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Abstract

A review of the effects of DT neutron damage in fusion reactor structural components and magnets is given. It is shown that the displacement rates in fusion systems are about equal or less than present day fission reactor facilities but gas production rates are one to two orders of magnitude higher. The effect of neutrons on mechanical properties such as yield strength, ductility, and creep is briefly reviewed and it is noted that very little, if any, direct data exists for 14 MeV neutron irradiated metals. Resistivity increases in magnet stabilizing metals and the reduction in critical properties of superconductors are shown to be minimal and subject to reduction by several design options. The level and decay of induced radioactivity is explored for various metals and levels of >1 curie per thermal watt are easily achieved at shutdown. The decay of afterheat, from shutdown values of 0.5 to 1% of the operating power is shown to be a function of materials choice and V and Al appear to offer some distinct advantages. Finally, a method to ameliorate the radiation damage problems is discussed.

1. Introduction

The use of a DT fuel cycle in fusion reactors means that a great many components of metallic, ionic and covalent bonding will be bombarded with energetic neutrons. The response of these materials to irradiation depends on many factors such as temperature, neutron flux and fluence, neutron energy, purity and stress state to name just a few. The purpose of this article is to provide a broad view of where we stand in defining the problems to be faced in fusion power reactors. The anticipated CTR materials and their environments will be outlined first and then the primary responses (e.g. displacement and transmutation rates) of these materials are explored. We will not concern ourselves here with charged particle leakage from the plasma and bombardment of the first wall as that is covered by other reviews in this volume.[1,2] Next, the various secondary responses (e.g. physical and mechanical property changes) will be examined and related to CTR application. Finally, some possible solutions to the radiation damage problem will be briefly discussed.

2. Materials Environment in CTRs

The basic feature of DT fusion reactors have been described elsewhere[3-12] and will not be repeated here. Table I attempts to 1) summarize the various

functions materials must satisfy, 2) list some of the materials which have been proposed, 3) give anticipated temperature ranges for material operation and 4) give some idea of the potential neutron fluxes to the various components. All fluxes are quoted for an average first wall neutron loading of 1 MW/m^2 which is ~4.4 x $10^{13}n/cm^2/sec$ of 14 MeV neutrons. Most current designs tend to use wall loading values within a factor of two or three of this number.

The neutron energy spectra in fusion reactors varies substantially from point to point and Figure 1 shows what a typical neutron spectrum looks like at the first wall, at the blanket-shield interface, and at the outside edge of a shield. It is noted that not only does the absolute neutron flux drop as one progresses into the blanket and shield, but the energy spectrum is considerably more thermalized.

The most critical components of a fusion reactor are those which must maintain vacuum tightness for the plasma and provide coolant passages in the face of fluctuating temperatures, stresses, pressure surges, corrosion and a general degradation of mechanical properties due to neutron irradiation. Various materials have been proposed as first walls in fusion systems ranging from austenitic (non-magnetic) stainless steel to high temperature refractory metals. Some have even proposed special aluminum alloys[13] and very high temperature covalent systems like SiC and graphite.[14] Such unprotected first walls face 14 MeV neutron fluences of ~1 x $10^{21}n/cm^2$/year and total neutron fluences (E > 1 eV) of ~6 x $10^{21}n/cm^2$/year.

Behind the first walls are usually lithium containing materials and in some systems, beryllium containing materials (especially for those which rely on solid lithium compounds). In the Li and Be, the direct displacement damage due to neutrons is minimal compared to the large amounts of gases generated. For example, every $Li^6(n,T)He^6$ and $Li^7(n,n'T)He^4$ reaction generates two gas atoms. Presumably, the tritium diffuses or is extracted out of the lithium containing material, while the insoluble helium atoms can collect into bubbles and cause dimensional instabilities. The same types of problems are associated with the use of Be for neutron multiplication via the $Be^9(n,2n)2He^4$ reaction.

Most reactor systems find it more economical to place neutron reflectors behind the breeding zones to both reduce the neutron leakage and to moderate the spectrum in order to take advantage of the high thermal cross section of lithium-6. High Z materials work well as reflectors in the 10-14 MeV region because of their high inelastic cross sections while carbon is particularly effective in thermalizing neutrons of energy <10 MeV. However, in the process of absorbing the kinetic energy of the neutrons considerable damage can take place in these reflectors (e.g. total neutron fluences of ~4 x $10^{22}n/cm^2$ over a 30 year plant like are incurred in graphite reflectors[5]).

After the primary neutron reaction functions have been satisfied (namely extracting the kinetic energy and breeding tritium) the remaining neutrons must be absorbed before they can do harm to sensitive reactor equipment like magnets or lasers and the operating personnel must be protected. Typical reactor designs[3-5] allow 1% less of their neutrons to escape <u>into</u> the shield region and for safety and economics reasons no more than 10^{-4}% are usually allowed to leak <u>from</u> the shield. The shields are commonly made of a mixture of moderators, neutron absorbers, and gamma ray attenuators. Suggested combinations of materials include water, kerosene, B_4C, Pb, and stainless steel. Since both the neutron energy and fluxes are usually low in this region ($<10^{20}cm^{-2}yr^{-1}$) the shield can operate at low temperatures. Relatively minor radiation damage problems are expected to be encountered in this region.

Some reactor types have special radiation problems. For example, pulsed systems like the theta pinch require electrical insulation to maintain 100 kV/cm[7] voltage gradients in the first wall and resistivities of $\geq 10^6$ ohm-cm.

Laser systems require a series of mirrors and windows to transmit the laser light to the cavity.[9] The neutron fluxes to the windows are usually as high as those to the first wall while the mirrors can be placed throughout the blanket and thereby minimize the neutron damage. One peculiar feature of laser systems is that the neutron fluxes are very high for a microsecond or so and then no neutrons are produced for as long as 100 ms to 1 second between shots. Since the average wall loading should be at the 1 MW/m^2 level for economic reasons, this means that instantaneous 14 MeV neutron fluxes of 10^{19}-$10^{20} cm^{-2} sec^{-1}$ might be incurred. Very little is known about the effect of such high instantaneous fluxes to laser windows or even structure materials.

Magnetic confinement schemes have their own particular problems too. Thermal insulation (typically using organic materials) is required for superconducting toroidal and poloidal field coils. However, the susceptibility of hydrocarbons to radiation damage is well known[15] and one must insure that 10^{15}-$10^{16} n/cm^2$/year and high gamma loads (5-10 Megarad per year) do not cause the insulator to lose its integrity. Another problem peculiar to magnetic systems is the retention of radiation damage at low temperatures (~4°K). Relatively little annealing takes place and even though the total fluence is quite low, the damage can accumulate until it produces significant physical property changes.

It will not be possible in this paper to survey all of the problem areas indicated in Table I so we will confine our attention to only those metallic systems in the first wall and blanket in addition to highlighting a few problems of superconductors. This is not to say that the other problems are unimportant, but it is generally recognized that the safe and efficient operation of all fusion reactors relies on the solution of problems faced by the first wall. The reader is referred to other reviews for the problems in non-metallic materials.[15-17]

3. Primary Material Responses to DT Neutron Irradiation

There are 2 major primary material responses to neutron irradiation.

A) Displacement of atoms from equilibrium position which in turn leads to

1. free vacancies and interstitials in the matrix.
2. sputtering of atoms from free surfaces.

B) Transmutation reactions which can lead to

1. production of gas atoms (H,He) inside the material.
2. a changing chemical composition consisting of both stable and radioactive elements.

Let us consider the displacement process first.

3.1 Displacement Damage

The mechanisms and magnitudes of neutron displacement rates in potential CTR materials has been recently summarized[18] and Table II lists some typical values of interest. The values have been normalized to 1 MW/m^2 and the unit of damage is the number times each atom is theoretically displaced during exposure, or the displacements per atom (dpa). These displacements can occur from reactions such as $(n,n'\gamma)$, $(n,2n)$, (n,α), (n,γ), (n,β^-) etc. Hence, dpa values of >1 are possible even though recombination, especially at higher temperatures, means only a small fraction of the original displacements (typically less than 1%) eventually survive.

A comparison of displacement rates in typical fusion and fission reactors (such as in EBR-II in the USA) is also included in Table II and displayed in Fig. 2a. It can be seen that in terms of absolute instantaneous displacement rates, fission test reactors currently can produce damage rates of 3-4 higher than anticipated for steady state CTR's and 1-5 orders of magnitude less than in pulsed systems. On a time averaged basis the dpa rates in pulsed systems are the same as for steady state devices. However, there are two other fundamental differences between fission and fusion systems. First, the higher neutron energy spectrum in a CTR (see Figure 1) will induce more deleterious transmutation reactions and second, one would like the components of a CTR to last the plant lifetime (~30 years) as opposed to a one to two years for fission fuel cladding. Therefore, the total dpa level for CTR materials could be greater than for a fission reactor component during its shorter lifetime.

Neutron sputtering, especially by 14 MeV neutrons, has been the subject of considerable controversy for several years. Sigmund[19] developed a theory for amorphous materials which predicts the sputtering coefficient for 14 MeV neutrons on Au to be ~4 x 10^{-5} atoms/neutron. Measurement by Keller[20] initially indicated values of S = 0.5 on Au but later experiments on an improved experimental apparatus yield S $\leq$ 6 x 10^{-4}.[21] Recent experiments by Behrisch et al.[22] yielded values of S = 2-3 x 10^{-4} and Robinson reports S < 10^{-4}.[23] On the other hand, Kaminsky has reported S = 0.06 to 0.22 for 14 MeV neutrons on Nb.[24] The major consequence of neutron sputtering is wall erosion (on both sides of a component). It can be shown that for S = 10^{-4} and 1 MW/m^2 the wall erosion rate is on the order of 10^{-4}mm/year, a completely negligible quantity. However, if the sputtering coefficient increases to 0.1, then ~0.1 mm/year could be eroded away. Since most first walls are 5-10 mm thick, such a high value could seriously impair the wall strength if the component is to last the entire reactor life. Furthermore, it is possible that the sputtered material, which is highly radioactive, might cause considerable contamination in the reactor coolant or vacuum system.

3.2 Transmutations

In contrast to the displacement situation, the production of hydrogen and helium in fusion systems is far more serious than in fission reactors. Table II shows that as much as one to two orders of magnitude more helium and hydrogen is produced in CTR structural materials than in fast fission reactors per year of exposure. Niobium produced the lowest absolute amount of helium and hydrogen in CTR environments while sintered aluminum product (SAP) (a fine dispersion of 5-10 w/o Al_2O_3 in Al) shows the highest gas production rates.

In general, the refractory metals have helium production rates of 20-60 atomic ppm per year per MW/m^2 wall loading. Iron or nickel based austentic alloys have ~200 appm/yr/MW/m^2 helium production rates while SAP has a rate of 400 appm/yr/MW/m^2. Very few if any materials have ever been tested with helium contents equivalent to even one year of CTR first wall exposure, let alone that corresponding to a 30 year value.

The helium production in carbon is quite high (2700 appm/yr) because of the large (n,3α) reaction. In beryllium, He is produced via the $Be^9(n,2n)2He^4$ reaction and in order to get a breeding in a typical solid breeder system, several thousand atomic ppm per year are produced.[5] The case for Li is especially critical and typically tens of thousands of atomic ppm per year is produced to attain breeding ratios of more than 1.0.

Another important point to note is that the ratio of He gas production in fusion systems is 1-3 orders of magnitude higher than in fission reactors (Figure 2b). Hence, properties which are sensitive to synergistic effects of

gas atoms and point defects cannot be satisfactorly duplicated in current fission reactors.

Non-gaseous transmutation reactions can also be quite serious. Recent studies[25-27] have shown that niobium is particularly sensitive to this problem. Table III summarizes some key transmutation products in CTR materials. It can be seen that metals such as Al, or V show only slight variations in composition while in 316 SS large increases in the Mn content may significantly effect the mechanical properties. Fusion neutron irradiation of Nb produces large amounts of Zr which could exceed the 10 at % solubility limit[28] if exposed to >20 years of 3 MW/m^2 loading. Second phase precipitation can change not only the reaction rates with point defects, it can also cause volume changes as we shall see later.

4. Secondary Materials Responses to DT Neutron Irradiation

The mere production of interstitials, vacancies and transmutations is just the beginning of the radiation damage problem. What happens to these defects after they are produced at either low or high temperatures in a fusion reactor? In the next few sections, we will try to highlight the problems posed by the accumulation and/or migration of these point defects into clustered arrays.

4.1 Dimensional Changes

4.1.1 Voids in Metals

This topic has been extensively treated elsewhere[29-31] and will not be reviewed here. It has been known since 1967 that irradiation of metals and alloys at 30-50% of their absolute melting point induces vacancies to migrate and precipitate into voids causing the metal to swell. The fluence threshold for swelling in metals has been found to be as low as 10^{19}n/cm^2 for pure Al[32] and as high as 10^{22}n/cm^2 for complex alloys like stainless steel.[29] Only three metallic systems have shown no swelling up to high fluences; Au and Ti to 3 x 10^{21}n/cm^2[33] and V-20Ti to 3 x 10^{22}n/cm^2.[34]

Examples of voids in pure vanadium and a Nb-1Zr alloy are shown in Figure 3a and Figure 3b shows how swelling in 304 stainless steel varies with neutron fluence at 480°C.[35] The equivalent dpa values are approximately 4.5 dpa per 10^{22}n/cm^{-2}. Using the numbers in Table II, irradiation equivalent to 4 years of 1 MW/m^2 wall loading may induce 8-10% swelling in 304 SS. Wiffen and Bloom[36] have conducted the only high fluence experiment to test the effect of high helium contents on void swelling in stainless steels. They irradiated 316 SS in a thermal reactor to dpa values of 52 to 121 and helium contents ranging from 1971 to 6090 appm. It was found that these specimens swelled by a factor of 2 or more greater than identical material irradiated to roughly the same fluence in a fast reactor where the helium content is only a few appm. The observed swelling also was not predictable from current swelling models indicating that the simultaneous generation of large amounts of helium and displaced atoms represents a region of little understanding.

No such results are available for the refractory metals but it is expected that helium will also play an important role here as well.

The absolute void swelling values for materials fusion reactors are certainly important but the swelling gradients (due to temperature or neutron flux gradients) could be even more devastating. As we shall see later, the material irradiated in a CTR for a year or more will be unable to stand very much (<1%) strain before fracture. If components swell unevenly, such strains could easily exist, posing severe maintenance and operation problems for the vacuum wall.

Another problem to be faced is the effect of displacement rate on swelling. Recent studies on steel by Johnston et al.[37] (Figure 4) have shown that

irradiation at 10^{-3} dpa/sec cause the swelling curve to be displaced to higher temperatures by 100-200°C when compared to materials irradiated at 10^{-6} dpa/sec. Theoretical treatment of this effect is not particularly satisfying yet but the reduced swelling at a given temperature is probably due to increased recombination at the higher displacement rates. The significance of this work is that while swelling measured in current fission reactors at ~10^{-6} dpa/sec may be applicable to fusion reactors at ~10^{-7} dpa/sec, it will probably not be applicable to the same material irradiated at the same temperature to the same fluence in a laser reactor at 10^{-1}dpa/sec in short pulses. (Figure 6) High damage rate and pulsed irradiation conditions have been simulated by Sprague et al.(38) who find that such an irradiation mode does not satisfactorily duplicate steady state high damage rate data.

4.1.2 Gas Bubble Swelling

It is well known from fission reactor fuels studies that the collection of large amounts of inert gas atom into bubbles causes considerable volume changes in metallic components.(39-40) The volume change depends on total number of gas atoms per unit volume, n_g, the VanderWaal's constant for the gas in the bubbles, b, the diameter, d, the surface energy of the bubble, γ, the temperature T and Boltzmann's constant k as,

$$\frac{\Delta V}{V_o} = n_g\left(\frac{kTd}{4\gamma} + b\right)$$

Hence, for a fixed amount of gas, the swelling will increase as the bubbles coalesce and d increases. Figure 5 illustrates the problem for beryllium. For comparison, note that ~2800 appm He/year/MW/m^2 is produced in Be used as a neutron multiplying medium in a typical fusion reactor utilizing solid breeding material.(5) Depending on the degree of bubble migration and coalescence, one can see that one year of reactor exposure would induce 2-3% swelling if all the bubbles were 100 Å in diameter and ~12% if they are 1000 A in diameter. Ten years of operation would increase these numbers to ~30 and >100% respectively. Such swelling is clearly intolerable and may even force certain Be components to be replaced during the plant lifetime.

4.1.3 Solid Transmutation Product Swelling

The production of some solid transmutation products can also cause large swelling in metals and alloys. This is particularly severe in Nb systems where Zr is the major product. For example, 1 year exposure at 1 MW/m^2 produces 0.18% Zr in Nb and 0.3% swelling. After 30 years this swelling would amount to 8% and the Zr concentration would be within a factor of 2 of the solubility limit. Second phase precipitation in this system is known to have severe embrittling effects.

4.2 Mechanical Properties

The production of point defects along with their subsequent migration and agglomeration into clustered defects can have profound effects on the mechanical properties of metals. Some of the effects are detrimental while some are actually beneficial as we shall see.

4.2.1 Yield Strength

Both point and clustered defects can pin or impede the motion of dislocations thus increasing the yield strength of metals. There is a considerable body of information for such effects on iron and nickel based alloys(41-43) and Figure 6 demonstrates a typical case for 304 stainless steel at 430°C.(44)

The yield strength increases until it reaches 3-5 times its unirradiated value and is relatively constant thereafter. Similar behavior is expected in refractory metals such as Nb, Mo and V.

There is no known mechanical property data on metals or alloys irradiated with 14 MeV neutrons above 0.0001 dpa but one would expect the same general effect on yield strength. The simultaneous high generation rate of gas atom may even accelerate the rate of yield strength increase but such a conclusion is only speculation at this time.

The significance of this yield strength increase is that the CTR blanket material will stay in the elastic regime up to very high stresses. That type of behavior will prevent the normal plastic deformation which occurs at stresses above the unirradiated yield point. However, the ultimate tensile strength does not usually increase as fast as the yield stress and eventually the two almost coincide which leads up to the next problem, embrittlement.

4.2.2 Embrittlement

There are at least five causes of embrittlement in fusion reactor materials.

1. Displaced atoms
2. Helium generation
3. Hydrogen generation
4. Interstitial impurity effects
5. Second phase precipitation

The first type of embrittlement is responsible for the yield strength increasing to near the ulitmate stress level leaving little room for plastic deformation to relieve stresses. Practically, this means that once the uniform stress exceeds a few tenths of a percent, the material will fail in a brittle fashion. This causes catastrophic failure as opposed to the usual plastic deformation in unirradiated metals. Figure 6 shows how fast the uniform elongation preceding failure drops off with displacement damage.(44) For a constant stress, once the uniform elongation limit is exceeded, the material behaves unstably and rapid failure follows. It is expected that the higher energy fusion reactor neutrons will also induce the embrittling behavior shown in Figure 6 and in fact, may accelerate it because the higher energy imparted to the primary displaced atoms will cause larger displacement spikes. Again no known data is available for dpa values above 0.0001 dpa.

The mechanism of helium embrittlement is different from the first mechanism in that the helium tends to form bubbles on precipitates and grain boundaries(45). Since the center of the metallic grains are strengthened due to displacement damage, the solids tend to want to deform at the grain boundaries. The presence of the bubbles reduces the strength of the grain boundaries and reduces their ability to slide over one another. Thus, when some critical stress is exceeded, cracks form at the grain boundaries and cause the metal to fail intergranularly.

Wiffen and Bloom(36) have demonstrated that the presence of large amounts of helium in 316 stainless steel can severely reduce its ductility at high temperatures. The higher the temperature, the more the tendency for bubbles to precipitate and hence the less deformation via grain boundary sliding before fracture. The high helium generation rates of CTR's will certainly aggravate these conditions and such embirttlement will probably place an upper temperature limit on the operation of metallic components.

Hydrogen embrittlement is only applicable to a few metals but its effects are well known inside as well as outside the nuclear community.[46] There are abundant sources of hydrogen in CTR environments from the D,T escaping from the plasma to the tritium bred behind the first wall, and that H,D,T generated by nuclear reaction in the metals. In contrast to helium embrittlement, which occurs mainly at high (>0.5 T_m) temperatures, hydrogen embrittlement is important at low temperatures[47]. This means that one must be careful about the shutdown of the reactor and lowering of the temperature of the metallic components. Subsequent start up may have to be accomplished with severely embrittled metals.

Unirradiated body center cubic metals (such as Mo,W,Ta,Nb,V) display a phenomena which has a characteristic ductile to brittle transition temperatures (DBTT). Below the DBTT, the metal fails in a brittle fashion while above the DBTT it fails in a ductile manner. This DBTT is sensitive to interstitial elements such as O,N,C, or H in unirradiated metals and several authors have found that it increases with irradiation exposure.[47] Wiffen[48] has shown that the DBTT in Mo is shifted from 175°K to ~550°C after a neutron exposure of ~10 dpa at 425°C. The significance of this disclosure is that the Mo may behave in a ductile manner at T > 550°C during irradiation but if the temperature is lowered below 550°C, for any reason (e.g. temporary shutdown, scram, etc.) the Mo will fail in a brittle fashion. (Figure 7) Similar behavior is found in Nb and V and it is expected that 14 MeV neutrons will produce the same effect. There is no known data on any bcc metal irradiated at high temperature with 14 MeV neutrons and tested for DBTT shift.

Finally, the build up of transmutation products can actually continue past the point at which they exceed the solubility limit in the metal. A perfect example of this is Nb in which Zr is the major transmutation product from 14 MeV neutrons. The solubility limit of Zr is Nb is ~10% which means that after 55 MW-yr/m^2 of exposure, one would start to get second phase precipitates that will cause severe embrittlement.

In summary, there are several mechanisms which will cause CTR materials to fail in a brittle manner, and some of these may actually show some synergism. This particular problem has already been identified as the determining factor in the useful life of a CTR first wall.[49] Values of ~2 years appear to be the limit for austenitic stainless steel.

4.2.3 Creep

The phenomena of creep in metals at high temperatures has been known for some time[50] and in fact this property determines the maximum temperature at which most metallic components can operate under high stress.[51] It has been amply demonstrated at high temperature that stresses on metals well below the yield stress can cause metals to fail over a long period of time, thus giving rise to a characteristic stress-rupture life. The effect of neutron irradiation on this phenomena is generally to increase the creep rate and lower the stress-rupture life at high temperatures[45] (Figure 8). On the other hand, this increased creep rate can help relieve stresses in metals due to inhomogeneous swelling, thereby ameliorating the strains induced in irradiated metals.

There should be little difference between fission neutron and 14 Mev neutron irradiation effects on creep in metals unless the large helium

concentrations tend to keep dislocations from climbing over obstacles. One feature which could considerably alter the creep rate in CTR metals is the pulsed nature of the damage in theta pinch and laser systems. No experience is available on very high damage rates (10^{-5}-10^{-1}dpa/sec) separated by relatively long annealing periods. Such effects need to be studied before more definitive statements can be made.

4.3 Physical Properties

The main effects in this area have to do with the electrical and magnetic properties of superconducting magnet materials. Unlike the previous analysis, this problem occurs at very low temperatures (~4°K) and therefore represents a completely different regime of discussion.

4.3.1 Electrical Resistance

It is well known that irradiation of metals at low temperatures can cause increases in their electrical resistance.(52) For fusion systems relying on magnetic confinement this is particularly important in the superconductor filament stabilizing materials such as Cu or Al. For example, the resistivity of OFHC copper at 4.2°K is 0.01 micro-ohm and the increase in copper to irradiation has been determined by Horak and Blewitt.(53) Figure 9 shows how this resistivity increase varies with displacement damage and demonstrates that a saturation level is reached at ~10^{-3} dpa. This is roughly 10 times the residual resistivity of OFHC copper and could be 100 times that for higher purity material. A similar situation exists for Al.

The significance of this resistivity increase is that if the stabilizing material is ever called on to perform its function in the magnet (namely to allow current to by-pass a superconducting filament which has gone normal until the element can be cooled below its T_c) then more heat will be generated in the cryoresistive material. This higher heat load represents a burden on helium liquifiers and in fact may reach the point where so much heat is being generated in the irradiated stabilizer that the helium coolant is unable to quench the "normal" condition. Under such circumstances, the entire coil might go normal causing great harm to the magnet. This limit has been examined recently for one fusion reactor design which allows a 5×10^{-5} dpa/yr. damage rate in the toroidal field coils.(4) From Figure 9 and noting that a reasonable limit on resistance increase is ~10% over the unirradiated value, reveals that such a limit would be reached in 2 years.

Fortunately, there are three relatively simple solutions to this problem. The first is to increase the thickness of Cu stabilizer to lower the heat generated by a fixed current. Secondly, a large fraction (80-90%) of the damage can be annealed by raising the temperature to ~300°K.(52) If this is done every two years then the magnets might last 20-30 years. The third solution is to build thicker shields which in turn requires bigger toroidal field coils and hence higher capital costs.

4.3.2 Superconductor Properties

For those fusion systems which rely on magnetic confinement by superconducting magnets there are currently two choices of materials, the NbTi alloy and the compound Nb_3Sn. Each of these respond differently to irradiation with Nb_3Sn being the most sensitive because it depends on atomic ordering for its properties. Any mechanism, such as displacement spikes or the production

of transmutations, which disrupts that order will degrade the critical temperature, current, and magnetic field of Nb_3Sn. NbTi is much less sensitive to irradiation as we shall see.

It is worthwhile to note at the beginning that there is very little experimental data which directly pertains to the irradiation stability of superconductors. First of all, it is very difficult to irradiate and test components at 4°K. Most experimenters irradiate at reactor ambient conditions and then test at 4°K. Secondly, there is no direct information on 14 MeV neutrons (or the associated neutron spectrum) effects to superconductors. Most of the current information comes from fission neutrons[54-58] and charged particle (H^+[59], D^+[60], or electrons[61]) bombardment.

Parkin and Schweitzer[57] have performed a recent experiment where both multifilamentary composite wires of NbTi and Nb_3Sn have been irradiated at 60°C to neutron fluences of $6 \times 10^{19} n/cm^2$ (~0.02 dpa). Measurements of the critical current as a function of magnetic field showed that whereas the NbTi was only moderately affected by irradiation, Nb_3Sn underwent a catastrophic reduction of I_c with an apparent threshold at 2 to $3 \times 10^{18} n/cm^2$ (~0.001 dpa) (Figure 10). At $6 \times 10^{19} n/cm^2$ the I_c for NbTi (40 kG) was still 82% of the unirradiated value while the I_c for NbTi was only ~4% of its pre-irradiation value. The value of T_c for NbTi was relatively uneffected (<1°K) whereas T_c for Nb_3Sn was reduced from 16.4 to 6°K. Post-irradiation annealing of Nb_3Sn for 1/2 hour at 400°C produced only a 19% recovery in I_c. Again, these results should be tempered with the knowledge that the irradiation was conducted at room temperature and not at 4°K.

Soell[58] has recently conducted an experiment where both NbTi and Nb_3Sn were irradiated at 4°K to 3-4 $\times 10^{18} n/cm^2$ (~0.001 dpa, roughly a factor of 20 below Parkin and Schweitzer's results.) He found that the response of NbTi depended very much on the pre-irradiation heat treatment and that decreases of less than 15% were found. In post-irradiation annealing of the Nb-Ti alloys restored the critical current density to 96% of the initial values.

The Nb_3Sn wires actually showed an increase in J_c at $4 \times 10^{18} n/cm^2$ (0.001 dpa) and ~1°K drop in T_c.

It is clear that more information is required at high fluences (~0.01 dpa) with a correct 14 MeV neutron spectra and at low temperatures before a final assessment of the problem can be made. The effect of periodic warm ups and the cost of increased shielding must also be assessed. However, there appears to be enough solutions available such that superconducting magnets can be used with fusion reactors.

4.4 Decay of Radioactive Species

The decay of radioactive transmutation products can lead to substantial operational and maintenance problems in fusion reactors. The gamma rays emitted during operation can represent safety problems to operating personnel and since a large fraction of the radioactive atoms have a half life > 1 day, maintenance of defective components can also be difficult. The heat generated by the decay can also be substantial (~1% of the operating power) and must be properly handled during disassembly and repair to avoid harmful temperature excursions in the critical metallic vacuum walls.

4.4.1 Radioactivity

Table IV gives a list of typical radioisotopes inventories in CTR systems which have already been proposed[4,6,13,27,62]. When all isotopes are included ($t_{1/2}>0$) it can be seen that all the structure in a D-T reactor can generate in the neighborhood of 10^6 curies/MW_t of power. Furthermore, while the activity is due to many different isotopes, the level of decay for the high temperature metals is $\geq 10^6$ Ci/MW_t for the first day. On the other hand, the activity from aluminum systems drops off quite rapidly, by 6 orders of magnitude in one week. However, the analysis of Al-Al_2O_3 system in the BNL reactor concept did not include radioactivity from impurities and activation of shield materials. The vanadium system shows considerable promise in that a 6 order of magnitude drop in radioactivity takes place in $\leq$6 months.

Fig. 11 shows that the reactor systems which use iron and nickel base alloys require approximately 2-4 years for a factor of 10 reduction in the decay rate and ~1000 years is required before a reduction of 10^6 is achieved. This means that the radioactive components will have to be stored for a long period after the reactor is shut down before they can be safely returned to the environment. Finally, the use of Nb as in the ORNL design results in about the same decay level as for Fe and Ni based alloys for the first year, it is slightly lower for the next 100 years, but is substantially higher after 1000 years. This stems from the production of ^{94}Nb which has a 20,000 year half life.

In summary, large inventories of radioactive species will be generated in the metallic components of fusion reactors. While these elements are normally quite firmly bound in the metal structure, their potential release during accident situations could present some safety problems. Another area of concern is the absolute levels of radiation in and around fusion systems during operation as well as radiation levels associated with components that need to be repaired or removed. The long term storage of these components should be also considered for high temperature systems such as Fe and Ni based alloys or refractory metals like Nb.

4.4.2 Afterheat

The release of energy due to the decay process can generate a significant amount of heat in a metal. Afterheat calculations on selected systems are also shown in Table IV. It is seen, again depending on material, that as much as 0.5-0.9 % of the equilibrium power level can be generated at shutdown and 10years of operation. (Figure 12) The short lived Al transmutation products cause this element to also have a relatively high afterheat (0.5%)although this decays away in a week or so. Stainless steel systems result in ~0.5% afterheat levels and require almost 2 years to decay to the 0.1% level. This means that as much as 5 MW_t of heat could still be generated from 316 SS structure in a 5000 MW_t CTR 2 years after shutdown. Finally, the Nb system has the lowest initial level of afterheat of all the systems considered here for high temperature operation, and remains lower than for steel 50 years after shutdown. The afterheat level in Nb approaches a constant value of ~5 x 10^{-5}% after 200 years due to the long lived ^{94}Nb isotope.The significance of afterheat in relation to reactor safety has been analyzed by Steiner[63,64] and Sze.[65] Both pointed out that in contrast to fission systems where the afterheat is generated in a relatively small volume, the afterheat density in fusion systems is quite low (0.5-0.9 watts/cm^3). This means that considerable cooling can

take place by conduction, convection, or eventually radiation such that temperature increase in currently envisioned CTR blankets should be no more than a few hundred degrees over the nominal operating temperature.

5. Possible Solutions

It has recently been proposed[66,67] that many of the aforementioned problems could be greatly reduced if the design of reactors were changed. In these papers it was suggested that most of the extraction of the kinetic energy as well as the breeding could be done inside the vacuum vessel. The consequence of such an approach is that, in principle, no neutrons are required to pass through the first wall. The four basic features of that proposal are:

1. To place a flexible, but porous 3-dimensional woven graphite structure in the vacuum, in front of the first wall. This structure could be from 20-50 cm thick (or 50% dense material) and would significantly degrade the energy spectrum so that the energy deposited in the first metallic vacuum wall is reduced. The heat generated in the graphite would be radiated to the first wall or conducted away by heat pipes.
2. To incorporate neutron multiplying pellets such as Be_2C, in the weave to increase the neutron production. This is done to enhance breeding in Li-6 with the thermalized neutrons.
3. To incorporate pellets of a high temperature breeding material such as $LiAlO_2$ (enriched in Li^6) in the weave. This material would absorb the thermalized neutrons before they strike the first wall. The tritium produced could diffuse into the vacuum chamber to be collected with the unburnt fuel. The lithium compounds would probably have to be covered with a pyrolytic carbon coating to reduce the evaporation of the compound at high temperature (~2000°C).
4. The first wall can be further protected by using a highly absorbing material such as borated graphite behind it to collect any neutrons which escape the thermalization and breeding zones and prevent their being reflected back into the first wall.

An example of the reduction in dpa and He production rates in vanadium for such a system in Fig. 13 as a function of the thickness of a pure carbon or graphite zone in front of the first wall. Note that a factor of 10 in dpa rate and a factor of 100 in helium and hydrogen production rates has been accomplished by 50 cm of a 50% dense mixture of the inner breeder zone. The long lived radioactivity is also reduced by a like amount as is the 14 MeV neutron sputtering.

The significance of this reduction can be realized by noting that if the unprotected first wall lifetime was determined to be 2 years due to embrittlement, then its lifetime may be 20-200 years in the protected scheme depending on whether it was pure displacement damage or gas atoms which was the major embrittling mechanism. The long term radioactive storage problems as well as the short term maintenance problems will also be greatly alleviated The reader is referred to reference (67) for more details on this particular concept.

6. Conclusions

The radiation damage problems facing the fusion reactor designer are truely immense. Many years of experimentation with the proper neutron spectra, materials and at correct temperatures will be required before one will be able to say

if conventional CTR blankets and shield will function reliably and safely. The use of internal spectral shapers and breeders may represent a solution to a multitude of these problems.

Acknowledgements

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Table I

Materials for Fusion Reactors

Function	Typical Examples	Top °C	$\phi t\ yr^{-1}\ (MW/m^2)^{-1}$ 14.1 MeV	$\phi t\ yr^{-1}\ (MW/m^2)^{-1}$ Total	Special Comments
			General		
First Wall and Blanket Structural Components	Austenitic Steels (AISI 304,316,347,etc.)	400-650	4×10^{13}	2×10^{14}	
	Nickel Based Alloys (PE16,Inconel,Incoloy,etc.)	500-700			Low Li Corrosion Resistance
	Refractory Metals (V,Nb,Mo or alloys)	600-1000			
	Other (carbon, SiC)	~1000			Low Tensile Strength
	Sintered Aluminum Product	~300-500			
Neutron Multiplier*	Be,	400-600	$\sim 1 \times 10^{13}$	1×10^{14}	High He Gas Production
	BeO, Be_2C	400-1000			
Breeding	Li	300-1000	$\sim 1 \times 10^{13}$	1×10^{14}	High Gas Production
	LiAl	300-600			
	$LiAlO_2$, Li_2O	600-1500			
Reflection (Moderator)	Graphite,Steel	400-1000	4×10^{12}	2×10^{13}	High He Gas Production
Radiation Shielding	B, B_4C, Pb, Steel	100-300	10^{11}	10^{13}	
			Special		
Electrical Insulation	Al_2O_3, MgO, Y_2O_3	500-800	4×10^{13}	2×10^{14}	~100 kv/cm, pulsed operation
Optics for Laser Systems					
Windows	Ge,NaCl,KCl,etc.	~100-200	up to 4×10^{13}	up to 2×10^{14}	pulsed neutron fluxes and temp. loads
Mirrors	GaAs,CdSe,etc	~100	variable	variable	
Thermal Insulation	Mylar	~-240 to -270°C	$\sim 10^6$	$\sim 10^8$	Large susceptibility to gamma irradiation
Superconducting Magnet Filaments	NbTi, Nb_3Sn V_3Ga	-269°C	$\sim 10^6$	10^8	Periodic annealing to R.T. can remove substantial damage
Superconducting Stabilizing Materials	Cu,Al	-269°C	$\sim 10^6$	10^8	"
Magnet Support Structure	Austenitic Steel	-200 to +30°C	$\sim 10^6$	10^8	

* Not necessary in all systems

Table II

Typical Displacement and Gas Production Rates in Metals

	Fusion Reactor First Wall 1 MW/m^2			First Fission Test Reactor-EBR-II (max)		
Material	dpa/yr	appm He/yr	appm H/y	dpa/yr	appm He/yr	appm H/yr
SAP[a]	17	410	790	76	7.9	50
316SS[b]	10	200	540	44	4.7	270
Nb	7	24	79	28	1	6.6
Mo	8	47	95	30	1.8	3.5
V	12	57	100	54	0.5	14
C	10	2700	Neg.	5	130	Neg.
Be[c]	(d)	2800	130[e]	(d)	3300	Neg.

a) SAP = Sintered Aluminum Product, 5-10% Al_2O_3 in Al

b) SS - Stainless Steel

c) ~Typical of 5 cm from first wall

d) displacement crossection not available

e) Tritium

Table III

Non Gaseous Transmutation Products in
First Walls of a DT Fusion Reactor

Base Material	Major Product	At.% per year per MW/m^2	Solubility Limit At % at 0.4 Tm
Al	Si	5×10^{-4}	0.06
	Mg	0.05	4
316 SS	Mn	0.15	60
	Ti	5.6×10^{-3}	3
	V	0.022	20
Nb	Zr	0.18	10
V	Cr	0.013	Complete Solubilit
	Ti	0.008	~70
Mo	Tc	0.12	~35

Table IV

Radioactivity and Afterheat Inventories Associated with Fusion Reactor First Wall Structural Materials

Material	Major Transmutation Product[a]	$t_{1/2}$	Level at t = 0, 10 year operation, 1.25 MW/m^2: Long Lived Curies[c,d] / kW_{th}	Afterheat - % of Operating Power for all Isotopes
316 SS	Fe-55	2.94y	140	0.7
	Co-58	72d	29	
	Mn-54	310d	24	
	Co-60	525y	47	
		Total[b]	310	
Nb	Nb-92m	10.1d	152	0.5
	Nb-95m	3.8d	50	
	Nb-95	35d	42	
	Sr-89	51d	30	
	Nb-94	50,000y	0.008	
		Total[b]	290	
V	Sc-48	1.8d	2.5	0.9
		Total[b]	5.6	
Al	Al-26	7.5×10^5y	0.004	0.7
		Total[b]	0.004	

a For those isotopes with $t_{1/2} > 1$d

b Including all isotopes at t=0, $t_{1/2} > 1$d

c Total activity in entire blanket usually on the order of 1.5 to 3 times that of first wall

d Ref. 27

Figure Captions

Figure	
1	Typical Neutron Spectra for a DT Tokamak Reactor[5]
2a	Instantaneous Neutron Displacement Rates in DT CTR First Walls and in a Fast Fission Test Facility, EBR II.
2b	Helium Gas to Displacement Ratios in Nuclear Reactors
3a	Typical Void Structures in CTR Materials Irradiated to 30 dpa 800°C with Tantalum Ions
3b	Void Induced Swelling in 304 Stainless Steel at 480°C as a Function of Neutron Fluence.[35]
4	Temperature Dependence of Void Swelling in Solution Treated Type 304 with 15 ppm of Injected Helium.[37] The Reactor Curve is from Bates and Straalsund, HEDL-TME-71-139(1971).
5	Effect of Helium Gas Bubble Size on Swelling in Metallic Beryllium
6	Effect of Neutron Irradiation on the Uniform Elongation and Yield Strength of Solution Treated 304 Stainless Steel.[44]
7	Total Tensile Elongation as a Function of Test Temperature for Molybdenum.[48]
8	Effect of Fast Neutron Irradiation on the Creep-Rupture Behavior of Annealed Type 316 Stainless Steel. E. E. Bloom p. 93 in Reference 51.
9	Resistivity Increase in Copper Irradiated at 4.2°K. After Horak and Blewitt.[53]
10	Effect of Neutron Irradiation on the Critical Current in NbTi and Nb_3Sn.[61]
11	Decay of Radioactivity Induced in the First Wall Structure of a 1000 MW_t DT Tokamak Reactor. Courtesy of W. F. Vogelsang.
12	Decay of Afterheat in the First Wall Structure of a 1000 MW_t DT Tokamak Reactor. Courtesy of W. F. Vogelsang.
13	Reduction of Neutron Damage in First Wall of a DT Fusion Reactor by Placing a Carbon Spectral Shifter in Front of a Vanadium First Wall.[67]

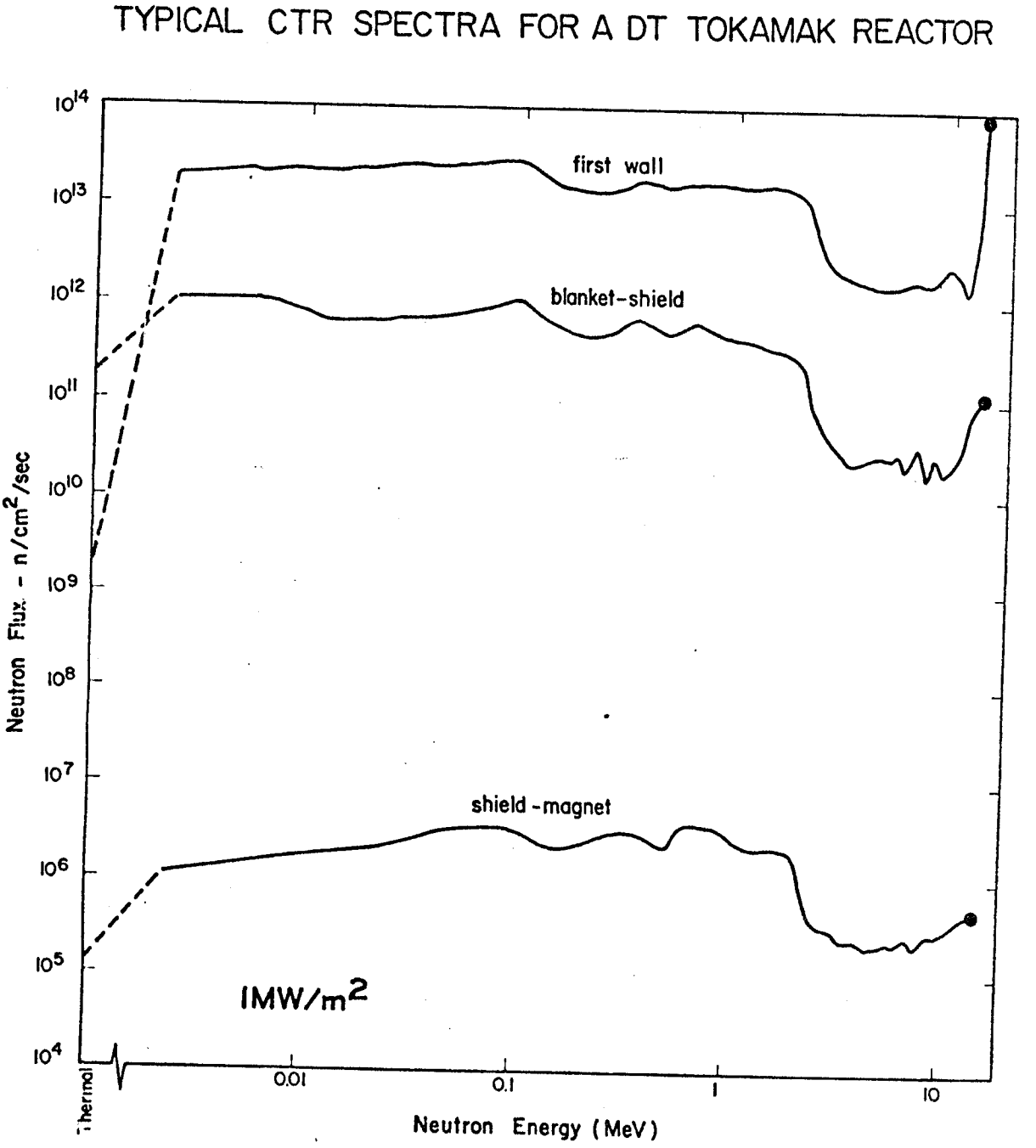

TYPICAL CTR SPECTRA FOR A DT TOKAMAK REACTOR
first wall
blanket-shield
shield-magnet
1MW/m2
Neutron Flux - n/cm2/sec
Neutron Energy (MeV)
thermal
0.01
0.1
1
10

Figure 1

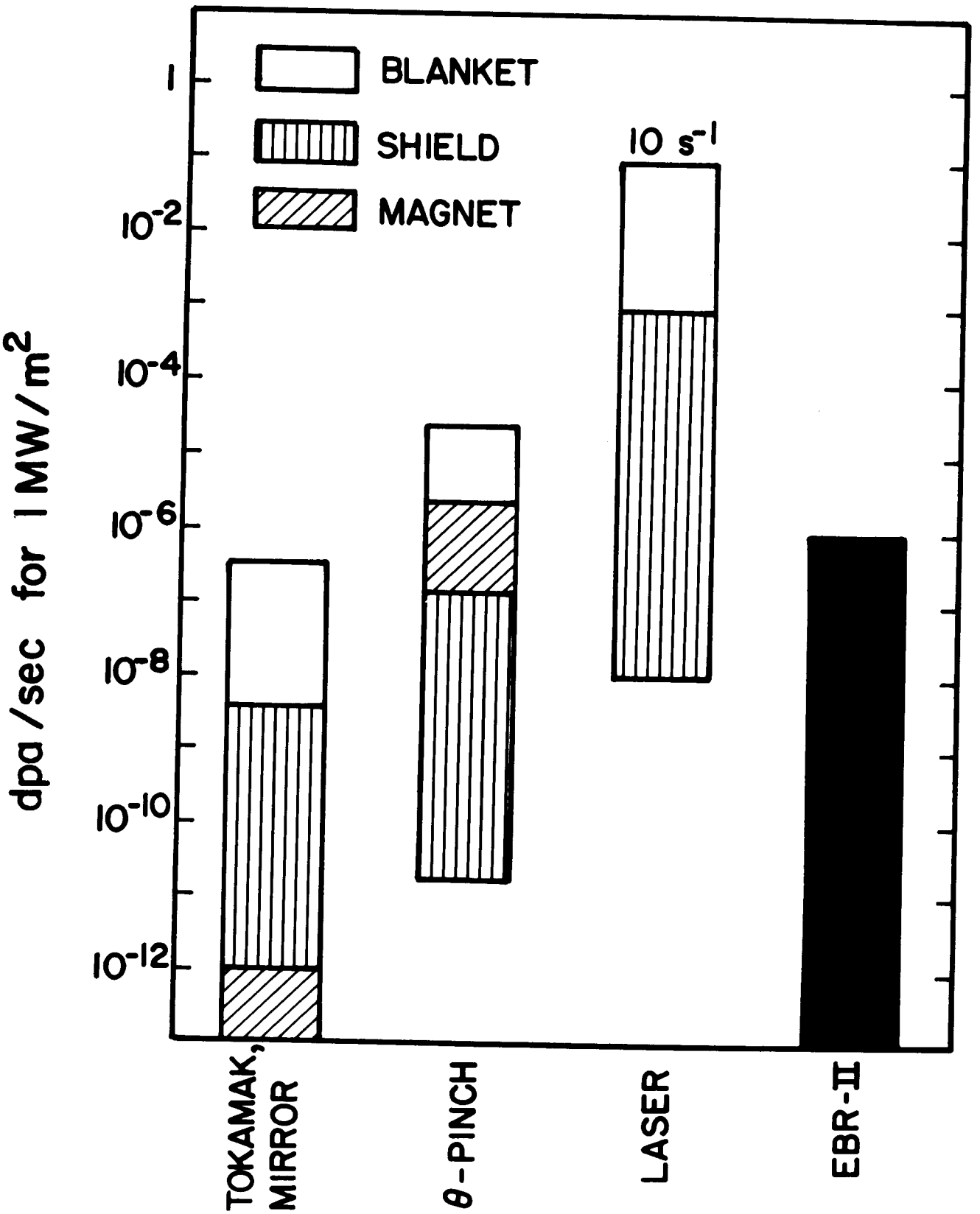

INSTANTANEOUS NEUTRON DISPLACEMENT RATES IN DT CTR FIRST WALLS
BLANKET
SHIELD
MAGNET
10 s⁻¹
dpa/sec for 1 MW/m²
1
10⁻²
10⁻⁴
10⁻⁶
10⁻⁸
10⁻¹⁰
10⁻¹²
TOKAMAK, MIRROR
θ-PINCH
LASER
EBR-II

Figure 2A

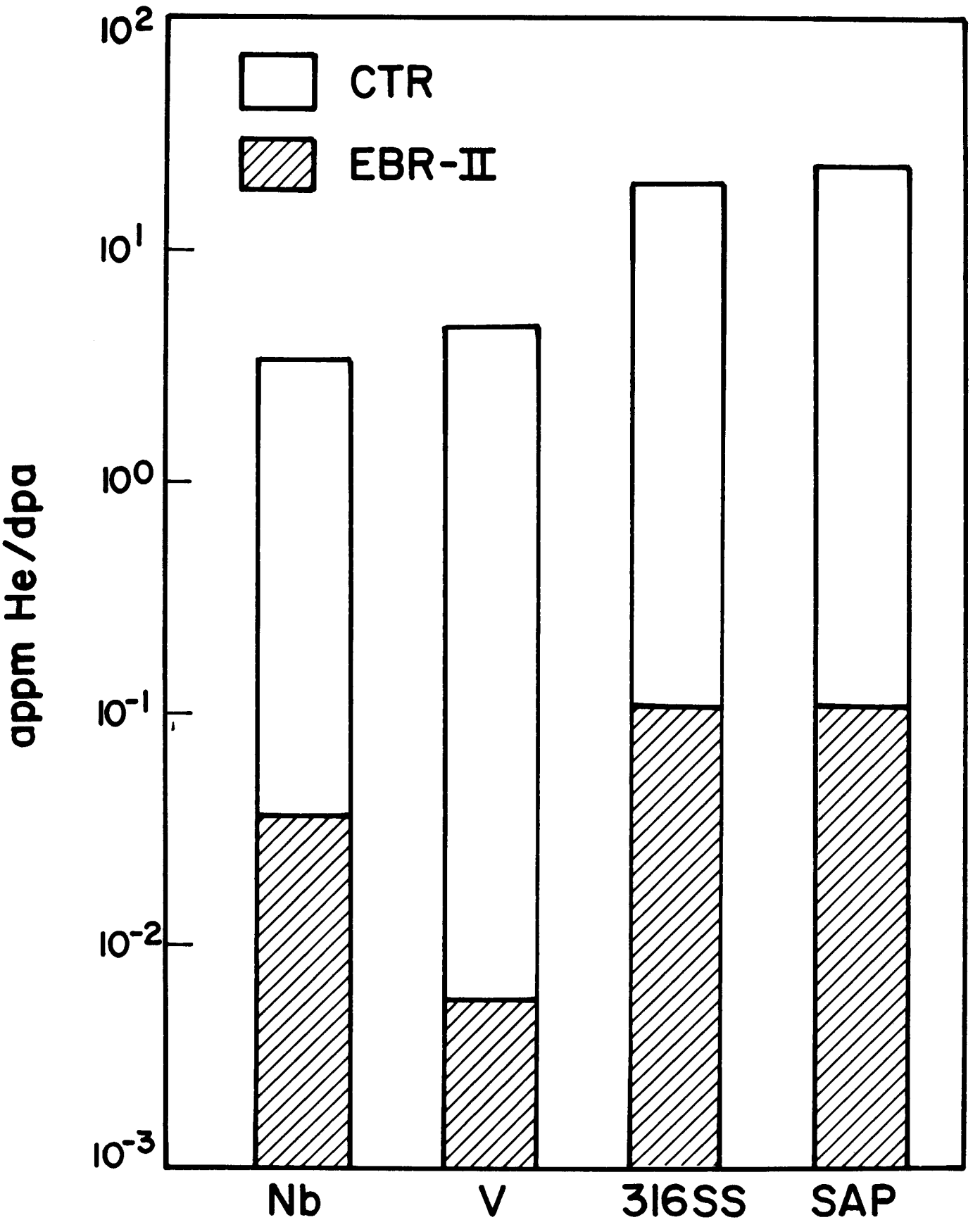
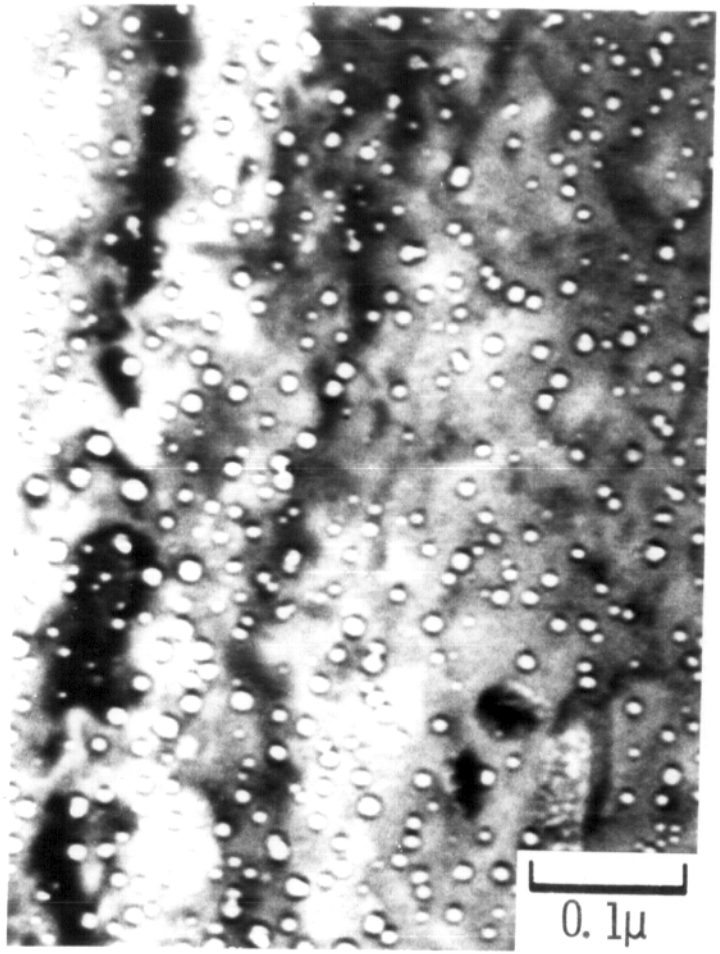
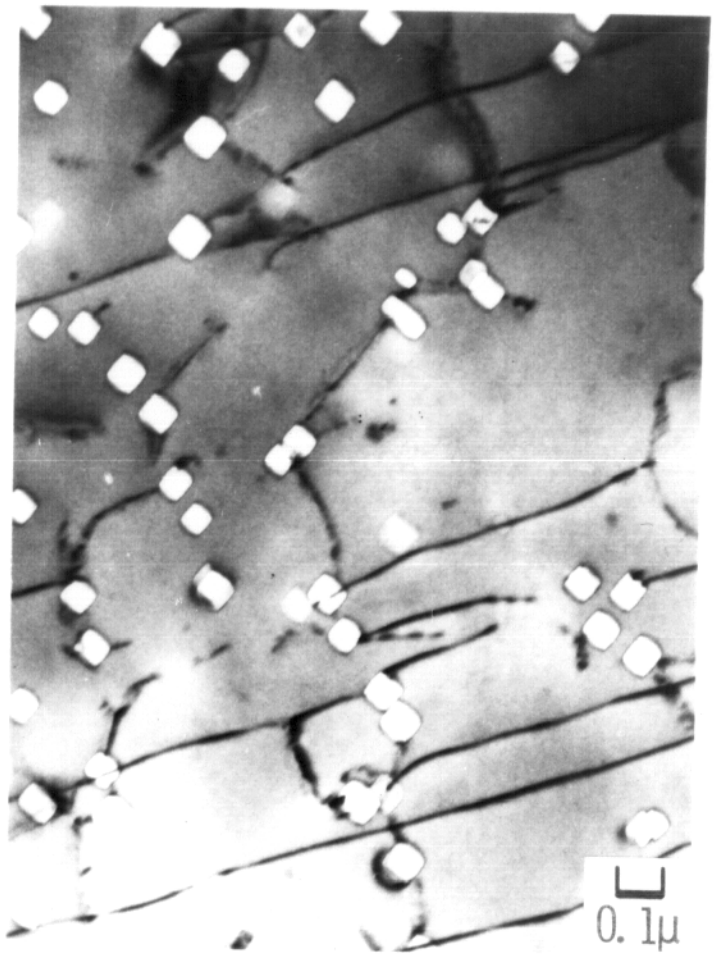

HELIUM GAS TO DISPLACEMENT RATIO
CTR
EBR-II
appm He/dpa
10^2
10^1
10^0
10^{-1}
10^{-2}
10^{-3}
Nb
V
316SS
SAP

Figure 2B



Nb-1Zr



VANADIUM

TYPICAL VOID STRUCTURES IN CTR MATERIALS IRRADIATED TO 30 dpa at 800°C WITH TANTALUM IONS

Figure 3A

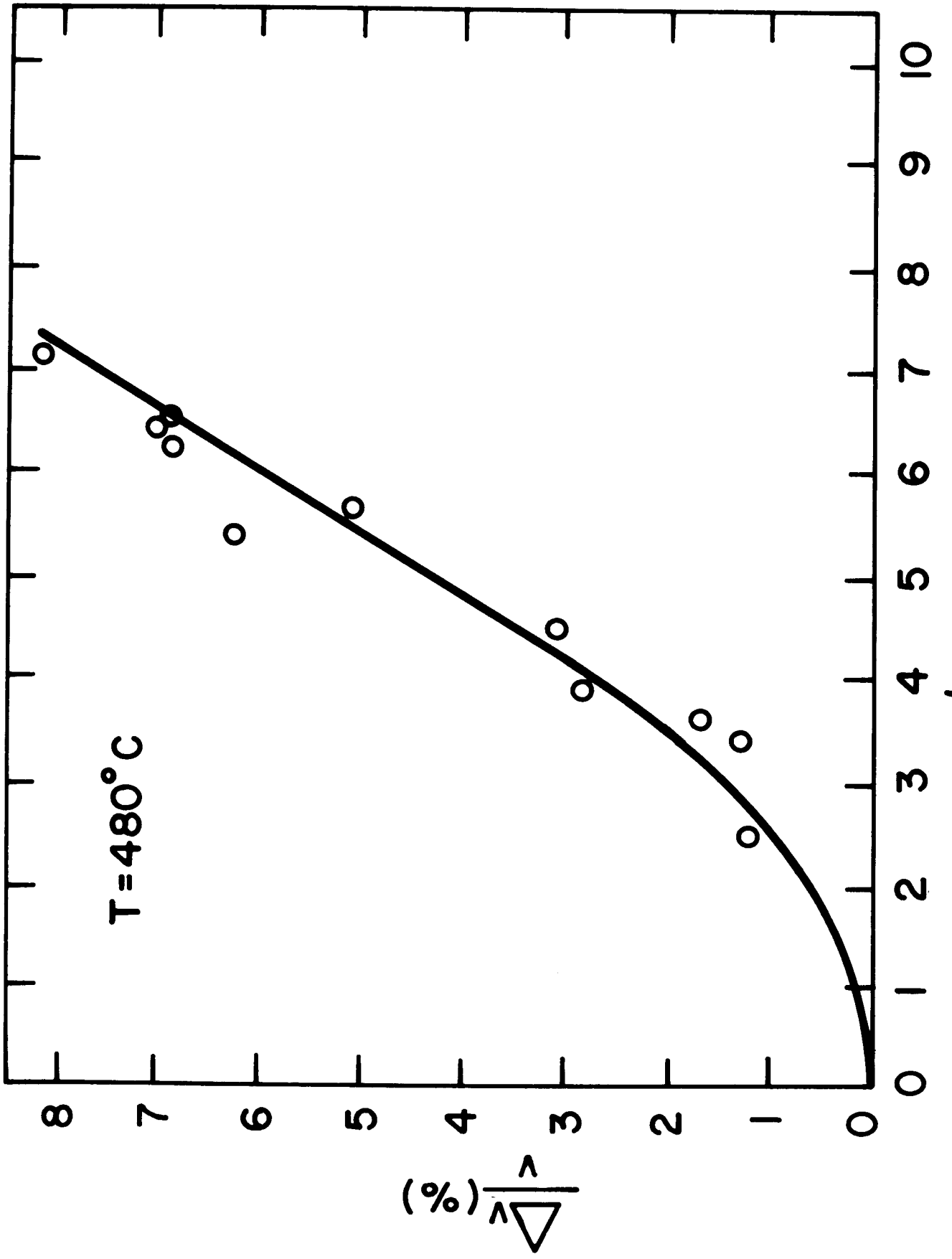

T=480°C
$\frac{\Delta\lambda}{\lambda}$ (%)
ϕ_t (10^{22} n/cm^2)
0
1
2
3
4
5
6
7
8
9
10

Figure 3B

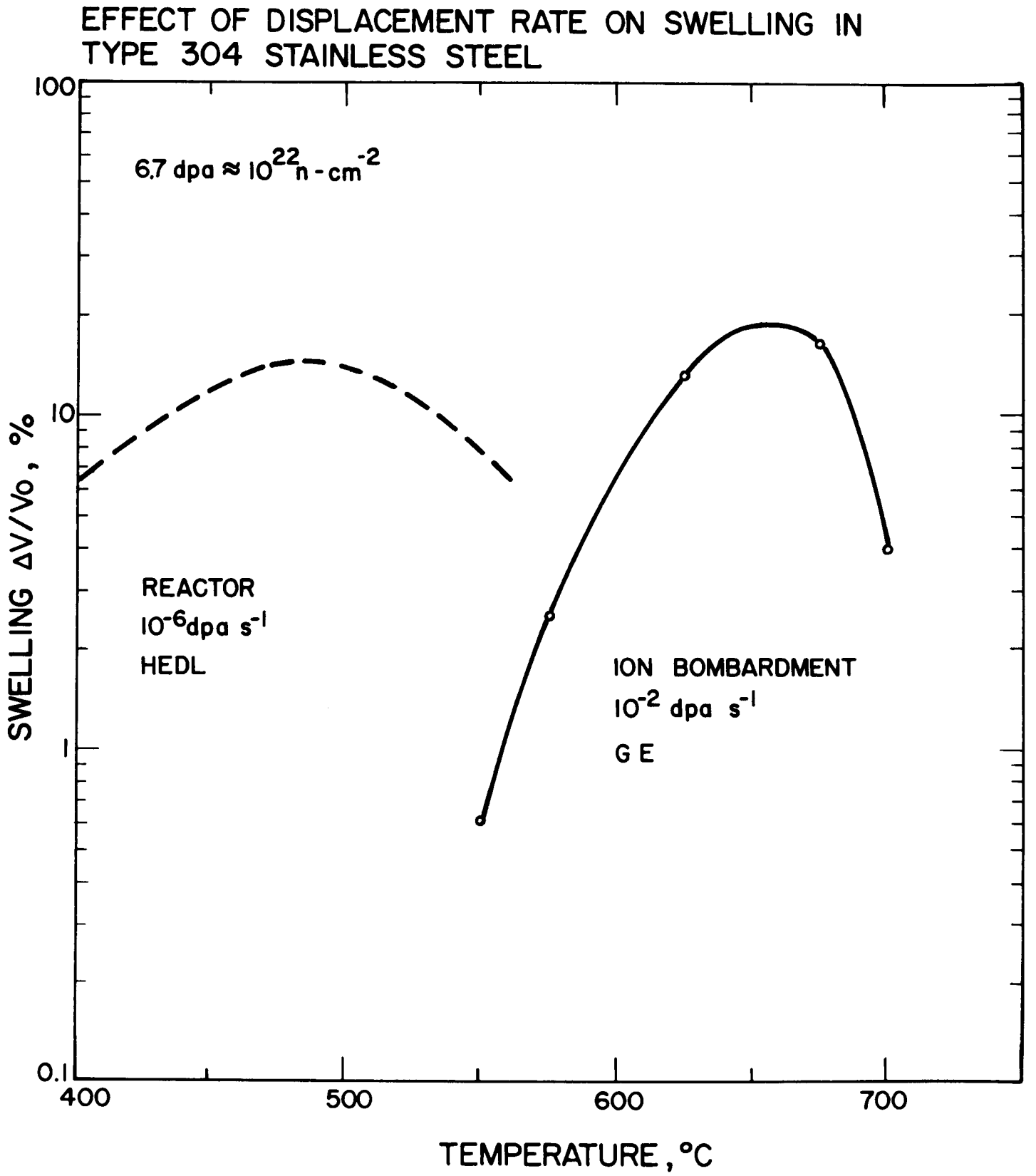

EFFECT OF DISPLACEMENT RATE ON SWELLING IN TYPE 304 STAINLESS STEEL
100
10
1
0.1
SWELLING ΔV/Vo, %
6.7 dpa ≈ 10^22 n - cm^-2
REACTOR
10^-6 dpa s^-1
HEDL
ION BOMBARDMENT
10^-2 dpa s^-1
G E
400
500
600
700
TEMPERATURE, °C

Figure 4

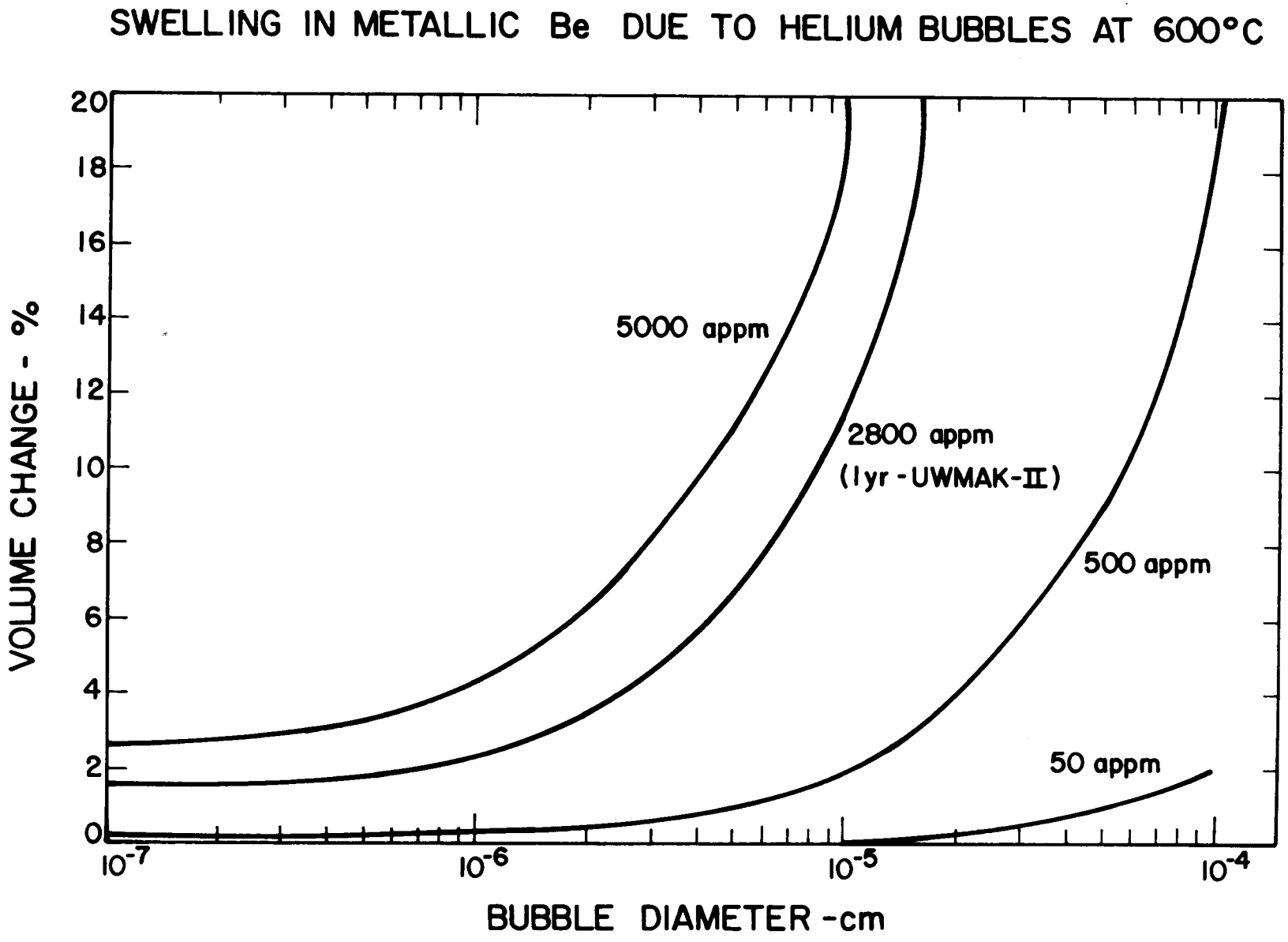

SWELLING IN METALLIC Be DUE TO HELIUM BUBBLES AT 600°C
VOLUME CHANGE - %
20
18
16
14
12
10
8
6
4
2
0
5000 appm
2800 appm
(1yr - UWMAK-II)
500 appm
50 appm
10^{-7}
10^{-6}
10^{-5}
10^{-4}
BUBBLE DIAMETER - cm

Figure 5

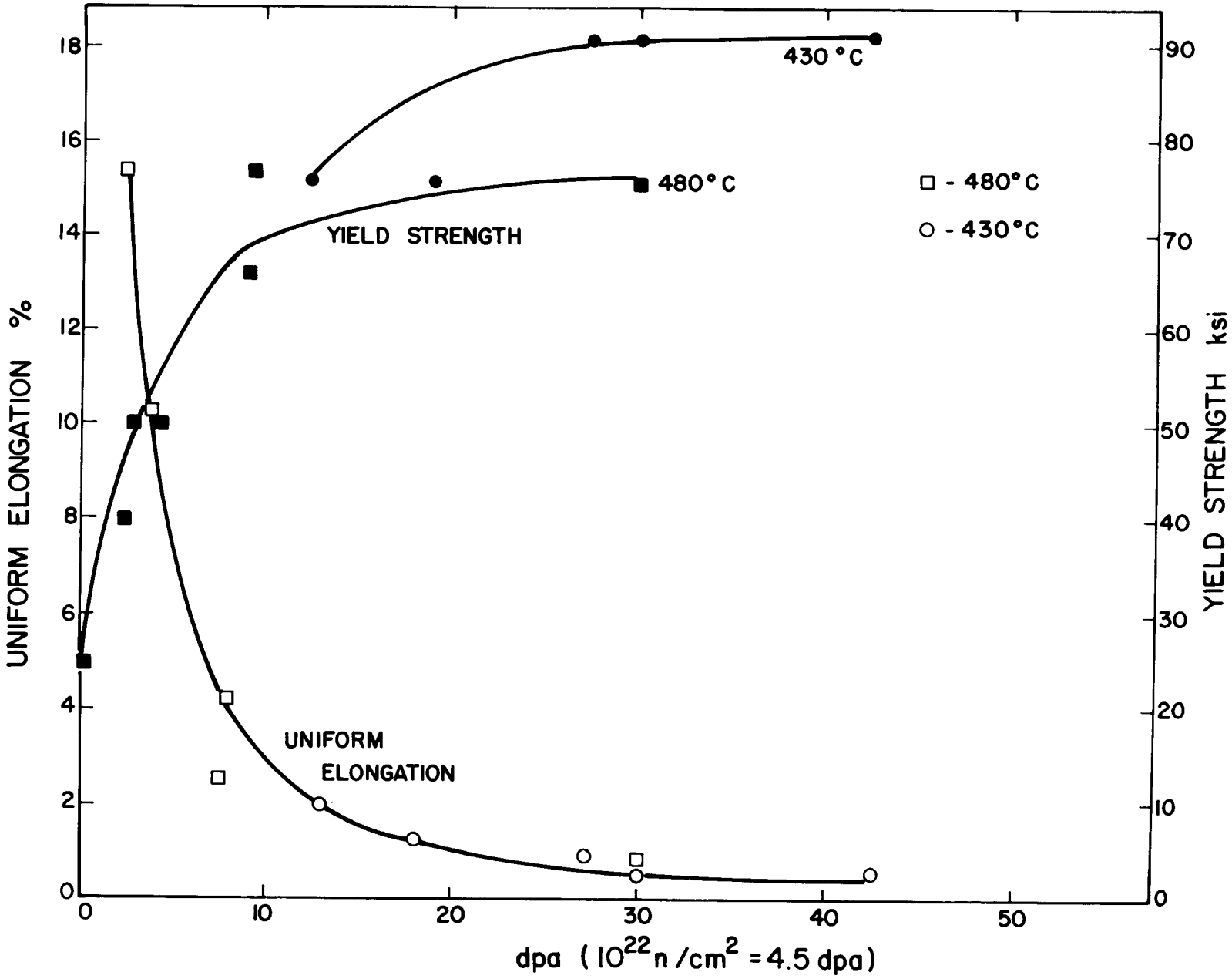
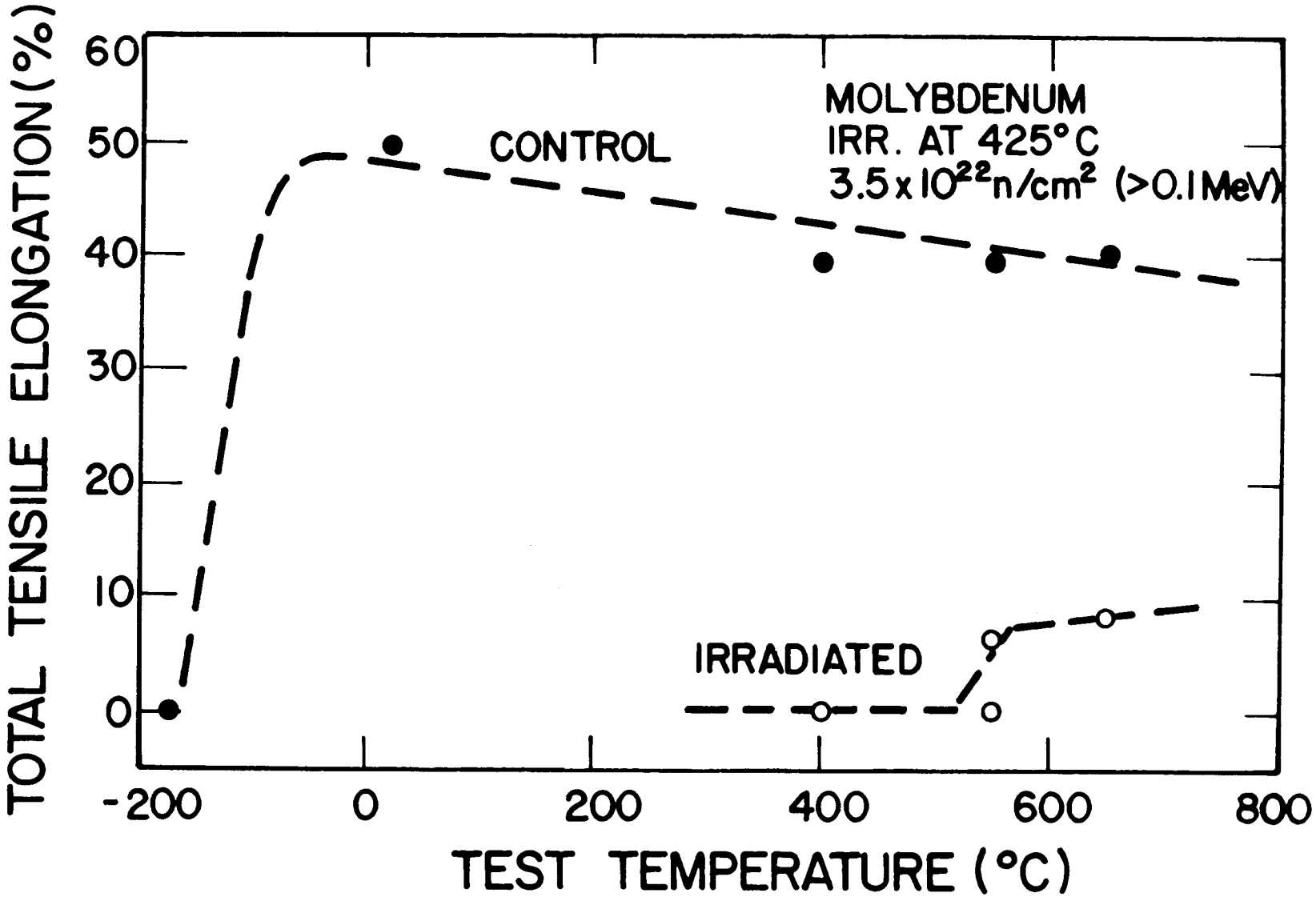
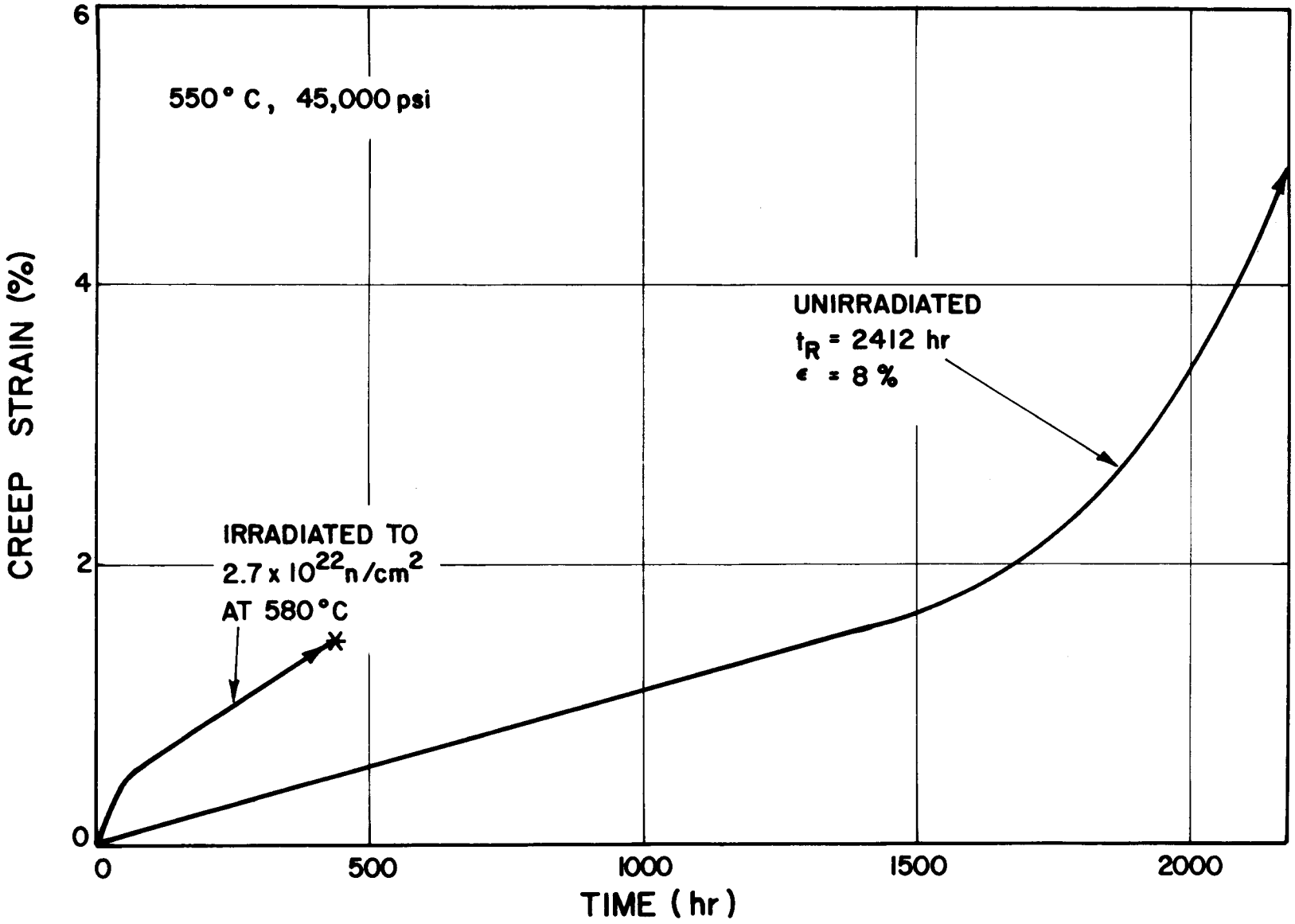

UNIFORM ELONGATION %
YIELD STRENGTH ksi
dpa (10^{22} n /cm^2 = 4.5 dpa)
430 °C
480 °C
YIELD STRENGTH
UNIFORM
ELONGATION
□ - 480°C
○ - 430°C
0
2
4
6
8
10
12
14
16
18
0
10
20
30
40
50
0
10
20
30
40
50
60
70
80
90

Figure 6



IRRADIATION EFFECT ON DBTT OF Mo

Figure 7



EFFECT OF FAST NEUTRON IRRADIATION ON THE CREEP-RUPTURE BEHAVIOR OF ANNEALED TYPE 316 STAINLESS STEEL. (45)

Figure 8

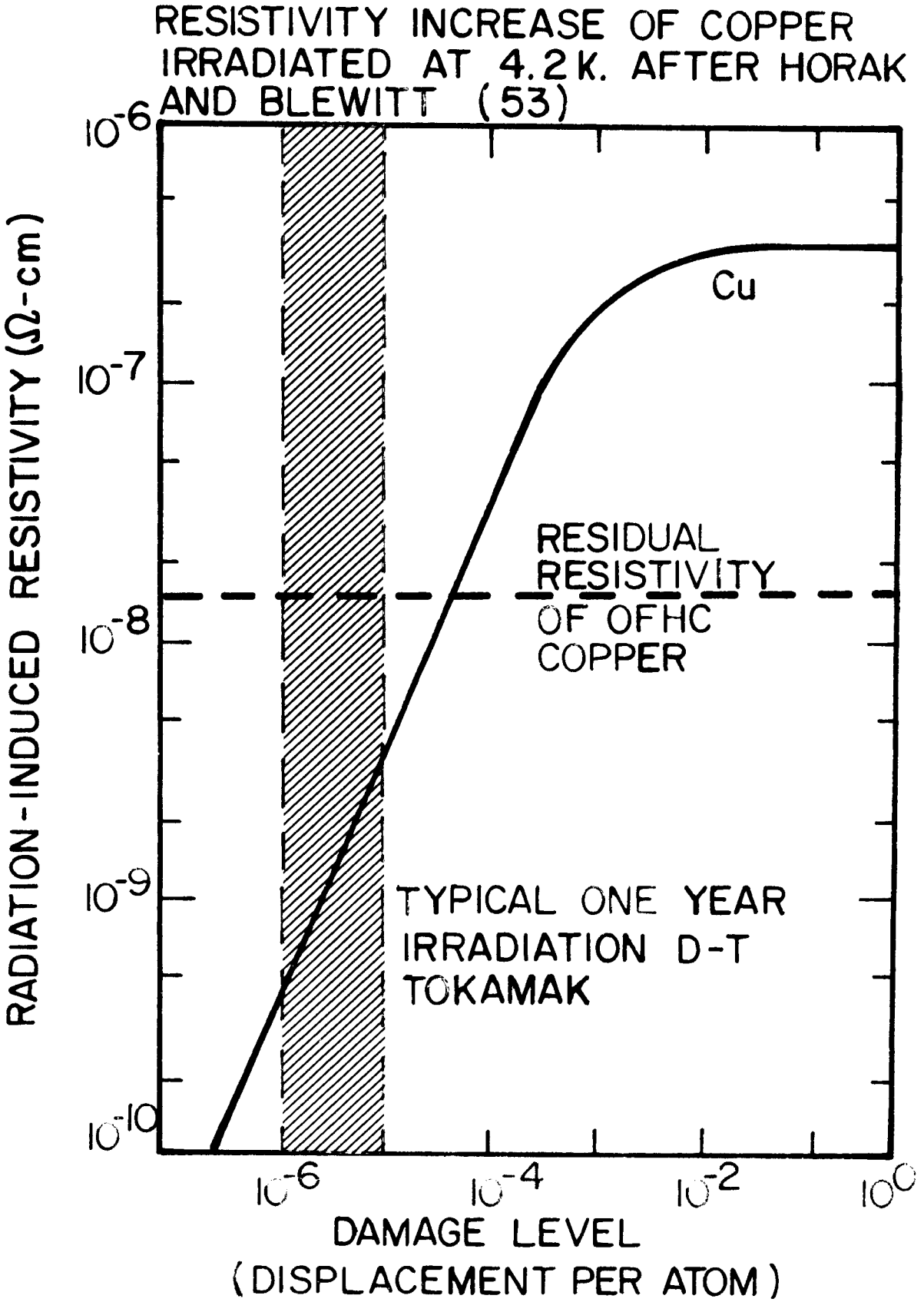
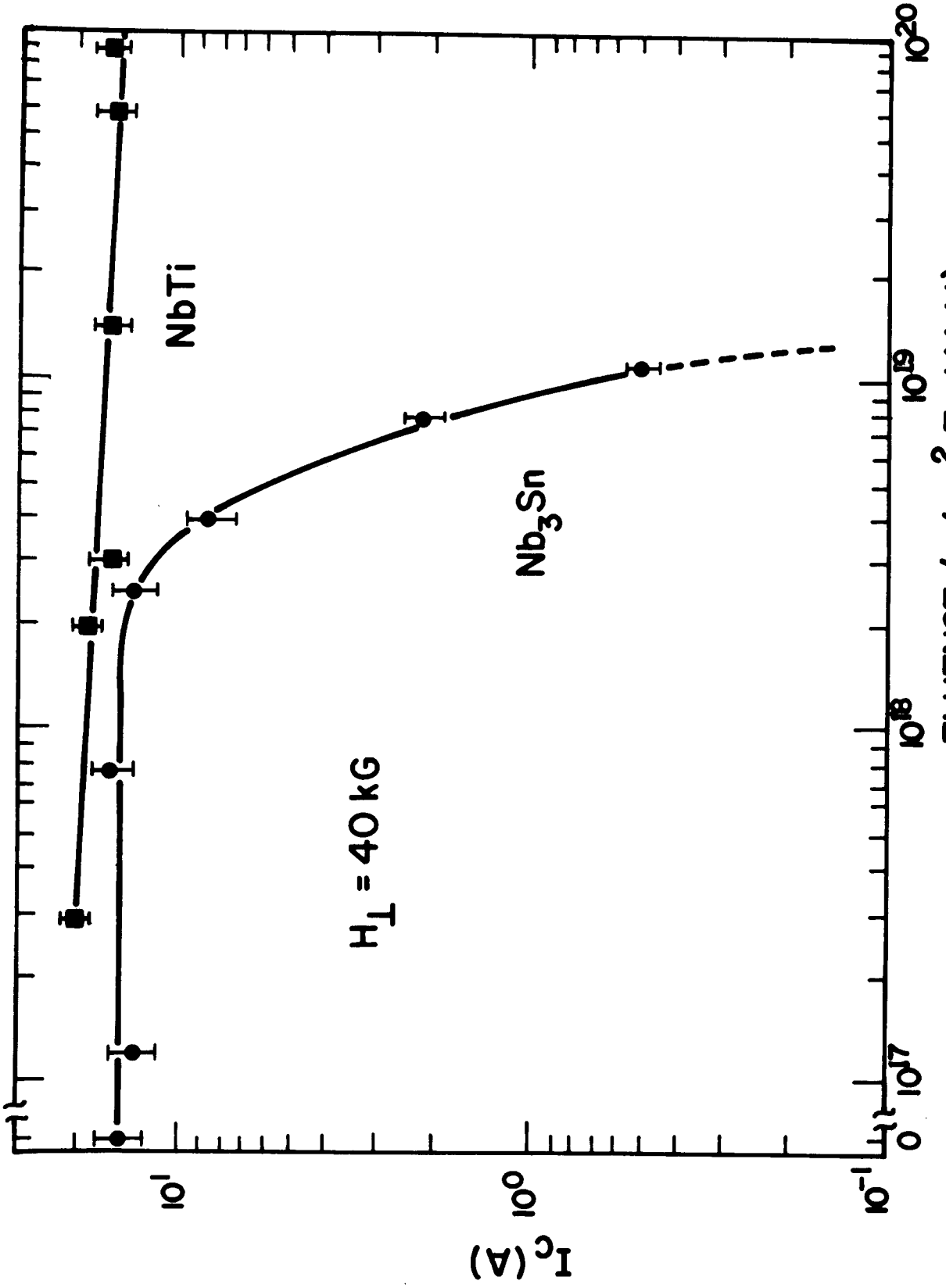

RESISTIVITY INCREASE OF COPPER IRRADIATED AT 4.2 K. AFTER HORAK AND BLEWITT (53)
RADIATION-INDUCED RESISTIVITY (Ω-cm)
10-6
10-7
10-8
10-9
10-10
Cu
RESIDUAL RESISTIVITY OF OFHC COPPER
TYPICAL ONE YEAR IRRADIATION D-T TOKAMAK
10-6
10-4
10-2
100
DAMAGE LEVEL
(DISPLACEMENT PER ATOM)

Figure 5



EFFECT OF NEUTRON IRRADIATION ON THE CRITICAL CURRENT IN NbTi AND Nb_3Sn. (61)

Figure 10

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