**STUDIES OF FUSION REACTOR BLANKETS WITH MINIMUM
RADIOACTIVE INVENTORY AND WITH TRITIUM BREEDING
IN SOLID LITHIUM COMPOUNDS:
A PRELIMINARY REPORT**

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**ENGINEERING DIVISION
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Abstract

The feasibility of fusion reactor blankets with low residual activity is examined. Several designs are examined with regard to activation, tritium breeding ratio, mechanical design, tritium removal from the blanket, and thermal cycle efficiency. Using aluminum (SAP) as a structural material, it should be possible to build CTR blankets with ~ 10 curies/MW(e) of long lived (half life one day or greater) residual activity (other than tritium), which is many orders of magnitude less than with Nb or stainless steel blankets. In the designs examined in this study, tritium is bred in solid lithium containing materials, e.g., LiAl alloy, which have high equilibrium tritium pressures. The tritium diffuses into either the helium coolant stream, from which it is removed by absorption, or into the vacuum region between the first wall and plasma. Depending on processing methods and blanket parameters, the tritium blanket inventory

should range from 10^2 to 10^3 curies/MW(e). Tritium breeding ratios range from 0.9 to 1.5 depending on blanket design, while thermal cycle efficiency is estimated to range from 35% to well over 40%, depending on design. Several module designs are developed in which helium coolant exit temperatures are substantially above the operating temperature limit ($\sim 400^\circ\text{C}$) for the aluminum (SAP) structure.

Introduction

As Steiner (1,2) and others (3,4) have pointed out, fusion reactor blankets using niobium and/or stainless steel as structural material will result in substantial amounts of long lived radioactive isotopes; in fact, the amount will be comparable to that in a fission reactor. While in general the biological hazards of such materials are less than those in fission reactors, they are still of concern. With vanadium, the long lived inventory is several orders of magnitude less, and results from impurities.

The use of refractory metals (Nb, V, Mo) or stainless steel is dictated because in the blanket concepts that have been proposed, tritium is bred in liquid lithium or a liquid lithium compound (e.g., $\text{LiF}\cdot\text{BeF}_2$ "flibe"). Refractory

metals or stainless steel appear necessary to resist corrosion by the liquid lithium or lithium compounds.

In this study we examine the possibility of a new type of CTR blanket in which tritium is bred in a solid lithium alloy or compound in the blanket. In one version of this concept, the bred tritium diffuses out of the solid lithium alloy or compound into a helium coolant stream, from which it is removed at a processing point outside the blanket. In another version of the concept, the bred tritium diffuses from the solid lithium alloy or compound into the plasma exhaust of the reactor.

One of the prime candidates for the solid lithium containing material is lithium-aluminum alloy. It has a high melting point, $\sim 700^{\circ}\text{C}$, and the vapor pressure of tritium from the alloy is much greater than from pure lithium. The fraction of blanket lithium converted to tritium is small, on the order of a few percent, over the 30 year life of a fusion reactor, so it is not necessary to ever replace the solid lithium alloy.

This concept thus permits a much wider range of blanket structural materials, since corrosion is not a problem. R. Hirsch and W. Gough (5) have suggested the desirability of a minimum radioactive inventory CTR. The use of a solid lithium alloy breeding blanket seems to be advantageous for this purpose since structural materials with very low long-lived activation can

now be used, where previously they would be ruled out because they were not compatible with liquid lithium or flibe. A prime candidate for a structural material is SAP (sintered aluminum product) in which pure aluminum is strengthened by the addition of 5-10% by weight of Al_2O_3 in the form of a very fine dispersoid in the aluminum matrix. This material has several advantages: it is reasonably strong, even at temperatures of 350-400°C; aluminum and Al_2O_3 can be made very pure to reduce activation of impurities; long lived activation products of Al and O for 14 MeV neutron are very small; it appears to have good resistance to radiation damage; has a high thermal conductivity; and it should be cheap.

In the rest of this report we examine several blanket designs based on this concept from the standpoints of materials, activation, breeding, tritium removal, heat removal, and cycle efficiency.

2. Summary-Implications for the CTR Program

On the basis of this preliminary study, it appears that CTR blankets can be designed to have very low inventories of long lived radioisotopes without a significant sacrifice in thermal cycle efficiency.

The total inventory of radioisotopes in a 1000 MW(e) reactor blanket is estimated to be:

- a) 10^4 curies of long lived activity (half life
≥ one day, but not including tritium)
- b) 10^5 to 10^6 curies of tritium
- c) 10^9 curies of Na^{24} (15 hour half life)

There is an additional $\sim 2 \times 10^9$ curies of short lived activity with half lives ranging from milliseconds to a few minutes. These activities will decay so quickly that they may be neglected.

Approximately 7000 curies of the long lived activity is Al^{26} (7.3×10^5 y half life). Most of the remainder is due to activation of various impurities in aluminum (titanium, zirconium, iron, etc.). If Be is used as a neutron multiplier, there will be a few hundred curies of Be^{10} (1.6×10^6 y half life), which should not be a significant biological hazard. There will also be ~ 100 curies of C^{14} from (n, α) reactions on O^{17} and (n, α reactions on C^{13} .

This amount of long lived activity is approximately 5 orders of magnitude smaller than the inventory in Nb-Li CTR blankets, and comparable to, though somewhat lower, than with V-Li. More information is needed about impurity levels in vanadium to assess how much lower.

The tritium blanket inventory depends on many factors: type, temperature, amount, and effective particle size of the fertile material (e.g., LiAl or $\text{Li}_2\text{Al}_2\text{O}_4$); amount of scavenging protium; and, if the tritium is released to the helium

coolant, what fraction of the coolant stream is processed to remove tritium. The inventory of 10^5 to 10^6 curies is estimated for LiAl alloy, depending on LiAl inventory, temperature, whether protium is used to scavenge tritium, etc. It may be possible to further reduce the tritium inventory with other fertile materials.

The Na^{24} inventory is large and comparable to the total radioisotope inventory in Nb-Li CTR blankets. The short half life should greatly reduce the hazard of the Na^{24} blanket inventory, however.

The tritium breeding ratio depends on blanket design. For designs with Li^7 in the fast neutron zone (type 1), the breeding ratio is ~ 0.9 . This can probably be increased to ~ 1.0 by optimization, but does not appear possible to raise it significantly above 1. This results from the necessarily high Al/Li ratio and thick Al first wall. If Be metal is used as a neutron multiplier (type 2 design), the breeding ratio is well above one (1.5 in one design), and breeding occurs principally by n,T reactions in Li^6 at relatively low neutron energies. If BeO is used instead of Be (type 3 design), the breeding ratio is slightly above one (1.1 in one design). It is probably not serious if the breeding ratio for minimum activity CTR's were slightly less than one, since a few specialized CTR's

(either operating on DD fuel or optimized for high breeding ratio) could make up the tritium deficiency for a much greater number of CTR's. However, there does not appear to be any problem in using Be or BeO neutron multipliers in the designs presented here; since helium coolant is used, all blanket materials are solid, and the Be or BeO is not used as a structural material. In this case one would have breeding ratios well above one.

The most promising low activity structural material is SAP, a dispersoid of micron size aluminum oxide particles in an aluminum metal matrix. The maximum operating temperature of this material is $\sim 400^{\circ}\text{C}$. A number of approaches are possible to achieve coolant exit temperatures substantially above 400°C , however. One such approach involves cooling the SAP structure with one helium stream, and cooling the rest of the blanket, in which most of the energy is deposited, with a separate helium stream. Another approach involves cooling the SAP structure with the inlet He stream which is then routed to the rest of the blanket. A number of modular designs incorporating these two approaches are examined, and depending on the design, it appears that thermal cycle efficiency can range from 35 to 50%.

Maximum stress in the SAP structure of the modules considered in this study is ~2000 psi, or ~20% of the stress rupture strength of SAP. The radiation damage resistance of SAP appears to be good for fission neutrons, at 300 to 400°C, but its resistance to 14 MeV neutron damage should be investigated, particularly as regards the effect of H and He produced by (n,p) and (n,α) reactions.

The tritium that is bred in the solid lithium containing material (e.g., LiAl alloy or $\text{Li}_2\text{Al}_2\text{O}_4$) diffuses out of the solid, either into the helium coolant stream where it is removed by absorption in a metal hydride bed from which it is recovered for injection into the plasma; or into the vacuum region between the plasma and the first wall, where it is recovered along with DT fuel escaping from the plasma and is recycled to the plasma. Depending on equilibrium tritium concentrations in the lithium containing material, as well as tritium diffusion rates, it may or may not be necessary to use protium to scavenge out the tritium bred in the solid.

There are several important implications for the CTR program arising from these studies:

- 1) Low residual activity CTR blankets should be superior in terms of siting and general public acceptance.
- 2) Low residual activity modular blankets should be much easier to repair or replace.

3) Blankets with solid lithium containing materials for tritium breeding should have less problems of corrosion or material compatability and should be easier to develop.

4) Low activity blankets for experimental and prototype CTR reactors should be advantageous since modifications can be more easily made after startup if necessary. Blankets with solid lithium material should offer less operating problems; further, they can probably be developed more rapidly than blankets with liquid lithium or flibe.

3. Blanket Materials

3.1 Properties

Table 3.1.1 lists the potential candidate materials for a fusion reactor blanket with minimum radioactive inventory and solid lithium breeding material.

Of the structural materials, SAP seems by far the best choice. It is ductile and has high strength (~10,000 psi) at elevated temperatures (~400°C), can be made in a very pure form, has good radiation damage resistance, high thermal conductivity, and is cheap (~\$1/lb). The other possible structural materials have significant disadvantages: beryllium is difficult to fabricate, expensive, and not reliable as a structural material; graphite and silicon carbide are brittle and low strength; and other Al alloys have considerably lower strength than SAP at

400°C. Magnesium could be a possible alternative to SAP, using a MgO dispersoid for strength, but its properties are not known.

Though SAP cannot be considered for long term use at temperatures much above 400°C, it can still be used in fusion reactor blankets operating at much higher maximum temperatures, if suitably cooled. This would require either cooling the SAP structure with incoming low temperature coolant (e.g., He at ~350°C), with the coolant then directed to the hotter blanket regions, or two separate coolant streams could be used, one for the SAP structure and the other for the hot blanket faces. Some form of thermal insulation would be needed to prevent heat exchange between the SAP structure and the hotter zones.

Of the possible moderating materials, there appears to be a much wider range of choices. There are several promising hydrides (or deuterides): TiH_2 (or deuteride) is attractive because of its high hydrogen density, high stability and low H_2 dissociation pressure (it could probably be used up to 600°C, where the H_2 dissociation pressure is ~300 mm Hg for $TiH_{1.5}$). However, it is strongly activated by fast neutrons in (n,p), (n, α), etc. reactions, and the threshold energy for neutron activation is rather low, on the order of 2 MeV. Its use would then be restricted to portions of the blanket and

shield where the fast neutron flux was small. Zirconium hydride has been used as a moderator for fission reactors, but it is activated by thermal neutrons and would probably not be suitable for a low activity fusion blanket, since it would give an inventory of $\sim 10^6$ curies/1000 MW(e). Vanadium monohydride is another possible moderating material. Its stability and H_2 dissociation pressure are comparable to TiH_2 , and it is not activated by thermal neutrons. Its threshold for activation by fast neutrons is low, however, on the order of 2 MeV. It would appear that of the two possibilities, TiH_2 would be a somewhat better choice.

Sodium hydride (or deuteride) is an interesting possibility. Its stability is rather poor (H_2 dissociation pressure is ~ 7 atm at $550^\circ C$), but it is cheap, can be easily obtained in pure form, and has a very high threshold energy for activation by fast neutrons. The only significant long-lived product is Na^{22} (2.6 yr half life) which is generated by an (n,2n) reaction with a threshold energy of 12 MeV. Potassium hydride is similar to sodium hydride, except that the activation threshold is much lower for production of A^{39} (269 yr half life) through an n,p reaction.

Magnesium hydride has a high H_2 dissociation pressure (1 atm at $350^\circ C$) and its use would severely limit the maximum temperature in the blanket.

All hydride moderators would have to be jacketed to prevent loss of hydrogen to a He coolant stream. The jacket material would probably be SAP. Since the jacket would essentially be in a state of zero stress, it could operate at higher temperature, i.e., $\sim 500^{\circ}\text{C}$. The jacket would serve principally as a diffusion barrier for H_2 , with a small pinhole in the jacket permitting inside and outside pressures to equalize (i.e., the gas inside the jacket would be mostly He, with a small amount of H_2), and to vent helium formed by (n,α) reactions a small amount of H_2 would be lost along with the vented He. H_2 has long term deleterious effects on aluminum at temperatures above 500°C , causing blistering and microcracks (6). However, since the SAP jacket is only a diffusion barrier and not a structural material, H_2 should not have any serious effect on the jacket.

Since the hydride moderator would be in the form of a powder, no radiation damage problems are anticipated. Molecules dissociated by radiation will quickly recombine, while helium produced by (n,α) and (n,He^3) reactions will be vented to the main coolant stream.

If Be and BeO ($\phi t \sim 10^{23}$) and graphite ($\phi t \sim 10^{22}$) are used only as moderators and not structurally, no problems arising from radiation damage are anticipated. There will be substantial volume increase after long irradiations, but sufficient space in the helium coolant channels can be incorporated to accomodate this volume increase.

Considerable information on SAP was developed in the course of the program on Heavy Water Moderated Organic Cooled Reactors (HWOOCR) in which it was intended for use both as a structural and cladding material. Much of this information is summarized in the SAP Handbook (6).

For this study we have adopted SAP 895 (89.5% by wt of Al) as the nominal alloy composition. Figures 3.1.1 and 3.1.2 taken from (6) show typical stress rupture and creep properties for SAP 895 in the unirradiated state. Figure 3.1.3 also taken from (6) shows the effects of irradiation to 4×10^{20} nvt ($E > 2.9$ MeV) at an irradiation temperature of 104°F . These irradiation effects disappear at higher irradiation temperatures and it is noted in (6) that "irradiation at 572°F to 4×10^{20} nvt ($E > 2.9$ MeV) had no significant effect on yield strength, ultimate tensile strength, or ductility up to 1112°F ."

Pearlman (7) notes that "The pressure tube life in the Heavy Water Moderated Organic Cooled Reactor was to be 30 years, at temperatures in excess of 350°C . The neutron flux for this duty was calculated to be 5×10^{22} nvt ($E > 1$ MeV). We did not consider this to present a major structural problem for the alloy (SAP).....Other than a tendency towards irradiation hardening at room temperature, we observed no effect of neutron irradiation up to 1.2×10^{21} nvt ($E > 1$ MeV). We think that this is probably due to the stable defect structure

AI-CE-MEMO-24, Rev 1

Typical Creep Curves for SAP-ISML 930 and 895
at 752°F (400°C)

Stress Rupture of SAP 895 (uniaxial tests
of bar and tube specimens)

ALCE-MEMO-24, Rev 1

3-15-66

695 7680-25582

Tensile Properties of SAP 938 Before and After
 Irradiation to 4×10^{20} nvt ($E > 2.9$ Mev) at 104°F(3-19)

inherent in the alloy preventing long range diffusion and a resulting effect on macroproperties. We agree, however, that a screening irradiation program should be an early priority in any CTR development program."

Pearlman further notes that irradiation may affect the life of the transition joints between SAP and stainless steel through increased diffusivity due to the production of Si by (n,α) reactions. However, in the blanket designs we are considering, any transitions would be outside the shield zone where neutron flux is negligible.

SAP would thus appear to have good radiation damage resistance in terms of a fission neutron spectrum. The irradiation of SAP in a fusion reactor blanket would be $\sim 10^{23}$ nvt ($E > 1$ MeV) at $1 \text{ MW(th)}/\text{m}^2$, which is comparable to that expected for the HWOCR. However the average neutron energy is considerably higher in a fusion reactor blanket. Of particular concern is the enhanced production in a fusion neutron spectrum of H and He atoms in the metal by (n,p) and (n,α) reactions. After 30 years irradiation at $1 \text{ MW(th)}/\text{m}^2$, ~ 7000 ppm each of He and H will have been generated in a SAP first wall. At temperatures above 500°C H can cause blistering and internal microcracks after long times (6). More information is needed to determine the effect of H at lower temperatures. In experiments with ion bombardment of SAP, "it was concluded that upon

Table 3.1.1

Potential Materials for Low Activity CTR Blankets

Structural Materials

Al (SAP); Al Alloys (5083, 5456)

Mg (with MgO)

Beryllium

Graphite

Silicon Carbide

Moderating Materials

TiH₂, TiD₂; NaH, NaD; KH, KD; VH, VD

MgH₂

Graphite

Beryllium Metal

Beryllium Oxide

Thermal Insulators

Al₂O₃

MgO

Breeding Materials

LiAl, LiSi, LiMg alloys

Li₂Al₂O₄, Li₂SiO₄ ceramics

Neutron Absorbers

B₄C, TiC

Coolant

He @ 10-20 atm

annealing at 400°C, the inert gases formed bubbles at the metal oxide/matrix interfaces (100 Å diameter); growth of bubbles did not occur at higher annealing temperatures until melting occurred" (6). This would tend to indicate that swelling due to He formation may not be serious.

Thus SAP potentially appears to be structurally adequate for fusion blankets. More work is needed to confirm its suitability and to determine what are safe operating stress levels. For purposes of this study we adopt a stress limit of 2500 psi for SAP, which is ~25% of its stress rupture limit at 400°C.

3.2 Activation

Since aluminum is the most promising structural material for low activity blankets, and also is present in one of the prime candidate breeding materials, Li-Al alloy, we first consider the question of neutron activation in aluminum. Figure 3.2.1 and 3.2.2 show experimentally measured activations of various grades of aluminum in a thermal spectrum. These experiments were carried out by ALCOA (8) in 1964 at the BNL graphite reactor. They have not been published before and we wish to express our appreciation to ALCOA for making them available.

The aluminum with the lowest activation is the ALCOA Process
(A)1199

grade. It is nominally 99.97% pure; typical impurity analyses are given in Table 3.2.1. This alloy was not developed for the purpose of low activation, but rather is one of ALCOA's standard grades, and costs on the order of \$0.50/lb. Kramer (8) points out that substantial further reductions in activation can probably be achieved by selecting the ore from which the Al_2O_3 feedstock is derived. Process modifications can also reduce the impurity level and activations. These techniques should not substantially increase the cost of aluminum beyond that for All99. Another technique would be to zone refine the All99 or equivalent grade. Such processing is now done routinely for the production of high resistivity ratio aluminum for cryogenic electrical applications. The present price of 5000 resistivity ratio (nominally 99.999% pure) aluminum is ~\$30/lb Al at an annual production of ~5000 lbs/year. In a study on the future cost of superconductors in a fusion reactor (9) economy, 5000 resistivity ratio aluminum was projected to be reduced to a cost of \$4/lb and 25,000 resistivity ratio aluminum (99.9999% pure) to a cost of \$8/lb. One could afford to pay \$10-20/lb for high purity aluminum for fusion reactor blankets.

The activation of All99 is only $\sim 10^{-4}$ curies/lb a day after it is removed from the reactor ($\sim 10^{13}$ n/cm² sec). For a 1000 MW(e) fusion reactor which would take $\sim 3 \times 10^6$ lbs of

aluminum at $1 \text{ MW(th)}/\text{m}^2$ in the blanket, an activity of 10^{-4} curies/lb corresponds to a total radioactive inventory in the reactor of only ~ 300 curies. However, the activation measured for Al199 in the BNL reactor would be different from that for Al199 in a fusion reactor. Three factors affect this difference:

a) The Al199 sample was only exposed for 20 days. The principal impurity activations were Sc^{46} , Zn^{65} , and Cr^{51} , which have half-lives of 83.8 d, 243.7 d, and 27.7 d. Activations with larger half lives, e.g., Co^{60} and Fe^{55} , did not approach equilibrium to the same degree as the Sc, Zn, and Cr isotopes. Exposures of 30 years might show other significant activations than the three observed, and one would expect that Al199 with longer exposures would tend to show higher residual activation.

b) The Al199 sample was exposed in a relatively high thermal flux, $10^{13} \text{ n/cm}^2 \text{ sec}$. Because of the large thermal absorption cross section of Li^6 , 942 barns, the thermal flux in a fusion reactor blanket will be considerably smaller than $10^{13} \text{ n/cm}^2 \text{ sec}$. This will tend to reduce the amount of thermal neutron activation in R-214 from what one would expect on the basis of activation at $10^{13} \text{ n/cm}^2 \text{ sec}$. The total source strength for a 1000 MW(d) DT fusion reactor (40% thermal efficiency) is 7×10^{20} neutrons/sec. For isotopes with half lives of 10 years or less, the equilibrium activity in the

blanket induced by thermal neutrons is:

$$(Ci)_j = \left(\frac{N_{Al}}{N_{Li}} \right) \left(\frac{N_j}{N_{Al}} \right) \frac{7 \times 10^{20}}{3.7 \times 10^{10}} \sum_{K=1}^{K=N} x_{jK} \frac{\sigma_{jK}}{\sigma_{Li}} \quad (3.2.1)$$

where

$(Ci)_j$ = total activity in curies induced by thermal neutron activation of j^{th} element impurity in aluminum.

$\frac{N_{Al}}{N_{Li}}$ = Al/Li atom ratio (average) in region of thermal absorption.

$\frac{N_j}{N_{Al}}$ = atom fraction of j^{th} element impurity in aluminum.

$\frac{\sigma_{jK}}{\sigma_{Li}}$ = ratio of thermal absorption cross section of Kth isotope of jth element to thermal absorption cross section of Li.

x_{jK} = fraction of jth element that is present as Kth isotope.

Table 3.2.2 lists the thermal neutron activations of significant long-lived impurities in aluminum, assuming a nominal impurity concentration (atom fraction) of 10^{-6} for each element, an Al/Li^6 ratio of 10/1 (counting aluminum in the SAP structure and Al_2O_3 insulators as well as that in the lithium breeding material). Scandium and cobalt are the potentially most significant impurities in terms of thermal neutron long lived activations. Long lived is defined here as any radioisotope

with half life of one day or greater. Some of the radioisotopes have very low energy and are much less hazardous than others, but no evaluation is made here of the relative hazards.

No scandium analysis is available for the All99 sample. However, we can obtain an upper limit for Sc concentration by assuming that all of the residual activity was due to Sc^{46} . (In fact, some was due to Zn^{65} and Cr^{51} .) The irradiation time, thermal flux, half life of Sc^{46} , and scandium thermal absorption cross section are known. The upper limit on Sc concentration is then 0.02 ppm (atom fraction). Thus Sc impurity seems to offer no problem for a CTR since its activation by thermal neutrons amounts to only 100 curies for a 1000 MW(e) reactor. It could be further reduced by suitable ore selection and/or processing, but this does not appear necessary.

A similar analysis can be made for Co. Co^{60} radiation was not observed in the R-214 sample. Assuming a detectability lower limit of 10% of total radiation, an upper limit to Co concentration is calculated as 0.02 ppm. Thus Co also seems to offer no problem for a CTR reactor, since its activation by thermal neutrons amounts to only 150 curies.

Taking the measured impurity concentrations for All99 grade in Table 3.2.1 (note that the concentrations are given in terms of weight, not atom fraction) and the activations for

a 10^{-6} nominal atom fraction given in Table 3.2.2, it is clear that the total activation by 1/V absorptions in aluminum impurities is considerably under 1000 curies for a 1000 MW(e) reactor. With non 1/V absorption included, the total long lived (half life ≥ 1 day) activation by all thermal and epithermal absorptions in the SAP structure, Al_2O_3 thermal insulators, and aluminum associated with the lithium breeding material is roughly estimated to be ~ 1000 curies for a 1000 MW(e) reactor.

The activation of long lived radioisotopes is small in either beryllium or graphite if these are used as moderators. The Be^{10} (1.6×10^6 year half life) activation is ~ 15 curies for a 1000 MW(e) reactor (Be/Li^6 ratio of 40/1) and ~ 60 curies of C^{14} (5730 year half life) at a C/Li^6 ratio of 60/1. It would appear fairly easy to obtain high purity graphite, if necessary using feedstock from thermal cracking of purified hydrocarbons. The problems of high purity beryllium have not been investigated yet, but will be in the next report.

There will also be some C^{14} produced by a thermal neutron absorption in O^{17} . This amounts to ~ 100 curies of C^{14} activity for a 1000 MW(e) reactor, with a $\text{O (natural)}/\text{Li}^6$ ratio of 10/1.

C. The aluminum structure in a fusion reactor blanket will be exposed to a fast neutron flux up to a maximum energy of 14 MeV. Activations due to fast neutron reactions like $(n,2n)$, (n,γ) , (n,p) , etc., then are possible and must be examined. These fast neutron activations may occur in aluminum, transmutation products, and/or impurities. Table A-1 in Appendix A gives the total transmutation chains for aluminum for the first, second, and third generations, for all possible reactions: n,γ ; $n,2n$; n,p ; n,α ; n,He^3 ; n,d ; and n,T . Reactions caused by γ rays are not included; however, the only significant one is γ,n and is equivalent to $n,2n$. The principal activations in the first generation for aluminum are Al^{26} ($n,2n$ reaction) with a 7.4×10^5 y half life and Na^{24} (n,α reaction) with a 15.0 hour half life. At equilibrium the Na^{24} inventory is 10^9 curies for a 1000 MW(e) reactor. Because of its short half life and because it does not concentrate in the body, this amount of Na^{24} seems to be much safer than the amount of radioactive waste products in a 1000 MW(e) fission reactor. The inventory of the $(n,2n)$ activation product Al^{26} , is very small. At the end of 30 years a 1000 MW(e) fusion reactor would have a total inventory of ~ 7000 curies of Al^{26} . This results from the long half life for Al^{26} , a very high threshold energy for the $(n,2n)$ reaction 13.5 MeV, and a small reaction cross section. This amount of Al^{26} activity does not appear to be

ALCOA Data [8]

ALCOA Data [8]

biologically significant.

The only other potentially significant activation from aluminum is Na^{22} (2.6 yr half life). The second generation route through an n,T reaction on Mg^{24} (the β^- decay product of Na^{24}) is below threshold). It first appears through several routes in the third generation. Activation through the third generation routes is very small. The Na^{22} activation for a 1000 MW(e) reactor will be small, on the order of 10 curies after 30 years.

All potentially significant long-lived activation chains for aluminum and the other main blanket component materials are summarized in Table 3.2.3. All potentially significant long-lived activation chains for impurities present in main blanket materials are summarized in Table 3.2.4.

Calculations of activation of these impurities by fast neutrons are being carried out. Preliminary findings indicate that of the various aluminum impurities, Ti and Zr will each cause $\sim 10^3$ curies of long lived activity in a 1000 MW(e) reactor, if they are present at the same concentrations as those in Al199 grade. Fast neutron activation of other impurities is much smaller than that due to Ti and Zr. With selected ore and perhaps some processing changes, the total long lived activation (half life ≥ 1 day) of aluminum impurities by fast

neutrons should be 10^3 curies or less in a 1000 MW(e) reactor.

Much purer grades of aluminum than Al199 can be obtained by zone refining. A study of superconductor costs (9) projects zone refined aluminum costs to be a few dollars/lb in large quantities. This cost appears to be practical for a CTR blanket. However, a total long-lived activation of 2000 curies per 1000 MW(e), which should be possible with some small modifications to the ALCOA process, appears to be sufficiently small that no further purification of aluminum should be necessary.

Activation of impurities in graphite or beryllium has not yet been examined. However, fast neutron activation of graphite impurities should be much smaller than the thermal neutron activation since graphite will not be used in the fast neutron zone.

The principal transmutation products of aluminum are Mg and Si. The concentrations of these elements are on the order of 1% and 0.1% after 30 years, respectively, and it does not appear that the mechanical properties of SAP should be adversely affected by them.

Table 3.2.1

Impurity Concentrations in ALCOA 1199 Aluminum

Concentration, ppm (weight) [(8)]

<u>Impurity</u>	<u>ALCOA 1199 (Standard)</u>	<u>ALCOA 1199 (Selected)</u>
Si	30	20
Fe*	10	7
Cu*	1	1
Mn*	10	9
Mg	1	1
Cr*	3	0
Ni*	0	0
Zn*	14	10
Ti*	2	2
V*	3	2
Pb*	1	1
Sn*	2	1
B	4	2
Be	0	0
Bi*	2	1
Ga	10	8
Zr*	3	4
Cd*	0	0
Sb*	0	0
Co*	<2	0
Mo*	<2	0
Na	1	<1
Ca*	<1	<1

* Impurities with long lived activation,
either by thermal or fast neutrons.

Table 3.2.2

Long Lived Activations by 1/V Absorption in
Aluminum Impurities

Nominal elemental impurity concentration = 10^{-6} (atom fraction)

Al/Li⁶ atom ratio = 10/1

Reactor power = 1000 Mw(e)

<u>Activation</u>	<u># of curies in reactor</u>	<u>Half life</u>	<u>Activation</u>	<u>No. of curies</u>	<u>Half life</u>
Co ⁶⁰	7400	5.3y	Sn ¹¹³	0.08	115d
Sc ⁴⁶	5200	84d	Sn ¹¹⁷	0.17	14d
Zn ⁶⁵	76	244d	Sn ¹¹⁹	3.8	245d
Ni ⁶³	30	100y	Sn ¹²¹	10.5	27h
Ni ⁵⁹	0.2	8×10^4 y	Sn ¹²³	1.4	129d
Fe ⁵⁹	0.7	45d	Sn ¹²⁵	0.06	9.6d
Fe ⁵⁵	25	2.7y	Cd ¹⁰⁹	2.2	453d
Cr ⁵¹	140	28d	Cd ¹¹⁵	2.3	45d (upper isomer)
Ca ⁴⁵	3.6	163d	Cd ¹¹⁵	17	54h
Ca ⁴⁷	.004	4.5d	Sb ¹²²	700	2.7d
Zr ⁹⁵	1.9	66d	Sb ¹²⁵	340	60d
Zr ⁹³	2×10^{-4}	10^6 y	Mo ⁹⁹	6.3	66h
Po ²¹⁰	38	(139d)			

Table 3.2.3

Potentially Important Activation Chains for
Main Blanket Materials

		<u>Aluminum</u>			
First Generation		Al^{27}	$n, 2n$	$*\text{Al}^{26}$	$(7.3 \times 10^5 \text{ y})$
		Al^{27}	n, α	Na^{24}	(15.0 h)
Second Generation		Al^{27}	n, α	Na^{24}	$\beta^- \text{ Mg}^{24} \text{ (S)}$
		Mg^{24}	n, T^+	$*\text{Na}^{22}$	(2.6 y)
		Al^{27}	n, γ	Al^{28}	$\beta^- \text{ Si}^{28} \text{ (S)}$
		Si^{28}	n, T^+	$*\text{Al}^{26}$	$(7.3 \times 10^5 \text{ y})$
Third Generation		Al^{27}	n, α	Na^{24}	$\beta^- \text{ Mg}^{24} \text{ (S)}$
		Mg^{24}	$n, 2n^+$	Na^{23}	(S)
		Na^{23}	$n, 2n$	$*\text{Na}^{22}$	(2.6 y)
		Al^{27}	$n, 2n$	$*\text{Al}^{26}$	$(7.4 \times 10^5 \text{ y})$
		$*\text{Al}^{26}$	n, α	Na^{23}	(S)
		Na^{23}	$n, 2n$	$*\text{Na}^{22}$	(2.6 y)
		Al^{27}	$n, 2n$	Al^{26}	$(6.4 \text{ s}) \quad \beta^- \text{ Mg}^{26}$
		Mg^{26}	n, α	Ne^{23}	$(38 \text{ s}) \quad \beta^- \text{ Na}^{23}$
		Na^{23}	$n, 2n$	$*\text{Na}^{22}$	(2.6 y)

Table 3.2.3 (Cont'd)

Oxygen

First
Generation

Second
Generation

Table 3.2.3 (Cont'd)

Oxygen (Cont'd)

Titanium

First Generation	Ti ⁴⁶	n, p	*Sc ⁴⁶	(83.8d)
	Ti ⁴⁶	n, T	*Sc ⁴⁴	(58.6h)
	Ti ⁴⁷	n, p	*Sc ⁴⁷	(3.41d)
	Ti ⁴⁷	n, d	*Sc ⁴⁶	(83.8d)
	Ti ⁴⁷	n, He ³	*Ca ⁴⁵	(163d)
	Ti ⁴⁸	n, p	*Sc ⁴⁸	(44h)
	Ti ⁴⁸	n, d	*Sc ⁴⁷	(3.41d)
	Ti ⁴⁸	n, T	*Sc ⁴⁶	(83.8d)
	Ti ⁴⁸	n, α	*Ca ⁴⁵	(163d)
	Ti ⁴⁹	n, d	*Sc ⁴⁸	(44h)
	Ti ⁴⁹	n, T	*Sc ⁴⁷	(3.41d)
	Ti ⁴⁹	n, He ³	*Ca ⁴⁷	(4.54d)
	Ti ⁵⁰	n, T	*Sc ⁴⁸	(44h)
	Ti ⁵⁰	n, α	*Ca ⁴⁷	(4.54d)

i means threshold energy for reaction >14.1 MeV

Table 3.2.4

Potentially Important Activation Chains for Blanket Impurities

Impurity	Long lived daughter isotope	Parent isotope	Generating reaction	Half life	Decay mode	Decay energy (MeV)	Max γ energy (MeV)	Final daughter
1. Sc	Sc ⁴⁴	Sc ⁴⁵	n, 2n	59h	β^+	3.9	2.6	Ca ⁴⁴
	Sc ⁴⁶	Sc ⁴⁵	n, γ	83.8d	β^-	2.4	1.1	Ti ⁴⁶
	Ca ⁴⁵	Sc ⁴⁵	n, p	163d	β^-	0.26	0.01	Sc ⁴⁵
2. Zn	Zn ⁶⁵	Zn ⁶⁴	n, γ	244d	β^+	1.4	1.1	Cu ⁶⁵
		Zn ⁶⁶	n, 2n					
	Cu ⁶⁷	Zn ⁶⁸	n, d	62h	β^-	0.57	0.18	Zn ⁶⁷
		Zn ⁶⁷	n, p					
	Ni ⁶⁶	Zn ⁶⁸	n, He ³	55h	β^-	0.20	--	Zn ⁶⁶
	Ni ⁶³	Zn ⁶⁶	n, α	100y	β^-	0.6	--	Cu ⁶⁵
3. Fe	Fe ⁵⁹	Fe ⁵⁸	n, γ	45d	β^-	1.6	1.3	Co ⁵⁹
	Fe ⁵⁵	Fe ⁵⁶	n, 2n	2.7y	E	0.23	--	Mn ⁵⁵
		Fe ⁵⁴	n, γ					
	Mn ⁵⁴	Fe ⁵⁶	n, T	312d	E	1.4	0.8	Cr ⁵⁴
		Fe ⁵⁴	n, P					
	Mn ⁵²	Fe ⁵⁴	n, T	5.6d	β^+	4.7	1.4	Cr ⁵²
4. Co	Co ⁶⁰	Co ⁵⁹	n, γ	5.7y	β^-	2.8	1.3	Ni ⁶⁰
	Co ⁵⁸	Co ⁵⁹	n, 2n	71d	β^+	2.3	1.6	Fe ⁵⁸
	Fe ⁵⁹	Co ⁵⁹	n, p	45d	β^-	1.6	1.3	Co ⁵⁹

Table 3.2.4 (Cont'd)

	Impurity	Long lived daughter isotope	Parent isotope	Generating reaction	Half life	Decay mode	Decay energy (MeV)	Max γ energy (MeV)	Final daughter
5.	Ni	Ni ⁶³	Ni ⁶⁴	n, 2n	100y	β^-	0.06	--	Cu ⁶³
			Ni ⁶²	n, ν					
		Ni ⁵⁹	Ni ⁶⁰	n, 2n	8x10 ⁴ y	E	1.1	--	Co ⁵⁹
			Ni ⁵⁸	n, ν					
		Ni ⁵⁷	Ni ⁶⁰	n, 2n	36h	E	3.24	3.2	Co ⁵⁹ (U)
		Co ⁶⁰	Ni ⁶²	n, T	5.7y	β^-	2.8	1.7	Fe ⁵⁷ (S)
			Ni ⁶¹	n, d					Ni ⁶⁰
			Ni ⁶⁰	n, p					
		Co ⁵⁸	Ni ⁶⁰	n, T	71d	β^+	2.3	1.7	Fe ⁵⁸
			Ni ⁵⁸	n, p					
		Co ⁵⁷	Ni ⁵⁸	n, d	271d	E	0.84	0.14	Fe ⁵⁷
		Co ⁵⁶	Ni ⁵⁸	n, T	78d	β^+	4.6	3.6	Fe ⁵⁶
		Fe ⁵⁹	Ni ⁶²	n, α	45d	β^-	1.6	1.3	Co ⁵⁹
			Ni ⁶¹	n, He ³					
		Fe ⁵⁵	Ni ⁵⁸	n, α	2.7y	E	0.23	--	Mn ⁵⁵
6.	Mn	Mn ⁵⁴	Mn ⁵⁵	n, 2n	312d	E	1.4	0.83	Cr ⁵⁴
7.	Cr	Cr ⁵¹	Cr ⁵²	n, 2n	27d	E	0.75	0.32	V ⁵¹
			Cr ⁵⁰	n, ν					
		V ⁴⁹	Cr ⁵⁰	n, d	331d	E	6.0	--	Ti ⁴⁹
		V ⁴⁸	Cr ⁵⁰	n, T	16d	β^+	4.0	2.4	Ti ⁴⁸

Table 3.2.4 (Cont'd)

	Impurity	Long lived daughter isotope	Parent isotope	Generating reaction	Half life	Decay mode	Decay energy (MeV)	Max γ energy (MeV)	Final daughter
8.	Ca	Ca^{47}	Ca^{48}	n, 2n	4.5d	β^-	2.0	1.9	Sc^{47} (U)
			Ca^{46}	n, γ					Ti^{47} (S)
		Ca^{45}	Ca^{46}	n, 2n	163d	β^-	0.26	0.01	Sc^{45}
			Ca^{44}	n, γ					
9.	K	Ar^{39}	K^{41}	n, T	269y	β^-	0.56	--	K^{39}
			K^{40}	n, d					
			K^{39}	n, p					
		Ar^{37}	K^{39}	n, T	35d	E	0.81	--	Cl^{37}
10.	Zr	Zr^{95}	Zr^{96}	n, 2n	65.5d	β^-	1.1	0.76	Nb^{95} (U)
			Zr^{94}	n, γ					
		Nb^{95}	--	--	35d	β^-	0.92	0.77	Mo^{95} (S)
		Zr^{89}	Zr^{90}	n, 2n	78h	β^+	2.8	0.90	Y^{89}
		Y^{91}	Zr^{92}	n, d	59d	β^-	1.55	1.2	Zr^{91}
			Zr^{91}	n, p					
		Y^{90}	Zr^{92}	n, T	64h	β^-	2.3	1.8	Zr^{90}
			Zr^{91}	n, d					
			Zr^{90}	n, p					
		Y^{88}	Zr^{90}	n, T	107d	β^+	3.6	1.8	Y^{88}
		Sr^{90}	Zr^{92}	n, He^3	29y	β^-	0.55	--	Y^{90} (U)
									Zr^{90} (S)

Table 3.2.4 (Cont'd)

Impurity	Long lived daughter isotope	Parent isotope	Generating reaction	Half life	Decay mode	Decay energy (MeV)	Max γ energy (MeV)	Final daughter
10. Cont'd								
	Sr ⁸⁹	Zr ⁹²	n, α	50d	β^-	1.5	0.9	Y ⁸⁹
		Zr ⁹¹	n, He					
11. Sn	Sn ¹²⁵	Sn ¹²⁴	n, γ	9.6d	β^-	2.4	2.3	Sb ¹²⁵ (U)
								Te ¹²⁵ (S)
	Sn ¹²³	Sn ¹²⁴	n, 2n	129d	β^-	1.4	1.1	Sb ¹²³
		Sn ¹²²	n, γ					
	Sn ¹²¹	Sn ¹²²	n, 2n	50y	β^-	0.39	0.04	Sb ¹²¹
		Sn ¹²⁰	n, γ					
	Sn ¹¹⁹	Sn ¹²⁰	n, 2n	245d	IT	0.065	0.024	Sn ¹¹⁹
		Sn ¹¹⁸	n, γ					
	Sn ¹¹⁷	Sn ¹¹⁸	n, 2n	14d	IT	0.16	0.16	Sn ¹¹⁷
		Sn ¹¹⁶	n, γ					
	Sn ¹¹³	Sn ¹¹⁴	n, 2n	115d	E	1.0	0.39	In ¹¹³
		Sn ¹¹²	n, γ					
	In ¹¹¹	Sn ¹¹²	n, d	2.8d	E	0.82	0.24	Cd ¹¹¹
	Cd ¹¹⁵	Sn ¹¹⁷	n, He ³	54h	β^-	1.45	0.53	In ¹¹⁵
	Cd ¹⁰⁹	Sn ¹¹²	n, α	453d	E	0.18	0.09	Ag ¹⁰⁹

4. Conceptual Blanket Designs

4.1 Mechanical and Thermal Aspects

The maximum operating temperature of SAP is $\sim 400^{\circ}\text{C}$. If the coolant exit temperature is also limited to $\sim 400^{\circ}\text{C}$, the thermal cycle efficiency is fairly low, on the order of 30%. This efficiency is probably acceptable, though it would increase the $\$/\text{KW}(\text{e})$ reactor cost.

It is possible to have coolant exit temperatures considerably above 400°C , however, since most of the fusion energy that is deposited in the blanket does not appear in the SAP structure, but rather in the moderator and lithium compound used for tritium breeding. In this case, the thermal cycle efficiency can be much greater than 30%. Depending on design, it should be possible to achieve thermal efficiencies of 35-50%.

We can achieve elevated coolant temperatures in either of two ways. In the first approach, a single coolant stream cools the SAP structure and then cools the moderator and lithium compound, which operate well above 400°C . The coolant exit temperature is now limited either by the maximum allowable operating temperature for the moderator and/or lithium compound, or by the ΔT chosen (i.e., the temperature use of the coolant through the SAP structure) and the energy fraction deposited in the SAP structure. One can optimize designs, but roughly

speaking, approximately one third of the energy released in the blanket appears in the SAP structure. Assuming a ΔT of 150°C for cooling the SAP structure (250 to 400°C), the coolant exit temperature from the module would then be $\sim 700^{\circ}\text{C}$ (assuming that the moderator and/or lithium compounds used could operate at such temperatures).

In the second approach, two coolant streams are used, one at relatively low temperatures ($\leq 400^{\circ}\text{C}$) to cool the SAP structure, and one at relatively high temperature (e.g., $\geq 1000^{\circ}\text{C}$) to cool the moderator and lithium compound. Material limitations then limit the exit temperature of the second coolant stream. In this approach, the blanket is somewhat more complex than in the first approach but the benefits of higher thermal efficiency may offset this.

Both approaches require thermal insulation (e.g., alumina) between the SAP structure and the rest of the blanket. The magnitude of the heat leak between the hotter and cooler regions depends on the thermal conductivity and thickness of the insulation. Evacuated insulation does not seem practical. Refractory materials of low density (with pores filled by helium) appear to be required, and the lowest thermal conductivity one can expect is on the order of 0.5 to $1.0 \times 10^{-2} \text{ w/cm}^{\circ}\text{K}$.

The allowable heat leak and the thermal conductivity fix the separation between hotter and cooler blanket regions.

This in turn strongly affects the physical scale of the regions. The blankets examined in this study are of the modular type, with the module diameters being on the order of 0.3 to 1 meter.

Figure 4.1.1 shows an idealized fusion blanket, with a first wall, a fast neutron zone (breeding in Li^7 or neutron multiplication in Be), a primary and secondary moderating zone (breeding in Li^6), and a shield zone. In the modular designs we have developed for this study, a given module has all of the above zones in it. Previous CTR modular blanket designs have not included the shield zone as part of the module. We have included this zone in the module, so that large numbers of modules can be easily removed from the blanket, if necessary. The individual modules will have very low residual gamma activation (~ 1 curie/module) and will require only minimum shielding to be handled.

The shield zone is usually conceived as involving layers of high density material (iron and/or lead) alternating with water layers, is operating at a relatively low temperature (e.g., 100°C).

In the blanket design examined in this study, the shield zone is the hottest part of the blanket since the helium coolant exits from the shield. Water cannot be used at the high temperatures contemplated (500°C), but metal hydrides (some of which have higher H densities than water) or graphite can be used.

IDEALIZED GEOMETRY OF FUSION REACTOR BLANKET

REGION	
1	PLASMA
2	VACUUM
3	FIRST WALL
4	FAST NEUTRON ZONE
5	FIRST MODERATING ZONE
6	SECOND MODERATING ZONE
7	SHIELD ZONE

REGION	TRITIUM BREEDING REACTIONS
4	Li^7 REACTIONS
5	Li^6 AND Li^7 REACTIONS
6	Li^6 REACTIONS
7	Li^6 REACTIONS (OPTIONAL)

**SCHEMATIC ELEVATION VIEW OF
SOLID BLANKET MODULE #1**

Figure 4.1.2

SCHEMATIC CROSS SECTION
OF SOLID BLANKET MODULE#1

Figure 4.1.3

BLANKET MODULE DESIGN NO.4

FIG. 4.1.4

SECTION A-A
(SEE FIG. 4.1.4)

BLANKET MODULE DESIGN NO. 4

FIG. 4.1.5

BLANKET MODULE DESIGN NO.7

FIG. 4.1.6

SECTION A-A
(SEE FIG. 4.1.6)

BLANKET MODULE DESIGN NO. 7
FIG. 4.1.7

Table 4.1.1

Blanket Designs Summary

Design #	Composition						Tritium release to	Max. exit temp.	# of He coolant streams
	Region #								
	1 & 2	3	4	5	6	7			
1	Plasma & vacuum	LiAl	TiD ₂	TiD ₂	TiD ₂	TiH ₂	He	~500°C (Type 1)	1
		SAP	LiAl	LiAl	LiAl	LiAl			
		Al ₂ O ₃	SAP	SAP	SAP	SAP			
			Al ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃			
2			LiAl	CaH ₂	CaH ₂	TiH ₂	He	~500°C (Type 1)	1
			SAP	LiAl	LiAl	SAP			
			Al ₂ O ₃	SAP	SAP	Al ₂ O ₃			
				Al ₂ O ₃	Al ₂ O ₃				
3			LiAl	C	C	TiH ₂	He	~500°C (Type 1)	1
			SAP	LiAl	LiAl	SAP			
			Al ₂ O ₃	SAP	SAP	Al ₂ O ₃			
				Al ₂ O ₃	Al ₂ O ₃				
4			Be	C	C	C	He	~700°C (Type 2)	1
			LiAl	LiAl	LiAl	TiC			
			SAP	SAP	SAP	B ₄ C			
			Al ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃	SAP			
5			Be	C	C	C	vacuum	~700°C (Type 2)	1
			SAP	LiAl	LiAl	TiC			
			Al ₂ O ₃	SAP	SAP	B ₄ C			
			LiAl	Al ₂ O ₃	Al ₂ O ₃	SAP			
6			BeO	C	C	C	He	~1700°C (Type 3)	2
			SAP	Li ₂ Al ₂ O ₄	Li ₂ Al ₂ O ₄	TiC			
			Li ₂ Al ₂ O ₄	SAP	SAP	B ₄ C			
						SAP			
7			"	"	"	"	vacuum	~1600°C (Type 3)	2
						Al ₂ O ₃			

Conventional high density shielding materials like iron are activated too strongly to be useful. A high density metal hydride, e.g., TiH_2 , could be used, or if graphite is desired instead of a hydride, a carbide like TiC_2 .

Figures 4.1.2 and 4.1.3 show an elevation and cross section view of a conceptual blanket module. In this design, termed type 1, the module is a cylindrical shell of SAP in which the solid blanket materials are placed. The module is closed at the bottom with a dished head which serves as the first wall. The helium coolant enters at the top of the module, passes down through the annular space between the SAP shell and the thermal insulator, and then returns through the interior of the module and exits through the outlet at the upper end. The thickness of the thermal insulator is approximately 1 cm.

All type 1 designs use as much Li^7Al as possible in the fast neutron zone to generate tritium through Li^7 reactions. The LiAl will limit the helium exit temperature, probably to something like 500°C . It probably would not be practical to make an exact stoichiometric LiAl alloy. If there is excess Al in the alloy or if Al (SAP) jackets it, some of the excess Al will form a liquid phase at $\sim 600^\circ\text{C}$. It therefore seems wise to limit the maximum LiAl temperature in the blanket to well below 600°C .

The other constraint associated with a type 1 design is that the tritium bred in the LiAl must be released to the

helium coolant. This is a result of depositing the major part of the 14 MeV neutron energy in LiAl. This energy must be transferred to the helium coolant at temperatures above 400°C. In other types of designs, where tritium breeding is predominantly through Li^6 absorptions, much less LiAl can be used, and thus the LiAl requires a much smaller fraction of the energy deposited by the 14 MeV neutrons. This then permits the option of placing LiAl outside the SAP pressure shell so that the bred tritium can be released directly to the vacuum region between the plasma and the first wall.

Various blanket compositions can be used in type 1 designs. In table 4.1.1, designs number 1 through 3 show some possible compositions for type 1 designs. There are many other possible type 1 designs. In this study we adopt the convention that the design refers to a specific set of blanket compositions in the various regions; if composition is changed in any of these regions, a new design # is used. Further, a letter designation after the design number (e.g., design #1A) refers to the specific set of blanket concentrations and dimensions used in the design.

The thermal cycle efficiency will be about the same for all type 1 designs since LiAl limits the helium exit temperature. Changes in moderator compositions, concentrations, and

dimensions will thus primarily affect the blanket activation and breeding ratio. These will be discussed in the next section.

The LiAl alloy can be in the form of rods made by powder metallurgy (to facilitate tritium diffusion from the LiAl it is probably necessary to have gas spaces between the particles of LiAl). These rods can be stacked in the module, with helium coolant flowing between the rods. Depending on the mechanical strength and radiation damage resistance of the LiAl alloy, it may be necessary to jacket the LiAl rods with a thin SAP tube, which would have sufficient holes to let the bred tritium escape to the helium coolant. (The SAP would have zero stress.)

If a metal hydride moderator is used, it will have to be enclosed in a sealed SAP tube to prevent loss of hydrogen. If graphite is used, it probably need not be jacketed.

At this point, we do not have enough knowledge to know what the safe operating stress in the SAP structure should be. Among other things this will depend on operating temperature, radiation damage effects, and the frequency of thermal cycling. For purposes of this study we take the maximum allowable stress in SAP to be 2500 psi, which is 25% of the yield stress. In any given design, the operating stress is a function of He coolant pressure, module dimensions and wall thickness, and temperature gradients in the SAP structure.

As an example, consider the module shown in Figures 4.1.2 and 4.1.3. The operating stress in the cylindrical shell is 2100 psi ($P_{\text{He}} = 10$ atm, module diameter = 30 cm, wall thickness = 1 cm). The dished head is 2 cm thick, and has a ΔT from the outer to inner surface of 20°C (total wall flux = $1 \text{ MW(th)}/\text{m}^2$) plasma radiation = $0.15 \text{ MW(th)}/\text{m}^2$). The maximum operating stress in the first wall (including thermal stress) is then ~ 1500 psi.

The range of helium coolant pressures that are feasible for these blanket designs appears to be ~ 10 to 20 atm. Pressures much below 10 atm degrade heat transport, while pressures much above 20 atm require excessive amounts of structure, due to the relatively low strength of SAP.

The first wall flux is also limited by the properties of SAP. The first wall has to be relatively thick to withstand the He pressure; this in turn causes a relatively large thermal stress. This can be overcome to some extent by using a ribbed or honeycomb construction as part of the first wall, in order that the necessary strength can be provided without causing an excessive ΔT across the first wall. However, the practical limit for first wall flux would appear to be $\sim 3 \text{ MW(th)}/\text{m}^2$.

On the type 2 blanket design (Figures 4.1.4 and 4.1.5) we depend on $(n,2n)$ fast neutron reactions in Be, followed by

subsequent neutron absorption in Li^6 to achieve good tritium breeding ratios. This means that a much smaller amount of LiAl is required in the module. If the Li^6Al (the Li must be enriched in Li^6) is suitably cooled, the temperature of the bulk of the module can now be well above 500°C , the limit for LiAl . In figures 4.1.4 and 4.1.5, the LiAl is located in the thermal insulator region, between the hot ($T \sim 700^\circ\text{C}$) and the cold ($T \sim 400^\circ\text{C}$) parts of the module. Part of the He coolant that passes down through the annular space between the module shell and the thermal insulator is channeled through the thermal insulator. This He stream both cools the LiAl and carries away tritium from it.

Approximately 30% of the He coolant is channeled through the thermal insulator; the remainder continues down the annulus, cools the first wall, and then returns through the interior of the module.

If a lithium compound that can withstand higher temperatures (e.g., $\text{Li}_2\text{Al}_2\text{O}_4$) and has satisfactory tritium release rates is used, it can be located in the hot region. This would eliminate channeling He coolant through the thermal insulator and would make module construction simpler.

With a type 2 design it is possible to place the LiAl outside the module shell so that the bred tritium can diffuse into the vacuum region between the first wall and the

plasma. The LiAl probably would be placed in SAP tubes that were bonded to the outside surface of the module shell. These tubes would be sufficiently open that the tritium could escape. Further, the LiAl temperature would have to be substantially higher (500 vs 350°C) than the SAP shell temperature to facilitate tritium release. Since the LiAl is a source of heat from the 4.7 MeV released per neutron absorption in Li^6 , this can be accomplished by thermally insulating the LiAl from the SAP tube that contains it, probably by a layer of Al_2O_3 .

The Be metal in zone 4 would probably be in the form of stacked rods of a few cm diameter, with He coolant between the rods. A similar construction could be used for the graphite in zones 5, 6, and 7, or one could use a solid block with coolant channel holes. The latter method of construction would probably not be as satisfactory because of radiation induced dimensional changes in graphite (total fluence $\sim 10^{22} \text{ n/cm}^2$). Graphite and Be swelling should not cause any blocking of the He flow channels. Further, each He channel must be properly sized so that the He flow rate in each channel is matched to the total energy deposited in the material cooled by the channel. This should not be difficult since all channels in the hot region will receive essentially the same amount of energy from the 14 MeV neutrons and γ rays. Further, this matching is more critical for type 1 design where the LiAl

temperature is limited to $\sim 500^{\circ}\text{C}$. For type 2 designs, materials like graphite and Be can tolerate temperatures well above the 700°C maximum coolant temperature (the limit established by the ΔT taken for the coolant across the SAP structure and the fraction of the total energy that is deposited in the SAP structure), provided swelling is not aggravated by temperature. Here mismatch is not very important; some channels can run hotter, others cooler than the average.

The choices of moderator materials for the type design is limited to Be, BeO, and graphite. No hydride materials seem suitable. Al_2O_3 is a possibility, but a somewhat thicker blanket would be required. Also, the thermal neutron diffusion length may be too low to permit LiAl to be placed outside the hot region, that is, too many of the neutrons reaching thermal energies in Al_2O_3 would be captured before they diffused out to the LiAl.

Figures 4.1.6 and 4.1.7 shows an elevation and cross section view of a type 3 module. There are two helium coolant streams. The low temperature stream cools the SAP module shell by circulating through tubes bonded to the outside of the shell. The tubes run down one half of the shell, across the first wall, and return up the other half of the shell. The tube separation can be fairly wide and the shell will still be cooled, because of the high thermal conductivity of SAP. For example, if the

first wall flux is $1 \text{ MW(th)}/\text{m}^2$ (total energy) and the separation between coolant tubes on the first wall is 6 cm, the ΔT between the tube wall and a point on the shell midway between tubes is only 20°C .

The high temperature He coolant stream comes in through channels in the thermal insulator and returns through the interior of the module. In the design shown in figures 4.1.4 and 4.1.7, the thermal insulator is $\text{Li}_2\text{Al}_2\text{O}_4$ and thus is also a source of heat and tritium through neutron absorptions in Li^6 . Alternatively, the $\text{Li}_2\text{Al}_2\text{O}_4$ could be placed in the hot region and an Al_2O_3 thermal insulator used. The latter alternative would permit faster tritium release rates and a somewhat higher average temperature for the coolant stream.

A third alternative is to use LiAl in tubes attached to the outside of the SAP module shell and let the bred tritium escape to the vacuum region.

Because of the high exit temperature desired for the hot helium coolant ($\sim 2000^\circ\text{K}$) the only moderator/neutron multiplier materials that can be used are graphite and BeO . Type 3 module diameters will have to be substantially greater, e.g. ~ 0.7 to 1 m. than those for type 1 and 2, due to the larger temperature difference between the hot and cool regions.

The use of 2 coolant streams instead of 1 has heat transport benefits. In the type 1 and 2 designs, the single

coolant stream first cools the SAP structure and then flows through the hot portion of the module. In order to get a helium exit temperature substantially above the 400°C limitation of SAP, the ΔT taken by the coolant across the SAP structure must be correspondingly large. Thus the helium inlet temperature must be relatively low, on the order of 250 to 300°C. In the type 3 design, the inlet temperature of the low temperature stream can be much closer to the exit temperature, e.g., the inlet temperature might be 375°C and the exit temperature 400°C. This will help to increase thermal cycle efficiency and reduce heat exchanger costs. Similar arguments apply to the high temperature stream.

A type 1 or 2 design is probably best coupled to a conventional steam cycle. A type 3 design could also couple to a conventional steam cycle, with the low temperature stream generating steam and the high temperature stream providing superheat. However, there are alternatives. The low temperature stream could generate steam for a separate turbine and the high temperature stream could be coupled to a direct cycle gas turbine, or a potassium topping cycle. It may even be feasible to use the high temperature stream in an MHD generator. The low temperature stream should give PWR level thermal efficiencies, e.g., 32%, for that portion of the heat carried away

by the low temperature streams (approximately one-third) while the high temperature stream could approach 55% thermal efficiency. The overall plant efficiency could then approach 50%.

It is possible to put a bank of separate SAP tubes carrying a separate He stream between the plasma and the dished heads of the blanket modules. These tubes would then function as a first wall. Most of the heat carried by the coolant from these tubes would be radiation from the plasma (~16% of the total energy generated by the DT fusion reaction, including neutron absorption energy), though a small fraction of the 14 MeV neutron energy would also be deposited in the tubes. This has certain advantages: the tube bank can have a number of layers, providing redundancy in the case of failure; and the conditions on the dished head of the modules are much gentler in terms of cooling, bombardment by plasma ions, and thermal stress. Further, the ΔT for the coolant in the tube bank could be small, which would increase thermal efficiency, and the helium inlet temperature for the modules could be greater, which would also increase efficiency. In type 3 modules the tubes on the outside of the SAP shell could be further apart if a tube bank is used.

Hopkins (10) has suggested the use of low activity SiC or pyrographite structural materials for CTR blankets. If the tube bank were made of SiC tubes, the first wall could operate

at much higher temperatures, e.g., 1000°C , and thermal cycle efficiency would increase. The SAP modules would then be placed behind the SiC tubes. One could contemplate also using SiC as a structural material for modules, but because of its brittle nature, it would seem much more suitable for small diameter tubes than for large complex modules.

The use of a tube bank will cause some degradation of breeding ratio, but probably not by a serious amount. The tube walls can be fairly thin, on the order of a couple of mm; further, some compensation could be made by having a thinner dished head for the module.

The designs in this study all employ cylindrical modules. This leaves some spaces between modules in the blanket/shield regions. To prevent neutron streaming it will be necessary to either fill these spaces with additional material and coolant circuits, or to change the shape of the modules so that they can fill the entire space. The latter approach would require non-circular modules, of varying cross sectional area along their length to accommodate the changing radius of the blanket. This is probably feasible, but more detailed blanket designs are needed for proper evaluation.

In an actual blanket the stress in the walls of the module should be substantially less than that calculated for individual modules, since adjacent modules can help to support

each other. If modules are designed to fill all space in the blanket, then there would be essentially no stress in the module side walls, though there would still be stress in the dished heads at the end of the modules. However, modules should probably still be designed as if there is no support from adjacent modules, i.e., vacuum is assumed outside the module. This precaution would permit operation if one module failed and had to be shut off.

Detailed stress calculations using two dimensional finite element codes are being carried out for the various types of modules, but are not yet complete. These calculations required a detailed thermal analysis with temperatures specified at all points in the module, so that thermal stress effects can be included.

Thermal analyses are also required to determine helium flow rates, pressure drops and thermal cycle efficiencies for the various designs.

Detailed thermal analysis requires use of a two dimensional neutron transport code to determine local neutron transport code to determine local neutron reaction rates plus a detailed breakdown of particle energies resulting from (n,α) , (n,p) , etc. reactions. Most importantly, it requires knowledge of the distribution of gamma ray sources, frequencies and intensities, in the blanket. The gamma ray source terms must then be used in conjunction with a gamma ray transport code to determine local heating rates.

Since time limitations did not permit this type of analysis for this report, the following approximate method is used:

1) Each zone is divided into cold, hot and insulator regions. Cold refers to the SAP structure at or below 400°C . Hot refers to the portions of the fast neutron, moderating and shield zone with temperatures above 400°C . Insulating refers to the low thermal conductivity (e.g., Al_2O_3) region separating hot and cold regions. In all designs zone 3 (first wall) has only a cold region. All other zones have hot, cold and insulator regions.

2) All of the particle energy (3.5 MeV) resulting from the DT reaction is assumed to be deposited on the first wall principally as bremsstrahlung radiation. This overestimates the load on the first wall, since some energy will be carried away as thermal energy of the hot plasma that escapes via the divertor. However, this should be a relatively small fraction of the 3.5 MeV source term.

3) The fast neutron reaction rates $[(n,2n), \text{etc.}]$ are then used to compute the local energy depositions due to particles resulting from the reaction. This included direct and indirect particle energy [e.g., the β^- particle from the Mg^{27} produced by an (n,p) reaction on Al^{27} . In the case of (n,γ) reactions, the γ photon is assumed not to deposit locally, but to deposit its energy in accordance with an average for all γ 's,

as discussed later. β particles resulting from daughters are assumed to deposit locally, however. A detailed breakdown of direct and indirect particle energies is not used because of the other approximations involved, and because the energy deposited via the direct and indirect particle route is a small part of the total energy balance. For our purposes we assume that direct particles (α, β, d, t) have an average energy of 2 MeV. The indirect particle energy depends on the particular isotope in question (e.g., an average energy of 1.7 is used for the $Mg^{27} \beta^-$ particle). Each zone is homogenized for the ANISN calculation, the zones are de-homogenized and the appropriate particle energy deposition assigned to each component. For example, in design 1A, 40% of the aluminum atoms in zone 4 belong to the cold SAP structure. After the total particle energy is calculated for all reactions on Al, 40% is then assigned to the cold SAP region. This introduces some error, since particles originating from nuclei in the SAP region could deposit some energy in the insulator zone, for example. However, similar events can occur with particles escaping from the insulator region into the SAP region, and the averaging process tends to make the errors relatively small.

4) The neutron energy deposition due to slowing down is then computed for each elemental component in each zone, and

the zone de-homogenized in the same manner as in 3. We thus know the neutron slowing down energy deposition for each region (hot, cold, and insulator) in each zone.

5) The total energy deposited via routes (3) and (4) are summed in terms of energy deposited/14 MeV neutron. The energy difference between this sum and 14.1 MeV is then taken as the γ ray energy. This approximation is more accurate than attempting to directly estimate the γ energy, as the energy deposited via routes (3) and (4) can be calculated more exactly, and because the sum of (3) and (4) is still a small part of the 14 MeV total.

6) The γ energy deposition is then apportioned among the components in all zones according to the relative masses of the components. Thus if SAP accounts for 25% of the total mass in the blanket (zones 3-6), 25% of the γ energy is deposited in SAP. This approximation rests on three assumptions: the mass attenuation coefficient μ_a/ρ , are the same for each element, and that γ ray path lengths are comparable to or exceed zone dimensions. The first assumption is fairly realistic over the range of 1 to 10 MeV, where most of the γ ray energy appears. The portion assigned to Li and Be tend to be overestimated by ~20%. The second assumption is more questionable, however, since zone dimensions tend to be greater than γ ray path lengths.

Table 4.1.2

Energy Deposition Balance

<u>Energy Deposition Mode & Temperature of Region</u>	<u>Module Design</u>		
	<u>Total Energy Deposited in Zones 3, 4, 5, & 6</u>		
	<u>#1A (MeV)</u>	<u>#4A (MeV)</u>	<u>#6A (MeV)</u>
<u>Hot Region</u>			
γ	7.60	3.20	4.8
α, β, p, d	0.366	1.62	0.90
$\text{Li}^6 + n$	3.83	7.11	5.18
Neutron slowing down (elastic)	<u>4.9</u>	<u>4.9</u>	<u>4.9</u>
Sub-total	14.7	16.83	15.8
<u>Cold Region</u>			
γ	3.8	1.60	2.6
α, β, p, d	0.434	0.44	0.47
$\text{Li}^6 + n$	-	-	-
Neutron slowing down (elastic)	-	-	-
Bremstrahlung from plasma	<u>3.5</u>	<u>3.5</u>	<u>3.5</u>
Sub-total	7.7	5.54	6.6
Total, Hot plus Cold Regions	22.4	22.4	22.4

Note: Units of energy are MeV/DT fusion.

Table 4.1.3

Thermal Balance for Modules

<u>Parameter</u>	<u>Module Design</u>			
	1A	4A	6A	
			<u>Cold He Stream</u>	<u>Hot He Stream</u>
Module Diameter, cm	30	45	75	
Total Module Power,* KW(th)	70	160	440	
Deposited Power in Cold Structure, KW(th)	24	40	130	-
Deposited Power in Hot Structure, KW(th)	46	120	-	310
Power Removed from Cold Structure,** KW(th)	30	51	162	-
Power Removed from Hot Structure,** KW(th)	40	109	-	278
Helium Flow Rate, lbs/sec per module	0.13	0.17	1.35	0.20
Inlet He Temperature, °C	280	245	300	1100
Outlet He Temperature, °C	500	650	350	
ΔP Across Module, psi ($P_{He}=10$ Atm)	2	2	2	2
$(\Delta T_{wall})_{max}$, °C	25	25	50	-

* $1 \text{ MW(th)}/\text{m}^2$ First Wall Flux (22.4 MeV/DT Fusion)

** Includes Heat Leak from Hot to Cold Region

7) The remaining energy input term is the energetic α (4.8 MeV) released by neutron absorption in Li^6 . No γ photon results from this absorption, and the α energy is released locally: in type 1 designs all of it appears in the hot region; in type 2 all of it appears in the insulator (if LiAl is used) or in the hot region (if a high temperature Li compound is used); in type 3 it either goes to the hot region or the insulator. The Li^7 reactions are a small part of the total neutron balance for type 2 and 3 designs and the α energy resulting from these reactions can be neglected.

8) The additional energy resulting from neutron captures in materials other than Li^6 is relatively small and can be neglected.

The various input terms are summed and appear in Table 4.1.2. In each design most of the energy appears in the hot region.

A thermal analysis of each design is then made, using the energy inputs determined by the above approximation. A summary of these thermal analyses is given in Table 4.1.3.

7) The remaining energy input term is the energetic α (4.8 MeV) released by neutron absorption in Li^6 . No γ photon results from this absorption, and the α energy is released locally: in type 1 designs all of it appears in the hot region; in type 2 all of it appears in the insulator (if LiAl is used) or in the hot region (if a high temperature Li compound is used); in type 3 it either goes to the hot region or the insulator. The Li^7 reactions are a small part of the total neutron balance for type 2 and 3 designs and the α energy resulting from these reactions can be neglected.

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The various input terms are summed and appear in Table 4.1.2. In each design most of the energy appears in the hot region.

A thermal analysis of each design is then made, using the energy inputs determined by the above approximation. A summary of these thermal analyses is given in Table 4.1.3.

4.2 Neutron Activation and Breeding

Representative compositions and dimensions selected for the type 1 (design #1A), type 2 (design #4A), and type 3 (design #6A) blankets. These designs are not optimized because of limited time.

Each zone in the module is homogenized for the one dimensional ANISN calculations. The compositions, volume fractions, dimensions and homogenized atomic densities for each zone are given in Tables 4.2.1, 4.2.3, and 4.2.5.

The cross sections used in this work were based on ENDF/B-III (11). For all of the nuclides except titanium, the cross sections used were processed into a multi-group set at Oak Ridge National Laboratory using the SUPERTOG (12) program and distributed through the Radiation Shielding Information Center as RSIC Data Library DLC-2/100G (13). This set represents up to a P_8 approximation to elastic scattering angular distributions, and a 100-group structure with energy boundaries identical to those in the GAM-II library (14). Only downscattering is allowed, and group 100 is a thermal group with cross section values based on a Maxwellian average at .0253 ev.

For ENDF/B-III titanium (MAT No. 1144) the processing was done at Brookhaven using the ETOG-2 version of the ETOG (15) program to generate a P_3 multigroup set with the same GAM-II group structure.

Retrieval programs are used to generate a P_0 cross section library tape for the ANISN (16) program from these cross section sets.

The neutron flux distributions are calculated using the ANISN program. The geometry is represented as an infinite cylinder with a vacuum boundary condition at the outer radius. A P_0 option is used for the order of anisotropic scattering, and an S_4 option for angular quadrature. The 14.1 MeV source neutrons are taken to have a uniform spatial distribution in the plasma region. In the GAM-II group structure, this energy falls within the boundaries of the highest energy group (13.5-14.918 MeV). For the neutronics calculations, the magnitude of the source is normalized to one neutron input to the system.

For the purpose of obtaining reaction rates, the ENDF/B-III data tapes are processed to provide group averaged cross sections within the GAM-II energy group structure, for all reactions for which data was available, for each of the nuclides present in the system. A neutron balance is obtained using these cross sections, the appropriate nuclide abundance, and the flux distributions computed by the ANISN program.

The resulting neutron balances are given in Tables 4.2.2 (design #1A), 4.2.4 (design #4A), and 4.2.6 (design #6A). The principal activations for these designs (not including impurities in aluminum) are given in Table 4.2.7.

The breeding ratio for design #1A is only 0.89, a result of the poor tritium production in Li^7 (0.1 per fusion neutron). The Al cross sections for fast neutrons are fairly low, but the aluminum first wall is much thicker and the Al/ Li^7 ratio much greater than the corresponding values for a Nb-Li blanket. The aluminum parasitically absorbs a significant fraction of the fast neutrons (~ 0.15) and degrades neutron energy so that the number of Li^7 reactions is reduced.

In this design the Al/Li ratio is 1.93 in zone 4 and the first wall thickness is 2 cm. By going to somewhat lower helium coolant pressures the aluminum can be significantly reduced and the breeding ratio increased. It is doubtful that the breeding ratio can ever be significantly greater than 1.0, however.

The titanium and oxygen have relatively little effect on the neutron balance. Breeding ratio could be slightly increased (on the order of a few hundredths) by increasing the thickness of the fast neutron zone (zone 4) since the fast neutron flux at the inner part of zone 5 is still significant. Increasing the thickness of the moderating zone (zone 5 and 6) would have almost no effect on the breeding ratio.

The fast neutron energy flux at the far boundary of the shield zone (zone 7) is $\sim 10^{-6}$ of the 14 MeV energy flux at the first wall. At a total first wall energy flux of

1 MW(th)/m², the leakage energy is small enough that its input to the superconducting magnets will be tolerable.

The blanket activation in design #1A is about what would be expected on the basis of earlier analysis except that the activation due to titanium (Ca⁴⁵ and the Sc isotopes) is too large. It appears that titanium hydride is not a suitable moderator if low blanket activity is required. It still could be used in the shield zone, however. Titanium hydride has been previously considered as a high temperature shield material. It has a high hydrogen density, a good inelastic fast neutron cross section, and good stopping power for gamma rays.

CaH₂ is probably a more suitable hydride moderator than TiH₂ or TiD₂ and is listed in design #2. Ca would activate to about the same degree as Ti, but the principal activity from Ca is A³⁷, a relatively non-hazardous isotope. The half-life of A³⁷ is relatively short, 35 days, and the allowable release concentration to the environment is quite high, 10⁻⁴ μCi/ml (17). In the very worst case, where all the blanket activity were dispersed in the air around the reactor, only about 1 km³ of air is required to dilute the A³⁷ to safe levels. This assumes that the allowable A³⁷ concentration in air for a single accidental discharge of the relatively short half life material is 100 times the concentration allowed for continuous restricted discharge to the environment.

The third alternative for type 1 blankets is to use graphite as the moderating material (design #3). This probably has the smallest inventory of long-lived activations, but requires a somewhat thicker blanket.

The optimum design for type 1 blankets will be affected by many factors, and some compromises will have to be made to get the best combination of low activity in the blanket, a thin blanket, high thermal efficiency, good breeding ratio, etc. On the basis of the neutronic calculations for design #1A, some clear directions for type 1 blankets emerge, however:

- 1) The Li^7 contribution to the breeding ratio is small. It can be increased by a somewhat thinner first wall and lower Al/Li ratio, but the increase will not be great.

- 2) If metal hydride moderators are to be used in the moderating zone (zones 5 and 6), the protide is preferable to the deuteride (e.g., TiH_2 is preferable to TiD_2). The only reason for using the deuteride in design #1 was to avoid the possibility of adverse flux disadvantage factors. The metal hydride moderator must be contained in separate tubes (typically 0.5 to 1.0 inch diameter) from the LiAl alloy. This arrangement might cause excessive enhanced neutron absorption in hydrogen (protium) if the neutrons are predominantly thermal. The neutronics calculations indicate that the neutron absorption

in Li^6 peaks at 20 ev, however, and that thermal neutrons contribute very little to the breeding ratio. The neutron spectrum is thus sufficiently non-thermal that the actual heterogeneous moderator-absorber structure can be considered as homogeneous without any significant error, even if protium is used as the moderator.

3) The contribution of the lithium in the shield zone to the breeding ratio is negligible. LiAl can then be eliminated in the shield and more effective shielding materials substituted.

4) The volume fraction of helium coolant in the shield zone is excessive, considering the small heat deposition per unit volume and the low flux. The He volume fraction in the shield zone can probably be reduced to a few percent. The helium volume fraction in the moderating zone could also be reduced but not by as much as in the shield zone, since heat deposition and radiation damage rates are greater.

5) By combining these improvements, the total blanket thickness (shield included) could be substantially reduced from the 1.6 meter value used in design 1A, while still meeting the condition that the neutron energy flux out of the shield be $\sim 1 \text{ watt/m}^2$. However no gamma ray shielding calculations have been done yet, and they may be limiting rather than the neutrons.

In design 4A the breeding ratio is considerably improved, reaching a value of 1.49, because of $n,2n$ reactions in beryllium. The breeding ratio is sufficiently large that considerable amounts of surplus tritium could be generated for other purposes, including making up the tritium deficiency of CTR reactors with breeding ratios less than one. The large breeding margin would also permit one to use much larger amounts of SAP structure, for example, a thicker first wall. This could be particularly important if the operating stress in the SAP structure must be much lower than 2000-3000 psi.

Approximately 500 lbs of Be metal is required per m^2 of first wall in design 4A. Since the Be metal is only used as a neutron multiplier and not a structural material, one does not need the high quality metal normally used for structures. The metal can have numerous cracks and flaws, and should be considerably cheaper than the cost of structural Be metal. Even so, cost may make it desirable to reduce the Be inventory. At \$10/lb the Be metal in design 4A would cost \$12/KW(e), for first wall flux of $1 \text{ MW}/m^2$. The detailed reaction rates from the neutronics calculations indicate that if the thickness of the Be zone were cut in half, the breeding ratio would be lower, 1.3, but still perfectly good for a self sustaining CTR.

After 30 years of operation at $1 \text{ MW}/m^2$, 1.3% of the Be

atoms would be destroyed by $(n,2n)$ and (n,α) reactions. This is the average fraction destroyed in the 30 cm thick Be zone; at the inner edge of the zone, the fraction destroyed would be ~4%.

The breeding ratio in design #6A is 1.08, using BeO as the neutron multiplier. This is adequate for a self sustaining CTR reactor, but it leaves very little leeway to increase the amount of structure, if necessary. The breeding ratio is lower than that in design 4A for three reasons:

- a) The lower Be atomic density increases parasitic absorption in the SAP structure.
- b) The much greater O^{16} density caused by using BeO degrades the neutron spectrum more rapidly.
- c) The thicker SAP first wall increases parasitic absorption.

Because of the low breeding ratio, the thickness of the BeO zone cannot be significantly reduced. Moreover, increasing the BeO thickness would not significantly increase breeding ratio since only 7% of the tritium breeding occurs in the graphite moderating zone (zones 5 and 6).

Titanium activation in the shield zone for design 6A is only 10^4 curies, which is probably acceptable since the daughter scandium isotopes have short half lives.

The thicker blanket and shield in design 6A (1.8 m vs.

1.6 m for designs 1A and 4A) reduces the neutron energy flux out of the shield to $\sim 0.01 \text{ watt/m}^2$, which is well under the maximum allowed flux of 1 w/m^2 .

Table 4.2.8 lists transmutation concentrations produced in the SAP first wall after 30 years of operation at 80% load factor and 1 MW/m^2 . This will be the critical structural region. The outer shell of the module will be operating at lower stress and will see a lower flux. The concentrations shown are for design 4A, but are quite similar for the other designs.

The activations due to impurities in aluminum have not yet been calculated for these designs. The ENDFB reaction cross sections are not available for the most part; however, they can be calculated with sufficient accuracy ($\pm 20\%$) for our purposes. Activations from all impurities in the various blanket designs will be calculated and presented in our next report. Since the impurities are present in very low concentrations, the ANISN generated flux spectrums will be unchanged and can be used to calculate the impurity activation.

The one-dimensional ANISN calculations should be adequate for comparing designs, evaluating the effects of various compositions and blanket dimensions, and calculating breeding ratios. However, the one dimensional model may not be adequate for determining heat deposition rates in different parts of

Table 4.2.1

Blanket Design #1A

<u>Compositions</u>	<u>Zone</u>				
	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>
Materials (%)					
1. SAP	100	17.44	21.0	21.0	21.0
2. Al ₂ O ₃	-	11.38	11.38	11.38	11.38
3. LiAl	-	52.06	26.03	26.03	26.03
4. TiD _{1.5}	-	-	22.47	22.47	-
5. TiH _{1.5}	-	-	-	-	22.47
6. Helium coolant	-	19.12	19.12	19.12	19.12
Atomic Densities (x10²²)					
1. Al (natural)	5.88	2.70	2.205	2.205	2.205
2. O ¹⁶	0.515	0.493	0.511	0.511	0.511
3. Li ⁶	-	0.142	0.632	0.632	0.632
4. Li ⁷	-	1.265	0.0703	0.0703	0.0703
5. Ti (natural)	-	-	0.845	0.845	0.845
6. D ²	-	-	1.27	1.27	-
7. H ¹	-	-	-	-	1.27
Thickness, cm	2	30	25	25	80

Table 4.2.2

Neutron Balance for Blanket Design # 1A

<u>Nuclide and reaction</u>		<u>Zone 3</u>	<u>Zone 4</u>	<u>Zones 5&6</u>	<u>Zone 7</u>
Al ²⁷	n, 2n	.010512	.017794	.0010896	.00001354
	n, γ	.0021246	.012557	.00338	.000021688
	n, p	.022603	.050575	.0045654	.000077491
	n, d	.0065754	.012199	.00085407	.000012055
	n, t	.000031229	.00005311	.0000032756	.000000041
	n, α	.032348	.065822	.0052436	.000082195
O ¹⁶	n, γ	1.32x10 ⁻⁸	1.65x10 ⁻⁷	6.67x10 ⁻⁸	1.26x10 ⁻⁹
	n, p	.00089801	.00356	.00032933	.0000048134
	n, d	.00030952	.0012041	.00010719	.0000015108
	n, α	.0030708	.014283	.0016290	.000027528
Li ⁶	n, 2n	--	.0018591	.00077143	.00001169
	abs	--	.00040837	.00025687	.000004698
	n, t	--	.30259	.47512	.0085838
Li ⁷	n, 2n	--	.012289	.00005969	.000000857
	abs	--	.0023216	.000012986	.0000001885
	n, t	--	0.10478	.00065622	.000011261
Ti	2, 2n	--	--	.0011212	.000014661
	n, γ	--	--	.0063199	.000083067
	n, p	--	--	.0021748	.000034422
	n, α	--	--	.000092504	.0000014581
D ²	n, 2n	--	--	.004852	--
	n, γ	--	--	.000015192	--
H ¹					
	n, γ	--	--	--	.0000057576

Breeding ratio = 0.89 tritons/fusion neutron

Table 4.2.3

Blanket Design # 4A

	Zone				
	3	4	5	6	7
Compositions					
Material (%)					
1. SAP	100	17	20	20	20
2. Al_2O_3	--	11.0	11.0	11.0	11.0
3. LiAl	--	10	10	10	--
4. Graphite	--	--	42	42	--
5. Be Metal	--	45	--	--	--
6. Ti(H) _{1.5}	--	--	--	--	52
7. Helium Coolant	--	17	17	17	17
Atomic Densities ($\times 10^{22}$)					
1. Al (natural)	5.88	1.53	1.70	1.70	1.70
2. O^{16}	0.515	0.477	0.492	0.492	0.492
3. Be (natural)	--	5.4	--	--	--
4. C (natural)	--	--	4.41	4.41	--
5. Li^6	--	0.243	0.243	0.243	--
6. Li^7	--	0.027	0.027	0.027	--
7. Ti (natural)	--	--	--	--	1.955
8. H^1	--	--	--	--	2.933
Thickness, cm	2	30	25	25	80?

Table 4.2.4

Neutron Balance, Blanket Design # 4A

<u>Nuclide and reaction</u>		<u>Zone 3</u>	<u>Zone 4</u>	<u>Zones 5 & 6</u>	<u>Zone 7</u>
Al²⁷					
	n, 2n	.01070	.00718	.00019	.00000
	n, γ	.00350	.00823	.00122	.00013
	n, p	.02511	.02530	.00134	.00001
	n, d	.00675	.00506	.00016	.00000
	n, t	.00003	.00002	.00000	.00000
	n, α	.03405	.02921	.00119	.00000
Total	Abs	.06944	.06782	.00391	.00014
O¹⁶					
	n, γ	.00000	.00000	.00000	.00000
	n, p	.0092	.00254	.00008	.00000
	n, d	.00032	.00085	.00003	.00000
	n, n	.00339	.01201	.00059	.00000
Total	Abs	.00463	.01540	.00070	.00000
Be					
	n, 2n	--	.71694	--	--
	n, γ	--	.00105	--	--
	n, p	--	.00005	--	--
	n, d	--	.00000	--	--
	n, t	--	.00000	--	--
	n, α	--	.06794	--	--
C					
	n, γ	--	--	.00000	--
	n, α	--	--	.00398	--
Li⁶					
	n, 2n	--	.00242	.00008	--
	n, t	--	1.3112	.17144	--
Total	Abs	--	.00079	.00006	--
Li⁷					
	n, 2n	--	.00019	.00000	--
	n, t	--	.00205	.00010	--
Total	Abs	--	.00004	.00000	--
Ti					
	n, 2n	--	--	--	.00000
	n, γ	--	--	--	.00378
	n, p	--	--	--	.00001
	n, α	--	--	--	.00000
H					
	n, γ	--	--	--	.00031

Breeding Ratio = 1.49 tritons/fusion neutron

Table 4.2.5

Blanket Design # 6A

<u>Compositions (Vol %)</u>	<u>Zone</u>				
	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>
1. SAP	100	15	15	15	15
2. $\text{Li}_2\text{Al}_2\text{O}_4$ (low density)*	--	30	30	30	--
3. Al_2O_3 (low density)*	--	--	--	--	30
4. Graphite	--	--	42	42	20
5. Beryllium Oxide	--	42	--	--	--
6. Titanium Carbide	--	--	--	--	20
7. Boron Carbide	--	--	--	--	2
8. Helium Coolant	--	13	13	13	13
 <u>Atomic Densities ($\times 10^{22}$)</u>					
1. Al (natural)	5.88	1.16	1.16	1.16	1.45
2. O^{16}	0.515	3.68	0.64	0.64	0.93
3. C (natural)	--	--	4.74	4.74	3.30
4. Be (natural)	--	3.04	--	--	--
5. Li^6	--	0.252	0.252	0.252	--
6. Li^7	--	0.028	0.028	0.028	--
7. Ti (natural)	--	--	--	--	0.99
8. Boron (natural)	--	--	--	--	0.22
Thickness, cm	3	40	30	30	80

* 40% of theoretical density

Table 4.2.6

Neutron Balance, Blanket Design # 6A

<u>Nuclide and reaction</u>		<u>3</u>	<u>4</u>	<u>5&6</u>	<u>7</u>
Al 27					
	n, 2n	.01474	.00441	.00002	0.0
	n, γ	.00419	.00537	.00032	0.0
	n, p	.0334	.01556	.00025	0.0
	n, d	.00929	.00314	.00002	0.0
	n, t	0.000 04	.0001	0.0	0.0
	n, α	.04626	.01796	.00019	0.0
Total	Abs	.07786	.04645	.00080	0.0
O-16					
	n, γ	0.0	0.0	0.0	0.0
	n, p	.00127	.01613	.00002	0.0
	n, d	.00044	.00538	0.0	0.0
	n, α	.00453	.07583	.0021	0.0
Total	Abs	.00624	.09734	.00023	0.0
Be					
	n, 2n	--	.34091	--	--
	n, γ	--	.00050	--	--
	n, p	--	.00002	--	--
	n, d	--	0.0	--	--
	n, t	--	.00547	--	--
	n, α	--	.03766	--	--
Total	Abs		.38456		
C					
	n, γ	--	--	0.0	0.0
	n, α	--	--	.00101	0.0
Li-6					
	n, 2n	--	.00204	.00002	--
	n, t	--	1.0089	.07011	--
Total	Abs	--	.00069	.00002	--
Li ⁷					
	n, 2n	--	.00016	0.0	--
	n, t	--	.00173	.00003	--
Total	Abs	--	.00003	0.0	--
Ti					
	n, 2n	--	--	--	1.3x10 ⁻⁷
	n, γ	--	--	--	3.0x10 ⁻⁵
	n, p	--	--	--	7.2x10 ⁻⁷
	n, α	--	--	--	2.7x10 ⁻⁸
Total	Abs	--	--	--	3 x10 ⁻⁵
B-10					
	n, d	--	--	--	0.0
	n, t	--	--	--	0.0
	n, α	--	--	--	.00057

B-11
(negligible)

-82-

Breeding ratio = 1.08 tritons/fusion neutron

Table 4.2.7

CTR Blanket Activation

<u>Nuclide</u>	<u>Half-Life</u>	<u>Generation</u>	<u>Curies in Blanket</u> <u>(30 years at 1000 MW(e))</u>		
			<u>#1A</u>	<u>#4A</u>	<u>#6A</u>
1. Al ^{26*}	7.3x10 ⁵ y	1st	1.1x10 ⁴	7x10 ³	7.4x10 ³
2. Be ¹⁰	1.6x10 ⁶ y	1st	-	350	180
3. C ¹⁴	5.7x10 ³ y	1st	~30	~30	~150
4. Na ²²	2.6 y	3 d	~10	~10	~10
5. Ca ⁴⁵	163 days	1st	~10 ⁶	~4000	~400
6. Sc ⁴⁶ , Sc ⁴⁷ , Sc ⁴⁸	84, 3.4, 1.9 days	1st	~4x10 ⁷	1.2x10 ⁵	~1.2x10 ⁴
7. Na ²⁴	15 hour	1st	1.9x10 ⁹	1.2x10 ⁹	1.2x10 ⁹

Activations of impurities not included.

* 50% yield of long lived Al²⁶ isomer.

Table 4.2.8

Transmutation Concentrations in SAP First Wall

Basis: 30 years of reactor operation
 80% load factor
 1 MW/m²
 nvt (=14 MeV) = 2.1×10^{22}
 Design #4A

<u>Transmutation Element</u>	<u>Concentration Atom %</u>
Si	0.060
Mg	0.76
Na	$\sim 4 \times 10^{-4}$
N	0.13
C	0.056
H	0.55
He	0.62

the module. Most of the 14 MeV neutron energy is deposited in the first 20 cm of the fast neutron zone. The diameter of the modules will be on the order of 30 to 100 cm, which is comparable to, or exceeds the depth of the fast neutron zone. It is therefore planned to carry out some 2 dimensional neutronic calculations on selected blanket designs, primarily for thermal design. However, breeding ratios and blanket activations will also be computed with the 2 dimensional code and the results compared with those from a 1 dimensional code.

4.3 Tritium Recovery from Blanket

In the blanket designs considered here, the tritium is bred in situ in solid lithium alloys (e.g., LiAl) or lithium ceramics (e.g., $\text{Li}_2\text{Al}_2\text{O}_4$). The tritium could be recovered either by letting the tritium diffuse from the solid and then be recovered, or by periodically removing the solid from the blanket and processing it. This latter approach is not very attractive because of cost, difficulty, and the large tritium inventories caused by the necessary infrequent processing. Leaving the solid in the blanket and recovering the bred tritium by diffusion from the solid seems to be quite feasible. The tritium holdup in solid breeding material in the blanket is small, on the order of 1 days production. We will concentrate on this method of tritium recovery in the rest of this study.

The simplest form of tritium recovery is to let the bred tritium diffuse directly into the vacuum region between the plasma and first wall, as in blanket design #7. The LiAl alloy is outside of the SAP module pressure shell, and the tritium that evolves from it diffuses along the spaces between modules into the vacuum region.

Two modes of tritium recovery are possible with this design:

a) The tritium atom fraction in the LiAl alloy approaches a steady state value with tritium diffusing continuously from the alloy at the same rate that it is bred. Since the amount of surface area is very large and reaction rates are rapid, this steady state concentration is essentially equal to the atom fraction in the alloy when it is at equilibrium with the partial pressure of tritium in the vacuum region.

b) If the tritium inventory in the blanket using method (a) is too large, it can be reduced by a scavenging technique. In this method, a high protium concentration is built up in the LiAl alloy by having a high pressure of protium (i.e., H_2 gas) for a short time. The protium gas is then pumped away and the evolved gas carries tritium with it. This cyclical process can keep the tritium inventory in the blanket much smaller than that with method (a).

Let us consider method (a) first: the pressure of tritium gas (T_2) above a dilute solution (single phase) of tritium atoms in LiAl is given by Sieverts Law

$$(P_{T_2})_{LiAl} = (K_T)_{LiAl}^2 [X_T]^2 \quad \text{torr} \quad (4.3.1)$$

where X_T = atom fraction of T in LiAl

$(K_T)_{LiAl}$ = Sieverts constant for T in LiAl (a function of temperature)

Sieverts' constants for H, D, and T in lithium are given by Maroni (18). No experimental information is available for hydrogen isotopes in LiAl alloy, though it is known that Sieverts' constants for LiAl will be much greater than those measured in lithium, since there is a strong reaction between Li and Al. The Sieverts' constants for LiAl can be estimated by the following approximations:

$$\left[\frac{(K_H)_{Li}}{(K_H)_{LiAl}} \right]^2 \approx \frac{\left(\frac{P'_{H_2}}{P'_{H_2}} \right)_{Li}}{\left(\frac{P'_{H_2}}{P'_{H_2}} \right)_{LiAl}} \quad (4.3.2)$$

$$(K_T)_{LiAl} \approx (K_H)_{LiAl} \approx (K_D)_{LiAl} \quad (4.3.3)$$

where $\left(\frac{P'_{H_2}}{P'_{H_2}} \right)_{Li}$ = H plateau pressure with H in lithium
(two phase region, LiH in Li)

$\left(\frac{P'_{H_2}}{P'_{H_2}} \right)_{LiAl}$ = H₂ plateau pressure with H in LiAl alloy
(two phase region, LiAl H₄ in Li)

The plateau pressures are at the temperature of interest (e.g., 500°C).

Equation 4.3.2 is exact if Sieverts Law is obeyed up to the boundary of the two phase plateau region. The Sieverts' constants for the different hydrogen isotopes are equal within ~50%, so this only introduces a small error.

$\left(\frac{P'_{H_2}}{P'_{H_2}} \right)_{Li}$ is given as a function of temperature by

Maroni (18) and $(P_{H_2})_{LiAl}$ is given by Salzano and Aronson (19) with these values (see Figure 4.3.2)

$$(P_{T_2})_{LiAl} \approx 2 \times 10^5 (X_T)_{LiAl}^2 \text{ torr}$$

temperature
 $\approx 500^\circ\text{C}$ (4.3.4)

If the vacuum region between the plasma and the first wall is maintained at a pressure of 10^{-6} torr, and for simplicity, if the background gas is assumed to be 100% tritium, the equilibrium atom fractions of tritium in LiAl alloy is $\sim 2 \times 10^{-6}$. A 1000 MW(e) reactor would then have a LiAl alloy blanket inventory of $\sim 750 \text{ m}^3$. At a tritium atom fraction of 2×10^{-6} , this tritium inventory would be ~ 160 gms or 2×10^6 curies. At a thermal cycle efficiency of 40%, ~ 300 gms of tritium will be burned in the plasma per day, and approximately the same amount will be bred in the blanket. Thus a tritium atom fraction of 2×10^{-6} represents less than a days inventory. This will not cause any problems with doubling time and inventory charges are negligible.

This tritium inventory is probably acceptable from a hazards standpoint, and is comparable to the inventory that would be expected for the refueling portion of the tritium cycle. If it is desired to further reduce the tritium blanket inventory, it is unlikely that it could be done by further

reducing the vacuum pressure, since 10^{-6} torr seems to be a practical lower limit. However, scavenging of the tritium by additions of protium seems to be a realistic approach.

Consider the following initial conditions in LiAl alloy:

$$(X_T)_{\text{initial}} \ll (X_H)_{\text{initial}} \approx (X_H)_{\text{EQ}} \approx (X_T)_{\text{EQ}} \quad (4.3.5)$$

where $(X_H)_{\text{EQ}}$ and $(X_T)_{\text{EQ}}$ are the concentrations that would exist in LiAl alloy under a pressure of 10^{-6} torr. If a protium pressure $\gg 10^{-6}$ torr is allowed to build up over the LiAl alloy, the protium concentration then would increase to

$$(X_H)^* \gg (X_H)_{\text{EQ}} \quad (4.3.6)$$

the pressure is then reduced to the normal operating pressure of 10^{-6} torr, then the final concentrations are

$$(X_T)_{\text{final}} = (X_T)_{\text{initial}} \frac{(X_H)_{\text{initial}}}{(X_H)^*} \quad (4.3.7)$$

$$(X_H)_{\text{final}} \approx (X_H)_{\text{initial}}$$

The tritium is evolved at HT molecules when the excess protium is removed. The reactor would then operate normally for awhile, until the tritium built up to the point where another batch processing became desirable. As an example, taking $(X_T)_{\text{initial}} = 10^{-7}$, $(X_T)_{\text{EQ}} = 2 \times 10^{-6}$, $(X_H)^* = 2 \times 10^{-5}$, then $(X_T)_{\text{final}} = 10^{-8}$.

The protium pressure necessary to scavenge the tritium is $p^* = 10^{-4}$ torr, and the scavenging must be carried out every 40 minutes. Using this scavenging method the maximum tritium blanket inventory is only 10^5 curies instead of 2×10^6 curies. The extracted tritium would be present in a concentration of ~1% in the evolved gas, and would be easy to extract from the scavenging protium. Separation could be carried out either by reaction with metal hydrides, distillation of liquid hydrogen or by distillation of tritiated water after conversion of the evolved hydrogen to water. This example is chosen to show the potentialities of the scavenging process and is not optimized.

The cyclical scavenging process could be carried out during normal reactor operation. The added protium could be bled into the spaces between the blanket modules, from where it would diffuse into the vacuum region between the plasma and first wall. The maximum pressure of 10^{-4} torr should not adversely affect plasma parameters.

A small amount of tritium would diffuse through the first wall and module pressure shells into the helium coolant for the blanket, but removal of this leakage would be easy, either by reaction with metal hydrides, or by oxidation to T_2O with subsequent absorption.

For blanket designs in which the bred tritium is released directly from LiAl or $\text{Li}_2\text{Al}_2\text{O}_4$ to the helium coolant stream, tritium recovery is still straightforward but more factors have to be considered. First,

- a) Tritium must be released at a fast rate to the helium coolant stream in order to minimize blanket inventory.
- b) Tritium must be removed at a fast rate from the helium coolant stream in order to minimize tritium inventory in helium.
- c) Tritium must be prevented from diffusing to the environment at a rate greater than the allowable limit, e.g., 1 curie per day.

Figure 4.3.1 gives a schematic view of a tritium removal flowsheet for an indirect cycle, with the primary helium coolant stream transferring heat to a steam generator. The steam then flows to a turbo-generator. Tritium can be present as T_2 , HT , T_2O , or HTO molecules in the helium stream. The amount of tritium recovered from the blanket is ~300 gms/day for a 1000 MW(e) reactor. The total helium inventory in such a reactor will be on the order of 10,000 lbs of helium, based on HTGR designs (10). The tritium concentration in the helium coolant would then increase by 100 ppm per day if it were not processed. It is clear that unless oxygen is deliberately introduced as an impurity, or unless a significant amount

of water vapor leaks in from the steam generator, that only an insignificant amount of tritium would be present as T_2O or HTO . Initially O_2 might be present in helium at a few ppm, but it would be quickly scavenged out. The makeup rate on helium inventory is only ~2% per year, so O_2 introduced via makeup gas would be insignificant on the order of parts per billion per year. Protium gas (H_2) might be deliberately introduced to strip tritium out of the LiAl alloy at a faster rate. If it is not introduced, some will still enter the helium coolant stream via (n,p) production on blanket materials. However, the amount thus introduced will be small (on the order of a few percent) compared to the rate of tritium introduction.

Taking the simplest case first, that is there is no oxygen or protium introduction and that protium production is negligible, the tritium blanket inventory can be calculated from the following relations for the case where part of the helium coolant flows through a metal hydride bed that removes T_2 from the helium:

$$\begin{aligned} \dot{N}_{T_2} &= \dot{N}_{He} [(X_T)_{He,In} - (X_T)_{He,out}] && \text{conservation of tritium} \\ P_T &= (X_T)_{He,In} P_{He} = K_s^2 (X_T)_{LiAl}^2 && \begin{array}{l} \text{Equilibrium of tritium} \\ \text{between solid LiAl and gas} \end{array} \end{aligned} \quad (4.3.8)$$

$$\dot{N}_{\text{He}} = \frac{N_{\text{He}} (860499) (f)}{\tau} \quad \text{Flow rate of helium through tritium trap, gm moles per day}$$

where

$$\dot{N}_{\text{T}_2} = \text{production rate of T}_2, \text{ gm moles/day}$$

$$(X_{\text{T}})_{\text{He,In}} = \text{atom fraction of T}_2 \text{ in He gas entering the tritium trap}$$

$$(X_{\text{T}})_{\text{He,Out}} = \text{atom fraction of T}_2 \text{ in the gas leaving the tritium trap}$$

$$P_{\text{T}} = \text{partial pressure of tritium, torr}$$

$$P_{\text{He}} = \text{pressure of helium, torr}$$

$$(X_{\text{T}})_{\text{LiAl}} = \text{atom fraction of T in LiAl}$$

$$N_{\text{He}} = \text{total inventory of He in coolant circuit, gm moles}$$

$$\tau = \text{time for entire He inventory to flow through blanket, sec}$$

$$f = \text{fraction of the coolant stream that flows through hydride trap}$$

We take the following values as representative,

$K_s^2 = 2 \times 10^5 \text{ torr}/(\text{atom fraction})^2$ [T of LiAl $\sim 500^\circ\text{C}$], $\dot{N}_{\text{T}_2} = 100 \text{ gm atoms/day}$ [1000 MW(e) reactor], $(X_{\text{T}})_{\text{He,Out}} \ll (X_{\text{T}})_{\text{He,In}}$, $f = 0.1$, $\tau = 1 \text{ second}$, $P_{\text{He}} = 15,000 \text{ torr (20 atm)}$ and $N_{\text{He}} = 10^6 \text{ gm moles (10,000 lbs)}$. The latter three values are typical of HTGR's with helium coolant (10). The value of $f = 0.1$ may be somewhat high in that the tritium trap volume will be large. A smaller value of f may be more optimum.

With these conditions, the atom fraction of tritium in LiAl alloy is $\sim 3 \times 10^{-5}$. This corresponds to a blanket inventory of ~ 2.7 kg of tritium, or 9 days of breeding. This represents no problem in terms of tritium inventory charges, but does necessitate an inventory of 3×10^7 curies of tritium in the blanket. The partial pressure of tritium in the helium coolant is 2×10^{-4} torr and the total tritium inventory in the gas is 0.03 gm or 10^3 curies which seems acceptable in terms of the small helium leakage rate ($\sim 2\%$ /year). The daily direct loss rate of tritium to the environment would then be ~ 0.1 curie/day.

The tritium blanket inventory in this case, where it is released to a helium coolant, is approximately an order of magnitude greater than that of the previous case, where it was released directly to the vacuum region between the plasma and the first wall. This results from having a slow helium flow rate (in terms of tritium recovery) through the blanket. The tritium inventory scales as $(\text{flow rate})^{-\frac{1}{2}}$, so that an increase of two orders of magnitude in flow rate would cut tritium inventory by a factor of ten. Such large flow rates would be impractical from a heat removal point of view, however, since pressure drops would be very large and ΔT too low for good thermal efficiency (the temperature of the incoming helium must be low enough that adequate strength is retained in the SAP structure).

The fraction of coolant flow through the tritium trap could be made greater than 10%, but this also appears impractical. The trap will require flow through finely divided metal with a large effective surface area, and slow flow velocities. The tritium blanket inventory could also be reduced by increasing the temperature of the LiAl alloy, which would increase K_s^2 . However, this does not seem compatible with materials limitations.

The tritium blanket inventory can be substantially reduced below 9 days inventory by stripping with controlled additions of protium gas, however. The rate controlling step in the tritium removal process is the flow rate of helium coolant through the tritium trap. Thus the tritium concentration in the helium coolant is fixed once the flow rate and tritium production rate are specified. However, if additions of protium gas are made, this tritium concentration can be achieved with a lower tritium concentration in the solid LiAl alloy. For the steady state case (the unsteady state is less advantageous) with controlled protium addition, we specify:

$$\dot{N}_{H_2} = S \dot{N}_{T_2} \quad \text{with } S \gg 1 \quad (4.3.9)$$

where

$$\dot{N}_{H_2} = \text{gm moles of } H_2 \text{ added to helium coolant stream per day}$$

$\dot{N}_{T_2}$ = gm moles of T_2 bred in blanket per day, also
amount removed from the coolant per day

Under these conditions there will be a mixture of H_2 , HT, and T_2 molecules in the helium coolant with

$$[H_2]_{He} \gg [HT]_{He} \gg [T_2]_{He}$$

The protium is added at a steady rate to the helium stream before it enters the CTR blanket. The protium and tritium concentrations are at steady state in the helium coolant and are determined by

$$\frac{d(X_T)_{He}}{dt} = \frac{1}{N_{He}} \left\{ 2\dot{N}_{T_2} - \dot{N}_{He} (X_T)_{He} \right\} = 0 \quad (4.3.10)$$

$$\frac{d(X_H)_{He}}{dt} = \frac{1}{N_{He}} \left\{ \dot{N}_{H_2} - \dot{N}_{He} (X_H)_{He} \right\} = 0$$

where the nomenclature is the same as above, except that

$(X_T)_{He}$ = atom fraction of HT molecules in He coolant

$(X_H)_{He}$ = atom fraction of H_2 molecules in He coolant

As before, the concentration of tritium and protium in the helium coolant leaving the tritium trap is taken as negligible compared to the concentration in the helium that enters the trap.

There exists in the LiAl alloy a steady state concentration of both tritium and protium, with the protium concentration much

greater than the tritium. Both species are in equilibrium with the gas phase:

$$\begin{aligned} P_{H_2} &= (X_H)_{He} P_{He} = (K_S)_{H_2}^2 (X_H)_{LiAl}^2 \\ P_{HT} &= (X_T)_{He} P_{He} = (K_S)_{HT}^2 (X_T)_{LiAl} (X_H)_{LiAl} \end{aligned} \quad (4.3.11)$$

with the Sieverts' constants essentially the same, i.e.,

$$(K_S)_{H_2}^2 \simeq (K_S)_{HT}^2 \sim K_S^2$$

under these conditions.

$$(X_T)_{LiAl} = \frac{2}{S^{1/2}} \left\{ \frac{N_T}{N_{He}} \frac{P_{He}}{K_S^2} \right\}^{1/2} = \frac{2}{S^{1/2}} (X_T)_{LiAl}^* \quad (4.3.12)$$

where $(X_T)_{LiAl}^*$ = tritium concentration in LiAl with no protium addition to the helium coolant ($N_{H_2} = 0$).

As an example, let us assume $S = 400$. The tritium inventory is then 270 gms (0.9 days of breeding) and is 10% of the inventory without protium addition. The recovered tritium must be separated from the added protium at a H/T ratio of 400/1. It would be economical to separate tritium from protium at this concentration, since D_2O is separated from H_2O at a H/D ratio of ~8000/1 at a low cost. The energy cost for recovery of bred tritium would then be less than that for obtaining D fuel for the reactor. Equipment would have to be capable of handling a radioactive process stream, however. The separated protium would then be recycled to the helium coolant stream, so as to avoid discharging

small amounts of tritium to the environment. At $S = 400$, the average heat supply required for the desorption of the hydrogen isotopes from the hydride bed would be $\sim 60 \text{ kW(th)}$, a negligible amount compared to the total heat budget of the reactor. The tritium blanket inventory could probably be reduced by another factor of 10, if desired, but the desorption heat and separation cost would now not be negligible.

We have not here considered the kinetics of the tritium removal from the LiAl alloy. The diffusion of hydrogen nuclei in the alloy is sufficiently fast that there seems to be no problem in getting the tritium from the interior of the alloy to the surface. The diffusion coefficient is on the order of $10^{-5} \text{ cm}^2 \text{ sec}^{-1}$. Assuming a characteristic dimension for the LiAl alloy of 0.1 cm (the alloy would most likely be made from sintered powder), the characteristic diffusion time is ~ 1000 seconds, which is negligible compared to the inventory time of ~ 1 day. The surface reaction rate for hydrogen absorption and desorption in metal hydride is very fast, in general, especially at high temperatures (20). The characteristic reaction times are on the order of seconds, which are negligible compared to the inventory time.

Because of these fast reaction times, it is not anticipated that the metal hydride tritium trap will pose much of a problem. The trap reaction volume will be on the order of 1% of the blanket volume.

Metallic zirconium, titanium, lanthanum, cerium, or yttrium would seem to be good candidates for the tritium trap after absorption to the desired level. The recovered tritium/protium would be desorbed off the bed by heat, and then sent to a tritium separation/fuel preparation unit.

An alternative method of tritium removal from the helium coolant would be to oxidize the tritium/protium to $\text{HTO}/\text{H}_2\text{O}$ in the coolant entering the tritium trap, absorb the water, and then separate and recover tritium by distillation and electrolysis. This would not seem to offer any advantage over the metal hydride bed. Watson (21) estimates that the tritium partial pressure must be $\leq 10^{-8}$ torr in order to keep the tritium leakage ≤ 1 curie/day 1000 MW(e) for a heat exchanger transferring heat from potassium to steam, at a temperature of $\sim 600^\circ\text{C}$. The stainless steel heat exchange tubes are clad with 12 mils of tungsten. For the design presented here, where 10% of the helium coolant is 1.8×10^{-4} torr. Since the permeation rate scales as the square root of partial pressure, the tritium leakage for the design presented here, where 10% of the helium coolant goes through the helium trap, the tritium partial pressure in the helium coolant is 1.8×10^{-4} torr, will be $\sim 10^2$ curies/day if the heat exchanger tubes had the same permeation resistance as Watson's example. Since the operating temperature of the heat exchanger will be $\sim 500^\circ\text{C}$ in this design, some increase in permeation resistance would be expected, on the order of a

factor of 10 (assumed activation energy of ~ 20 k cal/g atom for permeation), which would reduce the leakage to ~ 10 curies per day.

An alternate barrier could be copper, as proposed for the Princeton reference design (22). Davis (23) gives permeation constants for various copper alloys; the highest permeation resistance he quotes is for AZh 9-4 alloy (9% Al, 4% Fe). Its permeation resistance at 500°C is only 50% that of tungsten at 600°C .

It thus appears that some effort must be devoted to finding a suitable permeation barrier for the helium-steam generator if leakage into the steam circuit is to be kept at or below 1 curie/day. If it proves necessary, an intermediate helium coolant circuit could be used, from which the small amount of tritium that leaked through could easily be scavenged. This would slightly increase overall plant cost, and somewhat lower thermal efficiency.

Another alternate approach would be to use a direct cycle gas turbine for the power generation step. Permeation through the reject heat exchanger would be much smaller because of the much lower temperature. The coolant exit temperature from the blanket would have to be $\geq 800^{\circ}\text{C}$, however, for a practical cycle. This should be possible with some of the blanket design discussed earlier.

This discussion of tritium recovery from the blanket is idealized, of course. LiAl alloy will not be everywhere at the same temperature, and the breeding rate will vary with position in the blanket. The highest breeding rate/unit volume will occur where the neutrons become thermal or epithermal. This is also the zone of highest coolant temperature, which will tend to compensate to some degree. Future studies should look at more realistic breeding models with variable breeding rates and temperature.

This analysis of tritium blanket inventories and tritium removal methods has been very conservative (even so, the tritium inventories are low and certainly acceptable). There are three approaches that can probably reduce the tritium blanket inventory by at least an order of magnitude:

- a) Reduction of the blanket inventory of lithium alloys or compounds
- b) Use of lithium alloys or compounds with higher tritium pressure than possible with LiAl
- c) Periodic refueling of lithium alloys and compounds.

In the blanket designs analyzed earlier in this section, the LiAl inventory was assumed to be $\sim 750 \text{ m}^3$ for a 1000 MW(e) reactor. However at $1 \text{ MW(th)}/\text{m}^2$, lithium depletion in 30 years is equivalent to only 25 m^3 for a 1000 MW(e) reactor. A large

LiAl inventory is necessary if Li^7 reactions are essential to the tritium breeding ratio, since one then tries to maximize the amount of Li^7Al in the fast neutron zone. However, if blanket designs using Be or BeO as neutron multipliers in the fast neutron zone are adopted, then only Li^6 reactions are essential to the tritium breeding ratio. This would then allow one to greatly reduce LiAl inventory by using lithium highly enriched in Li^6 . If we assumed that the lithium blanket inventory is three times as large as that burned out after 30 years of reactor operation, the LiAl inventory would then be 75 m^3 instead of 750 m^3 . This would reduce tritium blanket inventory by a factor of ten from the values given earlier.

The second approach is to use lithium materials with higher tritium pressures than LiAl. One possibility is to use different lithium alloys (e.g., LiMg or LiSi) or to modify LiAl with additions of other metals to further lower lithium activity. The ability of lithium to hold onto tritium would then be reduced and tritium pressure would increase. $\text{Li}_2\text{Al}_2\text{O}_4$ should have a lower lithium activity, and therefore a higher tritium pressure, than LiAl. Further, $\text{Li}_2\text{Al}_2\text{O}_4$ can operate at much higher temperatures (up to $\sim 2000^\circ\text{K}$) than LiAl, which should help promote tritium release. However, to provide assurance that tritium is not bound in some hydroxide form,

experiments should be carried out to measure tritium pressure above $\text{Li}_2\text{Al}_2\text{O}_4$ as a function of temperature and tritium concentration. However, even if there were essentially no tritium concentration at equilibrium, there still would be a lower limit to tritium concentration corresponding to the characteristic diffusion time for tritium to escape from the solid lithium alloy or compound. This diffusion time will depend on the tritium diffusion constant in the particular lithium material and also on the particle size. Some type of low density sintered pellet would appear to offer the most surface area and shortest diffusion time. It should be possible to achieve a characteristic diffusion time of several minutes; this would correspond to a lower limit of ~ 1 gram of tritium inventory in the blanket.

In summary, there appears to be no problem in reducing tritium blanket inventory to ~ 1 days inventory (~ 300 g) with LiAl alloy. With optimization of design and reduction of LiAl inventory and/or use of lithium materials with higher tritium pressures, it may be possible to reduce this inventory by one to two orders of magnitude. The principal problem with the designs in which tritium is recovered from the helium coolant appears to be in keeping the tritium diffusion rate through the steam generator heat exchanger tubes within allowable limits.

TRITIUM LEAKAGE TO STEAM GENERATOR, 1 CURIE / DAY

A high resistance permeation barrier seems to be necessary.

It also appears practical to use an intermediate helium coolant circuit to keep leakage within acceptable limits if a suitable permeation barrier is not possible.

5. Potential Hazards of Fusion Reactor Blankets

The following types of accidents potentially could cause the release of radioactive materials from the blanket:

- 1) Localized melting and vaporization of the blanket by the hot plasma
- 2) Failure of a module structure
- 3) Loss of coolant
- 4) Injection of water into the blanket coolant circuit
- 5) Reaction between blanket and shield materials
- 6) Rupture of the magnet structure

In all cases, the minimum activity blanket concepts developed in this report would result in much smaller amounts of radioactive materials being released inside the reactor containment building or to the environment, than would occur with CTR blankets that use materials like stainless steel, niobium, and vanadium.

Accident types (4) and (5) can be prevented in all CTR blankets by proper choice of materials and coolants. For example, the shield materials should always be compatible with blanket materials (i.e., a shield using water would not be acceptable if there were lithium in the blanket). It probably will be necessary to install an intermediate circuit between the primary blanket coolant and the steam generator (if it is used) to prevent possible injection of water into the blanket coolant circuit.

Loss of coolant accident type (3) is not a potential hazard with the minimum activity blanket concepts described in this report, if the plasma can be turned off, which seems likely. The integrated after heat from radioactive isotopes in the blanket is so small that no cooling is required. (This is true of type 1 and 2 modules, but may not be true of type 3.) CTR blankets with stainless steel, Nb, etc. will have much greater integration after heating and may require emergency core cooling circuits. If these fail large amounts of radioactivity could be released to the containment building.

Accidents of types (1) and (2) are the most likely serious accidents to occur in a CTR blanket. The available thermal energy in the plasma is enough to melt a few m^2 of first wall if it is locally concentrated. The melting of limiters in tokamak experiments is an example of localized concentration. Type (2) accidents could involve the sudden disruption of a module due to a structural stress failure (for example, the head of a module could blow off). Parts of the module could be injected into the plasma region, scattering radioactivity around the inside of the reactor and possibly damaging other modules (which might in turn release more radioactivity and damage other modules, in a sort of chain reaction). A type (2) accident might also trigger a type (1) accident in which the plasma became unstable and dumped its energy locally.

Blankets with all solid components, like the concepts presented in this report, should be more resistant to this type of damage than blankets with liquid lithium or flibe. The principle concern would be with damage to those modules positioned where gravity could cause the solid components to fall out.

Assuming complete destruction of 0.1% of the blanket modules (e.g., approximately 10 modules), with the blanket concepts developed here, only about 10 curies of long lived activity (≥ 1 day) would be scattered inside the reactor, about 10^3 curies of tritium (which could probably be pumped away) and $\sim 10^6$ curies of Na^{24} (which would quickly decay). Repair of the blanket should be feasible. CTR blankets using stainless steel, etc. would have much greater releases of long lived activity ($\sim 10^6$ curies) and it would be very difficult, if not impossible, to repair the blanket. If the blanket could not be repaired, the plant investment would be lost.

A type (6) accident is much more unlikely than type (1) or (2) accidents, but much more serious in its consequences. It would correspond in severity to the rupture of the pressure vessel in a light water reactor. The total stored magnetic energy will be on the order of 10^{11} J in a large toroidal low β reactor, and the total mass of the magnet structure will be

many thousands of tons. Thus, for example, the rupture of a TF magnet coil in a tokamak conceivably could project many tons of steel at high velocities through the blanket structures, releasing most of its radioactive inventory, could cause the destruction of a large fraction of the remaining magnets, and could breach the reactor containment building.

This is a most unlikely event, of course. However, even if one assumes the absolutely worst possible accident in which all of the blanket radioactive inventory is released and dispersed to the environment, the effects will not be catastrophic if minimum activity blankets are used. The $\sim 10^4$ curies of long lived activity would require a much smaller dilution volume than the $\sim 10^9$ curies in a CTR blanket with stainless steel or niobium. The tritium release of $\sim 10^5$ to 10^6 curies would still be a small fraction of the natural tritium inventory in the world ($\sim 10^8$ curies). The local population exposure at site boundaries from the tritium release would probably be within the acceptable limit (25 R whole body) for such an accident. An estimate of the exposure effects from Na^{24} release will require more examination. However, its effects will be minimized because of the fast decay rate.

In summary, the use of minimum activity blankets should make the repair of damage to the blanket structure feasible, and should keep the effects of the worst possible accident, the

rupture of the magnet structure, much less than those from blankets with large radioactive inventories.

TABLE A-1
Transmutation Isotopes of Aluminum (24)
First Generation

Generating reaction	Parent isotope	Initial daughter	Half life	Decay mode	Decay energy (MeV)	Max γ energy (MeV)	Final daughter	Stable (S), unstable (U)
1. n, γ	Al ²⁷	Al ²⁸	2.2m	β^-	4.6	1.8	Si ²⁸	S
2. n, p	Al ²⁷	Mg ²⁷	9.4m	β^-	2.6	1.0	Al ²⁷	S
3. n, 2n	Al ²⁷	Al ²⁶	{ 6.4 s 7.3x10 ⁵ y }	β^+	3.2	--	Mg ²⁶	S
				β^+	4.0	1.8	*Al ²⁶	U
4. n, α	Al ²⁷	Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
5. n, d	Al ²⁷	Mg ²⁶	S	--	--	--	Mg ²⁶	S
6. n, He ³	Al ²⁷	Na ²⁵	60 s	β^-	1.0	--	Mg ²⁵	S
7. n, T	Al ²⁷	Mg ²⁵	S	--	--	--	Mg ²⁵	S

<u>Second Generation</u>								
8. n, γ	Si ²⁸	Si ²⁹	S	--	--	--	Si ²⁹	S
9. n, p	Si ²⁸	Al ²⁸	2.2m	β^-	4.6	1.8	Si ²⁸	S
10. n, 2n	Si ²⁸	Si ²⁷	4.2 s	β^+	4.8	2.2	Al ²⁷	S
11. n, α	Si ²⁸	Mg ²⁵	S	--	--	--	Mg ²⁵	S
12. n, d	Si ²⁸	Al ²⁷	S	--	--	--	Al ²⁷	S
13. n, He ³	Si ²⁸	Mg ²⁶	S	--	--	--	Mg ²⁶	S
14. n, T	Si ²⁸	Al ²⁶	{ 6.4 s 7.3x10 ⁵ y }	β^+	3.2	--	Mg ²⁶	S
				β^+	4.0	1.8	*Al ²⁶	U
15. n, γ	Mg ²⁴	Mg ²⁵	S	--	--	--	Mg ²⁵	S
16. n, p	Mg ²⁴	Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
17. n, 2n	Mg ²⁴	Mg ²³	12s	β^+	4.1	3.0	Na ²³	S

TABLE A-1 (Cont'd)

Second Generation (Cont'd)

	<u>Generating reaction</u>	<u>Parent isotope</u>	<u>Initial daughter</u>	<u>Half life</u>	<u>Decay mode</u>	<u>Decay energy (MeV)</u>	<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S), unstable (U)</u>
18.	n, 2n	Mg ²⁴	Ne ²¹	S	--	--	--	Ne ²¹	S
19.	n, d	Mg ²⁴	Na ²³	S	--	--	--	Na ²³	S
20.	n, He ³	Mg ²⁴	Ne ²²	S	--	--	--	Ne ²²	S
21.	n, T	Mg ²⁴	Na ²²	*2.6y	β^+	2.8	1.3	*Na ²² Ne ²²	U S
22.	n, γ	Mg ²⁵	Mg ²⁶	S	--	--	--	Mg ²⁶	S
23.	n, p	Mg ²⁵	Na ²⁵	60 s	β^-	3.8	1.0	Mg ²⁵	S
24.	n, 2n	Mg ²⁵	Mg ²⁴	S	--	--	--	Mg ²⁴	S
25.	n, α	Mg ²⁵	Ne ²²	S	--	--	--	Ne ²²	S
26.	n, d	Mg ²⁵	Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
27.	n, He ³	Mg ²⁵	Ne ²³	38 s	β^-	4.4	1.64	Na ²³	S
28.	n, T	Mg ²⁵	Na ²³	S	--	--	--	Na ²³	S
29.	n, γ	Mg ²⁶	Mg ²⁷	9.4m	β^-	2.6	1.0	Al ²⁷	S
30.	n, p	Mg ²⁶	Na ²⁶	1.0 s	β^-	8.7	1.8	Mg ²⁶	S
31.	n, 2n	Mg ²⁶	Mg ²⁵	S	--	--	--	Mg ²⁵	S
32.	n, α	Mg ²⁶	Ne ²³	38 s	β^-	4.4	1.6	Na ²³	S
33.	n, d	Mg ²⁶	Na ²⁵	60 s	β^-	3.8	1.0	Mg ²⁵	S
34.	n, He ³	Mg ²⁶	Ne ²⁴	3.4m	β^-	2.5	0.9	Na ²⁴	U
			Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
35.	n, T	Mg ²⁶	Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
36.	n, γ	Al ²⁶	Al ²⁷	S	--	--	--	Al ²⁷	S

TABLE A-1 (Cont'd)

Second Generation (Cont'd)

	<u>Generating reaction</u>	<u>Parent isotope</u>	<u>Initial daughter</u>	<u>Half life</u>	<u>Decay mode</u>	<u>Decay energy (MeV)</u>	<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S), unstable (U)</u>
37.	n,p	Al ²⁶	Mg ²⁶	S	--	--	--	Mg ²⁶	S
38.	n, 2n	Al ²⁶	Al ²⁵	7.2s	α^+	4.3	1.6	Mg ²⁵	S
39.	n, 2	Al ²⁶	Na ²³	S	--	--	--	Na ²³	S
40.	n, d	Al ²⁶	Mg ²⁵	S	--	--	--	Mg ²⁵	S
41.	n, He ³	Al ²⁶	Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
42.	n, T	Al ²⁶	Mg ²⁴	S	--	--	--	Mg ²⁴	S

Third Generation

43.	n, γ	Si ²⁹	Si ³⁰	S	--	--	--	Si ²⁹	S
44.	n, p	Si ²⁹	Al ²⁹	6.5m	α^-	3.7	2.4	Si ²⁹	S
45.	n, 2n	Si ²⁹	Si ²⁸	S	--	--	--	Si ²⁸	S
46.	n, α	Si ²⁹	Mg ²⁶	S	--	--	--	Mg ²⁶	S
47.	n, d	Si ²⁹	Al ²⁸	2.2m	β^-	4.6	1.8	Si ²⁸	S
48.	n, He ³	Si ²⁹	Mg ²⁷	9.4m	β^-	2.6	1.0	Al ²⁷	S
49.	n, T	Si ²⁹	Al ²⁷	S	--	--	--	Al ²⁷	S
50.	n, γ	Na ²³	Na ²⁴	15.0h	β^-	5.5	2.8	Mg ²⁴	S
51.	n, p	Na ²³	Ne ²³	38s	β^-	4.4	1.64	Na ²³	S
52.	n, 2n	Na ²³	Na ²²	*2.6y	β^+	2.8	1.3	Ne ²²	S
53.	n, α	Na ²³	F ²⁰	11s	β^-	7.0	1.6	Ne ²⁰	S
54.	n, d	Na ²³	Ne ²²	S	--	--	--	Ne ²²	S
55.	n, He ³	Na ²³	F ²¹	4.45	β^-	5.7	1.4	Ne ²¹	S
56.	n, T	Na ²³	Ne ²¹	S	--	--	--	Ne ²¹	S

TABLE A-1 (Cont'd)

Third Generation (Cont'd)

Generating reaction	Parent isotope	Initial daughter	Half life	Decay mode	Decay energy (MeV)	Max γ energy (MeV)	Final daughter	Stable (S), unstable (U)
57. n, γ	Na ²²	Na ²³	S	--	--	--	Na ²³	S
58. n, p	Na ²²	Ne ²²	S	--	--	--	Ne ²²	S
59. n, 2n	Na ²²	Na ²¹	23s	β^+	3.6	0.35	Ne ²¹	S
60. n, α	Na ²²	F ¹⁹	S	--	--	--	F ¹⁹	S
61. n, d	Na ²²	Ne ²¹	S	--	--	--	Ne ²¹	S
62. n, He ³	Na ²²	F ²⁰	11s	β^-	7.6	1.6	Ne ²⁰	S
63. n, T	Na ²²	Ne ²⁰	S	--	--	--	Ne ²⁰	S
64. n, γ	Ne ²²	Ne ²³	38s	β^-	4.4	1.6	Na ²³	S
65. n, p	Ne ²²	F ²²	4s	β^-	10.8	2.06	Ne ²²	S
66. n, 2n	Ne ²²	Ne ²¹	S	--	--	--	Ne ²¹	S
67. n, α	Ne ²²	O ¹⁹	27s	β^-	4.8	1.6	F ¹⁹	S
68. n, d	Ne ²²	F ²¹	4.4	β^-	5.7	1.4	Ne ²¹	S
69. n, He ³	Ne ²²	O ²⁰	14s	β^-	3.8	1.06	F ²⁰	U
70. n, T	Ne ²²	F ²⁰	14s	β^-	7.0	1.6	Ne ²⁰	S

TABLE A-2
Transmutation Isotopes of Oxygen (24)

First Generation

	<u>Initial daughter</u>	<u>Parent isotope</u>	<u>Generating reaction</u>	<u>Half life</u>	<u>Decay mode</u>	<u>Decay energy (MeV)</u>	<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S) or Unstable (U)</u>
1.	N ¹⁷	O ¹⁸ O ¹⁷	n, d n, p	4.2s	β^-	8.7	2.2	O ¹⁷	S
2.	N ¹⁶	O ¹⁸ O ¹⁷ O ¹⁶	n, T n, d n, p	7.1s	β^-	10.42	7.1	O ¹⁶	S
3.	N ¹⁵	O ¹⁷ O ¹⁶	n, T n, d	S	--	--	--	N ¹⁵	S
4.	N ¹⁴	O ¹⁶	n, T	S	--	--	--	N ¹⁴	S
5.	O ¹⁵	O ¹⁶	n, 2n	122s	β^+	2.8	--	N ¹⁵	S
6.	O ¹⁹	O ¹⁸	n, γ	27s	β^-	4.8	1.6	F ¹⁹	S
7.	C ¹³	O ¹⁶	n, α	S	--	--	--	C ¹³	S
8.	*C ¹⁴	O ¹⁶ O ¹⁷	n, He ³ n, α	5730y	β^-	0.16	--	C ¹⁴ C ¹³	U S
9.	C ¹⁵	O ¹⁸ O ¹⁷	n, α n, He ³	2.4s	β^-	9.8	5.3	N ¹⁵	S
10.	C ¹⁶ N ¹⁶	O ¹⁸ --	n, He ³ --	0.74s 7.1s	β^- β^-	8.0 10.4	NA 7.1	N ¹⁶ O ¹⁶	U S

Second Generation

11.	*C ¹⁴	N ¹⁵ N ¹⁴	n, d n, p	5730y	β^-	0.156	--	C ¹⁴ N ¹⁴	U S
12.	C ¹²	C ¹³ N ¹⁴	n, 2n n, T	S	--	--	--	C ¹²	S

TABLE A-2 (Cont'd)

Second Generation (Cont'd)

	<u>Initial daughter</u>	<u>Parent isotope</u>	<u>Generating reaction</u>	<u>Half life</u>	<u>Decay mode</u>	<u>Decay energy (MeV)</u>	<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S) or Unstable (U)</u>
13.	F ²⁰	F ¹⁹	n, γ	11s	β^-	7.0	1.6	Ne ²⁰	S
14.	B ¹³	N ¹⁵ C ¹⁴	n, He ³ n, d	17ms	β^-	13.4	3.7	C ¹³	S
15.	B ¹²	N ¹⁵ N ¹⁴ C ¹⁴ C ¹³	n, α n, He ³ n, T n, d	20ms	β^-	13.4	4.4	C ¹²	S
16.	B ¹¹	N ¹⁴ C ¹³	n, α n, T	S	--	--	--	B ¹¹	S
17.	Be ¹² B ¹²	C ¹⁴ --	n, He ³ --	11s 20ms	β^- β^-	11.6 13.4	 4.4	B ¹² C ¹²	U S
18.	Be ¹¹	C ¹⁴ C ¹³	n, α n, He ³	14s	β^-	11.5	8.0	B ¹¹	S
19.	*Be ¹⁰	C ¹³	n, α	1.6x10 ⁶ y	β^-	0.56	--	*Be ¹⁰ Be ¹⁰	7 S

TABLE A-3

Transmutation Isotopes of Carbon (24)

	<u>Initial daughter</u>	<u>Parent isotope</u>	<u>Generating reaction</u>	<u>First Generation</u>			<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S) or Unstable (U)</u>
				<u>Half life</u>	<u>Decay mode</u>	<u>Decay energy (MeV)</u>			
1.	C ¹⁴	C ¹³	n, γ	5730y	β^-	0.16	--	C ¹⁴ N ¹⁴	U S
2.	B ¹²	C ¹³ C ¹²	n, d n, p	20ms	β^-	13.4	4.4	C ¹²	S
3.	B ¹¹	C ¹³ C ¹²	n, T n, d	S	--	--	--	B ¹¹	S
4.	B ¹⁰	C ¹²	n, T	S	--	--	--	B ¹¹	S
5.	Be ¹¹	C ¹³	n, He ³	14s	β^-	11.5	8.0	B ¹¹	S
6.	*Be ¹⁰	C ¹³ C ¹²	n, α n, He ³	1.6x10 ⁶ y	β^-	0.56	--	*Be ¹⁰ B ¹⁰	U S
7.	Be ⁹	C ¹²	n, α	S	--	--	--	Be ⁹	S

TABLE A-4

Transmutation Isotopes of Titanium (24)

	<u>Initial daughter</u>	<u>Parent isotope</u>	<u>Generating reaction</u>	<u>Half life</u>	<u>First Generation</u>		<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S) or Unstable (U)</u>
					<u>Decay mode</u>	<u>Decay energy (MeV)</u>			
1.	Ti ⁵¹	Ti ⁵⁰	n, γ	5.8m	β^-	2.5	1.0	V ⁵¹	S
2.	Ti ⁴⁵	Ti ⁴⁶	n, 2n	3.1h	β^+	2.1	1.7	Sc ⁴⁵	S
3.	Sc ⁴⁹	Ti ⁵⁰ Ti ⁴⁹	n, d n, p	57m	β^-	2.0	1.7	Ti ⁴⁹	S
4.	* Sc ⁴⁸	Ti ⁵⁰ Ti ⁴⁹ Ti ⁴⁸	n, T n, d n, p	44h	β^-	4.0	1.3	Ti ⁴⁸	S
5.	*Sc ⁴⁷	Ti ⁴⁹ Ti ⁴⁸ Ti ⁴⁶	n, T n, d n, p	3.4h	β^-	0.6	0.16	Ti ⁴⁶	S
6.	*Sc ⁴⁶	Ti ⁴⁸ Ti ⁴⁷ Ti ⁴⁶	n, T n, d n, p	83.8d	β^-	2.4	1.1	Ti ⁴⁶	S
7.	Sc ⁴⁵	Ti ⁴⁷ Ti ⁴⁶	n, T n, d	S	--	--	--	Sc ⁴⁵	S
8.	*Sc ⁴⁴	Ti ⁴⁶	n, T	59h	e	3.9	2.6	Ca ⁴⁴	S
9.	Ca ⁴⁸	Ti ⁵⁰	n, He ³	S	--	--	--	Ca ⁴⁸	S
10.	*Ca ⁴⁷	Ti ⁵⁰ Ti ⁴⁹	n, α n, He ³	4.5d	β^-	2.0	1.9	*Sc ⁴⁷	U

TABLE A-4 (Cont'd)

First Generation (Cont'd)

	<u>Initial daughter</u>	<u>Parent isotope</u>	<u>Generating reaction</u>	<u>Half life</u>	<u>Decay mode</u>	<u>Decay energy (MeV)</u>	<u>Max γ energy (MeV)</u>	<u>Final daughter</u>	<u>Stable (S) or Unstable (U)</u>
11.	Ca ⁴⁶	Ti ⁴⁹	n, α	S	--	--	--	Ca ⁴⁶	
12.	*Ca ⁴⁵	Ti ⁴⁹ Ti ⁴⁸	n, α n, He ³	163d	β^-	0.26	0.01	Sc ⁴⁵	S
13.	Ca ⁴⁴	Ti ⁴⁷ Ti ⁴⁶	n, α n, He ³	S	--	--	--	Ca ⁴⁴	S
14.	Ca ⁴³	Ti ⁴⁶	n, α	S	--	--	--	Ca ⁴³	S

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