by Lindhard and co-workers.[27] However, these models do not take into account such possibly important factors as channelling of energetic recoils, and mutual recombination within the thermally excited cascade region (Nelson[28]). In addition, the vacancies which are left behind may not all agglomerate to produce voids since many forms of defect sinks exist in a real material. Finally, the process of void formation itself is highly temperature dependent, and also seems to rely on the presence of gas atoms to act as nucleation centers.

Perhaps the most useful thing one can do at this stage is to use the Lindhard formalism and compare displacement rates in the fusion system with those in a fast fission flux. The figures the author has calculated for niobium are 165 displacements per atom per year in the fusion spectrum and 78 in the DFR spectrum (Blow[29]). In PFR, however, the flux is 3-4 times greater (though the spectrum is softer) so the rate for niobium is probably from 180-240 displacements per year, i.e., a somewhat greater rate than in the first wall of a fusion reactor.

10. TRANSMUTATION IN THE FIRST WALL

Figure 5 shows the increase in time of transmuted materials for niobium structure in the first wall region (Blow[30]). The increase is roughly linear and at the end of twenty years about 23% of the original ^{93}Nb content has been changed, 13.5% to zirconium and 9.5% to molybdenum. This calculation is based on a 14 MeV (n,2n) cross-section of 1000 mb. If the true value is around 1500 mb, as recent measurements suggest, then the zirconium fraction would be about 20%. This transmutation rate is formidable, and of itself must produce radical changes in the mechanical properties of the niobium.

The problem is not nearly so severe in PE16 which for the purpose of this calculation is taken to consist of 43% nickel, 39% iron, and 18% chromium. PE16 is an alloy originally developed for use in jet engines in the range 550-700°C. It is of interest to the fast reactor program because of its resistance to void swelling. The total transmutation rate in PE16 is less than 1% after 20 years. There are two reasons for this. Firstly, the total transmuting reaction rate is only 0.3 of that in niobium. Second, and more importantly, there is a total of 13 stable nuclides in nickel, iron and chromium, and many transmutations take place between stable nuclides.

The transmutation in molybdenum is 8% after 20 years. Natural molybdenum consists of seven stable isotopes, ^{92}Mo, ^{94}Mo, ^{95}Mo, ^{96}Mo, ^{97}Mo, ^{98}Mo, and ^{100}Mo. Though reaction rates in molybdenum are very similar to those in niobium it is basically the two "end" isotopes, ^{92}Mo and ^{100}Mo, which are at risk from the dominant (n,γ) and (n,2n) reactions. Hence the smaller transmutation rate.

11. INDUCED ACTIVITY IN THE FIRST WALL

In calculating activities it is necessary to consider only those activated nuclides which have a half-life greater than a few hours, the reason being that after the reactor is shut down it will take a day or two before it is accessed for maintenance or repairs.

Figure 6 (from Blow[30]) shows the activity of niobium in the first wall region in units of curies per watt of thermal power. The activity rises rapidly at first owing to the creation of ^{92m}Nb with a half-life of 10 days. The activity remains approximately constant between 8 and 20 years at about 0.68 curies per watt. For a 5000 MW(th) reactor the first wall activity would therefore be 3.4×10^9 curies. When the reactor is switched off, the activity level will be controlled for the first 50 days by the ^{95}Nb (half-life 35 days). Thereafter, the activity will decay with the half-life of ^{93m}Nb at 13.7 years.

Steiner[31] calculated the activity in the total bulk of material to be $\sim$ 4.1 curies $watt^{-1}$ at the end of 20 years. If one deducts the 6.3 min ^{94m}Nb contribution the answer is 2.7 curies $watt^{-1}$ which compares well with the figure of 2.3 deduced by the author (Blow[30]).

The stable activity level of PE16 is about 0.42 curies per watt, some 60% of the niobium first wall activity. The stable level of activity in molybdenum will be about 0.2 curies per watt. However, over 90% of this activity has a half-life of 10 days or less, so the situation is considerably better than in niobium.

Recent calculations by Steiner reported by Postma[5] show that afterheat and activation in vanadium are a factor 10 down on the values for niobium.

12. AFTERHEAT IN THE BULK

For decay power (afterheat) calculations it is necessary to consider all species with a half-life of greater than a second, since it is necessary to know at what rate the coolant must flow immediately the reactor is shut down.

As may be seen in Fig. 7 (at time t = 1 sec) the stable levels of afterheat in a 5000 MW(th) reactor for structure of niobium, molybdenum, and PE16 are 45 MW, 32 MW, and 31 MW respectively. Steiner[31] found a value 48 MW for niobium. Molybdenum is seen to be clearly superior to either of the other two. Within 10 secs its afterheat drops below that of PE 16 and after 2 days the level is down to 5 MW.

Also shown in Fig. 7 for comparison is the decay power from the fuel of a typical fission reactor (Steiner[32]). It may be noted that the molybdenum curve lies below the fission-reactor fuel curve at all points in time.

There is one important point to emphasize with regard to the curves shown in Fig. 7. While the decay power of the fission-reactor fuel is uniquely determined by its thermal rating, this is not the case for the fusion reactor. The values are characteristic of the model depicted in Fig. 2. If 8% or 4% structure by volume were used then the curves would be increased or decreased proportionately.

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31. D. Steiner, "The Neutron-Induced Activity and Decay Power of the Niobium Structure of a D-T Fusion Reactor Blanket", Oak Ridge National Laboratory Report ORNL-TM-3094, 1970.

32. D. Steiner, "A Review of the ORNL Fusion Feasibility Studies", Oak Ridge National Laboratory Report ORNL-TM-3222, 1970.

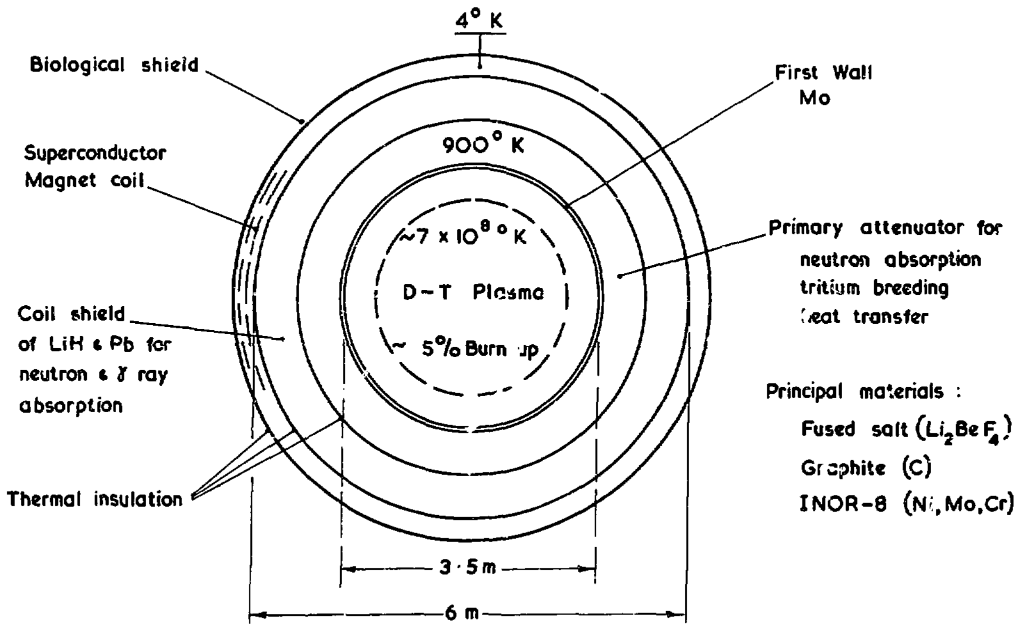

Fig. 1. Conceptual Design Cross-Section of a D-T Reactor Blanket after Homeyer.[2]

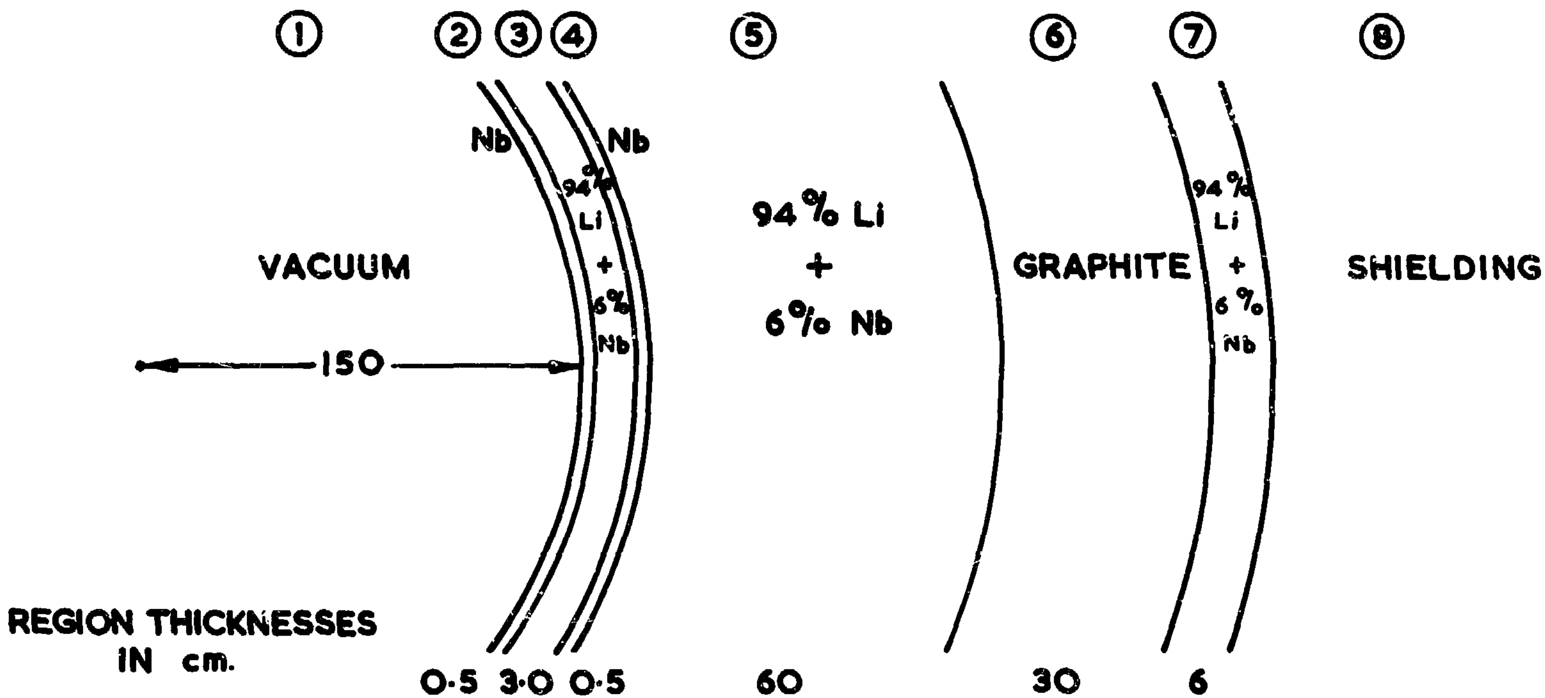

Fig. 2. Typical Blanket Model Showing Dimensions and Material Constituents in the Various Regions.

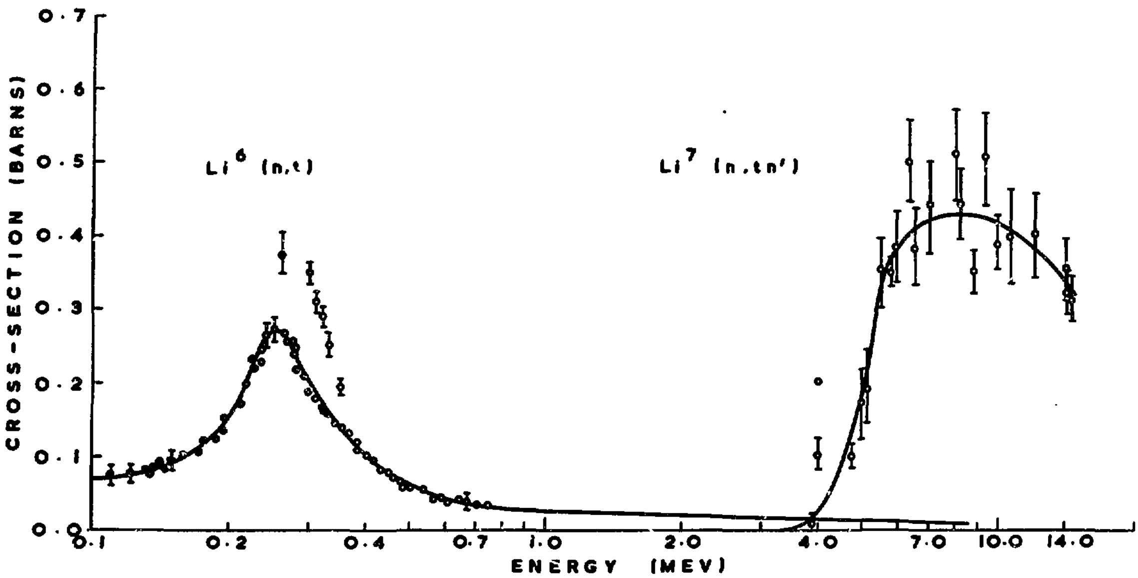

Fig. 3. Tritium Breeding Cross-Sections for Lithium Isotopes in the High Energy Region. The circles represent experimental points from several sources.

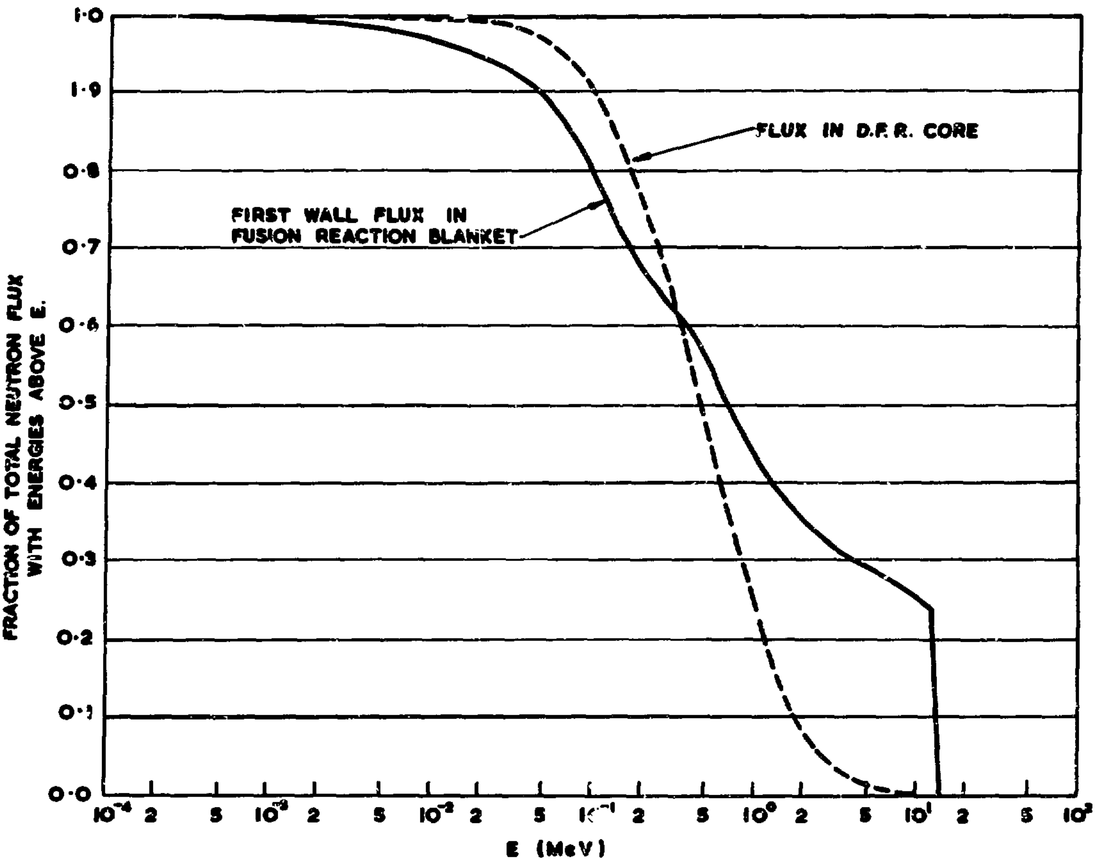

Fig. 4. The Fraction of the Total Neutron Flux with Energies Above E as a Function of E, for:

(a) Fusion Reactor Model First Wall, Total Flux = 2.77×10^{15} n.cm^{-2} sec^{-1}.

(b) Fast Reactor Core, Total Flux = 2.5×10^{15} n.cm^{-2} sec^{-1}.

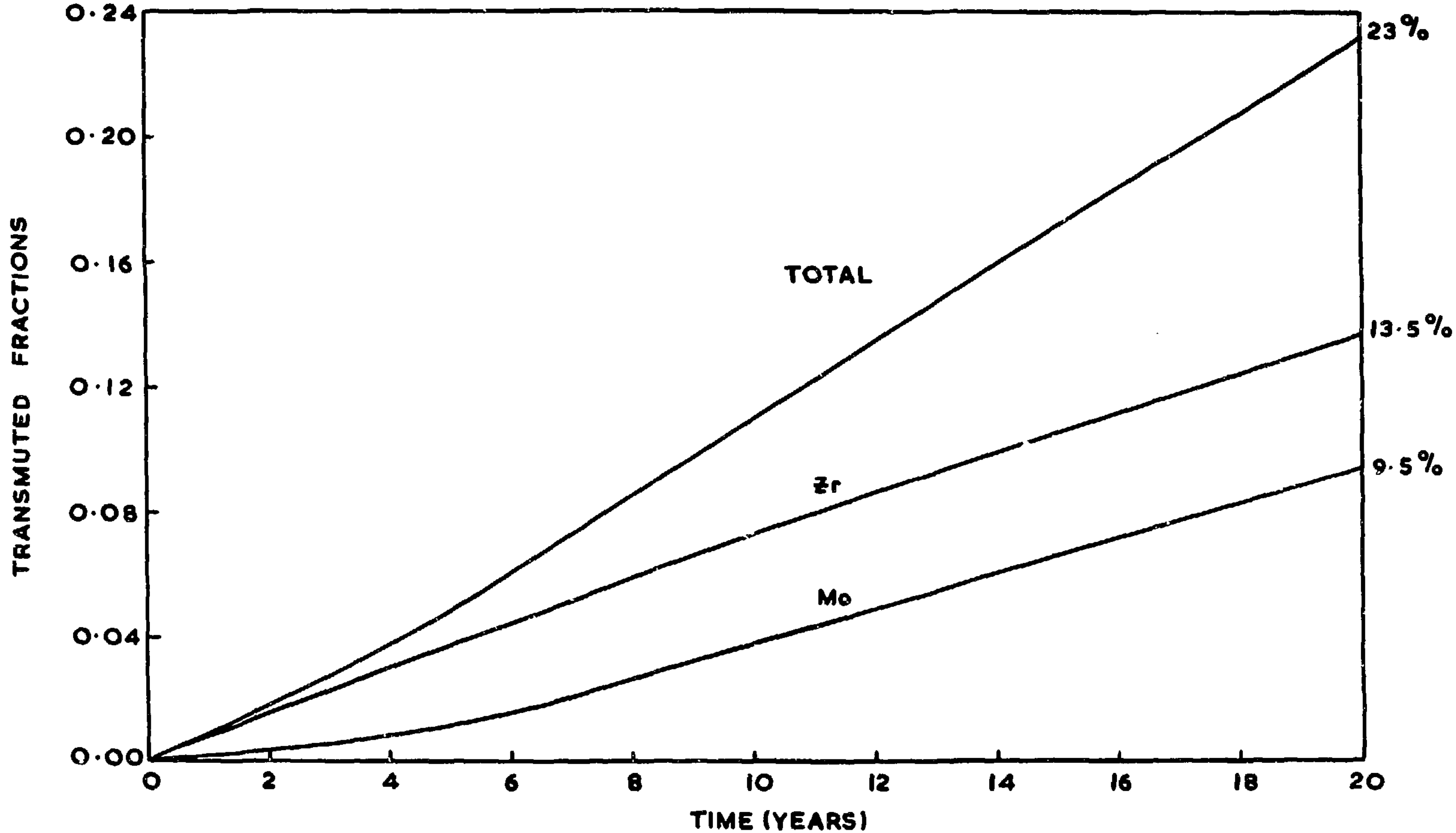

Fig. 5. The Transmuted Fraction of the Niobium First Wall of a D-T Fusion Reactor as a Function of Irradiation Time.

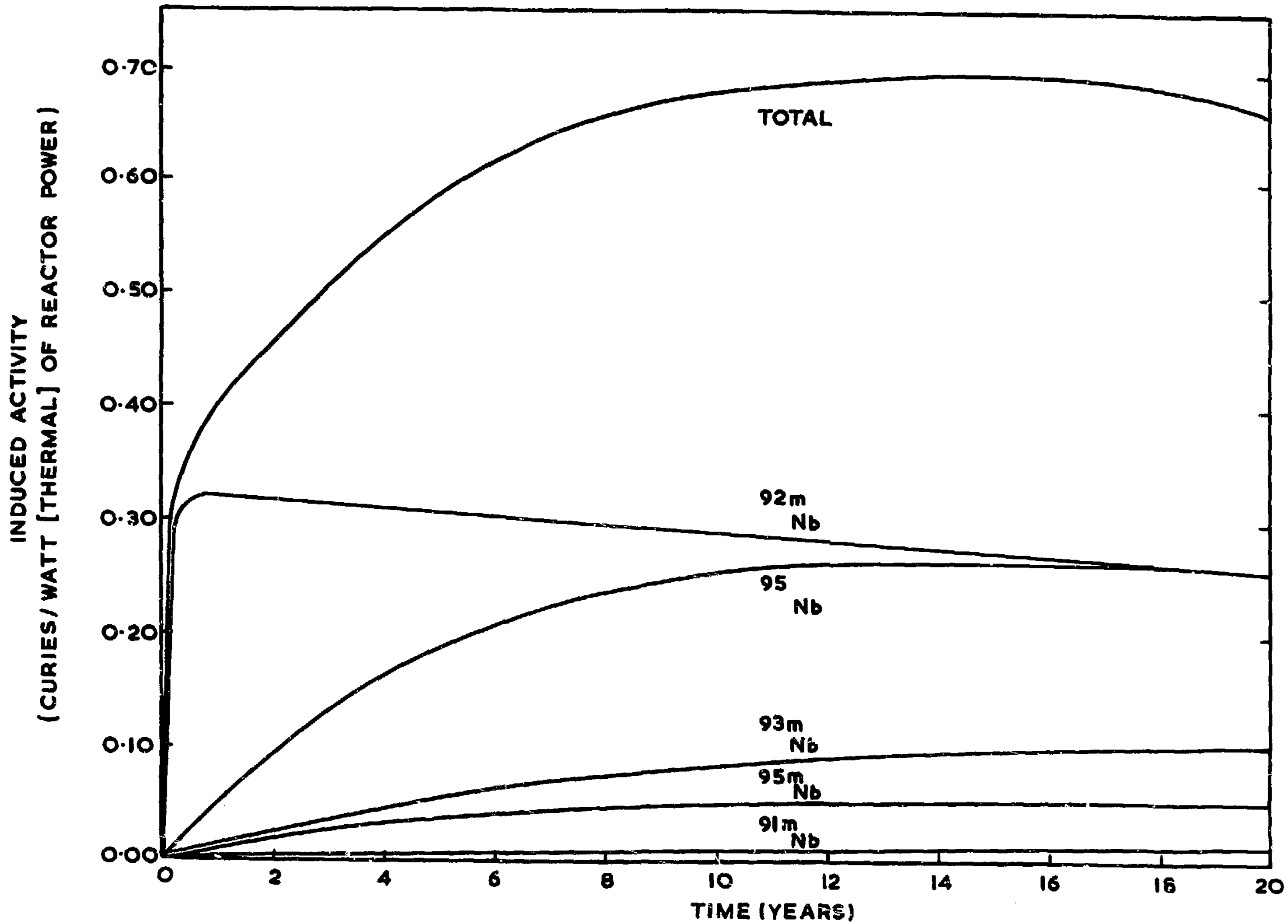

Fig. 6. The Induced Activity of the Niobium First Wall of a D-T Fusion Reactor as a Function of Irradiation Time.

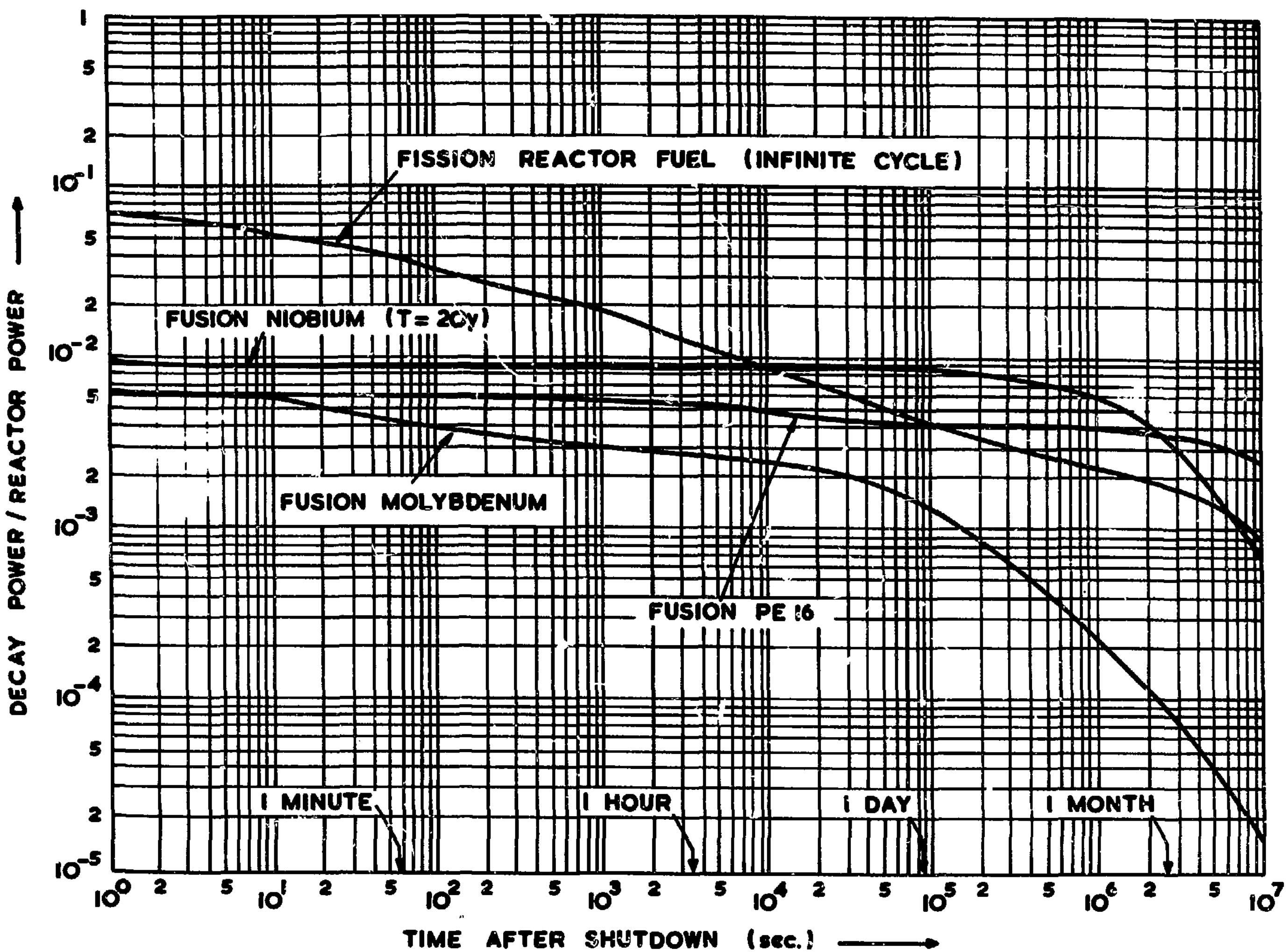

Fig. 7. Decay Power vs Shut-Down Time for a Fission Reactor System, and for Niobium, Molybdenum, and PE16 as Structure in a Fusion Reactor.

SUMMARY OF SESSION

NEUTRONICS

S. Blow

A.E.R.E., Harwell

In a sense, I do not think that I would disagree with Fred Ribe's remark made at the panel discussion last Tuesday evening that "neutronics is easy". Much neutronics work has been done in analyzing fission systems and a great deal of this is easily translatable to fusion systems. The problems that exist consist basically in refining computational techniques and extending cross-section measurements, and these can easily be accomplished within the time scale of CTR work with a modest budgetary expenditure. When I think of the problems of, say, fueling a fusion reactor or evaluating the radiation damage that will occur, then the problems associated with neutronics seem to fade into insignificance. I might add, however, in case it appears that I am doing myself or others out of a job, that of course it will be mandatory for any reactor study group to include a neutronics expert to evaluate the spectrum and flux of neutrons in different blanket regions and to indicate to engineering designers the ramifications thereof.

Now let me try to summarize the presentations that were made at the neutronics session on Tuesday. One of the factors that emerged was that for analysis of particular blanket models Transport and Monte Carlo codes produced values of tritium breeding which differed by about 12%. True, at the moment, one gets values of typically around 1.4 for triton production for one 14 MeV neutron, but these are for idealized systems and when realistic designs are produced which include input and exhaust fuel vents (and drawing on fast breeder experience!) it may well be that tritium breeding becomes marginal. In which case differences of 12% are very significant.

Most of the work presented was theoretical but an interesting experiment on the moderation of a 14 MeV neutron source through graphite was discussed. The commercial tube source produced 10^{11} neutrons per second. Foil activation was used for flux determination and the experimental values were checked with a two dimensional Transport calculation. The code appeared to predict too low a value for the ^{58}Ni (n,2n) reaction rate in the 6-14 MeV range, though the geometry of the experimental layout made analysis difficult. However, I think the result points up again the problem of whether Transport codes can handle anisotropic scattering satisfactorily.

Some excitement was generated in the discussion of fissile blankets and fission-fusion symbiosis. It seems to me that if you consider symbiotic schemes, you must have your objectives quite clear. Would a fission-fusion combination be a natural prototype step towards a solo fusion reactor? Are you going to use it to generate plutonium to cover the critical period when fast breeders are coming on-line? This latter intention is very limited in scope and may indeed not be necessary. Some estimates suggest that there will be enough plutonium generated in thermal reactors to charge the first generation of breeders. Now I think everyone knows the trouble mirror machine people have with their low Q values, and of course, putting fissile nuclides in the blanket can increase the energy release per fusion neutron by a factor up to 10. However, a whole host of problems arise such as what you do with the plutonium you inevitably produce and surely you have to worry about the neutron kinetics too. One scheme was proposed using the T-^{7}Li reaction, obviously applicable only to mirror machines, which produces 2 neutrons and 2 alphas. An attempt was made to burn up the plutonium produced in uranium near the wall in an outside region where the spectrum has thermalized, but it proves difficult to balance the reaction rates.

Considering fuel cycles other than D-T, a presentation pointed out that neutron activation is not very different for D-D and D-D-^{3}He cycles allowing for the difference in power densities. So do not think you can forget about neutrons by going to alternate cycles.

We have heard quite a deal about vanadium as a possible structural material, both here and at Madison, Wisconsin. The induced activity and afterheat of vanadium are much lower than those of niobium and also lower, though perhaps more comparable, with molybdenum (this is the activity and afterheat immediately on shutdown of the system). The great advantage of vanadium appears to be that only short-lived radioactive species are excited and that induced activity integrated over a thousand years is a factor of 1000 less in vanadium than in niobium.

At the concluding summary part of the working session, a number of suggestions for further work was made, but essentially they boil down to two proposals. Firstly, that an ad hoc group be set up which should determine an agreed standard blanket model and do a calculation using the same nuclear data. The method of calculation is to be the only variable. Such a group has met and agreed upon a standard blanket for calculation. (A description of the standard model and calculation is attached to this summary.) May I encourage other groups to agree on standardization of such things as say, the superconducting magnet costs, so that meaningful comparisons can be made between the different reactor systems? Secondly, another ad hoc group met with Sol Pearlstein (BNL) yesterday morning and presented him, eventually, with a list of cross-section measurement requirements in the CTR field. (The list of cross-section requirements is attached to this summary.) This list will be distributed among interested parties to survey the present state of affairs and to report back to Don Steiner any additional requirements so that a final specimen may be evolved and forwarded to appropriate groups involved in nuclear data measurement.

DISCUSSION

D. STEINER (ORNL): You mentioned that the difference between the Transport and the Monte Carlo calculations was about 12%. Could you indicate which is the higher and which the lower--this might be of interest.

S. BLOW: The figure quoted was the tritium production rate. For neutrons of 14 MeV you get a great deal of forward peaking in elastic scattering and it may be that Transport codes cannot handle this too well whereas Monte Carlo codes can. This is one of the things we hope to learn from our analysis of the standard blanket model. The present situation is that the Transport calculation yields the low value and the Monte Carlo calculation yields the high value, which suggests that the 14 MeV peak may persist deeper into the blanket than the Transport code would indicate.

F. CULLER (ORNL): Since you gave me the privilege at the beginning of the meeting to deliver observations that may be restraints on how you approach fusion from lessons learned in fission, one remark of yours reminds me to repeat something that I said earlier. You said that it would perhaps be desirable or maybe necessary to reach a consensus on the superconducting magnet costs. The bases for these remarks, I am sure, have been explored in some detail in the meetings, but there are dangers, I think, in optimizing on estimated costs where manufacturing techniques have notbeen developed and I caution you against being overly sensitive to estimated costs where a manufacturing technology does not exist. Such optimizations led us in the fission business, at the laboratory stage for instance, to predict with confidence that we could produce 1.5 mill electricity. Now that may be possible at some point in the future. The truth is, the situation as it develops is very different from what you assume in early optimizations, and it would seem to me to be a prudent course to design a system that works to some efficiency without being too constrained by economic optimizations based on estimates of costs where technology does not yet exist, and I offer that with due reservations concerning my own lack of knowledge of the system.

G. MILEY (U. Ill.): Was the problem of shielding the magnet discussed? How do the codes and calculations compare on the leakage flux into the magnet?

BLOW: In fact, we did not cover this topic too well. I think a very good evaluation has been done by Don Steiner and Art Fraas as part of the Oak Ridge analysis of the superconducting magnet system for a Tokamak type reactor. The sort of thicknesses of shield that were being proposed a few years ago were, I suppose, in the region of 50 cm. Steiner's analysis seems to indicate that, in order to get the nuclear heat deposition in the helium region down to about 1 kW for a 5000 MW thermal reactor, you need to have a thickness of about 100 cm of lead and borated water. I might just add a personal note here, that we are thinking of having iron in the shielding region for structural purposes, as John Mitchell has talked about and has been mentioned earlier today. Don Steiner is analyzing this system for us, and it may turn out that iron makes a better shielding material than lead, but this is still rather preliminary

H. POSTMA (ORNL): I see a coming back to almost a full cycle. As I recall Lyman Spitzer in an original proposal for stellarators and fusion reactors considered the primary reason was to be able to make material for fission reactors because of the great neutron capabilities of fusion. I see some discussion has evolved around that and I gather there was a thesis at MIT and some work by Lidsky. Now the problem is, and I understand there was some discussion of your group about this question, that you are taking a system which may be good and making it somewhat worse by the use of neutrons in making fissionable material and therefore radioactive waste products. Would you relate some of the conversations in your session because I think that it is fairly important to recognize what the orders of magnitude involved in these kinds of things really are before one proceeds to such an extent that you are making a bad name for fusion or fission.

BLOW: Well, I do not think I would personally like to add much to the remarks I have made, except that you must decide where you are going and why. You are inevitably going to breed plutonium if you utilize the fast fission in uranium for bumping up the Q value, so you have then to decide what you are going to do with that plutonium. Is it sensible to say you can have fast breeders systems running alongside fusion reactors and that you are going to fuel them with this plutonium? Then you have to look at the balance of reaction rates and so on. I would be happy for people to stand up and present their views as they were made during the session.

W. KÖHLER (Texas A & M): I made the point in the discussion that the development of a possible fusion reactor depends strongly on how the fast breeder reactor will develop, which is something that we will know in about 10 or 15 years. If it turns out that the fast breeder reactor does not work, which would mainly be because of safety reasons as it looks now, then one has the question of what one does with the energy reserves that we have in our uranium resources. I think in that case the question of burning the uranium in fusion reactors would really become an important question and a very good possibility because one has to consider that the technologies associated with this resource could not be efficiently used if the fast breeder reactor does not work. If the fast breeder reactor does not work, a situation which would have a tremendous impact on the development of the fusion reactor, one should probably look at fusion-fission symbiosis is order to efficiently use our uranium resources.

J. LEE (LRL): Just one quick comment in response to Postma. I think that we might get in trouble by saying that we might dirty up our systems or make them less attractive by including fission because that automatically assumes that fission is dirty and unattractive and I do not think we should say that. The idea of fission-fusion symbiosis or direct fission in the blanket has obvious advantages if you relate low Q values as being limits to D-T systems, and I think they are worth looking at and ought to be evaluated on the same grounds as any other system. They might lose out on the cost of making them safe, but let us look at them and see.

W. WOLKENHAUER (Battelle Northwest): There are a couple of points I would like to make. First, with regard to our slowing down experiment, we tried to correlate the experimental data both with the two-dimensional and a one-dimensional Transport code, using two different geometries with the one-dimensional code. The one point you brought out on underestimating the (n,2n) reaction is significant and, perhaps, just as significant is that the calculations seem to overpredict the thermal reaction rates by about a factor of 3 or so, depending upon which of the calculations you believe. Dr. Leonard is not here so I would rather not comment on our fission-fusion ideas. I think they have been covered by other people.

A. GIBSON (Culham): I would like to express my dismay at the apparent tendency of blankets to grow in thickness by 50 centimeters per conference, and ask what freedom for maneuver there is for reducing the thickness of the blanket?

BLOW: Very little, I am afraid. When we say that the blanket is growing by 50 cm, this refers just to the shielding region and obviously that is important for superconducting winding costs. It is just very hard to stop 14 MeV neutrons. Even if you were not worried about tritium breeding, but just wanted to shield from the vacuum wall against some kind of heat load or radiation damage in the coil, you still could not do it in much less distance. That is not strictly true, you could probably shave it down to about 75% of original thickness, but not much more. 14 MeV neutrons are very difficult to handle, they are very penetrating.

J. LEE (LRL): I personally think that the question of total blanket thickness will be more sensitive to considerations other than the question of tritium breeding. You mentioned that tritium breeding could be more marginal. I think that differences in cross sections and differences in codes will be much more important in the evaluation of overall blanket thickness than in the evaluation of tritium breeding alone.

STANDARD MODEL FOR COMPARISON OF NEUTRONICS CODES

At the Neutronics Session of the week of Working Sessions on Fusion Reactor Technology held at Oak Ridge National Laboratory in June 1971, some reported results of reaction rate calculations were in conflict.

It was suggested that an agreed standard blanket model be analyzed by different groups so that the only variable is the method of calculation.

A. Picture of the Model

The agreed standard blanket model is shown below

1. Geometry

Distances in cm	0 – 150	150 – 200	200 – 200.5	200.5 – 203.5	203.5 – 204	204 –		– 264	264 – 294	294 – 300
Origin →	Plasma	Vacuum	Nb	94 % Li 6% Nb	Nb	20 cm	94% Li 6% Nb 20 cm	20 cm	C	94 % Li 6% Nb
Zone Number	1	2	3	4	5		6		7	8
Region Number	1	2	3	4	5	6	7	8	9	10
Material	A	B	C	D	C		D		E	D
Number of Intervals Per Zone	1	1	3	6	3		30		15	3
Thickness (cm)	150	50	0.5	3	0.5		60		30	6

<u>Comment</u>: The intervals in each zone are of equal step length. There are 62 intervals all together.

2. Atomic Densities

The following number densities will be used for the different materials in the calculation:

Material Code Letter	Constituent	Number Density
A	Isotropic flux source of neutrons	
B	Vacuum	
C	Niobium	0.05556×10^{24}/cc
	Niobium	0.003334×10^{24}/cc
D	Lithium-6	0.003234×10^{24}/cc
	Lithium-7	0.04038×10^{24}/cc
E	Carbon	0.0804×10^{24}/cc

3. Nuclear Data

The ENDF/B and UK files for Nb, ^{6}Li, and ^{7}Li are the same. It is not clear how UK and ENDF/B files correlate on carbon. Use the latest versions available.

B. Assumptions of the Calculation

1) Use 1-D cylindrical geometry.

2) Use an isotropic source of 14 MeV neutrons distributed throughout a central cylinder of 150 cm radius.

3) Normalize on 10 MW m^{-2} of 14.06 MeV neutrons on the first wall $\equiv 4.43 \times 10^{14}$ n cm^{-2} sec^{-1}.

4) Those employing a transport code should use the S_4, P_3 approximation.

5) Neutron energy group structure will be left to the individual's discretion.

C. Action

1) Calculate absolute neutron flux values at the center of each of each of the 62 intervals. Present the results in graphical form. Monte Carlo workers must include statistical error bar.

2) Calculate Nb (n, 2n) reaction rates (normalized to one 14 MeV neutron) in each of the appropriate 45 intervals. Present graphs (M-C workers with error bar).

3) Do ^{7}Li (n,tn) as for (2) in each of 39 intervals.

4) Present tables for total Nb (n,2n) reaction rates and rates in regions 3, 4, 5, 6, 7, 8, and 10.

5) Present tables for total ^{7}Li (n,tn) reaction rate and rates in regions 4, 6, 7, 8, and 10.

6) Send graphs and tables to the following two addresses by September 1, 1971:

Dr. S. Blow
Materials Physics Division
Building 521.1
AERE, Harwell
Didcot, Berks., United Kingdom

Dr. D. Steiner
Reactor Division
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, Tennessee 37830 USA

Stephen Blow
Oak Ridge, Tennessee
Wednesday, June 30, 1971

CROSS SECTION REQUIREMENTS FOR FUSION REACTORS

At the International Working Sessions on Fusion Reactor Technology held at Oak Ridge National Laboratory June 28-July 2, 1971, an attempt was made to list the cross section needs of fusion reactor programs and indicate where data deficiencies currently exist. The list of data needs probably does not include data requirements of all fusion reactor design studies but is representative of the materials and kinds of information needed for the major fusion configurations. For each material and reaction considered the energy range, percentage accuracy desired, and justification for the data requirement are described. In some cases the status of current experimental information is indicated although this information admittedly may be incomplete or out of date and will be reevaluated by an ad hoc group subsequent to the meeting. As technology develops an attempt will be made to add or delete data requirements as found to be appropriate.

Sol Pearlstein
Oak Ridge, Tennessee
Thursday, July 1, 1971

CROSS SECTION REQUIREMENTS FOR FUSION REACTORS

Material	Reaction	Energy Range	Accuracy Desired	Present Status and Justification
^{7}Li	(n,n't)	3 - 14 MeV	< 10% at 14 MeV	Present measurements are accurate to about 25% at 8 MeV and 15 % at 14 MeV, leading to an uncertainty of about 0.1 in the breeding ratio. Better than 10% accuracy at 14 MeV is needed to determine more accurate breeding ratios for the varied engineering designs.
	$\frac{d\sigma}{d\Omega}$ (n,n)	$\sim$ 14 MeV	15%	Needed for reasonable shield and magnet cost estimates.
	$\frac{d\sigma}{d\Omega}$ (n,n't)	$\sim$ 14 MeV	15%	Needed for reasonable shield and magnet cost estimates.
	Secondary energy spectra for (n,n'x) reactions	thresh-14 MeV	15%	Energy spectra needed to evaluate effect of neutron capture competition between ^{6}Li and resonance absorbers in the calculation of the tritium breeding ratio. It is also needed to evaluate the transport of neutrons through the blanket and shields.
	(n,t)	similar comments apply as to ^{7}Li		
	$\frac{d\sigma}{d\Omega}$ (n,n)			

Material	Reaction	Energy Range	Accuracy Desired	Present Status and Justification
Nb	(n,2n)	14 MeV	10%	Experimental measurements for the Nb(n,2n) cross section range from 450 mb (activation) to 1400 mb (neutron production). Supporting measurements of 10% accuracy are needed to reduce uncertainty in neutron multiplication estimates to reasonable limits. Cross sections for the (n,p) and (n,α) reactions are needed for radiation damage estimates. Since appreciable transmutation effects occur, the (n,γ) cross section is needed to calculate formation of higher mass isotopes. The energy and angular dependence of the (n,2n) and (n,n'γ) secondary neutrons is needed to calculate transport of neutrons in the blanket and shield. Gamma-ray spectra measurements are needed to calculate heat generation in the blanket and shield. Spectral data exist only for a few incident energies and a few materials.
	(n,p)	14 MeV	20%	
	(n,α)	14 MeV	20%	
	(n,γ)	14 MeV	20%	
	(n,n'γ) energy spectra	thresh-14 MeV	15%	
	(n,γ) energy spectra	thermal-resonance range	15%	
	$\frac{d^2\sigma}{d\Omega dE}$ (n,2n)			
Mo, Al, V, Cr, Fe, Ni, Cu, Zr	Same reactions, energy range, and accuracy as above. Similar status and justification apply			New measurements of the (n,2n) neutron production cross section are in progress. Supporting measurements are needed to reduce uncertainty in neutronics calculations. Some (n,2n) activation cross sections are available for separated Mo isotopes.
C	(n,n'3α)	14 MeV		
F	(n,abs)	thermal to 14 MeV	10%	Poor experimental agreement.
	(n,n')	10 - 14 MeV	10%	Data needed for applications using LiF.

SESSION 4

SURFACE PHENOMENA

Chairman

M. Kaminsky
Argonne National Laboratory
Argonne, Illinois 60439

STATE-OF-THE-ART
PRESENTATION

SURFACE PHENOMENA LEADING TO PLASMA CONTAMINATION AND VACUUM WALL EROSION IN FUSION REACTORS AND DEVICES*

M. Kaminsky

Argonne National Laboratory, Argonne, Illinois 60439

1. INTRODUCTION

During the operation of a thermonuclear fusion reactor, such major components as container wall, blanket, shields, divertor walls, and beam dump of injector region will be exposed to the primary plasma radiations and/or to secondary radiations generated by the primary radiation-e.g., secondary radiation caused by (n,γ), $(n,1p)$, $(n,{}^3He)$, and other nuclear reactions and by the various secondary-particle and photon-emission phenomena. The following discussion will be restricted to certain phenomena that may occur when hot surfaces, both those of the vacuum wall enclosing the plasma and those of divertors, are struck by energetic particles and photons.

In reports and feasibility studies on fusion reactors,[1-15] it has been pointed out that plasma radiations striking the vacuum walls may (a) seriously damage and erode the wall and (b) release major quantities of gas which will contaminate the plasma and thereby cool it below the minimum temperature for thermonuclear reaction. For example, during the operation of large steady-state D-T fusion reactors, their vacuum walls are expected to reach 600-1000°C and simultaneously will be bombarded with high fluxes of energetic particles (e.g., MeV neutrons, neutral atoms formed by such processes as charge exchange near the plasma boundary, and ions leaking out of the confining fields) and energetic photons (e.g., bremsstrahlung, synchroton radiation, x rays, and γ rays). These primary radiations from the plasma affect the vacuum wall directly and also by causing secondary irradiation by energetic particles or γ rays from nuclear reactions, energetic displaced lattice particles, and the like. For example, the impact of 14-MeV neutrons on walls made of ${}^{93}Nb$ causes numerous nuclear reactions which result in energetic protons, alpha particles, etc.

*Work performed under the auspices of the U. S. Atomic Energy Commission.

The energetic particles and photons from primary and secondary radiations cause a variety of physical and chemical processes in the walls. These processes include physical and chemical sputtering, secondary-electron emission, x-ray emission, backscattering of particles and photons, release of absorbed and adsorbed gases, radiation blistering, radiation damage, photodecomposition of surface compounds, particle entrapment, re-emission of trapped particles, and the like.

The high temperatures of the walls not only may enhance the yields of some of these processes (including gas permeation and diffusion) but in addition may cause vaporization of the target material and of the imbedded impurities, thermal desorption, thermal emission of electrons and ions, and whisker growth. Some of these phenomena have been reviewed recently[16-19] and are shown schematically in Fig. 1.

The following sections will discuss plasma contamination and wall erosion by some of these particle- and photon-induced processes and by thermal mechanisms, with emphasis on processes that are common to various types of fusion reactors and fuels and on those that have received too little attention as potential sources of trouble in the operation of large fusion reactors. The former class of processes includes sputtering by MeV neutrons, gas release and surface-layer scaling under impact of energetic photons, and radiation blistering by impact of neutral particles, while the less frequently considered sources of trouble include whisker growth, thermal desorption, and gas permeation.

Plasma contaminants have an important effect on the reactor power losses due to bremsstrahlung. For a hydrogen-isotope plasma ($Z_1 = 1$), the ratio R of the power losses with and without the contaminant in a fusion reactor is $R = 1 + f(Z_2 + Z_2^2) + f^2Z_2^3$, where f is the fractional concentration of the impurity and Z_2 is its atomic number. One notices that R increases rapidly with Z_2. On the assumption that the efficient operation of a fusion reactor limits the tolerable increase in the power loss due to bremsstrahlung to 10% (R = 1.1), this expression can be solved for an upper limit on f. For fully ionized impurity atoms, some such limits are $f_{max} = 4.9 \times 10^{-3}$ for Be, 1.8×10^{-4} for V, 5.8×10^{-5}

for Nb, and 5.5 x 10^{-5} for Mo. Figure 2 is a plot of f_{max} vs Z_2. For any impurity whose atoms are not fully stripped, the values of f_{max} must be kept significantly smaller than those indicated in this plot.

It has been suggested that the structural integrity of the vacuum wall would be compromised if more than 20% of its thickness were lost. If a wall thickness of 1 cm is assumed and a wall lifetime of 20 years is considered as desirable, then the maximum permissible annual thickness loss is $\Delta\ell_{tot}$ = 0.1 mm = 100 µm. Therefore it is necessary that the sum over the individual thickness losses $\Delta\ell_\nu$ for the individual erosion processes (e.g., physical sputtering by energetic neutrons, by neutral atoms, and by ions, chemical sputtering, and photodecomposition), obeys the inequality

$$\Delta\ell_{tot} > \sum_\nu \Delta\ell_\nu = \sum_\nu \sum_\mu \left((\phi_\mu S_{\mu\nu} t)/N_\lambda\right) \Delta\lambda \qquad (1)$$

where ϕ_μ is the flux (projectile cm^{-2} sec^{-1}) of the particle species μ interacting with the wall, $S_{\mu\nu}$ is the yield (atoms/projectile) for a particular removal process by particle species μ (e.g., the sputtering yield caused by 14-MeV neutrons), t is the number of seconds in a year, $\Delta\lambda$ is the thickness (cm) of a monolayer, and N_λ is the particle density (atom cm^{-2}) of one monolayer. Therefore in order to estimate the total thickness loss one has to form the sum over the various erosion processes ν, and the sum over μ accumulates the individual contributions of the particle species or photons incident on the surface. In the sum over μ, particles with the same Z and A but different energy are treated as different particles since the $S_{\mu\nu}$ are energy dependent; i.e., this sum is over the yields from 14-MeV neutrons, 3.3-MeV alpha particles, 10-keV deuterium atoms, 20-keV deuterium atoms, etc.

The plasma contamination and wall erosion induced by certain phenomena which are caused by the impact of particles and photons and by thermal effects will next be treated individually in more detail.

2. PHENOMENA INDUCED BY PARTICLE IMPACT

A. Survey of the Effects Produced

When energetic neutral or charged particles impinge on and penetrate through solid targets, the resulting primary processes include momentum transfer between the projectile and the target atom, changes in the internal energy states of the projectile and/or the target atom, nuclear reactions, etc. Such primary processes in turn can cause displacement of lattice atoms, lattice atom excitation and ionization, x-ray emission, and other secondary processes. The primary and secondary processes may cause escape of the primary projectiles-either by backscattering (e.g. see Refs. 20, 21) or by re-emission after entrapment (e.g. see Refs. 22, 23)-and to the emission of secondary particles as an accompaniment to such diverse phenomena as physical and chemical sputtering, radiation blistering, secondary-electron emission, x-ray emission, desorption induced by particle impact, and nuclear reactions. Furthermore the structure, chemical composition, and thickness of the irradiated surface are altered by chemical reactions (e.g., those forming compounds that are volatile or are stable at high temperatures), embrittlement, particle removal by sputtering or evaporation, and the like. Although many of the processes mentioned are discussed in review articles and monographs,[16,17,24] their influence on plasma contamination and on wall erosion is commonly only poorly known for the wall materials, fuels, and operation conditions envisioned for fusion reactors.

The problems resulting from particle impact will be exemplified by a discussion of only two phenomena. The treatment in these two subsections is based on a recent review.[25]

B. Radiation Blistering

It has been observed that H^+, D^+, He^+, A^+, and other energetic particles penetrating through solids can form gas bubbles in the solid. Such bubbles can migrate and eventually burst in or near such energetically favored regions as the surface, grain boundaries, or dislocation lines. The result is to release bursts of gas and to pit the surface. The formation of such bubbles has been observed not only for certain noble gases (e.g., for He^+ on Cu,[26] He^+ on silicon,[27] and A^+ on Cu[28]), but surprisingly also for gases such as deuterium[29] and more recently hydrogen[9] which have markedly higher solubility and larger diffusion coefficients in many solids.

Figure 3 illustrates typical results[29] which we obtained when a (100) plane of a Cu monocrystal was bombarded under ultrahigh-vacuum conditions with 125-keV deuterons at normal incidence for a total charge per unit area of approximately 2.9 C/cm^2. Under these conditions the average number of pits per unit area in the irradiated region was rather large, $N \approx 2 \times 10^5$ pits/cm^2, a number which agreed surprisingly well with the observed number of deuterium gas bursts released during the irradiation. Mass spectrometric observations led to the estimate that on the average each burst released approximately 8×10^9 deuterium particles with thermal and near-thermal energies (as judged from the "higher energy tail"). Each burst also released other gas species, notably those characteristic of the target material. For a current density of 250 μA/cm^2 of 125-keV D^+ ions, the gas pulses from an actual irradiation area* of 0.13 cm^2 occurred at an average rate of approximately 2 per second. This is equivalent to about 15 pulses cm^{-2}s^{-1} or to deuterium emission at an average rate of 1.2×10^{11} particles cm^{-2}s^{-1}.

The pits shown in Fig. 3 are surprisingly large, with pit depths ranging between 0.3 and 0.5 μm and with an estimated average bubble volume to 5×10^{-13} cm^3. Our preliminary work on the bombardment of monocrystals with different orientations indicates that the crystallographic structure of the monocrystalline target influences the shape of the pits significantly. For example, when differently oriented copper crystals were bombarded with 125-keV D^+, the shapes of the pits in the (110), (100), and (111) planes of copper appeared to be respectively more rectangular, more square, and more triangular. These characteristic differences in shape are not artifacts. They have not resulted from the target-preparation procedures (prior to bombardment), since the surfaces were examined directly before and after the bombardment (bottom and top photographs in Fig. 3) without further treatment. Adsorbed oxygen and surface oxides are unlikely to have a serious influence on the pit formation because the continuous sputtering by the incident deuterons removes

*In Ref. 29, this number was misquoted as 3×10^{-2} cm^2.

oxygen and metal oxides faster than fresh oxygen arrives at the surface (oxygen partial pressure $\stackrel{\sim}{\sim} 3 \times 10^{-10}$ Torr).

An increase in the total charge per unit area increased the number of pits formed but did not change the size of the pits appreciably. This suggests an equilibrium size for the pits for a given set of irradiation parameters (e.g., for a given type of projectile, projectile flux and energy, surface energy, target material and structure, target temperature, angle of incidence, etc.). We also observed that the rectangular pits of a Cu(110) plane bombarded at normal incidence turned into more elongated grooves (such as those shown in Fig. 4) as the angle of incidence was changed from normal incidence to more oblique angles; and the deep elongated grooves were oriented along the incoming beam direction.

For the above mentioned special case of 125-keV D^+ bombardment of Cu(100) at normal incidence, the average volume of one pit is about 5×10^{-13} cm^3. From this and the rate of pit formation one can crudely estimate that pitting erodes away target material at an average rate of about 3.2×10^{-7} gC^{-1} cm^{-2}, which corresponds to $\sim 3 \times 10^{15}$ copper atoms per cm^2 per coulomb. This is about 18% of the erosion rate from the sputtering of a Cu(100) surface by 125-keV deuterons, for which the sputtering yield is $S = 2.85 \times 10^{-3}$ atom/ion. In this comparison one should realize, however, that the erosion due to blistering is quite localized, whereas sputtering removes wall material more uniformly from the irradiated area. Therefore averaging over a larger area, as in the above estimates, tends to veil the seriousness of the deep-pit formation by blistering.

While the above observations have not been made for Nb, V, Mo, or other materials suggested for the vacuum walls of a fusion reactor and not for conditions approaching those in an operating reactor, they suggest nonetheless that radiation blistering may seriously weaken the vacuum wall by deep pit formation and that the released gas contaminates the plasma. For the anticipated fluxes of energetic charged particles (e.g., nonconfined ions) and neutrals (e.g., those formed by charge exchange) penetrating the vacuum wall of a thermonuclear reactor, it can be expected that the wall erosion and gas release by blistering will be substantially worse than indicated above. On the other hand, the solubility and diffusivity of hydrogen and deuterium is much larger in Nb (for hydrogen permeability in Nb see Ref. 69) than in Cu. Therefore, especially at the high wall temperatures considered, the formation of hydrogen or deuterium bubbles is likely to be less severe in Nb than in Cu. Some of our preliminary experimental results on radiation blistering of polycrystalline and monocrystalline niobium by 125-keV deuterons for a dose density of 0.1 Coulomb/cm^2 at room temperature indeed suggest, that if bubbles are formed at all, their diameter would have to be smaller than approximately 500 Å (limit of resolution of metallograph used, - electron microscope studies with higher resolution are in progress). However, the formation of helium bubbles (by He or He$^+$ from D-T reactions, for example) in Nb may still be a severe problem. To what degree blistering will be aggravated by the simultaneous impact of high fluxes of different types of energetic particles is completely unknown.

Since the basic mechanisms underlying the blistering process are still only poorly understood,[10,16,17] it is impossible to make a reliable theoretical prediction of the amount of gas released nor to the size, shape, and number of pits formed in Nb, Nb alloys, V, or other materials considered for the vacuum wall under typical reactor operating conditions. Therefore it appears imperative to study radiation blistering for materials that have been considered for fusion-reactor vacuum walls. In particular, such studies should determine its dependence on such important irradiation parameters as the type of projectiles and their energy range, the projectile flux density, the total dose, the target temperature, the target material and structure, and the angle of incidence.

C. Sputtering

The process of target-particle emission under the impact of neutral or charged particles on the surface of a solid target is called sputtering. One often distinguishes between two cases, "physical" and "chemical" (or "reactive") sputtering, although in many experiments both physical and chemical sputtering act simultaneously on the solid surface. Physical sputtering occurs when the kinetic energy of the impinging projectile is high enough to displace target atoms from their sites by momentum transfer in a collision process and some of the displaced target particles move rapidly enough to be ejected into the gas phase. Chemical (or "reactive") sputtering occurs whenever chemically reactive gas particles (e.g., primary beam projectiles or component particles of the residual gas) interact with the surface of the solid and from a volatile compound. The kinetic energy of the interacting particle is of less importance in the latter case than in the former.

The sputtered particles include both "primary knock-ons" (target particles that have been displaced by collisions with energetic primary projectiles) and "secondary knock-ons" (particles in cascades of displaced target atoms ejected by sufficiently energetic "primary knock-ons").

The type of interaction between the collision partners depends on such parameters as the projectile energy and the distance of closest approach. For the range of sputtering experiments performed, a brief discussion of only three types of interaction will suffice.

(1) When an incident particle of energy E collides with a lattice atom, they can interact through the Coulomb repulsion of their nuclear charges (i.e., they undergo Rutherford collisions) if E significantly exceeds the lower limit given[30] by

$$E_B = 4E_R^2 Z_1 Z_2 (Z_1^{2/3} + Z_2^{2/3})^{1/2} M_1/M_2 E_d, \qquad (2)$$

where E_R = 13.68 eV is the Rydberg energy of hydrogen, E_d is the energy (20-30 eV for many metals) to displace one target atom from its lattice site, and Z_1, Z_2 and M_1, M_2 are the atomic numbers and atomic mass numbers of projectile and target atom, respectively. Some representative calculated values of E_B are listed in Table 1. In the energy region $E \gg E_B$ one finds in general that the sputtering yields S decrease with increasing particle energy E.

(2) At intermediate energies, the electron clouds of the colliding atoms partially screen the positive nuclear charges. These screened Coulomb collisions occur in the energy range $E_A < E < E_B$, where the lower limit[31] is

$$E_A = 2E_R Z_1 Z_2 (Z_1^{2/3} + Z_2^{2/3})^{1/2} (M_1 + M_2)/M_2. \tag{3}$$

Some typical calculated values of E_A are also listed in Table 1. In this region one finds in general that the sputtering yields are near their maxima and do not vary drastically with energy.

(3) At low energies ($E \ll E_A$), the electron clouds of the colliding atoms penetrate each other very little, and the collisions are approximately of the hard-sphere type.

No sputtering occurs below a sputtering threshold energy E_{th}. For many projectile-target systems the threshold energy values vary between [illegible] and 40 eV. In the energy range $E_{th} < E < E_A$ one observes in general that the sputtering yield increases with increasing projectile energy E.

For most of the plasma fuel cycles under consideration,[3-6] the particle energies are rather high [e.g. for D, T-fuel cycle it is desirable to have the fuel particle energy $E > 16$ keV, and for the α-particles (as one of the reaction products) $E = 3.5$ MeV]. Thus for typical vacuum-wall materials listed in Table 1, it is apparent that the collisions between primary plasma particles and the lattice atoms in the vacuum wall will be in the Rutherford collision region $E \gg E_B$. In this region, in which the sputtering yield falls as the projectile energy increases, operating the reactor at higher particle energies would tend to reduce the wall erosion by sputtering. It should be noted that our experimental observations[18,19,29,32] with light projectiles (H, D, He) indicate that for the region $E \gg E_B$ the sputtering yield is not affected by the charge state of the projectile (i.e., whether it is neutral, or of single, or double charge). One should keep in mind that the above considerations and estimates do not apply to neutron sputtering.

When using Eq. (1) to estimate $\Delta\ell_\nu$, the annual thickness loss due to sputtering process ν, it is necessary to know the flux $\dot{\Phi}_\mu$, and energy E_μ of particle species μ, and the sputtering yield $S_{\mu\nu}(E_\mu)$ for particle species μ and process ν, under the appropriate irradiation conditions (e.g., angle of incidence). At present our knowledge of the flux $\dot{\Phi}_\mu$ for

Table 1

Values of the limiting energies E_A between the hard-sphere collision region and the weakly-screened Coulomb collision region and E_B between the latter and the Rutherford collision region. The projectiles and wall materials shown are some of those of interest for fusion reactors. All energies are in keV.

Projectile	Vanadium		Niobium		Molybdenum	
	E_A	E_B	E_A	E_B	E_A	E_B
H^+	1.9	2.8	4.1	7.0	4.2	7.2
D^+	2.0	5.7	4.1	14.0	4.3	14.5
He^+	4.2	48.3	8.6	117.2	8.9	120.8

the various types of projectiles is too fragmentary to allow reasonable estimates of $\Delta\ell_{\nu}$; here tne plasma physicists will have to provide more realistic ϕ_{μ}-values. Furthermore, we have only an extremely limited knowledge of the sputtering yield $S_{\nu\mu}$ for the various types of projectiles, ranges of particle energy, and materials now being considered for the vacuum walls of fusion reactors. This is especially true for the extreme operating conditions envisioned for fusion reactors-e.g., for vacuum walls at temperatures of 600-1000°C bombarded by high fluxes of several species of energetic particles simultaneously. To provide at least a qualitative idea of the range of sputtering yields which might be expected for some of the projectiles and wall materials of interest, Table 2 lists sputtering yields calculated according to the theories of Pease[33] and of Goldman and Simon.[34] The limitations of these and other theoretical treatments of sputtering in the Rutherford collision region ($E > E_B$) are discussed in detail in Refs. 16 and 17. Since for this energy region experimental sputtering yields S are scare or nonexistent for most wall materials considered, both experimental and theoretical values are given for Cu to allow a better comparison. In view of the uncertainties in the experimental values and the rather severe approximations made in both theories, theoretical and experimental values differing by less than a factor of 3 are considered to be in satisfactory agreement. In particular, since neither theory takes account of such lattice effects as momentum focusing along close-packed rows of atoms when monocrystals are irradiated,[35] any relatively close agreement between the theoretical S values and the experimental ones obtained for monocrystalline Cu[18,32] should be considered as fortuitous. (For a more detailed discussion of experimental and theoretical results for the energy region $E > E_B$, see Ref. 16). It is of interest to note that the sputtering yields decrease with increasing energy over the listed energy ranges of all projectiles in Table 2. Note also that the sputtering yields for deuterons at the lower energies (e.g., at 20 keV) are significantly higher than those for the heavier helium ions at higher energies (e.g., at 3.5 MeV).

In order to estimate the annual thickness loss $\Delta\ell_{\nu}$ for a particular sputtering process ν by use of Eq. (1), it is also necessary to know the flux ϕ_{μ} and the energy E_{μ} of the incident particles (neutral atoms or

Table 2

Sputtering yields S (atoms/ion) for wall materials and projectiles of interest in fusion reactors. The values calculated according to the theories of Pease (P) and of Goldman and Simon (G) are compared with the experimental values (Exp).

Wall	H^+ Projectile energy E (keV) 10	20	50	100	200	500	D^+ Projectile energy E (keV) 10	20	50	100	200	500
V P	0.0097	0.0061	0.0031	0.0018	0.0010	0.0005	0.0240	0.0150	0.0072	0.0041	0.0023	0.0011
G	0.0030	0.0018	0.0008	0.0005	0.0003	0.0001	0.0069	0.0040	0.0019	0.0011	0.0006	0.0027
Nb P	0.0081	0.0056	0.0031	0.0018	0.0011	0.0005	0.022	0.014	0.0073	0.0043	0.0024	0.0011
G	0.0027	0.0016	0.0008	0.0005	0.0003	0.0001	0.0065	0.0038	0.0018	0.0010	0.0006	0.0003
Exp							0.0059[c]	0.0040[d] 0.0042[e]				
Mo P	0.0086	0.0060	0.0033	0.0020	0.0012	0.00055	0.0240	0.0150	0.0079	0.0046	0.0026	0.0012
G	0.0029	0.0018	0.0009	0.0005	0.0003	0.0001	0.0070	0.0042	0.0020	0.0011	0.0006	0.0003
Cu P	0.0160	0.0100	0.0053	0.0031	0.0018	0.0008	0.041	0.025	0.012	0.0070	0.0039	0.0018
G	0.0036	0.0021	0.0010	0.0006	0.0003	0.0002	0.0085	0.0049	0.0023	0.0013	0.0007	0.0003
Exp	0.022[a]				0.0013[b]	0.0010[b]				0.0028[f]	0.0020[f]	0.0011[f]

Wall	He^+ Projectile energy E (keV) 10	20	100	500	1000	3500
V P	0.2300	0.1350	0.036	0.0092	0.0050	0.0017
G	0.0630	0.0360	0.0093	0.0023	0.0012	0.0004
Nb P	0.2250	0.1370	0.0390	0.0100	0.0055	0.0018
G	0.0610	0.0350	0.0093	0.0023	0.0013	0.0004
Exp		0.051[e]				
Mo P	0.2410	0.1470	0.00[illegible]2	0.011	0.0060	0.0020
G	0.0660	0.0380	0.010	0.0026	0.0014	0.0005
Cu P	0.3910	0.2300	0.0620	0.0160	0.0085	0.0028
G	0.078	0.0440	0.0110	0.0028	0.0015	0.0005

[a]Ref. 37. Value measured for 8-keV H^+ on Au.

[b]Ref. 18. Value measured for monocrystalline Cu(111) target, theories not applicable.

[c]Ref. 7. Value measured for 12.2-keV D^+.

[d]Ref. 7. Value measured for 18.8-keV D^+.

[e]Ref. 8. The Nb target temperature was 1100°C.

[f]Ref. 16, p. 236. Values measured for monocrystalline Cu (100) target, theories not applicable.

molecules, non-contained ions). Unfortunately, these values also are subject to great uncertainties; not only will the actual fluxes and energies depend on the type of fusion reactor (e.g., high-β or low-β devices, toroidal or mirror machines) and the fuel cycles chosen, but also on the actual operating conditions (e.g., on the fuel burn-up fraction and stability of confinement). It is therefore not surprising to find flux estimates varying from zero (no charged or neutral plasma particles reaching the wall) to the maximum possible value (all fuel particles reaching the wall). Both extreme cases are probably unrealistic. For example, in estimating the lifetime of a Nb vacuum wall of a 5000-MW (thermal) reactor with a wall area[2] of 380 m^2, Summers et al.[7] assume a total particle flux (deuterium, tritium, and helium) of 2×10^{16} particles $cm^{-2}s^{-1}$, while Yonts[8] considers a 30,000-MW (thermal) reactor with a wall area[3] of 3000 m^2 and assumes a deuteron flux of 1×10^{16} $cm^{-2}s^{-1}$. Both estimates assume that all fuel particles reach the wall.

With these estimates for ϕ_μ and the criterion that not more than 20% of the Nb wall thickness could be eroded away without excessive loss of strength, Yonts concluded that deuteron sputtering alone would limit the lifetime of a 1/4-in.-thick Nb wall to 3.5 y (corresponding to an annual thickness loss $\Delta\ell_\mu$ = 0.035 cm). For a corresponding Mo wall, he estimated that the lifetime would be shortened to 1.5 y ($\Delta\ell_\mu$ = 0.080 cm). Summers et al. estimated the lifetime of a 1-cm-thick Nb wall to be approximately 1.7 y ($\Delta\ell_\mu \approx$ 1.2 mm/y for an average sputtering yield $\bar{S}$ = 0.008 atom/ion). Daniel and Finfgeld[11] estimated the lifetime of a 1-cm-thick Nb wall of a 30,000-MW (thermal) D-T reactor[3] (total wall area = 3000 m^2) as a function of the unburned-fuel fraction f_b (assuming that the total unburned fraction strikes the walls). For example, for burn-up fractions ranging from 0.5% to 10%, the wall lifetime (for eroding away 20% of the 1-cm-thick wall) ranges from 0.18 to 3.7 y.

None of the preceding estimates took account of the possibility that gas species with high Z may be sputtered, vaporized, or desorbed from the walls, may then gain energy by collisions with energetic plasma particles, and finally may return and strike the wall. In the process they may be ionized and possibly later neutralized by charge-changing collisions with plasma particles. The sputtering yields of these heavy particles, whose energies are much lower than those of the primary plasma particles, can be orders of magnitude higher than those for 20-keV deuterons. For example, from the Nb self-sputtering ratios calculated by Summers et al.[7] one can estimate that the sputtering yield for 1-keV Nb projectiles striking an Nb wall is approximately 100 times that for a 20-keV deuteron striking the same wall. This means that for equal rates of erosion by sputtering, the flux of 1-keV Nb would need to be only 1% of the flux of 20-keV deuterons (or deuterium atoms). Heavy metals present as impurities in the Nb would likewise contribute to the sputtering. Ultra-high-purity niobium would very likely be uneconomic for reactor construction; and even such ultrahigh purity niobium purified by triple-pass electron-beam float-zone refining contains Ta, W, and Mo impurities with concentrations varying between 100 and 500 ppm.[36]

The preceding discussions and estimates also took no account of wall erosion by energetic neutrons, though this process would occur for most fuel cycles and for most types of fusion reactors considered to date. As MeV neutrons penetrate a solid wall, they undergo elastic and inelastic scattering by the nuclei of the lattice atoms. Since the neutrons carry no charge, they impart momentum directly to the nuclei with which they collide. If 14-MeV neutrons interact with the nuclei of Nb lattice atoms, the neutron cross sections for elastic and inelastic scattering have approximately the same value ($\sim$ 2 barns).[38] (The cross section for displacement of a Nb lattice atom by impact of a 14-MeV deuteron is $\sigma_d \approx$ 4.8×10^3 barns-approximately 2400 times the value for neutrons of the same energy!)

The energy that can be transferred in an elastic collision between such an energetic neutron and the nucleus of a lattice atom ranges from zero to a maximum energy

$$E_{max} = \frac{4M_2 m}{(M_2 + m)^2} E_n,$$

where M_2 and m are the atomic mass numbers of the lattice material and the neutron, respectively, and E_n is the primary neutron energy. For a

14.1-MeV neutron, for example, the maximum energy transferable to the nucleus of a Nb lattice atom is E_{max} = 593.6 keV, and the mean energy $\bar{E}$ of a "primary knock-on" (a lattice atom displaced by a neutron) has been estimated by Myers[39] to be $\bar{E} \approx$ 181 keV, and by Robinson[38] to be $\bar{E} \approx$ 106 keV. The difference between these two values of $\bar{E}$ is not surprising in view of the different approximations used in calculating them. [Robinson's calculations were more refined in that, for example, he treats inelastic scattering by nuclear evaporation theory and treats other nonelastic scattering processes-such as (n, 2n) reactions-as if they were inelastic.]

It is important to realize that these high values of $\bar{E}$ for the primary knock-ons are several orders of magnitude larger than those for primary knock-ons resulting from MeV deuteron bombardment of Nb; e.g., the average knock-on energies $\bar{E}$ are about 364 and 328 eV for 14-MeV and 5-MeV D^+, respectively. Since the values of $\bar{E}$ for 14-MeV neutron bombardment are so much higher, it is not surprising to find that the mean number $\bar{\nu}$ of displaced atoms per primary knock-on is also orders of magnitude larger for bombardment by 14-MeV neutrons than for bombardment by 14-MeV deuterons. For bombardment of Nb by 14-MeV neutrons, $\bar{\nu}$ can be crudely approximated by the relation[40] $\nu \approx 0.561\ \bar{E}/E_d$, which for $\bar{E} \approx$ 146 keV gives $\bar{\nu} \approx$ 2800 displaced atoms per primary knock-on. For 14-MeV deuteron bombardment of Nb, an approximation formula[40] is

$$\bar{\nu} \approx 0.5\ E_{max}(E_{max} - E_d)^{-1}\ [1 + \ln(E_{max}/2E_d)]$$

$$\approx 5.4 \text{ displaced atoms per primary knock on.}$$

These considerations and estimates make possible a qualitative comparison between the sputtering yields for 14-MeV neutrons and deuterons. If one follows Pease's simplified model of sputtering, then the sputtering yield is $S \approx An'N$, where $A = \sigma_d n_o^{2/3}$ is the effective collision area available within an atomic monolayer of a solid with displacement cross section σ_d and particle density n_o (per unit volume), n' is the number of layers contributing to sputtering, and N is the total number of displaced particles per primary knock-on and is proportional to $\bar{\nu}$. (Pease suggests only proportionality to $\bar{\nu}$.) In using this formula to compare the values of S for 14-MeV neutrons and 14-MeV deuterons, one finds that $A(n)/A(d) \approx$ 1/2400, $n'(n)/n'(d) \approx$ 5-10, and $N(n)/N(d) \approx$ 560. This leads to the conjecture that the sputtering yields for 14-MeV neutrons on Nb are slightly larger than those for 14-MeV deuterons [$S(n)/S(d) \approx 1$-2]. For 14-MeV deuterons on Nb, the sputtering yield calculated according to the theory of Pease is $S(d) = 0.6 \times 10^{-4}$ atoms/ion

Since sputtering yields have not been measured for 14-MeV neutron sputtering of V, Nb, or other materials suggested as vacuum walls in fusion reactors, no direct comparison can be made. In fact, the information now available on MeV neutron sputtering is so scarce and contradictory that one cannot be reasonably certain of even the order of magnitude of the sputtering yields.

Keller and Lee[41] reported a yield S = 0.5 atom/neutron for the sputtering of polycrystalline Au by 4-MeV neutrons, but Keller[42] pointed out in a subsequent paper that this extremely high value may have been the result of systematic experimental errors. In this second paper he also reported experimentally determined upper limits for the sputtering yields for 14-MeV neutrons. These values range from $S < 3.9 \times 10^{-2}$ atom/neutron for Cu to $S < 6 \times 10^{-4}$ atom/neutron for Au. Anno et al.,[43] who bombarded Au with fission neutrons partially moderated in water (energies ranging from thermal up to $\sim$ 6 MeV), observed a yield $S = 1 \times 10^{-4}$ atom/neutron. They felt that only neutrons with energies $E > 0.1$ MeV were significant in sputtering. The sputtering of monocrystalline and polycrystalline copper under bombardment by neutrons from a reactor was studied for a range of neutron doses, and was reported recently by Garber et al.[44] Fedorenko et al.,[45] who studied the sputtering of a monocrystalline Au target by monoenergetic 14-MeV neutrons, reported a sputtering yield of 3×10^{-3} atom/neutron, which is five times the upper limit reported by Keller[42] for polycrystalline Au. Sputtering yields of the order of several times 10^{-3} atoms/neutron are qualitatively in good agreement with the values which we observed for MeV deuteron bombardment of Cu,[16,32] and they also agree with our value[16] $S = 1.3 \times 10^{-3}$ atom/ion for bombardment of Ag by 1.0-MeV D^+. This agreement seems to support our conjecture that MeV neutrons and deuterons should have similar sputtering yields (at least within one order of magnitude).

Sputtering yields of several times 10^{-3} atoms/neutron could also cause serious wall erosion and plasma contamination for the large total neutron fluxes that would be encountered in large fusion reactors. For example, for D-T fusion reactors of 5000-30,000 Mw (thermal) power, the energetic total neutron flux (including neutron backshine from the blanket) is estimated[2,5] to be $\sim 3\text{-}4 \times 10^{15}$ neutrons $cm^{-2}\ sec^{-1}$.

On the assumptions that the average neutron sputtering yield is 5×10^{-3} atoms/neutron for MeV neutrons, that the flux is 4×10^{15} neutrons $cm^{-2}\ sec^{-1}$, and that these energetic neutrons will cause approximately equal sputtering on the two sides of the vacuum wall, the erosion of a 1-cm-thick Nb wall by neutron sputtering alone would lead to an annual thickness loss $\Delta\ell_{\mu}$ = 0.38 mm, which would limit the wall lifetime (for 20% loss of thickness) to approximately 5-1/2 years. In view of the scarcity and contradictory nature of the experimental data on sputtering by MeV neutrons, such estimates are embarrassingly crude. A thorough study of MeV neutron sputtering of Nb and other materials proposed for the vacuum walls of fusion reactors is imperative. For example, the sputtering yield needs to be studied as a function of target temperature, of target materials and structure, of the partial pressures of H_2, D_2, and other reactive gases, and of neutron energy, flux, and total dose.

D. Desorption by Electron Impact

The desorption of gases from solid surfaces under impact of sufficiently energetic electrons has been observed by several authors (e.g., see Refs. 46-51) usually at electron energies ranging from 15 to 500 eV. The desorbed species leaving the surface include neutral atoms, molecules, and positive and negative ions-some in their ground states and others with varying amounts of excitation energy. For example, Redhead[52] has shown that when CO is adsorbed in one of the low-energy binding states on tungsten and is desorbed by electron impact, it appears as an excited neutral; and that O^+ can be desorbed both from the weakly bound state of O_2 and from CO adsorbed on tungsten. The threshold energy required for the desorption of an ion or neutral particle by electron impact is commonly in the range between 15 and 25 eV; the few measured values are reviewed in Ref. 24.

Desorption by electron impact can very well contribute significantly to plasma contamination. In an operating fusion reactor, the large fluxes of energetic photons and of particles (both ions and neutral atoms formed by charge exchange) cause copious emission of electrons from the walls; electrons are also emitted thermally from the hot walls; and as all these electrons leave the walls, they are turned back by the external confining field and cause further electron emission by electron impact. The energies of the electrons resulting from these diverse processes cover a wide range-from very low values up to energies comparable to those of the energetic photons. Thus large fluxes of secondary electrons must be expected in an operating fusion reactor, the magnetic confining field will deflect them so they return to the walls, and their impacts on the walls will be sufficiently energetic to desorb gas species adsorbed on the surface-and even to desorb species that are stable under high temperature and adhere to heated surfaces. These adsorbed gas layers are formed on the surface both by particles coming from the vacuum and by occluded gases and other species permeating through the bulk material.

Finally, as the energetic secondary electrons leave the surface and are brought back by the action of the magnetic confining field, they form an electron sheath on the vacuum side of the wall. This sheath in turn will be useful in ionizing the neutral gas species leaving the surface and will thus make it feasible to apply electric and magnetic fields (e.g., by divertors) to remove them from the vacuum vessel before they can contaminate the plasma.[53] Another effect of the electron sheath is that the "image charge" induced on the metal surface will significantly alter such electronic properties of solid surfaces as the work function.

3. PHENOMENA INDUCED BY PHOTON IMPACT

Energetic photons (e.g., synchrotron radiation, bremsstrahlung, x-rays, and γ-rays from nuclear reactions) impinging on fusion reactor components such as the walls of a plasma container or of a divertor produce a variety of processes. Some of these are: photo-desorption of adsorbed or absorbed gases, photo-decomposition of surface compounds, photo-catalysis leading to a reaction between adsorbed molecules and the vacuum wall, photoelectron emission, sputtering by the conversion of high electronic-excitation energy into displacement energy, rapid vaporization as a result of the high surface temperature and temperature gradients produced by photon absorption in layers near the wall surface, and possibly cracking of the bulk material and flaking of the surface.[54] One should also consider the desorption of adsorbed or absorbed gases under the impact of energetic photoelectrons, e.g., those that return to the wall under the action of a high external magnetic field.

These processes in turn can erode the walls and seriously contaminate the plasma with the gases they release, for example, contamination by only 1 atom % of the fully stripped oxygen will cause a 77% increase in the power lost through bremsstrahlung.

The spectra of both the bremsstrahlung and the synchrotron radiation cover a wide range of photon energies. Under typical conditions in a D-T fusion reactor, the bremsstrahlung can be expected to cover mostly the γ-ray, x-ray, and ultraviolet regions of the spectrum; and the synchrotron radiation will probably fall mainly in the infrared and conventional microwave regions. An excellent treatment of the possible power loss due to synchrotron radiation in a hypothetical D-T fusion reactor with small β has been given recently by R. G. Mills.[55] It is safe to assume that the energetic photons (and the photoelectrons they produce) desorb gases and erode the walls with varying effectiveness in different regions of the photon spectrum. Especially for very energetic photons (x-rays and γ-rays), the rates of gas release and wall erosion are completely unknown for Nb, Nb alloys, and other materials considered for the container walls and for the extreme operating conditions of a D-T fusion reactor. These effects for γ irradiation are indicated only by some fragmentary results[56] on the total outgassing rate of well cleaned and baked 6061-6063 aluminum irradiated with ^{60}Co γ-rays under high vacuum. The outgassing by synchrotron radiation has been studied in electron storage rings, in which the electrons producing the radiation are much more energetic than those envisioned in the operation of fusion reactors. In these studies, the wall surfaces were of Cu and stainless steel in the 3-GeV electron storage ring investigated by Fischer and Mack[57] and 304 stainless steel in the 200-MeV electron-positron ring investigated by Bernardini and Malter.[58]

Several reports (e.g. Refs. 59-65) indicate that desorption of gas from solids irradiated by visible or ultraviolet photons can potentially be a very serious source of plasma contamination. However, the available information is still too fragmentary to provide any reliable estimate of the rates of outgassing and erosion of materials considered for reactor components under the conditions of temperature and photon fluxes (especially of x-rays and γ-rays) envisioned during operation of a fusion reactor.

There is an urgent need for information on particle emission under photon impact on the vacuum walls. In particular, the phenomena need to be studied as a function of such parameters as photon energy, total photon dose, surface temperature, partial pressures of such gases as H_2, D_2, O_2, and N_2, and the nature of the irradiated surface (e.g., its gas content and its texture).

4. THERMAL EFFECTS

A. Summary of Thermal Phenomena

When a fusion reactor is in operation, not only the vacuum wall but also such internal components as beam limiters and divertors are heated both by the energetic particle and photon flux from the plasma and by secondary radiations. For particle radiations, the thermal loadings depend on the mass, atomic number, and charge state; and for all radiations, the loadings depend on the energy. For example, low-energy photons and particles have very short ranges in a solid so their energies heat the layers near the surface. Depositing all the energy in a shallow layer results in high surface temperatures (and perhaps surface melting) and sets up steep temperature gradients that may lead to cracking of the bulk material and flaking of the surface. More energetic radiations penetrate deeper into the solid and distribute the heat more uniformly. The heating by 10-15-keV bremsstrahlung is being studied by J. A. Phillips and collaborators.[54]

This heating of solid reactor components can cause thermal evaporation and desorption, thermal emission of electrons and ions, whisker growth, enhancement of the secondary-particle emissions[16,17,24] mentioned above, and acceleration of such processes as gas permeation and diffusion. These processes in turn can lead to serious plasma contamination and wall erosion. Only a few of these phenomena can be discussed in the following subsections.

B. Thermal Evaporation

The rate of vaporization of a solid wall heated in a vacuum depends on the vapor pressure of the material at the operating temperature. Some typical vapor pressure curves[65] for metals of potential interest for fusion reactor components are shown in Fig. 5. Over the temperature range shown, the vapor pressure of vanadium at any given temperature is higher than for any other metal shown. On the basis of the criterion that the fractional impurity concentration f resulting from the evaporation should not cause more than a 10% power loss due to bremsstrahlung, Craston et al.[14] conclude that the temperature of a vanadium wall should not exceed about 1080°C.

The formation of volatile compounds under certain reactor operating conditions may lead to substantially higher rates of evaporation than would be predicted for the elemental metals at the same temperature. For example, the vapor pressure of the molybdenum oxide MoO_3 is 10^{-4} Torr at 565°C and 10^{-2} Torr at 636°C. For the vanadium nitride VN, the vapor pressure is 10^{-4} Torr at 857°C and 10^{-2} Torr at 1027°C. Such high vapor pressures would lead to intolerably high plasma contamination.

The loss of wall thickness by evaporation may also affect the operation of a fusion reactor. The annual thickness loss Δ_{vap} (in mm/year) due to vaporization can be calculated from the simplified expression

$$\Delta_{vap} = 1.84 \times 10^7 \, (p/\rho) \sqrt{M/T},$$

where p is the vapor pressure (Torr) at temperature T (°K), M is the molecular weight, and ρ(g/cm^3) is the density of the solid material being evaporated. For example, for vanadium at 1227°C (1500°K), the annual thickness loss is $\Delta_{vap} = 0.055$ mm/year; and this, as explained in

Section 1, is 55% of the maximum permissible annual thickness loss $\Delta\ell_{tot}$ = 0.1 mm. Such a high $\Delta\ell_{vap}$ is clearly unacceptable when one considers that many other erosion processes (including but not limited to those discussed above) contribute to $\Delta\ell_{tot}$.

The rates of vaporization presumably will be altered by the structural and chemical changes induced in the solids by the severe plasma radiations, but the information needed to predict the importance of such effects still is completely lacking.

C. Absorption and Adsorption; Desorption

Gas particles impinging on solid surfaces with thermal energies interact by several quite dissimilar processes. Some penetrate into the bulk of the solid (e.g., by permeation) where they may form a compound (by chemical reaction) or a solution of the gas in the solid. This process is known as absorption or occlusion. Other gas particles stick to the surface and form a gas layer, a process called adsorption. Three limiting forms of adsorption can be distinguished in terms of the type of binding force between the adsorbate and the absorber. In order of increasing strength of binding, these types of adsorption are (1) physisorption (predominantly by van der Waals forces), (2) weak chemisorption (predominantly by exchange forces), and (3) strong chemisorption (predominantly by heteropolar binding forces). Typical bond energies range from values up to 10 kcal/mole in physisorption to 280-300 kcal/mole in strong chemisorption.

The bond energies in chemisorption depend both on the adsorbed species and on the substrate. In general, the bond energies for some common gas species chemisorbed on a given metal surface decrease in the order $O_2 > C_2H_2 > C_2H_4 > CO > H_2 > CO_2 > N_2$. For H_2, O_2, N_2, CO, and CO_2 adsorbed on different metals, the energies of chemisorption decrease in the order Ti, Ta > Nb > W, Cr > Mo > Fe. The actual gas/metal systems formed by chemisorption in any particular case will depend on the composition and pressure of the residual gas in the plasma vessel, on the material of this vessel, and on the concentration and species of gas occluded in it.

When a surface is heated sufficiently, the adsorbed gas layers are desorbed. The desorption of gases strongly chemisorbed on metals (especially on refractory metals) usually proceeds in steps occurring at different surface temperatures. In the case of oxygen adsorbed on Mo, a system in which molecular oxygen is weakly bound on top of a more tightly bound oxide layer, the molecular oxygen ("the α-phase desorption peak") is desorbed at about 400-500°K while the more tightly bound layer of molybenum oxide (the "β-phase desorption peak") remains stable up to about 1100°K. In the desorption of carbon monoxide from tungsten,[24] the α-phase desorption peak occurs at ∿ 500°K while the β-phase peaks occur at 900-1400°K. Thus it is apparent that the often suggested degassing of the vacuum walls of a fusion reactor at "typical outgassing temperatures" of about 400°C would do little to free the surfaces from strongly chemisorbed gases.*

When more than one chemically active gas is adsorbed, the desorption process becomes more complex. For a system in which the residual gases were H_2 and CO at a total pressure of 1×10^{-10} Torr, for example, Redhead et al.[66] observed that the desorption consisted in α-phase CO at about 400°K, in H_2 between 600 and 700°K, and in β-phase CO at 1600-1700°K.

Recent studies[67] of the velocity distributions of H_2 and D_2 desorbed from metal surfaces (e.g., nickel) showed that the mean energy of the desorbed molecules was 30-200% greater than the equilibrium energy in a gas at the temperature of the surface.

Alkali metals may also be desorbed from the hot wall surfaces. The wall materials may contain these metals as impurities and/or the liquid alkali metals used to cool the walls may have diffused through the wall. Some of these atoms will desorb as neutrals and some as ions, the relative abundances being a function of the surface temperature. (The desorption of alkali metals from W has been described by for

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*Particle impact desorption (see Section 2, D) will aid greatly in surface degassing.

example in Ref. 68, and a more general discussion of the "surface ionization" phenomenon has been presented in Refs. 16 and 24).

There is a clear need for more information of thermal influences on particle emission and wall erosion, especially for the materials considered for vacuum walls and for the divertor. These measurements should be performed under conditions comparable to those encountered in the actual operation of fusion reactors.

5. CONCLUSION

From the available fragmentary knowledge about surface phenomena induced by the impact of particles and photons and about the associated thermal effects, it has become obvious that many of these processes are potential sources of serious trouble in the operation of large fusion reactors. The problems of plasma contamination and wall erosion become especially acute when one considers the combined effects of the above processes under the extreme thermal and radiation conditions under which a fusion reactor is expected to operate.

As a prerequisite for even a first-order feasibility study of large fusion reactors, it will be necessary to obtain a set of first-generation yield values for each of the important secondary-particle-emission processes and for each of the vacuum-wall materials considered. In a second step, it will be necessary to study the yields under the extreme operating conditions of a large fusion reactor (i.e., to study the effects of simultaneously subjecting the walls to high fluxes of energetic particles and photons and to high temperatures). The results of such studies will then permit a more realistic second-order feasibility study.

With this as background, it will become more meaningful to initiate studies of possible methods of reducing (and possibly of eliminating) some of the major sources contributing to erosion of the vacuum wall and to contamination of the plasma.

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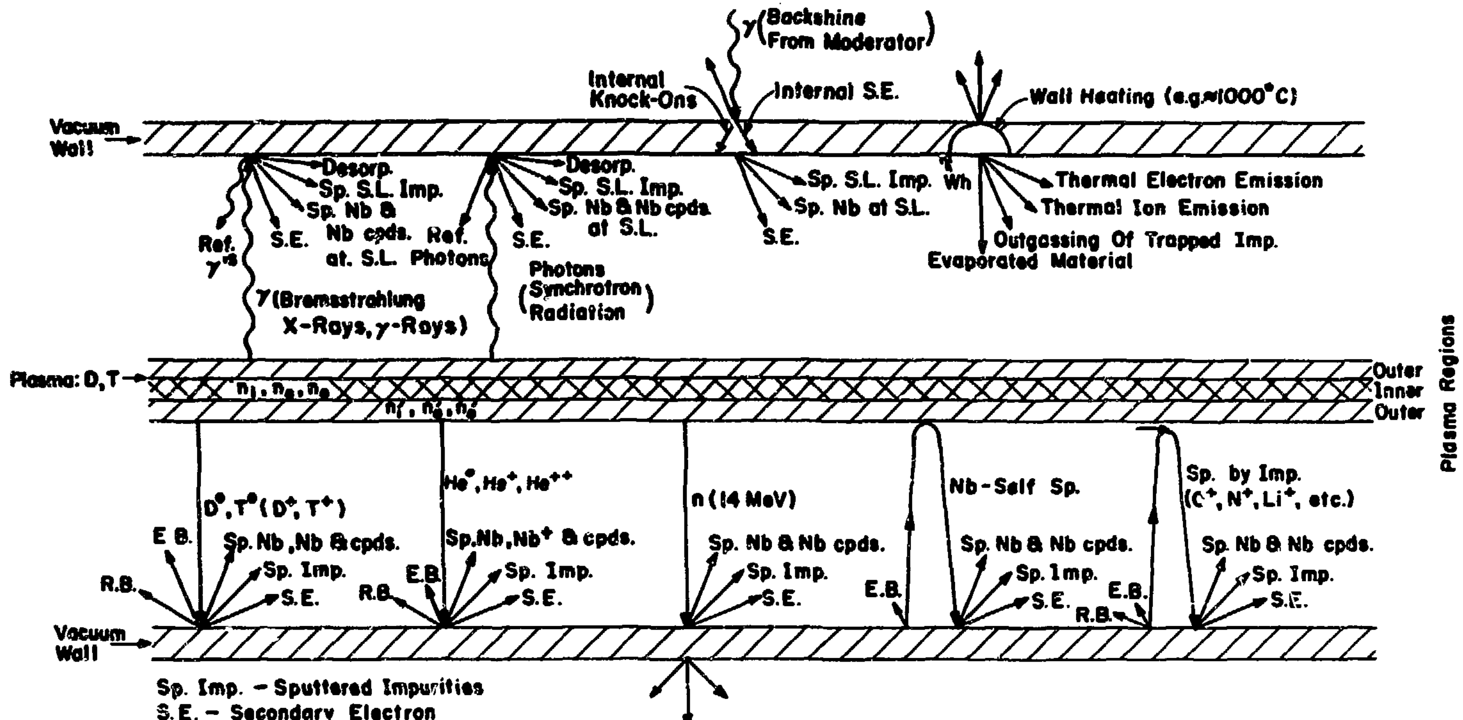

Sp. Imp. – Sputtered Impurities
S.E. – Secondary Electron
E.B. – Elastically Backscattered
Ref. – Reflected γ's or Photons
cpds. – Compounds
Desorp. – Desorption of Absorbed Gas Layers (Weakly and Strongly Bound)
S.L. – Surface Layer
R.B. – Radiation Blistering
Wh – Whisker Growth

Fig. 1. Secondary Phenomena Which May Occur at the Heated Walls of a Plasma Container Under the Impact of Energetic Particles and Photons.

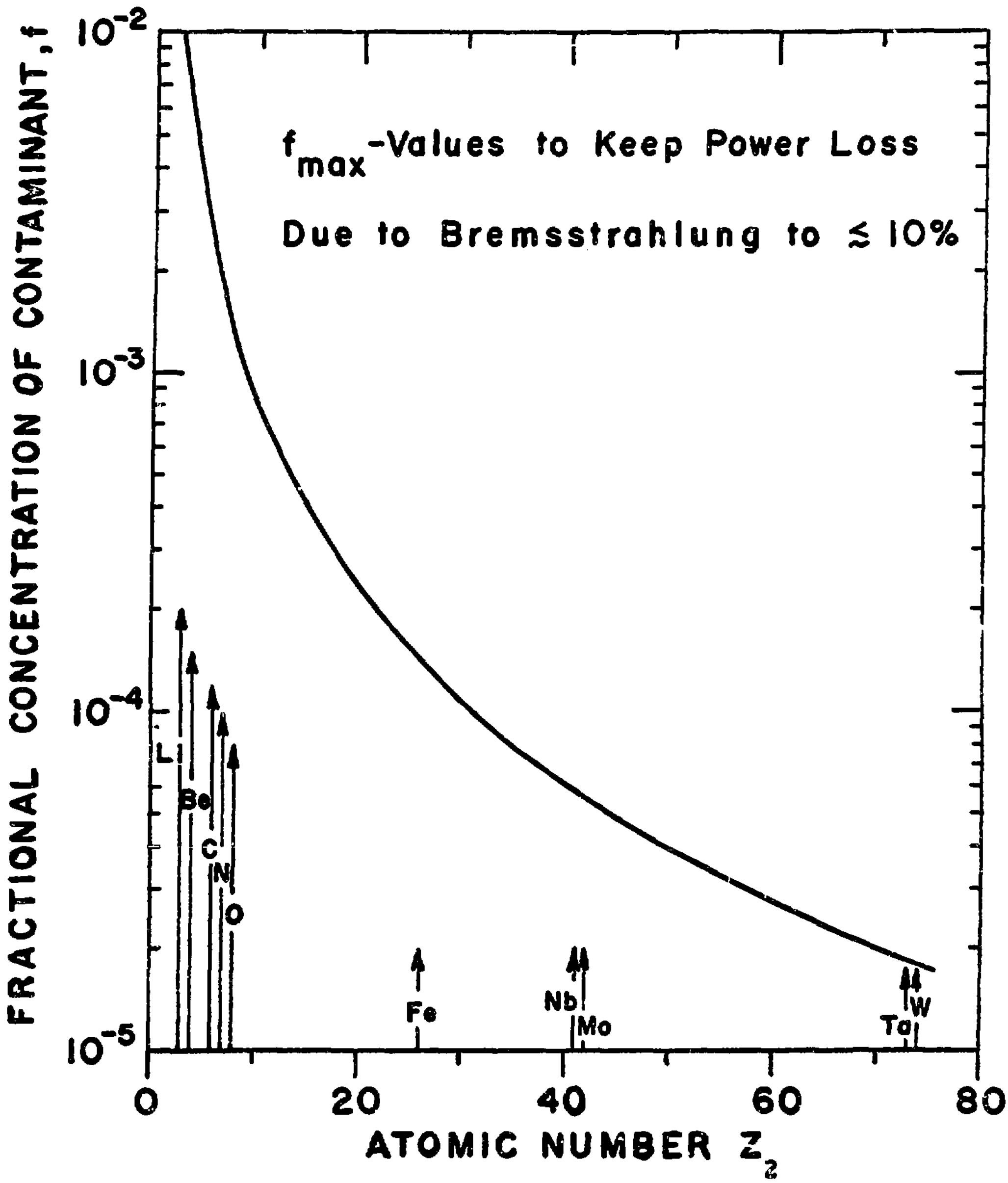

Fig. 2. Maximum Fractional Concentration of Contaminant, f_{max}, in Dependence on its Atomic Number, Z_2.

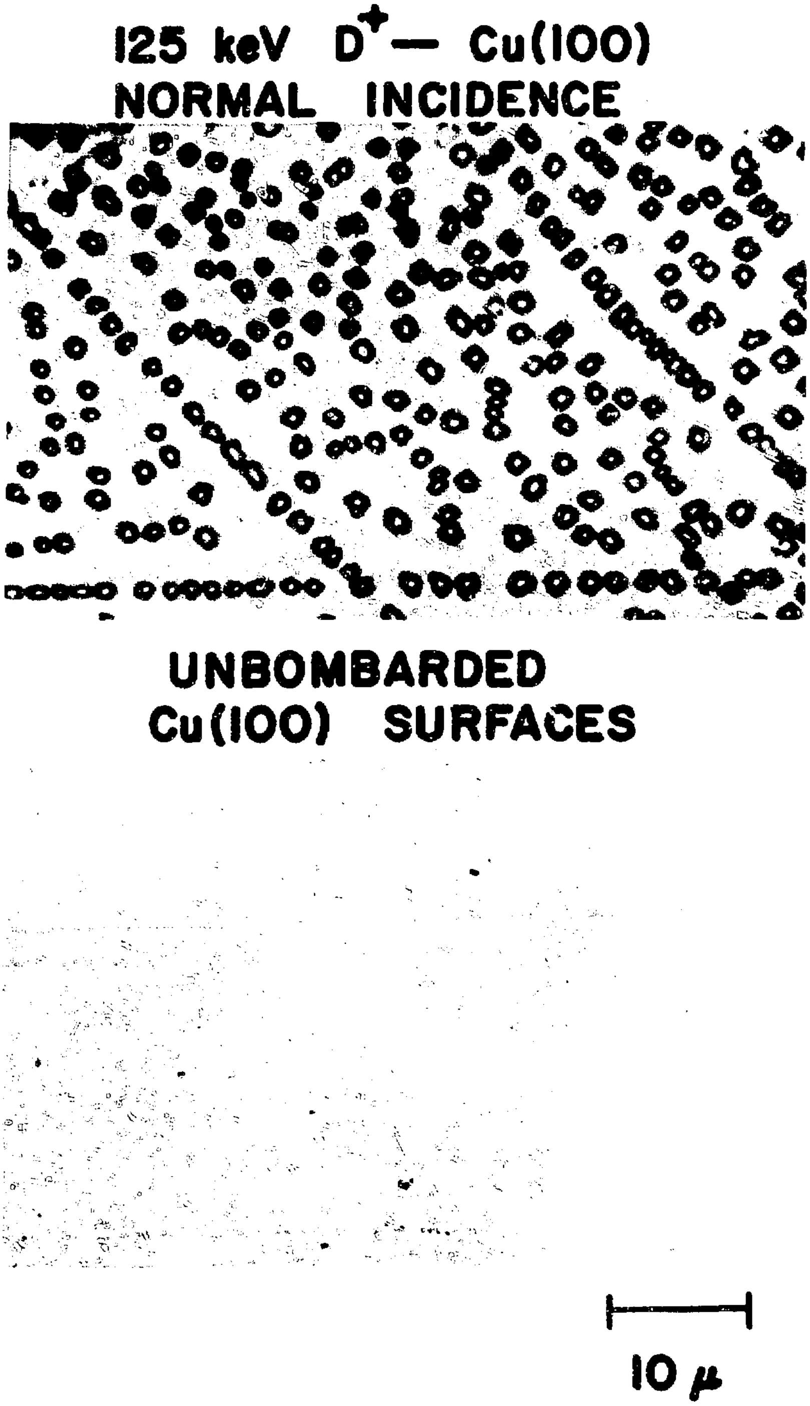

Fig. 3. Metallographs of a Cu(100) Plane Bombarded by 125-keV D' Ions at Normal Incidence for Approximately 800 μA-hr (Upper Micrograph) and for an Unbombarded Cu(100) Plane (Lower Micrograph) (M. Kaminsky).[29,69]

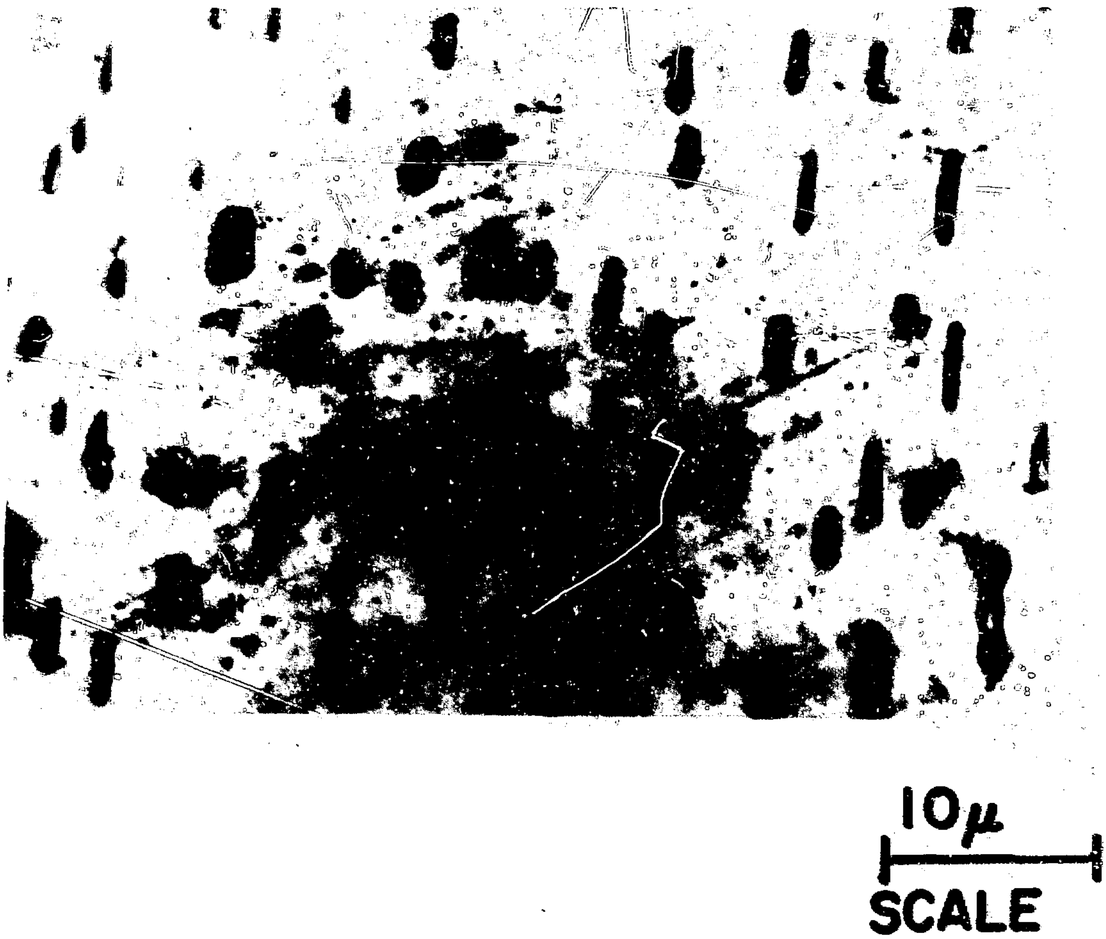

Fig. 4. Electron Micrographs of a Cu(110) Plane Bombarded by 800-keV D^+ Ions at an Angle of Incidence α = 45° for Approximately 110 μA-hr. The elongated grooves are oriented along the beam direction (M. Kaminsky[16]).

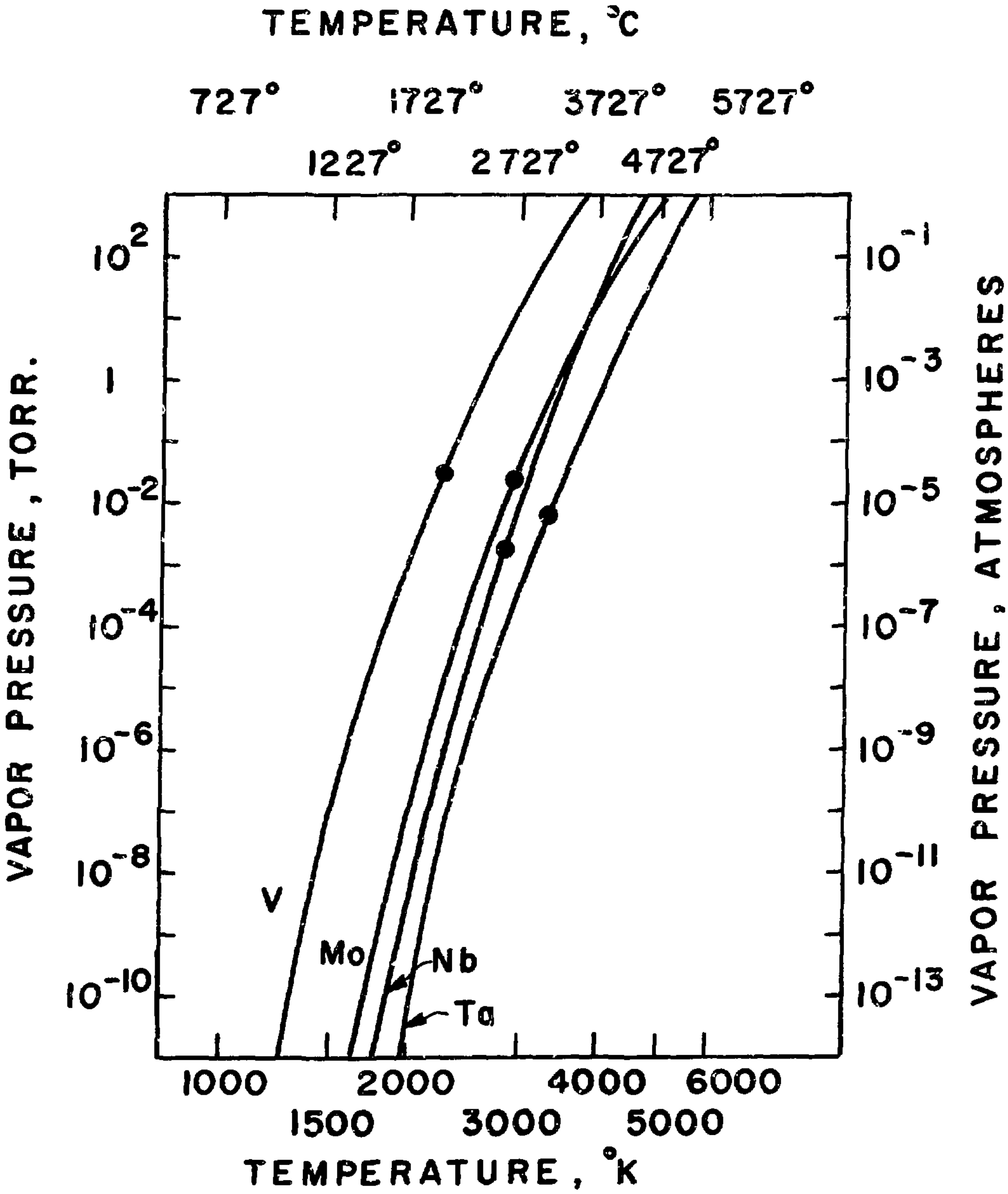

Fig. 5. Vapor Pressures of V, Mo, Nb, Ta. The solid points indicate the melting point (after Rosebury[65]).

SUMMARY OF SESSION

SURFACE PHENOMENA LEADING TO PLASMA CONTAMINATION AND WALL EROSION IN THERMONUCLEAR REACTORS AND DEVICES

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INTRODUCTION

The session was organized to cover three major topics: (A) Phenomena induced by particle impact, (B) phenomena induced by photon impact, and (C) thermal effects. The papers were followed by a general discussion which centered around the following questions: (1) How are wall erosion and plasma contamination affected by the combined action of various surface phenomena under the extreme thermal and radiation conditions under which a fusion reactor is expected to operate? (2) What directions should surface studies take in the near and far future to provide information for feasibility studies and the design of large fusion reactors? (3) What type of research is required to provide answers to ongoing work with plasma devices?

A. PHENOMENA INDUCED BY PARTICLE IMPACT

1. Physical Sputtering

Plasma contamination and wall erosion caused by physical sputtering were discussed by R. Behrisch,[1] C. Finfgeld,[2] G. M. McCracken,[3] and M. Kaminsky.

Behrisch reviewed data on physical sputtering with energetic ions and discussed the effect of surface contamination on sputtered yields. He pointed out that the walls will be contaminated not only during the start-up of a fusion reactor but also during its operation. At this time it is difficult to predict how the contamination of a fusion reactor wall will affect the sputtering yields and thereby the rates of wall erosion and plasma contamination. For example, oxide layers on certain materials tend to lower the sputtering yield, whereas adsorbed gas layers are generally sputtered off more readily.

Finfgeld presented some estimates for the lifetime of a 1-cm-thick Nb wall of a 30,000-MW (thermal) D-T reactor as a function of the unburned fuel fraction. The estimates are based on several simplyfying assumptions, for example that all the unburned fuel particles strike the wall, and that the sputtering yields for uncontaminated surfaces can be used. For burn-up fractions of 3% or less, it is assumed that the wall erosion is caused predominantly by sputtering from fuel-particle impact; for burn-up fractions above approximately 15%, the sputtering by the reaction products (He atoms and neutrons) begins to contribute significantly (> 10%) to the total wall-erosion rate. For burn-up fractions ranging from 0.5% to 10%, the wall lifetime (defined as the time for a 20% loss of thickness by a 1-cm-thick Nb wall) ranges from 0.18 to 3.7 years.

McCracken reported some sputtering yields, S, obtained recently by Summers, Freeman, and Daly[4] for the sputtering of polycrystalline Nb by deuterons, by protons, by helium ions, and by niobium ions. Typical yield values are S = 0.0013 atom/ion for ∿ 56-keV deuterons, 0.0015 atom/ion for ∿ 28-keV protons, 0.040 atom/ion for ∿ 57.5-keV helium ions, and 2.4 atom/ion for ∿ 20-keV niobium ions. These values were obtained under high vacuum conditions and with the niobium target at room temperature. He also reported on some observations made by Summers et al. on the dependence of the sputtering yields on the angle of incidence, θ, for the case of deuterons and helium ions impinging on Nb. Their observation that $S \propto (\cos\theta)^{-1}$ is in agreement with the results of other authors studying other systems.

Kaminsky presented estimates[5] for the sputtering yields of Nb, V, and Mo under the impact of 14-MeV neutrons and for H^+, D^+, and He^+ ions with energies ranging from 0.1-3.5 MeV. The estimates indicate that the sputtering of walls by 14-MeV neutrons could seriously limit the lifetimes of reactor walls. Reliable experimental results on 14-MeV neutron sputtering of such materials as Nb, V, and Mo are badly needed.

2. Chemical Sputtering

D. Gruen[6] discussed the possible emission of heteronuclear molecules in a chemical sputtering process during the operation of a thermonuclear reactor. He considered specifically the formation and ejection of metal hydride projectiles with atoms of a metal surface, (2) reaction between surface atoms and the hydrogen diffusion from the bulk material to the surface, and (3) the ejection of molecules originally formed as metal-hydrogen adatom complexes. Considerations of the binding energy for hydrogen on the metal surface, the dissociation energy of the diatomic metal hydride molecule, and the energy of an ejected metal hydride molecule suggests favorable conditions for the emission of stable metal hydrides which could be of interest to the controlled thermonuclear reactor program. An experimental program to study the emission of such chemically sputtered metal hydride molecules has been started by Gruen.

3. Radiation Blistering

Behrisch discussed studies of the radiation blistering of polycrystalline copper under bombardment by energetic protons (e.g., at 100 keV). This work, which he had done in collaboration with W. Heiland, revealed that large blisters were produced. He suggested that the blistering effect might perhaps be reduced by choosing materials in which hydrogen (or deuterium) is very soluble and by operating the bombarded walls at elevated temperatures.

Kaminsky reported some preliminary experimental results on radiation blistering of polycrystalline and monocrystalline niobium (at room temperature) by 125-keV deuterons for a dose density of 0.1 Coulomb/cm^2. No bubbles with diameters as large as $\sim$ 500 Å (the limit of resolution of the metallograph employed) were found. At least for this dose density of 125-keV deuterons, therefore, the blistering is less severe in Nb than in Cu—as discussed in more detail in Ref. 5. For helium bombardment of niobium, however, the formation of bubbles may still be a severe problem.

4. Surface Damage

Field-emission-microscope studies of the structure of surfaces after ion bombardment were described by H. Vernickel.[7] Materials such as Nb, Mo, and W were bombarded by argon ions with energies ranging from 0.5 to 5.0 keV with the dose densities up to 10^{16} ions/cm^2. As the temperature, T, of the bombarded sample was varied from 100 to $\sim$ 1000°K, the effect on the surface was found to depend on the relation between T and the melting temperature, T_m, of the target material. For $T < 0.1\ T_m$, a roughening of the bombarded surface was observed. For $0.10\ T_m < T < 0.33\ T_m$, certain specific surface structures appear. For $T < 0.33\ T_m$, the absence of any observable changes in surface structure suggests that the surface damage caused by argon impact is annealed as soon as it is produced. These observations made with low energy argon ions do not exclude the possibility of surface roughening (including blistering) by the impact of very energetic light projectiles such as D^+, T^+, or He^+.

Changes in the structure of thin ($\sim$ 1000 Å thick) vapor-deposited niobium foils under impact of Nb^+ ions were studied by P. Mohr.[8] In his experiments, the energies of the Nb^+ ions ranged from 80 to 100 keV, the ion dose densities ranged from 10^{13} to 10^{17} ions/cm^2, and the foil temperatures were varied from 25 to 850°C. For the foils bombarded at the lower foil temperatures (T < 600°C), transmission electron microscopy reveals a high density of small (< 100 Å) features which Mohr believes to be voids. At higher foil temperatures (> 600°C), the density of damaged regions is reduced. Considerable foil growth was observed after Nb^+ ion bombardment at all foil temperatures.

5. Particle Entrapment and Re-emission; Divertor Wall Problems

Studies of ion burial and thermal release of helium from a niobium monocrystal were reported by E. E. Donaldson.[9] In one type of experiment, the (100) face of a Nb monocrystal was bombarded by 1.40-keV He^+ ions for dose densities ranging from 4.80×10^{12} to 1.61×10^{14} ions/cm^2. Subsequently the thermal release of helium was determined for a target temperatures ranging from approximately room temperature to 1600°K. The helium re-emission rate was found to pass through several maxima at different target temperatures. For example, the highest re-emission rate was found to be near 1000°K if the ion dose density was 1.61×10^{14} ions/cm^2, and other pronounced maxima occurred at $\sim$ 600°K, $\sim$ 700°K, $\sim$ 750°K, $\sim$ 825°K, $\sim$920°K, and 1110°K.

The burial and thermal release was also studied for a plastically deformed niobium monocrystal. For the case of 1-keV He^+ ion bombardment of a (100) plane of a niobium monocrystal before and after stressing, it was observed that the deformation altered only two of the twelve peaks usually observed in the re-emission rate spectrum.

The trapping of energetic hydrogen and deuterium ions in such metals as niobium, titanium, zirconium, erbium, and lithium was reviewed by G. M. McCracken.[3] His investigations reveal that the trapping efficiency at high target temperatures is largely determined by the heat of solution of hydrogen or deuterium in the particular target metal. For example, for niobium (in which the heat of solution of hydrogen is 16 kcal/mole) the hydrogen trapping efficiency is greatly reduced for temperatures larger than 600°K, while for erbium (in which the heat of solution of hydrogen is as large as 55 kcal/mole) the trapping efficiency is near maximum for temperatures between 600° and 800°K and is greatly reduced for temperatures larger than 1000°K.

A decrease in the trapping efficiency was also observed for large doses at low temperatures. For example, for a total dose density of 10^{18} ions/cm^2 and a target temperature of 77°K, the trapping efficiency in all metals studied by McCracken had decreased to the order of 10% of the maximum trapping efficiency. This later observation can possibly be explained by the fact that the hydrogen diffusion rate in the metal is low at this low temperature (77°K), and that the trapped ions therefore accumulate and their greatly increased concentration correspondingly increases the probability that a trapped ion will be ejected by a subsequent incident ion.

In collaboration with S. K. Erents and P. Goldsmith, McCracken has also studied the trapping of 18-keV deuterium projectiles in solid and liquid lithium targets over the temperature range from 320° to 730°K. They observed trapping efficiencies of 97% in liquid Li for dose densities as high as 10^{19} ions/cm^2 at temperatures up to 700°K. At higher temperatures, the dissociation of lithium deuterides prevented further trapping. In the solid lithium, the trapping efficiencies were in the range from 70 to 95%.

The importance of an effective method of ionizing the neutral gaseous particles (e.g., those that are emitted from the walls as a result of plasma radiations) for subsequent removal by a divertor was discussed by G. Haas.[10] The outer plasma region ("screening layer") can serve as such an effective ionizer if it is sufficiently thick. The thickness depends on various parameters such as the diffusion perpendicular to the magnetic confinement field, the plasma flux in the inner plasma region (parallel to the magnetic confinement field), and the effective pumping speed of divertors at different locations along the confining field.

In his theoretical study, Haas finds that the thickness of the screening layer becomes considerably larger if a torsatron type of divertor is used instead of a classical (Stellarator) type. In these considerations, a classical diffusion perpendicular to the magnetic confinement field was assumed.

For dimensions typical of fusion reactors, he finds that approximately 50% of the neutral particles released from the walls can reach the inner plasma region if the screening layer has the small thickness attained with a divertor of the classical type, while no neutral particles will reach this region if the layer has the larger thickness found when the divertor is of the torsatron type.

6. Backscattering of Particles

If light projectiles such as protons and deuterons with energies of several tens of keV impinge on a solid surface, a significant fraction (of the order of several per cent) can be backscattered. Studies of the energy spectra and angular distributions of protons backscattered from both polycrystalline and monocrystalline copper and from nickel were reported by R. Behrisch.[1] The primary energies ranged from 40 to 120 keV. In some instances the observed energy spectra revealed the presence of such surface impurities as C and O. Studies of the backscattering of protons and deuterons (with primary energies ranging from 20 to 60 keV) from Nb and Ti which have been reported previously by McCracken et al.[11] were also reviewed. The observed energy distributions of the backscattered projectiles were compared with energies that had been calculated on the basis of a single-scattering model. The observed energies were found to be larger than the calculated ones, a finding which in part is attributed to the occurrence of multiple-scattering events.

7. Desorption by Electron Impact

Studies of gas release from solid surfaces under the impact of sufficiently energetic electrons were reviewed by E. E. Donaldson[9] and J. Peavey.[12]

Donaldson, who worked in collaboration with I. Newsham, J. Hogue, and D. Sandstrom, described the use of a time-of-flight technique to study the release of excited neutrals and ions from CO-covered monocrystalline and polycrystalline tungsten surfaces under electron impact. For the case of neutral CO molecules released from a (100) surface of a W monocrystal, for example, the time-of-flight distribution showed that the mean energy of the molecules was $\sim$ 3 eV. Furthermore, they observed that more neutrals left in a metastable state than in the ground state. They are now conducting similar experiments on niobium surfaces.

Peavey, who had worked in collaboration with D. Lichtman, described mass-spectrometric studies of the species and the charge states of gaseous particles desorbed from stainless steel and molybdenum under electron impact. A typical electron energy was 100 eV. They observed that neutral desorbed paticles were more abundant than charged ones. For both neutrals and ions, the desorption efficiency rose sharply as the electron energy exceeded a threshold value (typically between 12 and 20 eV), reached a maximum in the neighborhood of 100 eV, and then decreased slowly as the electron energy increased further. For oxygen-covered polycrystalline Mo surfaces under 100-eV electron bombardment, for example, desorption efficiencies for O^+ reached their maxima at 70 - 150 eV. The desorption efficiency, R, can have rather high values; e.g., $R \approx 1.5 \times 10^{-3}$ desorbed molecule per incident electron for CO desorbed from polycrystalline Mo surfaces by 100-eV electrons.

B. PHENOMENA INDUCED BY PHOTON IMPACT

1. Surface Heating by Energetic Photon Absorption

The absorption of Bremsstrahlung (e.g., in the x-ray region from 0.01 to approximately 100 Å) in various materials was discussed by J. A. Phillips,[13] who calculated the mean absorption depth associated with the

1/e photon energy attenuation. He found, for example, that photons with a wavelength of about 1 Å are practically completely absorbed in less than 1 mm of materials such as Mo, Cu, Al_2O_3, and Al, while for Be the required thickness is about 1 cm. (Note: For a hydrogen-isotope plasma with an electron temperature $T_e \approx 13.6$ keV, the wavelength at the maximum of the Bremsstrahlung spectrum is near 2 Å.)

The increase of the surface temperature of a wall due to absorption of Bremsstrahlung was calculated under the assumptions of complete absorption of the incident photons and a uniform power density, W (W/cm^2), during the photon pulse duration, τ. The surface temperature, T(°C), at any time, t(sec), during the pulse was approximated by

$$T = T_o + \frac{W}{2.1}\left(\frac{t}{\pi \kappa \rho c}\right)^{1/2} ,$$

where T_o is the initial surface temperature (°C) prior to photon bombardment and k, ρ, and c are the thermal conductivity (cal cm^{-1} sec^{-1} °C^{-1}), the density (g/cm^3), and the specific heat (cal cm^{-3} °C^{-1}), respectively. Phillips presented some calculated values of the temperature rise $\Delta T \equiv T - T_o$ for materials such as Al_2O_3 (1 mm thick), Ni (1 mm thick), Mo (2 mm thick) and Cu (2 mm thick). It was assumed that the power density was 10 kW/cm^2 and that the temperature of the outside surface of the wall was kept constant at T_o. He found that after $\sim$ 10 msec the values were ΔT = 307°C for Cu, 572°C for Mo, 715°C for Ni, and 1132°C for Al_2O_3.

Such fast temperature rises can be expected to occur in the first wall of a pulsed fusion reactor using the z-pinch, for which pulses of 0.01 sec duration may be repeated at intervals of 0.10 sec.

He also calculated the temperature of the inner surface of a 1-cm-thick copper wall of a pulsed fusion reactor when it was bombarded with photons with a uniform power density of 10 kW/cm^2 and the outer surface was held at 400°C. He found that each photon pulse (pulse duration = 0.01 sec, pulse period = 0.10 sec) resulted in a sharp thermal spike (a sharp increase in temperature, followed by a sharp decrease). The maximum temperature, T_{sp}, attained in the spike initially increased in successive pulses but then approached an equilibrium value; for example, $T_{sp} \approx 700$°C in the first pulse and $\sim$ 880°C in the tenth. The temperature, T, of the inner surface of the wall midway between pulses also increased gradually and then leveled off; for example, the 10 pulses incident in $\sim$ 1 second resulted in a temperature rise ΔT = 250°C. The thermal spikes are superimposed on this gradual change in the longer-term surface temperature.

The high temperatures reached in the thermal spikes can cause severe evaporation of the wall material, and the high temperature gradients existing in the layers near the surface can result in severe strains leading in turn to cracking of the bulk material and flaking of the surface.

2. Photon-Impact-Induced Desorption

Mass-spectrometric studies of photon-induced desorption, performed in collaboration with D. Lichtman, were discussed by Peavey.[12] Before the photon bombardment, the tungsten target was outgassed by ohmic heating and then exposed to the residual gases of the vacuum system (total pressure $\approx 1 \times 10^{-9}$ Torr) for more than 24 hours. The photon source was the synchrotron radiation from a 240-MeV electron storage ring. Only photons in the wavelength range from 400 to 3400 Å are thought to have contributed to the desorption process; photons with λ < 400 Å could not reach the target, while those with λ > 3400 Å are believed to be unable to cause photodesorption. The total photon flux in the 400 - 3400 Å region was estimated to be about 4.8×10^{12} photons cm^{-2} sec^{-1}. On the assumption that the average wavelength is 2000 Å, this corresponds to a power density of 4.7 $\mu W/cm^2$.

The photodesorption of CO_2, CO, and H_2 from the tungsten surface was observed, but photodesorbed argon and water could not be detected. The cross section for photon-induced desorption was crudely estimated to be $\sim 7.8 \times 10^{-21}$ cm^{-2}.

C. THERMAL PHENOMENA

The heating of components of an operating fusion reactor by irradiation with energetic particles and photons from the plasma and by secondary radiations can cause serious plasma contamination and wall erosion by several mechanisms. Such mechanisms include thermal evaporation and desorption, thermal emission of electrons and ions, and whisker growth. The heating can also seriously affect the chemical and mechanical proper ties of the heated component and tends to enhance secondary-particle yields and gas permeation.

1 . Evaporation

Phillips[13] presented calculations on the heating of an aluminum wall which had been irradiated for 1 sec with photons with a power density of 100 W/cm^2. During this 1-sec irradiation, the surface temperature would increase from the assumed initial value of $\sim$ 766°K to $\sim$ 790°K. At the latter temperature, the aluminum evaporates at a rate of 3×10^{13} particles cm^{-2} sec^{-1}.

In this connection, it is of interest to note that vaporizing the wall material at such a relatively high rate could lead to serious Bremsstrahlung losses. For a hypothetical fusion reactor vessel whose volume is of 100 m^3 and whose inner wall has a total surface area of 500 m^2, for example, a flux of vaporized wall material of 3×10^{13} particles cm^{-2} sec^{-1} could lead within 1 sec to an impurity-particle density $n_1 = 1.5 \times 10^{12}$ particles/cm^3. If one assumes a fuel-particle density $n_2 = 1.5 \times 10^{14}$ particles/cm^3, the impurity fraction, f, is $n_1/n_2 = 1 \times 10^{-2}$. This value is considerable larger than the maximum permissible impurity fraction $f_{max} = 4.5 \times 10^{-4}$ for aluminum (which can be read from Fig. 2 in Ref. 5).

Kaminsky discussed the vaporization of vanadium. (A more complete report can be found in Ref. 5.) For example, at 1092°C vanadium vaporizes at a rate of 1.32×10^{11} particles cm^{-2} sec^{-1}. If one assumes the dimensions of the above-mentioned hypothetical fusion reactor, one finds that within 1 sec the density of the vanadium impurity has increased to $n_1 = 3.4 \times 10^{10}$ particles/cm^3; again the fuel particle density n_2 is assumed to be $n_2 = 1.5 \times 10^{14}$ particles/cm^3. These values lead to an impurity fraction of 2.3×10^{-4}, while the maximum permissible impurity fraction for vanadium is $f_{max} \approx 1.9 \times 10^{-4}$.

2. Thermal Desorption of Neutral and Charged Gas Species

The desorption of gas from a sufficiently heated surface can lead to plasma contamination.

Stickney,[14] whose collaborators were A. E. Dabiri, T. L. Bradley, and T. E. Kenney, described experimental studies of the spatial and speed disbributions of H_2 desorbed from monocrystalline and polycrystalline nickel surfaces. The presence of surface impurities (predominantly S and C, as determined by Auger spectroscopy) on some of his nickel surfaces influenced significantly the spatial distribution of the desorbed H_2 molecules. For nickel surfaces contaminated with S and C, for example, his data on the spatial distribution of H_2 desorbed at surface temperatures ranging from 800 to 1300°K indicate that the distribution function is quite accurately proportional to $\cos^d\theta$, with $d \stackrel{\sim}{\sim} 4$. The value of d decreases (i.e., the spatial distribution becomes broader) as the concentrations of S and C are decreased by various surface cleaning procedures; for example, if the initial S concentration was reduced by a factor of 5, the value of d changed from 4 to 3. In contrast, when carbon was deposited on the nickel surface until the Auger peaks of nickel were undetectable, a value $d \stackrel{\sim}{\sim} 1$ was observed. The influence of surface impurities on the spatial distribution of desorbed H_2 molecules is similar for polycrystalline and monocrystalline nickel surfaces.

Dresser,[15] working in collaboration with L. Johnson and E. E. Donaldson, had obtained experimental results on the adsorption and absorption of hydrogen by niobium. For a clean niobium surface, the sticking coefficient was found to be 0.13. Studies of the sticking coefficient's dependence on the surface temperature and surface coverage reveal, for example, that for a coverage of 7×10^{13} molecules/cm^2 and for a surface temperature of 925°K the sticking coefficient is still as high as ~ 0.1. If the surface coverage was varied from 1.5×10^{14} to 2.7×10^{15} molecules/cm^2 at a surface temperature of 517°K, the value of the sticking coefficient decreased only slightly, from approximately 0.13 to 0.10. However, at a surface temperature of 925°K, the sticking coefficient dropped from 0.1 at a coverage of $\sim 7 \times 10^{13}$ atoms/cm^2 to 0.001 at a coverage of $\sim 1.2 \times 10^{14}$ atoms/cm^2.

Evidence for a precursor surface adsorption state and subsequent diffusion into the bulk material was found for the hydrogen-niobium system. Isothermal plots of the concentration of dissolved hydrogen as a function of the square root of the equilibrium pressure were obtained at 360, 388, 436, 543, 665, 801, and 923°K. These plots were found to be linear at concentrations below 1 atom % in agreement with Sievert's law; but anomalies were found at concentrations above this. The plot of the electrical resistance of niobium versus the concentration of dissolved hydrogen was found to depend on time and temperature. For a piece of niobium held at 443°K for $\sim$ 200 hours, for example, the hydrogen uptake was 4.3 atom % and the resistance decreased from the initial 0.625 Ω to $\sim$ 0.565 Ω.

Dresser reported also on studies of the thermal emission of sodium and potassium occurring as impurities in niobium samples. If the niobium surfaces were heated to a sufficiently high temperature, some of the emitted sodium and potassium were released as ions by the surface ionization mechanism. A characteristic diffusion time for each impurity was measured at a series of constant temperatures in the range from 1200 to 1800°K, and an activation energy, E, for the diffusion process was inferred from the slope of an Arrhenius plot. With a transient method, the activation energies obtained were E(Na) = 0.7 eV and E(K) = 1.3 eV; a steady-state ion-emission method yielded E(Na) = 0.9 $\pm$ 0.2 eV and E(K) = 2.1 $\pm$ 0.3 eV. Dresser suggests that the difference may be due to the thermal dissociation of alkali metal compounds, which would cause the apparent values obtained by the transient method to be smaller.

3. Effects on Chemical and Mechanical Properties (e.g., Surface Embrittlement, Strains, and Flaking

Phillips[13] discussed studies of the suitability of an all-metal or all-insulator discharge chamber for use in the fast toroidal z-pinch experiment at Los Alamos. Several important criteria for the fabrication and operation of the chamber must be satisfied--e.g., the material must have a high melting point, good vacuum properties (e.g., low outgassing rate), machinability to small tolerances, an ability to retain high electric fields (necessary to maintain the axial electric field in a z pinch), and good thermal conductivity. He believed that an all-metal chamber would satisfy more of these requirements than an all-insulator chamber. The difficulty of reconciling the requirement of good thermal conductivity with the ability to build up high electric fields ($\sim$1 kV/cm) at the surface was partly overcome by choosing aluminum with an anodized surface.

Tests were made to compare the performance of an all-insulator tube (high-purity alumina) with that of an all-metal tube (aluminum with certain anodized surfaces) in a pinch discharge. The aluminum tube consisted of 800 aluminum washers 0.010-in. thick stacked on top of each other. The flat surfaces of each washer were anodized and 300 V could be applied between adjacent washers without breakdown.

The experiment was performed with D_2 gas at partial pressures ranging from 25 to 50 mTorr, discharge voltages up to 30 kV, peak discharge current up to 0.45 mA, and discharge duration of 15 μsec. After several hundred discharges, the surfaces of the ceramic and the metal tubes were compared. It was found that a fine-grained brownish-white powder (identified as α-phase Al_2O_3) had flaked off from the alumina tube and covered a lower electrode. Furthermore, it appeared that small chips had been plucked out of the surface during the discharge. No such debris was observed with the aluminum tube. However, small local heating spots, surface pits, and a discoloration of the electrodes were observed; a study of the bombarded aluminum surface with a scanning electron microscope revealed small, nearly spherical protrusions. It is thought that most of these spherical protrusions ("droplets") were formed by melting projections that existed on the unbombarded surface.

4. Whisker Growth

Studies of the formation and growths of whiskers (protrusions) from heated surfaces of solids were reviewed by R. Vanselow.[16] Whisker formation can lead to unwanted surface roughening and thereby affect such properties as photon reflection coefficients and the yields of various particle-emission processes (e.g., sputtering, desorption, and secondary-electron emission).

In studies of the oxidation of Nb to form Nb_2O_5 by transmission electron microscopy, the formation of large numbers of long whiskers at surface temperatures ranging from 800 to 900°C and for oxygen partial pressures of 10^{-2} Torr have been observed[17] by transmission electron microscopy. In another study, field-emission-microscope emitters made out of Ta were heated at 800°C at a hydrogen partial pressure of $\sim 10^{-3}$ Torr. It was observed[18] that small crystallites were found and grew with increasing surface temperature but disappeared above 1250°K. The application of an external electric field did not seem to influence the crystallite formation of this H_2-Ta system. Vanselow et al.[19] exposed a molybdenum field-emission-microscope emitter to water vapor at a partial pressure of 1×10^{-4} Torr and heated it to temperatures ranging from 500 to 900°C. Under these conditions they observed the formation of molybdenum oxide whiskers. If the surface temperatures were increased further, the whiskers disappeared.

5. Solubility, Diffusivity, and Permeability

For the design of a fusion reactor, it is important to be able to estimate the amount of fuel (e.g., deuterium or tritium) that may get lost by permeation through the hot plasma containment wall. If it is assumed that the permeation rate is controlled by diffusion and not by surface reactions, and that outside of the containment wall the partial pressure of the permeating gas is zero, the permeation rate, Q, per unit time is given by

$$Q = (F/d)D\rho_S,$$

where F is the wall area, d the wall thickness, D is the diffusion coefficient of the gas in the metal, and ρ_S is the equilibrium density of the gas in the wall material for a given pressure.

Stickney[14] gave a brief summary of some experimental data and theoretical models pertaining to the solubility, diffusitivity, and permeability of hydrogen-isotope gases in metals. He emphasized that surface processes can significantly affect permeation rates.

The permeation rate of deuterium through a 1-cm-thick niobium wall at 800°C for a D_2 partial pressure of 10^{-2} Torr is estimated[20] to have the high value $Q = 0.2 \times 10^{-4}$ cm^3 gas (STP) per cm^2 and per sec. On the basis of this value, Völkl[20] estimated that for a vessel diameter of 2 m, one whole gas charge would be lost through the niobium wall in 8 sec.

D. GENERAL DISCUSSION

There was general agreement that the present lack of sufficient data precludes any meaningful prediction of how the problems of plasma contamination and wall erosion may be accentuated by the combined action of simultaneously occurring surface phenomena resulting from plasma radiations. However, several participants expressed suspicion that the combained radition effects may cause the various erosion processes to become nonlinear. No specific suggestions were made about the type of surface studies that should be undertaken to provide answers for ongoing work with plasma devices. Several people suggested that in the near future more of the surface studies should be directed towards these processes that are common to various types of fusion reactors and fuel cycles.

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SESSION 5

RADIATION DAMAGE

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STATE-OF-THE ART
PRESENTATION

RADIATION DAMAGE IN CTR'S

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INTRODUCTION

This and the accompanying papers on radiation damage will explore potential radiation damage problems in the first (vacuum) wall of a controlled thermonuclear reactor (CTR), examine the relevant data that has resulted from radiation damage programs in support of both fission reactor technology and fundamental solid state physics, and discuss some of the main experimental approaches that can be used to evaluate CTR radiation damage. This paper reviews the expected damage in a CTR first wall and briefly surveys much of the available fission reactor experimental results on bcc refractory metals. (The implicit assumption is that the first wall will be made from an alloy based on a bcc refractory metal. The choice seems to be restricted to those alloy systems based on niobium, vanadium, and molybdenum.) There is a large body of data on irradiation of these materials at temperatures too low for the CTR application. These data will not be reviewed here. The discussion of damage will be separated into two broad topics; component swelling and loss of ductility. Loss of ductility discussion will also include irradiation-produced changes in the strength properties.

The irradiation produced swelling will be important because the resulting changes in linear dimensions must be accommodated between the vacuum wall (which will be dimensionally unstable) and the connected vacuum, fuel handling, heat exchange, magnet and support systems. The swelling importance will be magnified greatly if gradients in neutron flux and/or temperature are present over the first wall. Such gradients result in different swelling rates for different segments of the wall and produce stresses that must be relieved by deformation of the wall material. The wall ductility during reactor operation must remain adequate for deformation to occur to accommodate stresses introduced by swelling. The stresses introduced by routine reactor operation, such as normal thermal cycling, will also require some deformation of the wall material. Successful reactor operation will require a wall material that retains at least limited ability to deform without fracture after several years of bombardment by the intense neutron flux of the CTR environment.

The discussion of the ductility loss produced by irradiation is supplemented by the following paper, "Irradiation Embrittlement in the BCC Metals" by J. Moteff. Following this, the paper "A Means of Studying Radiation-Controlled Creep in Refractory Metals for Fusion Reactor Design" by S. D. Harkness discusses the effects of irradiation on creep in the radiation field expected at a CTR wall and introduces one of the several simulation techniques to be applied to the investigation of CTR radiation damage problems. "Use of High Energy Charged Particle Bombardment to Simulate High Fluence Neutron Damage in CTR Materials" by G. L. Kulcinski discusses the advantages and limitations of ion bombardment in evaluating microstructural changes to be expected in CTR service and presents some examples of the use of this technique. Another example of the use of ion damage is "Ion Bombardment Simulation of 14 MeV Neutron Damage in Thin Niobium Films" by P. B. Mohr. "Helium Injection by The Tritium Trick" by W. V. Green, E. G. Zukas and D. T. Eash outlines a new method of introducing high helium concentrations into refractory bcc metals to simulate what is believed to be the most important of the transmutation products expected in the CTR wall.

Finally, two outlines of possible accelerator-connected sources of neutrons for possible use in radiation damage experiments, "LAMPF as a Neutron Source for Radiation Damage Experiments" by W. V. Green, D. Dudziak and E. Zukas and "Facility for Duplicating 14 MeV Neutron Effects in Fusion Power Reactors" by Harry Dreicer and Dale B. Henderson, complete this series of papers on the state-of-the-art of radiation damage.

THE IRRADIATION ENVIRONMENT AND DAMAGE PRODUCTION IN A CTR

The neutron flux spectrum expected at the first wall of a CTR is shown in Fig. 1, where it is compared directly with the spectrum of a fast fission reactor (EBR-II) and two thermal fission research reactors

(HFIR and ORR). The magnitude of the CTR flux may be too great by approximately 40%, according to a recent estimate,[1] but the important feature still remains that the average energy of the neutrons in the CTR is much higher than in the fission reactors.

The CTR flux is more damaging than an equivalent fission reactor flux, both in terms of the production of transmutation products by inelastic processes and in terms of the production of displacement damage by elastic scattering processes. The transmutation reactions, dominated by the highest energy neutrons, produce hydrogen, helium, and elements with atomic numbers near that of the target metal. Of these impurities produced in the wall material, helium is believed to be the most damaging; and this will be discussed in some detail below. The amounts of transmutation products produced per year of operation in a CTR with a total neutron flux of 3.7×10^{15} neutrons/(cm^2sec) are given in Table 1, for first walls made of niobium[1] and vanadium.[2] Transmutation rates for a molybdenum first wall are not given because the necessary cross sections are not as well defined as in the other two metals. Rates for molybdenum would probably be similar to those for niobium. For comparison, the rates of hydrogen and helium production in EBR-II are also given in Table 1. Note that in the values quoted no provision has been made for the effect of alloying elements on these transmutation production rates.

Displacement damage in a metal is created when a neutron strikes a lattice atom and displaces it from its equilibrium position. The unoccupied lattice position is a vacancy; and the displaced atom, when it comes to rest at a non-lattice position, is called an interstitial. The damage is further multiplied when the neutron transfers a large amount of energy to the struck atom and it in turn displaces many other atoms before itself coming to rest. At the temperatures at which a CTR first wall will operate, both vacancies and interstitials will be mobile. Most of the defects produced will be lost by annihilation at sinks or by recombination, but the few that survive in stable defect aggregates produce technologically important property changes. The nature of the stable damage produced under expected CTR irradiation conditions will be discussed below. Detailed calculations of damage rates under neutron irradiation have been made. The results have been evaluated by Robinson[4] and Martin[5] in comparing the damage effectiveness of 14 MeV neutrons compared to neutrons

TABLE 1

REACTOR TRANSMUTATION RATES

Metal		Helium	Hydrogen	Solid Product	Notes
		(ppm/yr)	(ppm/yr)	(%/yr)	
Nb	CTR Wall	270	890	1.4 (Zr,Mo,Y)	(a)
Nb	EBR-II Core	4	180	-	(b)(c)
V	CTR Wall	790	1500	∿ 0.7 (Cr,Ti)	(d)
V	EBR-II Core	3.5	100	-	(b)(c)

(a) Data from Steiner (Ref. 1,2). Total CTR flux 3.7 x 10^{15} neutrons/ (cm^2sec). Solid products initially 95% Zr, at end of 20-year life Approximately 66% Zr, 33% Mo.

(b) Cross sections from Alter and Weber (Ref. 3). Total EBR-II core flux taken as 3 x 10^{15} neutrons/(cm^2sec), 100% plant factor assumed.

(c) He production in EBR-II would be higher than value given because of contribution from nitrogen impurity content of typical Nb or V samples.

(d) Approximate values from Steiner (Ref. 2). Solid products 90% Cr.

of lower energies. These results, necessary to make a comparison of fission reactor data with expected results in CTR irradiation, show that in niobium a 14 MeV neutron is 2.5 to 4 times as damaging as a 1 MeV neutron. The actual ratio of damage production rates in a CTR first wall and a fission reactor will differ from these estimates when the effects of the whole neutron spectrum in each reactor is considered. The displacement rate expected in a niobium CTR first wall has been calculated[1] to be 210 displacements per atom[a] per year for a total first wall flux of 3.7×10^{15} neutrons/(cm^2sec).

The high mobility of individual vacancies and interstitials at CTR operating temperatures results in most of these point defects being annihilated soon after they are created. A small fraction of the defects created, however, are retained by forming stable vacancy or interstitial clusters. It is these stable defect configurations that will influence the properties of the CTR vacuum wall. Mobile interstitial atoms precipitate in a two-dimensional morphology, in the form of disks or partial extra atom planes bounded by dislocations. An interstitial dislocation loop is illustrated schematically in Fig. 2A. Vacancies can precipitate in two morphologies, shown in Figs. 2B and 2C. Under some conditions, including at least the lowest temperatures at which vacancies are mobile, the vacancies form dislocation loops, as illustrated in Fig. 2B. These loops are the vacancy analogue of the interstitial dislocation loop, and are a partial plane in the lattice with all atoms missing. At somewhat higher temperatures vacancies can precipitate into a three-dimensional morphology, the cavity shown in Fig. 2C. Two special classes of cavities must be considered in evaluating CTR radiation damage. Cavities that are essentially empty, formed by the precipitation of vacancies alone, are referred to as voids. Cavities that form by precipitation of both vacancies and insoluble gases (e.g., helium) are called bubbles and can exist in equilibrium with an internal pressure, P, given by

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[a]Displacements per atom is used as a measure of the displacement lattice damage. It represents the number of times, on the average, that each atom is displaced from its equilibrium position.

$$P = \frac{2\delta}{r}$$

where δ = surface tension of the metal and r = bubble radius. As a result of the internal pressure, voids and bubbles can be distinguished by their post-irradiation annealing behavior, with bubbles growing and voids shrinking on high temperature annealing. The two species can often also be separated without annealing by differences in their nucleation behavior. Examples of the two main components of irradiation damage seen in bcc metals examined by transmission electron microscopy are shown in Fig. 3. In this molybdenum sample, viewing under absorption contrast conditions, Fig. 3A, reveals a high concentration of voids but leaves any dislocation component of the damage nearly invisible. The voids seen in Fig. 3A are ordered on a superlattice parallel to the metal lattice, a feature observed in several bcc metals.[6] The same sample imaged under diffraction contrast conditions is shown in Fig. 3B. The most visible component of the damage is now the dislocation loops and dislocation segments that are believed to result from the growth and interaction of loops. Both components of damage occur throughout the sample and, of course, the number of vacancies contained in the voids will be in approximate balance with the number of interstitials that have precipitated to produce the dislocation loop-segment structure, so long as the loop structure is the main interstitial sink.

The service requirements of the first wall will, to a large extent, determine both the type and concentration of radiation damage to be expected in the wall material. By far the most important of these parameters (aside from the neutron flux) is the operating temperature. The range of possible operating temperatures that has been suggested is 500 to 1200°C, with a more likely range for the eventual choice being 600 to 1000°C. The choice of reactor coolants, with helium, lithium, potassium, and molten salts having been suggested, will influence the choice of materials for the first wall but outside of this will have little or no effect on the irradiation damage processes. The tritium concentrations maintained in the CTR metal components could influence the irradiation damage, but there are neither models nor data to establish any possible effect.

Radiation damage to metal components of a CTR is dependent on reactor model only in as far as the model choice determines the values of the parameters discussed in this section. There is no intrinsic dependence of radiation damage on reactor choice.

IRRADIATION PRODUCED SWELLING

Swelling in metals will occur both as a result of the transmutation component and the displacement component of the CTR first wall radiation damage. There is some meager information on swelling due to either of these processes separately, but no information on the swelling behavior with the two processes occurring simultaneously.

Swelling due to transmutation reactions is mainly a result of the precipitation in bubbles of the helium from α-producing reactions. Some swelling, additive to that produced by the other mechanisms discussed, results from the solid transmutation products. This effect is due to the slightly different lattice parameter of the alloy compared to the material before transmutations. For example, transmutation of 10% of a niobium wall to zirconium (about seven years of operation--see Table 1) will produce about a 2% increase in the volume of the wall material. On the other hand, transmutation of 10% of a vanadium wall to chromium (about 14 years of operation) will result in a volume decrease of about 2% in the wall material. Hydrogen produced in (n,p) reactions is not expected to affect the swelling processes because of its high diffusivity and permeability in the bcc refractory metals. Hydrogen produced in the wall material will diffuse to either wall surface, into the plasma cavity or into the blanket coolant, and will end up in the tritium-deuterium handling system a short time after it is created.

Swelling Produced by Helium

Helium in bcc refractory metals is not as innocuous as is hydrogen. It has an extremely low solubility in these, and in fact in all metals. It may also have a relatively low diffusion rate, possibly on the order of the self diffusion rate of the host metal.[5] The result is that the helium precipitates in bubbles satisfying the equilibrium condition

$$P = \frac{2\delta}{r}$$

given in the above discussion of cavities. Martin[5] has discussed in detail the properties and behavior of these equilibrium helium bubbles in niobium under CTR first wall conditions. Martin treats what is assumed to be the "worst case," with all the helium to be considered present at the start of analysis. Due to lower neutron fluxes assumed, the 20-year, end-of-life helium concentration analyzed for is 2750 ppm (2.75×10^{-3} atom fraction). This concentration corresponds approximately to the value after ten years' service in a niobium CTR first wall, using Steiner's parameters,[1] and three to four years' service for a vanadium first wall.

Martin[5] shows that for a fixed amount of gas the equilibrium swelling depends on the bubble diameter and is only weakly dependent on temperature in the range of CTR temperatures. For helium bubbles in niobium with diameters less than ~ 200 Å, the swelling at temperatures between 600 and 1200°C is approximately independent of bubble size and temperature and is set only by the helium content of the metal. For the 2750 ppm helium content assumed, the equilibrium swelling is near 1% for precipitation in bubbles with diameter less than 200 Å. For larger bubbles, the equilibrium swelling does depend on diameter and, less strongly, on temperature. For example, this same helium concentration produces 10% swelling at 1200°C if bubbles of about 2400 Å diameter form, and 10% swelling at 600°C for bubbles of 4000 Å diameter. The total swelling within these limits, then, is set by the number of bubbles into which the available helium precipitates. Martin has considered this, too, and gives a relationship for the number of bubbles per cubic centimeter as a function of bubble size. (Still for the fixed helium concentration, and for a single size of bubbles in each case.) For this case, the approximate upper limit on the bubble size for which swelling is size and temperature independent, 200 Å diameter, corresponds to a bubble concentration of two to three times 10^{15} cm^{-3}. Martin has further estimated the bubble concentration to be expected under CTR conditions in an attempt to better define the expected swelling. The estimates are necessarily very approximate due to a lack of experimental measurements of the parameters in the calculations. They do, however, suggest that the likely swelling due to helium in equilibrium bubbles is 0.6% at 600°C, with a concentration of

1.1×10^{18} bubbles/cm^3 and 1.0% at 1200°C, with a bubble concentration of 9×10^{15} cm^{-3}.

The only set of experimental data which provides an approximate check on Martin's analysis deals with equilibrium helium bubbles observed in molybdenum at several annealing temperatures.[7] Molybdenum and niobium are similar enough that this comparison provides an approximate check on the calculations. Helium was injected by α-ion bombardment at room temperature to an approximate concentration of one atom percent, four times the concentration used in Martin's calculations. Annealing temperatures from 640 to 1100°C gave bubble densities decreasing from 5×10^{17} to 5×10^{16} cm^{-3} at 1100°C. Bubble diameters varied from $\leq$ 20 Å at 640°C to $\sim$ 50 Å at 1100°C. The swelling calculated from these figures is 0.2% at 640°C, increasing to 0.33% at 1100°C. The bubble concentrations and total swelling are lower than predicted by Martin. The possiblity exists that the helium concentrations in the experiments was not as high as reported and also that the short annealing times used did not produce a true equilibrium in the helium bubbles. In any case, the data confirm Martin's calculations of swelling that is small and relatively independent of bubble size and temperature. The data also suggest that if Martin's calculations are in error, they probably predict swelling greater than will occur in practise.

Martin has also considered the effect of Brownian motion and typical CTR stress and temperature gradients on the swelling produced due to equilibrium helium bubbles. He finds that of these only temperature gradients can have an appreciable effect, and this only at operating temperatures of about 1000°C or higher. Making the "worst possible" assumptions, he calculates swelling of 3 and 30% in a temperature gradient of 250°C cm^{-1} at temperatures of 1000 and 1200°C, respectively. Clearly this is a problem that must eventually be investigated experimentally.

This discussion of the properties of helium bubbles has assumed equilibrium conditions, and equilibrium is not expected to prevail during the radiation damage to a CTR wall. This topic will be dropped now while the swelling due to displacement damage alone is considered. Following that discussion, the interaction between transmutation products and displacement damage in producing swelling will be examined briefly.

Swelling Produced by Displacement Damage

Swelling of metals in the approximate temperature range 0.25 to 0.5 T_m (where T_m is the melting temperature of the metal) results from the precipitation of vacancies to form voids, (the second type of cavity discussed above), accompanied by the precipitation of interstitials in dislocation loops. The greatest amount of swelling observed to date in a reactor-irradiated metal is 11% in stainless steel irradiated to 1.45×10^{23} neutrons/cm^2 (> 0.1 MeV) at 410°C. The more restricted data on unalloyed bcc metals at lower fluences show swelling slightly greater than in stainless steel. The maximum amount of swelling reported in a bcc metal is 4.8% volume increase in niobium at 585°C. The available data indicates that the swelling produced by void formation, acting alone, is at least an order of magnitude greater than the swelling produced by the formation of helium bubbles under equilibrium conditions.

The phenomenon of void formation is not well understood. Void nucleation, in particular, has not been adequately modeled theoretically. Nucleation seems to require the presence of some gas atoms in the embryo void to stabilize the three-dimensional void against collapse into a two-dimensional loop.[8] This view is supported by the demonstration in several systems that accelerator-injection of small amounts of helium can increase the void nucleation rate. In reactor irradiation, helium from (n,α) reactions is believed to assist in nucleation; and it is suspected, but not proven, that other gases are instrumental in void nucleation.

The excess of vacancies over interstitials necessary for both nucleation and growth of voids is established and maintained during irradiation by a preferential attraction of dislocation loops for interstitials over the attraction for vacancies.[8]

The general features of void formation as a function of irradiation parameters have been established in several systems. Within the range of temperatures in which voids form the void concentration decreases with increasing irradiation temperature at a fixed neutron fluence. This has been confirmed for several bcc metals by Wiffen[6] and Elen et al.[9] Void concentration can be very high in bcc metals, with concentrations as high as 2×10^{17} voids/cm^3 observed.[6] Void sizes are also dependent on irradiation temperature, with sizes increasing as temperatures increase.

The opposing temperature dependence of void concentration and void size results in a temperature dependence of swelling with a peak intermediate in the void formation temperature range. This temperature dependence of swelling can be partially defined for one bcc metal, niobium, with all available data on swelling and void parameters collected in Table 2. To examine the swelling dependence on irradiation temperature, these data have been extrapolated to a common fluence of 5.0×10^{21} neutrons/cm^2 (> 0.1 MeV), based on the assumption that the volume increase is linearly proportional to fluence. These data are plotted against irradiation temperature in Fig. 4. The data are not consistent at the lower temperatures but do suggest a maximum swelling temperature near 600°C (0.325 T_m) and an upper cutoff temperature near 900°C (0.436 T_m). As discussed below, it is possible that these temperature limits may be fluence dependent. The lower temperature limit for void formation in niobium is not defined by this set of data.

Similarly sparse data for vanadium and molybdenum show the same general trends as are shown for niobium in Table 2. Void concentrations, void sizes, and total swelling are comparable for the three metals and approximately the same dependence on irradiation temperature (expressed as a fraction of melting temperature) is exhibited in all three systems.

The shape and location of the swelling versus irradiation temperature curve will be dependent on the metal, and are probably also dependent on both neutron flux and fluence. The comparison of damage produced by neutron irradiation and by ion bombardment at several different atom displacement rates demonstrates that the whole swelling curve shifts to higher temperatures as the damage rate (flux) is increased.[12] At a constant flux, data on stainless steel[13] show that, in that system at least, the upper temperature for void formation increases with increasing fluence. No metal has been studied in sufficient detail to more fully determine these flux and fluence effects.

TABLE 2

VOID AND SWELLING DATA FOR NEUTRON-IRRADIATED NIOBIUM

	Irradiation		Void Parameters			
Material	Temperature (°C)	Fluence[a] (neutrons/cm^2)	Concentration (voids/cm^3)	Average Diameter (Å)	Volume Fraction (%)	Reference
Niobium	425	2.5×10^{22}	1.6×10^{17}	70	3.1	10
	585	2.5×10^{22}	2.1×10^{17}	71	4.8	10
	790	2.5×10^{22}	2.8×10^{15}	186	1.04	6
Niobium	470	3.9×10^{20}	3×10^{16}	20	0.01	9
	650	5.5×10^{20}	8×10^{15}	60	0.09	9
	750	4.1×10^{20}	5×10^{14}	125	0.05	9
Niobium	600	5×10^{20}	1×10^{16}	50-60	0.1-0.2	11
	900	5×10^{20}	0			11
Nb-1% Zr	790	2.5×10^{22}	$< 2 \times 10^{14}$	510	< 1.4	6

[a]Fluence for neutrons > 0.1 MeV.

The effect of fluence at a fixed irradiation temperature has not been determined for any bcc metal. In fcc metals three different types of fluence dependence have been identified. In one grade of nickel[14] the void concentration was found to be independent of fluence, but the average void size to increase with fluence. In stainless steel[15] the concentration increased with fluence, but the void size increased only very slowly with fluence; in aluminum[16] both the void concentration and size increased with increasing fluence. The aluminum void concentration showed a tendency toward saturation at less than 10^{15} voids/cm^3 at a fluence of 1 x 10^{22} neutrons/cm^2. The very high void concentration observed in bcc metals[6,9] suggests possible saturation in void concentration at low fluences, but this has not been substantiated.

The effect of alloying on void formation, examined in very few bcc metals, is not the same in all bcc systems. For a restricted range of irradiation conditions, it has been shown that high concentrations of small voids form in V, Nb, and Mo while no voids are found in V-3 to 20% Ti (ref. 6,17). Niobium-1% zirconium contained fewer and larger voids than niobium, while total void volumes remained nearly equal.[6] An example of the effect on void formation of starting with Nb-1% Zr instead of Nb is shown in the two microstructures in Fig. 5. The alloy very clearly contains fewer and larger voids than the pure metal. In a comparison of molybdenum-base alloys, one condition was found for which similar void populations formed in molybdenum and Mo-0.5% Ti while at a lower temperature the Mo-0.5% Ti contained more voids than did the molybdenum.[6] In another comparison at lower neutron fluences, TZM was found to be void-free, while molybdenum irradiated under the same conditions contained voids.[18] Impurity content has also been shown to affect void formation in bcc metals. Results on vanadium[6,17] and molybdenum[18] suggest that the void concentration increases as the interstitial impurity content of the sample is increased. These results on impurity and alloying effects are too limited to allow general conclusions to be drawn. The results suggest, however, that interstitial impurities may be influential in the void nucleation step, probably as stabilizing agents in embryo voids. The main effect of alloying is probably related to the effect on the precipitation and distribution of these impurities.

Some control of the amount of swelling produced by void formation may be possible by control of the pre-irradiation microstructure. In stainless steels the very high density of dislocations introduced by cold working has been shown to be effective in suppressing swelling.[15,19] Suppression has also been achieved in a nickel base alloy treated to contain a very high concentration of small precipitate particles.[19] Both of these treatments are believed to be effective because they provide sites at which single defects can be trapped and annihilate with defects of opposite type. To be effective, the trapping sites must be of a high enough concentration that essentially all vacancies as well as interstitials end up on them, even though the interstitial mobility is much higher than vacancy mobility. Similar treatments may be effective in controlling the swelling of bcc metals, and need to be investigated in detail.

Extrapolation of the available results on void-produced swelling to predict the swelling that will occur at the CTR first wall clearly cannot be done with any accuracy. The available models of void swelling have not been sufficiently tested to be used, and the experimental data do not yet define the fluence dependence of swelling for the metals that will be used for a CTR wall. The data in Fig. 4 show a maximum swelling in niobium of approximately 1% near 600 °C for a fluence of 5×10^{21} neutrons/cm^2. If this data is extrapolated assuming the swelling is linearly dependent on fluence, the assumption made in deriving Fig. 4 from the data in Table 2, then a swelling of 100% is predicted for a fluence of 5×10^{23} neutrons/cm^2 (about five years in a CTR). On the other hand, if swelling is assumed to be dependent on fluence to a power somewhat less than first order, the swelling will be less. As an example, Bates and Pitner[20] report a fluence dependence of swelling in tantalum of 0.4. If this same fluence dependence is assumed for niobium, the 4.8% swelling observed at 2.5×10^{22} neutrons/cm^2 extrapolates to 16% swelling at 5×10^{23} neutrons/cm^2.

Prediction of swelling in the CTR first wall will require an early determination of the fluence dependence of swelling. Contrary to the two assumptions above, this dependence will likely be a function of both temperature and fluence. The swelling probably will saturate at some point, but the swelling value at which saturation may occur cannot now be predicted.

Swelling under CTR Irradiation

During CTR irradiation both transmutation and displacement reactions will be occurring. The two effects cannot be separated as they have been in the discussions above. The cavities formed during CTR operation will probably be neither true bubbles nor true voids, in the sense defined earlier. Instead, the cavities formed will likely be void-like, containing some helium at a pressure less than that required for equilibrium. Observed swelling in neutron irradiated niobium and molybdenum[6] exceed either that predicted due to helium in equilibrium in niobium[5] or observed for molybdenum[7] containing 0.27 and 1.0 atom precent helium, respectively. Since the neutron-irradiated material was essentially helium-free, this demonstrates that at least for some possible operating temperatures there will be ample cavity volume to accommodate the helium and the swelling in the CTR first wall will not be governed by equilibrium helium bubble considerations. The helium bubble analysis discussed earlier can thus be regarded as setting a lower limit on the swelling to be expected in practice.

The transmutation products will also complicate the swelling beyond the "pure void" case considered in the previous section. As indicated above, the helium has been found to be instrumental in promoting void nucleation in the several metals in which it has been investigated. Void nucleation is therefore expected to be much more rapid in CTR service than in the fission reactor irradiations discussed above. What effect the helium will have on void growth, and thus on the overall swelling behavior beyond the nucleation stage, is not clear. The solid transmutation products will also have some effect on the swelling. As the metal becomes more highly alloyed by the build-up of transmutation products, the general tendency for alloying to surpress swelling may prevail, and reduce the swelling below that which would be predicted for the starting composition of the first wall. The swelling or shrinkage due to the change in alloy composition will either add to or subtract from swelling from other sources, depending on the wall material used.

In summary, the presently available data on swelling are adequate only to predict some general trends to be expected in CTR first wall swelling. The data are inadequate to predict either the amount of swelling that will occur in service nor are they adequate to define what material compositions and operating temperatures would minimize the swelling. The swelling will be very dependent on the purity, composition, and metallurgical state of the wall materials used. These variables will be manipulated to develop an alloy, probably a very complex alloy, that will give an optimum balance in reducing the effects of the swelling described in this section and the mechanical properties degradation discussed in the following section.

IRRADIATION EFFECTS ON MECHANICAL PROPERTIES

Partly as a result of the aura of newness surrounding the work on irradiation produced swelling, and partly as a result of lack of data, the area of mechanical properties has been relatively neglected in discussion of irradiation effects in the CTR first wall. Among the changes produced in the mechanical properties, it is the loss of ductility which will have the most serious consequences. Very little data are available on mechanical properties of the bcc refractory metals irradiated at temperatures representative of CTR service. Fluences, too, are one to three orders of magnitude below those typical of CTR operation. This section will consider briefly the effect of irradiation on strength properties, ductility reduction by three different mechanisms, and the effect of irradiation on creep during reactor service.

Strength Properties

During irradiation at temperatures below about 0.5 T_m radiation damage, in the form of the defect clusters discussed earlier, accumulates in the metal lattice. In general, this damage builds up more rapidly at lower temperatures than at higher temperatures, and at a fixed temperature the amount of damage increases with increasing fluence. These defect clusters, whether dislocation loops or cavities, provide obstacles to the dislocation motion that is the atomic scale process responsible for plastic deformation. The result is that all measurements of strength properties show increases produced by neutron irradiation. If the strength properties are measured in a tensile test, the yield strength is increased, often to several times the pre-irradiation value. The ultimate tensile strength is also usually increased, but is not affected as much as the yield strength. In a creep test the strength increase produced by irradiation is usually reflected in a decreased minimum creep rate. This effect of irradiation hardening on reducing the minimum creep rate in creep tests is illustrated in Fig. 4 of Moteff's following paper.[21] Recent results on molybdenum[10] illustrate the effect of irradiation on the strength properties observed in tensile tests. Figure 6 of this paper shows two tensile curves of a molybdenum alloy irradiated to a high fluence (by present standards) at a temperature just below the range of CTR interest. The ultimate strength of the irradiated samples is more than twice that of the control material. The ductility effects seen in these tests will be discussed below.

The available data on the mechanical properties of the bcc refractory metals has recently been compiled and summarized by Kangilaski.[22] These data in general show the hardening effects discussed above, but in using this compiled data it must be remembered that the bulk of the data represents the effect of irradiations at low temperatures (typically 50 to 100 °C). The data generally show major changes in properties measured at room temperature but a rapid recovery of the properties if tests are conducted at higher temperatures. These results cannot be used directly to predict behavior under CTR conditions because: (a) damage accumulation in the form of defect clusters is much slower at CTR temperatures than at the lower temperatures, (b) damage produced at higher temperatures is generally stable to higher temperatures, and therefore testing at temperatures above the irradiation temperature can be misleading, and (c) the void component of irradiation damage is only produced at irradiation temperatures above $\sim 0.25\ T_m$ and its effect is thus not evaluated in low temperature irradiations followed by higher temperature tests. The available data are not sufficiently comprehensive to even provide a ranking of the order of severity of irradiation hardening expected in alloys based on V, Nb, and Mo under CTR service conditions.

A more detailed discussion of the mechanisms responsible for irradiation hardening is given in Moteff's paper.[21]

The Ductile-to-Brittle Transition

All of the bcc refractory metals show a transition from low temperature brittle behavior to higher temperature ductile behavior at a temperature known as the ductile-to-brittle transition temperature (DBTT). The DBTT is dependent on the exact condition of the specimen being tested and on the method of test. It is generally near room temperature for molybdenum and below -100°C for niobium and vanadium. The DBTT is usually raised by any process that hardens the metal, and this includes neutron irradiation. It has been demonstrated in molybdenum that low temperature, low fluence neutron irradiation can produce DBTT shifts of up to +120°C (see the data reviewed in reference 22). Data [10] for higher temperature irradiation to 3.0 x 10^{22} neutrons/cm^2 (> 0.1 MeV), represented in Fig. 6, show that fractures may be completely brittle, with zero elongation, to test temperatures at least as high as 550°C. Figure 6 shows that ductility in a tensile test also depends on the rate of loading during test, with slower test rates showing better ductility properties. The fractography included in the figure shows the cleavage fracture mode in the specimen tested at the faster rate. This fracture mode is characteristic of brittle failure of bcc metals at low temperatures.

The available data on irradiated niobium and vanadium do not define a change in DBTT for these metals.[22] While some increase would be anticipated, the low value of the DBTT for the unirradiated material suggests that the effect may be of less importance in these metals than it is in molybdenum.

The Ductility Loss Due to Lattice Hardening

The most commonly observed effect of neutron irradiation on ductility is a continually decreasing ductility with increasing fluence, a ductility decrease that accompanies the strength property increase. An example of the ductility loss in a tensile test on irradiated molybdenum is shown in curve B of Fig. 6. This test, at a lower strain rate than the brittle test shown in curve A, shows zero uniform elongation and less than 10%

total elongation to failure. The fracture surface, shown in the fractograph of Fig. 6, shows the dimple features characteristic of a ductile failure mode. These data are interpreted as showing that the irradiated molybdenum has little or no ability to work harden and fails above the DBTT by local deformation with a very reduced tensile fracture elongation. The effect of higher temperature and higher fluence irradiation on the tensile ductility of molybdenum is unknown. Niobium and Nb-1% Zr irradiated at lower temperatures and lower fluences[23] show behavior similar to the molybdenum sample discussed above. The samples tested at room temperature and above showed limited, non-uniform elongation but did not fail in a brittle mode. There are no data on niobium alloys irradiated at higher temperatures.

Vanadium alloys irradiated at 50 to 100°C to 1.4×10^{21} neutrons/cm^2 (> 0.1 MeV) were tensile tested at room temperature and higher test temperatures.[24] Room temperature tests showed very low values of uniform elongation and reduced total elongation in the range 1.1 to 2.5%, compared to control values of 14.5 to 21.8%. Tests conducted at 650°C showed complete recovery of pre-irradiation properties in some alloys, while test temperatures of 750°C were required for complete recovery in other alloys. These results suggest that the irradiation effects in vanadium and in niobium alloys may be similar. Two vanadium base alloys irradiated at higher temperatures (500 to 660°C) to fluences of 1.6 to 4.9×10^{21} neutrons/cm^2 showed essentially no effect of the irradiation on the tensile ductility measured at room temperature.[25] Similar specimens[26] of V-20% Ti irradiated to fluences up to 3×10^{22} neutrons/cm^2 showed tensile ductilities unaffected by the irradiation for test temperatures in the range 400 to 750°C. These results, and the observed lack of swelling in some vanadium alloys discussed earlier, suggest that vanadium alloys are more resistant to some forms of irradiation damage than are other bcc alloys.

The loss of ductility due to high fluence neutron irradiation has been most extensively studied in the austenitic stainless steels. An example[27] of the stainless steel results is shown in Fig. 7, where the post-irradiation ductility during creep tests of type 304 stainless steel is given as a function of the neutron fluence. This is perhaps one of the most extreme examples of the effect irradiation can have on ductility, with ductility decreasing in a regular manner from 20 to 0.15% as the neutron fluence increases to 6.5×10^{22} neutrons/cm^2. The only results available on the creep ductility of bcc metals irradiated at elevated temperatures provide some hope that ductility will be much less severely affected in bcc metals than it is in the stainless steels. Creep tests on V-20% Ti, irradiated near 600°C to 1 to 2×10^{22} neutrons/cm^2 and creep tested at 650°C showed minimum creep rates and creep elongations near the values for unirradiated specimens.[28] However, this alloy does not form voids for these irradiation conditions.[6] Moteff[21] shows that the elongation in 750°C creep tests of molybdenum is little affected by irradiation to 1.6×10^{20} neutrons/cm^2 at irradiation temperatures of 70, 700, and 1000°C. These same samples did contain voids, with a void concentration of 3×10^{16} cm^{-3} for the 700°C irradiation.[29]

This subject is discussed further in the following paper.[21]

The Ductility Reduction Due to Helium

Helium produced in transmutation reactions has an effect on reducing the ductility of many metals. In fcc metals the helium generally becomes detrimental to ductility at test temperatures above about 0.5 T_m (refs. 30,31). In the only bcc metal (other than commercial steels) in which the effect of helium was studied, a concentration of 1×10^{-6} atom fraction helium introduced by α-implantation was found to reduce the tensile ductility of vanadium alloys at test temperatures above 800°C (about 0.5 T_m) but to have no effect below this temperature.[32] The most severe ductility reduction produced by the helium was in V-20%Ti-10%Nb, where the ductility dropped from 20% to less than 1% in the helium containing specimens. In these vanadium alloys, as in the fcc metals, helium reduces ductility by weakening grain boundaries, resulting in an intergranular fracture mode. The effect of helium on the other bcc metals is unknown, but embrittlement similar to that found in vanadium and in the fcc metals seems to be typical and should be anticipated. It is also likely, but not proved, that with the much higher helium contents produced in CTR service the temperature at which helium is detrimental to ductility will be lower than the 0.5 T_m "cut-off" exhibited at the low concentrations.

Helium embrittlement effects have not been seen in reactor-irradiated bcc metals because fluences have probably been too low to produce significant amounts of gas, and test temperatures have generally been below the 0.5 T_m level at which the effect might become important at these very low gas concentrations.

Radiation-Controlled Creep

It is likely that metals stressed in the irradiation environment of the CTR will deform by creep at rates higher than would be predicted from results of out of reactor tests. The creep rate, set by the rate of motion of dislocations, will be enhanced by the very high concentrations of vacancies and interstitials in the metal and the high flux of these defects to the dislocations. This topic is discussed in Harkness' paper[33] in this volume. It is possible, too, that deformation in the irradiation field may enhance the amount of deformation that a metal can sustain before failure. This very important consideration, which might provide some relief from the severe limitations on ductility determined from post-irradiation testing, has not been studied experimentally in any detail. The effect is suggested, however, by the successful operation of such fast spectrum, high flux fission reactors as EBR-II in spite of the results of post-irradiation measurements such as shown in Fig. 7.

Mechanical Properties under CTR Irradiation

The severely limited amount of data make predictions of irradiation effects on the CTR first wall difficult. The solid products of transmutation reactions, as for example the zirconium that will be produced if niobium alloys are used, will by themselves produce some moderate strengthening with only slight effects on the alloy ductility. As damage accumulates in the metal lattice and increases the strength properties of the metal, the DBTT will be raised. If a molybdenum alloy is used for the first wall, it would be brittle under such conditions as rapid thermal shock loading during shut-down periods if the wall temperature fell below 500°C. Some shift in the DBTT for niobium or vanadium base alloys is expected but there are not sufficient data to suggest whether or not the DBTT could be raised to where it overlapped the CTR temperature cycle.

At temperatures above the irradiation-effected DBTT, in the operating temperature range, the ductility will probably be gradually reduced below the unirradiated values, the ductility decreasing as irradiation time increases. Available data suggests that the amount of ductility that can be accommodated may all be nonuniform ductility (with the wall material having no ability to work harden) and that the available ductility will be greatest in accommodating the most slowly applied loads.

As the helium content in the first wall builds up, it is likely that it will completely dominate the ductility behavior. The helium will certainly promote low-ductility failures by the intergranular failure mode at temperatures exceeding 0.5 T_m for the particular wall material in question. This will limit the upper temperature at which a vanadium wall could be used to about 800°C and niobium and molybdenum wall to about 1100 and 1200°C, respectively. More severe restrictions will prevail if, as is expected, the very high helium concentrations affect the fracture mode at lower temperatures.

It is hoped that the phenomenon of irradiation-enhanced creep will provide some relief from these limited ductilities by allowing the metals to deform in service greater amounts than is predicted by post-irradiation testing.

SUMMARY

This paper has reviewed the presently available data that aid in predicting the effects of CTR neutron irradiation on the refractory bcc metals that will be used in the reactor first wall. Data available are often for irradiation temperatures outside the range of CTR interest and are always at fluences of magnitude and average energy much below those that will be reached in CTR service. The CTR irradiation environment will be more damaging than the environment in any present fission reactor, with displacement damage rates higher by two to five times those in fast fission reactors and transmutation rates, especially for the important helium production, 50 to 200 times those in fission reactors.

The high helium concentration will influence the irradiation-produced swelling of the wall material. Probably it will be most influential in nucleating cavity formation. The cavities will be void-like. The observed greater amount of swelling in reactor irradiated bcc metals than the amount that is predicted due to helium equilibrium suggests that the cavities will contain a pressure of helium below the pressure that would exist in equilibrium bubbles. It seems probable that the swelling, outside of the helium influenced nucleation stage, will be similar to the swelling produced in helium-free void formation. The controlling factor governing the swelling will be the supply of excess of vacancies over interstitials that reach the cavities. Saturation in swelling is expected but the data available are not adequate to predict the values of swelling at which saturation will occur.

The most important effects of neutron irradiation on the CTR first wall will be on the mechanical properties. The most limiting to CTR operation will be the ductility reductions in the wall material. Ductility will be affected by three processes:

1. The lattice hardening produced by irradiation will raise the ductile-to-brittle transition temperature (DBTT). This may limit the lower temperature CTR operation.
2. The lattice hardening will limit the amount of both uniform and total elongation available during service at temperatures above the DBTT. This becomes especially important at rapid loading rates.
3. The transmutation product helium will produce a transition to a low-ductility, intergranular fracture mode. This will be especially important in limiting the upper operating temperature of a CTR.

Some data on bcc refractory metals irradiated and tested at typical CTR temperatures suggest that the ductility in these metals is not as severely affected by irradiation as is stainless steel. The possibility of irradiation-enhanced creep contributing to available ductility under service conditions is also encouraging.

Finally, Fig. 8 defines graphically the probable temperature ranges over which the various irradiation effects are expected to be important.

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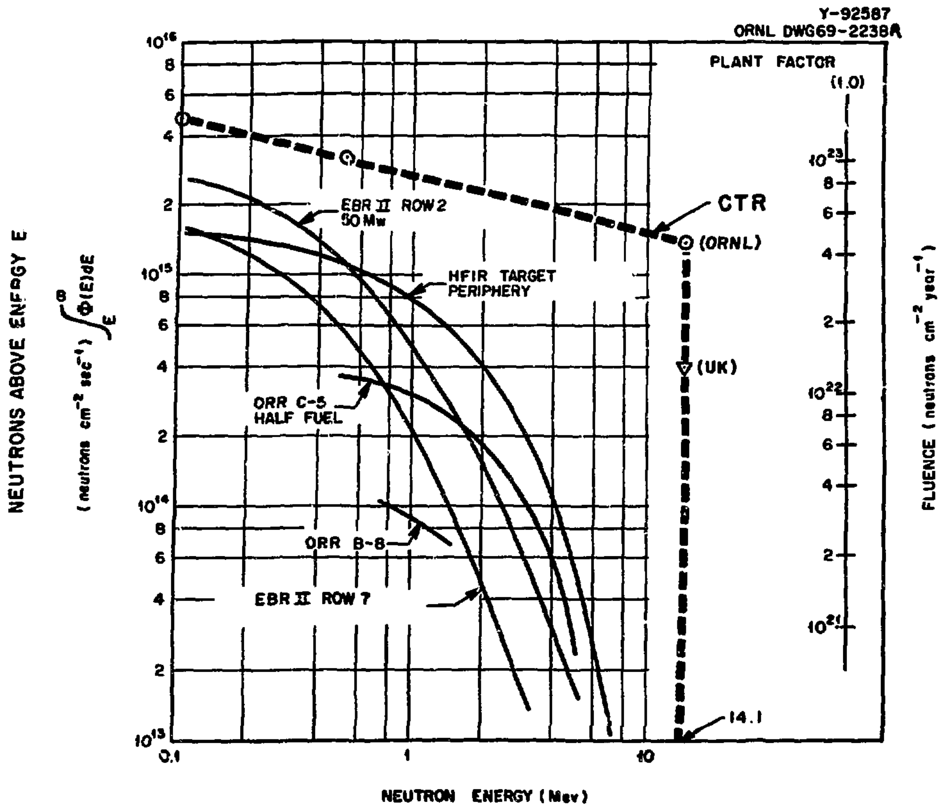

Fig. 1. Integral Neutron Flux on First Wall of Proposed CTR Compared to Core Flux Levels in Three Fission Reactors.

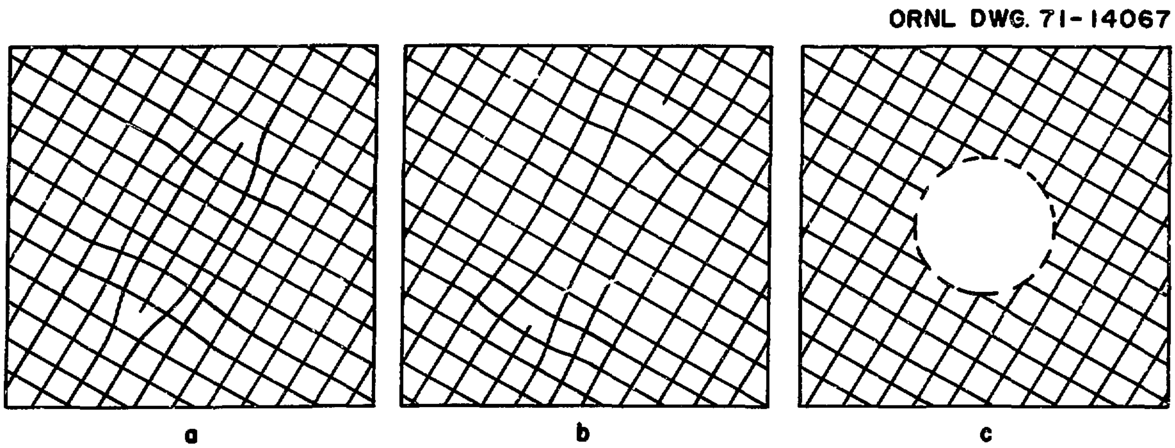

Fig. 2. The Major Types of Defect Clusters. (a) Interstitial loop, viewed on edge, forms by precipitation of interstitials into two-dimensional disks. (b) Vacancy loops form similarly as vacancies precipitate in the disk morphology. (c) Cavities are three-dimensional vacancy clusters with little strain at the cavity-lattice boundary. The distinction between the two types of cavities, gas filled bubbles and empty voids, is given in the text.

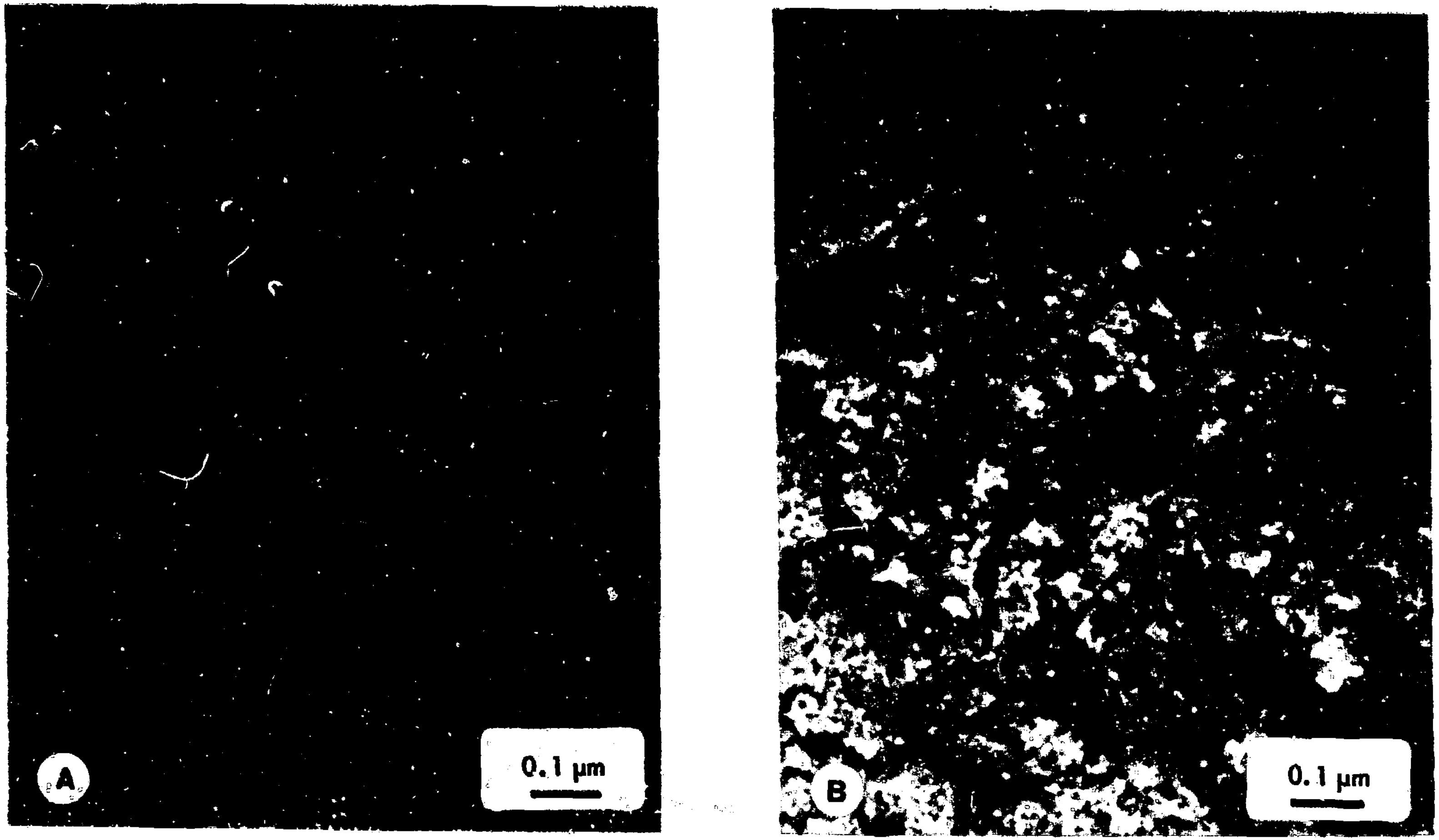

Fig. 3. Examples of the Two Components of Microstructural Damage Seen in BBC Metals Irradiated in the Temperature Range 0.3 to 0.5 T_m. This is a sample of molybdenum irradiated at 585°C to 2.5 x 10^{22} n/cm^2 ($E > 0.1$ MeV). (a) Shows voids seen under absorption contrast conditions. (b) Dislocation loops and segments seen under diffraction contrast conditions. Both components of damage occur throughout the specimen.

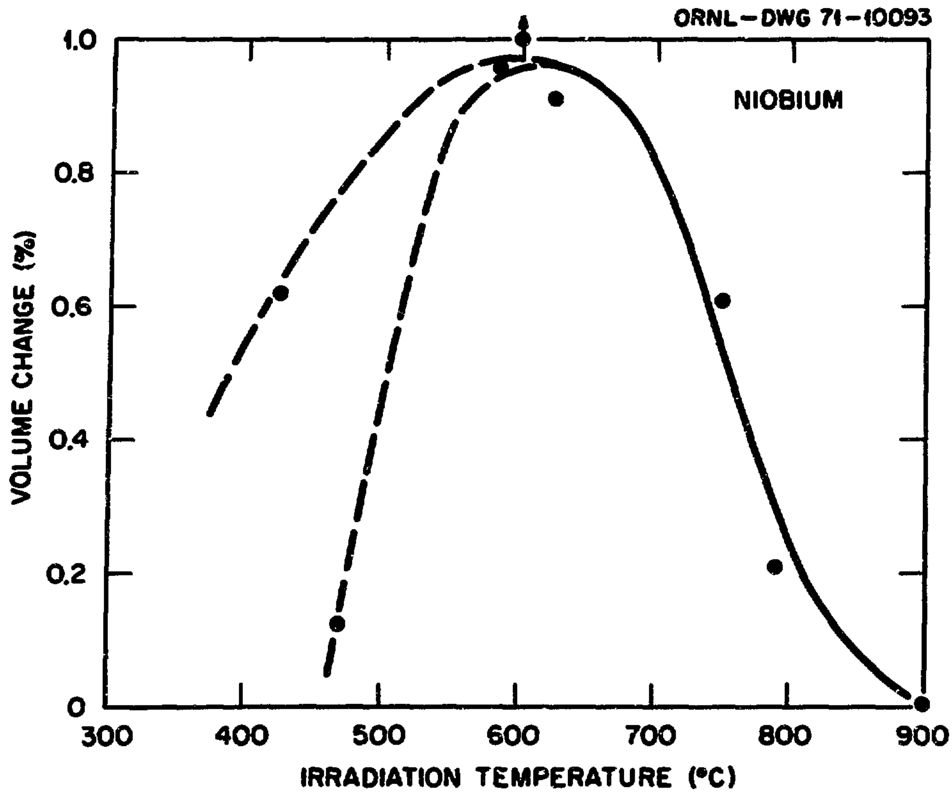

Fig. 4. Swelling of Niobium as a Function of Irradiation Temperature. Source of data can be identified in Table 2. The data have been normalized to the intermediate fluence of 5×10^{21} n/cm^2 (E > 0.1 MeV) using the assumed model that swelling is linearly dependent on fluence.

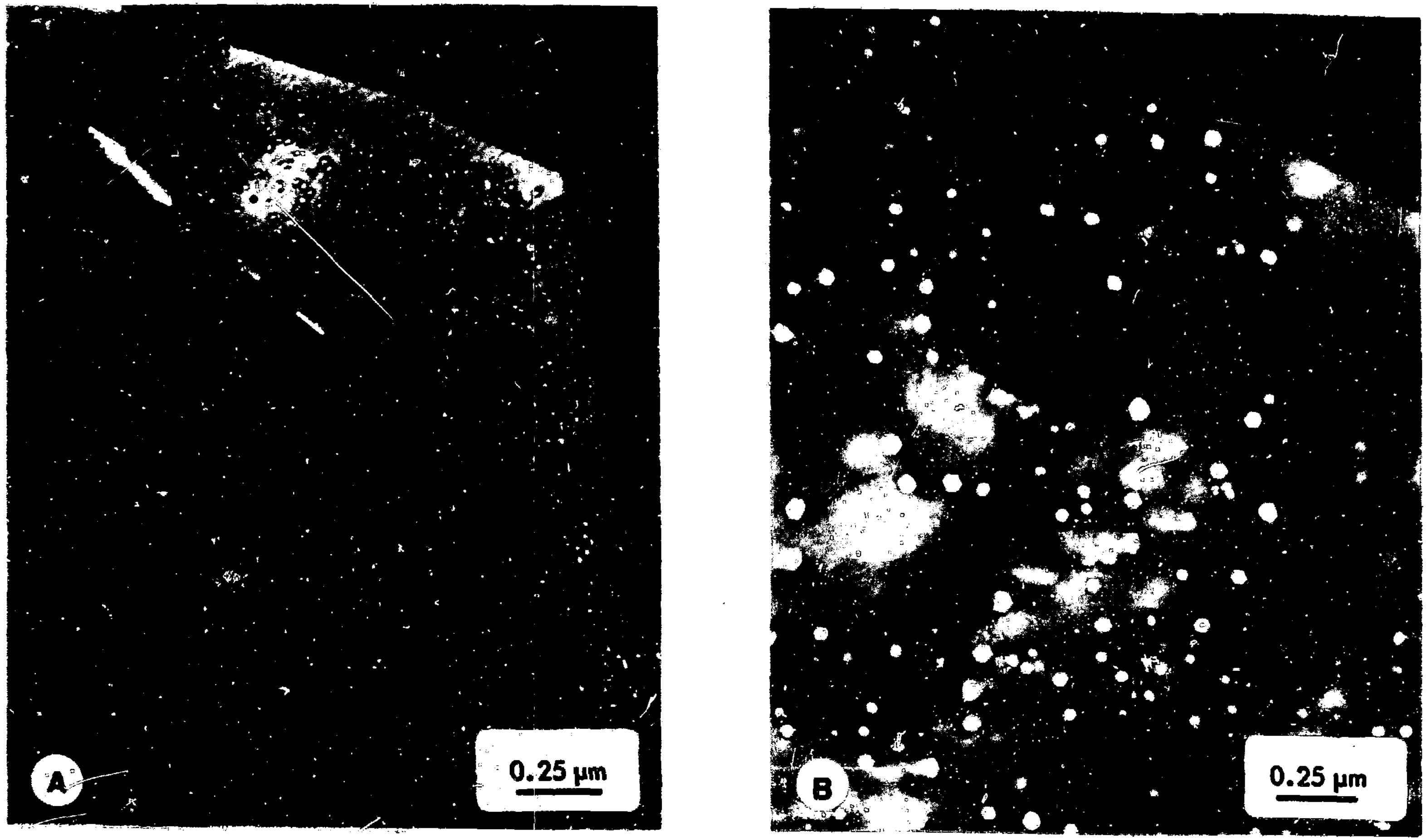

Fig. 5. Void Microstructure in (A) Niobium and (B) Niobium-1% Zirconium. Irradiated to 2.5×10^{22} n/cm^2 (E > 0.1 MeV) at 790°C. The niobium void concentration is 2.8×10^{15} voids/cm^3 with average diameter 186A°. The alloy contains $< 2 \times 10^{14}$ voids/cm^3 with an average diameter of 510A°. The swelling in each case is near 1% volume increase.

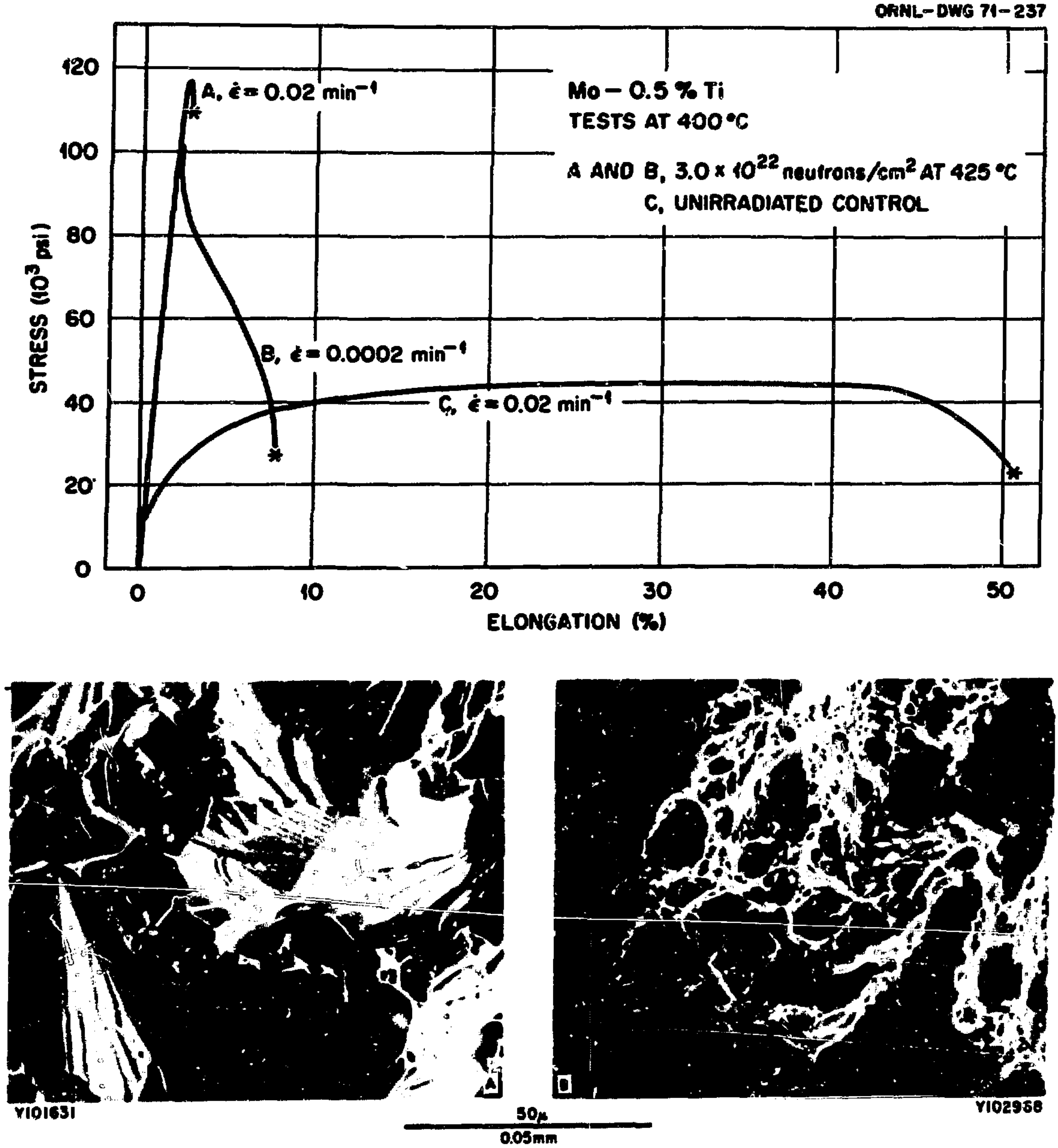

Fig. 6. Tensile Results on Irradiated and Control Samples of Mo-0.5% Ti. The scanning electron micrographs of the fracture surfaces, corresponding to tensile curves A and B, show that test A terminated in a brittle cleavage fracture and test B in a characteristic ductile mode fracture

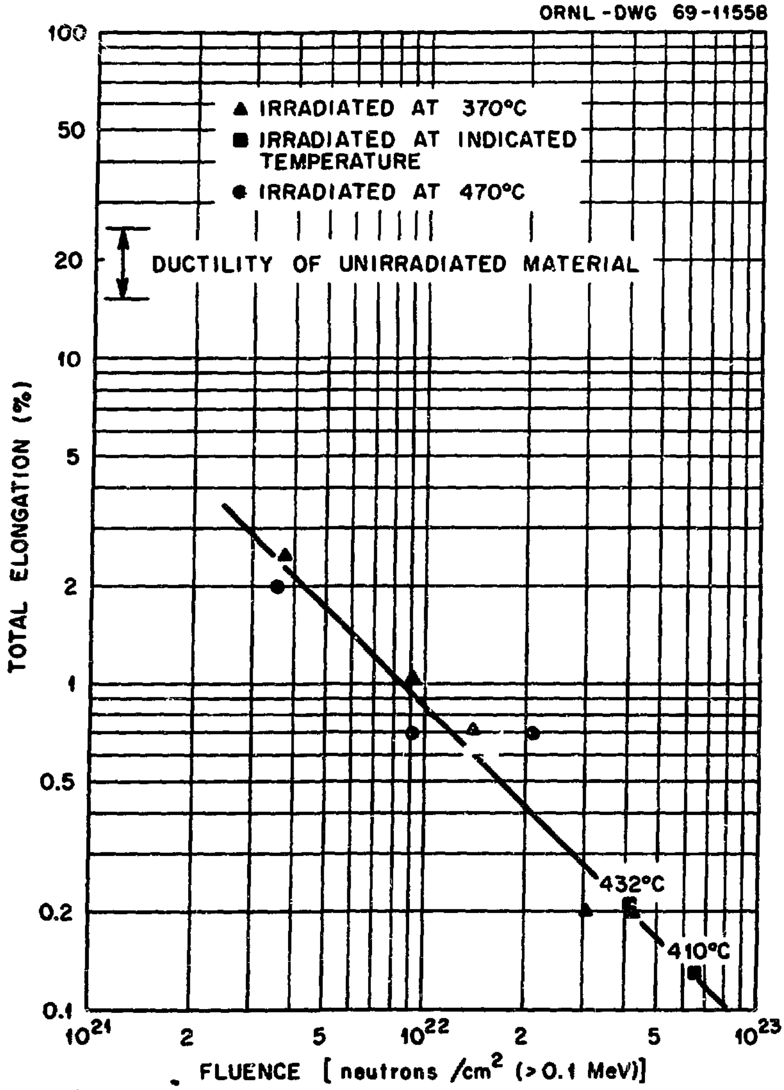

Fig. 7. Post-Irradiation Ductility of Type 304 Stainless Steel, Creep Tested to Failure at 600°C. Initial stress 27,500 psi. All irradiations in the EBR-II fast reactor at conditions indicated (From Bloom and Stiegler, Ref. 27).

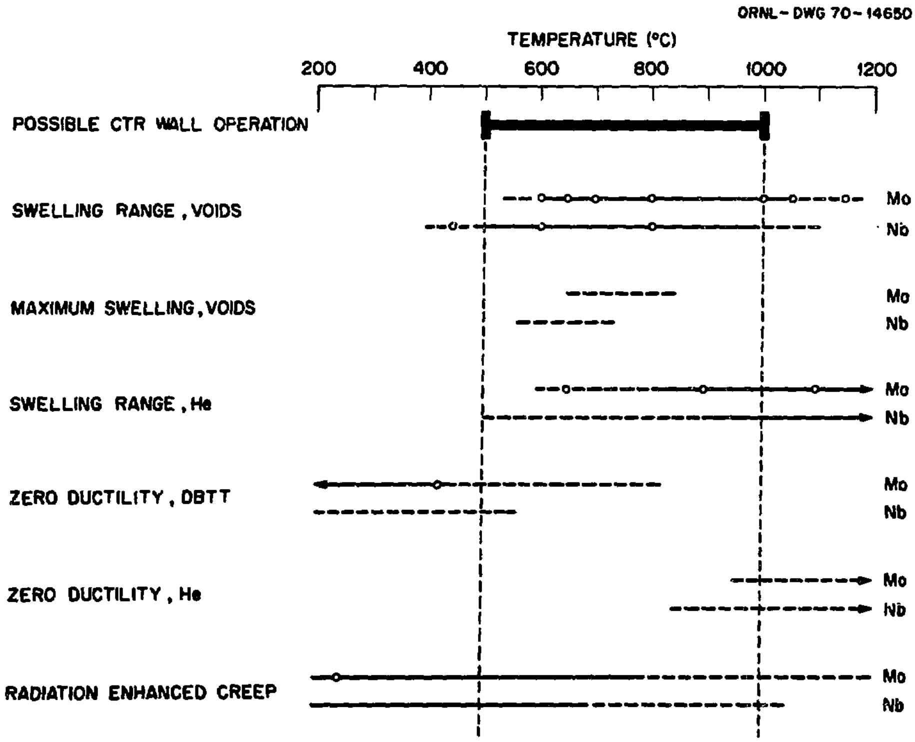

Fig. 8. Probable Temperature Ranges in Which the Various Effects Discussed in This Paper Will be Problems to Successful CTR Operation. The ranges are given for molybdenum and niobium, two candidate first wall materials. Although all effects are uncertain, broken lines suggest regions of lesser importance than regions defined by solid lines. Circles are data points verifying the processes.

IRRADIATION EMBRITTLEMENT IN THE BCC METALS

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The study of irradiation embrittlement in metals and alloys in the past 25 years has been concentrated primarily on the FCC metals such as the austenitic stainless steels and nickel-base alloys, the HCP zirconium based alloys and the BCC ferritic pressure vessel steels. Unfortunately, very little information has been generated on the mechanical properties of the high temperature BCC metals (Mo, Nb, W, Ta) and alloys.

Based on the knowledge generated on the FCC metals and alloys and, in part, on preliminary information available on the BCC high temperature alloys, it can be safely stated that irradiated embrittlement may be separated into two categories, depending on the irradiation and/or testing conditions of the material under investigation. Although there may be several substages, it would be adequate for the purpose of this discussion to consider the following two categories:

a. Low Temperature Embrittlement; $T<0.5T_m$
b. Elevated Temperature Embrittlement; $T>0.5T_m$

As discussed earlier in this session by Dr. Wiffen, the types of defects generated by neutron irradiation include point defects consisting of free interstitials and vacancies, planar defects (interstitial or vacancy loops) and compact defects (voids). In addition, foreign atoms are generated as a result of transmutation reactions with perhaps the more important being helium atoms. The helium atoms form the bubbles which are so well known and strongly believed to be responsible for the elevated temperature embrittlement in irradiated austenitic stainless steels. Finally, the irradiation induced defects may also form various complexes with the impurity interstitial atoms (B,C,N,O) normally present in these BCC metals. All the above defects can, in some way, affect the mechanical and physical properties.

Plastic deformation, which is a measure of ductility, will depend on the ease with which dislocations can move within each grain. Obstacles to dislocation motion, such as the irradiation induced defects described briefly, will tend to harden (strengthen) the metal with a resulting decrease in ductility. The general trend is shown schematically in Fig. 1. Therefore, the test temperature will play an important role in the embrittlement of irradiated metals in that (a) the thermal stability of the defects are such that they will anneal (dissolve) out with an increase in temperature and (b) for a given defect (one that is relatively stable over a range of temperatures) the dislocation may be able to surmount the barrier with greater ease with an increase in temperature. Both events lead to improved ductility. This circumstance is true for the case of cold-worked metals, for the precipitation hardened alloys and there should be no reason for any difference in material containing irradiation induced defects.

Annealing studies show that most of the point and planar type defects are removed when the temperatures reach about 0.31 and 0.41 T_m respectively. Actually the 0.41 T_m temperature range is that normally recommended for recrystallization processing of commercially pure BCC metals. Although not confirmed, it is believed that the irradiation induced voids are removed by annealing at temperatures approaching 0.7 T_m.

For the case of low temperature embrittlement, it may be stated that to a first approximation the BCC metals may behave quite similarly to the FCC metals in that the embrittlement will be directly dependent on the number density of planar or compact clusters. The strength is increased and the ductility decreased typical to that sketched in Fig. 1.

For the case of elevated temperature embrittlement, very little data exist on mechanical properties above 0.5 T_m. Trends observed on the existing data, however, do show that the type of elevated temperature embrittlement known to exist in the FCC alloys does not occur in the BCC metals. Consider Fig. 2 where the elongation of tungsten is plotted as a function of creep-rupture test temperature. It is obvious that the ductility tends to show an increase beyond the 0.5 T_m temperature regions. (The value of 0.5 T_m in tungsten is about 1570°C). The valley in the elongation curve for control and irradiated specimens at about 1700°C is typical to that observed in commercial powder metallurgy tungsten. It is also interesting to note the difference in ductility for those specimens irradiated above or below the 1250°C (0.41 T_m) temperature range. This is the recrystallization region and it is believed that the number density of planar loops and possibly voids are much greater for the irradiations below 0.41 T_m. This leads directly to a strengthening effect with a corresponding reduction in ductility.

In addition to the irradiation and/or test temperature as an important parameter for the ductility behavior of BCC metals at test temperatures below 0.5 T_m, it is found that the strain rate plays a significant role. As an example, tests performed on molybdenum at three different strain rates show that the flow stress will (a) increase with a decrease in test temperature for a fixed strain rate and (b) decrease with a decrease in strain rate for a fixed temperature. When the magnitude of this flow stress reaches the brittle-fracture stress the specimen will fracture with nil ductility. This behavior is clearly shown in Fig. 3. Unfortunately similar studies have not been performed on the irradiated BCC metals and alloys. The presence of the irradiation induced planar defects (and voids) tends to increase the yield stress over and above that of the unirradiated metal so that the brittle-fracture stress is reached at a higher temperature and thereby increasing the brittle-ductile transition temperature.

For the case of test temperatures above 0.35 T_m, creep-rupture tests on irradiated molybdenum specimens which were irradiated at three different temperatures are shown in Fig. 4. In this experiment, sets of specimens were irradiated so that the substructure would consist primarily of (a) small planar loops, (b) mixture of large loops and voids and (c) voids. The neutron fluence was essentially the same for the three conditions. It is obvious that the irradiated induced strengthening (i.e. longer time to rupture for the same applied stress) is not accompanied by any significant reduction of ductility. At these test temperatures, the planar loop defects are not very effective in hardening the BCC metals. This is due to the fact that the loop density decreases rapidly in this temperature range and that the remaining defects are not considered to be hard barriers. The specimens irradiated at $\sim$ 70°C will have only planar type defects.

On the other hand, the irradiation induced voids are effectively hard barriers to dislocation motion and this is reflected in the large increase (factors of 12 to 18) in the time to rupture for the specimens irradiated at 700 and 1000°C. The void concentration is about 3 x 10^{16} and 5 x 10^{14} voids/cm^3 for the 700 and 1000°C irradiation condition, respectively.

Based on the relatively small amount of data available on the ductility of irradiated BCC metals and alloys the following conclusions are made:

1. Low temperature irradiation embrittlement mechanisms for both BCC and FCC metals are quite similar.

2. Large increases in the brittle-ductile transition temperature may be produced by irradiation.

3. The strain rate plays an important role in the degree of irradiation embrittlement for the case of test temperatures lying between 0.15 and 0.35 T_m. The brittle-ductile transition temperature is believed to decrease with a decrease in strain-rate.

4. The ductility of the irradiated BCC metals is not significantly affected for the case of test temperatures above 0.41 T_m, especially for the slower strain rates (creep).

5. The elevated temperature helium embrittlement, characteristic of the FCC metals, does not appear to be as significant in the case of the BCC metals.

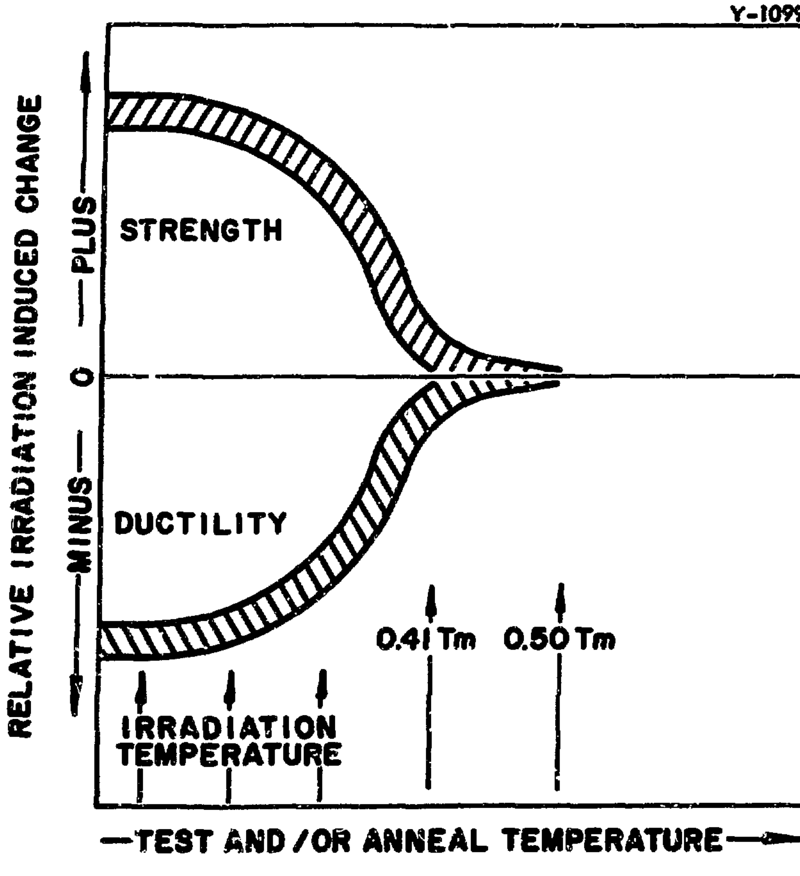

Fig. 1. Sketch Illustrating the General Behavior of Low Temperature Embrittlement of Irradiated BCC Metals and Alloys Following Low Temperature Irradiation.

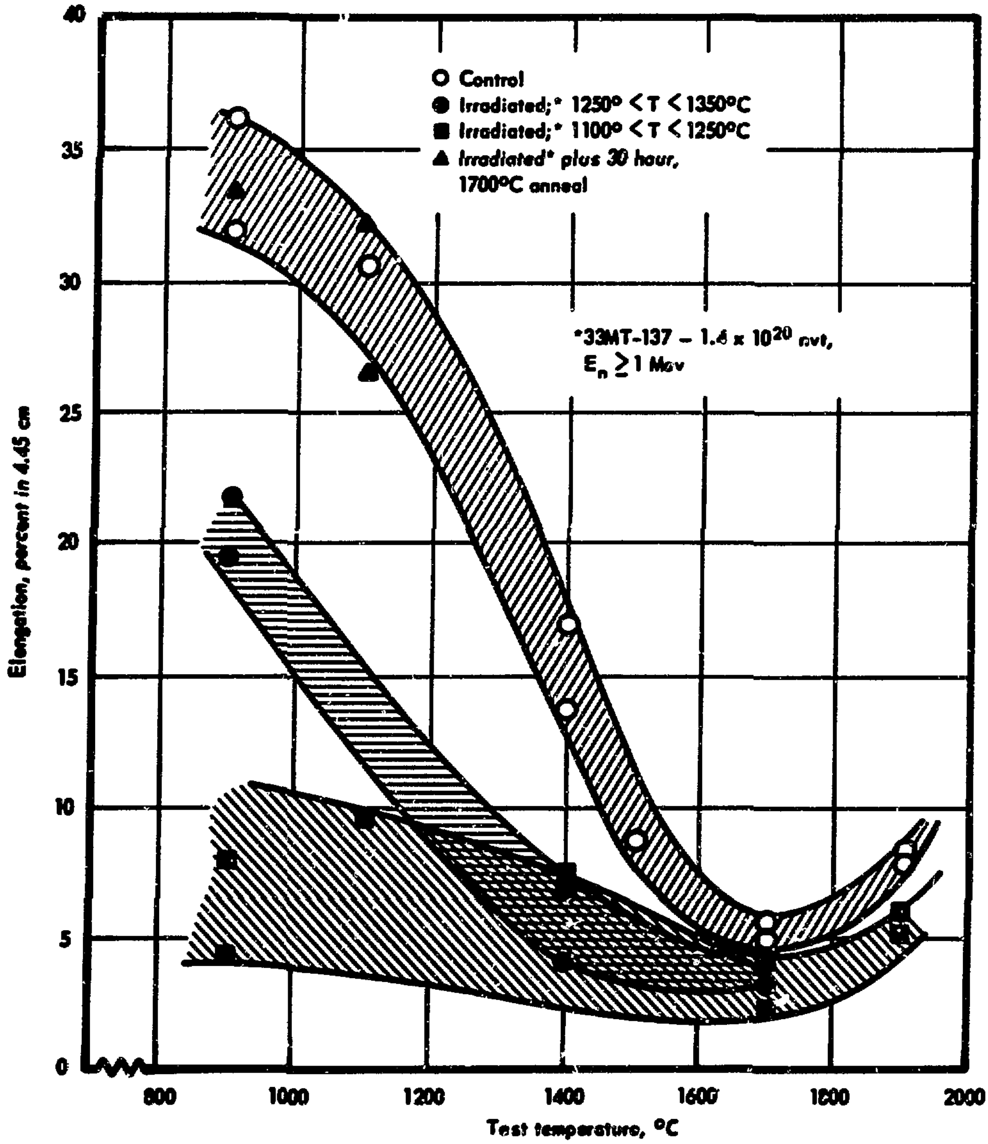

Fig. 2. Creep-Rupture Ductility of Tungsten as a Function of Test Temperature for Specimens Irradiated at Elevated Temperatures and Control Specimens. (From Moteff, to be published.)

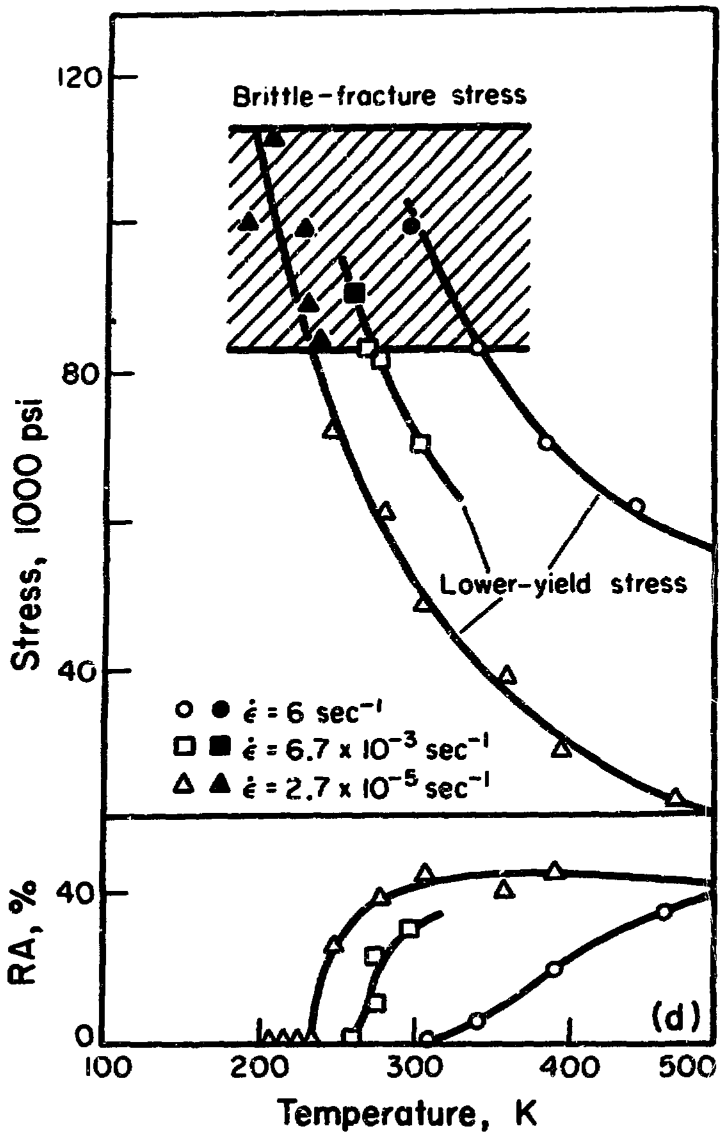

Fig. 3. Ductile-Brittle Transition of Molybdenum Tensile Specimens Determined by the Intersection of Yield and Cleavage Strength Curves. (RA, reduction in area, is a commonly used measure of ductility.)(From Rosenfield, Votava, and G. T. Hahn, Ductility, p 78, ASM Publication 1968.)

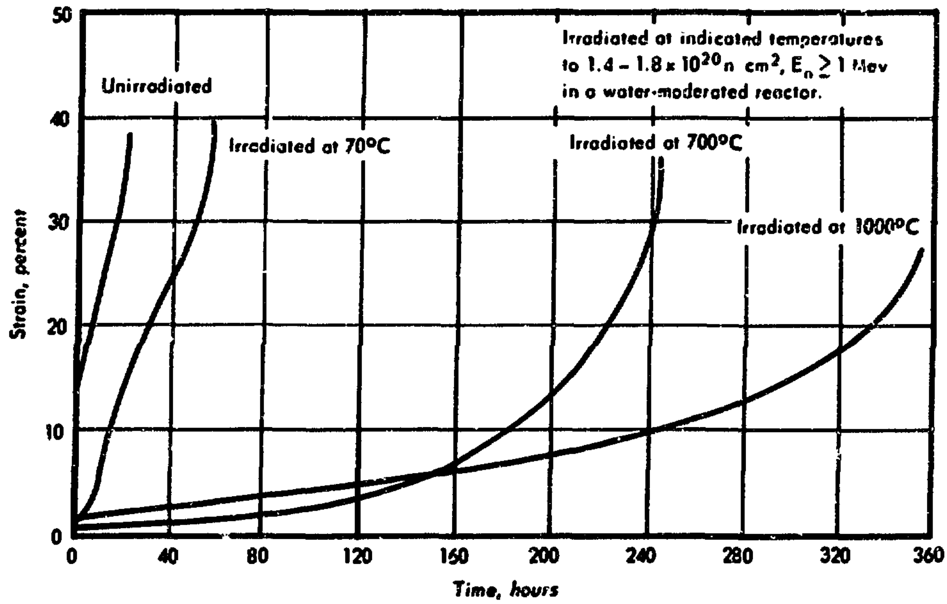

Fig. 4. Creep-Rupture Properties of Polycrystalline Molybdenum at a Test Temperature of 750°C and an Initial Stress Level of 25,600 psi Following Neutron Irradiation at Three Temperatures to 1.4 - 1.8 x 10^{20} Neutrons/Cm^2 (E > 1 MeV) in a Water-Moderated Reactor. (From Moteff, Rau and Kingsbury, Radiation Damage in Reactor Materials, Vol. II, p 274, IAEA, Vienna, 1969.)

A MEANS OF STUDYING RADIATION-CONTROLLED CREEP IN REFRACTORY METALS FOR FUSION REACTOR DESIGN*

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Many components in current fusion reactor designs will be subjected to stresses over long periods of time. How the components creep under these stresses will determine the ultimate configuration of the reactor. Experience gained in fission reactor studies[1] of radiation-controlled creep has indicated that the mechanisms controlling the phenomenon differ from those operative during thermal creep. The stress (σ) dependence of the radiation-controlled creep rate, $\dot{\varepsilon}_r$, is found to be much less than that of thermally controlled creep rate, $\dot{\varepsilon}_t$, (i.e., $\dot{\varepsilon}_r \propto \sigma^{1-2}$ vs $\dot{\varepsilon}_t \propto \sigma^{5-8}$). Furthermore, at constant stress, radiation-controlled creep has been found to be nearly athermal, whereas thermal creep is highly temperature dependent.

All mechanisms developed to explain radiation-controlled creep depend in one manner or another on the point defects generated during bombardment. Currently popular mechanisms involve biased nucleation of interstitial loops, biased growth of interstitial loops, and increased dislocation climb velocities over radiation-produced obstacles. The creep rate obtained in the formulation of each of these mechanisms is proportional to the difference between the interstitial and vacancy flux reaching the loop or dislocation.

As discussed elsewhere[2] this difference is proportional to neutron flux when a large number of sinks (loops, voids, etc.) are present and nearly independent of temperature for a given sink structure. At higher temperatures the thermal vacancy population becomes greater than the radiation produced concentration and normal, thermal creep processes again

*This work was performed under the auspices of the U.S. Atomic Energy Commission.

[1] E. R. Gilbert and L. D. Bl. kburn, WHAN-FR-30, (Oct. 1970).

[2] S. D. Harkness and Che-Yu Li, to be published, International Sumposium on Radiation Produced Voids, Albany, N.Y., June 1971.

become dominant. The temperature where this occurs is dependent on the neutron flux level. For expected CTR conditions thermal and irradiation controlled creep should be equal at about half the absolute melting point of the materials (1170, 1097, and 814°C for molybdenum, niobium, and vanadium, respectively). Radiation controlled creep would be expected to dominate at lower temperatures, thermal creep at higher.

To assess experimentally the temperature, flux, fluence, and stress dependencies of radiation-controlled creep of vanadium, niobium, and molybdenum alloys under conditions of interest to fusion reactor design, is difficult. Experiments can be performed in fast reactors, although, with the present facilities, temperatures must be limited to <700°C. These tests, however, are expensive and time-consuming. An alternate to this type of testing has been developed at Argonne National Laboratory that uses a beam of 22-MeV deuterons to generate the high point-defect concentrations responsible for the enhanced creep rates under irradiation.

A schematic of the uniaxial creep rig used during deuteron bombardment is presented in Fig. 1.

The bombardment is carried out with the 60-in. cyclotron operated by the Chemistry Division of Argonne National Laboratory. Deuterons were chosen as the projectile because they create more damage than protons while retaining a high diffusivity. This high diffusivity results in low, steady-state deuterium concentrations as opposed to the high gas concentrations that would result from an alpha-particle bombardment. Deuteron currents up to 10 $\mu A/cm^2$ on the sample at energies of 22.4 MeV are possible with the present experimental arrangement. In practice, the deuteron beam is degraded by a rotating (360 rpm) linear aluminium wheel such that an equal number of deuterons stop throughout the sample thickness. This scanning rate is much faster than the expected[3] vacancy lifetime of 1 sec, thereby insuring that the vacancy concentration will not rapidly rise and fall with time.

[3]Che-Yu Li, D. G. Franklin, and S. D. Harkness, ASTM STP 484, 347, (1970).

What is, of course, desired in this work is an equal number of Frenkel pairs created throughout the thickness of the sample. The equations for the energy loss of the deuteron-to-electron excitation and for the elastic interactions were obtained from Thompson.[4] Although the absolute magnitude of the calculated generation rates may be in error by a factor of 3, it is felt that the generation profile should be roughly accurate.

Based on these results, an average Frenkel-pair generation rate of $1.7 \times 10^{16}/\text{cm}^3/\mu\text{A}/\text{cm}^2$ is used to compare the results obtained in the cyclotron with those obtained during neutron bombardment. This compares with the estimated effective Frenkel-pair generation rate of $1.5 \times 10^{16}/\text{cm}^3$-sec at the EBR-II core center, or roughly comparable to that expected in a fusion reactor vacuum wall.

RESULTS

Experiments to date on stainless steels have indicated three major points:

1. Radiation-controlled creep is nearly athermal.
2. The stress dependence of radiation-controlled creep is much lower than for thermal creep.
3. Radiation-enhanced creep must be expected to decrease as radiation-induced microstructures (voids and dislocation loops) are developed.

The first conclusion was obtained by changing the test temperature by 40°C during two successive runs (see Table I) and observing that no detectable change occurred in the creep rate. Such a temperature change would result in a two-order of magnitude change in thermal creep rate.

The second point was established by changing the stress while maintaining other variables constant. The measured values of m ($\dot{\varepsilon}_r \propto \sigma^m$) were 0.8, 1.5, and 2.5 in the three tests. Stainless steel has m values of 6-9 under thermal creep.

The third point was determined by testing a sample previously irradiated in EBR-II to a fluence of 5.6×10^{21} n/cm^2 to develop a void-dislocation loop microstructure. This sample crept at a factor of 50 lower rate than a similar sample that had not been EBR-II irradiated.

[4] M. W. Thompson, Defects and Radiation Damage in Metals, Cambridge Univ., 1969, p. 155.

Although all the previous results were for austenitic stainless steel, similar results are expected for the refractory metals. Further work will be aimed at obtaining experimental results on the functional dependencies of radiation-controlled creep in these metals.

CONCLUSIONS

Charged-particle bombardment has proved to be an effective means of studying radiation-controlled creep. It is suggested that this technique be extended to a study of the creep of refractory metals during bombardment. It is felt that this information will be of value to designers of initial fusion reactors.

TABLE 1

CREEP RESULTS OBTAINED FROM 22-MEV DEUTERON BOMBARDMENT

Type of Stainless Steel	Previous EBR-II Fluence $n/cm^2 > 0.1$ MeV	Stress, ksi	Temp., °C	Deuteron Current, $\mu A/cm^2$	Creep Rate, hr^{-1}	Length of Test, hr
304	0	66.4	330	1.6	5.2×10^{-4}	6.5
304	0	66.4	372	1.7	5.0×10^{-4}	13.2
304	0	40.1	380	1.8	3.37×10^{-4}	13
304	0	40.1	380	0	$< 1 \times 10^{-5}$	55
304	5.6×10^{21}	65.7	390	1.7	1.1×10^{-5}	21.8
316	0	18.9	492	0	$< 1 \times 10^{-7}$	200
316	0	18.9	501	2.0	1.1×10^{-5}	12.8
316	0	34.8	506	2.0	2.7×10^{-5}	7.5
316	0	48.7	508	2.2	6.1×10^{-5}	14

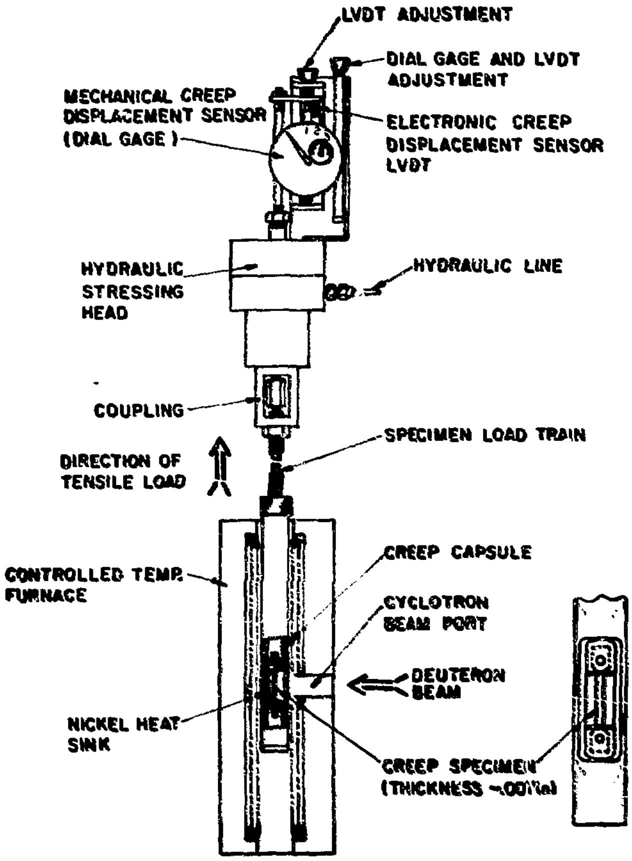

Fig. 1. Schematic of Uniaxial Creep Apparatus Used During Deuteron Bombardment.

USE OF HIGH ENERGY CHARGED PARTICLE BOMBARDMENT TO SIMULATE HIGH FLUENCE NEUTRON DAMAGE IN CTR MATERIALS

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The materials used to construct future fusion reactors will be subjected to the most severe high temperature radiation environment that has ever been imposed on solids.[1,2] The displacement rates due to the 14 MeV neutrons alone will be at least 3 to 4 times higher than the present damage rate in our most powerful fast test reactors. In addition, several hundred parts per million of hydrogen and helium gases will be neutronically produced in metals for every year of operation.[2] The fact that such high displacement and transmutation rates will be occurring at temperatures well above that required for diffusion of these defects means that they will migrate to sinks where they will precipitate into voids or bubbles. Among other things, these latter defects will cause the metals to swell, in some cases nonuniformly, which in turn will impose severe stresses on structural components. It is vital for the materials scientist and design engineers to know the magnitude of this swelling, its fluence and temperature dependence as well as its sensitivity to material parameters.

The most logical approach to this problem would be a well designed irradiation program covering a wide range of possible CTR materials, temperatures, and anticipated neutron fluences. Unfortunately, the irradiation facilities presently available to us cannot fulfill these needs in a reasonable time (i.e., within the next 10 to 15 years). The situation is summarized in Fig. 1 which illustrates the damage-temperature regimes for one year of irradiation in EBR-II, future LMFBR's and future CTR's. It is possible to simulate CTR temperatures in fission reactor by the construction of small, heated capsules which can be inserted into the core. One can even attempt to simulate the large amounts of neutronically produced gases by preinjecting the samples with appropriate amounts of these elements before irradiation (although this approach is not widely accepted as a true simulation especially when more than a few ppm of insoluble gases are used). But what one cannot do, is to significantly increase the displacement rate so as to even approach the displacement rate in CTR's, let alone exceed it, so that data pertinent to a 20 year exposure can be obtained in the reasonable lifetime of a government funded project.

British scientists R. S. Nelson and D. J. Mazey[3] recognized the limitation in 1969 when working with LMFBR materials and devised a rather clever experiment that accelerated the damage rate in 316 stainless steel by three orders of magnitude above that available in current fast reactors. In that work they used a few hours of irradiation with 100 keV protons, carbon ions, oxygen ions, and iron ions to duplicate a high temperature damage structure which normally required up to a year of fast neutron irradiation to produce. They extended this work to higher equivalent damage states with 20 MeV carbon ions[4,5] and also applied the technique to pure nickel.[4,6] The field was immediately expanded by the use of 1 to 2 MeV protons[7,8] and 5 to 10 MeV Cu,[9] Se,[10,11] Ni,[12,13] and Ta[13,14,15] ions to study the temperature and fluence dependence of void formation in metals and alloys of interest to both fission and fusion reactors. The current damage temperature regime that can be reached with heavy ion bombardment is also shown in Fig. 1 along with the irradiation time, in hours, required to produce the dpa value on the left hand side of the figure. Other workers have now used high energy beams of deuterons,[16,17] nitrogen[18,19] ions and electrons[20-22] for such studies and the general technique (as well as significant limitations) is now widely accepted and utilized by many laboratories.[23-26]

Perhaps the most significant achievements of these studies have been the establishment of a saturation level in void induced swelling for stainless steels[6] and pure nickel.[6,11,14] These studies have indicated that the swelling may be limited to 5 to 15% which would at least allow reactor designers to take into account these dimensional changes. A significant amount of basic information about the nucleation and growth mechanisms of voids has also been obtained. For example, ion bombardment studies[3,6,9-14] have shown that inert gas atoms such as helium are not essential for void formation, but may possibly alter the size and number distribution if more than 10 ppm of helium is present. This is important to fusion reactor materials in which several hundred ppm of helium will be neutronically produced during one year operation.[2] The production of voids during high energy electron bombardment studies[20-22] has also eliminated the displacement spike as a required nucleation mechanism because only one, or at the most, two atoms are displaced per incident electron.

The ion bombardment studies have also produced some new and unique void structures, which were simultaneously discovered in high temperature, high fluence neutron damage studies. The three dimensional superlattices of voids have been found in Ni,[11] Mo,[18,19,27] TZM,[19] Nb,[14,27] and Ta[28] in an array coincident with the crystal lattice of the host metal (i.e., fcc in Ni, and bcc in Mo, Nb, Ta, and TZM). The existence of such ordered structures only at high damage states may be of considerable significance in the approach to the saturation of swelling.[11]

There are other advantages of the ion bombardment technique besides that of reducing the time to achieve high damage level states. There is no induced radioactivity when heavy particles are used. The cost per sample is considerably reduced, mainly because of the ease with which the samples may be handled during and after irradiation. The temperature of the irradiation may be varied in any manner desired because unlike nuclear reactors, only the sample, and not the radiation generating equipment, is heated. The lack of neutronically produced gases allows one to include, or exclude, the gases normally inherent to any high energy neutron bombardment. Thus, the role of gaseous impurities in void nucleation may be isolated. The fact that the irradiations are conducted in a vacuum also allows a much closer control of interstitial and substitutional impurities. Finally, the rate at which ions bombard the surface, and hence, the rate at which atoms are displaced may be varied over several orders of magnitude. Such experiments will assist in developing theoretical models on the kinetics of void formation and growth.

There are, as with any experimental technique, certain drawbacks which must be recognized. First of all, the short range of heavy charged particle (1 - 10 microns) is usually insufficient to obtain significant mechanical property data. The exception to this is the very high energy ($\sim$ 30 MeV) light particle work described by Harkness at this conference.[17] In general, it is expected that meaningful mechanical property data will have to come from neutron irradiation of bulk samples. The second disadvantage is directly associated with the primary advantage of this study and that is the shift in effective irradiation temperature due to the extremely high production rates. Bullough and Perrin[29] have shown that one should ion bombard samples at somewhat higher temperatures than those of an equivalent neutron irradiation experiment in order to obtain comparable results. Finally, long term aging processes which might occur during neutron irradiation, may not take place during the much shorter ion bombardment times.

The reader is referred to the many papers in the literature describing the advantages, disadvantages, and results of high temperature bombardment studies,[4,6,7,9,12,14] only a few examples will be shown here. A study of the temperature and fluence effects of void formation in Nb and Nb-Zr alloys, Mo, and TZM, and V has been conducted at BNW. The samples were irradiated over a temperature range of 650 to 1000°C. Figure 2 shows the typical microstructure found in Mo, TZM, and Nb after a 900°C irradiation to only 5 dpa (only a few weeks in future CTR's). The void induced swelling in the Mo and Nb is $\sim$ 1% while that in TZM is $\sim$ 0.5%. Figure 3 shows typical voids in V and Nb-1 Zr after an 800°C irradiation to $\sim$ 30 dpa (a few months in a CTR). The swelling in the vanadium is $\sim$ 3% while that in the Nb-1 Zr is $\sim$ 1%. A more complete picture of this study will appear shortly, but it is clearly evident that all of the candidate materials for CTR reactors which have been irradiated thus far will have a considerable amount of dimensional changes after 1 to 20 years in these reactors.

In summary, high energy heavy ion bombardment studies can make a significant contribution to the area of CTR technology studies if conducted under the proper circumstances. The initial thrust of such a program can be to screen a large number of possible structural materials to ascertain their susceptability to swelling in the operating temperature and fluence range of future CTR reactors. Studies can also be conducted in parallel to shed more light on the exact mechanisms of void nucleation and growth such that more swelling resistant alloys can be developed.

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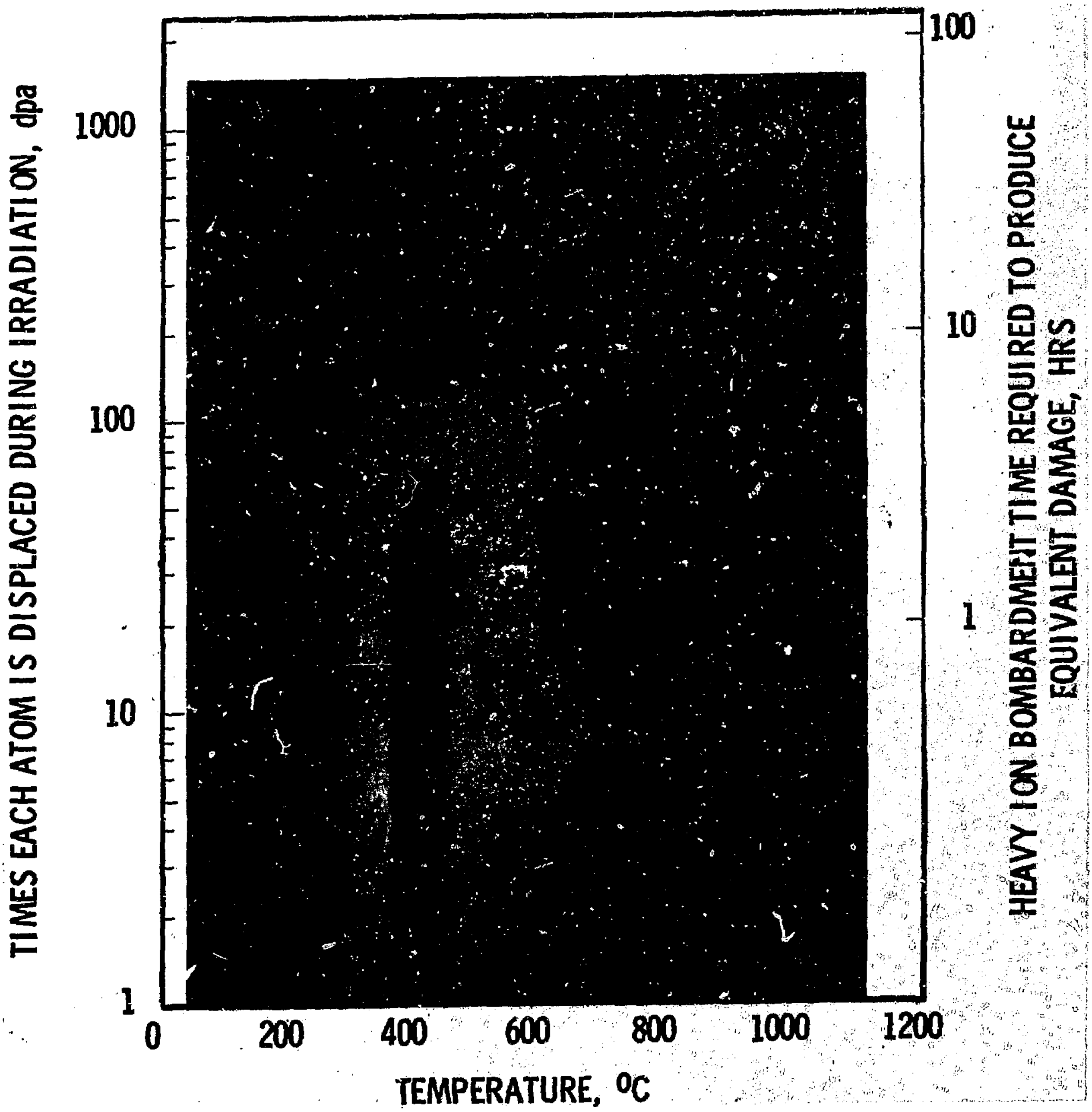


Fig. 1. Characteristic Damage Conditions for One Year of Irradiations in Nuclear Reactors.

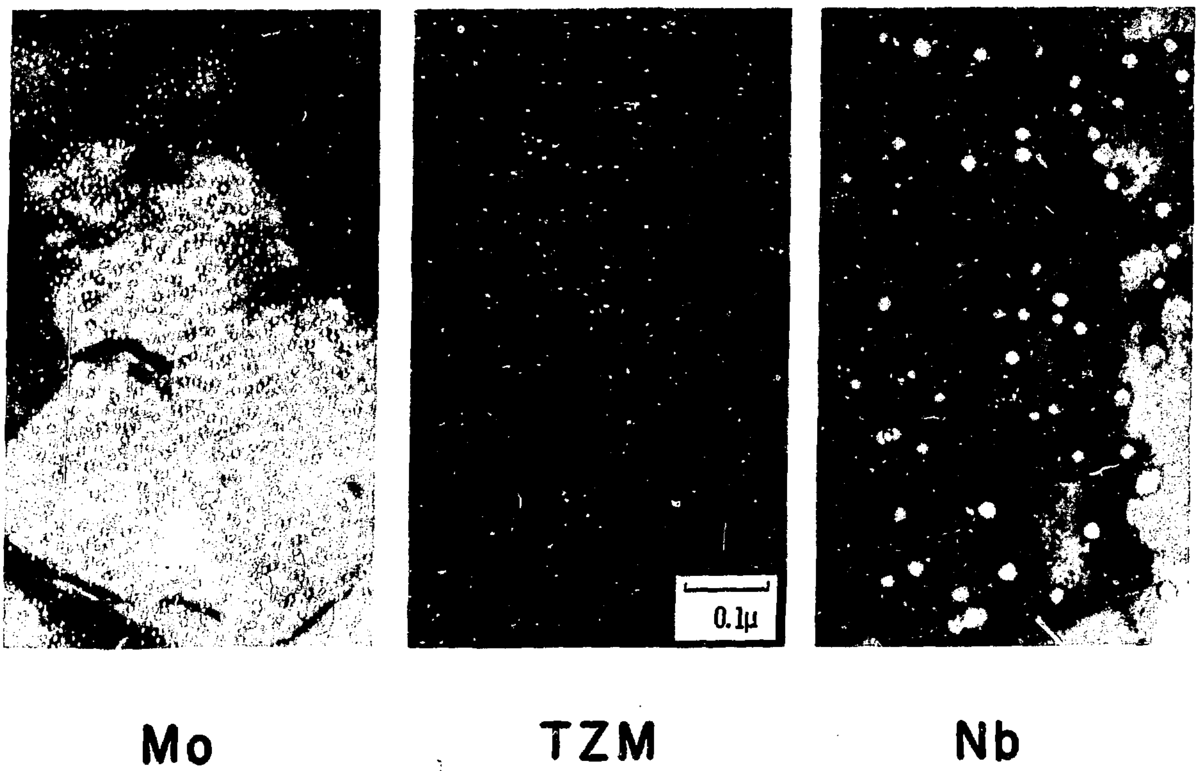
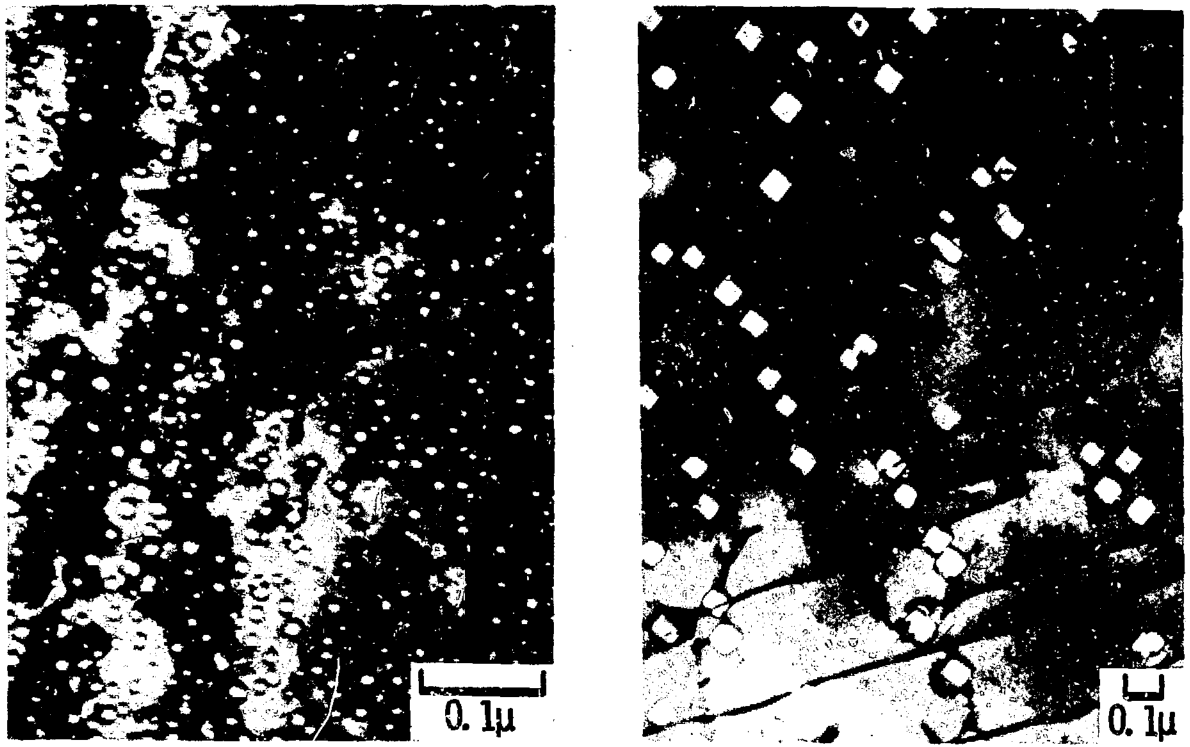


Fig. 2. Voids Induced by 5 MeV Nickel Bombardment to Approximately 5 dpa at 900°C.



Nb-1Zr VANADIUM

Fig. 3. Voids Induced by 7.5 MeV Tantalum Bombardment to Approximately 30 dpa at 800°C.

ION BOMBARDMENT SIMULATION OF 14 MeV NEUTRON DAMAGE IN THIN NIOBIUM FILMS

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Thin (∿ 1000Å) vapor deposited foils of niobium have been bombarded with Nb^+ ions in the as-deposited and annealed condition to ion doses between 10^{13} and 10^{17} ions/cm^2. Bombardment temperatures ranged from 25 C to 850 C. Incident ion energy was 80 to 100 keV.

Examination of foils bombarded at the lower temperatures within this range, by transmission electron microscopy, reveals an extremely high density of very small (< 100Å) interstitial dislocation loops. The range of the induced features was measured to be 700-800Å from the incident face. The apparent (number) density increases with dose. Foils bombarded at somewhat higher temperatures (i.e., 600 C) showed a greatly reduced density of larger (100-1000Å) oriented features believed to be bombardment induced precipitates. Considerable foil growth was observed with bombardment at all temperatures. Some foil growth was observed upon heating in 10^{-6} Torr vacuum.

Foils deposited and released from warm salt substrates have very small grain size (∿ 200Å) and many extensive tear-like defects. These defects "healed" and the foils annealed readily above ∿ 800 C. At annealing temperatures above 1200 C, relatively large (up to 2.0 μ) grains of {100} orientation are commonly formed having essentially no resolvable substructure. Electron beam annealing of local areas on the microscope stage produced grain structures of the greatest clarity. Thin unsupported foils were found to be extremely fragile under all conditions investigated and especially subject to gross contamination at the higher temperatures.

Attempts to perform mechanical properties tests on thin unsupported niobium foils were largely unsuccessful. Sheet specimens (0.5 mm thick) bombarded on one face showed no measurable or detectable surface strain differences attributable to bombardment when strained at room temperature to point of maximum load.

HELIUM INJECTION BY THE TRITIUM TRICK*

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Helium is thought to be very insoluble in solid metals as it is a noble gas. When forced in solution in some way, it is trapped in bubbles at high pressures. Small quantities, less than 1 ppm helium atoms per metal atoms are sufficient to modify high temperature mechanical properties. Just how, and to what extent properties like creep ductility will be degraded by large quantities of helium (200 ppm increase per year over 20 years) is unknown, but large effects are likely.

Helium will be produced in unprecedented amounts in wall materials of proposed power producing fusion reactors, the above figure of 200 ppm per year being typical for such a reactor design, while fast breeder fission reactors would produce about 0.3 ppm per year. Helium effects, therefore, could be very important to fusion reactor technology.

Large controlled amounts of helium in uniform concentration throughout thick samples can be readily obtained through radioactive decay of dissolved tritium gas to ^{3}He. We coined the name "tritium trick" when helium added by this method is used to simulate (n, α) production of helium in (simulated) hard flux radiation damage studies.

Tritium decays to helium with a half life of about 12 years so that about 1/2 percent decays per month. The only radiation emitted is a very weak, 12 keV, beta particle. It is too weak to cause significant damage. Tritium, being an isotope of hydrogen, is very soluble in many metals. It also diffuses into and out of metals quickly. Thus tritium can be dissolved, allowed to decay to the desired helium content, and pumped away selectively relative to the ^{3}He. Only ^{3}He will remain to change properties.

Published phase diagrams show the amount of hydrogen (and therefore tritium) that will dissolve in various metals at any temperature and pressure. As an example, at 1000°C (the operating temperature likely for some fusion reactor walls) and at 1/100 atmosphere pressure of tritium, the ^{3}He

*Work performed under the auspices of the U. S. Atomic Energy Commission and with the support of National Aeronautics and Space Administration.

production rate in niobium (presently the first choice metal for structural parts in the walls of fusion reactors) via beta decay will exactly equal the ^4He production rate at full power via (n, α) reactions. This amount of tritium in uniform solid solution would only be 1/4 atom percent. As another example, niobium charged at 600°C under one atmosphere of tritium gas will collect 200 ppm helium in only ten days, thereby allowing accelerated damage rate studies. The tritium trick should not be difficult, nor expensive, once a tritium handling facility is available. Initial charging experiments are now in progress. We hope to establish how helium diffuses, collects in bubbles and modifies creep properties with electron transmission microscopy, internal friction and creep experiments.

LAMPF (LOS ALAMOS MESON PHYSICS FACILITY) AS A NEUTRON SOURCE FOR RADIATION DAMAGE EXPERIMENTS*

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The neutron flux spectrum in fast breeder reactors is substantially softer than that for proposed D-T fusion reactors. Ideally, a 14 MeV D-T source, perhaps such as that suggested here by D. Henderson, should be used in radiation damage experiments if the results are to be relevant to fusion reactor technology. But such a source is not now available. We believe that high temperature radiation damage experiments with an accelerator called LAMPF will allow us to begin to learn about radiation damage as it will occur in fusion reactors. LAMPF (short for Los Alamos Meson Physics Facility) is scheduled to be fully operating in early 1973. It is an 800 MeV linear proton accelerator with a water cooled copper beam dump. High energy neutrons will be generated in this target. At the best location for radiation damage experiments, the neutron flux will contain a copper evaporation component of nuclear temperature 4 MeV and an intranuclear cascade component of lesser intensity with energies up to 200 MeV in the tail.

The flux at our proposed experiment has been estimated to be at least 10^{13} n/cm^2-sec from the duty cycle, the proton beam current, and the number of neutrons that will be generated by each incident proton. As an independent check, we scaled the calculations for the proposed, but abandoned Canadian ING (Intense Neutron Generator) accelerator to our case. This check estimated the flux at about 10^{14} n/cm^2-sec. Good agreement, since we, by intent, were conservative. As a measure of the combined effect of flux and spectrum, LAMPF will produce about 1 ppm helium atoms per niobium atom per year in niobium. If the Canadian calculations are correct, then this value will be increased accordingly. In fusion reactors, about 200 ppm of helium will be produced per year; in EBR-II, a fast breeder fission reactor, about 0.3 ppm per year would be produced.

We propose to augment the (n, x α) production of helium in LAMPF by the "Tritium Trick" described in a companion presentation. In this way, the helium production rate will be made equal to that anticipated in fusion reactors.

*Work performed under the auspices of the U.S. Atomic Energy Commission and with the support of National Aeronautics and Space Administration.

Mechanical stresses will be present in fusion reactors. Pressure difference across the vacuum wall, mechanical constraint on swelling, differential swelling, and changes in the magnetic field will all cause stress. The structure will be at some elevated temperature, such as 1000°C, so that the combination of sustained stress and temperature will cause creep (the slow time dependent deformation of a material) and its microstructural damage. To be relevant to fusion reactor technology, high temperature radiation damage studies must be concurrent with creep. The radiation damage facility at LAMPF will be built to accomplish this. The great abundance of space at LAMPF will make the experimental procedures much easier than similar experiments in the core of typical reactors. Possible interaction between creep microstructural damage and high temperature radiation damage were suggested. These include:

1. A possible increase of swelling rate due to an increase in the density of dislocations that are able to climb.

2. A reduction in creep ductility because creep causes grain boundary migration. This, in turn, would tend to sweep up radiation induced voids into the grain boundary cracks thereby causing creep failure. The cost necessary to add the radiation damage experiment to LAMPF is small, but the increased capability to evaluate radiation damage as it might occur in fusion reactors is significant.

FACILITY FOR DUPLICATING 14 MEV NEUTRON EFFECTS IN FUSION POWER REACTORS*

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The study of the neutron effects that will occur in a controlled fusion reactor requires a 14 MeV neutron flux one thousand times larger than presently available. This paper describes a practical means for achieving such a flux.

We have recently surveyed[1] the possible sources which might be employed to duplicate the neutron flux (in both intensity and spectrum) which is anticipated from an operating D-T power reactor. Our survey is summarized in Table I and compared with a fusion power reactor. Of all possible sources considered and listed in Table I only an Ion Accelerator, which utilizes a dense gas target for neutron production from the D-T reaction, can satisfy all of the conditions required for the demonstration and study of neutron effects. As one measure of the efficiency of each of the neutron sources listed, Table I gives the rate of helium build-up due to (n, xα) reactions in niobium for each source and compares it with the helium production expected for fusion reactor neutrons. Table I shows that only the gas target Ion Accelerator would be comparable to the fusion reactor in this respect. All other sources are several to many orders of magnitude too weak, and may have an incorrect neutron spectrum as well.

Our conception of the gas target Ion Accelerator, shown schematically in Fig. 1, is similar to the earlier independent suggestion of Colombant and Lidsky.[2] It utilizes a one ampere tritium ion beam from an ion source of the type developed at Oak Ridge National Laboratory, ORNL, a standard 300 keV accelerator column [design codes for which exist at LASL from the meson factory (LAMPF) development], a dense ($\sim 10^{19}$ molecules-cm^{-3}) deuterium gas target in the form of a supersonic jet directed across the ion beam

*Work performed under the auspices of the U.S. Atomic Energy Commission.

[1] H. Dreicer and D. B. Henderson, Facility for Duplicating 14 MeV Neutron Effects in Fusion Power Reactors, Report LA-4709-MS, 1971.

[2] D. G. Colombant and L. M. Lidsky, High Intensity 14-MeV Neutron Source, in Research Laboratory for Electronics (MIT) QPR No. 91, p. 133, Oct. 1968.

to minimize differential pumping requirements,[3] and a tritium recovery dump which utilizes the existing LASL Tritium Facility for recovery and processing to allow reuse of the tritium (Tritium cost = $14 x 10^3/gm). Our preliminary design of the target features a hypersonic wind tunnel at Mach 5 with entrance and exit holes for the tritium beam protected by differential pumping sections.

The major components of the system - ion sources, accelerators, power supplies, hypersonic wind tunnels, pumps, and tritium facilities - are within existing technology. This should make the design and construction of such a neutron source a relatively straightforward matter if a sufficient need for it is judged to exist.

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TABLE I

Neutron Source	Spectrum D-T	Spectrum Type	Flux $n/cm^2 sec$	Helium ppm/month
Fusion Power Reactor	Yes		1×10^{15}	30.
Ion Accelerator - Gas Target	Yes		3×10^{14}	9.0
- Metal Target	Yes		5×10^{11}	.015
LAMPF Beam Dump	No	Copper	2×10^{12}	.017
Dense Plasma Focus at 5 MJ	Yes		1×10^{12}	.030
Experimental Breeder Reactor II	No	Fission	3×10^{15}	.008
Electron Linac (e, γ, n)	No	Uranium	6×10^{11}	.0001
Boosted Electron Linac	No	Fission	6×10^{12}	.0003
			n/cm^2	ppm
Thermonuclear Bomb	Yes		1×10^{17}	.0012
minimum needed for metallurgy				.001

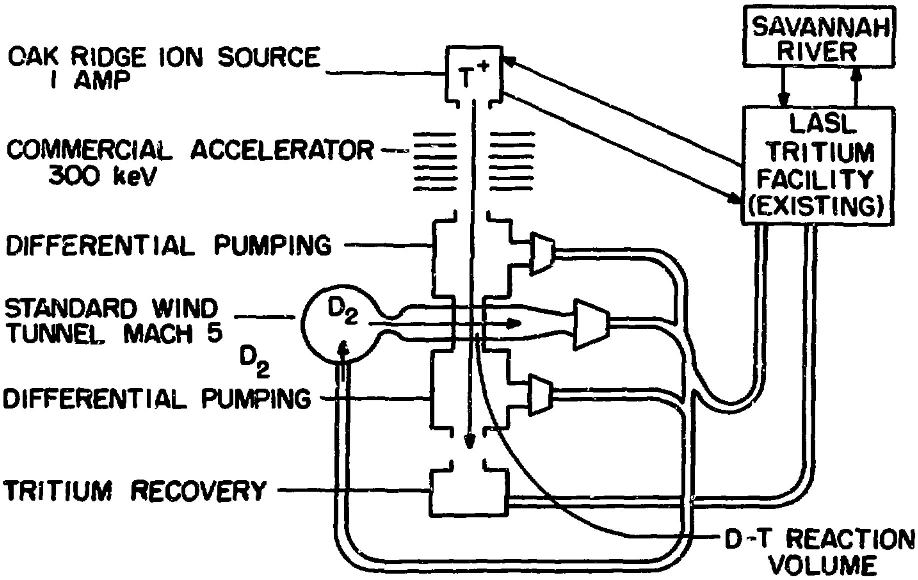

Fig. 1. Gas Target Ion Accelerator for Neutron Production from the D-T Reaction.

SUMMARY OF SESSION

RADIATION DAMAGE

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INTRODUCTION

With the help of Mark Robinson, Gerry Kulcinski, John Moteff, and Sam Harkness we reviewed the last ten years of radiation damage studies related to fission reactors. We tried to evaluate from this background what the problems will be in the controlled thermonuclear reactor (CTR) system and what is known about these problems. Peter Mohr, Walter Green, Dale Henderson, and John Weeks also made brief contributions. Beyond this I am not going to acknowledge individual contributions in this summary. We did establish that most of the problem areas remain uncertain. The data available now are not good enough to predict CTR component life, but are adequate to define likely problem areas. More questions remain unanswered than we answered. The data required to predict the integrity of the first wall have not been obtained. The required neutron fluxes to evaluate CTR response do not exist and simulation techniques will have to be used. Only part of the anticipated radiation damage problems can be looked at in any one study. Fission reactor irradiations cover only part of the need. We need to use those reactors and we need to use ion bombardment damage. Although both of these experimental approaches have severe limitations, they are the best tools presently available.

Floyd Culler reminded us in his introduction that we made mistakes in predicting radiation damage effects in the past ten years, rather bad mistakes in trying to extrapolate beyond existing data. The two problems that were missed were ductility loss due to helium in metals and swelling due to the condensation of radiation produced vacancies into three dimensional voids. I think that we had a concensus that these are two of the main problems anticipated in a thermonuclear reactor, and I hope we now have a better ability to evaluate CTR radiation effects. For one thing, we can work in the correct temperature ranges for the first wall of the CTR reactor. This is a pitfall that was made in early work, using temperature extrapolation in predicting radiation effects. Secondly, we have available techniques now which will let us get to the required level of displacement damage (measured conventionally by calculating the number of times an individual atom is displaced during the radiation exposure). The required damage levels are 10 to 1000 displacements per atom. This was the second problem in earlier work, extrapolation beyond the low experimental irradiation damage concentrations. At the same time, this accelerated technique, ion bombardment, does not produce the damage state that would be produced in a thermonuclear reactor. We can produce the number of displacements, but do not get the number of transmutation reactions that would be produced by 14 MeV neutrons. There are ways that we can work around these problems. Perhaps we can inject the transmutation product simultaneously, especially the one that really concerns us, the helium produced by (n,α) reactions. Perhaps we can introduce some of the transmutation products by other means. In any case, this simulation will require an accurate knowledge of the transmutation reactions and the cross sections for these reactions in the wall material so we can simulate them correctly. It seems doubtful the presently available cross sections are as good as we desire. There are two other major disadvantages of ion bombardment techniques. The kinetics of the process are much more rapid than would obtain in a reactor. A few hours of ion bombardment can produce the amount of displacement damage that would be produced in a CTR in twenty years of operation. This is the important advantage, in that we can reach the damage state required, but there are some pitfalls. Perhaps we know how to model, or at least think we know how to model, this kinetics difference, but we must always recall that it is a limitation. Secondly, we cannot readily determine the required effects on mechanical properties by using these techniques. In fission reactors the fluxes are lower than needed for CTR simulation, but fission reactor irradiations are apparently the only present source of specimens on which to measure mechanical properties. Since we need to guarantee the integrity of this first wall, it is the effect of irradiation on mechanical properties we ultimately have to determine, and so we must continue fission reactor neutron irradiation in addition to using the other simulation techniques. Effects observed by the ion bombardment technique must also be calibrated against a standard, and the standard we have is the fission reactor neutron irradiation.

Again let me emphasize that although these techniques do not really meet the exact needs, they are the only tools available at the moment. We can expect some help from computer modeling of the damage state in the programs now in progress, but, of course, this never replaces data. At best it helps interpret data. We also look forward to development of better radiation sources. Perhaps the accelerator connected neutron sources that were discussed briefly in the radiation damage session will be of some help. Better fission test reactors are being developed in this country and elsewhere, and we would expect to use these as they become available.

In the remainder of this summary I will review some of the main points discussed in the radiation damage session.

CHOICE OF FIRST WALL MATERIAL

We had some discussion of factors other than radiation damage which limit the selection of materials for first wall and other major components. Limits on choice of materials are set by service requirements that include at least:

1. Mechanical properties requirements at the service temperature. This requirement is so far stated without reference to the effect of irradiation on these properties.
2. Compatibility with liquid lithium and other possible coolants.
3. Neutronic considerations involving the decay heat that must be removed during reactor shutdown and the consideration of the long-time radioactivity and solid waste storage and disposal of replacement reactor components.

In discussion it is often implied the wall material will be a pure metal, but strength and fabricability requirements dictate an alloy, not a pure metal, be used. The choice of first wall materials would seem to be restricted to alloys based on the following systems:

1. Niobium
2. Vanadium
3. Molybdenum
4. Tantalum
5. Stainless steels, or other iron-base alloys.

I think this list is approximately in order of decreasing preference, based on what is known today. Niobium and vanadium alloys seem to be leading candidates. The neutronic properties of niobium are advanced in arguments against its use, but most other considerations recommend it. Vanadium is offered as attractive due to activation characteristics, but it is suspect because of high helium transmutation rates (approximately three times that in niobium). Vanadium-base alloys restrict operating temperatures below those allowed by niobium and also suffer from lack of an existing fabrication technology.

The second category of materials includes molybdenum and tantalum. Several sessions raised the question of using molybdenum but the fabrication technology, especially welding, is difficult as it exists today. There may be some hope it will be better before we have to build a fusion reactor. Tantalum is probably out of the question because of the activation problems, but perhaps the afterheat cooling requirements of the reactor could be met. The third category of material listed is stainless steel and similar iron-base alloys. This was brought up several times during the course of our discussion. The requirement of compatibility with liquid lithium restricts the use of stainless steel to about 500°C or lower. This rules out stainless steel as a candidate material for advanced systems, but it was pointed out that prototypes must be built long before power reactor systems, and these prototypes will use existing technology. They will not require a twenty year lifetime, either. Perhaps stainless steel will be used in these prototype systems.

RADIATION DAMAGE TO THE FIRST WALL

The prime objective of this session was to evaluate the radiation damage that can be expected at the CTR first wall. The two broad classes of problems anticipated at the first wall are: 1. Loss of ductility, 2. Swelling. The most important problem is loss of ductility of the wall material. It is impossible to design a structure of the size we are talking about that does not have some relative motion between two points under the service loading. The metal has to be able to deform to accommodate the loadings due to the thermal cycling of the machine, due to the differential swelling of various components, and due to other load cycle requirements. The irradiation-produced losses of ductility of materials can be very severe. There are conditions in which a metal with twenty to fifty percent elongation to failure when tested in the unirradiated condition sustains only 0 to 1/10th of a percent elongation to failure after irradiation. Designers cannot work with these strain restrictions. Not all materials will be damaged the same amount. We need to find out which of the different candidate CTR materials have the highest resistance to this ductility loss. There are at least two mechanisms by which ductility is reduced by irradiation. Transmutation-produced helium, acting alone, can precipitate at grain boundaries and result in severe ductility losses, especially at elevated temperatures. The metal is also hardened by the formation of displacement damage and this hardening is accompanied by a ductility loss, especially important at lower temperatures. The interaction between the helium-produced ductility reductions and the displacement damage-produced ductility reductions is not fully understood. While we do not know the details of either ductility loss mechanism, it remains clear both mechanisms will be important under CTR conditions.

I want to emphasize that the loss of ductility is one of the most important predicted effects of radiation damage on the CTR first wall. A large fraction of the radiation damage work today is concerned with the examination of microstructures. This is very important, but we still have to keep in mind that although it is not as easy to do, we have to determine the effect of irradiation on the ductility of these materials.

Irradiation is known to increase creep rates under some irradiation conditions, and this was discussed in a paper on irradiation controlled creep in the radiation damage session. It is also possible that irradiation may enhance ductility for materials deformed in the radiation field as compared to the ductility values determined in post-irradiation creep or tensile tests. This would provide some relief from the very strict limits on ductility set by post-irradiation testing. Experimental verification of any ductility enhancement is not yet available.

The second major problem is irradiation-produced swelling. There are at least two mechanisms by which metals swell in an irradiation environment; due to the condensation of helium into bubbles and due to the condensation of vacancies into voids. The two types of swelling have a number of different characteristics, but we do not know for sure how the two mechanisms interact. Cavity formation resulting in volume increases of 10 percent or greater has been found in reactor experiments and in excess of 40 percent swelling in some ion bombardment experiments. The characteristics of the swelling phenomena are only partially understood. Swelling will probably saturate after some large neutron exposure, but this expected saturation has only been demonstrated experimentally under a few specialized conditions. Stainless steel has been irradiated to fluences nearly one-twentieth of the fluence we require. This produces a large amount of damage (swelling of 11% volume increase), but there are still no indications of saturation in the swelling. It is safe to predict volume increases greater than 10% but we need to know exactly what the swelling will be as a function of first wall operating conditions. Swelling cannot go on forever, but we do not known when saturation will occur.

Some input from reactor designers would be useful in establishing how much work is justified in trying to define the parametric dependence of this volume swelling. As an example, could a reactor be designed in which a maximum volume increase in wall components of 10 percent and a differential swelling of 5 percent between wall segments can be accommodated?

Only a limited amount of data is available on the effect of irradiation under conditions of interest to the CTR. Too few materials have been studied, the fluences are too low (and thus too little damage has been introduced), and irradiation temperatures are usually below those required for CTR. Also, the transmutation product effects, especially the effects of helium, have not been adequately included in damage studies.

EXPERIMENTAL EVALUATION PROGRAM

The effect of irradiation on the CTR first wall is being investigated experimentally. Most of the work now under way is directed towards evaluating the magnitude of the problems discussed in the previous section and establishing base-line data on irradiation damage in refractory bcc metals. It is probably too soon to begin detailed screening studies for the selection of first wall materials or to try to generate engineering data on how first wall materials will behave in service. The limited effort available for this program requires that we maintain close cooperation and coordination to avoid duplication of effort.

We already dwelled on the use of fission reactors. This is of prime importance because these are the best facilities available; not because the fluxes are as high as we need, but because they are the best we have. The first requirement is the determination of the effect of irradiation on mechanical properties, keeping in mind that it is the ductility that concerns us more than any other property. We also have to examine the swelling behavior of the irradiated metals. Microstructural examination will help monitor swelling and provide basic information necessary to understand and explain the mechanical property changes. Knowledge of the mircrostructural changes produced by neutron irradiation will also be used to evaluate other simulation techniques. We need somehow to simulate transmutations. Fission neutrons do not give the same transmutation rates as do 14 MeV neutrons. We have to find ways of introducing helium, perhaps before we start irradiation, perhaps by employing alloy additions that form helium during the irradiation. Boron is far from ideal as an alloying addition but boron has a high cross section for the (n,α) reaction, and we can produce helium in this manner. We can perhaps simulate some of the transmutation reactions by direct alloying, adding zirconium to niobium, for example. Helium can also be injected by alpha bombardment. In some experiments helium and displacement damage can probably be injected simultaneously. Heavy metal bombardment from one accelerator produces displacement damage while a second accelerator focused on the same specimen is used to inject helium. Alternatively, sequential heavy ion damage and helium injection bombardments could be used.

We also intend to simulate the CTR radiation damage by ion bombardment techniques. Primarily we can look at changes in the microstructure produced by this technique. This will be used in swelling studies to screen a large number of material variables in a short time. It will also be useful in establishing the dependence of microstructural changes or irradiation parameters, including bombardment temperature, damage rate, and damage concentration. The technique may also have some value in determining expected CTR microstructural damage and using this information to predict mechanical properties changes. However, this usefulness in predicting properties assumes development of better models to explain radiation damage than the models presently available. The ion bombardment techniques will be very useful as long as the limitations of the technique are always kept in mind. These limitations include rate effects, surface effects, lack of direct production of transmutation products, and the general inability of the technique to evaluate mechanical property effects.

There is one mechanical property that we can measure during ion bombardment, that is irradiation enhanced creep. Some work at Argonne is very helpful in this direction. A material in which the damage microstructure has been established by previous neutron irradiation is exposed to a beam of deuterons to generate the concentration of vacancies and interstitials typical of reactor irradiation. The specimen is stressed and the rate of creep deformation compared for beam-on and beam-off conditions. Another use of ion accelerators is in neutron generators. They may be useful in some cases but there seems to be some questions as to whether the flux level will be high enough to simulate the CTR environment.

Another major problem to be investigated is the effect of helium in metals. There is not very much information available on the behavior of high helium concentrations in metals. Accelerators and the "tritium trick" can be used to introduce the helium. Accelerators are used by stopping a beam of alpha particles in thin targets. The tritium trick is something new. Refractory metals will dissolve large amounts of tritium. The tritium is quenched in, the sample is held at low temperatures for days to weeks, and ^3He is produced by decay of the tritium. This looks like a very useful technique for introducing the quantities of helium that we need. It is effective in large samples, and can be used to dope samples for mechanical property measurements. We also need to define the diffusion rates of helium in these metals and to study the swelling due to helium alone so we know this as a component of the total radiation damage problem.

Very close attention should be paid to fast fission reactor programs. This is a very important item, because here is a "free ride" for us. There is a very large effort to develop breeder reactors. These programs use the highest flux neutron sources available and the prime positions in experimental reactors. If we monitor these programs very carefully, perhaps a small body of data on materials and conditions pertinent to CTR application can be expanded and extrapolated by use of irradiation damage models developed in the fast fission reactor development programs. Finally, theoretical and computer modeling of radiation damage may allow us some extrapolation beyond the available data, if the modeling is done correctly. The modeling work, too, is largely supported by other programs.

RADIATION DAMAGE TO OTHER CTR COMPONENTS

The main intent of this session was to consider the problems of radiation damage at the CTR first wall. A few other potential radiation damage problems were raised in discussion and it may be useful to list these and comment on them briefly. The following points were considered:

1. Resistivity increase in magnet copper.
2. Damage in magnet superconductors.
3. Breakdown of electrical insulators.
4. Swelling of graphite moderator.
5. Hold-up of tritium in structural components.
6. Radiation-enhanced corrosion.

The main difficulty in evaluating radiation effects in the magnet components will be in determining the neutron flux and spectrum at the magnets. The radiation will increase the electrical resistivity of the magnet copper during 4°K operation. The rate of resistivity increase has been reported by a number of investigators. Resistance as a function of magnet operating time can be determined from this existing data when the neutron flux has been calculated. An approximate value can be given for this effect. If the magnet coils are shielded to receive only 10^{-5} of the first wall neutron flux, one year of continuous operation at 4°K would produce an increase in the copper electrical resistivity of approximately 0.1 μΩ-cm, near 6% of the room temperature resistivity of copper. Continued operation would lead to saturation of the resistivity increment produced by irradiation, with the saturation resistivity in the range 20 to 30% of room temperature resistivity. The accumulated damage can be removed by periodic annealing. Raising the magnet temperature to room temperature would remove 80% of the irradiation-produced resistivity increment; annealing at 200°C gives essentially complete return of pre-irradiation properties.

The point defects introduced by irradiation of superconductors at their operating temperature can be expected to produce changes in the magnet properties. There are little or no data available where the irradiation was performed at the superconducting temperature so that the damage would be typical of that introduced during CTR operation. This is information that can easily be obtained in existing reactor facilities.

It came as a surprise to many of us to realize that electrical insulators are required in high flux locations of a CTR. Some data are available on insulators irradiated in the temperature range of interest. As with other data, the neutron fluences are low compared to CTR service requirements. However, limited data on alumina, the most commonly suggested insulator, show that after irradiation to 5×10^{21} neutrons/cm^2 ($E > 0.1$ MeV) in the temperature range 600 to 1000°C samples had fractured by separation of grain boundaries. The fragmentation occurred without any external loading of the insulator samples. These and other very limited data suggest that major developmental work will be required to produce insulators that can withstand the neutron flux at or near the CTR first wall.

The dimensional instability of graphite under neutron irradiation has been under investigation for many years to define graphite behavior in fission reactors. The general response of graphite is a rapid densification, followed by a more gradual swelling. Typically the densification may reach 6 to 8 percent volume decrease at a fluence of 10^{21} to 10^{22} neutrons/cm^2 before swelling begins. Swelling for several graphite grades has been observed to exceed 8% volume increase at a fluence 2 to 4×10^{22} neutrons/cm^2 at 700°C, with no evidence that the rate of swelling would decrease at higher fluences. The swelling rate is of course dependent on the irradiation temperature and grade of material. Current research programs are developing grades of graphite that are more resistant to swelling; some of these results will be of interest to possible CTR use. If a CTR blanket cannot be designed to accommodate the large swelling expected in these compact grades of graphite, there is an alternative approach that may be attractive. Since the swelling of graphite is mainly due to the development of porosity between grains of material that swell very little, dimensional stability could be maintained if unbonded graphite flour is used. This powder would be contained in metal cans with the useful life of the module probably set by the neutron damage in the can rather than in the graphite. An obvious disadvantage of this approach is in the reduced thermal conductivity in the unbonded graphite flour relative to the bonded grades of graphite. Further irradiation damage studies directly related to graphite use in CTR's would not seem to be justified until more detail is available on blanket design.

The final two subjects considered were possible effects of irradiation on the tritium held up in reactor structural components and on the corrosion of components in liquid lithium. Radiation is not expected to have any appreciable effect on the corrosion processes, but there are no experimental data avilable to support this opinion. The corrosion rates of refractory metals in liquid lithium can be measured using well established techniques. Measurement of effects of irradiation will be very difficult and have not yet been attempted. The properties of tritium in refractory metals (solubility, diffusivity, and permeation rates) suggest that tritium will not be held up in structural components. However, the interactions between gases in metals and radiation-produced cavities are complex and are certainly not understood in detail. I think the question must remain open, and the experimental solution of the problem presents a difficult challenge.

NEEDS FROM OTHER SPECIALISTS

Optimum planning of radiation damage programs in support of CTR technology will require cooperation among materials people and those people working on reactor design and other technologies related to CTR development. The range of CTR operating parameters suggested in conceptual designs is very large. Refinements in design to better define conditions, especially operating temperatures and neutron and charge particle fluxes at the first wall, will reduce the range of parameters that must be examined experimentally. For example, will the energy loading at the first wall be 10 MW/m^2 or 1 MW/m^2? Both values have been suggested. First wall operating temperatures have been discussed ranging from 500 to 1200°C. It is important to define the requirements on these parameters set by economics, reactor physics, heat transfer, and other considerations. This will help set limits within which parameters can be specified to optimize first wall material performance. Good lines of communication between design and materials specialists will continually refine the direction of radiation damage studies as design innovations and modifications are introduced.

The values of cross sections for nuclear transmutation reactions in the first wall material are important in modeling expected radiation damage. Improvements in these cross sections over the neutron energy spectrum expected at the first wall should be passed on to radiation damage personnel. The most important transmutations (from the standpoint of radiation damage) are those that produce helium. This point is emphasized because in the fission reactor programs there is a two-step reaction involving thermal neutrons and ^{58}Ni that has been recently discovered. Although completely unexpected, the reaction in some thermal spectrum reactors produces helium in stainless steel that can be approximately two orders of magnitude greater than would have been predicted two years ago. We hope surprises of this type will not be common in CTR development.

One final point to be considered is the question of tritium concentrations. It would be nice to know what tritium concentrations can be expected in various regions of a CTR to better evaluate possible interactions with structural materials.

DISCUSSION

E. R. Wells (ORNL): There are several areas in which there is a grave concern. The ductility of new materials in cryogenic service is important. We have a tendency of not taking the manufacturer's data and like to verify properties for ourselves. Facilities are handicapping us on the low temperature end. Do you have facilities available for checking the radiation damage at low temperature?

Wiffen: Since structural materials that will operate at cryogenic temperatures will be well shielded, the neutron flux will be very much lower than the first wall flux. There are a few facilities in this country that could be used to measure low temperature mechanical properties after low temperature neutron irradiation. Other irradiation and testing facilities could easily be developed in existing research reactors if they are needed.

P. J. Persianni (Argonne): You made a point on monitoring fast reactor development programs. I am a fast fission reactor man and I want to say that is a very good idea. There is a lot of information from fast reactor programs which is pertinent to all sessions that we have covered at this meeting, and all attempts should be made to utilize such data in neutronics, in critical facilities, and in materials. I thought it was a very good point.

Wiffen: I think it is a point that is natural. The CTR radiation damage programs are small enough that the same people are also involved in the various fission reactor radiation damage evaluations. I think this "free ride" on other reactor technology will be valuable.

D. Steiner (ORNL): What kind of accuracy do you feel is sufficient in calculating the transmutation rates in order to allow you to narrow in on the range of a parametric variations for your studies?

Wiffen: Of course we would like 100% accuracy. We hope we can be given transmutation rates that are accurate to better than a factor or two. Perhaps it is fair to ask for an accuracy that is about as good as that with which neutron fluxes can be calculated for the CTR.

Steiner: One of the factors that may reduce the uncertainty, in the first wall at least, is that the major contribution to the (n,α) and (n,ρ) transmutation rates comes from the 14 MeV flux and that at this energy the cross section when measured certainly has the kind of accuracy you are talking about, i.e., a factor of two.

SESSION 6

PLASMA FUELING AND RECOVERY

Chairman

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SUMMARY OF SESSION

PLASMA FUELING AND RECOVERY

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Session 6 was divided into four topics: Fuel injection, plasma processes, plasma removal, and the external loop.

Three methods of fuel injection were suggested, neutral injection, droplet or pellet injection, and a possible default option of cycling the machine (flush it out, put in a new fuel charge, go through a burning cycle, and then repeat). This latter idea, however, poses more mechanical problems due to thermal cycling, not always compatible with long lifetime.

For perspective on the injection problem, a typical hypothetical reactor has the following parameters: densities, a few times 10^{14} particles/cm^3; confinement times of the order of a second; and volumes of the order of 10^8 cm^3, i.e., total ions in the plasma approaching 10^{23}. Since this must be fueled and removed in about 1 second, equivalent currents approach 10,000 amperes, and the equivalent pellet injection for hydrogen isotopes is approximately 1 cm^3/sec. These numbers vary and are not rigid from one machine to another.

Speaking on neutral injection, L. D. Stewart (Oak Ridge) described the work he, W. L. Stirling, and O. B. Morgan have done on the duoPIGatron. They have achieved 4-ampere ion sources in the 25-40 kV range, operating with a 10% duty cycle for 0.1 second. These sources are limited by their power supplies, following the $V^{3/2}$ law for the first half of the operating range until encountering arc current limitations. Thirty percent of the beam falls within a one degree half angle without a magnetic lens, and about fifty with.

R. F. Post (Livermore) reported on the neutral injection work done there by J. E. Osher, R. V. Pyle, and others. Although the state-of-the-art is essentially 1 ampere, they are optimistic that they can reach 10 very promptly and retain good angular divergence. His 100 MW conceptual mirror reactor requires only 500 amperes of injection, and Post suggested that 10 units of approximately 50 amperes each appears reasonable for future development.

Discussing the range of energetic neutrals in fusion reactor grade plasmas, he noted that a 10 keV neutral will penetrate approximately 15 cm, whereas at high energies, a 250 keV neutral goes about 120 cm. This is fine for mirrors, but rather unsatisfactory for closed geometries.

He also mentioned the problems of generating these beams with high neutralization efficiency, stating that this can be done with negative ions (D^-) with about 90% efficiency, due to the low binding energy of the electrons.

C. D. Hendricks (Univ. of Illinois) described his excellent droplet acceleration technique and showed motion pictures of controlled collisions. He can accelerate any size particles over a range of five orders of magnitude (0.01 -1000 microns) with near-precision control to various velocities (e.g., 0.01 μ Ne to 8 x 10^6 cm/sec). He believes 10^7 cm/sec D_2 is possible. Positively charged droplets can hold about 10^8 V/cm field gradient at the surface and negatively charged, about 10^7 V/cm.

As illustrated in the film, the droplets that are produced are extremely uniform in properties and, when accelerated, the impact parameter can be controlled to 0.1 micron. Considerable discussion followed as to whether the pellets, despite their proven performance, will be able to penetrate deeply within the plasma. It was unanimously agreed the question remains unanswered at this time.

Further remarks on this problem were made by F. H. Tenney (Princeton) who reported on work done there in 1954 by L. Tonks, and D. J. Rose (MIT) who spoke of work accomplished at Culham. E. Thompson (Culham) joined in the discussion, a summation of which follows. A thesis in preparation at Columbia by a student of R. A. Gross, S. L. Gralnick, is devoted to a part of this problem.

The problem is very complicated. When the droplet encounters the high energy density present in the plasma, a shock wave or a deflagration wave propagates through the solid, and a sudden generation of high density, low energy plasma follows at the surface of the particle. Ionization and charge exchange take place within this region and, under the influence of the magnetic field, plasma can spread along the field direction. Locally a $\beta = 1$ plasma may form, excluding the magnetic field and preventing further energy influx from the plasma.

If the droplet "blows a hole" in the plasma, does this act against the magnetic field structure of the system and suddenly stop, or does the field flow around the particle in a hydromagnetic manner and allow free particle propagation across the field and out the other side? An overall answer in principle to this question notes that the plasma with which the pellet is interacting is magnetically confined and is limited in its transport properties across the field. Determination of the amount of energy or its maximum rate of transfer from the plasma into the particle indicates that if the particle is big enough, it can't be evaporated and will go on across. As the size is scaled down, the rate of energy transfer will drop off by the square of the linear dimensions; however the needed energy drops off by the cube. If the particle is small enough, it won't go through.

There is a range in size, as yet undetermined, which would allow particles to be injected from either side by means of Dr. Hendrick's method and have them stopped in the middle by mutual collisions. The risk here is that these particles may be so large that an extremely large perturbation in the plasma will result with accompanying disturbance of the confinement system. Experiments in this area would be very valuable.

There were three speakers on plasma processes. T. Kammash (Univ. of Michigan) discussed charged particle heating; M. Ohta (JAERI), reactor plasma thermal instability and control; and J. R. McNally (ORNL), higher Z fusionable nuclei.

Dr. Kammash reviewed some of the classical work on the cooling rates of a test particle in a field, extended them somewhat, and showed certain modification to the rates of cooling.

Dr. Ohta pointed out that if a plasma is operated below a certain critical temperature, it becomes unstable, and some type of feedback mechanism is needed to maintain optimum operating conditions. This work is in its initial stages, and an important factor yet to be considered is the radial dependence inherent in this problem. Since one is not dealing with an infinite homogeneous medium, one must examine how the control of this instability varies as a function of the radius. This appears to be closely associated with the injector problem.

Terming his scheme the "wet wood fusion burner", Dr. McNally considered systems that cannot sustain themselves independently but require a steady source of energy from the outside. Under these conditions one can consider a wide variety of fusion reactions among the elements with Z slightly higher than those of the two hydrogen isotopes and the one helium isotope on which fusion power forecasts are usually based.

Dr. Post spoke rather briefly on direct conversion. His unbalanced mirror plan takes material from one end of the machine only. A mirror that is slightly stronger on one end than the other is used, and preferential losses through the weak mirror result. Weak losses from the strong mirror are partially returned by an electrostatic system, a method of suppressing the leak from the strong mirror to provide a single exiting stream to feed the direct conversion apparatus.

Divertor design problems were discussed by G. H. Miley (Univ. of Ill.) and stimulated a great deal of audience participation. He noted that the divertor, which is subject to a wide variety of design problems as yet unsolved, appears to be the critically important part of a closed reactor system. Although a divertor has been operated successfully in the Princeton stellarator, this is not necessarily relevant for a fusion device. The Model C divertor actually did transport higher particle flux rates than will be necessary in a reactor, but they were cool plasmas, and it was a transient, not a steady state, system. Other problems include neutron shielding (escape to the outside world is possible unless the open passage from the plasma is curved) and pumping. Higher pressure in the divertor chamber will result in a flux back into the system. Some of this will be ionized by the exiting stream, but some will creep along in the dead region between the plasma and the wall. Perhaps the divertor will prevent the bombardment of the wall, but, on the other hand, the question of injection is very closely associated with wall problems because the neutral pellets coming into the plasma from the surface will introduce large quantities of neutral gas, at least in a transient manner, into the plasma. Ionization is a rapid process, but so is charge exchange. Much more fuel will be introduced than burned. Depending on the model, approximately 0.5 - 2.5 percent only will be consumed per cycle. If a substantial amount of the excess fuel becomes charge-exchanged and neutralized near the surface of the plasma, this may bombard the internal wall, the divertor notwithstanding. Thus the injector, surface problems, and the divertor are all associated.

Dr. Miley showed conceptual designs of several different types of divertors. Among these were a transverse (or toroidal) model, which he called the classical divertor; and a longitudinal (or poloidal field) divertor. He also presented conceptual sketches of Tokamak types including M. Yoshikawa's suggestion of a triangular-shaped aperture for the plasma in which the three corners are not completely symmetrical. One represents a separatrix, and the other two lie beyond the plasma, thereby providing a continuous divertor removing the plasma from the reactor volume.

E. F. Johnson (Princeton Univ.) gave the only talk concerning the external loop. He pointed out that equilibrium constants of any process are important, but even more so are the reaction rates. If fuel is to be circulated rapidly, the reaction rates must be examined to make certain that the chemical system being designed is realistic.

Professor Johnson does not feel the separation problem will be difficult, nor that the tritium inventory will be critical. Control of the total inventory may be based on time requirements: how many seconds, days, or hours of fuel are needed behind the injector. There may be a lower limit of a few hundred grams of tritium, but the decision as to how much reserve is required might raise this to 1 or 2 kilograms. Although this inventory may seem cause for concern, it should be remembered that it will be located in a small region of the plant and consequently can be protected to rather high safety factors without excessive expense.

In summary, what is the outlook for fusion power reactors from the standpoint of the problems of the fuel cycle? Adopting a highly conservative position, one can say that we don't know how to get the fuel in; we don't know how to get it out; and we are not very sure what will happen to it while it is inside; but if we carry out these operations, our chemical engineering friends will separate the exhaust and give us back the proper mix for reinjection.

DISCUSSION

P. PERSIANI (ANL): In the pellet injection scheme, you said a large pellet goes through the machine, and too small a pellet doesn't allow enough energy to be deposited.

MILLS: It will be evaporated very rapidly. It doesn't need much energy, so it won't penetrate far.

PERSIANI: Am I right in picturing this: There is an optimum size that would be partial to the surface to volume ratio. Has anyone attempted to make a stab at determining this optimum size?

MILLS: I don't know whether Ernie Thompson would like to comment on this or not. The calculations have been back-of-the-envelope type calculations so far.

S. BLOW (AERE Harwell): When you think about ice pellets going into a plasma, I can't help thinking about the poor snowball in Hell. Although one mentions velocities of 10^7 cm/sec for presumably small liquid pellets, when you think about the size of the particles that you are probably going to have to put into the plasma, then it is not going to be nearly so easy to accelerate them to such high velocities. I was wondering, are there any new thoughts about this question? As I recollect, John Chubb and David Rose, two or three years ago, were talking about velocities of 10^6 cm/sec for an ice pellet, and you've got a big problem if you are going to get up to these velocities for a large pellet.

MILLS: The high velocities apply only to small particles, and at the larger particle sizes the velocities we talked about were $\sim 10^5$ cm/sec, which are, I think, probably practical.

SESSION 7

ENGINEERING DESIGN OF MAGNET SYSTEMS

Chairman

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STATE-OF-THE-ART PRESENTATION

SUPERCONDUCTING MAGNET TECHNOLOGY*

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INTRODUCTION

The rapid growth of superconducting magnet technology that has occurred in the last decade stems from the discovery by Kunzler et al.,[1] that Nb_3Sn wire can carry high current density at 88 kG. At this tenth anniversary, we will briefly look at the most recent advancements in materials and magnets in this field. The history and other aspects can be found in a number of recent and excellent reviews.[2-8] Before proceeding on a discussion of superconducting magnets, let me briefly mention that only in the last couple of years (with one exception) have superconducting magnets been used in fusion research, but the truly rapid progress in this technology almost assures a more extensive utilization in the coming years.

SUPERCONDUCTING MAGNET MATERIAL

In Table 1, I have listed the commercial material that is now in production and use and some other interesting future possibilities for superconducting magnets. The materials are listed in order of increasing value of the upper critical field. At the moment, the most popular materials are the ductile NbTi alloys and ternaries based on the NbTi alloy system for use below about 85 kG and the brittle compound Nb_3Sn for fields in the range 80-150 kG. Only very recently has V_3Ga been available, and curiously enough, although the upper critical field of V_3Ga is lower than Nb_3Sn, it has much higher current density in the field range above about 120 kG. Among the many possibilities for new magnet material, we mention PbBi alloys which are embedded in porous glass (pore sizes of order 40 Å) under pressure and exist as very fine fibers. It has not yet been produced in long lengths and thus remains to be tested as a magnet material, but it does have useful current density at fields as high as 80 kG, and it also possesses a very high normal state resistivity which may be a desirable property for special applications, e.g., persistent switches. Niobium nitride, though very brittle, has been measured in thin film form and shown to have the highest pinning force of any

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

material measured to date.[9] We also mention the newest discovery by Tachikawa and coworkers: $V_2Hf_{0.5}Zr_{0.5}$, a C-15 type compound which apparently is the first material of this class to have such a high critical field and like the other high field materials (since BBC, B-1, and A-15 structures can be described as cubic) has a cubic symmetry. Whether it can be fabricated into wires and be applicable for superconducting magnets remains to be demonstrated. The V_3Si is listed only because of its high critical temperature and high critical field. No one has produced material with reasonable current density at high fields, and like the other A-15 compounds, it is very brittle. It thus offers no apparent advantage over Nb_3Sn. The last two in the table are the most interesting discoveries particularly because of their high critical fields. As we discuss later, both are being developed for possible magnet material.

In Table 2 most of the many commercial manufacturers of available magnet material have been tabulated. The material used most extensively to date for superconducting magnets has been NbTi so we shall take a closer look at this alloy system before proceeding to the high field materials and more recent developments in materials. The critical temperature and the upper critical field are shown plotted against at. % Ti for the NbTi system in Figs. 1 and 2, respectively. From the latter graph, one sees that the upper critical field is above 100 KG from about 35 at.% Ti to the limit of the cubic phase which occurs about 78 at. % Ti, and in fact commercial material is available from one manufacturer or another over most of this range. There is no one composition which is optimum owing to the complex dependence of current density on field and composition. It turns out that the materials with lower at.% Ti are best at high fields, and materials with a high concentration of Ti are more favorable (higher current density) at lower fields so the particular application would dictate which exact alloy composition one would wish to use. The most widely used compositions in this country have been Nb-56% Ti, Nb-64% Ti, and Nb-78% Ti. The individual manufacturers have optimized the heat treatment for these compositions to obtain the maximum critical current density.

Some of these features are shown in Fig. 3 where the most important parameter from the standpoint of the magnet designer, the short sample current density, is plotted against field for a number of commercial NbTi superconductors. The dimensions, amount of copper, and other details are listed in Table 3. A Calculation of the intrinsic current density (NbTi only) from this data shows, as expected on the basis of composition, that the Cryomagnetics material (high at.% Ti) has higher current density at low fields and lower current density at high fields than the Airco and Supercon material.

In comparing current densities of rectangular conductor with square or round conductor, we took the short sample current with the field perpendicular to the broad face of the rectangular conductor. In general as is indicated by the data in Table 4. for two Supercon conductors,[10] the critical current is higher with the field parallel to the broad face than when it is perpendicular. In both cases the field is transverse to the transport current. The difference in current density between the two directions of field depends on the aspect ratio (width to thickness of the normal matrix) and the distribution of the superconducting filaments, but it can be as large as 40% for material with aspect ratio of only 2:1.

More recently the trend has been to even finer diameter superconducting filaments and at present twisted multifilament NbTi with strands of from 2 to 10 μm are readily available from more than one manufacturer. In addition because of the increased stability associated with the small size and the fully transposed braiding, a material with less copper to superconducting ratio can now be employed in magnets yielding higher overall current densities than was attainable only a few years ago. Figure 4 shows j vs B for the latest thin multifilament NbTi at different copper to superconductor ratios from two manufacturers: Air Reduction Company (Airco) in this country and Imperial Metals Industry (IMI), England. Other manufacturers are not appreciably different. The Nb_3Sn data are shown for comparison, and it is much higher than any of the NbTi compositions over this field range. Note also that NbTi, when it is used in fully stabilized configurations, often employs copper to superconductor ratios of 10:1 or even higher which would reduce the current density much more than is shown in any of these curves. If we were to compute the cost per A-ft. for this material, NbTi is more economical below about 85 kG.

Another recent advance in the NbTi wire is the successful bonding of NbTi in an aluminum matrix.[11] Because Al has a smaller magnetoresistance than copper as well as a low resistivity, this means that a smaller aluminum to superconducting ratio can be tolerated for fully stabilized operation and hence a greater overall current density can be realized with an Al bonding than with Cu. On the other hand, Al is not as strong as Cu and cannot withstand the same stresses.

The short sample critical current density of high field materials as a function of field is shown in Fig. 5. Both the GE Nb_3Sn and the V_3Ga from Japan are in ribbon configuration. The dimensions are shown on the figures in millimeters. The first figure is the width; the second is the thickness. The thickness of a typical Nb_3Sn composite consists of a 1 mil core comprised of diffusion reacted Nb_3Sn (5 to 10 μm) on both sides of Nb, 2 mils of copper and 1 mil of stainless on both sides of the core--all bonded together by solder. The Sumitomo conductor consists of V_3Ga layers (approximately 10 μm thick) on a vanadium core with about 1-1/2 mils of copper on each side. A small diffusion layer separates the V_3Ga from the copper, and the entire composite is covered with Mylar insulation (about 7 μm thick). The Nb_3Sn and the V_3Ga from Sumitomo are commercially available, and the V_3Ga from Vacuum Metallurgical will soon be. The V_3Ga designated Tachikawa was made in Japan and measured by Iwasa at the U.S. National Magnetic Laboratory. One has to be careful in comparing these current densities because of instabilities in transverse fields when the width of the ribbon gets too large. The only commercially available V_3Ga at present is the Sumitomo material which is shown in the graph. Its current density is well below that of the small quantity produced by Tachikawa and measured by Iwasa under carefully controlled conditions. It is hoped, of course, that the production material can be increased to the laboratory experimental values in the near future. Even so, as shown in this figure, the

Sumitomo material is better than the old GE copper-coated material which was available up to last year. It is also better than the latest, newest improved type GE material bonded to aluminum above 140 kG so this illustrates in a clear manner that although V_3Ga has an upper critical field smaller than Nb_3Sn, it may prove to be a more useful magnet material above 140 kG. The Vacuum Metallurgy material is superior to Nb_3Sn above approximately 120 kG. One word of caution in using this figure for magnet work; it is known that ribbons are very susceptible to instabilities in perpendicular magnetic fields.[2, 12-14] Such fields, for example, are present as radial field components at the end of a solenoid. It has been found that additional high conductivity normal material, such as high purity Cu or Al can be used as a magnetic damper to stabilize the ribbon against flux jumps initiated by transverse fields.[15,16] This is the so-called dynamic stabilization technique which has been known for a long time but only recently has come to be utilized to any extent.[17,18] The amount and thickness of the Al necessary to insure stability depends, of course, on its resistivity and on the width and thickness of the superconducting material and the maximum perpendicular field and its rate of increase, so one would have to consider the superconducting material plus Al stabilizer before comparing overall current densities at any particular field value. Not shown on this figure, but recently available in the commercial market, is an Nb_3Sn conductor formed by plasma spray[19] rather than the diffusion process used by GE and the vapor deposition technique pioneered by RCA.

It is the hope of many workers in the field that Nb_3Sn and V_3Ga as well as other high field materials will be developed into a flexible wire form with thin filaments, and the latest developments in this area look promising. Multifilament Nb_3Sn (diam = 0.043 cm) with up to 73 filaments each having a diameter of 10 μm has been produced by Kaufman and Pickett[20] of the Whittaker Company, but at present the work is proceeding slowly. Filamentary Nb_3Sn research is also being conducted by Suenaga and Sampson at Brookhaven.[21] A multifilament wire form of V_3Ga has been produced by Tachikawa and coworkers[22] as well as at Brookhaven,[23] and the results look promising. The material is prepared by embedding a vanadium rod into a CuGa alloy tube. The composite is cold-drawn into a wire and then heat treated. The formation of V_3Ga

leaves a coating of copper on the outside which is ideal for stability and low loss connections.[22] The newest material and potentially the most important of the new technology is the recent work on Nb_3Al in ribbon form which was produced by Rose's group at MIT.[24] The material prepared in ribbon form has high critical temperature and high critical field, but no data on the critical current density have been reported yet for any field value. Similar techniques might also yield the other very high critical field material $Nb_3(Al_xGe_{1-x})$.

Before proceeding to a discussion of magnets, what are the prospects for further discoveries of high field materials in the near future? If we look at the date and discovery of each new increase in critical temperature of superconductors over the last 60 years, we discover that there is a new advance each decade in the raising of the critical temperature. The data are tabulated in Table 5 and plotted conveniently in a linear scale in Fig. 6. One sees that if the trend continues we can expect a new high temperature superconductor above the boiling temperature of liquid hydrogen within the next 8-1/2 years. Of course, if these data are plotted on a semilog graph, the increase in T_c would appear much smaller as the graph then resembles a saturation curve which is more likely the true course of events until a radical departure generates a step discontinuity. Linear extrapolations can be deceptive. In general, superconductors with high T_c also have a high upper critical field. Some question has been raised as to whether there really is value in increasing the critical temperature a few more degrees even above the hydrogen point because it is the critical current density and critical field at the operating temperature which is important. Hence, if the critical temperature were just a few degrees above the hydrogen point (21 Kelvin), we would still not expect to have a very high critical field and high current density in liquid hydrogen, and such materials would still have to be operated in liquid helium. Even if we assume that the critical temperature could be raised to 30 or 35 K, there is still some question as to whether one would risk the inherent safety problems of operating a magnet in liquid hydrogen or sacrifice the additional refrigeration cost and take the convenience and safety of liquid helium or helium gas.[25] Of course, in this event, the high T_c material would be more easily stabilized because one would not have to be so concerned with the transition from nucleate to film boiling with the accompanying temperature jump from 4.2 K to around 20 K provided that the critical field and critical current density are still not exceeded at this temperature. All in all one would consider the effort well spent to increase T_c provided that increase in j_c were not neglected.

SUPERCONDUCTING MAGNETS

A. State-of-the-Art

Before discussing superconducting magnets, I should like to show three figures--one of the conductor, the second of the winding machine, and the third one is one half of the solenoid of the new CERN bubble chamber magnet.[26] This magnet is a very impressive piece of engineering. It has been completed and is scheduled for testing some time before the end of the year--November 1971 is the projected date. Figure 7 shows the conductor of the CERN magnet. The components are as follows: One is the NbTi conductor in a copper matrix 6.1 cm wide and 0.3 cm thick. Four is the Al heating strip 0.01 cm thick which is used to remove the residual field (frozen in flux) for calibration work. Seven is a 0.2 cm thick 316L stainless steel reinforcing strip. Eleven is a very involved copper ribbon cooling strip 0.19 cm thick machined out of a single ribbon of copper. Eight and ten are insulation for the first turn only; all the others are thin insulators of either polyester sheets or polyamide film. Figure 8 shows some of the multilayer strip being wound onto one of the pancakes. The involved winding operation and cost is easily appreciated from such a picture. Figure 9 is a picture of one completed section of the CERN bubble chamber magnet. This magnet, the largest completed to date, has stored energy of over 800 MJ or one order of magnitude larger than the successful Argonne bubble chamber magnet. The design of this magnet is sufficiently conservative that a field of 50 kG could be generated were it not for the attractive forces between the two halves exceeding the machanical strength of the supports.[27]

In the following tables, I have listed most of the high field and physically large coils completed in the last few years, but these tables are by no means complete. They are just selected for illustrative purposes. Table 6 shows the high field magnets which have been built. The highest field produced to date is the 150 kG solenoid with a small bore built by RCA utilizing their own vapor-deposited Nb_3Sn ribbon which they no longer offer commercially (the material will shortly be available from a new Canadian firm, CSCC). A few coils using GE diffusion bonded ribbon have achieved fields in the 140 kG range. The largest coil which has produced more than 100 kG in the working volume (117 kG in an 18 cm bore) is the magnet HYBUC built by American Magnetics, Inc., for a small bubble chamber; the maximum field at the windings was calculated to be 123 kG. It is worthwhile to note that no high field magnet, i.e., over 100 kG anywhere in the material, has ever been built with a bore larger than 0.51 m. We emphasize this point because fusion reactor magnets will require large working volumes and high fields at the windings. The last magnet listed in this table uses the latest material V_3Ga, and as we shall see below there are other magnets being planned which will also use this material.

In Table 7 the very large magnets of axial symmetry are listed. Here the important point is that few magnets over 1 m in bore size have achieved their design value after the initial construction and testing. Clearly the first magnet listed in the table, the Avco-NASA Lewis magnet built for fusion research, must be considered one of the most successful large magnets built to date. It has achieved the combination of both very high fields and very high stored energy. The RCA coil assembly also built for NASA Lewis did not achieve its design value even after pumping below the λ point (2.2 K) of liquid helium. The Saclay magnet BIM needed to have one section rewound, and then it performed according to specifications. The protection circuit was very well designed, and tests indicated that more than 98% of the stored energy could be removed and dissipated in an external resistor on quenches initiated by exceeding the critical current. The Argonne bubble chamber magnet did not quench, and it is fully expected to meet the design value, but the definitive tests will not be carried out until some stress calculations involving the structure are completed. It worked at a high enough field 18.5 kG to satisfy the bubble chamber experimenters, and the prevailing opinion was that there was no need to risk structural failure by going to the design value of 20 kG. The Brookhaven bubble chamber coil quenched just before reaching its design value, but at this moment it can be considered a successful coil as the field reached 28.2 kG is sufficient for bubble chamber operation. The large sizes are accompanied by increased forces which require more careful designs and construction considerations to avoid premature quenching of a superconducting magnet.

In the next table, No. 8, we turn our attention to nonaxially symmetric magnets. With the exception of Baseball and IMP, all nonaxially symmetric coils designed and built to date have maximum fields less than 50 kG. Although neither Baseball nor IMP has been fully tested, it is again worth noting that the initial test for both did not reach the design value. Of course to be fair to the designers, it must also be pointed out that they were not afforded a proper amount of time for testing, and large coils often suffer premature quenches as a result of the "bedding down" resulting from the large electromagnetic forces. The coils were designed for use in plasma experiments and at present operate at a field value sufficient for the initial plasma physics work. The definitive test will come only when all the low field plasma experiments are completed. Some difficulties were encountered in the construction of the Jülich magnet "Argas" also, and it had to be rewound. The second model, although expected to reach the design value, suffered a premature quench during its first vigorous testing. Difficulties are now believed to be due to an incompatibility between the coil and the power supply. Serious oscillation problems between a highly inductive superconducting coil and the capacitive output of some solid state power supplies are well known and often can be avoided by shifting the resonance point through introduction of resistance into the lead circuit. The fairly recent introduction of improved NbTi utilizing twisted multifilaments of small diameter in a transposed braid shows that high current densities are readily obtainable in relatively small, low field magnets. In addition to the successful d.c. dipole (bending) and quadrupole (focusing) coils, there is much work on a.c. dipole coils for synchrotron magnets mainly at Brookhaven, LRL (Berkeley), Rutherford, Saclay, CERN, Karlsruhe, Grascati, and the Radiotechnical Institute in Moscow. Pulsed 50 kG superconducting coils for synchrotron application have been built operating at high current density and energizable at rates in excess of 100 kG/sec without quenching. We fully expect that large magnets made from this new improved material, even though only partially stabilized (adiabatic stabilization, see the review paper by Hancox), will be realized in the near future. Of course, large magnets have large forces present and the same high current densities will not be attained; some decrease in this parameter will have to be accepted to offset the increased size.

Table 9 lists all the large magnets that have been projected for the near future. Both the NAL bubble chamber magnet and the CERN "Omega" coil are expected to be completed on schedule with the Omega magnet to be tested by November of this year. This latter magnet uses hollow superconductors cooled by forced circulation of compressed helium. The electrical circuits for the various pancakes are in series while multiple parallel paths are used for cooling. The insulated joints that are able to sustain high pressure in a vacuum are also a unique development by Morpurgo. The Garching stellarator is an ambitious project, and it is the first large system employing nonstabilized coils. The stored energy is high, 50 MJ, and the maximum field is fairly high, 65 kG. This system as it is presently conceived will be a major advance of the state-of-the-art. Two very high field, small bore coils are being planned in Japan, both utilizing V_3Ga ribbon conductor. The ease with which these magnets meet their design criteria will be an indication of the stability and usefulness of V_3Ga in further magnet projects. In addition to the superconducting magnets already mentioned which are to be employed in fusion experiments, a number of levitated superconducting coil systems are either under construction or already completed. A summary of these machines which are presently under development at Culham, Princeton, LRL (Livermore), and Garching is shown in Table 10.

8. Stored Energy and Current Density

Data for the total stored energy $E_s = (1/2)\ LI^2$ as a function of the magnet current density $\langle j \rangle$ for all the known large superconducting magnet systems presently in operation, under construction, or in the design stage are shown in Fig. 10. Since the practical operating field for each superconducting material is fixed, the stored energy is just another measure of the overall size of the magnet. Similarly the operating current density greatly affects the size and weight of any magnet design. The line marked achievement boundary shows the envelope of magnets that have been successfully tested as of March 1971. It is to be noted that the area under the curve has been increasing each year. As mentioned previously, the largest magnet in operation up to the present time is the 4.8 m bore, 18.5 kG, Argonne bubble chamber magnet with a stored energy of about 80 MJ which is operated at a conservative design value of $\langle j \rangle = 800\ A/cm^2$. This magnet was wound with multifilament NbTi in a copper matrix. The largest magnet built but as yet untested is the CERN bubble chamber 4.7 m bore, 35 kG magnet with $E_s = 800$ MJ and $\langle j \rangle = 1.1\ kA/cm^2$. The highest field so far produced by a superconducting magnet is 150 kG in a 4 cm bore coil made from a Nb_3Sn ribbon composite no longer commercially available.

For magnets with stored energy less than about 20 kJ, the design value of average current density $\langle j \rangle$ in the range 20 to 60 kA/cm^2 has been achieved mainly in coils utilizing very thin, twisted, multifilaments of NbTi in a low copper to superconducting matrix which is stabilized against degradation due to movement by vacuum impregnating the coil in epoxy, oil, grease, wax, etc. Some magnets with stored energy in the range of 60 kJ to 1 MJ have failed to operate at the design current although understandably it is hard to get complete data concerning such failures. In any event it is known from practical experience that large magnets necessitate operation at lower overall current densities than small magnets producing the same field. For economic reasons and ease in handling, one would like to have design criteria for operating at the maximum possible current density and obtain a coil of minimum size and weight for a particular field. Figure 10 presents an incomplete picture in that there are some valid reasons why large magnets are designed to operate at lower current density. The intrinsic superconducting current density decreases with increasing magnetic field, and thus high field magnets (i.e., the field at the windings) must be designed for lower current densities than low field magnets utilizing the same material. Also, and more important, is the unavoidable need for additional structural support in large magnets to contain the hoop stresses and axial compression forces which result in a lower overall current density than small magnets producing the same field. However, besides these two undeniable facts, we will show in the next section that the very principle utilized for the design of large stable magnets invariably leads to a lower current density than would be dictated by considerations of the magnitude of the field alone.

C. Relation Between Size and Current Density

All of the magnets built with a stored energy above 10 MJ and many in the decade 1 to 10 MJ are completely stabilized (cryogenic stabilization), i.e., the superconductor is embedded in a sufficient amount of high conductivity normal material that a local, normally conducting zone either arising from a local disturbance or created by exceeding the critical current of the superconductor will not propagate. Barring failure of the cryogenic, vacuum or power supply system or some mechanical breakage, such magnets will not quench and dissipate their stored energy. This technique, coupled with the need for structural support to contain large hoop stresses, precludes operation at large overall current densities, at least with the presently available commercial superconductors and normal matrix materials. One might almost classify these magnets as cryogenic coils which operate at 4.2 K without joule dissipation because of the shorting by the superconductors. When reliability is of prime importance, as in an electric utility or bubble chamber application, and cost and size are secondary, a completely stabilized magnet is the proper choice. Extrapolation of the present results to larger size magnets must be done with extreme care for experience has taught us that each significant increase in magnet technology to date has been accomplished only after overcoming difficulties that were not always anticipated.

The conservative design approach assumes that a local disturbance such as a flux jump (a collapse of the induced magnetization currents and a sudden penetration of the magnetic field) leads to a local temperature rise above the critical temperature of the superconductor for that field and current value. Since the normal state of superconductors is one of very low electrical conductivity, this causes all the current in the superconductor to be shunted into the low resistivity normal matrix. This normal material must then have a high enough thermal conductivity and a sufficient area in contact with the liquid helium bath that all the joule heat produced can be adequately transferred without exceeding the nucleate boiling limit between the solid surface and the cooling liquid. Exceeding this limit, or critical heat flux, results in a transition to film boiling and subsequent thermal runaway. If the temperature rise associated with this power level is less than 1 K, in most cases the normal zone will not propagate but will collapse as the current transfers back to the superconductor, but only if the transient disturbance occurs at a current level below the superconducting critical current which is a function of both the field strength and the temperature. In actual magnets with narrow cooling channels, the critical heat flux value is more practically given by the nucleate boiling heat flux which remains below the value of the heat flux associated with the transition from film to nucleate boiling, and this has a maximum value in restricted spaces of about $\dot{q} = 0.4$ W/cm^2. By choosing a nucleate boiling heat flux below the film boiling value, complete stability is assured. This is the only type of stabilization technique that assures magnet operation at the short sample current value. Surface condition and geometrical considerations are significant parameters in determining the precise value associated with the heat transfer to liquid helium and the collapse of film boiling. A number of recent studies have been directed toward measurements of heat flux in channels of various kinds at liquid helium temperatures.[28-34]

In any magnet project an important design parameter that precedes most other assumptions is the overall current density. We anticipate that the size of the magnet needed for a fusion reactor is at least three orders of magnitude larger than any superconducting magnet that has been built and tested to date. In view of what has been stated above, what can we assume to be a reasonable current density with some measure of confidence that we are not overlooking some pitfall? The following discussion and analysis will attempt to answer this important question. In attempting to find the allowable current density for extremely large magnets, we shed some light on why magnets with large stored energies require low current density design values quite apart from considerations involving structural support.

The condition for complete stability, the power balance between joule heat produced and heat transfer to liquid helium is given as

$$\frac{I^2 R_n}{S} \leq \dot{q} \tag{1}$$

where S is the surface area in intimate contact with the helium bath, $R_n = \rho_n \ell / A_n$ is the resistance of the normal matrix with field dependent resistivity $\rho_n(H)$ and cross section area A_n, and $\dot{q}$ is the critical heat flux which we will assume from here on to be the film boiling heat flux in long narrow cooling channels. The maximum operating value for I is given by using the equal sign in Eq. (1). Although not necessary for the following arguments, it will simplify the expressions and be sufficiently accurate to assume that the ratio of normal to superconducting material is large. It will also be convenient to assume that the compound conductor is round.

Montgomery has analyzed the case of a round conductor with large copper to superconducting ratios to determine the theoretical maximum possible conductor current density for fully stabilized operation in a coil considering various overall diameter wires and assuming reasonable values for insulation and interlayer spacings.[35] He concluded that for small coils utilizing small conductors, a maximum stable winding current density from 13 to 17 kA/cm^2 was achievable. We assume the mean value 15 kA/cm^2 to be a realizable maximum for a coil with stored energy in the range of 10 to 20 kJ which is the largest completely stabilized size where one would estimate that this high current density could be achieved. It will be noted that nonstabilized magnets also start to show degradation above this size. We wish to design and compare a series of completely stabilized simple solenoids each of rectangular cross section and uniform current density. To simplify matters the inductance and the magnetic field will be kept essentially constant. The series of coils is obtained by increasing the bore, operating current, and size of the compound conductor. If we start with a physically small solenoid of bore diameter 2 a_1 operating at a modest current I carried by a small diameter wire and compare it with a larger solenoid (increased a_1) operating at a higher current (increased I) carried by a larger conductor (increased D) but otherwise constructed of compound conductor having similar cooling properties and resistivity of the matrix, then the answer we seek is how does the increase in stored energy compare with the change in overall current density without violating the complete stability criteria?

Utilizing Eq. (1) a useful expression can be obtained for the maximum current of a stabilized conductor. For any conductor the cooled surface per unit length S/ℓ can be expressed in terms of the total area as $k\sqrt{A}$ where k is a dimensionless constant independent of A varying between approximately 0.75 and 3 for practical cases.[36,37] For example, for a circular wire $k = 2\sqrt{\pi}\, f$ where f is the fraction of the surface in intimate contact with liquid helium, for a square conductor $k = 4f$. and for a rectangular conductor with edge cooling only $k = 2f/\sqrt{\eta}$ where η is the aspect ratio, width to thickness. With these substitutions and assumptions, a relation between the current and conductor current density at the maximum operating point can be obtained from Eq. (1) by eliminating the area

$$I = \frac{1}{j^3} \left(\frac{k\dot{q}}{\rho_n}\right)^2 . \tag{2}$$

Had we chosen to eliminate I, we would have found that $A = j^{-4}(k\dot{q}/\rho_n)^2$, and $I = jA$ is not violated. This means that an increase of I by enlarging the conductor dimensions requires a more rapid increase in A to maintain complete stability resulting in a lower j. We are assuming here that the superconducting to copper ratio is small so that a correction $[1 + (A_{sc}/A_n)]^{-2}$ can be neglected in Eq. (2). Since the second term in the above equation $(k\dot{q}/\rho_n)^2$ is fixed once the conductor geometry and cooling technique is chosen, we can consider it a constant in our "Gedanken experiment". For a solenoid with N turns and bore diameter 2 a_1, the field and self-inductance are $H = K_1 N I f_1(\alpha,\beta)/a_1$ and $L = K_2 N^2 a_1 f_2(\alpha,\beta)$, respectively. If we now keep both α the ratio of OD to ID and β the ratio of length to ID constant, the series of solenoids can be envisioned as ones with I increased by some factor ν, a_1 increased by $\nu^{2/3}$, and N decreased by $\nu^{-1/3}$. Thus if ν is assumed to cover a range of 10^3 (i.e., the conductor enlarged so that the current density is decreased by ten, the current carrying capacity is increased by 10^3), a_1 must increase by 100, and N must decrease by ten for L and H to remain constant which are not unreasonable range variations for these parameters.

The stored energy E_s which varies as I^2 for constant inductance then covers a range of 10^6. As a result of Eq. (2), the stored energy depends on average overall current density $<j> = \lambda j$ where λ is the packing factor as

$$E_s = \frac{\text{constant}}{<j>^6} . \qquad (3)$$

A decrease of $<j>$ by a factor of ten leads to an increase in stored energy by a factor of 10^6. In Fig. 11 the dependence of stored energy on the average current density is shown for a series of solenoids starting at 15 kA/cm^2 and E_s = 10 kJ or E_s = 20 kJ. Because of the sixth power, the assumed high current density starting point is not too crucial. We also note that for any one solenoid in this series the stored energy is directly proportional to j^2 as expected, e.g., $E_s = 1/2\ LI^2 = 1/2\ LA^2j^2$ = constant x j^2. What we have computed above is the dependence of stored energy on current density if the wire size and current capacity is increased and the inductance held constant. Also shown in this figure are all the large magnets constructed or designed on this stability principle. Although there is a fair correlation between increasing size and decreasing current density, we note that Eq. (3) and the Gedanken experiment is pessimistic and magnets with larger stored energies are possible for the current densities given by the line in the Fig. 11. Larger magnets such as the sizes indicated in the figure are always constructed with higher rather than lower self-inductances than small magnets (α and β are not kept constant) although the ones shown have been designed conservatively with respect to current density. Although one would like to keep L small, there is a practical upper limit at the present time for I in the range 5 to 10 kA due to limitations imposed by well regulated power supplies and joule losses arising from the introduction of high current leads into a dewar. If reliable persistent switches capable of operating at high current values are developed, no doubt magnets would be designed to operate at higher current levels than those presently under construction.

RADIATION EFFECTS

All of the irradiation studies involving superconducting Nb and Nb-alloys reported in the literature have utilized either neutron, proton, deuteron, or electron sources and, for the most part, small size specimens. No x-ray or γ-ray studies have been published.

The published literature shows that there are either no changes produced or that the changes induced by the irradiation are small. In the few cases where both the irradiation and the measurements were performed at low temperature, the changes while larger than those produced at or near room temperature did anneal out on cycling to room temperature. A few short experiments were performed on magnets, and no deleterious effects were noted. A short summary of just the experiments involving magnets will be given because unlike the short samples most of the information has not been available in the open literature.

The magnet studies, while not extensive, are significant enough to lend credence to the belief that small radiation doses will not be harmful. A magnet made with Nb-25% Zr nylon coated wire producing a field of only 15 kG was tested while in persistent mode in a slow neutron beam for one week duration without suffering any harmful effects.[38] In the second experiment with this magnet, it operated without degradation in a neutron beam under an irradiation exposure for 24 h for a total slow neutron does of 6×10^6 neutrons/cm^2. Another slightly larger Nb-25% Zr coil wound from 0.25 mm diam wire producing 22 kG also operated in a cyclotron beam without ill effects under an irradiation of 1.2×10^9 neutrons/cm^2-sec.[39] The energy of the neutrons was in the range 3-13 MeV and peaked about 8 MeV. A total dosage of 4×10^{12} neutrons/cm^2 caused no observable change in the magnet. A total dosage of 7.2×10^{17} neutrons/cm^2 (fission neutrons above 0.7 MeV from the CP-5 reactor) also caused no deterioration of this coil provided the dose rate was kept low enough to avoid heat spikes or in this experiment about 2.8×10^9 neutrons/cm^2-sec. However, at an exposure rate of 8.3×10^{11} neutrons/cm^2-sec, sufficient nuclear heating occurred that the coil showed severe degradation. An even larger Nb-25% Zr coil wound from about 300 m of 0.25 mm diam wire producing a central field of 37 kG was energized to almost its critical current and then exposed to a 440 MeV proton beam of intensity as high as 9×10^6 protons/cm^2-sec without undergoing a quench.[40] No ill effects were noted after exposure to an integrated flux of 8×10^{10} protons/cm^2. Perhaps the most encouraging result, although performed on only a 4 m length of 0.25 mm diam NbZr wire, was obtained from the following experiment.[41] The wire was exposed to neutrons and some γ-rays in a reactor for a six-week period until it became radioactive. Then after a period of two years (to allow safe handling although the wire was still "hot"), a small magnet was wound from this wire and compared against a similar but unirradiated piece of wire. The performance of the irradiated material was similar to that of the control magnet.

Although all the radiation experiments seem to give little cause for concern, a word of caution is warranted. Besides the fact that no experiments were performed on NbTi or Nb_3Sn magnets, it is important to note that the above experiments neither utilized high energy neutrons nor the extended exposure times anticipated for reactor magnets.

While no specific x-ray or γ-ray studies on superconductors have been reported, it is believed that no degradation would result from exposure to even an intense source. For a superconducting magnet in operation, the prime consideration would be the x-ray heating. If the energy is not sufficient to cause 0.2 W/cm^2 dissipation over some surface, then no problem is anticipated. Even this level (but probably not one larger than 0.4 W/cm^2) might not initiate a propagation of a normal zone resulting in loss of the stored field energy.

A greater source of anxiety for fusion reactor magnets is the irradiation effect on the copper, stainless steel, and insulating material since the superconductor in a completely stabilized design occupies only a small fraction of the magnet cross section. Further study of irradiation effects particularly for materials at 4.2 K will be necessary to determine if the blanket designs under present consideration attenuate the radiation sufficiently well to avoid problems with normal materials. In another paper these radiation effects are discussed fully.[42]

STRUCTURAL MATERIAL

The need for higher field magnets and the emphasis on larger bore magnets for all field ranges means increased structural problems. In addition to the electromagnetic forces, one must also pay proper regard to the stresses arising from the differential thermal contraction especially in large unstabilized coils where excessive movement might lead to a quench and loss of stored energy. There are a number of important recent advances in materials both in strength and in other properties which are useful for magnet designers. A short tabulation of some of these materials is shown in Table 11. With regard to strength, note in particular the 21-6-9 stainless steel and the MP-35N multiphase which have extremely large yield stress at 4.2 K. Both materials have been used recently in large coils for fusion research, Baseball II and IMP discussed earlier in the report. The Armco 21-6-9 is a low permeability steel, and it is stablized against martensite and ferrite precipitation. For some applications it is useful to know that many other stainless steels suffer an irreversible ferromagnetic martensite phase transition due to thermal cycling[44] Another material with high yield also shown in the table is E.L.I. an alloy of Ti, Al, and Sn which is also light weight but expensive so it is useful for application where specific strength is more important than cost. It was employed for instance in the cryogenic storage system of the Apollo space capsule.[45] There is a class of materials useful in low temperature designs where high strength is needed along with very low thermal conductivity and electrical insulation. Epoxy fiberglass has high specific strength and light weight and extremely low thermal conductivity and thus is particularly useful as support members operating between room temperature and 4.2 K. It has also been employed as interpancake insulation in large coils because it can provide electrical insulation and withstand the axial compression. One word of caution is its relatively low modulus of elasticity approximately 12×10^6 psi at room temperature and thus large elastic strains of up to 2% can be present when epoxy figerglass is used in construction of pressure vessels operating at high stress levels. Recently fibers of boron and carbon have been used with epoxy providing a material with modulus of elasticity of 60×10^6 psi and $30\text{-}60 \times 10^6$ psi, respectively.[46] These materials are more costly than the epoxy fiberglass and do not have as high a tensile strength. The tensile strengths of boron and carbon fiber epoxies are 180 kpsi and 80 kpsi, respectively.

A number of magnet designs utilize spacers for improved homogeneity, spacers for cooling channels, and end plate mounting boards. Although micarta is often used in this application, it is worthwhile to note that there are at least two other materials available in a variety of shapes and sizes which machine well, are very light, have reasonable strength, and have much lower moisture absorption than micarta. There are the Hysol cast epoxy made by the Dexter Corporation and Slipshod manufactured by Ireland Industries, Ltd. The cast epoxy has a high thermal contraction and in this respect lies between nylon and Teflon (polytetrafluoroethylene). The thermal contraction of many materials has been tabulated by Corruccini and Gniewek.[47] It has also been noted recently by Wood that Bakelite (cloth filled phenolics) performs well with stainless steel and brass at low temperatures.[48] When matching of thermal contraction is pertinent and the filling of a void required, then the filled epoxies are ideal. Stycast 2850 GT[49] matches the thermal expansion of brass from 300 to 4.2K when the resin is mixed in the proper proportion to the catalyst. Similarly, Stycast 2850 FT matches aluminum, and with the addition of 300 mesh fused quartz powder, it matches 304 stainless steel in thermal contraction.[50] These or similar filled resins have also been employed in the potting of the low copper to superconducting ratio NbTi fine filament conductor. Although the thermal contraction of all resins and plastics is much higher than metals, the technique of filling them with low expansivity powders, fibers, or fabrics is quite general and works very well. There are experimental tricks employed in vacuum impregnated coils with epoxy resins for there have been some reports of coils which show training or multiple series of quenches before reaching the operating current. According to Smith[51] such training effects can be correlated with stored strain energy in the potting material either during impregnation or during cooldown. To avoid this problem, Smith has also used potting materials incapable of storing appreciable strain energy, such as wax or oil, to achieve essentially short sample performance with negligible training. In addition to these materials, both the Rutherford group and the Brookhaven group have also tested various greases, metallic bonding (staybrite and InTl), and thermosetting plastics (polybondex 180) as potting agents. The work is still in progress, but the results of potting superconducting coils are sufficiently complex that there appears to be no single technique which is applicable for all types of magnets. In extremely large coils, the thermal conductivity of the various components plays an increasingly important role. The thermal conductivity of the common metals has been conveniently tabulated and compared to many low temperature adhesives by Denner.[53] For example, the insulating cement adhesive Fortafix[54] has a thermal conductivity higher than stainless steel and many superconductors at 4.2 K or 5 mW/cm-K although unfortunately it is still orders of magnitude below the good metals like Cu and Al and suprisingly Pb. The insulating varnishes like the GE 7031 used on their Nb_3Sn ribbon conductor has a low thermal conductivity of about 0.7 mW/cm-K.[55] Other insulating materials that are often interleaved such as Mylar (polyethylene-terephthalate) and Kapton (polymide) also have low thermal conductivity. A practical solution to the complex problem of finding a suitable interturn insulation and a suitable low temperature adhesive was the adaption of a two-component system by the designers of the ANL bubble chamber.[56] It consisted of an insulating film, F.E.P. film (fluorinated ethylene-propylene) and an epoxy adhesive.

We have only briefly touched on the mechanical and thermal properties of normal materials used in superconducting magnets and have not duplicated the listings given in some magnet papers,[8,57] but a more detailed review and tabulation of the available data would be useful.

CONCLUSION

A survey of the most recent developments in materials and magnets shows that the United States is losing its preeminent position. The newest commercial high field material V_3Ga was developed and is being produced in Japan. Aluminum bonded NbTi is commercially available from France, and it has also been successfully accomplished in Japan. Vapor-deposited Nb_3Sn developed by RCA is now being produced by a new Canadian firm.[58] The largest superconducting magnet in the world was recently built at CERN, and a large coil using hollow superconducting material cooled by forced circulation of compressed helium is presently being built also at CERN. The largest system using adiabatically stabilized

NbTi material will soon be underway at Garching. Rutherford Laboratory, which pioneered in the development of thin twisted multifilament NbTi conductor, continues as the leader in the field of pulsed superconducting magnets. The Laboratoires de Marcoussis has already built a 600 kJ, 1600 A energy storage coil. This listing is not meant to be exhaustive but rather to show the recent trend.

It is clear that the development of superconducting magnet technology has come a long way in its first decade of growth, and it is altogether fitting that we are determining the requirements anticipated for fusion reactors. The future stages in the construction of superconducting magnets will include: Large bore (1 - 3 m then 5 - 10 m) coils made with partially stabilized NbTi including extension to magnets with fields up to 80 kG at the material; large bore, high field magnets utilizing Nb_3Sn, V_3Ga, or a combination to 150 kG; and finally large bore magnets to fields of 200 kG with material as yet unavailable. In all likelihood we do not have to discover any new high field superconductors, but rather we need to further develop our presently known ones into a high current density, economical form--preferably thin, multifilament, twisted wires.

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TABLE 1

SUPERCONDUCTING MAGNET MATERIAL

Material	T_c(K)	H_{c2}(4.2 K)	(Ref.)
Commercial Production:			
NbZr	9 - 11	70 - 90	1
NbTi, NbTiZr, and NbTiTa	8 - 10	90 - 120	2, 3
V_3Ga	14.5	210	4, 5, 6
Nb_3Sn	18.3	225	5, 7
Interesting Candidates:			
PbBi (glass)	8.6	125	8
NbN	14.7 - 17.3	140	2, 9
$V_2Hf_{0.5}Zr_{0.5}$	10	230	10
V_3Si	17	235	5, 6
Nb_3Al	18.7	295	5, 11, 12
$Nb_{3.76}(Al_{0.73}Ge_{0.27})$	20.7	410	5, 13

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TABLE 2

SUPERCONDUCTING MATERIAL MANUFACTURERS

	NbTi	Nb + 25 wt % Ti + 10 wt % Zr	NbTiTa	Nb_3Sn	V_3Ga
USA	Airco[a] Cryomagnetics Supercon[b] Supertechnology		Cryomagnetics	Intermagnetics General Co.[c] Union Carbide Corp. (Linde)	
UK	Imperial Metals Industry			G & G Electronics, Ltd.[d]	
France	Thomson Houston-HB Compagnie Générale d'Electricité (Marcoussis)			Thomson Houston-HB	
Germany	Vacuumschmelze GmbH (Hanau)				
Japan	Vacuum Metallurgical Co., Ltd. Toshiba Electric Co., Ltd. Mitsubishi Electric Corp.	Hitachi Ltd.	Mitsubishi Electric Corp.		Vacuum Metallurgical Co., Ltd. Sumitomo Electric Ind., Ltd.
Switzerland	Brown, Boveri & Co., Ltd. (Oerlikon)				

[a]Now associated with Magnetics Corporation of America

[b]No longer a division of Norton Company

[c]Formerly GE Superconducting Products Division

[d]Formerly Plessey Superconducting Division

TABLE 3

CHARACTERISTICS OF THE NbTi MULTIFILAMENT WIRE

Sample Number	Manufacturer	Size (mils)	Number of Filaments	Diameter of Filament (mils)	Cu/SC Ratio	$\frac{R_{300\ K}}{R_{4.2\ K}}$	Number of Twists/Foot
1	Supercon-24T	51 x 123	24	10	2.33	173	4
2	Supercon-15	57 x 114	15	11	3.56	154	None
3	Airco-187T	D = 30	187	1.4	1.6	(a)	36
4	Airco-68T	D = 30.7	68	1.9	3.0	(a)	24
5	Cryomag-51	D = 20	51	1.6	2.0	(a)	None
6	Cryomag-62T	50 x 125	62	5.5	3.2	198	12
7	Cryomag-62	50 x 125	62	5.5	3.2	198	None

[a] Not measured

TABLE 4

RECTANGULAR CONDUCTOR

	Conductor Size (mils)	Number of Filaments	Diameter of Filaments (mils)	Cu/SC Ratio	Twist Rate per Foot	Short Sample Current					
							H(kG)	75	60	45	30
1	51 x 123	24	10	2.3	4	I(A)	$(\perp)$	480	740	1075	-
							$(\parallel)$	650	1000	1475	-
2	57 x 114	21	11	2.3	4	I(A)	$(\perp)$	570	930	1400	2000
							$(\parallel)$	655	1100	-	-

TABLE 5

PROGRESS IN RAISING THE TRANSITION TEMPERATURE

Decade	Ref. No.	Date and Discoverer		Material	T_c(K)
Initial Discovery	1	1911	Onnes	Hg	4.16
10's	2	1913	Onnes	Pb	7.2
20's	3	1929	de Haas and Voogd	PbBi	8.8
30's	4	1930	Meissner and Franz	NbC	10.1 - 10.5
	5	1932	Meissner et al.	Transition Metal Carbide and Nitrides	12
40's	6	1941	Aschermann et al.	NbN	15
50's	7	1953	Hardy and Hulm	V_3Si	17
	8	1954	Matthias, Wood, et al.	Nb_3Sn	18
60's	9	1967	Matthias, Geballe, et al. Ageev, Alekseevskii, et al.	$Nb_3(Al_{.8}Ge_{.2})$	20.05
	10	1969	Matthias, Corenzwit, et al.	$Nb_{3.76}(Al_{.73}Ge_{.27})$	20.7
70's			?	?	?

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TABLE 6

HIGH FIELD MAGNETS (AXIALLY SYMMETRIC)

	Magnet Description	Superconducting Material	Bore (m)	Length (m)	Design B(kG)	Test Result B(kG)	Test Result B_{max}(kG)	Test Result $\langle j \rangle$ (kA/cm^2)	Test Result E_s(MJ)
1	RCA - 3 Section Solenoid Experimental Use	Nb_3Sn	0.04	0.16	150	150	-	14.4	0.35
2	GE - Experimental Use	Nb_3Sn	0.03	0.2	-	-	143	19.3	-
3	RCA - NASA Lewis	Nb_3Sn	0.15	0.35	140	135	-	8.6	2.00
4	American Mag. Corp. - Vanderbilt and Max Planck Inst.	Nb_3Sn inner NbTi(T) outer*	0.18	0.6	117	117	123	10.0	1.66
5	Sumitomo (Japan)	V_3Ga	0.03	-	100	100	-	-	-

*(T) designates twisted multifilament compound conductor.

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TABLE 7

LARGE MAGNETS (AXIALLY SYMMETRIC)

	Magnet Description	Superconducting Material	Bore (m)	Length (m)	Design B(kG)	Test Result B(kG)	Test Result B_{max}(kG)	Test Result ⟨j⟩ (kA/cm^2)	Test Result E_s(MJ)	Comments
1	Avco - NASA Lewis	NbTi outer Nb_3Sn inner	0.51	0.64	88	88	103.4	5.0	8.5	
2	RCA - NASA Lewis 4 Coil Assembly	Nb_3Sn	0.51	0.86	72	59.3	-	14.9	4.8	Achieved at 1.8 K only. Extremely bad results at 4.2 K.
3	Saclay - BIM	NbTi	1.0	1.0	40	36.8	50	4.75	8.5	Published result.
						41	55.6	5.3	10.5	After rewinding.
4	CERN - Bubble Chamber	NbTi	4.72	4.0	35	-	(51)*	(1.1)*	(830)*	To be tested Nov. 1971, design values indicated by ()*
5	Brookhaven - Bubble Chamber	NbTi	2.44	2.4	30	28.2	38	2.5	64	Partially tested.
6	Argonne - Bubble Chamber	NbTi	4.8	2.8	20	18.5	-	0.78	80	Partially tested.

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TABLE 8

NON-AXIALLY SYMMETRIC MAGNETS

	Magnet Description	Superconducting Material	Bore (m)	Length (m)	Design B(kG)	Test Result B(kG)	Test Result B_{max}(kG)	Test Result $\langle j \rangle$ (kA/cm^2)	Test Result E_s(MJ)	Comments
1	LRL - Baseball II	NbTi	1.2	1.2	20	14.6	55	2.92	8.9	Partially tested. B_{max}(design) = 75 kG.
2	ORNL - IMP Quadrupole	Nb_3Sn	0.15 x 0.18	0.6	20	14	61	10	1.23	Partially tested. B_{max}(design) = 88 kG.
3	Jülich - Argas	NbTi(T)	0.1 x 0.08	0.8	42.5	33	-	5	5.8	Partially tested.
4	Hitachi - Saddle	NbZrTi	0.38	1.8	45	47	56	3.2	4.5	For MHD.
5	CERN - Dipole	NbTi(T)	0.13	1.4	45	45	49	22	1.4	Bending coil.
6	LASL - Quadrupole	NbTi(T)	0.15	1.0	30	30	44	23	0.05	Focusing coil. Gradient = 3 kG/cm.
7	NAL - Dipole	NbTi(T)	0.04 x 0.1	0.76	22.5	22.5	30	20	0.008	Bending coil. 0.1% uniformity.
8	CERN - Quadrupole	NbTi(T)	0.1	0.75	20	20	50	11	0.2	Focusing coil. Gradient = 5.4 kG/cm.

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TABLE 9

PARTIAL SUMMARY OF FUTURE MAGNETS

	Magnet Description	Superconducting Material	Bore (m)	Length (m)	Design B(kG)	B_{max}(kG)	$\langle j \rangle$ (kA/cm^2)	E_s (MJ)	Comments
1	IGC (GE) - Split Solenoid[a]	Nb_3Sn	0.05	-	160	-	-	-	For 1971.
2	NAL - Bubble Chamber	NbTi(T)	4.26	2.93	30	51.5	2	400	For April 1972.
3	Rutherford - HFBC	NbTi(T)	1.9	2.55	70	82	1.6	300	?
4	SLAC - Proposed BC	NbTi(T)	1.47	1.51	70	82	2.5	100	?
5	Saclay - Quad + Mirror	NbTi(T)	0.3	1.15	15	80	6	8	40 kG at mirror proposed design.
6	CERN - Omega	NbTi[b] hollow	3.0	-	18	35	1.4	50	For spark chamber by Nov. 1971.
7	Garching - W7 Stellarator	NbTi(T)	1.04	12.6[c]	40	65	10 - 15	50	40 separate coils by 1973.
8	Vacuum Metal (Japan)	V_3Ga inner Nb_3Sn + NbTi outer	0.03	0.25	150	150	-	-	For 1971 using own V_3Ga.
9	Sumitomo (Japan)	V_3Ga	0.03	-	150	150	-	-	For 1972 using own V_3Ga.

[a] Intermagnetics General Corporation was formerly the Superconducting Department of General Electric.

[b] Hollow conductor cooled by compressed helium circulated internally.

[c] 2π times major radius of toroidal system.

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TABLE 10

SUPERCONDUCTING LEVITRONS

	Location	Material	Max. Currents (kA)	D (m)	d (cm)	Operational
1	Culham	NbTi (IMI)	500	0.6	9	By end of 1971
2	Princeton	Nb_3Sn (GE)	375	1.5	-	Yes
3	LRL (Livermore)	Nb_3Sn (GE)	600	0.8	9	Yes
4	Garching	Nb_3Sn (GE)	400 200	1.2 0.6	- -	1972

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TABLE 11

MECHANICAL PROPERTIES OF MATERIALS USED IN SUPERCONDUCTING MAGNETS

	Material	Yield Stress (4.2 K) (10^3 psi)*	Tensile Strength (4.2 K) (10^3 psi)	Young's Modulus (10^6 psi)	Thermal Contraction ($\Delta L/L$) from 300 K - 4.2 K (10^{-5})
	NbTi	140	170	14.2	131
	Copper	13 - 16	-	21	330
	Brass	50	-	15	397
	304 Stainless Steel	40 - 80	235	28.9	305
1	21-6-9 Stainless Steel	196	245	-	-
	Epoxy Fiberglass	250	320	11 - 13	60
2	Nomex	128	-	-	-
3	MP-35N Multiphase	332	349	-	-
4	2014-T6	90	110	10.6	-
5	CP4-4285 Cast Epoxy	-	11	-	1800
6	Slipshod	-	4	0.4	> 4000
7	ELI (Ti, Al, Sn Alloy)	210	230	16	-

*For stress in kg/cm^2, multiply stress in psi by 7×10^{-2}.

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1. 21% Cr, 6% Ni, 9% Mn, 1.5% other, 62.5% Fe manufactured by Armco Corp., Pittsburgh, Pennsylvania.
2. Aromatic Polyamide heat treated silicone impregnated manufactured by Dupont Co., Wilmington, Delaware.
3. Standard Pressed Steel Company.
4. Aluminum Alloy with 92% Al and 4.5% Cu.
5. Hysol Division of Dexter Corp., New York.
6. A medium density polyethylene, Ireland Alloys Ltd.
7. Extra-low interstitial (ELI) alloy of Ti with 5% Al and 2.5 % Sn.

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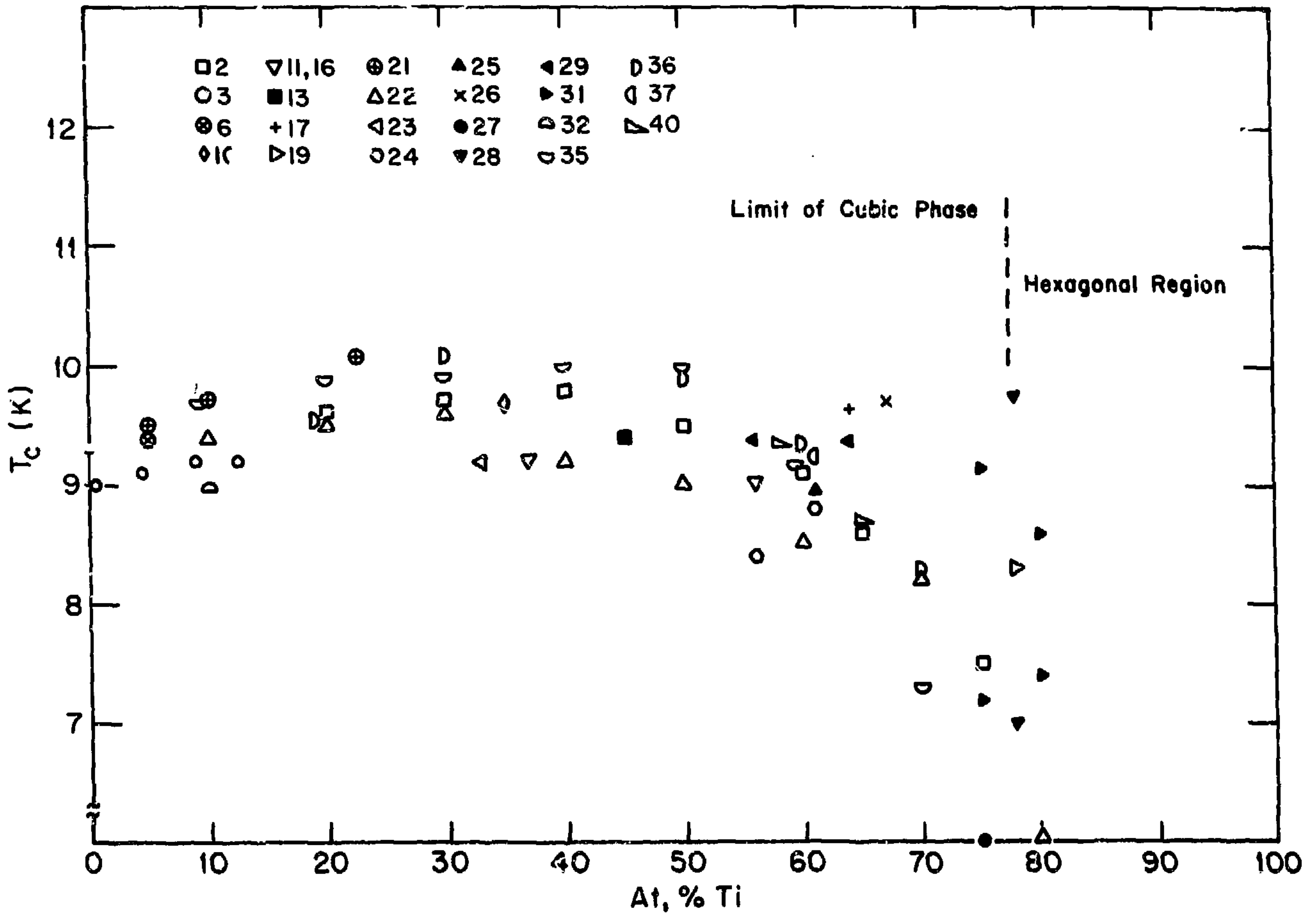

Fig. 1. The Critical Temperature at Zero Field for Nb-Ti Alloys.

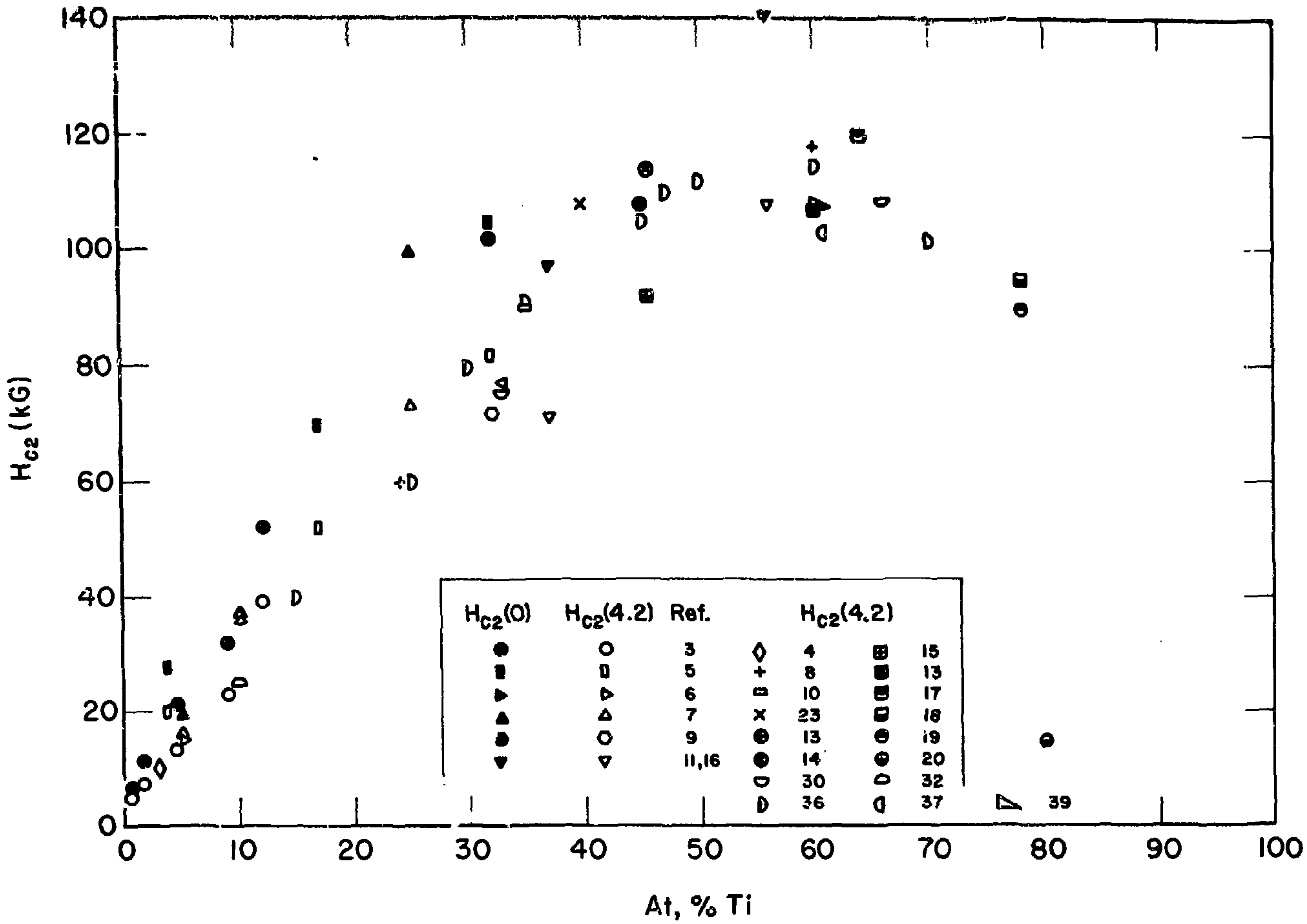

Fig. 2. The Upper Critical Field at T = 0 K and T = 4.2 K for Nb-Ti Alloys.

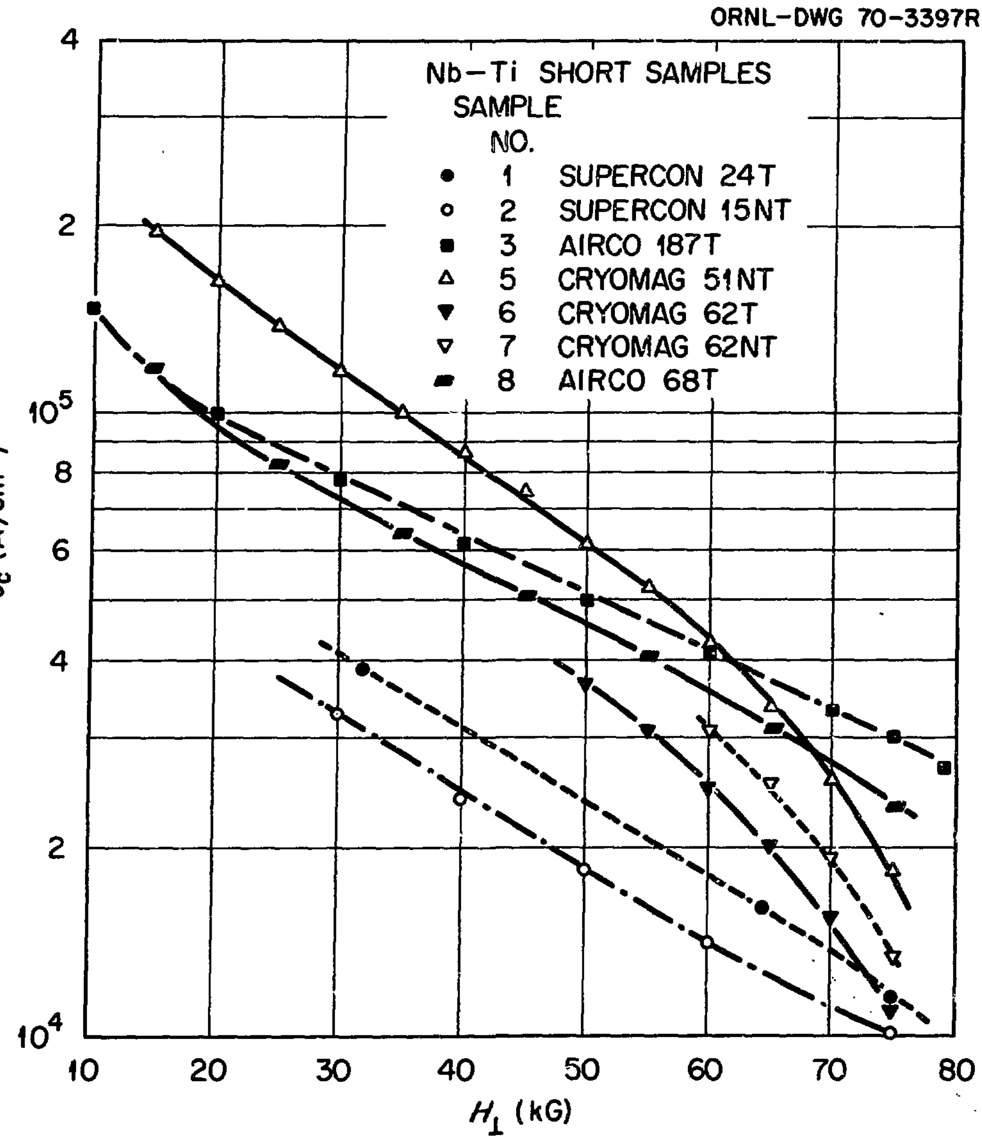

Fig. 3. Natural Log of Overall Current Density J_c vs Preset Transverse Magnetic Field in Unrestricted Helium for a Variety of Nb-Ti Short Samples. The first number after the manufacturer's name gives the number of filaments and the T or NT designates whether or not the filaments were twisted. See Table 3 for other details. [Taken from D. L. Coffey, W. F. Gauster, and M. S. Lubell, ORNL-4545, Aug. 1970.]

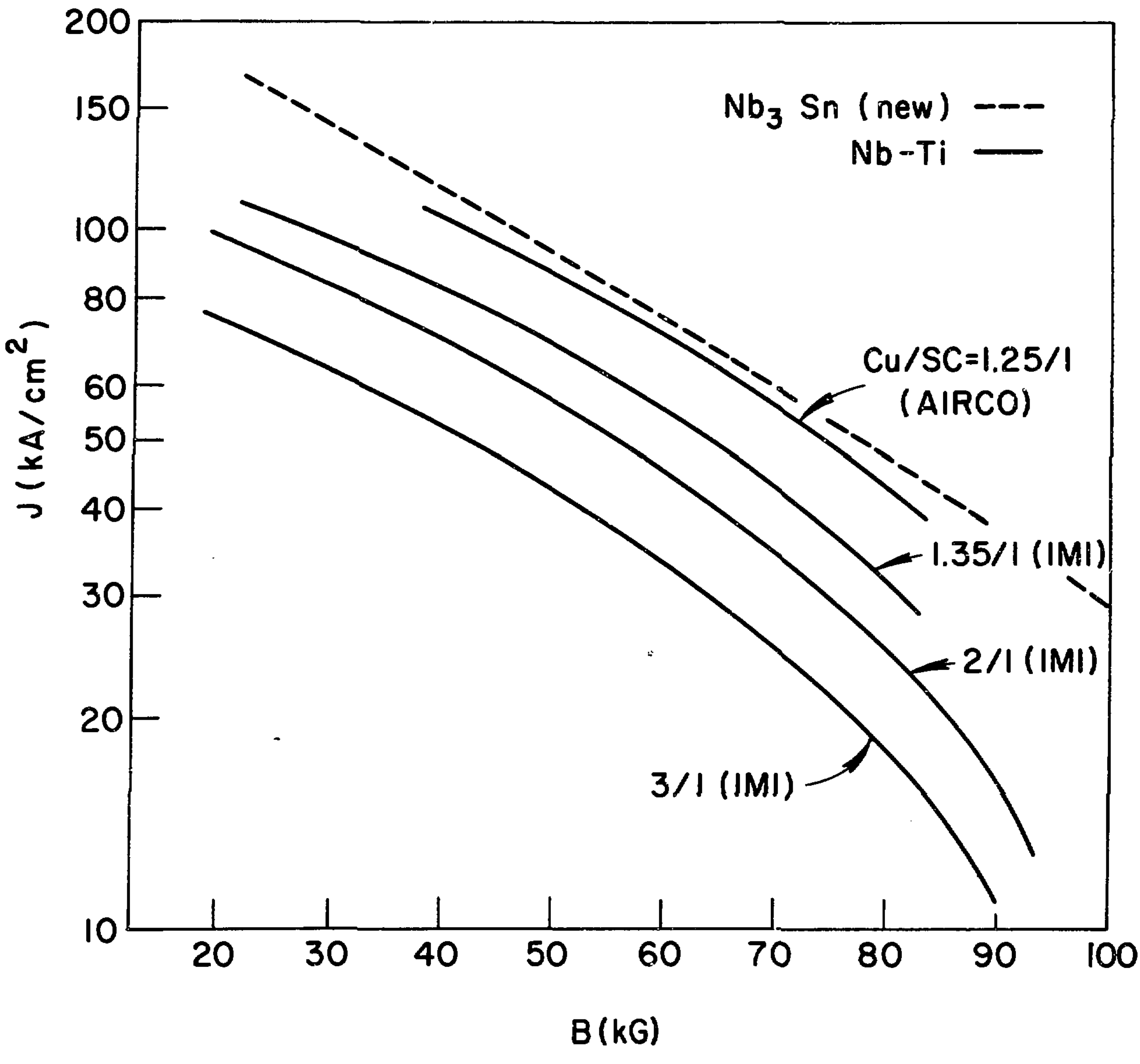

Fig. 4. Overall Short Sample Current Density vs Transverse Field. Nb_3Sn (7 mm x 0.19 mm) data derived from data supplied by P. Swartz, Intermagnetics General Corp. (formerly G. E. Superconducting Dept.), Schenectady, N. Y. The NbTi data were obtained from R. L. Stoecker, Air Reduction Co., Murray Hill, New Jersey and A. R. Mortis, Imperial Metals Industries (Kynoch) Ltd., Birmingham, England.

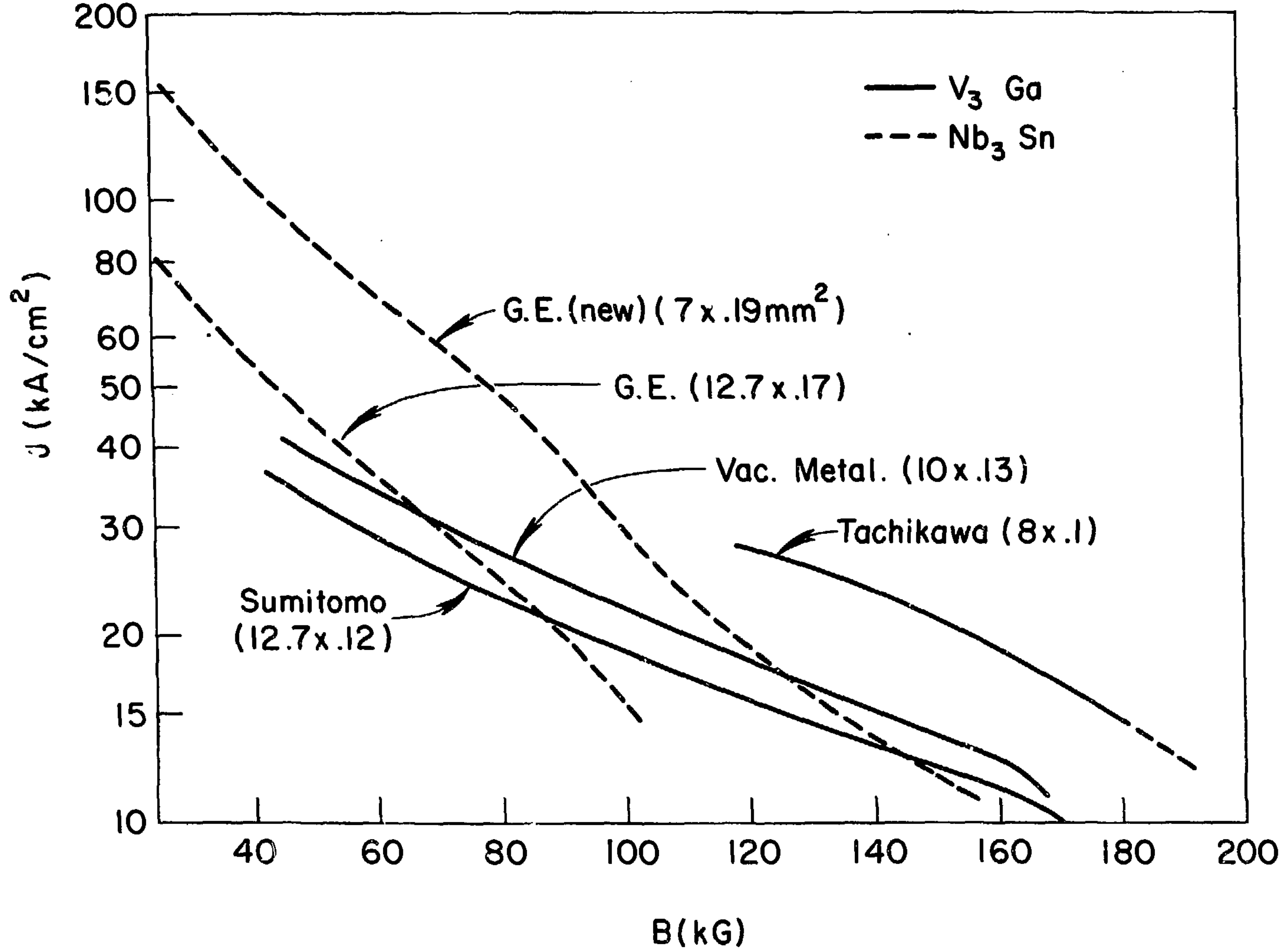

Fig. 5. Composite Short Sample Current Density vs Field. The Nb_3Sn data were derived from G. E. data sheets plus information supplied by P. Swartz, Intermagnetic General Corp. (formerly G. E.), Schenectady, N. Y. The V_3Ga data supplied by Y. Muto, Vacuum Metallurgical Co., Ltd., Yokohama, Japan; Y. Matsuda, Sumitomo Electric Industries, Ltd., Osaka, Japan; Y. Iwasa, National Magnet Laboratory supplied the data on Tachikawa's material. The width and thickness are given in mm.

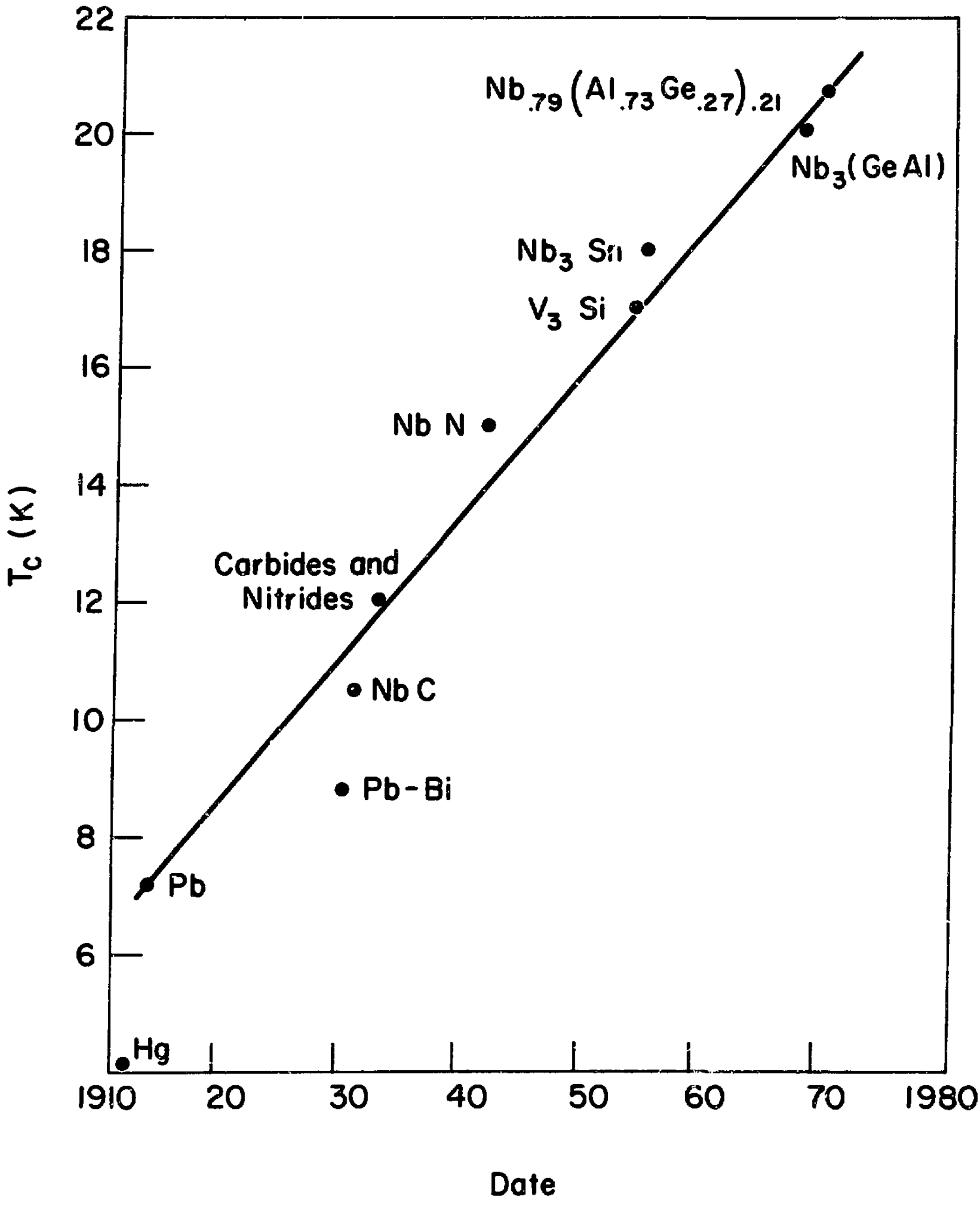

Fig. 6. Critical Temperature of Superconductors with Highest T_c vs Date of Discovery. [See Table 5 for discoverer.]

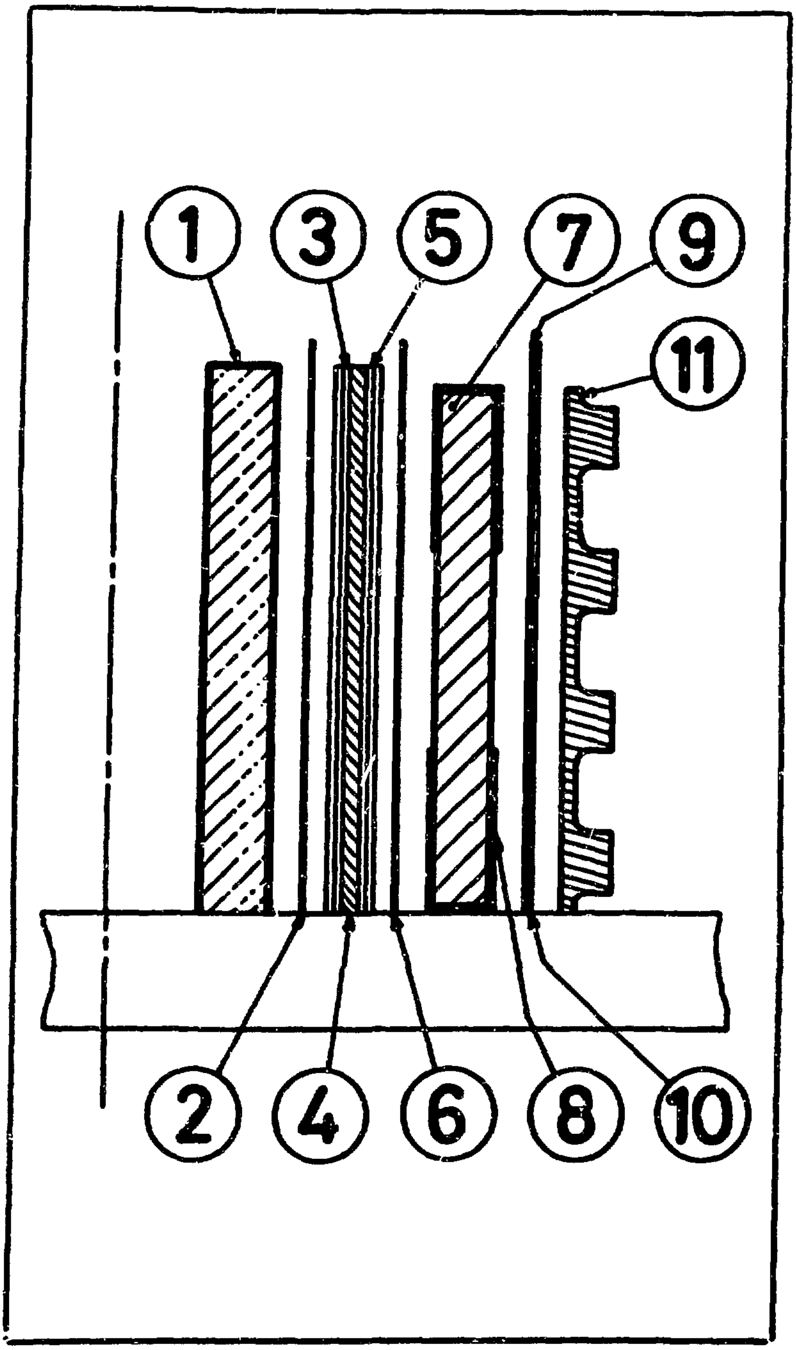

Fig. 7. Compound NbTi Superconductor Used in CERN Bubble Chamber Magnet. 1- NbTi conductor 6.1 cm x 0.3 cm with 200 filaments; 2,3,5,6,8,9,10 - thin insulators (polyester sheets and Polyamide film); 4 - Al heating strip 0.01 cm; 7 - 316L stainless steel reinforcing strip 6 cm x 0.2 cm; 11 - Cu cooling strip 6 cm x 0.19 cm; 8,10 - insulation for the first turn only.

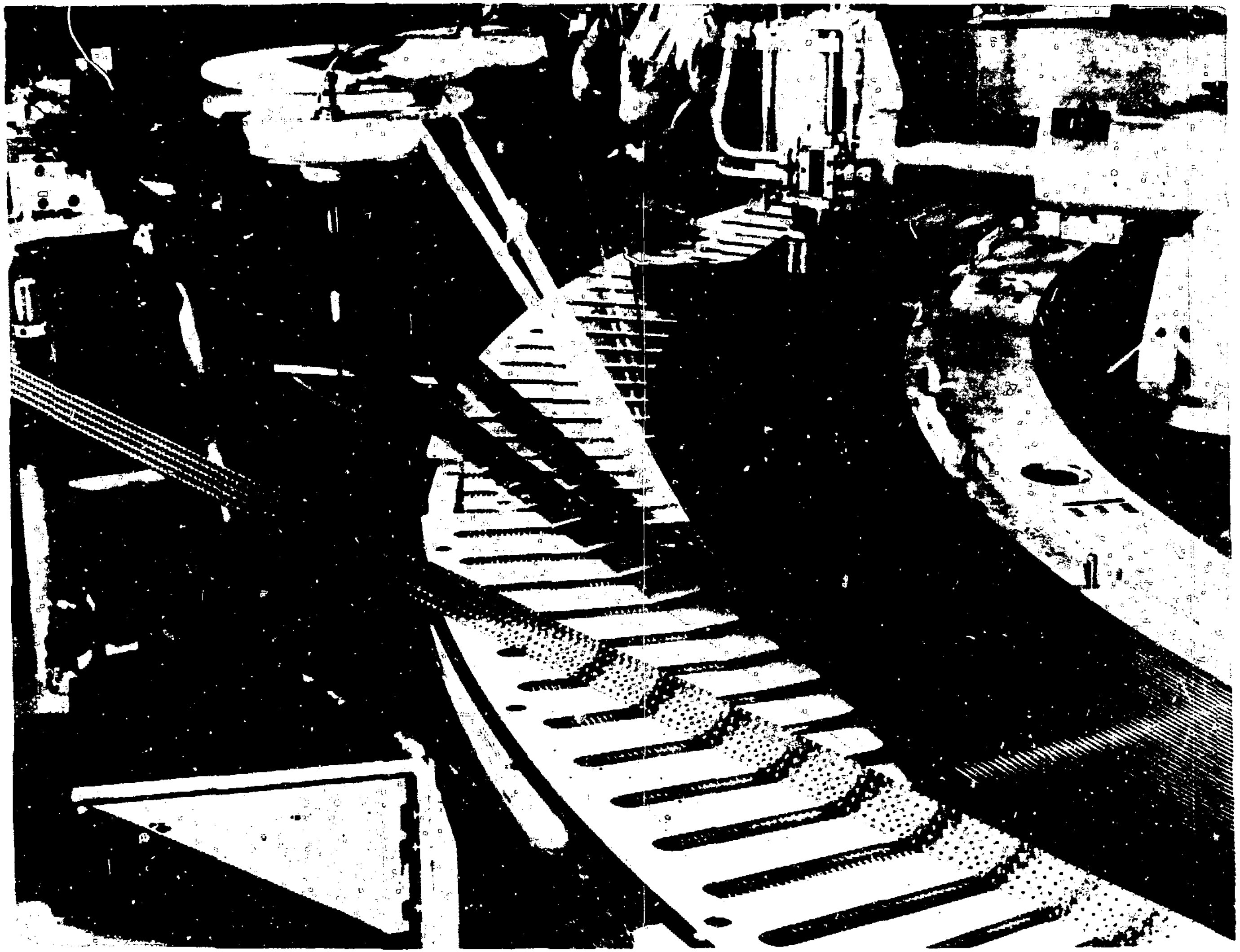

Fig. 8. Winding Operation of CERN Bubble Chamber Magnet.

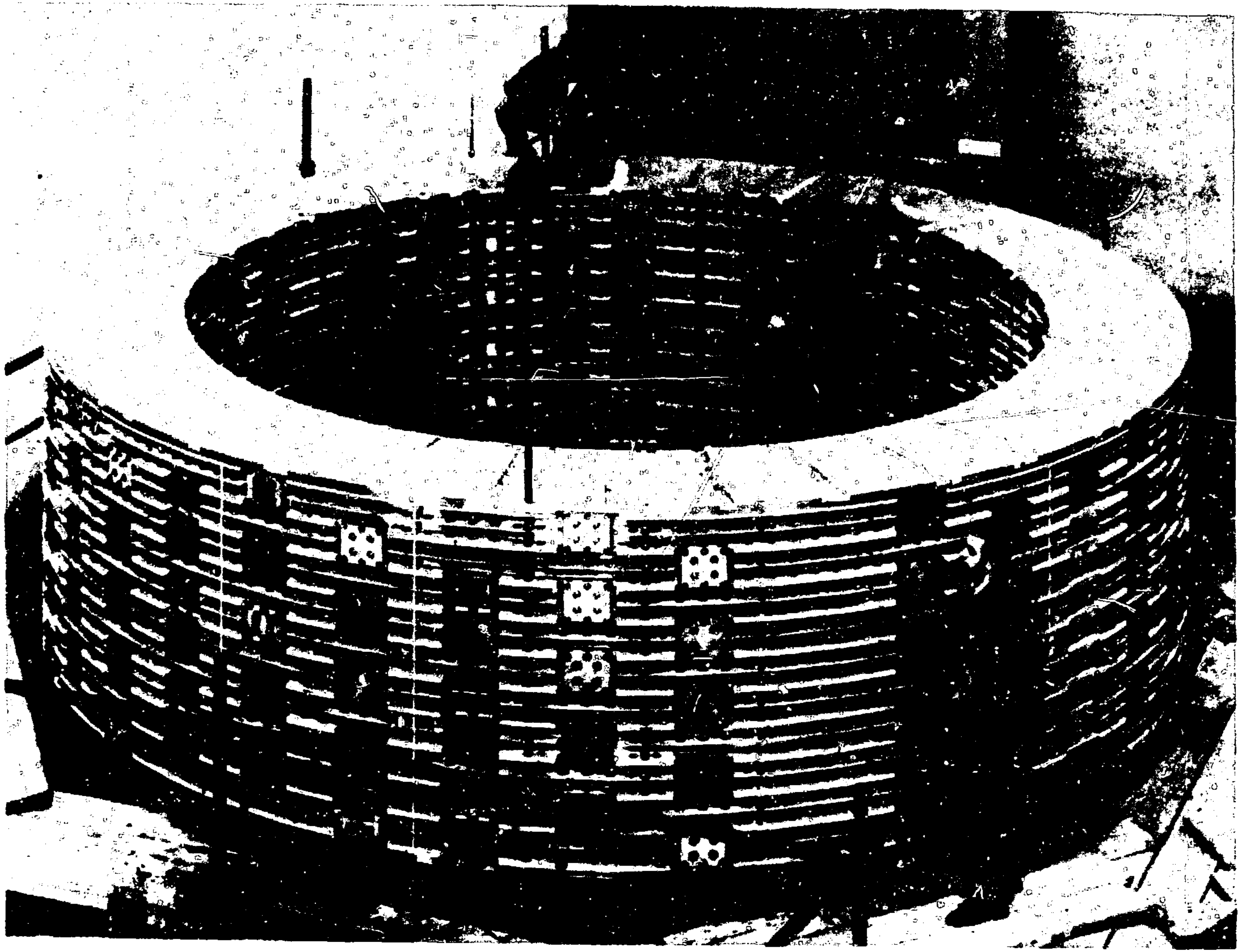

Fig. 9. One Section of the CERN Bubble Chamber Magnet.

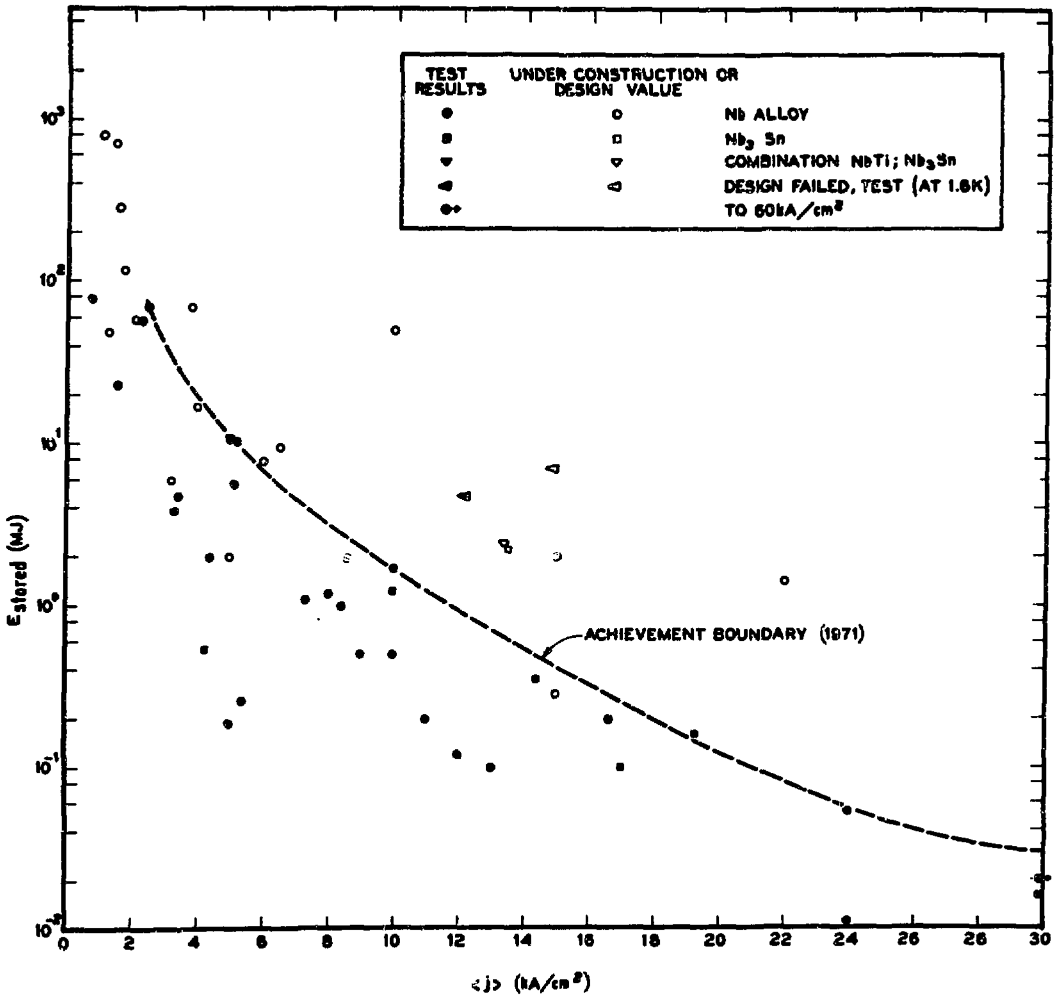

Fig. 10. Total Stored Energy vs Average Overall Current Density for Many Superconducting Magnets.

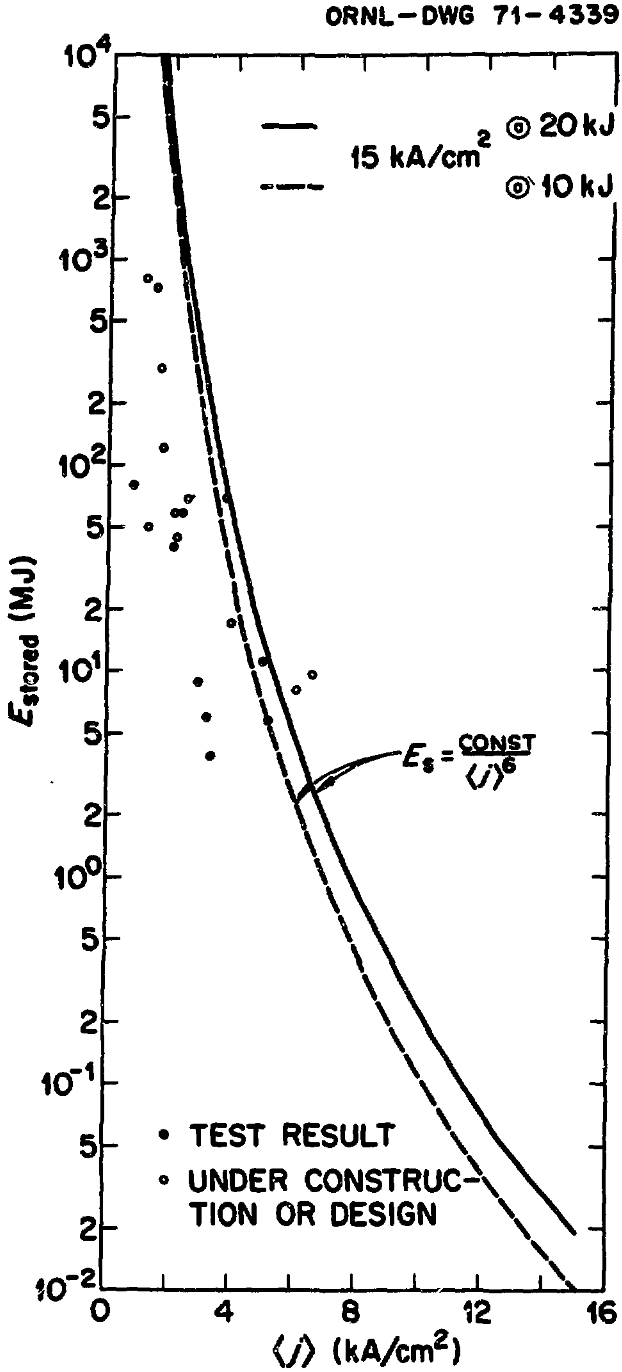

Fig. 11. The Calculated Stored Energy vs $1/\langle j \rangle^6$ for a Series of Solenoids with Constant "Geometry Factor", Inductance, and Field. The points indicate E_s and $\langle j \rangle$ for a number of completely stabilized magnets.

ENGINEERING DESIGN OF MAGNET SYSTEMS
REVIEW OF STABILIZATION TECHNIQUES

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The previous review has dealt with the superconducting materials which are either commercially available now or may become available during the next decade. To use these materials in a magnet, however, we must understand the behavior of superconductors in static and changing magnetic fields and the methods which can be adopted to prevent premature quenching.

The stabilization techniques to be considered are:

(a) Cryogenic stabilization, which depends on the provision of an alternative low resistance path for the current when the superconductor is driven normal;

(b) Dynamic stabilization, which is obtained by the addition of sufficient normal material to magnetically damp any flux jumps so that the energy released can be removed by conduction and possible instabilities prevented from growing;

(c) Adiabatic stabilization, which requires the use of fine filaments of superconductor so that the energy associated with a flux jump is too small to drive the conductor normal.

Of these three, cryogenic stabilization is used in large magnets and the design criteria are well developed and proven in practice, but overall current densities are low and typically 2000 to 5000 A/cm^2. The other two methods are at present more important in small magnets where higher overall current densities are required, and 10,000 to 25,000 A/cm^2 are possible. Adiabatic stabilization also has the advantage that it gives stability against pulsed magnetic fields.

Before discussing these stabilization techniques in detail it is important to outline the specification of the magnet in which the superconductor will be used. Since this is a reactor conference a specific reactor model will be considered to give some feel for the problems involved. Futhermore, since the reactor will not be built for several years yet it is possible to consider conductor configurations which are not commercially available, such as filamentary niobium-tin conductors.

REACTOR MODEL

The model chosen is a steady state toroidal reactor based on the stellarator geometry discussed by Gibson, Hancox and Bickerton.[1] The reactor has an electrical output of 2500 MW(e) obtained from steam turbine/alternators supplied by thermal energy from the nuclear blanket placed between the plasma and the superconducting magnet. Approximate dimensions of the reactor are:

Major radius of the toroidal system	10.00 m
Minor radius of the plasma	1.75 m
Minor radius inside nuclear blanket	2.20 m
Minor radius outside nuclear blanket	3.70 m

The magnet will be a combination of windings around the minor cross section producing an axial (or longitudinal) field and a helical field. For an $\ell = 3$ system, there will be six helical conductors inside the axial field winding. Possible parameters would be:

(a) Helical winding -

Mean winding radius	4.30 m
Cross section	1.25 x 1.25 m
Maximum magnetic field	160 kG
Total current in winding	31 MA turns
Current in conductor	10 kA
Overall current density	2000 A/cm^2
Length of conductor	1550 km
Mass of conductor	3500 tons

(b) Axial winding -

Mean winding radius	5.10 m
Cross section of winding	1.00 x 0.40 m
Maximum magnetic field	95 kG
Mean magnetic field on axis	47 kG
Total current in winding	250 MA turns
Current in conductor	10 kA
Overall current density	2000 A/cm^2
Length of conductor	650 km
Mass of conductor	1500 tons

The current in a conductor is determined by the requirements of protection in the event of a fault which necessitates rapid discharging of the magnetic energy. The total energy in the field system is around 100,000 MJ, so that even discharging at a rate of 200 MW will require 10 minutes to bring the current to zero. The total voltage during discharge will be 20 kV, although it should be possible to divide this between several circuits.

The current density in the superconductor at the peak field of 160 kG is assumed to be 400 kA/cm^2. This is a factor four higher than is possible in present commercial material but has been approached by materials under development. The reason for the assumption is that substantial reductions in the cost of superconductors will be required for reactor magnets, and increases in current density as well as reductions in processing costs may be necessary if fusion is to be economically viable. It follows, however, that of the 5000 tons of conductor required for the reactor only about 1% is actually superconductor, the rest being stabilizing material such as copper or aluminum.

CRYOGENIC STABILIZATION

Cryogenic stabilization requires good contact between the conductor and liquid helium, and this implies either an open winding with liquid helium cooling channels or hollow conductors with liquid helium flowing through them. The design criteria[2] is

$$J_{cu} < s\, I^{-1/3} \qquad \ldots (1)$$

where

$$s = \{Q_c \kappa/\rho\}^{2/3}$$

and Q_c is the limiting heat flux for nucleate boiling of the liquid helium ($\sim$3000 W/m^2), κ is a shape factor ($\sim$2), and ρ is the resistivity of the stabilizing material ($\sim 3 \times 10^{-10}$ Ωm) which is taken to be copper.

The current density J_{cu} is related to the cross sectional area of the stabilizing material, and it is seen from Eq. (1) that for a high current density the current in the conductor I should be low.

A similar criterion determines the design of the magnet from the point of view of protection,[2] and is

$$J_{cu} < P\, I^{1/2}$$

where ... (2)

$$P = \{f(\theta)\ V/E\}^{1/2}$$

and here the important parameters are E the energy in the magnet ($\sim 10^{11}$ J), V the voltage at which it is discharged ($\sim$ 20 kV), and a function of the allowable temperature rise θ. For a high current density the current in the conductor I should be high.

Combining Eqs. (1) and (2) gives an optimum current and a maximum attainable current density in the magnet, and for the reactor magnet these are around 10 kA and 4000 A/cm^2. The overall current density will be further reduced by the need for mechanical structure, and will probably be about 2000 A/cm^2. The cross sectional area of the stabilized conductor will be 2.5 cm^2, which is not much bigger than is currently used in large bubble chamber magnets.

An additional requirement for cryogenic stability is that each filament of superconductor must also be stable, and this sets a maximum filament size.[3] The criterion is

$$r_c < \sqrt{K_s \Delta T/\rho\, J_{cu} J_s} \qquad \text{... (3)}$$

where K_s is the thermal conductivity of the superconductor ($\sim$ 0.1 W/m-K), ΔT the difference between the operating temperature and the critical temperature of the superconductor ($\sim$ 10 K), and J_{cu}, J_s are the current densities related to the areas of stabilizing material and superconductor, respectively. For the reactor this leads to a maximum filament diameter of 0.3 mm. The current in each filament will then be about 250 A, and a 10 kA conductor will require a minimum of 40 filaments of superconductor.

Since each conductor must be in close contact with the liquid helium, the total helium inventory of the magnet is estimated to be 16 tons (200,000 liquid liters).

DYNAMIC STABILIZATION

Dynamic stabilization is used in strip wound magnets. It is currently of importance in niobium tin magnets, but can be applied to any system using thin strips of superconductor.

The principles of dynamic stabilization have been discussed by Hart[4] and Hancox[5] but are not as well understood as other forms of stabilization. The two most important criteria are:

$$H_{\perp}^2 < \pi^2 K_{cu} T_o / 4\rho \qquad \ldots (4)$$

which gives the maximum allowable magnetic field perpendicular to the tape in terms of the thermal conductivity K_{cu} and resistivity ρ of the stabilizing material, and a characteristic temperature T_o ($\approx$ 5 to 10 K), and

$$J_s^2 d_s^2 < \pi T_o (d_n/d_s)(K_s/\rho) \qquad \ldots (5)$$

which effectively gives the maximum thickness d_s of the superconducting strip relative to the thickness d_n of the normal material and is limited by the thermal conductivity K_s of the superconductor.

Equation (4) can only be evaluated after a fuller analysis of the magnetic field in and around the winding, and may be a serious limitation in the helical windings where perpendicular fields will be high. Equation (5) is more generally applicable and is quite severe since the thickness of the superconductor can only increase as the cube root of the thickness of the stabilizing material. Using the parameters for the reactor magnet, the superconductor is limited to about 6 μm, so that the current carried by the strip will be 24 A/mm width. For a total current of 10 kA an effective width of 400 mm will be required, which implies the stacking of several strips with copper interleaving--for example, ten strips of 40 mm width. Such a conductor is technically feasible but may not be acceptable mechanically, especially if the superconductor is brittle or on a relatively thick substrate as with present niobium tin strip.

ADIABATIC STABILIZATION

If a superconducting wire is thin enough the energy involved in a flux jump is too small to drive it normal, and it is adiabatically stable. If a large number of such wires or filaments can be contained in a conductor in such a way that each reacts independently to any magnetic field change, then the whole conductor is stable. This can be achieved by fully transposing the filaments within the conductor or, to a more limited extent, by twisting the conductor.

The simple criterion[6] for the filament size is

$$J_s^2 d_s^2 < 2\ C\ T_0/\mu \qquad \ldots (6)$$

where C is the product of the specific heat and density of the superconductor ($\sim 10^3$ J/m^3). This criterion is most severe where the current density in the superconductor J_s is highest and so instabilities tend to occur in the low field regions of a magnet. If the highest critical current density is of the order 2×10^6 A/cm^2 the filament diameter must not exceed 10 μm, giving a current in each filament of 0.25 A and a total of 40,000 filaments in the conductor. If the filaments are embedded in a high conductivity matrix, however, there will be an additional stabilizing effect due to the damping of flux motion,and the size could probably be increased[5] by a factor $(D_t/D_m)^{1/4}$ where D_t and D_m are the thermal and magnetic diffusivities of the matrix. Thus a 30 μm filament should be stable, giving a current of 2 A and 5000 filaments in a conductor.

If the decoupling of the individual filaments is achieved by twisting rather than by transposition, there is a limitation on the size of the conductor due to a self-field instability. The criterion for overcoming this is the same as for the individual filaments except that the overall current density and specific heat are used. This gives a maximum size of 5 mm. Such a conductor might contain 500 filaments and carry 1000 A, so that eleven such sub-conductors could be transposed to form a complete 10 kA conductor.

The criterion for the twist length in a sub-conductor is

$$\ell_c^2 = 2J_s d_s \rho/\dot{B} \qquad \ldots (7)$$

giving a twist length of about 1 meter if the magnet is energized in 30 hours. The transposition length of the complete conductor would be much longer than this provided the individual sub-conductors are insulated, and will depend mainly on field gradients along the length of the conductor.

Adiabatic stabilization offers three main advantages. Firstly, higher overall current densities are possible since less stabilizing material is required. Secondly, the conductor can also be stable against pulsed fields such as would be experienced in an ohmically heated toroidal reactor. Thirdly, intimate contact of the conductor with liquid helium is not required so that a mechanically compact winding is possible and the helium inventory can be reduced.

At first sight it appears that the cost of producing conductors containing large numbers of fine filaments will be greater than the cost of a cryogenically stabilized conductor. It may be, however, that the sub-conductors of an adiabatically stabilized winding are closer to the type of conductor which will already be in use for other applications, and therefore could use existing production facilities.

CONCLUSION

Three types of stabilization have been considered. Of these both cryogenic and adiabatic stabilization could be used in the reactor magnet. The choice will depend on a variety of factors, of which the most important will be the current density required in the winding and the cost of fabricating and winding the conductor. Whichever is chosen, our rapidly improving understanding of the behavior of superconductors and growing experience in building large magnets suggest that the design of a reactor magnet will not present insuperable problems.

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SUMMARY OF SESSION

ENGINEERING DESIGN OF MAGNET SYSTEMS

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The emerging importance of superconducting magnet systems not only for fusion reactors (which after all are still a long way off from being realized) but also for fusion research was clearly indicated by the great number of papers presented at this session. There were twenty-four talks including two state-of-the-art reviews, and thus this summary will attempt to provide a general overall impression of the session without reviewing each contribution in depth. An appendix follows listing the topics and speakers in order of presentation and a short abstract which most of the participants provided for their contribution. Many subjects were discussed, and a wide variety of problems mentioned. The major problems raised and still to be investigated for very large scale systems are: (1) superconducting material cost particularly for high fields, (2) large bore high field magnets, (3) superconducting instabilities in magnets which are not completely stabilized (4) structural problems associated with static and dynamic force containment, (5) switches for inductive energy storage devices and also persistent switches for superconducting magnets, (6) protection of superconducting magnet systems including dewars and structural members against a sudden quench, and (7) refrigeration and field erected vacuum systems.

Superconducting magnet technology, although a younger field of investigation than fusion research, has kept abreast of the needs for fusion research on a laboratory scale, but it is not yet able to meet the requirements for most of the full scale reactor plants proposed at this meeting. It is generally agreed that we need either improvement in the current carrying capacity of the presently used high field superconducting materials, further development of known high field material, or discovery of new superconductors. We do not yet have an adequate material for reactor sized magnets for the field range 120 - 200 kG (see the state-of-the-art review by the author). The cost of superconducting material, although decreasing over the last five years, is still high enough to be a major cost figure in any magnet system for a full scale reactor plant. It might be anywhere from 33 to 70% of the reactor cost depending upon the field range and design.[1,2]

A number of large bore, completely stabilized (see both state-of-the-art reviews) low field magnets have been successfully built and operated, and the design features and parameters of two bubble chamber magnets were discussed by Purcell and Jensen, but the stellarator project W-7 at Garching as outlined by Wipf will apparently be the first to undertake the ambitious project of constructing a large partially stabilized magnet system. To date only relatively small size, low stored energy magnets have been built utilizing the twisted and transposed thin multifilament NbTi (adiabatically stabilized) compound conductor as was discussed in the review. One definite advantage of this material is its use in the construction of relatively complicated windings (dipoles and quadrupoles) at very high average current densities over 20,000 A/cm^2 as described by Rogers. McInturff discussed some designs using either or both dynamic and enthalpy stabilization (see the review by Hancox for discussion of stabilization) that can even be pulsed to within 95% of short sample critical current with rates of energizing larger than 100 kG/sec. We caution the reader to take cognizance that these fine results apply only to relatively small (< 500 kJ), low field (< 60 kG) magnets. Nevertheless, the coils built with this small filament NbTi material when properly stabilized and potted do meet predicted design values. Figure 1 is the measurement of the loss per cycle vs field for a solenoid (ID = 2.5 cm, OD = 7.5 cm, and ℓ = 4.5 cm) with and without vacuum grease potting. The coil contains 300 m of Airco wire in 33 strand flat braid fully transposed with each 8 mil diam formvar insulated strand containing 121 NbTi filaments twisted in a matrix with copper to superconducting ratio of 2 to 1. The conductor motion in the formvar coils (unpotted) is sufficient to actually produce audible noise. In Fig. 2 both pancakes and dipoles wound from the same Cryomagnetic 132 strand 1.6 cm wide braid and with similar load lines (current vs field) are compared. Each strand is an 8 mil diam wire with 210 NbTi filaments in a matrix with copper to superconducting ratio of 1.3 to 1 and insulated with either formvar or a AgSn intermetallic insulation. The unpotted formvar pancake coils show severe degradation while the pancake with intermetallic insulation reached short sample performance, and the dipole carried 95% of the short sample current.

The types of instabilities which occur in Nb_3Sn ribbons subjected to transverse field were discussed by Coffey in connection with a large hybrid system (NbTi outer coil and Nb_3Sn inner) and also Efferson with regard to the quadrupole coils for the IMP system. A picture of the former system in the process of being wound is shown in Fig. 3. This is the largest bore magnet with an operating field over 100 kG constructed to date (total cost about $200,000). It has the further distinction of working right up to the short sample value. The IMP quadrupole is the largest non-symmetrical system employing Nb_3Sn. The instabilities were reduced to a harmless level by interleaving high conductivity aluminum with the superconducting ribbon the so-called dynamic stabilization technique. The instabilities in ribbon conductors would render reactor size magnets particularly vulnerable to the pulsed fields that would also be present in some of the fusion systems proposed. Experiments on the stability of a magnet in persistent mode subjected to a large pulsed transverse magnetic field have not been performed. In some cases one can make convincing arguments that the effect would not be catastrophic. In other situations, the relative orientation of the fields make any theoretical conclusion just a wild guess.

One divergence of opinion between the W-7 project of Garching and other toroidal systems is that the former will use a multiple dewar system with all the structure members which are used to contain the forces brought out of the cryogenic environment through vacuum walls to room temperature supports. Furthermore to keep the heat leak at a manageable level, the supports are only sufficient to handle the radial forces, and any azimuthal forces that arise as a result of a partial quench of the 40 coil system will necessitate a total shutdown of all the magnets. Both the NASA-Lewis present mirror system which is the first fusion experiment to employ superconducting coils and the bumpy torus project described by Roth, and the Princeton toroidal force free proposal discussed by File contemplate having the structural reinforcing members at liquid helium temperature, and some of the other reactor systems--our own tokamak fusion reactor for example--also envision having all the structural components cooled to cryogenic temperatures. In this manner, the heat load and structural problems are reduced, but the cool-down problem is significantly increased. The characteristics of the NASA-Lewis superconducting bumpy torus facility which is 90% completed are: major radius = 76 cm, minor radius = 9.5 cm, 12 coils at a total cost of about $300,000. The maximum magnetic field on the axis will be 30 kG, and the mirror ratio on the magnet axis is 3.1. It has generally been standard practice to weld all tubing that is at liquid helium temperature. To avoid complications in welding tubing containing electrical cables, all of the large number of joints are standard A and N demountable fittings which are capable of transmitting liquid helium under vacuum without leaking. The mating surfaces were cleaned free of scratches and coated with liquid teflon before mating and tightening at a torque of 100 ft-lb. Although not brought up at the meeting, the insulating joints developed by Morpurgo[3] to carry liquid helium are capable of withstanding 5000 V in vacuum with an internal pressure over 300 Atm.

Protection of large superconducting magnet systems was not really given an adequate forum at this meeting partially because it has not yet proven to be a problem, however we suspect that a definitive solution has not been employed as yet. The present systems utilize a dump resistor and a switch which disconnects the magnet from the power supply. The magnet simply discharges exponentially with the L/R_d time constant. In many cases, as much as 99% of the energy can be removed in this manner, and the maximum voltage rise across the magnet R_dI at the onset of discharge can be tolerated. However, this is because the present operating currents are not large. For future reactor systems which will be designed for higher currents to avoid high inductances, the maximum voltage appearing across a quenched coil may be too high for the presently used insulations and should be a source of concern. In the stellarator discussed by Hancox, 20 kV was proposed for the discharge voltage. In addition even 1% of the stored energy dumped in a cryostat could cause a severe problem. For the magnet systems proposed at this meeting, 1% of the stored energy would be sufficient to vaporize all the helium in the dewars resulting in a rapid and large overpressure for which the dewars must be designed. A possible protection mechanism is the fast removal of liquid helium through large pressure operated orifices in the dewar. This allows the heat capacity of the total magnet system to absorb the stored energy. For large cryogenically stabilized magnets, the temperature rise is never high when the energy is equally distributed as would be the case on the fast removal of helium because normal zones would propagate very rapidly once the system is no longer in liquid helium.

The problem of switches was discussed in connection with inductive energy storage by Laquer and Ribe. At present, switching times as short as 100 nanoseconds are achieved, and experiments are in progress on a 25 kJ, 1000 A coil with plans to increase the stored energy by an order of magnitude within a year. This will then compare with the energy storage coil at Laboratories de Marcoussis which is a 600 kJ, 1600 A solenoid with a 76 cm bore, a length of 60 m, and insulated for 100 kV.[4] Laquer and Ribe envision eventually a really large energy storage device as the next generation Θ-pinch experiment as shown schematically in Fig. 4. Such an experiment would typically be designed for a Blumlein-line voltage of 50 to 60 kV (200 to 240 kV around the shock heating coil) and a compression field of 85 kG, having a rise time of 2 msec and a plasma duration of at least 6 msec at a temperature of 5 keV and a density of 1.7×10^{16} cm^3. For a major radius of 25 m, the superconducting magnet would store about 850 MJ, and assuming the energy transfer efficiency is 25%, then about 200 MJ can be delivered to the 85 kG plasma compression coil. Owing to the utilization of cryogenic magnetic energy storage, such a device would cost about five cents per joule of compression - coil energy installed (assuming coil cost and switch cost is each $7000 per meter section), as opposed to 60 cents per joule for Scyllac (capacitor storage). Wipf has also looked at the problem of switching, and he showed that existing techniques are not adequate to both stabilize a superconducting switch and keep the losses at a low level. Schmitter discussing the power limits for inductive energy storage showed convincingly by analyzing a Brooks' coil that a large stored energy must be split up into and stored in smaller units to discharge at a sufficient speed. It is clear from the discussions of both Los Alamos and Garching that there is much needed work on inductive energy storage devices. Very little experimental work has been reported to date and much improvement is almost certain to come with further research and development. Certainly at the present time, there is agreement that capacitive energy storage discussed both by Knobloch and James is more advanced than superconducting inductive storage. Both considered the jump to a 100 MJ bank as quite feasible with only a little more development of the present capacitor technology.

Force containment is and will continue to be a major problem in high field systems and is likely to ultimately prove to be the limiting problem (bottleneck) in attaining large steady state superconducting fields over 200 kG. This suggestion has already been made by Dave Rose. Some interesting, fairly new materials were discussed which are useful for alleviating force problems in large systems. Nelson indicated that the containment of the electromagnetic forces in the LRL Baseball II was significantly aided by the use of Armco 21-6-9 stainless which has a high yield stress of 250,000 psi at 4.2 K. The bobbins of the IMP quadrupole were also made from this material. Another strong material particularly useful for low thermal heat leak is epoxy fiberglass. Brewer reported on a set of supports of 1/2 in. cross section fabricated here in Oak Ridge that were capable of withstanding 28 tons in tension. This support system and a unique heat shield solved the two major design problems of the dewar made for the magnetic mirror facility for NASA-Lewis. Figure 5 shows the assembly drawing of this magnet system. The support straps are angled to take the 1/2 G side load and the 8 G downward load and to provide the longest path for heat flow from the magnet can. Adjustment of the support straps is made from the outside of the vessel. Although epoxy fiberglass has favorable properties, it was pointed out by Grieger that it will undergo creep somewhat when subjected to a continual high load under tension. A study of this very important problem is being investigated for Garching at both Siemens and Grenoble.[5]

Of all the major components that go to make up a complete superconducting magnet system, the cryogenic system as discussed by both Long and Jensen is relatively inexpensive when compared to the superconducting material and reinforcement structure. Jensen believes that 25% efficiencies for refrigerators are quite easily achieved in large plants and is a reasonable goal. He points out that present costs for 300 W machines (power at 4.2 K) are $500/W, and estimates for 10 kW machines appear to be $150/W in one-of-a-kind production. Some decrease could be expected if a given size were produced in large numbers. Long reinvestigated the very useful state-of-the-art data of Strobridge[6] and verified that the generalized phenomenological curves of cost vs input power and percent of carnot efficiency vs capacity for cryogenic refrigerators is still applicable. Although not quite in the same state of advanced development, large field erected vacuum chambers and unusual shaped dewars are likewise to be considered ahead of the present needs of fusion research as pointed out by Murphy. He also said that present materials and techniques with only a little more development will most likely be adequate for the requirements of fusion reactor systems.

The most interesting points of discussion for fusion reactor studies unfortunately came at the end of the session when many of the attendees had already left. Both the tokamak studies of Ohta et al. and the stellarator investigations by Georgievsky et al. indicate that higher fields than are presently available will be needed. Ohta performed an optimization study on a tokamak reactor and finds that the reactor will be impossible for the small fields that have been proposed. Georgievsky concluded that an economical fusion plant based on the stellarator will have a total thermal output power of 10,700 MW and provide energy at a cost of approximately $0.26 per KWh if a superconducting magnet system is employed. Hubert analyzed the "ultimate torsatron" which is reduced from the classical stellarator by removing one set of helical coils, the toroidal field coils, and the vertical field coils leaving only a set of helical windings. However, a compensating loop is necessary to remove the effects of the stray field. He compared his work with the ORNL tokamak reactor[1] discussed by Long and showed that by using the same values for the central field, minor coil radius, and minor plasma radius, the cost of the required superconductor for the torsatron would be only 30% more than for the tokamak. In addition the torsatron has a divertor which is not the case for the tokamak, and if one is needed it will certainly reduce the gap in cost between the two. Although the magnets required from these in depth analyses (and I would include the reactors discussed earlier in the session by Hancox and File as well) undoubtedly cannot be constructed at the present time, they are not so very far above present capabilities that one becomes discouraged. In all likelihood we will be able to meet the needs of fusion reactors when they are ready to be constructed provided that research and development in superconducting magnet systems continues. Studies of this kind which ask more than the present state-of-the-art but yet do not represent an unreasonable extrapolation are quite valuable in indicating the direction and the distance that we still have to proceed.

Although no one expects that the state-of-the-art of superconducting magnets and the information in this fusion reactor session on magnets can be adequately summed up in one figure or table, I believe the essence can be gleaned from Fig. 6. The present achievements in superconducting magnets, Bore vs B_{max} (see tables in review paper by M. S. Lubell), is shown along with the magnet systems required in the various recent fusion reactor proposals (see Table I). This should not be taken to indicate that magnets outside the regime are beyond present capabilities but rather as the limit to what has been attempted and accomplished. As the need arises, larger sizes and fields will unquestionably be attained. For the large bore relatively low field NbTi magnets, I should point out that almost without exception no magnet over one meter in bore size achieved the design value at the initial construction and testing. One should also note that no high field magnet with more than 100 kG at the windings has ever been built with a bore larger than 60 cm. Present technology is not adequate to meet the needs of plasma physics especially when one considers the pulsed fields which will be present in some fusion systems. More R & D will be required. The amount of extrapolation of present technology to achieve the sizes and fields required by the proposed reactors is easily seen in Fig. 6. The triangles represent systems discussed at this meeting by Ohta, Georgievsky, Hubert, Hancox, File, and Long and the Culham work presented at the recent IAEA meeting in Madison. In addition previous systems proposed by Mills and Carruthers both recently analyzed by Hubert are also included. See Table I for comparative details and the appendix for more information. Only three of the proposed systems--the ORNL, CEA, and modified Princeton--are at a low enough field that puts them within an order of magnitude in size from what has already been achieved. All the others require an even larger extrapolation at yet higher fields. A large bore (11 m) superconducting magnet of moderately high fields ($B_{max} \leq 80$ kG) constructed along completely stabilized principles[7] is technically feasible at present.[1] It still remains to be demonstrated that large bore superconducting magnets which use other forms of stabilization can be reliably operated. There is general agreement that large bore high field magnets (> 140 kG) are beyond present capabilities (see the state-of-the-art review). Further development and improvement of known materials will be required. In general one can say that instability and degradation problems become more severe as the size and maximum field of a superconducting magnet is increased. To a certain extent it is understandable that a superconductor is less stable when it is subjected to a field close to its upper critical field. However, a fully adequate explanation for degradation of magnets due to increased size (stored energy) has not been given. Until a full understanding of all superconducting instabilities is available, one must approach each major extrapolation in size and field with some caution.

There is universal agreement that a superconducting magnet for a fusion reactor whether of a field value that we can construct today or of a size and field that puts it off until some unknown time in the future, will be expensive. The engagement between nature's coldest and hottest phenomena has been a long one and the impending marriage will be costly. Let us hope, however, that this does not deter the union from taking place and producing bountiful offspring.

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TABLE 1

RECENTLY PROPOSED FUSION REACTOR SUPERCONDUCTING MAGNET SYSTEMS

Ref. No.	Laboratory	Type	Power [MW(th)]	Wall Loading [W/cm²]	Major Radius [m]	Plasma Radius [m]	Blanket Thickness [m]	Inner Radius of Windings [m]	Multipolarity and Period Per Helix	Average Current Density [A/cm²]	Magnetic Field on Axis [kG]	Maximum Magnetic Field [kG]	Stored Energy [GJ]
1	ORNL	Tokamak	5000	345	10.5	2.8	2	5.6	---	1550	37	80	40
2	PPL, Princeton	Toroidal (D-T) Toroidal (D-D)	(3500) (3500)	(1040) (220)	5 10	1.7 4.0	(0.5)[a] (0.5)[a]	(3.25) (6.18)	--- ---	6800 7000	91 88	160 160	15.4 112
3	JAERI	Tokamak	5000	1300	5.2	1.1	1.2	2.8	---	---	80	184	(20)
4	Princeton-CEA	Tokamak	2200	1000	4.7	1.0	1.2	2.5	---	---	65	95	9.8
5	Culham-CEA	Tokamak	6000	1300	6.6	1.25	1.25	3.05	---	---	100	125	56
6	PTI, Kharkov	Stellarator	10,700	1300	16.8	1.0	1.0	2.0 (H)	3 and 5	---	80	160	---
7	CEA, Fontenay-aux-Roses	Torsatron	10,000	(460)	22	2.0	2.0	5.5	3 and 4	1340	37	97	---
8	Culham	Stellarator	5830	(675)	10	1.75	1.5	3.7 (H) (4.6)(A)	3 and 7	2000 (H) 2000 (A)	47	160 (H) 95 (A)	100
9	Culham	Tokamak Stellarator	5830 5830	(985) 1000	7.5 8.4	1.6 1.4	1.5 1.5	(3.5) 4.0 (H)	--- 3 and 6	2000 2000 (H)	(106) 60	200 200 (H)	(80) ---

All values given in ( ) are calculated by the author from the data given in the article.

[a]Calculated assuming $r_p = r_w$ and the smallest radial distance of magnet windings.

H and A refer to helical and axial windings.

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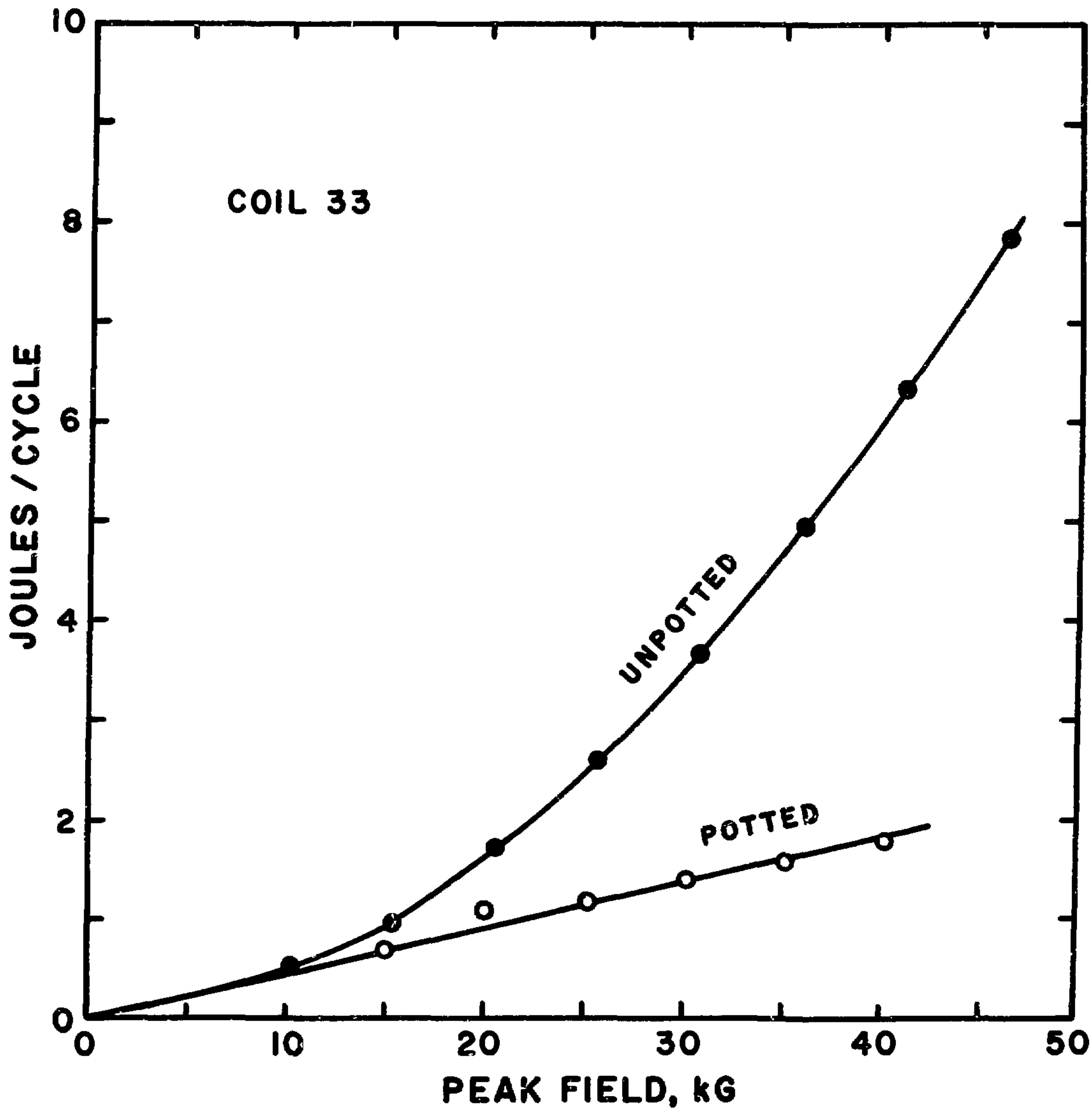

Fig. 1 The loss per cycle comparison between a potted and unpotted pulsed superconducting solenoid wound from a braid of NbTi (taken from A. D. McInturff et al.).

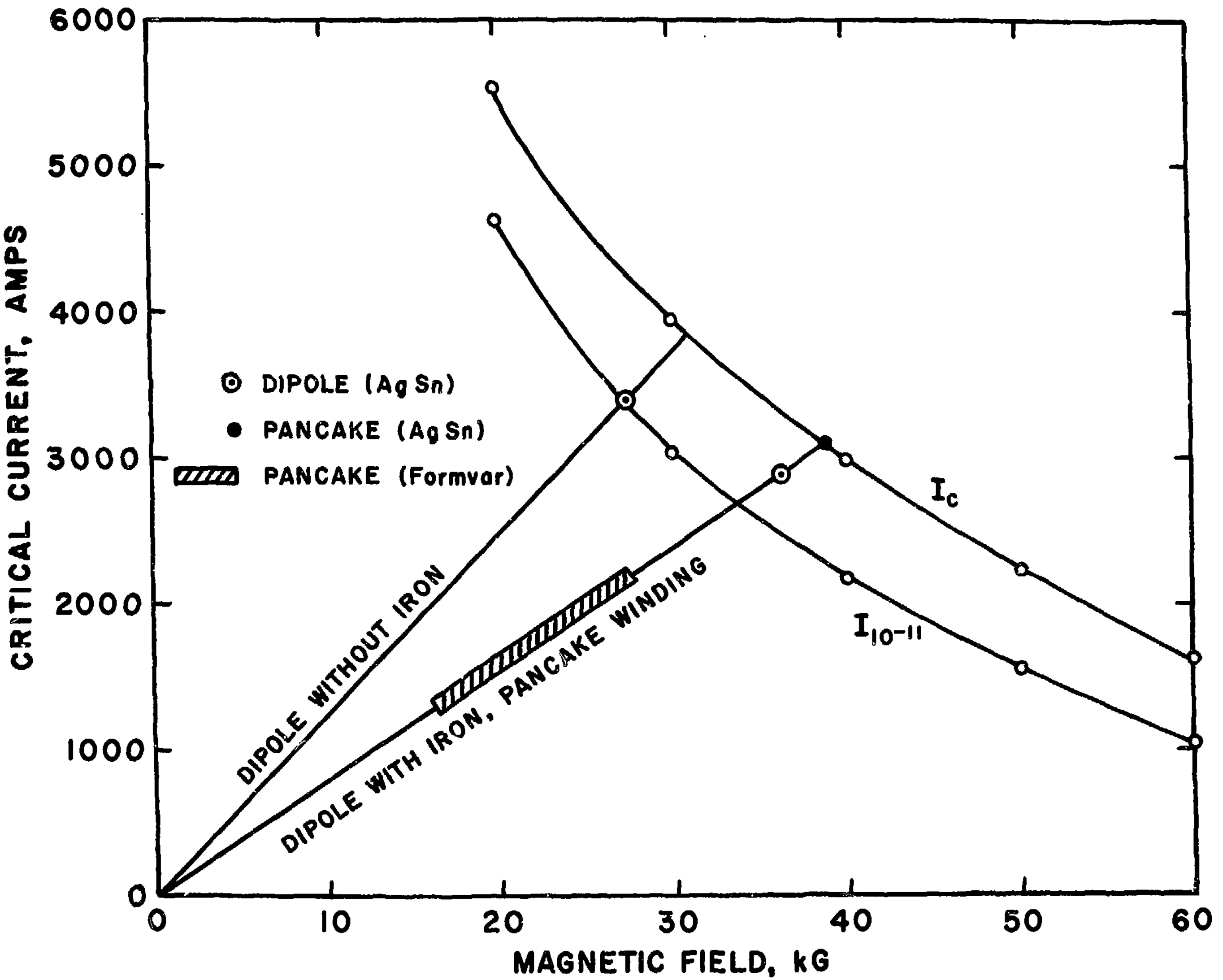

Fig. 2 A comparison between formvar and metallic coated wire in the magnet performance of pancakes and dipoles (taken from A. D. McInturff et al.).

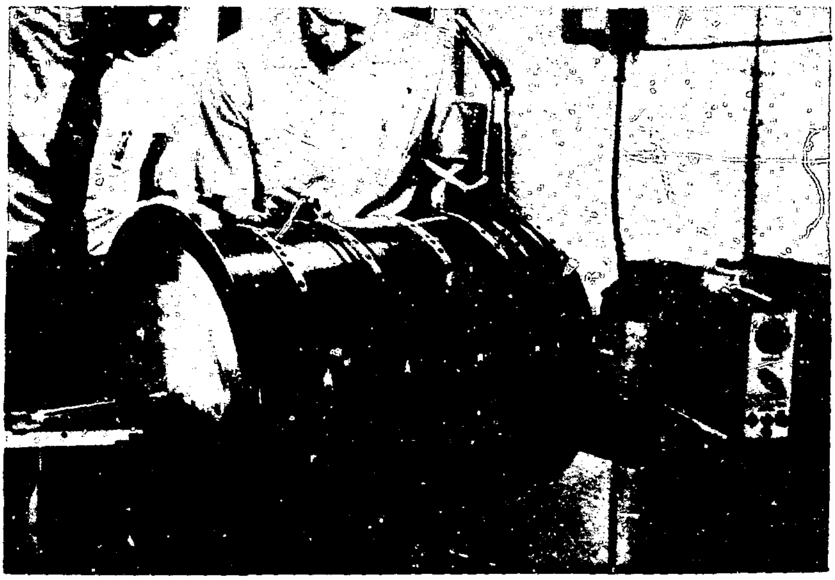

Fig. 3. A view of the magnet HYBUC being wound (taken from D. L. Coffey).

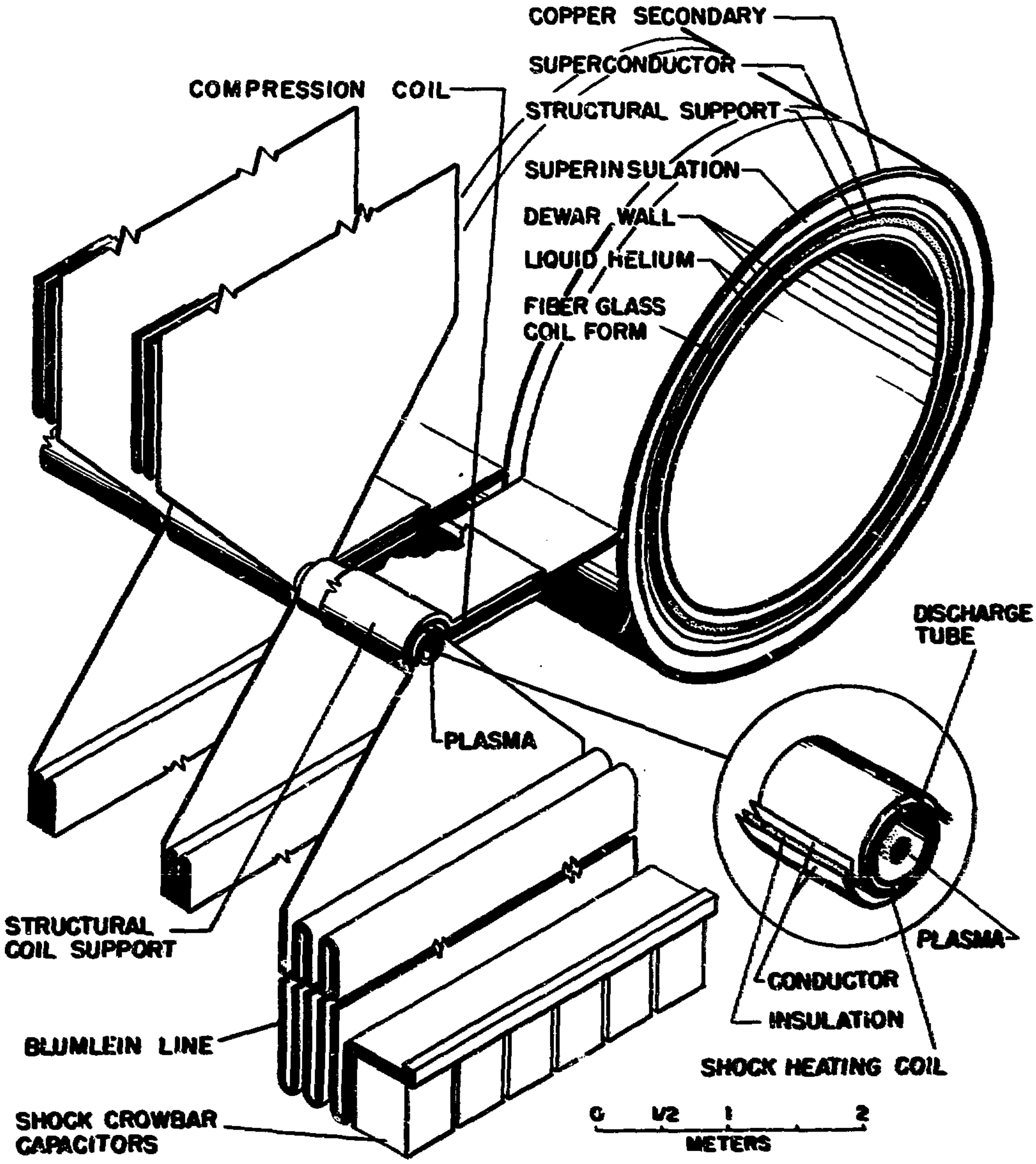

Fig. 4. Schematic drawing of proposed future Θ-pinch experiment using a superconducting energy storage coil (taken from H. L. Laquer and F. L. Ribe).

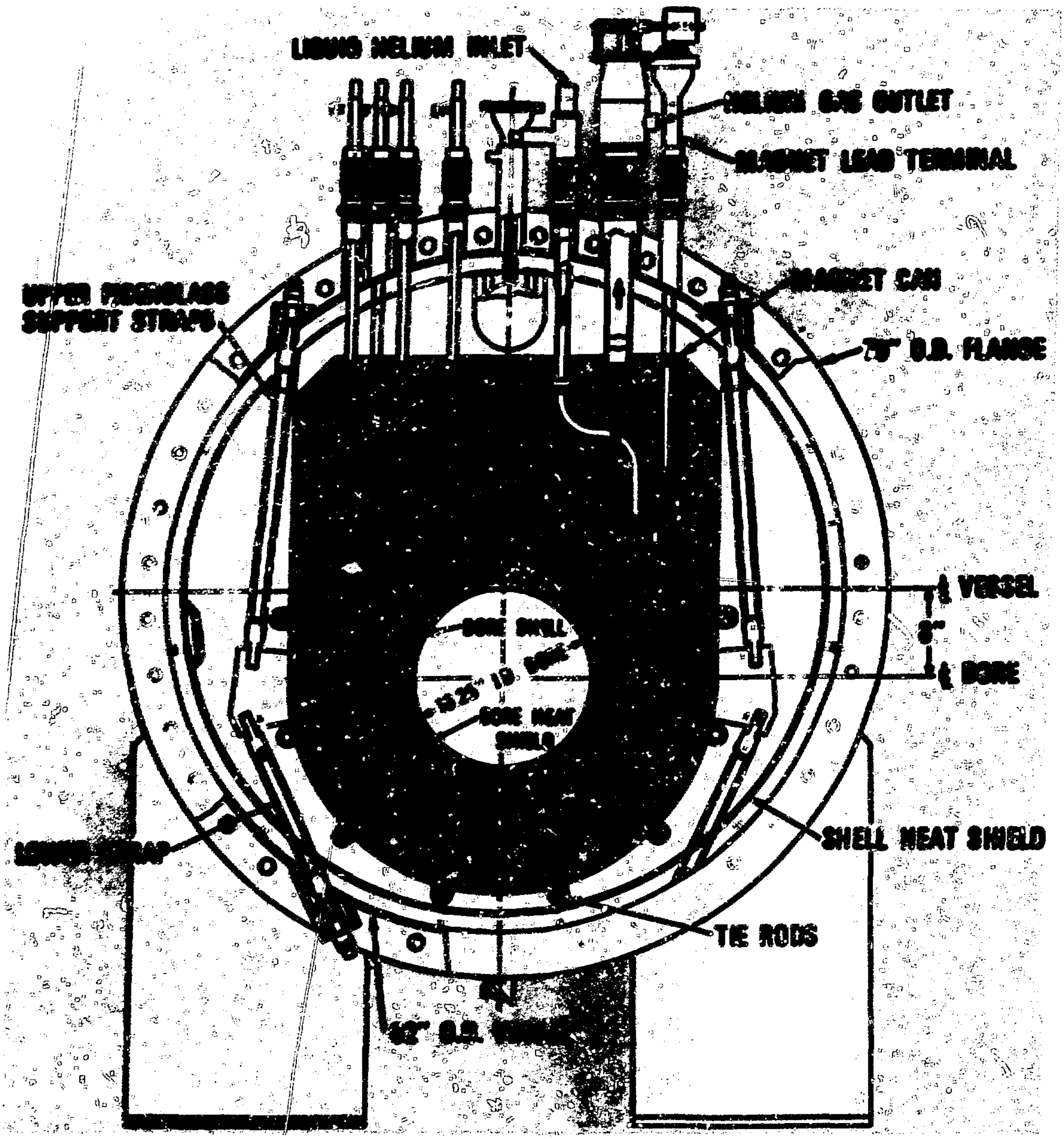

Fig. 5. Assembly drawing of the dewar for a superconducting magnet mirror facility (taken from J. E. Brewer).

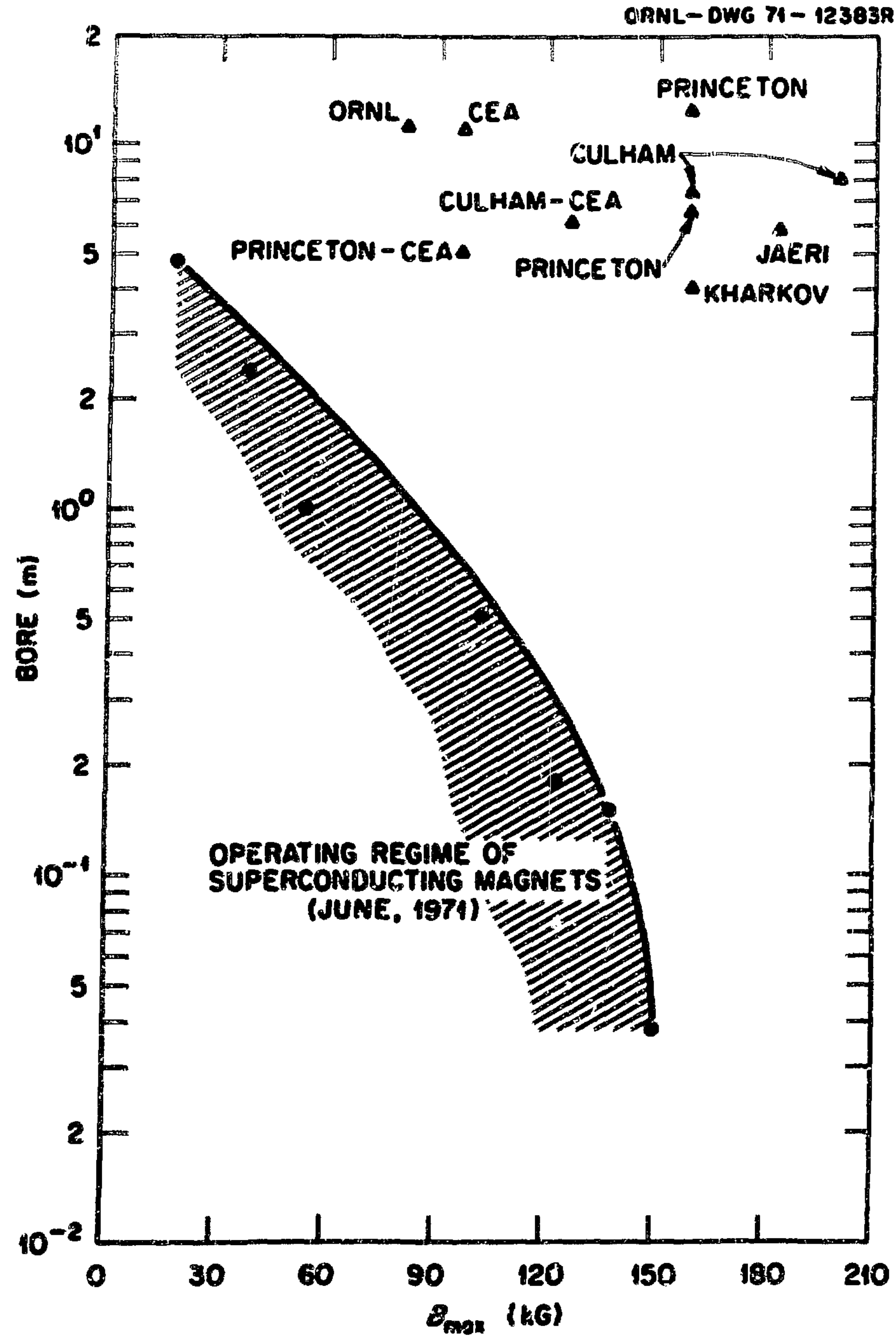

Fig. 6. The state-of-the art of superconducting magnets, Bore vs B_{max} (10 kG = 1 Wb/m^2 = 1 T). The points giving specific target parameters were taken from the tables in the review paper (M. S. Lubell). The triangles are for recently proposed fusion reactor systems (see Table I).

DISCUSSION

S. BURNETT (LASL): I would like to address myself to the comment that Stephen Blow made this morning and that you eluded to in the pricing of superconductors. At Los Alamos we have attempted to price future prototype machines by using the present day costs. Even restricting ourselves to present day costs, there is a wide range of variables concerning stabilization with copper and then eventually the stainless steel that might be used to reinforce the windings. I would like to have comments from other people who might suggest something other than using today's cost at least as a relative basis on which to price future prototype machines.

LUBELL (ORNL): For NbTi compound conductor with circular cross section or rectangular cross section with an aspect ratio (length to width) no larger than 5:1 (considering twisted filaments), we have taken the present cost averaged over a number of manufacturers in so many dollars per amp foot per kilogauss and assumed that a large quantity order would allow a reduction in this cost by one half. A couple of manufacturers agree that this is a reasonable estimate provided the quantities are approaching the order of 10^6 feet. The normal quantity runs for which you will pay the present prices are 5000 ft to perhaps 3×10^5 ft. Komarek (Jülich) has shown that the cost of NbTi has been steadily decreasing over the last five years and is presently one half of the 1966 price. However, the trend is saturating and any further decreases will probably only result from a significant increase in demand resulting in steady production runs. The future cost for Nb_3Sn or V_3Ga is harder to predict. According to present estimates by the manufacturers, a price reduction by a factor between 2.5 and 4 can likely be anticipated for extremely larger orders (3×10^5 m - 3×10^7 m) of Nb_3Sn ribbon. If we further assume that the current carrying capacity can be improved by a factor of two, the cost appears to be a factor of between 5 and 8 less than the present normalized costs. V_3Ga appears to have a similar price per foot as Nb_3Sn.

R. HANCOX (Culham): Somehow I seemed to get a reputation last night for being a pessimist, but let me put the record straight because when it comes to the cost of superconductors, I am an optimist and surely the price is going to come down. The cost of Nb_3Sn is going to come down at least by an order of magnitude if not more. The cost of Nb_3Sn at the moment in terms of weight of actual superconductor is $8000 per kg. The cost of raw materials, even being pessimistic, is $80 per kg. There are two orders of magnitude there and surely someone can make some inroads into that difference.

LUBELL (ORNL): I would like to make one more comment. The price will come down only if there are large orders and more competition. The Japanese are now producing V_3Ga, and this may help force the price down on Nb_3Sn. The only producers of Nb_3Sn at the moment are Intermagnetics General Company (which was formally the General Electric group) in this country, the spin off company from Plessey in England (but I am not sure if they are still in business), and Thomson-Houston HB in France. So if there are not enough orders to keep the companies in business and not enough manufacturers remain to keep up the competition, the price of Nb_3Sn may not come down as fast as we would prefer or as rapidly as it has for NbTi.

C. LAVERICK (ANL): I tend to agree with Roger Hancox and I guess I am an optimist too. As far as the protection of the large magnets is concerned, I think you are making too much of the problem. Even if you have 100,000 MJ, I do not think there is a problem that cannot be solved. I do not see any difference between protecting that type of magnet and protecting the big magnets that have already been built and using those as subsections of a larger system. As far as the future in superconductivity goes, I think we have to realize that in a very short time with quite a small effort we have come a long way. So we can afford to be really optimistic. I think that we should concentrate on more materials research, and it is a shame that after having done so much in the United States, for instance, the ball seems tc have been passed to Europe for development. We are getting behind in that field. Then I think that we are running into a problem of resources in the developed countries, and copper is on the list of one of the things that is going to be in short supply. So here is one good reason, but not the only one, for developing aluminium stabilized conductors as quickly as possible. There is a lot of aluminum, and even porous glass is plentiful since there is a lot of silica in the world.

LUBELL (ORNL): I do not think all the techniques for protecting large magnet systems have been used or in some cases refined, but I did not mean to imply that there were any problems which cannot be solved. Remember too that all the really large systems have utilized the completely stabilized design principle. However, I believe that as we develop more confidence in the partially stabilized coils (using adiabatically stabilized fine filament NbTi wire) and construct large ones, we probably will require more sophisticated protection schemes (including detection of the onset of a quench) than have yet been employed.

I think you raise a good point concerning the aluminum bonded superconductors. The research labs of CGE at Marcoussis is producing aluminum stabilized NbTi conductor, and Intermagnetic General Company is also offering aluminum bonded Nb_3Sn ribbon.

R. WELLS (ORNL): In regard to the coil cost, I think we should pay some attention to the fabrication of these coils. If we use a cost per pound, the tooling cost is going to increase when we jump into the large magnets even though the cost of the superconducting materials or the base metals of the coil cases may decrease. The tooling cost is going to rise when you start fabricating these large coils. I think this should be taken into consideration and some effort should be made to contact industry and keep them informed of some of the large systems that are anticipated. In one of the papers yesterday on vacuum tanks, the problems that one faces in fabricating a large vessel in the field were discussed. A simple thing like welding, which one takes for granted on some of the small experiments within the shops, becomes more difficult and requires new techniques when employed with field erected equipment.

LUBELL (ORNL): I agree with you. I think in particular that the first large superconducting magnet system will be very expensive to construct. Thereafter perhaps the techniques that are refined will be utilized at other locations and save some fabricating costs. It is difficult to anticipate all the problems that will arise when a major scale up in size is attempted. For this reason we used more than one rule of thumb in estimating the winding costs for our tokamak reactor system.

APPENDIX

SPEAKERS FOR SESSION ON ENGINEERING DESIGN OF MAGNET SYSTEMS

I. Review

1. M. S. Lubell (Oak Ridge National Laboratory)
2. R. Hancox (Culham Laboratory)

II. High Field Systems

3. D. L. Coffey (American Magnetics, Inc.)

III. Large Magnet Systems

4. J. R. Purcell (Argonne National Laboratory)
5. J. Jensen (Brookhaven National Laboratory)

IV. Toroidal Magnet Systems

6. S. L. Wipf (Institute for Plasma Physics, Garching)
7. J. R. Roth (NASA-Lewis Research Center)
8. J. File (Plasma Physics Laboratory, Princeton)

V. Non-Symmetric Magnets

9. J. D. Rogers (Los Alamos Scientific Laboratory)
10. R. Nelson (Lawrence Radiation Laboratory, Livermore)
11. K. R. Efferson (Oak Ridge National Laboratory)

VI. Energy Storage and Pulse Magnets

12. H. Laquer (Los Alamos Scientific Laboratory)
13. K. H. Schmitter (Institute for Plasma Physics, Garching)
14. A. Knobloch (Institute for Plasma Physics, Garching)
15. T. James (Culham Laboratory)
16. A. D. McInturff (Brookhaven National Laboratory)
17. S. L. Wipf (Institute for Plasma Physics, Garching)

VII. Associate Techniques

18. H. M. Long (Oak Ridge National Laboratory)
19. J. Murphy (Pittsburgh-Des Moines Steel Company)
20. J. Jensen (Brookhaven National Laboratory)
21. J. E. Brewer (Oak Ridge Gaseous Diffusion Plant)

VIII. Fusion Systems

22. M. Ohta (Japanese Atomic Energy Research Institute)

23. A. V. Georgievsky (Physical Technical Institute of Academy of Sciences of Ukranium SSR, Kharkov)

24. P. L. Hubert (Euratom-CEA, Fontenay-aux-Roses)

THE HYBUC MAGNET SYSTEM

D. L. Coffey
American Magnetics, Inc.

A high field (110 kG) superconducting magnet system has been designed and built for the Max-Planck Institut für Physik and Vanderbilt University. It is being used with the hyperon bubble chamber in Geneva.[1]

Details of the magnet design and construction were given, including general, electrical, mechanical and cryogenic considerations. The magnet employs a concentric set of NbTi and Nb_3Sn coils which are operated in series electrically. The NbTi coil section produces 85 kG in a 26 cm bore. It operates at 440 A with graded current densities of 6000 to 14000 A/cm^2 in multifilament NbTi conductor. The Nb_3Sn coil is wound of 12.7 mm wide ribbon and generates 25 kG in a bore of 17.2 cm. Together the magnets have generated 118 kG with calm conditions, or 110 kG with the vibrations associated with the operation of the bubble chamber expansion system. The magnet system includes unusual design features in the magnet cooldown provisions, superconductor stabilization, series operation (with trimming), normal state transition protection, emergency scram and vibration sensitivity provisions. It achieved full short sample performance.

[1]See The CERN Courier, No. 3, March 1971.

BUBBLE CHAMBER MAGNET COMPARISONS

J. R. Purcell
Argonne National Laboratory

TABLE A1

DESIGN VALUES OF SUPERCONDUCTING MAGNETS

	ANL	NAL	BNL	CERN
Winding ID (ft)	16	14	7.8	14
Current (A)	2000	5000	6000	5700
Ampere Turns (10^6A)	5	15	5.5	20
Field, Central (kG)	20	30	30	35
Field, Maximum (kG)	20	51.5	40.5	51
Average Current Density (A/cm^2)	800	1885	2860	1050
Inductance (H)	40	31.7	4	51
Stored Energy (MJ)	80	400	72	800
Coil Weight (tons)	50	80	20.5	100
Coil Compressive Load (tons)	200	12,000	---	9920
Refrigerator Capacity (W)	500	500	250	1500
Cost (10^6\$)	2.5	2	1	4.1

TOROIDAL MAGNETS

S. L. Wipf
IPP, Garching

The toroidal field for stellarator "Wendelstein VII" is composed of 40 separate superconducting ring coils. At present W7 is in its final design stage so the data given are not yet absolutely fixed and may still be changed somewhat.

Major torus radius R = 2m and minor radius of coil winding r = 52 cm. For better access each coil has its own separate dewar with dimensions I.D. = 80 cm, O.D. = 130 cm. This leads to special force problems. When fully energized, each coil has a centripetal force of $7 \cdot 10^5$ N. Epoxy-fiberglass supports transmit this force from liquid helium to room temperature. To guarantee the spacing between coils, 8 further supports are distributed between the side faces of neighboring coils, again transmitting the azimuthal forces between cryogenic temperature via room temperature. Hydraulic adjustments are provided. In case of the failure of one coil, all coils will be simultaneously de-energized, because the spacing supports could not cope with the full unbalancing azimuthal force caused by one missing coil.

The heat leak through the supports account for 70% of the refrigerator losses; the rest is distributed between current leads (15%), helium transfer tube (10%) and loss through cryostat walls (5%). The required refrigeration power output for all 40 coils is 400 W. In addition there will be a loss of up to 80 kJ for each pulse of the (normal conducting) helical windings. This energy appears in the copper casing at helium temperature which shields the superconducting windings from the pulsed fields.

The maximum field at the windings is 65 kG; the total field energy E_s = 65 MJ with an overall current density of approximately $\langle j \rangle$ = 10 kA/cm^2 in the winding of twisted multifilament conductor having a NbTi : Cu ratio of 1 : 2.5. According to the present state-of-the-art as given in Lubell's review, the maximum values just quoted may be too ambitious, and one must be prepared to settle for a field which is perhaps 20% lower.

At present Siemens is developing and building one prototype coil only, to be finished towards the end of the current year.

LONG-TERM RELIABILITY AND PERFORMANCE OF A SUPERCONDUCTING MAGNET FACILITY USED FOR PLASMA PHYSICS RESEARCH

J. Reece Roth
NASA Lewis Research Center

Eventual fusion reactors, whether for space or ground-based applications, must use superconducting magnets and operate reliably for long periods of time. A superconducting magnetic mirror machine has been in operation at the NASA Lewis Research Center for 6-1/2 years, as part of a research program on high-temperature plasma physics. This superconducting magnet facility was the first of its kind to be used in plasma physics or controlled fusion research.

The facility consists of two superconducting coils with 10 mil diam. NbZr wire in a mirror configuration with a 2.6:1 mirror ratio, and a 17 cm diam. clear bore. The design magnetic field at the mirror throats was 25 kG, and 20 kG are reached on a routine basis. Further information on the facility itself has been published,[1] as have the results of the ion heating experiments conducted in it.[2-4]

The coils first went superconducting on December 2, 1964, and were first used in an experiment on January 12, 1965. As of June 17, 1971, the facility experienced 556 liquid helium loadings, 525 experimental runs with the magnets charged, and 107 coil normalcies. As of this writing, the coils still operate satisfactorily, and without degradation of performance from that initially achieved. These coils each had a persistent switch, one of which failed in April, 1967, roughly 2-1/2 years after it went into service. The liquid level indicator system consisted of eight carbon resistors, a high and low level indicator in the liquid nitrogen and liquid helium canisters of each dewar. These failed one by one, until none were left by the summer of 1970. The failure mode was an open circuit in the carbon resistors, which was also the failure mode of the persistent switch heater resistor. Apparently carbon resistors are not reliable over long-term thermal cycling. The coil-to-ground insulation failed in June, 1969. This was dealt with by floating the power supply above ground.

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[1] J. R. Roth, D. C. Freeman, Jr., and D. A. Haid, Rev. Sci. Instr. 36, 1481 (1965).
[2] J. R. Roth, Rev. Sci. Instr. 37, 1100 (1966).
[3] J. R. Roth, Plasma Physics 11, 131 (1969).
[4] J. R. Roth, Phys. Fluids 14 (1971), in press.

LARGE SUPERCONDUCTING MAGNET DESIGNS FOR FUSION REACTORS*

J. File, R. G. Mills, and G. V. Sheffield
Plasma Physics Laboratory, Princeton

It will be important in fusion reactor technology to take advantage of the highest fields that available superconductors are capable of producing. These fields are already so high (> 150 kG) that the structural design appears to be the principal problem. We describe methods that may prove useful for supporting the forces that will result. In particular we treat the problem of designing a large bore 160 kG magnet. Such a magnetic field can exert a pressure on a conductor of about 15,000 psi. In a toroidal magnet, the field strength within the useful volume varies inversely with the radius from the axis of symmetry. In almost all cases the conductors generating such fields will be subject to bending moments in addition to an effective internal pressure. Our approaches are based on three ideas: (a) to remove the moments, i.e., to put the conductors into pure tension throughout much of the winding; (b) to take the necessary net forces on a simple structural element (a cylinder in compression); and (c) to make use of the techniques of force reduction. The coil presented here, is taken only as an example and no attempt was made to optimize either costs or use of several materials.

The Moment-Free Coil

A conductor tethered at either end will be stable in position if it is in pure tension and therefore not subject to any bending moments. A further advantage accrues to a superconducting or other cryogenic magnet using conductors in this manner, since large forces will not have to be transmitted through the thermal insulation.

When the conductor is in pure tension, T, it will lie in a curve of radius of curvature, ρ, such that the lateral force per unit length of conductor equals T/ρ. The lateral force per unit length on a conductor perpendicular to a magnetic field is proportional to BI, where B is the magnetic field and I the current in the wire. In a toroidal magnet field, B, varies inversely as the radius, r, from the axis of symmetry.

*Published in IEEE Transactions on Nuclear Science NS-18, 277 (1971).

Since we wish to take the net forces on a cylindrical structural element, we seek the curve tangent to a cylinder possessing a radius of curvature proportional to the distance from the axis. Except where the conductor lies flat against the support, its radius of curvature, ρ, is proportional to the radius, r, or

$$\rho = kr$$

Since the radius of curvature, ρ, is given by

$$\rho = \pm \left[1 + \left(\frac{dr}{dz}\right)^2\right]^{3/2} \Big/ \frac{d^2r}{dz^2} \, ,$$

we need to find the solution of:

$$r \frac{d^2r}{dz^2} = \frac{\pm 1}{k} \left[1 + \left(\frac{dr}{dz}\right)^2\right]^{3/2} .$$

This cannot be integrated in closed form, but can be readily solved numerically.

Use of the foregoing analysis to design toroidal magnets for 160 kG yields satisfactory dimensions and stresses for two coils tabulated in Table A2. They are coils for a hypothetical DT reactor, and a hypothetical catalyzed D-D reactor, both of 1500 MW(e).

TABLE A2

PARAMETERS FOR PROPOSED FUSION REACTOR TOROIDAL COILS

Item		D-T Reactor	Catalyzed D Reactor
Superconductor		Nb_3Sn	Nb_3Sn
Width of Conductor	cm	1.27	1.27
Thickness of Conductor	cm	0.022 - 0.116	0.020 - 0.089
No. of Pancakes/Coil		14	30
Maximum Field at Conductor	kG	160	160
Major Radius	m	5	10
Plasma Radius	m	1.7	4
Field at Major Radius	kG	91	88
Ampere Turns/Coil	10^6 A	5.62	11.0
Minimum Current Density	A/cm^2	4000	3000
Average Current Density	A/cm^2	8800	7000
Current/Conductor	A	1500	1000
Energy Stored in Field	GJ	15.4	112
Inductance of Torus	kH	13.7	224
Weight of Cylinder	10^6 lb	0.55	3.9
Weight of Coils	10^6 lb	1.9	8.8
Weight of Assembly	10^6 lb	2.5	12.7
Centering Force/Coil	10^6 lb	37.3	146
Heat Leak (Structure and Insulation)	kW	4 0	15.0

HIGH FIELD GRADIENT MAGNETS, 15 AND 25 CENTIMETER AMBIENT TEMPERATURE BORE SUPERCONDUCTING QUADRUPOLES*

John D. Rogers
Los Alamos Scientific Laboratory

A 15 cm ambient temperature bore superconducting quadrupole doublet for beam focusing has been operated with a 30 kG field at the conductor at the center of the 30-cm-long straight section and with a 3 kG/cm field gradient. Also superconducting quadrupole doublet and triplet magnet systems complete with shielded cryostats, persistent mode switches, power supplies, and 4.5 K refrigerators have been designed. These have a 31 kG field at the conductor at the center of the 45-cm-long straight section of the magnets and a 1.58 kG/cm field gradient for focusing secondary particles from the Los Alamos Meson Physics Facility (LAMPF). A 90° segment of the magnets has been tested using twisted multifilament Nb-Ti superconductor imbedded in a Cu matrix of 1.52 mm diameter at currents up to 485 A. The room-temperature beam aperture of the magnet cryostat is 25-cm. The magnet system focal length, as for the 15-cm bore doublet is 1 m for 500 MeV pions. The magnets are to be mounted in a cryostat similar to that used for the 15-cm bore doublet with thermally compensating supports to minimize motion upon cooling and an outer iron shell to reduce external fields to less than 100 G. Commercial fabrication costs have been ascertained and a cost comparison with conventional quadrupole magnets made.

*Work performed under the auspices of the U.S. Atomic Energy Commission.

THE IMP SUPERCONDUCTING COIL SYSTEM*†

K. R. Efferson, D. L. Coffey,†† R. L. Brown,
J. L. Dunlap, W. F. Gauster, J. N. Luton, and J. E. Simpkins
Oak Ridge National Laboratory

The IMP superconducting coil system consists of two mirror coils surrounded by a quadrupole coil set (four coils) which produces a minimum-B in the region between the mirrors. The mirror coils are designed to operate in the field of the quadrupoles and produce a central field of 20 kG at 383 A while the spacing between mirrors is fixed at a position to give a 2:1 mirror ratio. The maximum design magnetic field in the system is about 85 kG, and it occurs in the quadrupole windings at a quadrupole current of 815 A. The mirror coil superconducting material is a 15-strand, multifilament NbTi by Supercon (57 x 114 mils), and the quadrupole material is a laminated ribbon of Nb, Nb_3Sn, Cu, and stainless steel by General Electric (0.008 in x 0.5 in overall). It was found necessary to parallel the Nb_3Sn ribbon with an Al ribbon (0.006 in. x 0.5 in., $\rho_{300k}/\rho_{4.2k} > 2000$) for improved stabilization. The coil system has been completed and partially tested. Sixty-two percent of the design field in both sets of coils is required for plasma experiments. This point was achieved two times with no transitions. A transition occurred with mirror current fixed at 172 A (45%) when the quadrupoles were charged to 605 A (74%). The coil system has been installed in the IMP machine and are being used operationally. Further testing has been postponed until the plasma physicists have need of higher magnetic fields. Design, construction details, and experimental work leading to the construction of the quadrupoles were discussed.

*Research sponsored by the U.S. Atomic Energy Commission under contract with the Union Carbide Corporation.

†Published in IEEE Transactions on Nuclear Science NS-18, 265 (1971).

††Present address: American Magnetics, Inc., P. O. Box R, Oak Ridge, TN 37830.

ENERGY STORAGE AND SWITCHING WITH SUPERCONDUCTORS*†

H. L. Laquer and F. L. Ribe
Los Alamos Scientific Laboratory

Inductive magnetic energy storage with superconductors or cryogenic aluminum.conductors will probably be used to provide the magnetic fields needed in pulsed thermonuclear reactors and in some large scale pulsed plasma physics experiments designed to demonstrate the scientific feasibility of controlled fusion. In the latter case, the problem of switching large currents between inductors may also be handled cryogenically, i.e. by a superconducting "switch" which is made to change to the normally resistive state.

On the basis of an analysis done by R. R. Hake at Los Alamos during the summer of 1970, we can state that the minimum volume (V) and hence cost of superconductor in the switch is determined only by the maximum voltage (E) and maximum current (I) produced by the storage unit, and by the critical current density (J_c) and normal state resistivity (ρ_n), of the switch material thus

$$V = EI/J_c^2\ \rho_n$$

For high speed transfer (high E) this volume would be prohibitively large. For this reason we propose a separated shock Θ-pinch with low energy high voltage (μsec) Blumlein lines to initially heat the plasma, followed by slower (msec) adiabatic magnetic compression and containment.

We have shown that the simplest way of switching a current carrying superconductor into the normal state is by increasing the current density and that in short samples fractional microsecond switching times are attainable. In the limit of sufficiently large rates of current rise these times

*Work performed under the auspices of the U.S. Atomic Energy Commission.

†Paper presented at Energy 1971, Boston, Massachusetts, August 6, 1971.

depend only on the diameter of the superconducting wire. Braiding individual wires to accomodate larger currents does not appear to cause undue degradation either in switching times or current densities.

Work is presently underway on a 30 kJ, 1000 A, 30 kG storage and switching experiment. We plan to do an order of magnitude larger (300 kJ) experiment with a transformer coupled load in the near future. The ultimate goal is a 50 m major diam. toroidal inductor storing 850 MJ and capable of delivering 200 MJ to an 85 kG plasma compression coil.

FUTURE 100 MJ FAST CAPACITOR BANKS

T. E. James
Culham Laboratory

It is possible that fast high current pulsed power supplies of a few 100 MJ with current rise times of about 10 μsec will be required for the next generation of fusion research experiments or ultimately for fusion reactors. I would like to consider some of the problems that will arise in 40 kV capacitor banks of this size which is an order of magnitude greater than the largest existing bank (11 MJ Syllac bank at Los Alamos).

Most fast megajoule banks operating at present were designed in the period 1960-65 using comparatively large numbers of low performance components. Subsequent development of capacitors and spark gap switches has increased their individual ratings about five times, to 15 kJ/unit and 50 coulombs/switch respectively, which has reduced both bank size and complexity appreciably. The overall size of a 50 kV, 1.0 MJ, 15 MA bank now being built is 4 x 4 x 3.5 metres, including switches, connections, subcollector plates and space between units for maintenance, which is equivalent to an energy density of 18 kJ/m^3.

The major technical problem envisaged in fast 100 MJ banks is associated with the reliability of the switching system due to the larger number of components involved. The failure rate of existing single spark gap switches at 50 coulombs/pulse is of the order of 1 failure in 10^4 shots, and therefore it is desirable to limit the total number of switches to about 10^3, resulting in switch ratings of 100 kJ and 100 coulombs/pulse for a 100 MJ bank at 90% voltage reversal. Preliminary spark gap tests suggest that an acceptable failure rate will be achieved at these ratings.

Another significant parameter in large banks is their overall size which is mainly dependent on the energy density of the capacitors and the manner in which the various components are arranged and interconnected. Based on an energy density of 18 kJ/m^3, discussed above for megajoule banks now being built, a 100 MJ bank would occupy a volume of 5,500 m^3, possible dimensions being 100 m long (based on 1 MJ/metre length), 11 m high and 5 m deep. However, it is reasonable to suppose that over the next ten years the energy density of capacitors will be doubled which should enable banks with an energy density of about 30 kJ/m^3 to be built. In this event, the total volume of a 100 MJ bank would be 3,300 m^3 with dimensions of about 100 x 8 x 4 meters.

RECENT DEVELOPMENTS IN PULSED MAGNET MULTIFILAMENT SUPERCONDUCTING WIRES

A. D. McInturff, P. F. Dahl, W. B. Sampson, and K. E. Robins
Brookhaven National Laboratory

For a period of less than a year, it has been evident that the severity of motion stability, as well as heat transfer problems, might limit the maximum superconductor to normal metal matrix volume ratio that could be used reliably in organically insulated multifilament wires, thereby reducing the maximum current density obtainable in a pulsed magnet.

Metallic insulated multifilament wires, on the other hand, are very reliable and only limited by the short sample characteristics of the superconductor,[1,2] having greatly increased both adiabatic and dynamic stability and excellent heat transfer. The early metallic conductors, however, had a very low $\dot{B}_{min}$ for coupling between the individual wires, thus reducing their utility to dc or very low frequency devices of a few tenths cycles per minute.

At Brookhaven a program was started to increase $\dot{B}_{min}$. The preliminary measurements of the effectiveness of the various metallurgical steps on the time constant of interwire coupling utilized magnetization techniques. These measurements were made in a manner similar to those described in earlier papers.[3,4] Such a metallic and intermetallic insulation was fabricated and indeed showed decoupling in the magnetization tests at reasonably synchrotron frequencies. Long lengths of such material were fabricated, and performance and loss tests were performed on test solenoids.

For completeness, a review of dipole data will also be included, both of our laboratory and of other groups.

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[1] W. B. Sampson et al., Particle Accelerators 1, 173 (1970).

[2] W. B. Sampson et al., IEEE Trans. Nucl. Sci. NS-18, 660 (1971).

[3] A. D. McInturff and A. Paskin, J. Appl. Phys. 40, 2431 (1969).

[4] A. D. McInturff and J. Claus, Proc. 3rd Intern. Conf. Magnet Technology, Hamburg, 1970 (in press).

It is clear that if the dipole is not required to have a high $\dot{B}$ > 10 kG/sec the metallic and intermetallic insulations offer a much more stable and reliable system as well as enhanced performance per unit superconductor volume. At present investigations are being carried out utilizing thermoplastics that show great promise, and in the future probably the best system will be a combination of both.

Another promising system that seems very compatible with the metallic insulation technique was recently reported on by Suenaga and Sampson utilizing V_3Ga multifilamentary wire.

PULSED COILS FOR INDUCTIVE ENERGY STORAGE

S. L. Wipf
IPP, Garching

Superconducting inductive energy stores can offer advantages if they are faster than flywheel generators and big enough to be cheaper than capacitive energy stores. Coils in the MJ range or larger are needed with discharge times substantially shorter than 10^{-1} sec without causing significant losses. Each of these requirements, if posed separately, bring us to the limit of the present state-of-the-art (see contributions of Lubell and of McInturff), and together present as yet unsolved problems. Some problems are discussed:

(A) Energy Transfer

The discharge of an initially shorted current carrying storage coil (persistent mode) into an inductive load by changing the shorting link from zero to a high resistance (i.e. opening a switch) results in the loss of at least half the stored energy in the switch (or in a parallel resistance e.g. at higher temperature). The process is similar to the inelastic collision of two masses. The equivalent of the lossless elastic collision is achieved by paralleling the switch with a capacitor capable of storing half the energy. For large enough energy stores the resistive transfer will be used.

(B) Normal-Superconducting (n-s) Switch

A high resistance in the open switch gives a low cryogenic loss. The rating of the n-s switch material is given by $\rho_n j_c^2$. For NbTi ($\rho_n \approx 5 \times 10^{-5}$ Ωcm, $j_c \approx 10^6$ A/cm^2) this is 5×10^7 W/cm^3, which in an actual switch is reduced by the packing factor. Two experimental alternative materials are the PbBi in porous glass superconductor[1] which has $\rho_n \approx 5 \times 10^{-3}$ Ωcm, $j_c \approx 10^5$ A/cm^2 or the same power density as NbTi; and sputtered NbN[2]

[1] J. H. P. Watson, J. Appl. Phys. 42, 46 (1971), also private communication.

[2] J. R. Gavaler, M. A. Janocko, C. K. Jones, and A. Patterson, J. Appl. Phys. 42, 54 (1971).

($\rho_n \approx 10^{-3}$ Ωcm, $j_c \approx 10^7$ A/cm^2) with 10^{11} W/cm^3. It is clear that the actual energy dissipated has to be limited to less than 10^3 J/cm^3, corresponding to the heat needed to melt the material. The serious problem is the stabilization of the switch, which in its s-state is part of the large storage coil. A measure for the stability is the power dissipation per unit surface in the n-state, $\rho_n \cdot j^2 \cdot D$ (D being the volume/surface ratio of the material) which should not exceed approximately 1 W/cm^2. This requirement reduces the potential power density of a n-s switch by many orders of magnitude and leads to switch volumes which surpass the volume of the storage coil.[3]

(C) AC Losses of Coil

If the superconductor is thinner than ~ 10 μm, it can stand an adiabatic field change of 80 kG without heating above T_c (~ 10 K). Modern composite conductors have even smaller filaments, but for frequencies above ~ 10^2 Hz the losses become orders of magnitude higher because they are then governed by the diameter of the whole composite. It will be necessary to develop a filamentary conductor with an isolating matrix; the stability problems mentioned under (B) will have to be considered. With many filaments in parallel and lacking perfect transposition, the necessity of resistively transferring energy which is stored inductively between filaments requires the dissipation of energy as mentioned above under (A); this aggravates the stability problem further.

The problems under (B) and (C) could be solved with an intrinsically stable material, i.e. material characterized by $dj_c/dT > 0$ over a sufficient range of temperature.[4]

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[3]A similar problem exists for flux pumps and is discussed in: S. L. Wipf, "Cryogenic Engineering," Proceedings of the 1st International Cryogenic Engineering Conference Kyoto, 1967 (Heywood-Temple, London), p. 137.

[4]M. S. Lubell and D. M. Kroeger, presented at the Conference on the Science of Superconductivity, Stanford, August 1969. To be published in Physica.

CONSTRUCTION OF LARGE VACUUM CHAMBERS AND RELATED CRYOGENIC STORAGE

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The development and construction of large vacuum chambers has seen major advances in sealing material selection, pumping methods, advances in cryogenics, low temperature refrigeration, and insulation.

Design for the shell of a large vacuum chamber is developed from, but not limited by the ASME Unfired Pressure Vessel Code, Section VIII. Shell material is generally an austenitic type stainless steel. Large penetrations of the shell require analysis of vessel loads around the opening and across the opening. Design of the door seal must consider vacuum loading and sealing. A seal independent of the structural load provides a good seal with advantages in fabrication.

Welding procedure for vacuum vessel construction should be to the standards of the ASME Code. Final assembly of the large vacuum chamber is in the field where special attention must be given to material handling, welding, polishing, cleaning and leak checking, and acceptance testing.

The large vacuum chamber is generally the focal point of the laboratory, hence the laboratory and supporting equipment must be designed around it.

The design of large scale storage dewars for liquified products of air is developed on the basis of the liquid pressure and temperature, heat leak, refrigeration, flow rates, necessary valving, and instrumentation.

CRYOGENIC ENGINEERING ASPECTS OF PULSED AND STEADY STATE SUPERCONDUCTING MAGNET SYSTEMS

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(A) Introduction

The relationship of cryogenic engineering to superconductivity is so close that an attempt to consider the problems of the latter without regard for the former is to invite failure. Cryogenic engineering must be considered in the broadest sense, including mass and heat transfer phenomena in the fluids and solids within the device as well as the basic refrigerator and cryogenic enclosure.

A summary of "state of the art" refrigerators is presented, including size, efficiency, estimated costs as a review. A discussion of the available modes of heat transfer is given, with some advantages and disadvantages of each. Some basic problems of heat transfer within the device and power leads and the effects of these on the overall system are presented.

(B) Summation

1. Refrigerators themselves do not present any new problems that the present technology cannot overcome. Large sizes will come with the demand and the prices will probably be lower than present (on a per watt basis), particularly if a number of machines are required. Reliability will come with operating experience and should be quite high.

2. The type of cooling mode used must take many factors into consideration, but there appears to be sufficient choices.

3. Heat transfer phenomena within the superconductor package is complex and requires a great deal of experimental and mathematical work to be fully understood.

4. Current leads may require large amounts of refrigeration in proportion to the rest of the loads and should be considered very carefully.

DEWAR DESIGN FOR THE SUPERCONDUCTING MAGNET MIRROR FACILITY

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Oak Ridge Gaseous Diffusion Plant

The Engineering Division of the ORGDP designed the dewar for the "Superconducting Magnetic Mirror Facility" for NASA - Lewis Research Center, Cleveland. The system will be used in plasma research directed toward controlled nuclear fusion.

The dewar is 62 in. O.D., 16.25 in. I.D., and 110 in. long, with the bore axis horizontal. The research equipment will be installed in the bore. The dewar houses 4 - 50 kG superconducting magnets and consists of 9 separate sections. All sections are flanged and machined to close tolerance for interchangeability of magnet and spacer sections. The 6 in. and 9 in. thick spacers contain 4 viewing ports at 90° for monitoring equipment and power lines. These spacers can be increased to 12 in. and 15 in. spacers. The unit can be assembled without any spacers with minor modifications.

<u>Internal Design Pressure</u>: High vacuum to 40 psig

<u>Design Forces</u>: 80 tons in either axial direction
80 tons vertical - downward
15 tons horizontal - either side

(Only one of these forces is assumed to occur at one time.)

Magnetic attraction (spacer loading) = 640 tons.
Separating force if one magnet is connected wrong = 50 tons (tie rod loading).
Magnet Weight = approximately 1.25 tons each.
Helium steady state heat loss: 38 W
Bore, viewing ports, and outer shell to be room temperature and pressure.
Tie rods, magnets, magnet can and spacers temperature: -452° F.
Dewar material of construction: 310 and 304L stainless steel (310 at LHe temp.)
Stored energy in magnets = 20 MJ.

The unique features of the dewar are the supporting system and heat shield system that solved the two major design problems.

The heat shields are copper plates roll bonded together and the waffle pattern inflated to 0.15 in. overall thickness, which permitted the heat shield and 2 vacuum spaces to fit into a 7/16 in. space. These commercial units are rated for 150 psig. (Deformation occurs at 300 psig and failure at 900 psig.) These heat shields will be gold plated to prevent oxidation and loss of emissivity.

The filament wound fiberglass support strap has a 9000 lb. design load. The test strap 13 in. long and approximately 0.28 in. in area failed at the rounded end at 17 tons, at 122 kpsi tensile stress in straight section and E of 6.4×10^6 psi.

At 520° F Temperature difference, the heat transmission for a 12 in. long strap 0.50 $in.^2$ area is approximately 0.30 BTU/h and calculated thermal contraction is approximately 0.01 in.

Adjustment of the support straps is made from outside the vessel and "O" ring gaskets maintain vacuum seal.

Fiberglass straps were fabricated from "S" glass, unidirectional wound filament, 4080 filaments per strand with epoxy resin, 70% glass (Ref. - Goodyear Aerospace GER-11214S/11.) Fiberglass strap cost is approximately $80 each.

Approximate dewar cost = $50/Lb. of magnet weight (does not include any NASA participation). Replacement magnet cost is $100,000.

SUMMARY OF TOKAMAK SYSTEM

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We discuss the design procedure of a tokamak type fusion reactor under the various physical and technological restrictions. The optimum design is considered to be the maximum power output for a given major radius of the reactor. This optimum design is also equivalent to that of the maximum power density for a given major radius.

The three important parameters are derived during the optimization. Those can be named as (a) the plasma parameter, (b) the reaction parameter, and (c) the geometrical parameter. The first is determined from the equilibrium and stability conditions. The second is a function of temperature related to the reaction rate and the third is expressed by the dimension of the structure and the intensity of toroidal field of the reactor. Because the magnetic field is limited either by the mechanical strength or by the critical field of the superconducting materials, as a result the maximum of this factor is determined only by the geometry.

An example of the optimum design is given in Table A3.

*Toshiba Electric Company.

TABLE A3

TOKAMAK REACTOR MODEL

Total power, P_t	MW	5000
Apparent heat flux, P_w^a	W/cm^2	1300
Effective heat flux, P_w^e	W/cm^2	200
Temperature, T	keV	20
Plasma parameter, ℓ		0.3
Confinement time, τ	sec	0.6
Current duration time, τ_I	sec	1000
Major radius, R	m	5.2
Plasma radius, a	m	1.1
Aspect ratio, A		4.7
Core radius, b	m	1.5
Radial thickness of toroidal coils, s	m	0.9
Toroidal magnetic field, B_{t0}	kG	80
Maximum toroidal magnetic field, B_{tm}	kG	184
Magnetic field of the core, B_b	kG	75
Maximum allowable stress, σ	ton/cm^2	1

INVESTIGATION OF MAGNETIC SYSTEM OF THE STELLARATOR - A FUSION REACTOR

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A promising scheme of the fusion reactor comprises the closed system where plasma is isolated from the chamber walls by means of a high magnetic field. At present there are two types of closed magnetic traps for confining hot plasma which are distinguished mainly by the methods of producing a magnetic field with rotational transformation (namely, "Tokamak" and stellarator).

The stellarators, as to their size and results obtained, are today second to tokamaks, but at the same time, comparison between the best stellarator systems and the tokamaks under equal operating conditions offers every reason to consider the both types of closed traps to be a rather promising basis for a controlled fusion reactor in the future.

Not the least of the factors beneficial for developing systems with external conductors would be the possibility of working in stationary conditions and the possibility of applying various methods of plasma heating. Besides, it is possible to produce the required field configurations without participation of plasma in stellarators, and as Gourdon showed, it is possible in torsatrons to solve an important problem about the divertor which does not disturb the configuration of the confining magnetic field.

The problem of producing an economical controllable reactor on the stellarator basis amounts to developing a rather simple and effective magnetic system.

The object of the given work is both to choose such a stellarator-type magnetic system which would be suitable for the controlled fusion reactor, and optimize its parameters as well.

Conclusion

Calculations of the reactor-stellarator magnetic system with the output P = 4500 MW(e) support our confidence in its prospects as the basis of a stationary fusion reactor.

The accomplished work allows the following conclusion to be made:

1. The torsatron scheme of the stellarator has considerable advantages over the classical scheme, as it allows for using the divertor which does not disturb the topology of the magnetic field and reduces production and operational costs of the magnetic system, as a necessity in longitudinal field winding falls away, and ponderomotive forces acting upon the helicoidal winding in optimal torsatron version are much lower than the forces acting upon the helicoidal winding of the classical stellarator.

2. The maximum value of the magnetic field strength H_{max} on the turns of the helicoidal winding in optimal version is approximately 2 times more than H_0 in the centre of the plasma volume and does not change greatly according to a scheme of magnetic system (for classical stellarator this ratio is 10 - 15 percent lower) and the law of conductor winding (for equally - inclined winding this ratio is 10 - 15 percent higher).

3. The ratio between minimum and maximum values of magnetic field strength in the helicoidal winding pole is equal to $\sim 1/3$, thus giving possibility for using combined poles consisting of various conductor materials according to the (jH) value for the conductor.

4. Economic comparison between magnetic systems with superconducting windings and windings made of pure aluminium shows the cost of electrical energy produced by the reactor with aluminium windings to be about 20% higher at approximate equality of capital investments for erecting magnetic systems.

Economic showings of the reactor-stellarator with optimized magnetic system are not worse than similar showings for the reactor with conventional toroidal winding of longitudinal field.

FORCE BALANCED TORSATRON COIL

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In a torsatron coil, force balancing is achieved when the average magnetic force towards the main axis is compensated by the tension of the wire. This situation can be obtained with an adequate combination of helical pitch modulation and of vertical field intensity. Simultaneously the production of vertical field can be arranged in order to suppress the stray field produced at large distances by the system.

As an application, the features of a coil for an eventual torsatron reactor are calculated in a simplified fashion. The results are listed in the table.

In conclusion it appears that the ratio : superconducting material weight divided by power output, is about 1.5 greater for the Torsatron than for the Tokomak. This apparent disadvantage is not a real one because the margin of uncertainty is probably larger. Moreover with the quoted amount of superconductor the Torsatron is provided with a divertor and with ample free space for injection. This is not the case for the Tokomak. It is doubtful whether the introduction of a divertor and of access holes in Tokomak design can be envisaged without increasing the cost by a sizeable factor.

TABLE A4

MAIN PARAMETERS FOR AN EVENTUAL TORSATRON REACTOR

Magnetic Field on Axis, B_o	kG	37
Average Small Radius of Winding, r	m	5.5
Large Radius, R	m	22
Aspect Ratio, A		4
Multipolarity, ℓ		3
Periods per Helix, m		4
Current in Each Helix, I_H	10^6A	34
Plasma Radius, r_p	m	2
Blanket and Shield Thickness, t	m	2
Relative Plasma Pressure, β		0.14
Thermal Power Output, W(th)	GW	10
Radius of Conductor, a (Circular Cross Section Assumed)	cm	90
Average Current Density, j	A/cm^2	1340
Maximum Field on Conductor, B_{max}	kG	97
Total Compound Conductor Weight, M	10^6kg	3.3
Tensile Stress, T	kg/cm^2	940
Compensating Loop, R_c	(m)	42
Compensating Loop, I_c	10^6A	28

SESSION 8

ENERGY CONVERSION SYSTEMS

Chairman

R. W. Werner
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SUMMARY OF SESSION

ENERGY CONVERSION SYSTEMS

R. W. Werner
Lawrence Radiation Laboratory

The session on Energy Conversion was divided into three principal subsections. One was on direct conversion of fusion energy to electricity, which is characterized in one principal aspect by not being Carnot limited in efficiency. The second section was one in which conventional thermodynamic cycles are used and where obviously Carnot limits on efficiency exist. Finally, we had a section on fuel cycles, that is D-D, D-T, etc. Fourteen separate papers were presented and in looking back on yesterday's session a unifying theme is difficult to formulate. It is apparent, however, that many models of fusion reactors exist. Because of the existence of these many models and the fact that at some time in the future we must, of necessity, narrow the choice of candidates down to a lesser number, I concur with Dave Rose who remarked at the conclusion of the Culham Conference that the statement "Many were called but few were chosen" is an appropriate caveat as regards the number of different reactor concepts we presently have versus the number with which we must end up. It is evident we have not yet reached that point in time where we logically can have a lesser number of reactor candidates. As a matter of fact, we are properly going the other way. We have the antithesis of lesser candidates and the counter statement that "Many were called and even more showed up" would be appropriate to this meeting. This is evident by the much larger than anticipated number of people in attendance, each with his own ideas and his own thoughts. These many ideas and thoughts are applicable to reactor design, fuel cycles, blanket designs, conversion systems, etc. Every facet of fusion reactor technology has grown since we met at the Culham Conference and many new facets and many new ideas have been added. Some old ideas have stood the test of time and improved. This growth of ideas is a healthy attitude. We must now begin to increase the work on problems beyond the fringe of plasma physics and study the technological problems that we face in reasonable detail. We must recognize that there are and will be a potentially large number of solutions to these technological problems. We recognize that there is no one answer to one particular problem at this point in time. Our meeting encompassed a rather wide spectrum of philosophies, the consequence of which served to create different design situations and different design solutions.

The philosphy ranged from that in which technologists and scientists think in terms of pilot plant reactors in which ordinary materials such as stainless steel are used to that philosophy in which they think in terms of the ultimate reactor. Thus, for instance, at this meeting we had proposals and concepts and ideas ranging from very ordinary material usages to the use of very high temperature refractory materials and complex systems.

A little ray of financial sunshine emerged from the talk that opened the session, which was given by Walt Rosengarten (Philadelphia Electric Co.). In his opinion, fusion reactors could have a lower impact on the environment. That is to say they could generate less waste heat, particularly if the concept of direct conversion is adopted. The public utilities are showing increased interest in fusion and may support some research in a financial way. This is something they have never done before; that is, underwritten any basic research to any great extent, and they hope to do it at a level like one or two percent of their gross revenue. So we in the fusion area have a chance of being funded by the utilities to some degree. Rosengarten indicated he felt fusion was good for the future but probably costly and still a long way off.

In the first technical paper that was delivered, Husseiny discussed a pulsed reactor in which ^3He plasma is bombarded by energetic deuterium atoms.[1] No neutrons are released and the electron temperature is low so that radiation is reduced compared to other concepts. This system which is termed "Fast Fusion Reactor" is characterized by its small metawatt size and absence of neutron activation and afterheat problems. Table 1 shows a comparison between conventional and fast fusion reactors.

TABLE 1

	$n\tau$ (sec/cm^3)	Neutron Flux $(cm^{-2}\ sec^{-1})$	Radiated Power On The Wall (kW/cm^2)
CONVENTIONAL			
T = 60 keV	5×10^{14}	2.88×10^{13}	21.8
FAST			
W_D = 480 keV	$n_3\tau = 2.34 \times 10^{13}$	———	6×10^{-4}
T_e = 5 keV	$n_D\tau = 2.8 \times 10^{10}$		

Small size reactors based on toroidal geometry, and by small size I mean size in terms of the number of megawatts and not physical size, is a new trend that seems to be emerging at this meeting. Fraas and Mills, for instance, have both suggested smaller megawatt size plants than we have heard about before this. At Culham it was demonstrated that plants had to be on the order of 5,000 megawatts or greater to be economically attractive particularly for toroidal systems. Here at this session, some seem to be going the other way and reversing this idea. Systems that are 100 megawatts or in that size bracket are discussed. And the reasons for these smaller plants and their economic credibility is not particularly clear except in mirror machines with direct conversion where high betas are involved and different scaling laws also are involved. The logic that allows considering small megawatt toroidal reactors is not clear unless it is necessary (but not particularly economic) because radiation damage time can be extended by the expedient of cutting down on the wall fluxes an order of magnitude.

Yoshikawa discussed the use of divertors in toroidal systems and the integrating of divertors with a direct convertor in Tokamaks.[2] I think insufficient attention has been given to divertors for toroidal systems up to this time. We have considered them as small black boxes for which no one has wanted to be responsible. Bob Mills of Princeton has provided some good initial input on divertors.[3] I am pleased to say that Yoshikawa is also now beginning to look at the problem. Much more effort in this area is required. Yoshikawa considered the possibility of incorporating a divertor or a direct energy convertor to a Tokamak. The configuration considered was a Tokamak with a non-circular cross section, such as a triangular cross section. Outside the plasma the field lines move away from the plasma and can be connected to an axisymmetric divertor or an energy convertor.

The extence of the divertor protects the vacuum wall from sputtering It also provides the thermal insulation of the plasma from the wall and thus greatly reduces the heat conduction loss of the plasma.

According to the neoclassical transport theory, a typical Tokamak reactor has a much longer confinement time than required to keep the temperature constant. In order to prevent a temperature run away, it is necessary to enhance the particle loss by some external means. The particles lost can be fed to a divertor or a direct energy convertor. The power released at the divertor generates electricity via an ordinary thermal cycle or alternately, a direct energy convertor may be connected to produce electricity at high efficiency. The cost of the direct convertor seems excessive, if only one outlet is provided for the plasma. However, by providing several outlets to the plasma and stacking the direct energy convertors connected to them, the cost seems to be lowered to a competitive level.

In the area of direct conversion of charged particle energy to electricity, one very important observation can be made. This is that actual experimental work is being carried on by Moir and his associates at LRL and their results are very encouraging.

As is generally recognized, the direct conversion technique under study resembles somewhat the operation of a Van de Graff accelerator in reverse. Escaping ions and electrons are guided and directed magnetically in a structure called an "expander", following which they are separated from each other and pass into "collector" structures within which decelerating electric fields are maintained. As each particle loses its kinetic energy, it is deflected and collected at high electrical potential, thereby creating a source of high voltage dc power (spread over a range of potentials corresponding to the random spread in energies of the incident particles). Conventional inverter and rectifier circuits are used to convert these separate currents to a source at a common potential, for use as HVDC. The critical element of this direct convertor system is the collector, within which the final deceleration, sorting and current collection is performed. Scale model laboratory tests on which Moir is working plus computer simulation calculations, have been performed that prove out the design approaches being taken. Using an ion beam of variable energy to simulate the plasma ion streaming, measured efficiencies in excess of 80% were achieved. Steps are now underway to further improve the collector structure to the end of achieving efficiences approaching 90% under even more realistic operating conditions. Following these tests it is expected that a direct convertor would be attached to 2X II or Baseball II to test the idea under operating conditions that would be much closer to those to be met in an actual reactor.

Those of us that are in the technology group (reported by J. D. Lee) have considered some of the economic implications of Q as it relates to mirror machines with direct conversion.[4] By Q we mean the ratio of fusion power to net injected power. In our studies where we look at Q as an open parameter, we found generally that in D-T systems with expected Q values that are in the range of 1 to 2, the reactor systems are economically attractive. In D^3He systems, on the other hand, with their inherently lower Q values, (values like .2 or .4) present systems are economically marginal at best. This strong influence that Q has on economics means that the physicists must do something to enhance the Q values of D^3He systems if they are to be competitive with other systems. There should be a number of possibilities or potential ways of doing this. I believe that the Culham people have some ideas along these lines and Don Sweetman may want to make some comments. Preliminary studies show direct electrostatic conversion in mirror reactors can have a dramatic effect on overall system performance. A good example is the Q's required for a break-even power balance with and without direct conversion. With direct conversion, Q_{DT} must be ≈ 0.8 and $Q_{D^3He} \approx 1.2$. But with direct conversion added both fuel cycles require Q values of only about 0.2. As we get beyond the break-even point and into the net power producing area, the effect of Q on estimated system costs is equally striking.

While preliminary, these engineering studies on direct conversion at LRL indicate that D^3He mirror reactor systems have the potential of both high efficiency and competitive costs at values of Q above 0.6.

In the section of the session on thermodynamic cycles, I had mentioned in my opening comments that it is my opinion that we must move off the 40% efficiency plateau provided by steam cycles and move upward to topping cycles in which efficiencies of perhaps 60% are realizable. I think that most people generally agreed. By topping cycles, I mean those that include MHD and gas or vapor turbines that were discussed by Petrick[5] and Hoffman and Förster[6] and Biancardi.[7] A two phase flow liquid metal MHD power system is being developed at ANL which can be coupled to a fusion reactor. The system uses lithium as a magnetohydrodynamic working fluid and helium as the thermodynamic working fluid. Probable overall cycle efficiencies for the system lie between 35% at 1200°F to > 50% at 2000°F. The key to achieving the attractive overall cycle performance is the development of an efficient two phase flow MHD generator. Experimental and theoretical studies completed thus far have indicated that the electrical and theoretical studies completed thus far have indicated that the electrical end losses of the generator can be readily controlled by field shaping. The critical phase relative velocity problem is now under study. If the variation of the phase velocity ratio (V_g/V_L) of the two phase mixture expanding through the generator is small:

$$\left(\frac{V_g}{V_L}\right)_{\text{exit}} < 1.2$$

then this attractive performance can be realized. Initial generator experimental results support this position. Either d.c. or a.c. power can be produced directly through the use of a Faraday or induction generator. The energy conversion system is completely compatible with the materials - liquid metal technology that will need to be developed for the fusion reactor blanket.

The use of plasma MHD generators in Brayton topping cycles combined with modern steam bottoming plans was discussed briefly by Hoffman. While he pointed out that these generators have great performance potential, they require very high stagnation gas temperatures on the order of 2000°K (3140°F) or higher. However, this requires very advanced, high-temperature blankets for steady-state reactors using the D-T cycle. Consequently, the use of plasma MHD generators with steady fusion reactors is probably far in the future. However, there is some possibility of using these with pulsed fusion concepts.

To get to the higher efficiencies that we desire, higher operating temperatures are clearly necessary and it is interesting to note that in the energy conversion loop it is the blanket and its operating temperature which is the limiting component and not the external units. MHD systems must and gas turbines can be run at significantly higher temperatures and we must therefore work on blanket temperatures and provide a means for increasing blanket operating limits. As Biancardi states, although a closed-cycle gas turbine system could utilize the high-temperature energy and provide a net station efficiency above 50%, the maximum efficiency advantage of the future high-temperature capabilities of fusion reactors may be obtained by using a binary system with a metal-vapor cycle as the topping system. The metal vapor selected for the topping cycle would operate between a maximum temperature determined by the available material capabilities and a minimum temperature suitable for heat rejection into a conventional steam system. The steam system could, for instance, operate at a maximum turbine inlet temperature of 950 to 1050°F and have one stage of reheat and several stages of feed-water heating. The net station efficiency obtainable from the potassium cycle varies from 20 to 32% and results in a binary system efficiency of approximately 60% if the plant were sized for an output of approximately 250 MWe.

This need to upgrade blanket temperatures suggests to me that the use of vanadium, which has been discussed at some length at this meeting, should be viewed with some skepticism because its useful operating temperature is around 1000°K and in operating at 1000°K, considered in terms of thermodynamic efficiencies, we would be back to the 40% level under which we currently operate and thus we have really gotten nowhere. Structurally, vanadium is not significantly superior to stainless steel. Now molybdenum on the other hand has a tendency to be dismissed for consideration as a blanket material by s me because of its currently difficult problem of fabrication but it can operate at temperatures that are quite high, say 1450°K which does provide potential for realizing higher cycle efficiencies. So I do not think it sufficient to choose vanadium on one basis such as neutronic benefits without first considering its other limitations. Nor do I think that it is sufficient to dismiss molybdenum because it is currently difficult to fabricate. I strongly believe our general attitude and philosophy should be one of looking forward to advances in technology for our design thoughts and not looking backward to using technology that existed 20 years ago. If in our design considerations, we do things that may seem to be a little far out, so be it; after all the whole concept of fusion is one involving advanced and imaginative thinking anyway and we may as well keep the system elements consistent.

As we get more and more involved in complex studies of reactor systems, it is probably advisable that some standard of comparison be devised and adhered to. Paul Persiani made a plea for standardization of the parameter Q and for standardization of another parameter, ε, which he defined as equal to the fractional recirculated power. He states that a survey of power-producing fusion reactor studies seem to indicate that for some reactor systems the optimum design may require a complex of multiple energy conversion subsystems and/or several plasma containment, heating, and injection devices. To establish a basis for an intercomparison of these complex power systems, it seems necessary to generalize and standardize the definitions of the two important power balance parameters mentioned.

I believe that a standard of comparison is always a good idea and I feel that everyone else would agree. The only reservation I would make to Persiani would be that the standard be carefully considered before adoption.

Bob Mills discussed what he terms a catalyzed deuterium reactor.[8] In this system a reaction is made to take place by injecting or reinjecting tritium and ^{3}He in such a way that the rates of reaction are balanced. No tritium breeding is necessary in this blanket. The system that Mills discussed because it is a D-D system must be signigicantly larger physically than a D-T cycle for a given power output but despite this increased size, costs could be attractive since the materials that are used in the blanket and throughout the system are generally of the conventional stainless steel variety. A thermodynamic conversion cycle for this catalyzed D-D reactor was discussed by Citrolo.[9] This steam cycle and boiler design of Citrolo's serves to illustrate the point that I made previously about design philosophy. I think that this concept can be construed as a pilot plant or test plant because it uses currently available material and conventional construction. This is an interesting idea because it demonstrates that a fusion power plant, although not an optimum one, might be designed for the most part in a conventional way. For Citrolo's initial study, steam alone was evaluated as the working fluid. Since the reaction is basically D-D, the design problems of liquid metal usage and tritium production and recovery were avoided. Other advantages are cited. The principal value of this study is that it illustrates the coupling of a fusion heat source to a highly conventional steam cycle.

In an area in which interest is increasing, John Russell discussed another possible approach to fusion, an inertially confined reactor. This inertially confined reaction is a class of reactions in which ignition of a fuel pellet is achieved by highly focused energy sources such as a laser or an electron beam. Ignition of the fuel pellet is followed by a burn and the reaction is sustained until the rapid expansion terminates the reaction. Thus, the only forces holding the reacting products together are inertial ones. The energy release is in the form of an explosion so Russell has traded the magnetic confinement problem for a problem in hydrodynamics.

For these inertial confinement systems, a preliminary study was made based on a plant using a graphite dust suspended in helium which is used as a coolant. The graphite dust absorbs the neutrons and the shock from a 2×10^{11} joule fusion burst. Repetition rate or time between burst was one minute.

The motivation for this study lies in the possibility of circumventing the radiation damage problem inherent in magnetically confined fusion approaches. In particular the explosions can occur in a large vat of liquid (or fluidized bed) such that the neutrons are absorbed in the liquid rather than the vessel wall. Additional advantages include elimination of the magnetic field requirements as well as opening the possibility of smaller economical systems. The penalty paid is the complication of shock containment.

In the last paper George Hopkins reviewed helium cooled lithium moderated blankets which have some distinct advantage over flowing lithium cooled blankets because of the chemical inertness of helium and the lack of pumping problems. In my opinion, helium cooled blankets have come on a little bit stronger since Culham because we now have more definitive data about pressure drops with lithium particularly in toroidal systems. Helium is attractive because some of the pressure drops that one may experience in machines like Tokamaks may not be allowable if lithium is used as a flowing coolant because flow conforming closely to field lines does not seem possible.

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9. J. C. Citrolo, "Steam Conditions in a Catalyzed D Reactor".

DISCUSSION

H. LAQUER (LASL): You mentioned a possible interest of the utility companies in supporting fusion research. Would you care to comment some more about this possibility?

WERNER: I see a volunteer in the back, named Rosengarten, who would like to answer this question.

W. ROSENGARTEN (Philadelphia Electric): First, I'd like to clarify one point a little bit. Dick indicated the utility industry had not supported basic research or fusion to any degree in the past. Well, we haven't done it to any large financial extent but we have always supported basic research to some degree. For social and economic reasons, we have not been able to spend large amounts of money. At the present time, I think the industry is putting something of the order of a half a million dollars into fusion, so we are doing something. It's hoped that the changing social and political situation will permit the utility companies to spend more money for research. Something which I think we all feel we should be doing. At the present time, the Electrical Research Council, which is an organization of both public and privately owned utilities, is attempting to develop a program which will permit them to spend something of the order of one or perhaps two percent of the gross revenues of the industry on research. Now this is in line, though perhaps somewhat less than most of the manufacturing industry spends on research, but I think it's in line with the kind of thing which we ought to be doing.

A. FRAAS (ORNL): Siegfried Förster asked me to make a comment for him. I think I can make it as he asked me to make it just before he left to catch his plane. He said that he feels strongly that, while my concern over the losses in gas turbine systems will result in a poorer thermal efficiency than what he had estimated, he feels a far greater concern that in these direct conversion systems the accumulated losses in each of these various components in the direct conversion system may lead to a vastly lower thermal efficiency than has been estimated in many cases.

WERNER: Our analysis looks into all elements of the system of direct conversion and looks at these elements in terms of their individual efficiencies and in terms of overall efficiencies. We do therefore look at a total system. I think J. D. Lee pointed out that we consider at least 10 separate elements in doing our analysis on direct conversion and its total plant efficiencies. Therefore, it is not just the direct conversion itself that we consider. We look at injectors, we look at accelerators, we look at all elements and all thermal aspects are calculated. To my knowledge no element is left out. There may be some naivetes in some of our economic calculations but our efficiency calculations are valid and applicable.

G. GRAVES (LASL): I'd like to open a slightly different set of problems for discussion. This pertains to the shielding of these systems, which I think has been neglected a bit at this meeting. I'd like to say, before I sound like Roger Hancox's pessimism, that I'm also an exponent of fusion but some of the things that I have to say may not sound that way. For instance, I had found myself getting progressively more favorably inclined toward this direct conversion system. However, last night as I was thinking about the problem of shielding this system, I became concerned. You must remember you have here an aperture which is of the order of one-half meter by one meter, I think this was the information I got from Mr. Moir earlier. This feeds a semicircular system then which is really a very large duct, the order of 80 to 100 meters long, whose minimum thickness and width is about 1 meter and whose maximum is several meters. This is a system into which neutrons can flow through the open end of the mirror. We have spoken of the blanket attenuation problem at this meeting several times, and we have been speaking as if this were a simple lateral system where we could achieve some number like 10^6 attenuation and be in good shape. I must remind you that we are dealing with two kinds of problems at least. One is the general question of protecting magnets where people have spoken of something like 10^6 attenuation and the other is a question of bioligical protection and general radiation around one of these systems. Now if you have an area which you attenuate to 10^6, but an adjacent area which you attenuate not at all, then the bulk of the radiation loss is determined by that hole. For instance, in an area the size of a football field attenuated by 10^6, one square foot will give you twenty times the radiation than emerges behind the rest of that attenuated area. Moreover, when you are concerned about biological shielding, you are talking about 10^8 or 10^9 attenuation. So over this entire very large semicircular area you would need massive shielding against the emergent neutron flux. I think that would be true in the Tokamak design, too, where you have ducts for injection of your plasma and you have a substantial duct with a substantial wall area for the extraction of the plasma. You can't expect the wall loading on the ducts to be very much down from the wall loading inside the vessel. These problems are really problems of streaming. They are going to transfer radiation damage to new areas, problems of secondary radiations calculated from corners are very very difficult. The same thing will be true to divertors. I would even speculate,for the person who lamented the fact that the shield thicknesses seem to be going up, that if anything they are likely to go up still more. We really should keep in mind that we have a very complicated system problem and,frankly, I haven't yet seen a very good design here which looks like it's compatible with shielding needs.

S. BLOW (Harwell): I'd like to ask Dick about this problem of increasing the temperature at which you operate. It is true that you are limited in the efficiency of the thermodynamic cycle by the temperature at which you can operate the coolant fluid, but at the same time you've got a trade-off in rapidly developing materials problems associated with the increasing temperature, e.g., corrosion, etc. Therefore, although you may have increased your thermodynamic efficiency from 40 to 60%, when you consider the increased capital cost associated with developing these specialized materials, have you gotten anywhere in producing cheaper electricity?

WERNER: Well, I don't know that there is a definitive answer to this. A tendency in the United States and elsewhere is to concern outselves with waste heat and ask the question as to whether we shouldn't charge a penalty for waste heat generated and I think the answer is "Yes". We're going to have to pay for the waste heat. If we can reduce the waste heat, then this cost savings, in a sense, defrays the cost of the refractory metals which we may have to use. My feeling about increased efficiency is based on the President's Clean Energy Bill where he directs the breeder people to look towards more highly efficient systems for the reasons I'm citing about waste heat, and I think if they are going to do it, we also should. Additionally, we're talking of systems that are going to be fabricated, hopefully, fifteen years hence at the earliest and certainly in fifteen years' time we can do things to defray the cost of these metals. Roger Hancox made a point about the costs of superconductors. You might get superconductors costs down to the cost of the base materials and that's 2 orders of magnitude lower than what you buy it for now. You might be able to do the same thing with refractory metals for structure. You should be a little bit wary of using present day costs for future programs. It's very misleading.

G. MILEY (U. Ill): I'm very enthusiastic about the catalyzed reactor concept. You described it as a pilot plant and I guess, as far as the blanket is concerned, it is. However, it looks like it has tossed the problems back to the plasma where you have a 35 keV plasma on your hand. I'm not sure it's a good pilot plant from that point of view. It'd be nice if we did have a pilot plant that was truly one in terms of the blanket and in terms of the plasma.

WERNER: It's a pilot plant only in the sense that it uses conventional materials and eliminates breeding problems which get rid of two difficult design problems. I meant it in that context.

EPILOGUE

PERSPECTIVES ON FUSION REACTOR DESIGNS

SKETCHES

by

David J. Rose

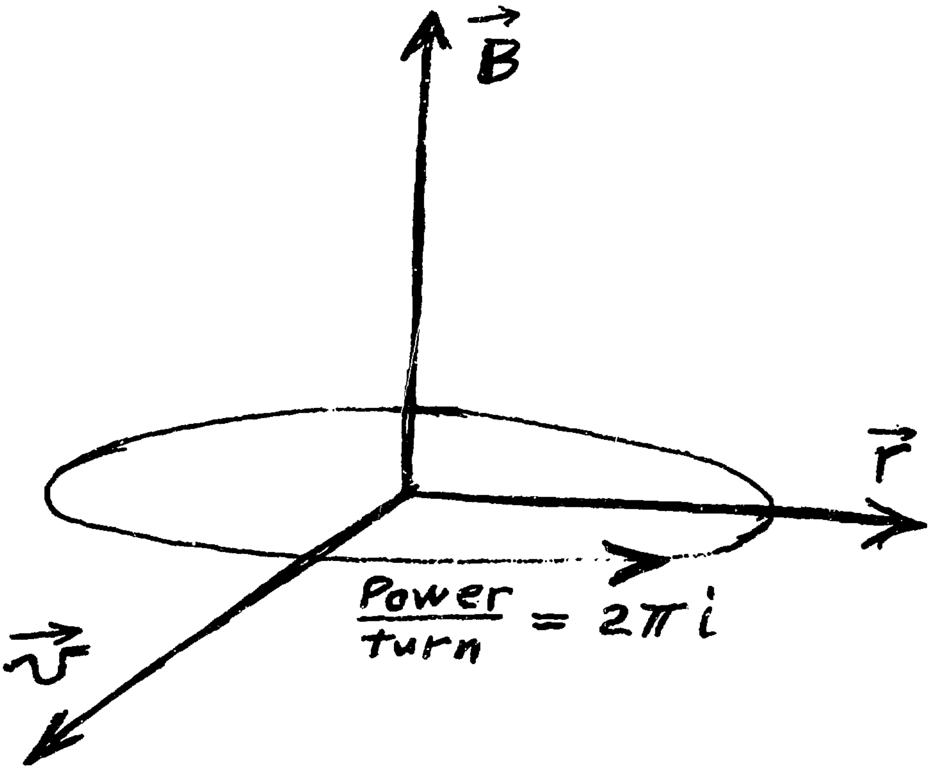

Fig. 1. Theorist's View of the Fusion Reactor.

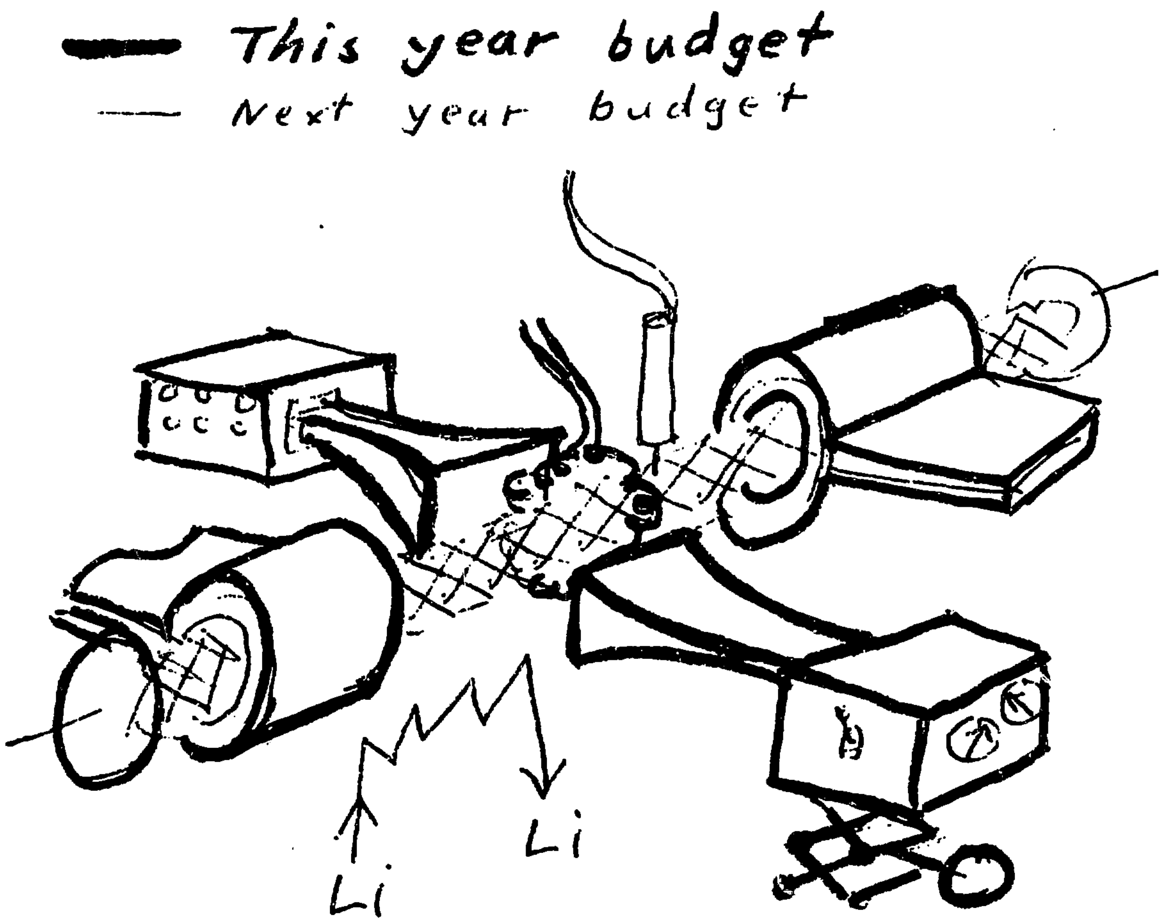

Fig. 2. Experimentalist's View of the Fusion Reactor.

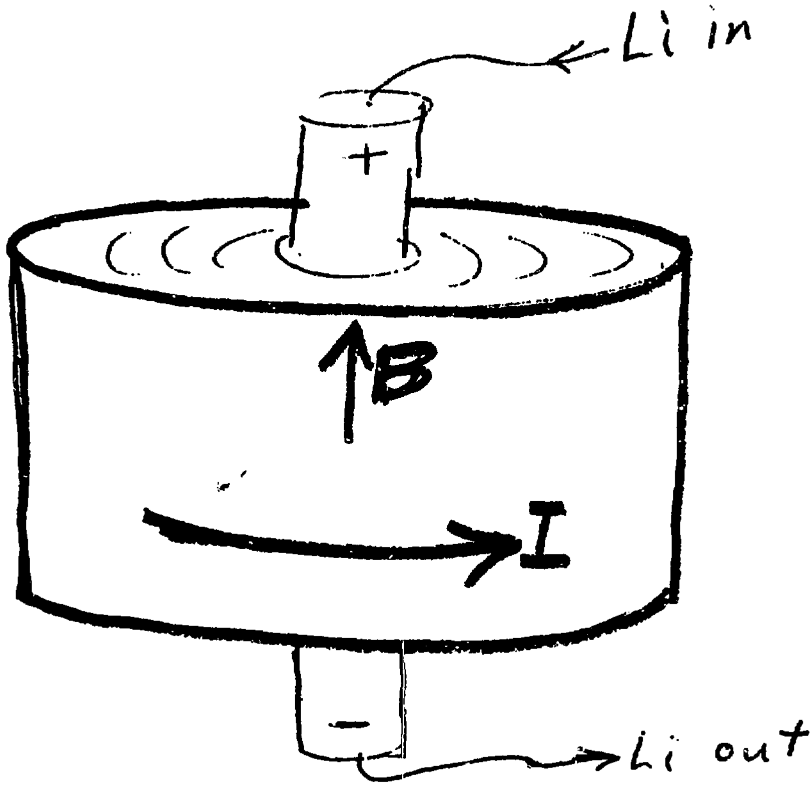

Fig. 3. Magnetic Field Designer's View of the Fusion Reactor.

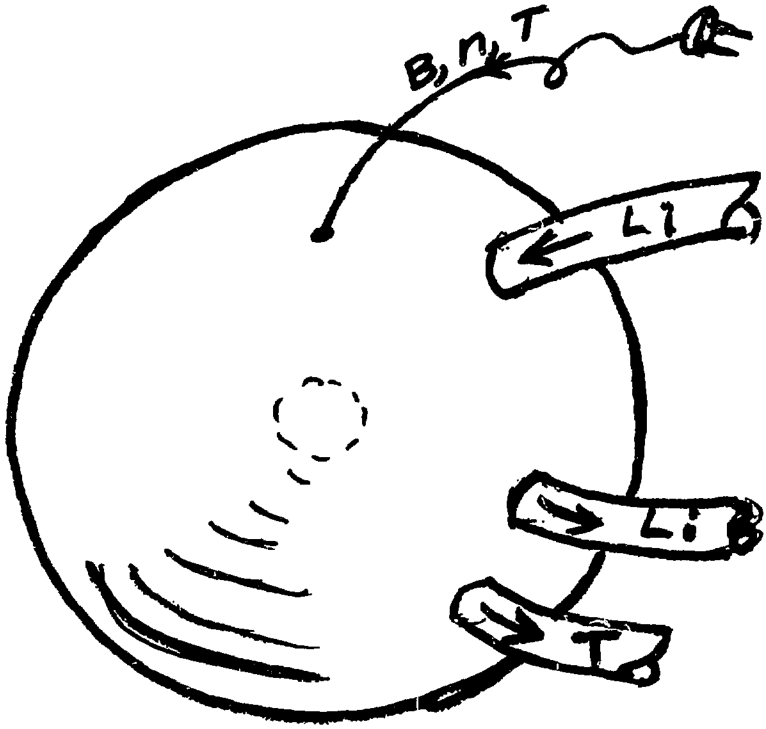

Fig. 4. Tritium Breeder's View of the Fusion Reactor.

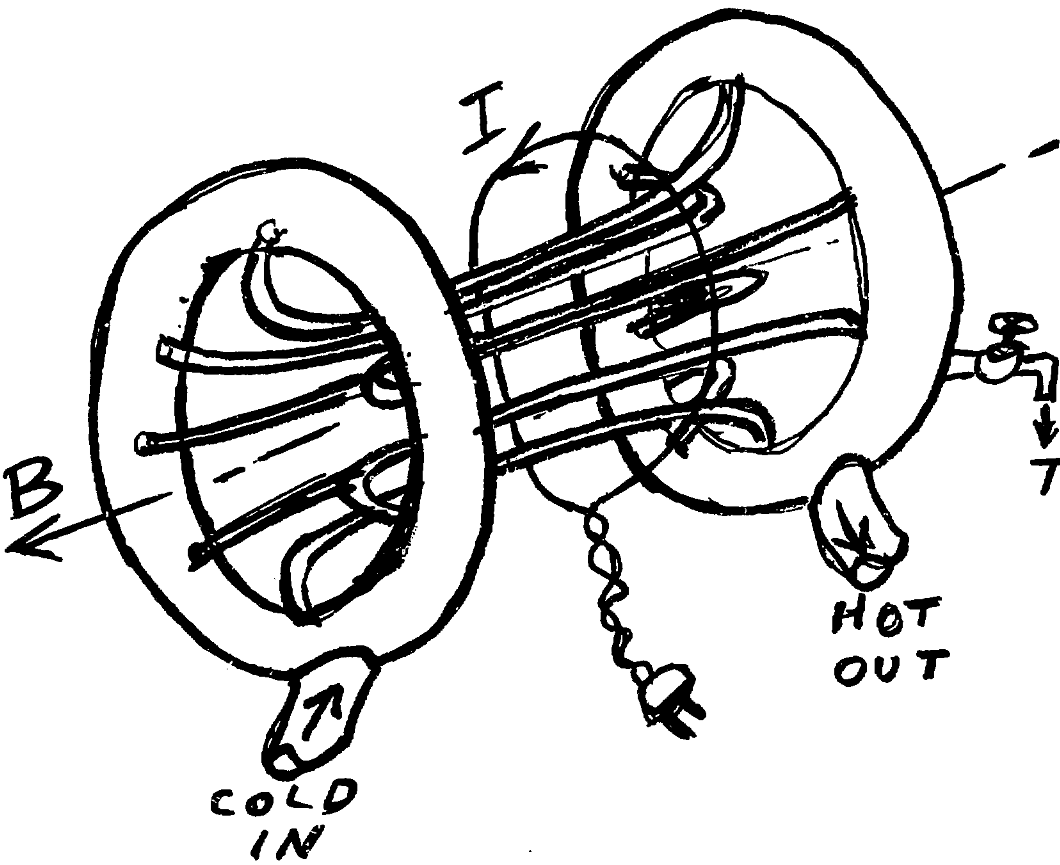

Fig. 5. Heat Exchange Designer's View of the Fusion Reactor.

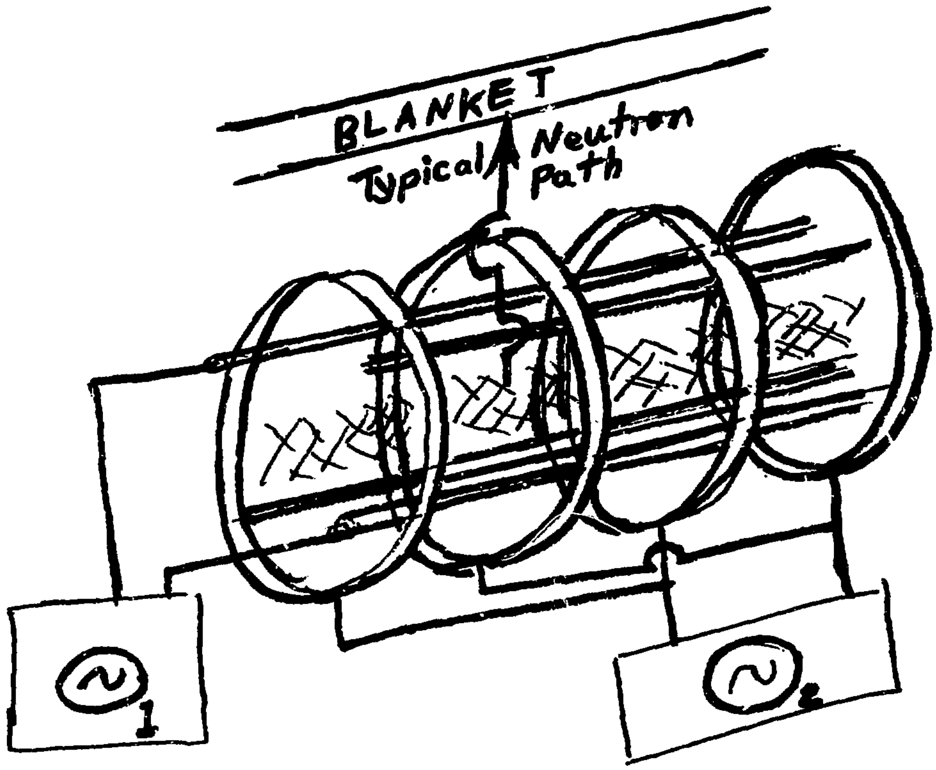

Fig. 6. R.F. Heater's View of the Fusion Reactor.

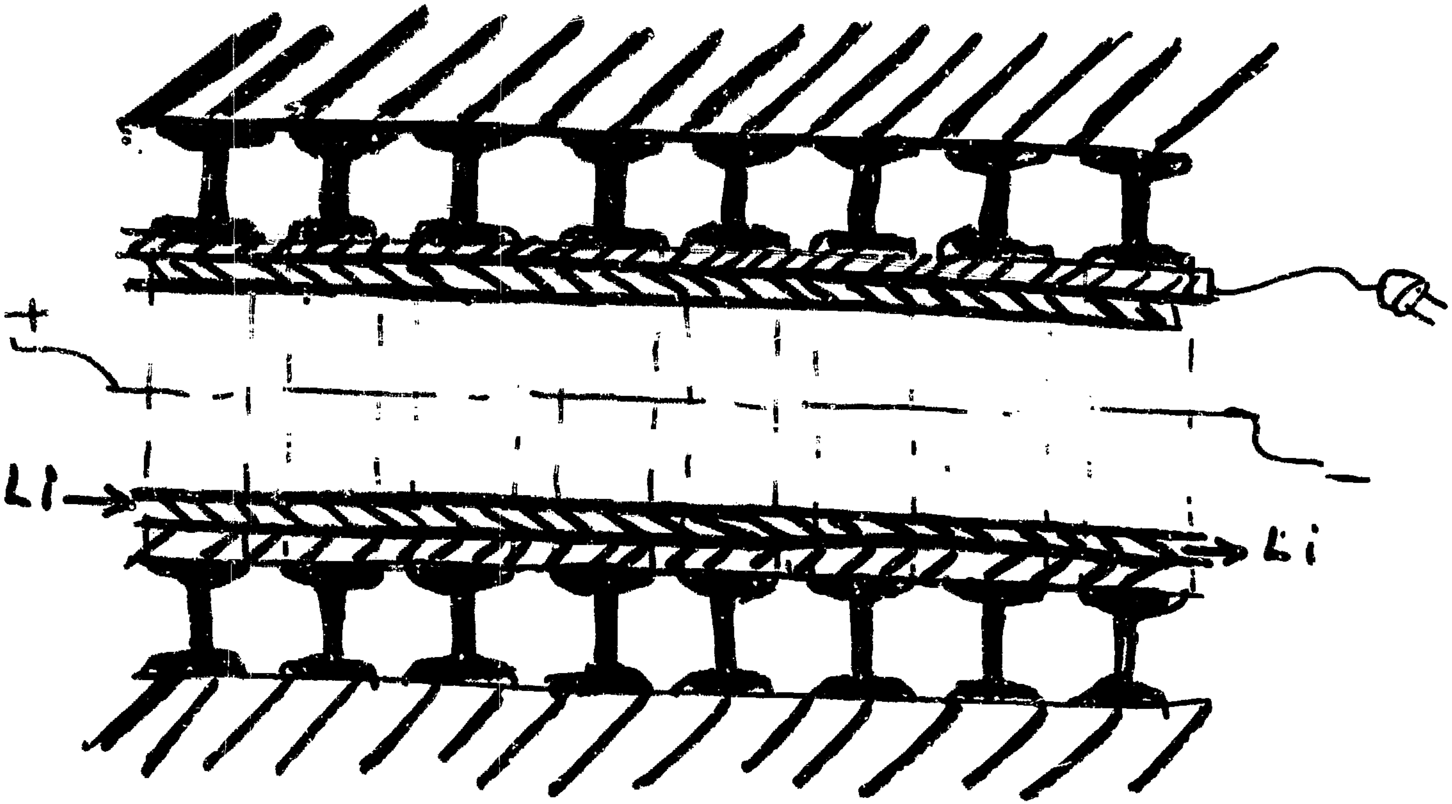

Fig. 7. Structural Engineer's View of the Fusion Reactor.

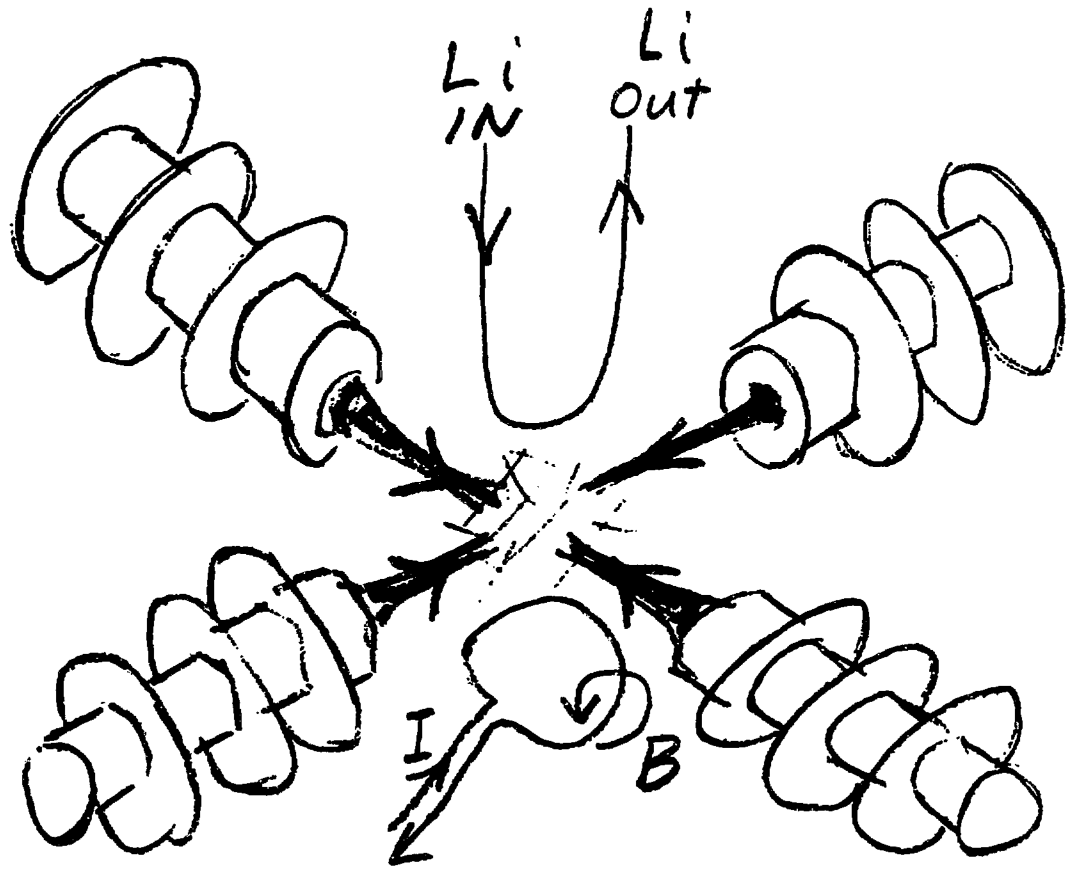

Fig. 8. Ion Injector's View of the Fusion Reactor.

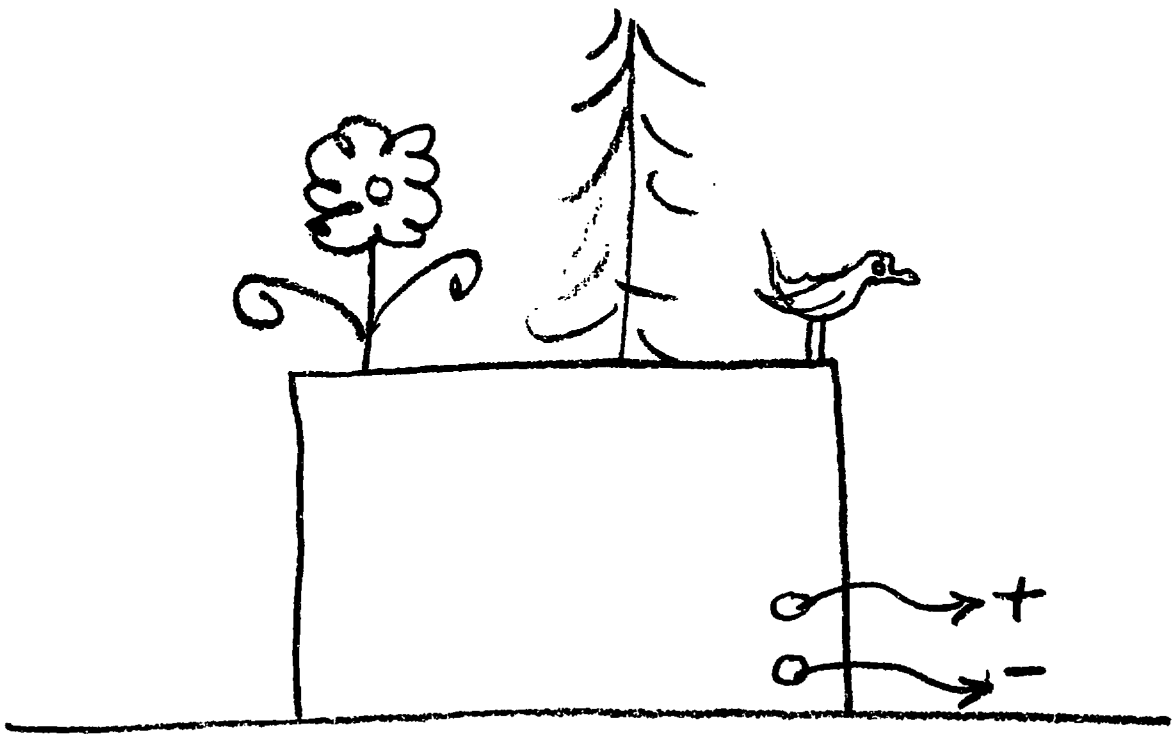

Fig. 9. Environmentalist's View of the Fusion Reactor.

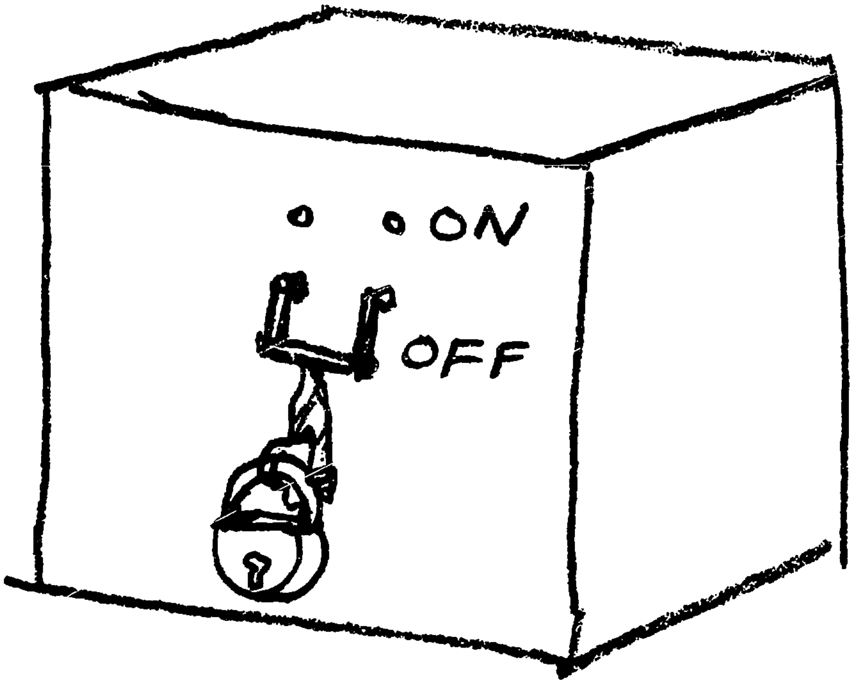

Fig. 10. Safety Engineer's View of the Fusion Reactor.

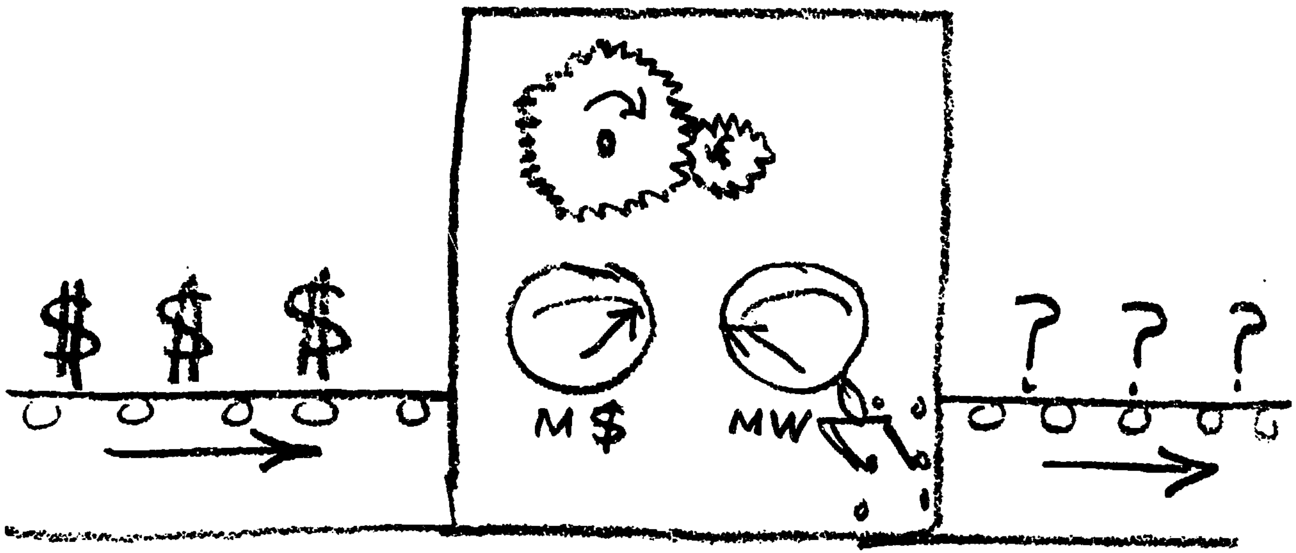

Fig. 11. Office of Management and Budget's View of the Fusion Reactor.

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