Fusion Power System - Technology and Engineering Considerations*

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INTRODUCTION

Looking schematically at a thermal electric power plant cycle, the fusion thermal energy source replaces, for example, the fossil fuel or fission energy source. In all three processes energy is released as heat which in turn is used to raise the temperature of a working fluid which may or may not be water. Assuming that we generate steam, directly or indirectly, to cause a turbine to rotate and turn a generator to produce electricity, we see that in this context the fusion reactor does not change the basic power plant cycle. What is, of course, fundamentally different compared with the previous sources is the fusion energy source.

In magnetically confined fusion and the deuterium-tritium fuel cycle, a large fraction of fusion energy will be released as high energy neutrons which must be slowed down in a relatively thick structure, a blanket, which surrounds the plasma. The kinetic energy of the neutrons is converted to high temperature heat which, as we noted, must be removed from the blanket by a coolant system which in turn is converted to electricity in a standard thermal cycle.

In addition to blanket technology and the power conversion system, fusion technology includes the magnet system, vacuum and wall technology, high energy neutral beams for plasma heating as well as tritium technology. The current presentation simply focuses on the blanket which itself poses a formidable technology challenge. Several general problems are discussed and with a specific blanket design, we show by example what technology considerations must be accounted for, at least as far as the blanket is concerned. That it is complete is by no means the case, but it should provide a beginning to understand a few of the problems. Lastly, this represents a particular point of view and does not attempt to provide a general overview to particular blanket designs.

I. BLANKET REQUIREMENTS

In all fusion reactors using the deuterium-tritium (D-T) fuel cycle, a large fraction (~80%) of the fusion energy will be released as ~14 MeV neutrons. These neutrons must be slowed down in a relatively thick structure, the blanket, surrounding the plasma, (for reference [1] see Figure 1, the cylindrical cans constituting the blanket; reference 1 is a design study of several aspects of a fusion reactor and power plant) and their kinetic energy converted to high temperature heat. In general this heat must be continuously removed from the

blanket by a coolant system, although a discontinuous blanket coolant system has been proposed by Powell [2]. The variant is the pulsed graphite blanket. There are no coolant tubes incorporated in the graphite structure. The reactor is operated in the pulsed mode with no cooling while the plasma burns. After an extended burn, typically 10 minutes in length, the graphite blanket is quickly cooled by direct contact with helium. The plasma volume is pumped down and a new burn started. Finally, a portion of this heat is converted to electricity in a standard thermal cycle, i.e., a steam or gas turbine cycle.

In addition to being a good moderator of neutrons with sufficiently good heat transfer characteristics for efficient heat removal, the blanket must be a breeder of tritium. For the D-T fuel cycle, it is necessary to have lithium present in the blanket to breed tritium to replace that consumed in the plasma. Tritium production occurs via the neutron-induced reactions, ${}^6\text{Li}(n,\alpha)$ and ${}^7\text{Li}(n,n'\alpha)$, that is, known sources of tritium are the isotopes ${}^6\text{Li}$ and ${}^7\text{Li}$ via the neutron induced reactions:

Lithium may be present in the blanket in liquid metal form, which may also be used as the blanket coolant, molten or fused salt, a mixture of lithium-beryllium-fluoride (Flibe) or in solid form, such as lithium aluminate.

The blanket also serves as the first stage of neutron + gamma ($n+\gamma$) shielding of the magnet coils (see Figure 1) from excessive radiation damage and nuclear heating.

In summary the primary twofold purpose of the blanket is:

- 1) to convert the kinetic energy of the 14 MeV neutrons to heat;
- and
- 2) breed tritium from lithium.

II. BLANKET ENVIRONMENT

The fast 14 MeV neutrons seriously activate many candidate blanket materials. Moreover, the radiation damage resulting from these fast neutrons is a serious problem and may necessitate frequent replacement of the blanket. Thus, it is desirable that the blanket be made from materials that exhibit relatively

little or no residual radioactivity. This not only greatly eases repair, maintenance, and replacement operations, but also greatly reduces the radioactivity of damaged materials that must be handled and stored. The lifetime of a blanket for a fusion reactor operating on D-T fuel is not yet well defined, but it is clear that lifetimes will be at most a few years, rather than the 30 years or so desired for commercial power plants. This assumes that no failure occurs such as a leak in the blanket which may necessitate earlier removal of a blanket module. Thus the entire blanket or individual blanket modules must be replaced a number of times during the life of a fusion plant. What is important is that this must be done rapidly if plant utilization factors are to be reasonably high. A recent study by Mitchell and Hollis [3] is a good source for such considerations.

III. BLANKET MATERIALS

Having placed in context the blanket requirements and, briefly, the blanket environment, the next question we might ask ourselves is out of what materials do we construct the blanket? In addition to activation there are, of course, a number of other factors which impinge on blanket material choice. For example, there are temperature considerations. In general, high temperatures imply higher thermal efficiencies, less waste, more economy. Also high temperatures may imply materials that can retain their strength at high temperatures.

While these comments may be true in general they do not necessarily preclude the use of materials with relatively lower temperature capability. The element of design also enters in. A case in point is the Brookhaven aluminum [1] and SAP (sintered aluminum product) [4] blanket designs. While the allowable temperature of the structural shell is indeed lower than that of a comparable stainless steel shell, the bulk of the high temperature heat is extracted not by the coolant of the shell but by a separate coolant which passes through the hotter breeding-moderator region. These temperature limits are considerably higher than the shell.

Returning to high temperature materials, refractory metals such as niobium, molybdenum and vanadium are high temperature metals. For example, the useful temperature of niobium is 1500°C . On the other hand, there is a limited niobium industry and fabrication technology which would be required for reactor blanket components. The same is true of vanadium. In the case of molybdenum there is an industry for fabricating molybdenum, but it is notorious for unreliable welds. In summary it appears that the refractory metals are not yet available for large-scale construction [5].

On the other hand, stainless steel is available, is well known, strong, workable and weldable. Its useful temperature is on the order of 600°C , which is lower than the refractory metals.

Aluminum is also well known and available. The useful temperature of a wrought Al alloy (5000 series) is of the order of 200°C . A higher temperature can be achieved by using SAP (sintered aluminum product, that is a dispersion of aluminum oxide in an Al matrix). The allowable temperature is $\sim 400^{\circ}\text{C}$ and the thermal conductivity of aluminum is significantly better than stainless steel. The attractive feature of several of the aluminum series is the relatively low residual radioactivity which the alloys exhibit.

Ceramic materials such as graphite and silicon carbide offer allowable temperatures greater than 2000°C and 1500°C , respectively. In general the appeal of the ceramic materials is: a) low residual radioactivity; b) low atomic number for low plasma impurity effects, i.e., the low atomic number is especially significant for first wall applications in minimizing plasma energy losses from wall derived plasma impurities; 3) very high temperature capability for potentially high thermal efficiency; and d) abundant and inexpensive raw material supply. The ceramic nature poses special problems in engineering design beyond those normally encountered with ductile metal materials.

In a later section, First Wall Heat Load, several physical phenomena are identified which also impinge on the choice of the blanket material, in particular, the first wall.

IV. TRITIUM BREEDING

If lithium is present as a liquid metal or fused salt, blanket materials that resist corrosion have relatively high residual radioactivities (i.e., stainless steel or niobium) and many years are required for activated materials to decay.

On the other hand, if a solid form of lithium is present in the blanket, the choice of blanket constructional materials is no longer limited by lithium corrosion behavior. In fact the blanket can now have only solid components and can be cooled by an inert gas such as helium. Constructional materials with very low residual activation can then be used. The use of solid lithium compounds to breed tritium in fusion reactor blankets was proposed at BNL several years ago [6] and has been examined in more detail since then [7].

In summary solid lithium compounds have the following three important advantages for fusion blankets: a) elimination of the corrosion of structural materials by blanket coolants; b) reduction of the tritium inventory in the blanket by an order of magnitude or more; and c) reduction of the residual long term blanket activation by many orders of magnitude.

The first advantage results from using solid lithium compounds with gaseous blanket coolants, i.e., helium. With an all solid blanket and gas coolant, the possibility of corrosion can be virtually eliminated (except perhaps for corrosion of refractory metal structures by impurities in the coolant gas).

The second advantage results if the solid lithium compound is selected to have very low retentivity for tritium. For example, compounds can be selected to give mean holding times of only a few minutes for the bred tritium. Total blanket tritium inventories for a 1000 MW(e) reactor can then be kept at $\sim 10^4$ curies, or almost two orders of magnitude less than possible with liquid metal or salts.

The third advantage results from selection of blanket structural materials, e.g., Al, C, or SiC, with very low residual activation. These materials do not appear compatible with liquid lithium metal or salts. Residual activations of ≤ 1 curie/MW(e) are possible, compared to $\sim 10^6$ curies/MW (e) with structural materials like stainless steel or niobium.

4.1 Solid Lithium Compounds for Tritium Breeding. There are dozens of solid lithium alloys (binary and ternary) and compounds with melting points high enough to be suitable for fusion reactor blankets. Selection of an optimum alloy or compound will depend on many factors besides the physical properties of the solid such as compatibility with choice of neutron multiplier and/or moderator materials, blanket coolants, blanket temperatures, etc.

The first criterion in the selection of candidate solids for the BNL designs is that they do not acquire long-lived induced radioactivity under blanket conditions since a primary goal of the solid-blanket concept is to achieve a system with a low inventory of radioactive, and therefore, hazardous, material. This goal also requires, and the concept postulates, that bred tritium will be removed from the blanket after a very short average residence time.

Of concern in choosing a solid breeder is that the lithium compound or alloy will become, as a consequence of sintering, a coherent mass due to heating without melting, thereby resulting in unacceptably large tritium inventories in the blanket. Recent experiments [7] at Brookhaven have been done to find out whether tritium could indeed be moved from a lithium-bearing solid of suitable nuclear properties at reasonable rates under practically attainable thermal conditions. Of the lithium-bearing solids investigated, lithium aluminate (LiAlO_2) has been

identified as a promising candidate for breeding tritium and is compatible with the overall BNL design concepts. The tritium diffuses out of the LiAlO_2 quite readily and can be cold-trapped from the helium stream. In other words, sintering does not appear to pose a problem at least within the context of the experiments performed.

The attributes of LiAlO_2 are that: a) it has a very high melting point (1900°C) permitting high temperature operation of the blanket; b) very low tritium retentivity and residual activations; and c) is stable and inert.

Of interest in the breeding blanket is that the tritium breeding ratio, that is, the number of tritons produced in the blanket per fusion neutron or equivalently, per D-T reaction, be greater than one. In order to have net tritium breeding the solid lithium alloy or compound must either be principally lithium or lithium alloy plus a neutron multiplier incorporated in the blanket to provide additional neutrons for those lost in nonuseful capture. LiAlO_2 does not have a high enough Li content to breed without incorporating some type of neutron multiplier, i.e., Be or Be compounds or Pb or Pb compounds, in the blanket.

4.2 Modes of Tritium Release in Solid Lithium Blankets. There are three possible release modes for the tritium that is bred in the solid lithium alloy or compound: a) diffusion from the solid into the main He coolant stream; b) diffusion from the solid into a separate He processing stream, which is essentially isolated from the main He coolant stream (a small amount of mixing would probably be tolerated); and c) diffusion from the solid into the vacuum region between the first wall and the plasma.

In modes A and B, the bred tritium would be recovered from the He streams by some form of trapping. The various alternative trapping methods are discussed in [7]. In mode C the bred tritium would mix with the escaping plasma, and would be recovered either from the divertor exhaust or from the reactor vacuum pumps.

Figure 2 shows these different release modes for a modular blanket concept. All 3 release modes are feasible with enriched lithium alloy or compound modular blankets. However, with aluminum modular shells (shown in Figure 2), the module diameter must be reasonably large, e.g., 0.3-0.6 meter, to effectively thermally insulate the cool aluminum structure from the hot interior. This is necessary to achieve good overall power cycle efficiency, since the hot interior receives ~70% of the fusion energy. If the entire blanket had to operate at the temperature allowed for the aluminum shell, the resultant overall cycle efficiency would

be unacceptably low. The minimum module diameter restriction for aluminum makes modes B and C, in which the lithium compound must be localized, unattractive.

The large module diameter results in relatively high flux disadvantage factors and inefficient use of neutrons. A further drawback of mode C for aluminum structures is that the 4.8 MeV neutron absorption energy in Li^6 is deposited in the cool aluminum shell instead of the hot interior, as in mode A. This substantially reduces the overall power cycle efficiency.

With structures other than Al, i.e., SS, Nb, V, Mo, the blanket modules can be operated with a much hotter structure. Thermal insulation is not required, and modes B and C are as neutronically and energy efficient as mode A. Mode B remains more mechanically complex than the other two modes, however. The choice between mode A and C for these blankets must depend on more detailed design considerations.

With natural enrichment solid Li alloys or compounds (e.g., LiAl , Li_4Si , etc.), the solid Li must fill virtually the entire module shell, since Li^7 reactions are very important to obtain good breeding ratios. Almost all of the fusion energy is then deposited in the solid Li compound, and modes B and C do not appear to be feasible. This holds for all choices of blanket structure, including Al, SS, Mo, V and Nb. Mode A release is quite acceptable, however.

With high temperature graphite blankets (Figure 3) all 3 release modes appear feasible, with the preferred mode being either C or A. The graphite blanket is essentially a thick (30-50 cm) structure of graphite with internal slots to permit radiant heat transfer to readily spread thermal energy deposited by neutrons and gammas through the blanket. Internal coolant tubes receive this energy by thermal radiation from the hot graphite. These coolant tubes are protected against radiation damage by the graphite structure. In mode C release, solid LiAlO_2 pellets can be incorporated into holes in the cooler graphite regions. The bred tritium will then flow out into the vacuum region by way of small spaces between the graphite blocks. In modes A and B release, the solid Li compound would be placed into some or all of the coolant tubes, and the bred tritium would enter the helium streams. Mode C would appear to be optimum for this type of blanket. In a variant of this blanket, the pulsed graphite blanket [2], Figure 4, there are no coolant tubes incorporated in the graphite structure. The reactor is operated in the pulsed mode with no cooling while the plasma burns. After an extended burn, typically 10 minutes in length, the graphite blanket temperature has increased by $\sim 500^\circ\text{C}$, and the blanket is quickly cooled (typically 2 minutes) by direct contact with helium at ~ 2 atm. The plasma volume is then

quickly pumped down from 2 atm to a few microns in a short time (typically 2 minutes) and a new burn is started. Here mode C release is the only possible mode. A small fraction of the bred tritium enters the pulsed He coolant flow, and can be recovered by a small process side stream.

V. HEAT REMOVAL PROBLEMS

5.1 Overall Description. To obtain electrical power from the D-T fusion reaction, a standard thermal cycle, i.e., a steam or gas turbine cycle must be used. Since the thermodynamic efficiency of such a cycle is governed by the operating temperature of the heat source, the problems of heat removal from a CTR are of considerable importance. The choice of the coolant and flow network impinges on such aspects as capital cost and power-plant efficiency, and as we have discussed already tritium breeding, radioactivity, material and structural criteria, and the reactor lifetime.

It is generally assumed in the case of a commercial reactor that to maximize the wall loading, i.e., the ratio of the maximum rated power output of the reactor divided by the torus surface area facing the plasma, is desirable since the reactor size and the capital costs are directly related. Even though such factors as radiation damage and plasma-stability restraints may strongly affect the allowable wall loading, the ability to cool the wall appears to be the major factor involved in establishing its limit. The coolant or coolants are introduced so as to absorb the energy of the neutrons released in the fusion reactions along with the energy (neutron + gamma generation) developed in the blanket with or without tritium breeding. In addition the coolant or coolants must absorb a certain percentage of the energy from the plasma, primarily in the form of radiation, which impinges on the blanket surface exposed to the plasma. The neutron and gamma energy results in volumetric heat generation within the blanket while radiation to the surface is responsible for an imposed surface heat flux.

5.2. Operating Regimes. In order to carry out a heat transfer analysis it is necessary to establish whether the phenomena is time dependent or not. In the case of commercial reactors where burn times may be on the order of 30 minutes or so it is anticipated that steady state conditions should be achieved in the blanket. By steady state it is meant that the temperature distribution does not change with time.

For our graphite blanket designs a general purpose heat conduction code was developed [8] allowing for thermal radiation exchange amongst multiple surfaces

as well as inclusion of coolant tubes. Our calculations do not assume a steady state, that is, set the time derivative, $\frac{\partial T}{\partial t}$, equal to zero. The steady state temperature distribution is generated from arbitrary initial conditions as the asymptotic solutions of the nonsteady temperature build-up in the blanket. Since the explicit method of solution is a reasonably simple time-dependent method, the flexibility of being able to achieve the transient solution, if desired, is attractive. The start-up and shut-down phases of the reactor correspond to the starting and ending transients relative to the steady state operation and must be considered in blanket analyses.

For an experimental power reactor (EPR) it has been found that a steady, periodic temperature behavior persists. Assuming plasma on-off times of 30 seconds (typical projected times for the EPR), steady, periodic temperature profiles are generated in the blanket, i.e., the integral of the transient temperature term over the period, τ ,

$$\int_0^{\tau} \frac{\partial T}{\partial t} dt$$

is equal to zero.

For commercial reactors and the pulsed graphite blanket, that is, an intermittently gas-cooled blanket, the dominant mode of operation is the transient state. In this case the integral of the transient term over a period τ does not equal zero.

5.3 First Wall Heat Load. In the design of blankets and the subsequent problems of cooling and heat removal, the first wall looms as the critical region of concern. Consequently, it is singled out for separate study and discussion. As the first material surface interfacing the plasma region, all of the emissions from a reacting plasma may intersect this wall. These include [9] ions and neutral particles, primary neutrons, X-rays (bremsstrahlung) and cyclotron radiation. In addition, scattered neutrons and gamma radiation generated in the blanket regions exterior to this first wall are also imposed on it. Finally, D-T gas surrounding the plasma may react chemically with the first wall materials. One result of these many interactions is the generation of appreciable heat which can be up to 20% of the plasma output; thus provision must be included for adequate cooling. In most analyses the above effects are usually lumped together as a radiant flux. While the incident fluxes are absorbed in a small depth of any first wall material, the usual assumption is to treat the lumped radiant flux as if it were simply deposited on the surface facing the plasma. In heat transfer analyses it is an applied heat flux boundary condition. The phenomena of sputtering and

blistering, the formation of gaseous reaction products, and material vaporization processes may represent the major sources of impurity in the plasma. Thus the first wall can be a limiting component in reactor power output due to a large power generation within it; and it may also limit plasma performance through impurity levels. As already noted, the useful first wall lifetimes in a power reactor may be severely limited through radiation damage and surface erosion. The attendant frequent replacement could unduly add to operating costs and require an excessive plant down time for replacement. Thus the design and material selection for this first wall is a vital element in the design of a fusion power reactor. In a number of designs, for example, [9], the first wall is constructed as a liner attached to the remainder of the blanket to absorb the brunt of the thermal loading. This is not the case in the BNL designs.

In addition to the above concerns with respect to the first wall, it is necessary to consider additional effects which the first wall will experience as well as potential scenarios which could be disastrous for the first wall and cause subsequent failure of the blanket as a whole. These all have a bearing on the design of the blanket, need for careful analysis, as well as heat removal and cooling aspects of the blanket.

Assuming that the plasma does not undergo any excursions to the first wall under normal operation, the first wall will be subjected to thermal cycling. This is not necessarily a problem in itself but rather the temperature excursions per pulse which the structure could be subjected to. What could occur are cracks in the first wall as a consequence of thermal stresses, the propagation of these cracks leading to the ultimate failure of the blanket.

5.4 Energy Dump. Plasma excursions and contact with the first wall is of considerable concern, in particular the problem of an energy dump. The plasma energy dump could possibly cause surface melting of the first wall. If 100% of the plasma energy in a UWMAK-I plasma were dumped to the first wall in 1 sec, the heat rate to the affected first wall would be $100 \text{ J/cm}^2 \text{ sec}$. This would raise the surface temperature by $\sim 50^\circ\text{C}$. The surface temperature rise in a water cooled shell would be somewhat less. If only a fraction of the first wall surface receives the plasma dump, the aluminum surface temperature increase will be proportionately greater. A limiting value would appear to be a dump to $\sim 20\%$ of the first wall surface area. Dumps to a smaller fraction of the first wall would result in a melting of the surface of the aluminum shell. With other first wall materials (e.g., SS, Nb, V, etc.), the dump could occur to a somewhat smaller fraction of the first wall before melting occurs, e.g., 5-10%. However, choices

as to first wall material on the basis of surface temperature rise during a plasma dump seem arbitrary, since one has no real idea as to the surface area or time scale that will be involved. Plausible scenarios can always be constructed so that portions of a metallic wall will be melted, no matter what material is used. If plasma dumps are a problem because of sudden unexpected instabilities, it probably will be necessary to line the first metallic wall with a protective layer like graphite or SiC that can ablate when the dump occurs.

In order to obtain an estimate of the temperature rise of the first wall due to a excursion which comes in contact with the wall, we consider the following analysis. We assume a semi-infinite slab exposed to a constant heat flux, q_0 . A plate exposed to convective or radiative environments always behaves as a semi-infinite solid in its early moments of heating [10] so that the assumption of a semi-infinite medium seems justified. The temperature history is given by [10].

$$T - T_0 = \frac{2q_0}{k} \sqrt{at} \operatorname{ierfc} \frac{y}{2\sqrt{at}}$$

where T_0 represents the slab temperature at $t=0$ (the assumption implying that the first wall surface temperature and temperatures at any value of y at $t=0$ are uniform; this seems to be a reasonable assumption for small y); $\operatorname{ierfc} ()$ is the complementary error function. The surface temperature rise at $y=0$ for any t is simply

$$T_s - T_0 = 1.128 \frac{q_0}{k} \sqrt{at}$$

Assuming a blanket structure made of SAP, α , the thermal diffusivity, $\approx 1 \text{ cm}^2/\text{sec}$, $k \approx 2 \text{ w/cm}^\circ\text{K}$ so that for an energy dump taking place over 1 sec and a surface temperature rise of 250°C , the heat dump $Q(= tq_0)$ would be 440 J/cm^2 (for 22% of the first wall area). In other words the 250° temperature rise will cause the surface of a He cooled SAP module (400°C operating temperature) to reach the melting point. Hence, the allowable heat dump to cause this would be 400 J/cm^2 .

For any other material "X" the allowed heat dump (no convective cooling of the first wall is assumed) will be given by

$$\frac{Q_X}{Q_{Al}} = \frac{(\Delta T_s)_X}{(\Delta T_s)_{Al}} \left(\frac{k_X}{k_{Al}} \right) \left(\frac{\alpha_{Al}}{\alpha_X} \right)^{1/2}$$

A comparison of other metallic first walls results in

Material	Q_X/Q_A	$\frac{(\Delta T_S)_X}{(\Delta T_S)_A}$	$\left(\frac{k_X}{k_A}\right)^*$	$\left(\frac{\alpha_A}{\alpha_X}\right)^{1/2}$
Al	$1=(440/440)$	$1=(250/250)$	$1=(2/2)$	$1=(1/1)^{1/2}$
SS	1.5	3.8	0.086	4.5
Nb	3.8	6	0.35	1.83
Mo	5.4	6.4	0.52	1.62

* Neglects temperature dependence of k and α . A water cooled aluminum shell at $\sim 200^\circ\text{C}$ operating temperature would be better than stainless steel, since the ratio Q_{SS}/Q_{Al} would now be ~ 0.8 .

VI. MATERIALS SELECTION FOR RADIATION SHIELDS

While it has not been explicitly stated as such, the blanket serves as the first stage of neutron and gamma shielding of the magnet coils from excessive radiation damage and nuclear heating. This can be inferred from Figure (1). In general the gamma-rays generated in the blanket because of inelastic scattering of neutrons as well as the neutrons themselves are not totally moderated in the blanket so that additional shielding of the magnet coils is required. In general cooling of this shield will be necessary. To effectively protect the magnets, the shield materials must degrade and absorb as much of the incident neutron and gamma-ray flux as possible. The materials must also be chemically compatible and capable of maintaining their structural integrity under the thermal and radiation loads imposed on them [9]. To insure attenuation of the total neutron spectrum, the shield composite must contain materials possessing varied characteristics. An inelastic scattering medium is an effective way to provide degradation of the high energy component (>3 MeV) of the spectrum. Adequate removal of the neutrons is then obtained by the presence of a moderator-absorber, in the BNL designs, boron carbide (B_4C). The structural shield material, aluminum, also attenuates the gamma-ray flux. In the recent BNL designs the inner wall of the shield forms the primary vacuum seal.

Additional shielding of the magnets may be required and the primary containment vessel, a concrete structure beyond the magnets, is also a shield to the external environment. In addition shielding of vacuum ports, beam liner, etc.,

must also be provided.

Besides the questions of radiation damage and nuclear heating of components other than the blanket, there is the attendant question of maintenance accessibility. That is, to what extent can hands-on-maintenance be performed? The choice of structural material will play an important part in determining the answer. Dosage calculations [11] for the BNL minimum-activity blankets and shields have been performed for a man standing outside the fixed shield (1 meter-thick, with Al structure and B₄C and water coolant) 15 days after reactor shutdown, assuming a 1 year reactor operation. Results indicate that it should be possible to perform hands-on-maintenance at selected areas on the outside surface of the shield. The dose to personnel would be less than 10^{-2} rem per day.

In the future, it is planned to investigate dosages due to beam and pumping ports using Monte Carlo codes. It does not seem likely that these structures will eliminate the capability for hands-on-maintenance. There will be streaming through openings in the reactor shield, but these effects can be minimized. In addition for vacuum ports the lines can be constructed of low activity structure, e.g., aluminum, and the lines can be stepped or curved to essentially eliminate activation due to streaming.

VII. AFTERHEAT

Afterheat is the release of energy which is generated after reactor shutdown due to the decay of radioactive material. In addition to normal shut-down there is concern of a loss-of-coolant accident. If the afterheat is not removed, it could damage the activated components of the reactor. This implies that afterheat generation depends strongly on the structural material chosen.

Extensive afterheat calculations have been made, for example, for the UWMAK-I fusion reactor [12] where time dependence of afterheat following shut-down is considered. The UWMAK calculation as well as others [13, 14] indicate that removal of afterheat is not likely to be a severe problem; the order of magnitude being 0.5% of the reactor thermal power at shutdown for niobium or stainless steel structures. On the other hand, the afterheat will decay relatively slowly. In contrast the afterheat of a SAP structure will decay rapidly following shutdown and within weeks would be controlled by impurities in the material. Certain choices of structural material virtually eliminate the afterheat problem. Carbon and SiC would exhibit practically no afterheat. The afterheat of a carbon or SiC structure will decay very rapidly following shutdown and within hours would

be controlled by impurities in the material. In general relative to fission reactors, the associated afterheat power densities at shutdown suggest that afterheat removal will be quantitatively less of a problem in fusion reactors [15].

Lastly, although the rate of afterheat release is low and one would rely on conduction, radiation and in some cases natural convection as loss mechanisms, there must be a sink. This should be reflected in the designs.

VIII. FUSION BLANKET DESIGN

8.1 Physical Description. To place the previous discussion in context, we consider a specific blanket design. Hopefully it will give the reader a better perspective on additional technology requirements of a fusion reactor especially those aspects related to the blanket.

We view the overall blanket region as a short cylinder with an octadecagon cross section. The cylinder has vertical inner and outer walls and horizontal wedge shaped top and bottom surfaces. The individual blanket modules completely cover these surfaces in the case of the EPR design and cover all but the divertor slot in the case of a commercial reactor.

Figure 5 shows schematically how the blanket and shield are assembled. The shield forms a fixed structure from which the blanket modules are supported and also provides the primary vacuum seal. Blanket canisters are mounted on heavy aluminum backing plates forming modules which are in turn supported from the fixed shield.

Figure 6 shows how the modules compare in the case of the EPR* and a commercial reactor with a UWMAC size plasma. For the EPR, the module is 1.0m wide x 4.8m long x 0.7m thick. It holds 3 canisters, each 33cm wide x 4.8m long x 50cm thick for a total weight of about 7 mtons. Modules for the UWMAC size plasma are basically similar and are somewhat larger with 6 canisters per module, each canister being 11m long.

Continuing with the EPR case, Figure 7, a representative 20° sector of the torus, shows the three vertical outer modules and the one vertical inner module. It also shows how the tapered section associated with each 20° will be covered with modules. The central unit is similar to the vertical side wall modules while left-hand and right-hand modules are composed of foreshortened canisters to match the sector taper. These foreshortened canisters intrude into the taper space of the adjacent sector. This figure also indicates that some of the

*EPR $\equiv$ experimental power reactor.

severely foreshortened canisters might be eliminated entirely to provide beam holes, vacuum ports, etc.

The complete blanket consists of 72 vertical modules and 108 horizontal modules arranged so that any one of the 10 modules associated with each 20° sector can be removed through two relatively small access ports located between the magnets bounding each sector. Figure 8 shows an elevation and section of the lower access port. It should be noted that the size of this access port is somewhat larger than a single module, is located on the sector centerline and also contains all the header connections to the lower horizontal modules. The upper access port is similar but is designed to permit access to top modules as well as outside vertical modules. Access ports are closed by a removable shield plug which may be water filled, the water being discharged when the plug is to be pulled.

Suitable track arrangements in the fixed shield structure [Figure 9], together with the access ports just mentioned permit replacement of any module in the blanket. For example, to remove the lower central module, valving in the coolant headers would be shut off, after which the quick disconnects in the lines outside the shield plug would be opened; then the vacuum seals around the shield plug would be cut. All this is done on the outside of the shield and can be accomplished by direct access. The removable shield plug is then removed and finally the bottom central module rolled out on its roller pad supports. To remove either the left or right side tapered modules, the central module would be removed followed by movement of the side module, first to the center module position (i.e., sidetracking) then out through the access port.

The upper horizontal modules and outside vertical modules are removed in a similar manner through the upper access port, in each case the central module being removed before the side module can be removed.

Replacement follows the same procedure in reverse order.

Figure 10 shows one of the design concepts being considered for the canisters. This concept embodies three oval-shaped cylindrical canisters bolted to the aluminum backing plate to form a unit. Presently we prefer this concept if stresses can be held to within reasonable limits. A modification of this concept embodies 6 circular cylindrical canisters stacked 2 high on equilateral triangular pitch. This concept increases the header and support problems but stress-wise would be better than the oval concept. Figure 11 shows schematically the shell cooling configuration. The inlet and outlet headers run the full length of the canister and form part of the canister wall. From these headers, individual coolant passages carry the shell coolant circumferentially around the canister wall.

Alternate dimensions are shown on Figure 11 for the water coolant passages (water in at 189°C, out at 204°C, at 300 psia) and for the helium coolant passages (helium in at 200°C, out at 380°C at 600 psia). The water cooled shell uses a wrought aluminum alloy (5000 series) while the helium cooled shell uses SAP.

A summary of the blanket process conditions for both the water cooled first wall and for the helium cooled first wall are shown in Table 1.

8.2 Technical Analysis of EPR Design. Design of anything as technically involved as a tokamak blanket is by its very nature an iterative process. Starting with basic premises, broad decisions are made which are then examined, carefully analyzed and evaluated and then a second round of decisions made or modification made to earlier decisions and the whole cycle of analysis repeated. Depending on the number of iterations made, one can refer to the design as a conceptual design or a feasibility study or in the ultimate an optimum design. Engineering studies of fusion blankets are at best conceptual designs or feasibility studies. True optimization is necessarily well off into the future although some measure of local optimization is possible in feasibility studies.

The basic criteria used for the EPR analysis were: 1) a plasma burn time of 30 sec on and 30 sec off; 2) a wall loading of 1 MW/m² during the on cycle, resulting in an average loading of 0.5 MW/m² for the entire cycle; 3) ANL-TEPR plasma dimensions (see Figure 5) and 4) no breeding. Other basic criteria had to do with limiting materials temperatures. In the case of the wrought aluminum alloy (5000 series) the temperature was limited to about 200°C while in the case of SAP the temperature was set at 400°C. For the ceramic materials inside the canister, no attempt was made to push them to their temperature limits but rather to base the hot helium temperatures on existing HTGR technology, it being fully recognized that at some future date an optimization might indicate higher temperatures.

These basic premises led to a canister design essentially 33cm x 50cm x 5m with an aluminum wall about 2.5cm thick. This aluminum canister constitutes the first wall. Immediately behind the insulated aluminum wall and extending for approximately 20cm is a region of silicon carbide blocks in which most of the neutron moderation takes place. Behind the SiC blocks, there is 30cm of B₄C in which the remainder of the neutron slowing down and absorption occurs. Thus the bulk of the fusion energy is absorbed by these high temperature ceramic materials from whence it is transferred to the helium stream at temperatures of the order of 700-800°C.

8.3 Neutronics. Each blanket module was subdivided into a number of zones and

TABLE I

Typical Thermal and Hydraulic Characteristics of the
Aluminum Blanket at 0.5 MW/M² Wall Loading for an EPR

Nominal Reactor Power	300 MW
Fraction of Power to Cold Circuit	0.45
Fraction of Power to Hot Circuit	0.55

Cold Circuit Conditions

	Option	
	5000 Series	
	Al	SAP
Coolant	Water	He
Inlet Temperature, °C	189	200
Outlet Temperature, °C	204	380
Operating Pressure, psia	300	600
Flow Rate, g/sec	4.30×10^6	2.87×10^5
Channel Velocity, cm/sec	420	3500
Heat Transfer Coefficient, w/cm ² - °C	4.59	0.35
Blanket Pressure Drop, psia	29	13.5
Pumping Power as Fraction of Total Power	0.003	0.022

Hot Circuit Conditions

Coolant	He
Inlet Temperature, °C	400
Outlet Temperature, °C	800
Operating pressure, psia	300
Flow Rate, g/sec	7.9×10^4
Channel Velocity, cm/sec	2200
Heat Transfer Coefficient, w/cm ² - °C	0.08
Blanket Pressure Drop, psia	8
Pumping Power as Fraction of Total Power	0.01

Blanket Conditions

	Option	
	5000 Series	
	Al	SAP
Maximum First Wall Temperature, °C	215	400
Maximum Carbide Bed Temperature, °C	1450	1450

within each zone, the components were homogenized. Neutron and gamma-ray fluxes were calculated by the computer program ANISN, using 100 neutron and 21 gamma-ray energy groups, a cylindrical geometry and a P_3S_8 order of expansion, together with the neutron and gamma-ray kerma factors are supplied by the data library DLC-37 [16]. The 14.1 MeV source neutrons are taken to have a uniform spatial distribution in the plasma region. The wall loading was 1.0 megawatt (thermal)/ m^2 .

The spatial distribution of the neutron flux as well as the heat deposition rate due to neutron and gamma heating are shown on Figure 12 as a function of distance from the first wall. The neutron energy escaping from the shield is about 10 eV/fusion neutron for the circular aluminum case.

8.4 Thermal Analysis. Since two coolant circuits are contemplated, i.e., one to cool the Al canister and the other to cool the hot interior, it is imperative to determine the total heat flow in each circuit. We now examine the heat flows to the canister circuit.

The incident energy flux amounts to 3.5 MeV of the total 17.4 MeV deposited or 20%. This is all deposited within the first few millimeters of the first wall and is essentially a surface effect. Neutron and gamma ray heating of the first wall amounts to about 4.4 W/cc which for a 2.5cm thick wall amounts to an additional 11%. Neutron and gamma ray heating of the side walls can be approximated by

$$q = q_0 e^{-.083x}$$

where q_0 is the heating rate at the first wall and x is the distance away from the first wall. For the basic canister dimensions selected, this amounts to an additional 8.1%. Heat leakage from the hot interior region depends on the temperature levels in the canister wall and in the interior, the amount and quality of insulation used, and the contact area. Using 2.5 cm thick layer of graphite felt with a thermal conductivity of $0.0036W/cm^{\circ}K$ results in an additional heat flow of 5.5% in the water cooled case and somewhat less in the helium cooled case.

Thus the total heat pickup in the canister wall cooling circuit amounts to 45% of that generated by the plasma plus neutron and gamma reactions in the wall structure. The remaining 55% is absorbed in the hotter interior. For a 5000 MW(th) commercial reactor (UWMAK-I [12] plasma dimensions), wall loading of $2.5 MW(th)/m^2$, and He cooled Al shell, the energy split between the cold and hot

regions is 34% and 66%, respectively. The presence of the breeding and neutron multiplier materials in the hot region account for the increase; the larger Q value (MeV/fusion) reducing the fraction in the cold region.

Having determined the heat pick up in the canister wall, the coolant flow can be established consistent with previously established aluminum temperature limits. Since in the present instance, the coolant passages occupy but a small portion of the shell wall, detailed thermal analysis seemed warranted. Optimization of the location of these coolant passages or of their dimensions has not been attempted. Based on an arbitrary (seemingly reasonable) choice of dimensions (see Figure 13), the two dimensional temperature distribution in the canister wall was determined using the heat conduction code, HEATING-3 [17] and is shown in Figure 14 for the water cooled wall (the temperatures are relative to a fluid temperature of 0°). It should be pointed out that these results are all for the "plasma on" phase and are therefore based on an actual wall loading of 1 MW/m^2 . The aluminum shell has a small heat capacity compared to its interior contents and it will be necessary to either almost stop flow during the plasma off period, or maintain the shell coolant inlet temperature constant by means of thermal ballast in the exterior cooling circuit. The first alternative would appear preferable since it would minimize the amount of stress change due to thermal cycling, but the latter course would probably also be acceptable with water coolant. In this latter case, the average shell temperature will vary about 15°C . In the former case, the average temperature will vary by a much smaller amount. With a He cooled shell, on the other hand, the ΔT across inlet to outlet will be considerably greater or $\sim 180^{\circ}\text{C}$. In this case it would probably be necessary to essentially stop the He coolant flow to the shell during the plasma off period.

The characteristic Reynolds number is such that for the water cooled shell, EPR and commercial reactor, the flow is turbulent ($\text{Re} \sim 4 \times 10^4$ to 10×10^4). For the helium cooled SAP shell, $\text{Re} \sim 5 \times 10^3$ to 14×10^3 .

Having established the water flow and temperatures, all that remained was to establish the pressure required to suppress local boiling anywhere in the coolant path.

The remaining 55% of the heat to be recovered from the ceramic internals represents a much easier design problem since 1) the filler material has no structural function and 2) the operating temperatures in the He circuit were established based on HTGR technology and are well below the melting points of the carbides ($>2000^{\circ}\text{C}$).

The carbide bed could be formed in a variety of shapes and sizes. What was selected for this design was a rod bundle composed of rectangular rods with spaces to provide coolant passages between the rod clusters. In the EPR design, the hot helium flows through the packed bed for the full length of the module, turns around and flows back through the full length of the module. The commercial module being more than twice as long as the EPR module would have excessive pressure drop if this same flow pattern was used. Instead, the hot helium flows through a header space which runs the full length of the module. The helium leaving the header space flows radially along the 70 cm depth of the bed to the outlet collection header. This much shorter path results in a reduced pressure loss and permits for larger mass flows. In either case flow through the bed is laminar. To maximize the shielding function, the voidage permitted for these coolant passages was limited to 15%. Since the thermal capacity of the ceramic rod bundle is large relative to the thermal capacity of the helium coolant stream, it is perfectly acceptable to maintain the same coolant flow for both the plasma on and plasma off phase of the cycle and to withdraw the heat at the average rate based on 0.5 Mw/m^2 .

Headers or manifolds are important in a design such as this (or any blanket design)-important inside each canister to insure uniform cooling of the canister wall and equally important external to the blanket proper to insure uniform flow to each of the blanket modules.

There are two counteracting pressure effects of concern in these manifolds. On the one hand, flow friction causes the static pressure to decrease with distance through the header. On the other hand, the continuous removal of fluid through the manifold connections results in a decrease in velocity head in the feed header which causes the static pressure to increase with distance through the header. In the return header the effect of flow friction and of velocity head operate in the same direction causing a more pronounced change in static head with distance through the header. It is of course the difference in static head in the feed header and static head in the return header (at a location corresponding to the same coolant tube) which determines the flow through the coolant path.

A detailed analysis based on the methods of Acrivos [18] of the internal canister headers lends to the conclusion that the maximum deviation in flow is only 3% for either the water or helium cooled shells which should not present any problem. As might be anticipated for most of the flow path, the flow is turbulent then laminar as it approaches the end of the log.

8.5 Stress Analysis - A two dimensional, plane strain finite element analysis has been used to determine stress in the blanket modules due to the 300 psi internal pressure. Canisters formed by two hemicylindrical heads connected by straight sections, with a wall thickness of 2.5 cm, exhibit unacceptable stress levels when they are located at the ends of the module where lateral support from adjacent canisters cannot be assumed. Excessive bending causes stresses in the 25 ksi range. The shape shown in Figure 15 is designed to reduce the bending by stiffening the sides of the canisters. As shown, the vertical portion of the 2.5 cm shell wall is reinforced by a tapering support 5.5 cm at its base made of fiber product (FP) aluminum material. This support is cantilevered from the module's base plate and attached to the shell wall. Fiber product is an extremely tough aluminum composite with an elastic modulus of 30×10^6 psi. The indicated support would only be used at the outside walls of the two extreme canisters in each module. A small boss at the shoulder of the canister is included to butt up to a similar boss in the adjacent module thereby restricting lateral movement in the outside canister wall. By using a small raised surface, instead of the entire vertical wall, for lateral support between modules the machining tolerance problems are considerably lessened.

With the FP reinforcement and lateral support from adjacent modules the stress in the canisters are greatly reduced. The maximum stress is 3500 psi in the aluminum wall with average stresses through the wall around the 2000 psi level. Figure 15 shows the stresses at various critical locations for this case. As can be seen all stress values are now at acceptable levels. Even if for some of the canisters the adjacent module does not provide lateral support the stress values in the FP reinforced design are significantly reduced from those in the unsupported design. Now a maximum stress of 12400 psi is reached at the canister shoulder where the FP support begins. Bending stresses at top center of the canister are also higher, reaching a value of 7800 psi. In both cases average stresses through the wall are much lower, so that, while local yielding at the highly stressed edges may occur, the integrity of the wall will be preserved. In the finite element analysis of this shape the canister is assumed to be symmetrical about its vertical center line. In actuality the inside wall of any outside canister is supported by the wall of the adjacent internal canister in the same module so that the stresses calculated using the symmetry assumption are probably larger than would actually be experienced in the outside wall. (It should be noted that module canisters internally located have their vertical walls supported laterally along their entire height by adjacent canisters. Under

these conditions stresses in the canister walls are below 2300 psi everywhere. Therefore the relatively high stress values discussed above refer only to the outside canisters of each module.) The 12400 psi stress value as well as the other stress values in Figure 15 can be further reduced by thickening the FP support and bringing it up higher on the outside shell wall. Better fairing of the support with the wall will also reduce stresses.

It must be pointed out that such effects as thermal cycling or prolonged radiation which will influence canister strength have not been taken into account in the above analysis.

SUMMARY

These are but a few of the considerations in the area of blanket technology for the design of a fusion reactor, the analyses being far from complete. They do provide, however, a starting point for an appreciation of some of the problems and possible solutions. As in other high technology areas, one thing becomes apparent and that is the multidisciplinary nature of the efforts required for a final design.

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FIG. 1 MODULAR MINIMUM ACTIVITY
BLANKET

FIGURE 2 - MODES OF TRITIUM RELEASE IN MODULAR BLANKETS

FIG. 3 HIGH TEMPERATURE GRAPHITE BLANKET

FIG. 4 INTERMITTENT DIRECT COOLED GRAPHITE BLANKET

EPR - CIRCULAR PLASMA SHOWING MODULE REMOVAL SCHEMES

FIG. 5

COMPARISON OF BLANKET MODULES
FOR OUTER SIDEWALL OF FUSION.
REACTORS

DETAIL OF 20° SECTOR OF TORUS
 SHOWING INNER & OUTER VERTICAL MODULES
 AND TOP OR BOTTOM MODULES

FIG. 8

PLAN VIEW SHOWING POSITIONING TRACKS IN FIXED SHIELD
FOR BOTTOM MODULE INSERTION AND REMOVAL

MODULE SHOWING THREE INDIVIDUAL CANISTERS

FIG. 11

NEUTRON FLUX & VOLUMETRIC HEAT DEPOSITION RATES vs DISTANCE

FIG. 12

TWO DIMENSIONAL MODEL USED TO COMPUTE TEMPERATURE
 DISTRIBUTION IN FIRST WALL OF WROUGHT
 ALUMINUM BLANKET

FIG. 13

TEMPERATURES SHOWN ARE IN °C
ABOVE COOLANT TEMPERATURE

TEMPERATURE DISTRIBUTION IN WROUGHT ALUMINUM FIRST WALL

FIG. 14

	STRESSES IN psi		
	INSIDE	OUTSIDE	AVERAGE
AA	7770	-2995	2100
BB	3200	990	2030
CC	-6110	12,400	1840
DD	-3710	1510	-2067
MAX. STRESS IN FP = 17,500 psi			

	STRESSES IN psi		
	INSIDE	OUTSIDE	AVERAGE
AA	1370	1960	1640
BB	1840	1550	1650
CC	3080	1030	1700
DD	3510	1150	2180
MAX. STRESS IN FP = -10,580 psi			

STRESSES IN EDGE CANISTER WITH AND WITHOUT ADJACENT SUPPORT