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Abstract

The major conclusions from the conceptual design of a Tokamak fusion reactor, UWMAK-I, are presented. The implications of the basic results and areas of major uncertainty are described. Many of the technological problems posed by such a reactor appear solvable within reasonable extensions of existing technology but some major advances remain to come in two general areas, plasma physics and technology, and materials, particularly first wall problems, before such fusion reactors are possible.

1. Introduction

A detailed description of the conceptual Tokamak fusion reactor, UWMAK-I,[1] has been presented in another paper in this volume.[2] The purpose of the UWMAK-I study has been to carry through a conceptual design in sufficient detail to identify and quantify some of the major technological problems that will face the designers of fusion power reactors. Some problems are specific to the Tokamak confinement concept but many are applicable to deuterium-tritium fueled fusion reactors in general. In this paper, we consider the major conclusions and technological implications of this work. The paper is organized such that each major technical area is treated separately but the impact of problems in one area on those in other areas is considered throughout.

2. Plasma

It is clear that a number of serious uncertainties remain in the plasma physics and operation of Tokamak devices and these will have a serious impact on the ultimate design and operation of Tokamak fusion reactors. These uncertanties relate to the startup of a large Tokamak discharge, the plasma scaling laws that will ultimately govern the plasma physics of the reactor regime, the effects of impurities and impurity control, the operation and efficiency of divertors for impurity control, and methods for fueling the plasma. The last two areas are critical if long burn times, on the order several minutes to an hour, are to be achieved.

The startup of a large Tokamak discharge is an important open question. The use of a moving limiter for UWMAK to perform such a function is being studied presently[3] and such operation will be studied on PLT.[4] The use of an expanding magnetic limiter is also a possibility and this may be studied on the Poloidal Divertor Experiment currently being designed.[5] For UWMAK-I, we have studied the trade off between the startup time and the peak power required from an energy storage unit. The energy required to provide the requisite transformer action is 16 MW-hr based on a plasma resistivity of 3.5 times the Spitzer value.

This anomolous increase is to account for either neoclassical effects[6] or the presence of impurities. Table 1 summarizes the results. It is assumed that it will be possible to purchase 500 MW_e from the line when a reactor such as UWMAK-I is operational. The National Accelerator Laboratory now purchases 200 MW_e from Commonwealth Edison of Chicago. The conclusion is that there exists a serious competition between short startup times and the desire to minimize the cost and size of energy storage systems for Tokamaks.

Table 1

Current Rise Time and Energy Storage

Startup Energy for UWMAK-I - 16MW-HR
($\eta = 3.5\ \eta_{sp}$)

Startup Time(sec)	Energy from Line (MW - hr) (Based on $500MW_e$)	Energy From Storage Unit (MW-HR)	Peak Power (MW)
1	.139	15.9	10^5
10	1.39	14.6	10^4
50	6.94	9.1	2×10^3
100	13.9	3.2	10^3
200	16.0	0	5×10^2

The effective plasma resistivity impacts on the transformer design as well as the stored energy requirement. The design discussed in Reference 1 can accomodate anomolous resistivities up to a factor of 8 times the Spitzer value before the maximum field of 8.6 Tesla appears at any of the transformer coils.

Plasma heating using neutral beams has been studied for the UWMAK-I system and the questions of the profile for beam energy deposition, beam energy requirements for penetration, and beam power requirements to achieve ignition or a prescribed heatup rate have been analyzed.[7] The analysis, based on pseudoclassical electron heat transport,[8] neoclassical ion heat transport,[9] and injection tangent to the magnetic axis, shows that large Tokamak reactors such as UWMAK-I can be ignited at low density ($\sim 3 \times 10^{13}$ particles/cm^3), using moderate levels of neutral beam power (10-75 MW), beam energies of several hundred KeV, and in times on the order of seconds. Ohmic heating alone is insufficient to ignite the plasma. For UWMAK-I, a 500 KeV beam of deuterons or tritons is adequate to provide injected power deposition profiles that are peaked on axis. Lower energy beams, such as 100 KeV, do not penetrate the plasma and thus yield temperature profiles peaked towards the plasma edge. However, in terms of heating, such beams would be adequate and can actually produce more rapid heatup rates. It has been found, for example that a 100 KeV beam produces a faster plasma heatup rate following termination of injection than a 350 KeV beam.[7] The low density beam heating phase is used to enhance beam penetration while reducing plasma losses. From this analysis, neutral beam heating appears to be an effective way to ignite a large power reactor such as UWMAK-I. It will be important, however, to assess the effects of alternate plasma scaling laws on these results. This work is in progress.

We noted in Reference 2 that plasma operation in UWMAK requires confinement times that are several hundred Bohm times but are ~2-3 orders of magnitude shorter than neoclassical confinement times. These requirements are for plasma operation at quasi-steady state during a burn time. The most economical temperature range for the plasma operation in a β-limited plasma is 10-20 KeV. If the scaling on τ_E and τ_c are proportional to $T^{1/2}$ then an energy equilibrium in the 10-20 KeV range is thermally unstable. Thermally stable operating conditions are possible at higher temperatures (T_i > 25 KeV) but require larger toroidal magnetic fields to produce the same power. For UWMAK-I, thermally stable and unstable equilibria have been calculated and operation at the stable point would require magnets that are approximately \$100 x 10^6 more expensive than for operation at the unstable point (still within NbTi superconductor technology.)

The method of producing thermally stable plasma operation in the desired n,T,τ range remains an open question. We have recently examined the effects of predicted microinstabilities[10] on Tokamak plasma operation.[11] In particular, using the predicted scaling for the dissipative trapped ion mode given by Dean et.al.[12] it is found that thermally stable plasma operation can be attained for temperatures in the range from 10 to 20 KeV. As a quantitative example, using UWMAK-I machine parameters for q, β_θ, A, B_ϕ (on axis), and the plasma current, I_p, of 20.7 MAmps, a thermally stable plasma equilibrium producing 5000 MW_{th} occurs at a plasma temperature of 14.5 KeV for $Z_{eff}= 1$. No impurities are purposely added to the plasma. Should Z_{eff} be larger, the stable equilibrium temperature increases.

When the confinement time is the order of several seconds, as is implied by trapped ion mode scaling, the problem of fueling becomes very important if long burn times, such as the 90 minute burn time for UWMAK, are to be achieved. Fueling with large pellets (~2mm diameter D-T pellets) appears formidable if the accelerating velocities required for penetration are to be achieved.[13] Larger pellets will constitute either too large a fraction of, or actually exceed, the number of particles in the plasma. Little effort has been devoted to detailed analysis and experiments in this area and more work is required to better understand the physical processes involved in pellet fueling of reacting plasmas.

The plasma stability factor, q, and the plasma Beta, β, are critical parameters for Tokamaks and while q > 1 (the Kruskal-Shafranov limit) is generally accepted, present devices generally have q at the plasma edge greater than 3. In addition, the poloidal beta, β_θ, (β in the weak magnetic field) is usually around 1/2. It is important to know the upper limit on β_θ and the lower limit on q for stable plasma behavior since this strongly effects plant economics. Figure 1 shows the scaling of magnet costs (toroidal field, transformer and divertor field coils) as a function of the safety factor, q(a), for different poloidal beta limits using UWMAK-I as the reference point. Along a constant β_θ line, the plasma current is fixed which means the cost of the transformer and divertor windings are also fixed and only the toroidal field magnet costs vary. On the other hand, all magnet costs scale as one moves vertically along a line of constant q since both B_ϕ and B_θ change such as to keep the ratio constant. It is clear that the best case is the line of largest constant β_θ, taken here to be β_θ = A. The right hand scale translates magnet costs to dollars per kW_e and all costs are for NbTi superconducting magnets. Along a constant β_θ line, q can only be increased by increasing the toroidal field, B_ϕ, which means increasing the maximum field at the toroidal field magnets. We have set 100 Kilogauss as the upper limit for NbTi coils (allowing for pumping on the helium) and have indicated this by the hatched line in Figure 1. For β_θ and q combinations above this line, the superconductor must be changed, the technology will change, and the costs will show a different scaling. It suffices to conclude that there is strong incentive for achieving high β and low q in a Tokamak (also for other β-limited devices, such as stellarators) and the cost figures give some quantitative indication of the economic penalties involved in various choices.

Long burn times also require control of impurities in Tokamak plasmas. Divertors such as that included for UWMAK-I may be the basis for such impurity control and can also serve to protect the first wall from bombardment by energetic particles. The advantages of the double null poloidal divertor, described in Reference 2 and elsewhere,[14] are that it is compatible with the desire for a small aspect ratio, it produces a vertically elongated D-shaped plasma that is consistent with the D-shaped toroidal field coil design, and it leads to low ratios of maximum field to toroidal field in axis, the value for UWMAK being 2.25. This last point is quite important since it allows the use of NbTi superconducting magnets and thus does not require a large extrapolation of present technology.

Poloidal divertors which preserve axisymmetry appear feasible but many open questions remain. The major questions that remain are the efficiency of such divertors for collecting particles diffusing out of the plasma, (~3.5 x 10^{22}

particles per second for UWMAK) and metallic atoms coming off the walls. Both experiment and analysis are required to better understand the physics in the region between the plasma edge and the first wall, and including plasma-wall interactions, the effects of neutral gas in the diverted field zone, the impact of neutron sputtering,[15] and the blistering and sputtering caused by charged particle bombardment of the first wall.

3. Magnets

Large bore, superconducting magnets are essential for the economic development of fusion power. The toroidal field coils design developed for UWMAK-I[1,2] is based on the recognition that complete cryogenic stability is the only viable design at this time. The alternative of using unstable magnets requires considerable development before any reliability can be predicted. In addition, from the UWMAK-I design, it appears that the required toroidal field on axis in Tokamaks can be achieved with current densities of about 1000 Amps/cm^2, a typical cryogenically stabilized value.

Many magnet shapes were possible for UWMAK-I but only two were seriously considered: A circular cross section[16] with external reinforcement rings to resist bending and a "D" shaped magnet[17] which provides a constant tension winding region without external rings. The constant tension design makes best use of a uniaxial stressed member and for this reason, and to accomodate the diverse requirements of the plasma and divertor, the constant tension design was chosen. Figure 2 shows the outline of this shape as curve (a), which is the true "D" shape and the only one for which the tension is constant for all portions of the conductor, including the straight vertical inner portion. For curve (b), the tension is constant only over the portion from the outside up to point B. For curve (c), the small starting radius and smaller tensile load does not permit the shape to reach the desired inner dimension. (This is constant tension from the outside to point C but from C to 0 in Figure 2, the extra magnetic loading cannot be carried by tension in the winding frame and is instead transmitted to the central core.) With these options, it has been found possible to economically accomodate various plasma shapes by modifying the basic constant tension "D" shaped magnet. Also, the effects of winding thichness have been included. It is also proposed to prestress the conductor during winding in order to make better use of the stainless steel. Shape (c) was chosen because it is the only one which assures the retention of the pre-stress.

Conductor design questions have also been addressed. It appears extremely difficult to conceive of winding such large bore, high field magnets from wire or tape. Several large conductor designs (~2cm x 2cm) have been considered[18] and a composite conductor of NbTi in copper sized to carry the total design current at 5.2°K was chosen. This conductor will be inserted into spiral grooves of varying width and depth on the face of forged, stainless steel pancakes. The magnets are then constructed by stacking individual pancakes[19] and the design uses cooling on both sides of the individual "D" shaped discs.

A liquid helium refrigeration system has been sized to meet the total magnet system heat loss rate of 11 kW and is found to be a relatively small cost factor, namely, about 5% of the total costs of the transformer, divertor, and toroidal field coils. However, the supply of helium is a potential problem area. The magnet system requires 250,000 liters with 200,000 liters in storage to allow for 24 hours of operation during a refrigerator malfunction.

The magnetic fields external to the plant coming from stray toroidal fields and from the divertor and transformer coils as described in I are also a potential problem. In UWMAK, the poloidal field associated with the system of toroidal currents, including the plasma current, are the main contributors to the field external to the plant. The mod B contour plot shown in Figure 3 indicates that the field drops under 1 gauss at about 500 meters and is falling off as a dipole-field. The average value of the earth's magnetic field in Wisconsin is approximately 1/2 gauss.

4. Neutronics

The neutronics and photonics studies of fusion reactor blankets and shields is central to the analysis of fusion power reactors. The neutronics studies for UWMAK-I[1,2] have shown that tritium breeding in stainless steel, lithium cooled systems is large enough that most conceivable variations in nuclear data are unlikely to prevent such systems from breeding. This is to be coupled with the fact that short doubling times, on the order of months, are possible in fusion systems even at lower breeding ratios than the 1.49 value for UWMAK. The same conclusion has been found if Nb, V, or Mo were substituted as the structural material. Variational calculations have shown that alterations in the percentage of structeral material in the tritium breeding zones away from 5%, as used in UWMAK, do not prevent SS, V, or Mo systems from breeding.[20] However, 15% Nb in the breeding zones lowers the breeding ratio to 1 in a UWMAK-type blanket.

Methods for calculating space dependent nuclear heating in fusion systems have been developed at Wisconsin[21] and the results have shown that the total energy per fusion reaction is about 10% lower than the nominal value of 22.4 MeV often assumed in fusion work. This is particularly relevant since the reactor power output is directly proportional to this value. From these studies, we have also concluded that enrichment of natural Li in ^{6}Li does <u>not</u> significantly effect energy multiplication and, given that it is an expensive process, such enrichment does not produce an economic gain. For UWMAK-I, enrichment to 50% ^{6}Li increases the energy multiplication by less than 1% over the natural lithium case. The energy produced can be improved, however, by addition of Be in the blanket and this appears economical. For example, 4 cm and 10 cm of Be increases the total energy production by 9% and 19%, respectively. The additional costs appear to be less than half the decrease in cost per unit power. On the other hand, as we shall discuss shortly, the low reserves of Be both inside and outside the U.S. appears to forclose the possibility of widespread use of Be in a 10^6 MW_e fusion economy.

The reflector in the UWMAK-I blanket is stainless steel rather than graphite. A SS reflector improves energy multiplication and energy attenuation. It also allows for a thinner reflector zone which makes the blanket and shield thinner and brings the magnets closer to the plasma, thus saving on magnet costs. The breeding ratio is lowered slightly but 1.49 for UWMAK is perhaps too high in any case.

The shield design given in ref. 2 is based on optimization studies to minimize cost when the costs of the magnets, the shield itself, and refrigeration for the magnets, are considered. The optimum materials choice for the shield is 70% lead and 30% boron carbide. In practice, 20% stainless steel has been included for structure and the lead content has been lowered to 50%. The shield thickness of 77 cm yields a blanket-shield system thinner than is usually employed. The blanket could actually have been even thinner and still produced adequate tritium breeding because the blanket size was governed by heat transfer and coolant flow considerations.

Another area of considerable importance which we have studied is the gas production rates as a function of position in fusion blankets. In the first wall of UWMAK,[1,2] the hydrogen production rate is 636 appm/year and the helium production rate is 298 appm/year. The implications of such high production rates on the materials in UWMAK-I will be discussed shortly (see also, reference 22). However, a relevant point here is to note that a serious problem in this area is the paucity of gas producing reaction cross sections themselves. This has been pointed out in a recent comparative study of various fusion reactor blanket designs,[23] and also discussed by Barschall.[24] In view of the magnitude of the gas production rates and given the relevance of such information to radiation damage[22] in fusion systems, experiments are required to obtain this data.

5. Radiation Damage

The effects of radiation on the materials in fusion reactors has been recognized for some time as perhaps the major technological uncertainly aside from the plasma physics itself. The radiation damage studies for UWMAK-I have allowed us to quantify the extent of certain types of damage and to point to areas of large uncertainty. The major information from the UWMAK-I study was reported in References 1 & 2. These included information on void swelling, gas bubble swelling, irradiation effects on yield strength, transmutation effects, loss of ductility and surface effects.

The most severe limitation on the lifetime of the first wall of the UWMAK-I reactor is the loss of uniform ductility below a design value of 0.5%. This embrittlement is the result of two mechanisms; helium embrittlement and matrix hardening.[25-28] Helium embrittlement results from the generation of helium gas within the matrix by (n,α) reactions and the collection of helium gas bubbles at grain boundaries. These bubbles can grow under the action of applied stress by diffusional properties until the grain boundaries fail. Ultimately, the bubbles may lead to failure by assisting crack nucleation at second phase particles within the boundaries. This effect is dominant at high temperatures ($>0.5T_m$ or>650°C in steel). Matrix hardening can lead to premature failure by forcing most of the deformation to be absorbed by the grain boundaries. Such an effect results in excessive grain boundary shearing which leads to high stresses and subsequent failures initiated at grain boundary triple points. This effect occurs mainly at medium to low temperatures ($<0.4\ T_m$ or <500°C).

It is important to note that when helium is present at boundaries and irradiation hardening is significant, ($0.4\ T_m < T < 0.5\ T_m$), the helium embrittlement and matrix hardening can combine to produce ductility losses more severe than the two processes acting alone.[27] Under such conditions, strain is concentrated at the grain boundaries which, because of helium embrittlement, are less able to withstand shear than before irradiation. These conditions result in grain boundary failure at much smaller strains than simple matrix hardening alone.

The temperatures in UWMAK-I have been limited to <500°C due to excessive lithium corrosion at higher temperatures. Thus, most of the ductility loss can be assumed to be due to matrix hardening and current fast reactor data can be used. Application of this data[25-28] to UWMAK-I conditions reveals that after one year of irradiation, the first few centimeters of the blanket will have reached the 0.5% uniform elongation design limit. Unfortunately within the next year, the ductility of the first 20 cm of the blanket (the entire heat removal cell region) will fall below this limit. After 3 years, 30 cm of the blanket will have uniform elongation values less than 0.5%. The conclusion is that the heat removal cells (the first 20 cm of the UWMAK blanket) will have to be removed approximately every 2 years due to this radiation induced embrittlement. The recycling of these cells means almost 246,000 kg of radioactive solid waste will have to be disposed of per year.

It must be emphasized that the above analysis does not take into account the possible synergistic effects of large helium concentration and displacement damage. We feel that inclusion of such effects would tend to reduce the wall lifetime to even lower values. Experimental information in this area is absolutely vital to the safe and economical operation of fusion reactors, regardless of the material of construction.

The swelling behaviour of the blanket materials has also been analized in References 1 & 2. A design limit of 10% has been used although future reactor designs will have to be closely scrutinized for more precise limits. Both solution treated (ST) and 20% cold worked (CW) 316 SS was investigated. It was found that 20% cold worked 316 SS would undergo much less swelling than ST 316 SS during the 2 year first wall lifetime. The calculated valves were 7.9% for ST 316 SS and 0.25% for CW 316 SS. However even 20% CW 316 SS swells as much as 10% at 500°C in 6 years of operation so that if the heat cells didn't have to be replaced because of embrittlement they would have to be replaced due to swelling.

The production of helium at the rate of 298 appm/yr and its subsequent collection into bubbles will cause less than 1% swelling in 2 years and less than 7% in 30 years. However, the effects of such high helium concentrations on the void swelling phenomenn are unknown. It is extremely important to assess the combined effect of high displacement damage (>30 dpa) and high helium contents (>600 appm) on the ductility of 316 SS at 300-650°C. Alloy development programs may have to be initiated to find more ductile but still readily available CTR structural materials if one wishes to construct economic CTR's.

The first wall of fusion reactors will be subjected to intense fluxes of both neutrons and high energy ions during operation, as shown in Reference 2. In UWMAK, the major contribution to the wall erosion rates is the sputtering caused by the 14 MeV neutrons. This effect accounts for 75% of the total wall erosion rate which is 0.22 mm/year. Other major mechanisms and their fractional importance are (D,T) sputtering (5%), self ion sputtering (10%) and back scattered neutron sputtering (10%). The erosion rates due to charged particle bombardment are calculated assuming the divertor is 90% efficient in collecting particles which diffuse out of the plasma before they strike the first wall. However, since the neutrons constitute ~75% of the wall erosion rate, the removal of the divertor would mean that the thinning rate would increase by only a factor of 3 to 4. Clearly, if no catastrophic self ion build up occurs, one could operate the first wall at this increased wall erosion rate for a few years. However, if we wish to limit the wall erosion due to particle bombardment to less than 1 mm (the maximum $\Delta \ell$ allowed by stress in 316 SS is ~3.3 mm including corrosion loss) then there is a 5 year limit at UWMAK-I operating conditions (using a 90% efficient divertor).

Transmutation effects on the properties of 316 SS appear to be minor provided the walls are changed every 2 years. A thirty year exposure would have the major effects of increasing the Mn content to >6%, increasing the Ti to ~0.2% and the V to ~0.9% while reducing the Fe content from 62% to 58%, and Ni from 14% to 13%.

As a final point we have considered the effects of radiation damage on the superconducting magnets. This damage can manifest itself as a reduction in the critical current of the superconductor and as increased resistance of the copper stabilizer. We have found that radiation damage of the superconducting material (NbTi) does not appear to be serious over the lifetime of the plant. There will probably be less than a 1°K drop in T_c and <2% change in J_c of the superconductor. Radiation in the form of neutrons and gamma rays does not appear to be a problem for the Mylar super-insulation in the magnets. The most severe problem for the superconducting magnets is the increase in resistivity for the Cu stabilizer. However, by increasing the Cu/Superconductor ratio and periodic annealing to room temperature, this problem can be solved. Therefore, in designing the shield for a system like UWMAK, it is the heat load in the magnet, and not radiation damage, which governs the design.

The major integral wall loading limitations for UWMAK-I are listed in table 2. They are expressed in terms of MW-years/m^2. This table shows that ductility will limit the first wall to 2MW-years/m^2 which, to a first approximation, could be obtained by a 2 year exposure to a 1 MW/m^2 wall loading or a 1 year exposure to a 2 MW/m^2 wall loading, etc. Table 2 also shows that the present limits of 10% swelling and 1 mm wall erosion place equally restrictive limits the UWMAK-I first wall. Other radiation damage mechanisms are considerably less restrictive and are discussed in more detail in Reference 1.

Table 2

Major Radiation Damage Limitations for the UWMAK-I First Wall

Phenomena	Integral Neutron Wall Loading Limit (years)
Embrittlement (uniform elongation >0.5%)	2
Swelling (<10%)	6
Surface Erosion (<1 mm)	6

6. Blanket and Shield Design

One of the most difficult features of a toroidal reactor is the repair and maintenance of the fusion blanket. This statement automatically assumes that there will be random failure of the reactor components aside from the scheduled maintenance of first wall. The UWMAK-I has been designed with the philosophy that the reactor should be segmented (12 modules in the present case) so that any substantial repair is done in hot cells outside the reactor itself. This approach is necessary because the radiation levels in the reactor after shutdown are in the neighborhood of several megarads. It also means that if spare segments are available, the down time of the reactor is only governed by the time it takes to remove and reinsert one module. An added feature is that malfunctions in the toroidal field coils could be repaired outside the reactor, something which would be very difficult to do in the inner part of the torus. A major disadvantage of this approach is that rather massive components (~several thousand metric tonnes) must be transported over distances of ~100 meters to and from hot cell facilities. Considerable detail on how such an operation is to be performed in UWMAK can be found in Reference 1.

An important feature of the UWMAK-I design is the flow patterns of the lithium in and out of high magnetic field regions.[29] Since the pressure drop is proportional to $\bar{v} \times \bar{B}$, it is obvious that both the magnitude and direction of the fluid can be utilized to reduce pumping losses. Large flow channels (~3-4 cm in diameter) are provided to reduce the velocity to ~4 cm/sec. The flow is also predominantly perpendicular to the main toroidal field and the flow pattern is such that eddy current losses are minimized. The electrical pumping power required for all of the Li coolant is only 22 MW_e or 1.5% of the plant output. This number should be compared to the total electrical auxillary requirement of ~125 MW_e for UWMAK-I. Such an analysis shows that with proper care, liquid metals can be used economically in magnetically confined fusion reactors.

Perhaps one of the most severe limitations of the UWMAK system is the high corrosion rates between 316 SS and lithium at temperatures >500°C.[30] It has been found that as much as 1500-2500 kg of steel may be dissolved into the lithium per year. Such a large amount of corrosion product could represent a problem in the fouling of heat transfer surfaces, the deactivation of tritium extraction beds or the actual plugging of tubes in heat exchangers. Even more serious is the fact that a large amount (~10%) of the corrosion product will come from the first wall and therefore is highly radioactive. High radiation levels in loops outside the reactor may severely hamper normal maintenance. It must be noted that even those materials which exhibit good corrosion resistance may cause high radioactivity levels in the coolant. Such activity could come from neutron sputtering of the first wall into the coolant. It has been calculated, strictly on the basis of high 14 MeV sputtering yields[15], that this mechanism could contribute up to 50% as much radioactivity as chemical corrosion alone. This needs to be investigated experimentally.

Finally, there is clearly room for ingenuity in the blanket and shield design. Clever assembly and disassembly schemes, methods of detecting leaks once they form, and mechanisms for shutting the reactor down quickly and safely are required.

7. Tritium

Tritium as a fuel for fusion reactors clearly implies that detailed consideration must be given to tritium handling, extraction, and expected leakage rates. Some tritium release is probably inevitable during operation of a fusion reactor and the principal escape routes in UWMAK are: permeation through the shield into the reactor hall; permeation into the helium cooling system of the magnet shield; permeation through headers and piping into the reactor hall and heat exchanger cells; leaks through any valves or joints; and permeation through the heat exchanger into the steam system. The escape route which appears most difficult to control is permeation through the heat exchangers into the steam system.

Tritium recovery from the steam would be expensive but the leakage rate from the system will be small compared to the production rate. As reported in Reference 2, the tritium release rate from UWMAK would be 10.1 Ci/day and it is released almost entirely via the primary lithium coolant through the heat exchangers to the steam cycle. We would point out, however, that this 10.1 Ci/day would be diluted in 12000 gpm blowdown from the condenser and would be discharged into a large body of water.(31) The resulting tritium concentration is about 1.5 x 10^{-4} $\mu Ci/cm^3$ or about 2% of the regulatory limit from 10 CFR 20, Table II, Column 2, appendix B, and would result in a dose of 2 x 10^{-4} rem/week to a person obtaining a normal daily intake of water from this source.

The tritium production rate in UWMAK-I is 1.05 kg/day and the buildup of tritium in the lithium is 1.2 wt ppm/day. In contrast, tritium diffusing from the plasma at a rate of 7.4 kg/day is primarily collected by the lithium collectors of the divertor. Included with this tritium is about 5 kg/day of deuterium. The importance of this is that significantly more tritium must be handled by the divertor system than is bred in the primary coolant system. The steady state inventory of tritium in the primary lithium and extraction beds is 9.7 kg while the inventory associated with the divertor and its extraction beds is 3.8 kg. When one includes the tritium inventory in reserve, in the distillation system, and in the fueling system, the total inventory is about 30 kg. Therefore, even if the system did not use lithium cooling and did not breed tritium, the tritium inventory would be high, perhaps 15 to 20 kg.

Finally, the use of yttrium extraction beds(32) appears to be satisfactory based on thermodynamic considerations. The flow rate through the beds and the construction of the beds are reasonable. However, before such a system could be operated, it is necessary to verify the proposed equilibria at the temperatures of interest and to determine the effects of dissolved oxygen and nitrogen on the chemical reactions and the effects of corrosion products on the efficiency of such a system.

8. Radioactivity and Afterheat

The presence of 14 MeV neutrons as a principal fusion reaction product implies that fusion reactors based on the D-T cycle will have a large amount of induced radioactivity and that an afterheat will be associated with this activity. However, the generation of radioactive isotopes is not a serious problem if they can be contained, have short half lives, and are not particularly harmful to man. In particular, the induced activity in the structural materials of fusion systems will constitute a solid, rather than a gaseous, waste and the low afterheat makes release of this activity to the environment highly improbable.

The radioactivity for UWMAK is characterized by a rapid buildup to levels of about 10^9 Ci in less than 1 hour (regardless of the structural material, 316 SS, Nb - 1Zr, V-20Ti, or aluminum). This level is roughly maintained during operation. As reported in Reference 2 and elsewhere,(33) the decay of this activity is relatively slow, in 316 SS, dropping successively to 5 x 10^8 Ci in 2 years, 5x10^6 in 20 years, and 3.5 x 10^2 Ci in 200 years. The early time decay is governed primarily by ^{55}Fe ($t_{1/2}$ = 2.7 years). The impact of these results is that remote maintance and handling will be required even after modules of the reactor are removed for servicing or other purposes. Also, since radiation induced embrittlement limits first wall life to about 2 years, one is faced with removal of an average of 246,000 kg per year of solid radioactive waste. Compaction and on-site storage appears to be feasible since the afterheat is relatively low, namely, 29 MW_t at shutdown after 2 years of operation. Vogelsang et.al.(33) have also compared fusion with advanced fission systems(34) on the basis of Biological Hazard Potential (BHP). They conclude that a fusion system such as UWMAK-I will have overall BHP indices that are a factor of 100 to 1000 lower than an advanced fission system. Thus, it appears that fusion systems like UWMAK can offer a real advantage in terms of radioactivity and afterheat, compared with advanced fission reactors.

The total afterheat at shutdown in the 316 SS blanket is 29 MW_{th} after 2 years of operation and 33 megawatts after 30 years. Almost 50% of this afterheat is generated with the heat removal cells and this drops by a factor of 30 in 20 years. The maximum rate of temperature increase in a loss of flow accident was calculated to be 0.1°C sec^{-1} but it appears to be more like 0.01°C sec^{-1} when convection and conduction forms of heat loss are considered. It is concluded that afterheat represents no serious problem even in the event of a loss of flow accident in UWMAK-I, whether or not the plasma remains on or is quenched. However, a loss of coolant accident without plasma quench can cause more rapid first wall temperature increases. This is being investigated. A detailed analysis of the loss of coolant accident with the plasma operating should be made to insure that major damage will not occur in the CTR blanket.

The total afterheat in UWMAK-I after shutdown and 10 years of operation is not particularly sensitive to blanket material. The value at shutdown varies from 0.55% to 0.1% of the operating power for the three metals studied here (316 SS, V-20Ti, Nb-1Zr). However, the afterheat in V-20Ti decays faster than the steel and Nb alloy and is at half the shutdown level after a month of decay.

From such considerations, it is also concluded that methods should be developed to collect, store and ship several metric tonnes per year of radioactive corrosion products from the stainless steel. Also more detailed calculations are required on the radioactivity levels at various maintenance points outside the reactor due to the deposition of 316 SS corrosion products. Finally, long term solutions to the concentration and storage of spent reactor components must be addressed. In particular, methods of handling ~250 metric tonnes of 316 SS heat removal cells per year should be studied. On site storage may be feasible but central burial facilities should be investigated.

9. Resources

A preliminary assessment of the availability of certain metals that appear important to a fusion power industry based on UWMAK type reactors has been made.[35] The results are summarized in Table 3. We have purposely divided the reserve estimates into U.S. and external to U.S. since self-reliability in a commodity as basic as electricity has become rather crucial. The conclusion is that for a UWMAK type system, the U.S. reserves of Ni, Cr, and Li are less than would be required for a 10^6 MWe economy (~1/3 the projected electrical generation capacity in the year 2020) and that lead is only about three times the requirement. It is important to note that the lithium reserves in a system where lithium is used as a coolant do not greatly exceed the requirement. However, if lithium is only used for breeding, the reserves far exceed the requirement. At three times present prices, all metals except Cr exceed the requirement for a 10^6 MWe economy. Since the reactor itself only constitutes ~1/3 of the total plant cost, a price increase of 3 times the present value only contributes ~70% to the cost of electricity from fusion. This percentage is further reduced by the fact that more than half the cost of reactor materials is in fabrication and assembly. Therefore it may be possible to use materials which cost as much as three times present prices and still not increase the cost of electricity by more than 35%. The use of beryllium has been frequently discussed as a means for improving energy production in fusion systems. However, given the very low reserves of Be in the context of the requirements of a 10^6 MWe economy, the use of large amounts of Be either in solid form in UWMAK or in FLIBE does not appear to be practical.

10. Summary

We have presented in this paper a discussion of the major conclusion of the UWMAK-I conceptual design[1,2] and treated in some detail the implications of these results. Overall, most of the technological problems posed by Tokamak fusion reactors appear to be solvable within reasonable extensions of existing technology. The two general problem areas which cannot be categorized this way and which have large remaining uncertainties relate to materials, particularly first wall problems, and plasma physics & plasma technology, including scaling, impurity control, and fueling. A list of problem areas is given in Table 4 which fall in these two general categories. We would conclude, therefore, that while many of the identified technological problems appear solvable, major advances are required in the state of plasma physics and certain reactor technology problems before a reactor such as UWMAK-I could be built.

Table 3

Resource Requirements and Availability

(Metric Megatons-Based on UWMAK in Year 2000)

Metal	Requirement For 10^6MWe	U.S. Reserves @ Present Prices	U.S. Reserves @ 3x	ex-U.S. Reserves @ Present Prices	ex-U.S. Reserves @ 3x
Fe	15	8500	$>10^5$	170×10^3	Large
Ni	1.5	.14	14.4	24	922
Cr	1.9	0	1.2	370	890
Li	0.95	.8	2.7	.2	.2
Cu	3.25	20?	>100	>100	320
Pb	11.0	35	>100	50	>500
Be	.12(for Flibe system)	.018	.018	?	.03

Table 4

Areas Within Extendable Technology	Areas With Large Uncertainties
. Multi-Ampere, Megawatt Neutral Beams	. Plasma Startup and Scaling
. Lithium-Stainless Steel Systems	. Impurity Control, Divertors
. Tritium Breeding, Extraction and Leakage	. Fueling, Long Burn Times
. Modular Design; System Dis-assembly	. First Wall Life; Materials
. Large NbTi S/C Magnets	. Large Nb_3Sn S/C Magnets
. Energy Storage and Transfer	. Unstable Magnets
. Afterheat and Radioactivity	
. Power Cycle	

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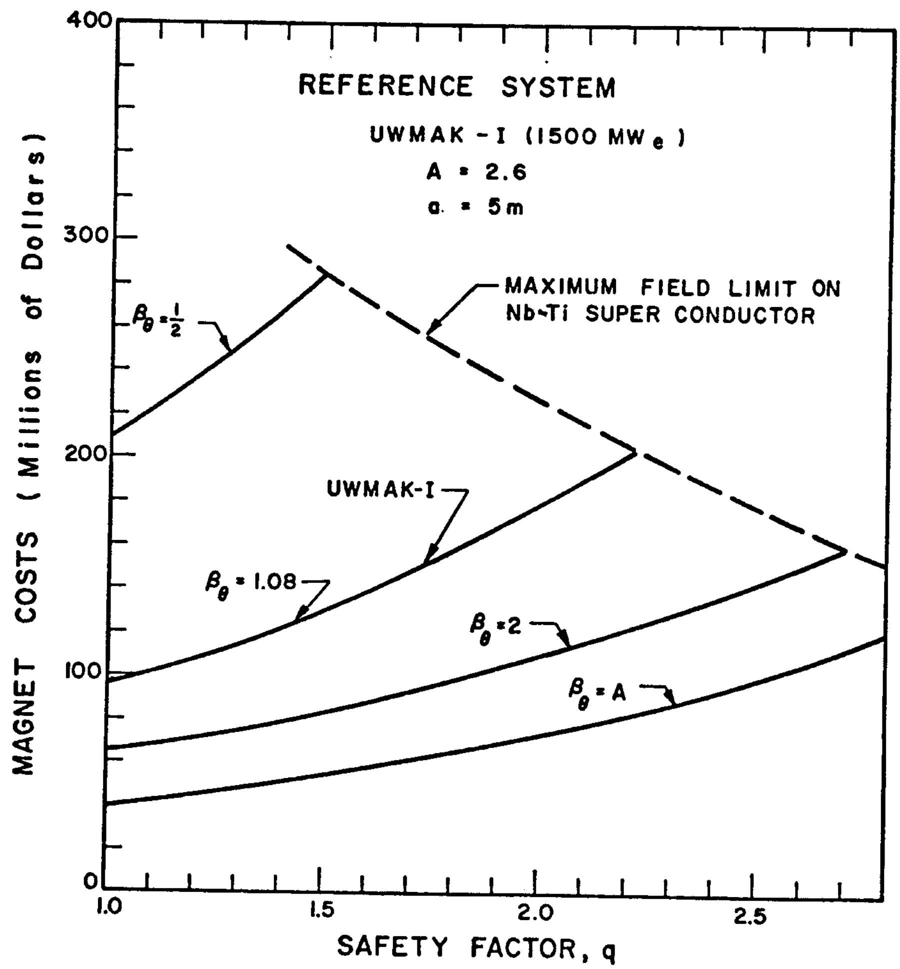

Figure 1. The scaling of magnet costs (toroidal, transformer, and divertor field coils) as a function of the safety factor, q, for different values of poloidal beta.

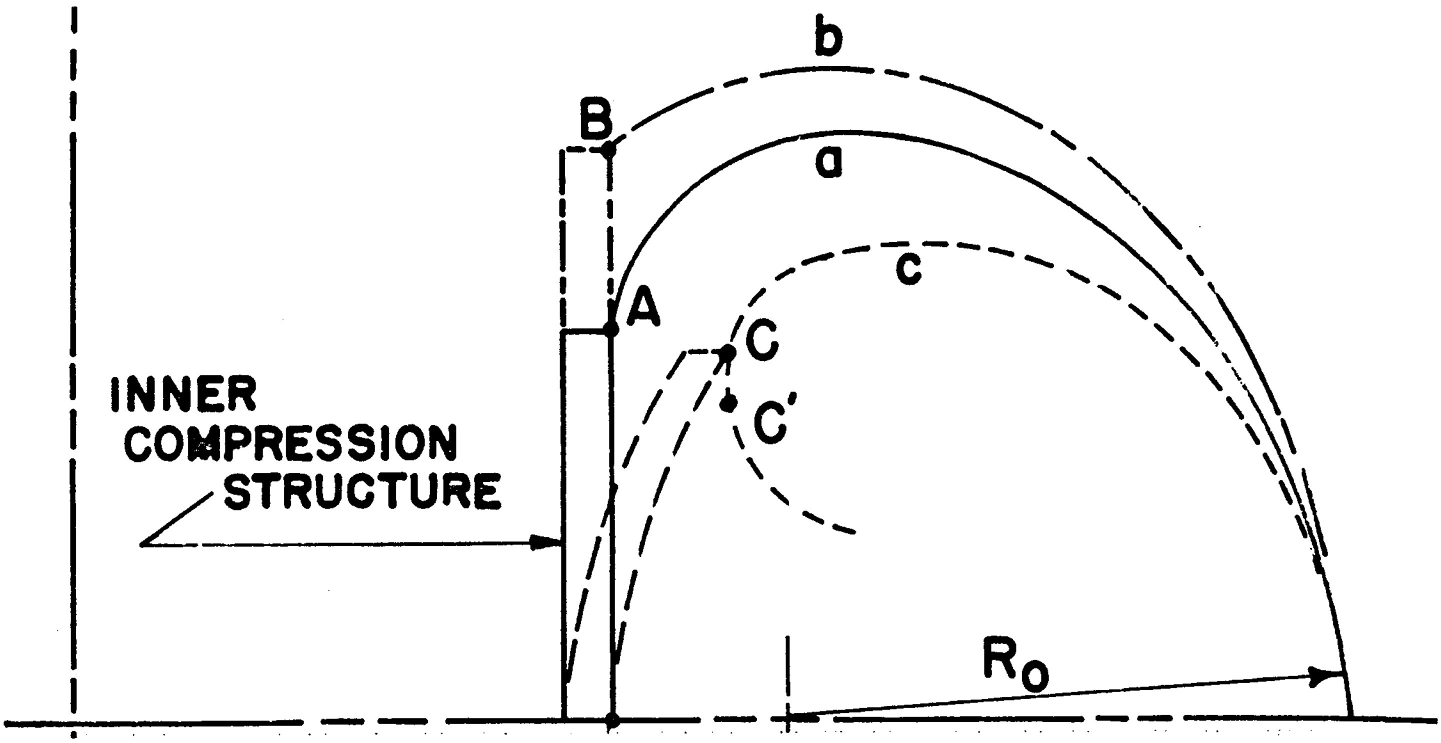

Figure 2. Three typical "D" Shaped Magnet Designs. Only curve 'a' yields constant tension for all portions of the conductor.

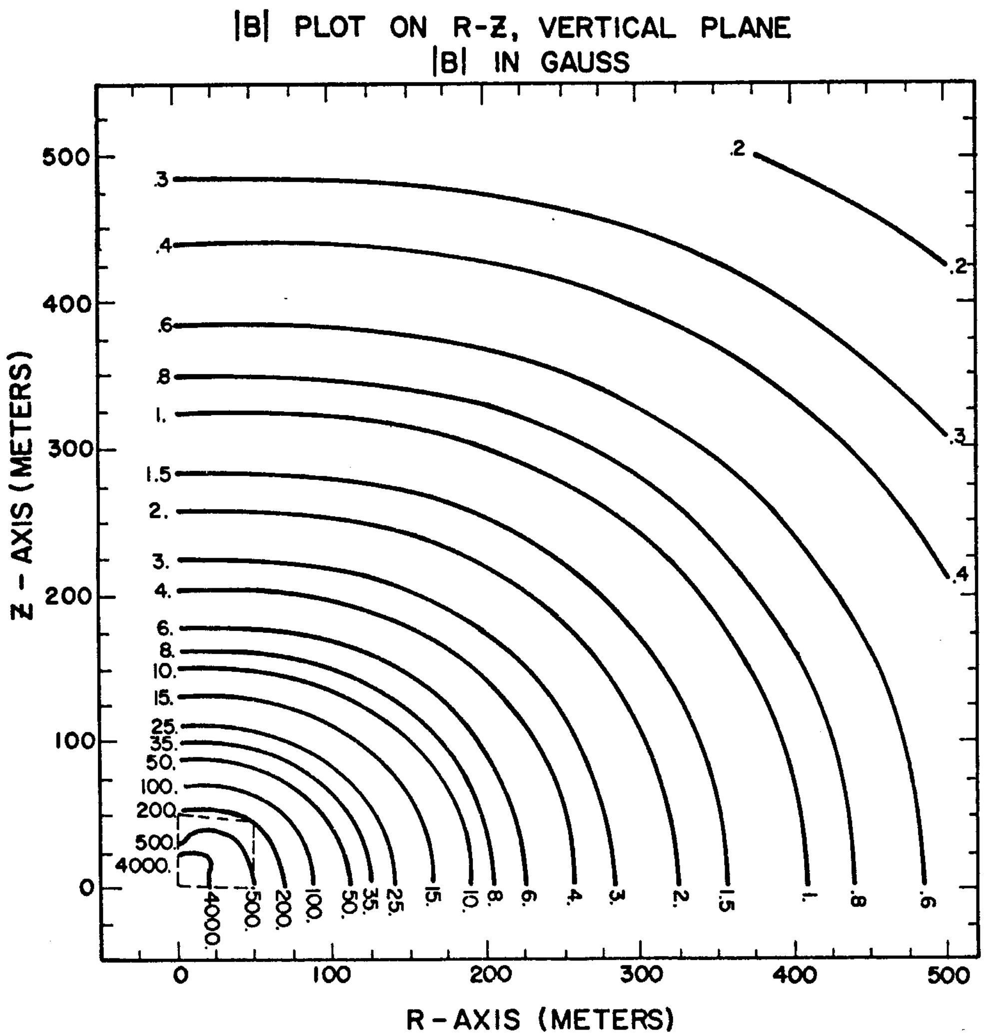

Figure 3. Mapping of mod B of poloidal field on a vertical plane. The toroidal field is negligible except within 100 meters. The dashed lines indicate the reactor building size.