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and Experimental Power Reactors \*

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## MINIMUM ACTIVITY BLANKETS FOR COMMERCIAL AND EXPERIMENTAL POWER REACTORS

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### Summary

A new kind of tritium breeding blanket for fusion reactors is investigated. All components in the blanket are solid, including the lithium containing material, which eliminates problems of corrosion and materials compatibility. This permits the use of structural materials with very low neutron activation. A promising structural material is SAP, a pure aluminum strengthened by an  $Al_2O_3$  dispersoid. Three types of blanket module designs are investigated in which the helium coolant exit temperature is substantially greater than the maximum operating temperature of SAP ( $\sim 400^\circ C$ ). Exit temperatures range from  $500^\circ C$  to above  $1000^\circ C$ , while breeding ratios range from 0.9 to 1.5, depending on design. Structural stress is  $\sim 2000$  psi, which is  $\sim 20\%$  of failure stress. The bred tritium diffuses from the lithium alloy or compound (e.g.,  $LiAl$ ), which has a much higher tritium pressure than pure lithium, into the helium coolant stream from which it is recovered. Depending on processing rates, temperature,  $LiAl$  inventory, and the amount of scavenging by added  $H_2$ , the tritium blanket inventory ranges from  $\sim 10^5$  to  $10^6$  curies. Short lived activities, principally  $Na^{24}$  (15 hour half life), decay to a negligible amount in a few days, leaving a long lived inventory of  $\sim 10^4$  curies/1000 MW(e) [principally  $Al^{26}$  ( $7.3 \times 10^5$  year half life)]. This material appears to present no problems of permanent waste disposal. Because of the very low inventory of long lived activities, blanket repair or replacement seems relatively simple.

### Introduction

The ideal fusion reactor blanket should have the following characteristics:

1. Very low inventory of tritium and long lived isotopes
2. Good resistance to radiation damage
3. Easy repair or replacement
4. No corrosion, either by coolant or breeding material
5. High operating temperature (good thermal efficiency)
6. Adequate tritium breeding ratio ( $> 1.05$ )
7. Low cost

It is unlikely that all goals can be simultaneously achieved and some tradeoffs are necessary.

Previous blanket designs have used either liquid lithium metal or flibe (a fused salt mixture of  $LiF$  and  $BeF_2$ ) as the breeding material [ $Li$  captures neutrons and generates tritium to replace that burned in the plasma]. The choice of structural materials to contain liquid lithium or flibe is rather limited -- niobium, stainless steel, or vanadium seem the only reasonable contenders. The first two are severely activated by the fast neutrons from the DT plasma, with a total inventory of  $\sim 10^6$  curies of long lived isotopes in a 1000 MW(e) reactor. This is comparable to the inventory in a fission reactor. Pure vanadium is not activated, but is expensive. In addition pure vanadium is very susceptible to oxygen impurities in the coolant/liquid lithium, and it is quite possible that it will have to be alloyed with elements which are themselves severely activated.

In this study a new type of fusion reactor blanket is examined in which tritium is bred in a solid lithium alloy or compound in the blanket. The bred tritium diffuses out of the solid lithium alloy or compound either into a helium coolant stream, from which it is removed outside the blanket, or into the plasma exhaust of the reactor.

Lithium-aluminum,  $LiAl$ , is a prime candidate for the breeding material. It has a melting point of  $\sim 700^\circ C$ , and the pressure of tritium above  $LiAl$  is much greater than above pure lithium. It is not necessary to replace the solid  $LiAl$ , even if the reactor operates for 30 years.

A much wider range of blanket structural materials is now possible since corrosion is not a problem. R. Hirsch and W. Gough<sup>1</sup> have suggested the desirability of a minimum radioactive inventory CTR. A solid breeding blanket seems advantageous since structural materials with very low long-lived activation can be used which would not be compatible with liquid lithium or flibe. A promising 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.<sup>2</sup> This material was developed for the organic cooled reactor program. It is reasonably strong ( $\sim 10,000$  psi) at  $350-400^\circ C$ ; aluminum and  $Al_2O_3$  can be made very pure; long lived activation products of Al and O from 14 MeV neutrons are very small; it appears to have good resistance to radiation damage; has a high thermal conductivity; and it should be cheap ( $\sim \$1/lb$ ). The other possible structural materials have disadvantages: beryllium is difficult to fabricate, expensive, and not structurally reliable; graphite and silicon carbide are brittle and low strength; and other Al alloys are much weaker than SAP at  $400^\circ C$ . Magnesium with a  $MgO$  dispersoid could be a possible alternative to SAP, but its properties are unknown.

There are several promising moderators, i.e. graphite, Be,  $BeO$ , and metal hydrides. Graphite, Be and  $BeO$  are capable of high temperature operation and should be activated very little; further Be and  $BeO$  are good neutron multipliers and can boost the tritium breeding ratio. Metal hydrides are attractive since they can minimize blanket thickness; however, most high temperature hydrides are strongly activated by neutrons (e.g.,  $TiH_2$ ,  $ZrH_2$ , etc.)  $MgH_2$  will not be activated, but can only operate to about  $400^\circ C$  -- above this temperature, the  $H_2$  dissociation pressure is too high.

Aluminum is a promising structural material for low activity blankets, and also is present in a promising breeding material ( $Li-Al$  alloy). Figure 1 shows experimentally measured activations of various grades of aluminum in a thermal neutron spectrum.<sup>3</sup> These experiments were carried out by ALCOA in 1964 at the BNL graphite reactor.

The aluminum with the lowest activation (curve 26) is the ALCOA 1199 grade. It is normally 99.97% pure; typical impurity analyses are given in Table 1. This grade was not developed for low activation, but is one of ALCOA's standard grades, at  $\sim \$0.50/lb$ . Further reductions in activation can probably be made

without substantial cost increase by selecting the ore from which the  $\text{Al}_2\text{O}_3$  feedstock is derived<sup>3</sup> and by process modifications.

#### Blanket Designs--Mechanical and Thermal Aspects

The maximum operating temperature of SAP is  $\sim 400^\circ\text{C}$ . At a coolant exit temperature of  $\sim 400^\circ\text{C}$ , the thermal cycle efficiency is on the order of 32%, comparable to a PWR. This efficiency is probably acceptable. If the blanket operates at  $400^\circ\text{C}$ , it can be designed as a multilayer tube bank with the moderator and LiAl inside the SAP tubes. This should be highly redundant and reliable. Coolant exit temperatures above  $400^\circ\text{C}$  are possible since most of the fusion energy does not appear in the SAP structure, but rather in the moderator and the lithium compound used for tritium breeding. The thermal cycle efficiency can be much greater than 30%, probably in the range of 40-50%. In this paper we only consider designs with elevated coolant temperatures.

Elevated coolant temperatures can be achieved by using a single coolant stream to first cool the SAP structure and then to cool the moderator and lithium compound, which operate above  $400^\circ\text{C}$ . The coolant exit temperature is limited either by the maximum allowable temperature for the moderator and/or lithium compound, or by the  $\Delta T$  across the SAP structure and the energy fraction deposited in the SAP structure. Roughly one third of the energy/fusion appears in the SAP structure. With a  $\Delta T$  of  $150^\circ\text{C}$  across the SAP ( $250$  to  $400^\circ\text{C}$ ), the coolant exit temperature from the module is  $\sim 700^\circ\text{C}$ .

Alternatively, two coolant streams can be used, one at low temperature ( $\sim 350^\circ\text{C}$ ) to cool the SAP structure, and one at high temperature ( $\sim 1000^\circ\text{C}$ ) to cool the moderator and lithium compound. Materials limit the exit temperature of the second coolant stream. Here the blanket is more complex than in the first approach, but the higher thermal efficiency may offset this.

Both approaches require thermal insulation (e.g. alumina) between the cool SAP structure and hot portion of the blanket. The magnitude of the heat leak depends on the thermal conductivity and thickness of the insulation. Evacuated insulation does not seem practical. Refractory ceramics of low density (with pores filled by helium) appear necessary. Thermal conductivity is  $\sim 0.5$  to  $1.0 \times 10^{-2}$   $\text{w/cm}^2\text{K}$ .

The allowable heat leak and the thermal conductivity fix the separation between hotter and cooler blanket regions, which in turn sets module diameter (typically 0.3 to 1.0 meter).

Figure 2 shows an elevation view of a conceptual type 1 blanket module. The module is a cylindrical shell of SAP with solid blanket materials inside. It is closed at the bottom with a dished head which is the first wall. The helium coolant enters at the top, passes down the annular space between the SAP shell and the thermal insulator, returns through the interior of the module, and exits at the upper end. The thermal insulator is approximately 1 cm thick.

The type 1 design uses as much  $\text{Li}^7\text{Al}$  as possible in the fast neutron zone to generate tritium by  $\text{Li}^7$  reactions. The LiAl limits the helium exit temperature to  $\sim 500^\circ\text{C}$ .

The LiAl alloy is in the form of rods or balls made by powder metallurgy (to facilitate tritium release there are gas spaces between the particles of LiAl). These rods or balls are stacked in the module, with helium coolant flowing between. The LiAl may require a SAP jacket with holes so the bred tritium can escape to the helium coolant. (The SAP would have zero stress.)

Metal hydride moderators must be enclosed in a sealed SAP tube to prevent loss of hydrogen. Graphite probably need not be jacketed.

Figure 3 shows an elevation view of a type 2 module. The coolant flow is similar to that for type 1 modules. The interior materials are Be for neutron multiplication and graphite for neutron moderation. The LiAl is placed in the thermal insulator between the hot and cold regions and cooled by a He bleed stream. Coolant exit temperatures well above  $500^\circ\text{C}$  are possible with this module type.

Figure 4 shows an elevation view of a type 3 module. The low temperature He stream circulates through tubes bonded to the outside of the SAP shell and cools it. The tube separation can be fairly wide (typically 6cm) because of the high thermal conductivity of SAP.

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 Figure 4, the thermal insulator is  $\text{Li}_2\text{Al}_2\text{O}_4$ , which is a source of heat and tritium. Alternatively,  $\text{Li}_2\text{Al}_2\text{O}_4$  could be placed in the hot region with an  $\text{Al}_2\text{O}_3$  thermal insulator.

Because of the high helium exit temperature ( $\sim 2000^\circ\text{K}$ ), graphite and BeO must be used. Type 3 module diameters must be on the order of 0.7 m due to the large  $\Delta T$  between the hot and cold regions.

The thermal balance for the three module designs is shown in Table 2. Helium outlet temperatures of  $500^\circ\text{C}$  or over are feasible. Larger module diameters increase the ratio of hot/cold power because of reduced heat leak.

A type 1 or 2 design can be coupled to a conventional steam cycle. A type 3 design can also couple to a conventional steam cycle, with the low temperature stream generating steam and the high temperature stream, superheat. Alternatively, the low temperature stream could power a turbine and the high temperature stream could couple to a direct cycle gas turbine, a potassium topping cycle, or possibly an MHD generator.

Stress distributions have been calculated for the module shells with a two dimensional axisymmetric finite element code (2000 elements). Figure 5 shows stress contours in a typical module with a coolant pressure of 20 atm. Thermal stresses are included, and the first wall loading is  $1 \text{ MW}(\text{th})/\text{m}^2$ . Maximum stress occurs at or near the junction of the dished head and cylindrical shell. With coolant pressures of 10 atm., maximum stress is  $\sim 2000$  psi or about 20% of the failure stress.

A bank of separate SAP tubes can be placed between the plasma and the dished heads of the blanket modules. The tubes would be the first wall. The coolant in the tubes would receive the radiant energy from the plasma plus a few percent of the 14 MeV

neutron energy. This has advantages--the tube bank can have several layers, providing redundancy in the case of failure; the dished head of the modules are protected against bombardment by plasma ions; cooling is easier; thermal stress is less; and the  $\Delta T$  for the coolant in the tube bank can be small, which will increase thermal efficiency. In type 3 modules the tubes on the outside of the SAP shell can be further apart if a tube bank is used.

Hopkins<sup>4</sup> has suggested low activity SiC and/or pyrographite structural materials for CTR blankets. If the tube bank were SiC tubes, it can operate at much higher temperatures, e.g., 1000°C, increasing thermal cycle efficiency. The SAP modules would sit behind the SiC tubes. Because of the brittle nature of SiC, it seems much more suitable for small diameter tubes than for large complex modules.

A tube bank will reduce the breeding ratio somewhat, but not seriously. The tube walls are fairly thin, on the order of a few millimeters, and a thinner dished head for the module is then possible, which will partially compensate for the tube bank.

#### Neutronics--Breeding Ratio

Representative compositions and dimensions are selected for the type 1, type 2, and type 3 blankets. These designs are not optimized, however.

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 Table 3 for a type 2 blanket.

The cross sections used in this work were based on ENDF/B-III.<sup>5</sup> 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 SUPERDING<sup>6</sup> program and distributed through the Radiation Shielding Information Center as RSIC Data Library DLC-2/100G.<sup>7</sup> 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.<sup>8</sup> 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<sup>9</sup> program to generate a  $P_3$  multi-group set with the same GAM-II group structure.

Retrieval programs are used to generate a  $P_0$  cross section library tape for the ANISN<sup>10</sup> 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.

To obtain 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 balance for a type 2 module is given in Table 4.

The breeding ratio for design 1A is only 0.89 because of the poor tritium production in  $Li^7$  (0.1 per fusion neutron). The Al cross sections for fast neutrons are low, but the aluminum first wall is much thicker and the Al/ $Li^7$  ratio much greater than the corresponding values in a Nb-Li blanket. The aluminum parasitically absorbs 15% of the fast neutrons (and degrades neutron energy, reducing the number of  $Li^7$  reactions).

The Al/Li ratio is 1.93 in zone 4 and the first wall thickness is 2 cm. With lower helium pressures the aluminum can be significantly reduced and the breeding ratio increased. It is doubtful that the breeding ratio can be greater than 1.0, however.

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 (zones 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)/m<sup>2</sup>, the leakage energy is small enough that its input to the superconducting magnets will be tolerable.

With type 2 modules, the breeding ratio is much better, 1.49, because of n,2n multiplication in beryllium. Considerable amounts of surplus tritium could be generated for other purposes, such as making up the tritium deficiency of CTR reactors with breeding ratios less than one. The large breeding margin also permits one to use more SAP structure, for example, a thicker first wall. This could be 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<sup>2</sup> of first wall in design 2A. Since the Be metal is only used as a neutron multiplier and not structurally, high quality metal is not needed. It can have cracks and flaws, and should be much cheaper than the cost of structural Be metal. Even so, it may be desirable to reduce the Be inventory to reduce cost. At \$10/lb the Be metal in design 2A costs \$12/KW(e) for a first wall flux of 1 MW/m<sup>2</sup>. The neutronics calculations indicate that if the Be zone is cut in half the breeding ratio would be reduced to 1.3, which is still quite good.

After 30 years of operation at 1 MW/m<sup>2</sup>, 1.3% of the Be atoms would be destroyed by (n,2n) and (n, $\alpha$ ) reactions. This is the average fraction destroyed; at the inner edge of the zone, about 4% of the Be would be destroyed.

The breeding ratio in design 3A is 1.08 using BeO as the neutron multiplier. This is adequate for fusion, but it leaves little leeway to increase the amount of structure. The breeding ratio is lower than that for design 2A because of the reduced Be density and competition from oxygen for neutrons. Because of the low breeding ratio, the BeO zone cannot be significantly reduced. Increasing BeO thickness would not significantly increase breeding since only 7% of the breeding occurs in the graphite.

The thicker blanket and shield in design 3A compared to 1A and 2A (1.8m vs 1.6m), 2A reduced the neutron energy flux out of the shield to  $\sim 0.01$  watt/cm<sup>2</sup>.

In summary, there should be no difficulty in achieving breeding ratios well above 1.0 if Be or BeO are used as neutron multipliers. If Be or BeO are not used, the breeding ratios will approach 1.0, but makeup tritium from outside sources (other types of fusion reactors, accelerators), may be needed.

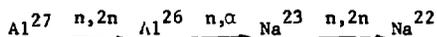
#### Neutronics--Activation and Transmutation

Long lived activation of the blanket is defined here as those isotopes with half lives of one day or greater. This is to some extent arbitrary, since a half life of a year or less is short lived in terms of permanent waste disposal, while a half life of a few hours could be long lived in case a major reactor accident dispersed some of the blanket to the environment. A criterion of one day is useful, however, from the standpoint of blanket repair or replacement. All activities with half lives of one day or less will die out sufficiently in two weeks or less so that they can be neglected when repair or replacement starts.

Blanket activations can be divided into two groups - those resulting from neutron reactions on blanket constructional materials (Al, Al<sub>2</sub>O<sub>3</sub>, Be, BeO, C, LiAl, etc.) and those resulting from neutron reactions on impurities in the blanket constructional materials. Impurity concentrations are low enough (ppm level) that neutron activation of transmutation products of the various impurities can be neglected. Activation of transmutation products of blanket constructional materials must be considered, however, since they are present in appreciable amounts. In fact, transmutation chains are investigated up to the 3d generation to ensure that no important activations are overlooked.

Considering the aluminum structure first, the two important first generation long lived activations of aluminum are Al<sup>26</sup> [(n,2n) on Al<sup>27</sup>] with a half life of  $7.3 \times 10^5$  years (long lived isomer) and Na<sup>24</sup> [(n,α) on Al<sup>27</sup>], with a half life of 15.0 hours. Strictly speaking, the latter is not a long lived activity, by the definition we have adopted, but its half life and inventory are large enough that its decay must be considered when evaluating possible hazards and cool-down time before repair or replacement.

No important activations of aluminum are formed in the second generation since the required reactions have a threshold energy  $> 14.1$  MeV. Na<sup>22</sup> (2.2 yr half life) is produced in the third generation by three routes, for example:

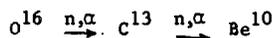


The long lived Al<sup>26</sup> activity is relatively small, typically a few thousand curies in a 1000 MW(e) reactor after 30 years. The inventory is small because of the long half life and the high threshold energy (13.06 MeV) for the (n,2n) reaction on Al<sup>27</sup>. There is a short half life Al<sup>26</sup> isomer (6.4 seconds) also produced by the (n,2n) reaction on Al<sup>27</sup>. This activity can be neglected because of its short half life. A 50-50 yield is assumed for the two isomers.

The Na<sup>24</sup> activity is much greater, on the order of  $10^9$  curies in a 1000 MW(e) reactor. Because of its short half life it has essentially disappeared after two weeks. The inventory of Na<sup>22</sup> is so small ( $\sim 10$  curies in a 1000 MW(e) reactor, after 30 years) that it can be neglected compared to other activa-

tions. The small inventory is due to Na<sup>22</sup> only being generated in the 3d generation by reactions with small cross sections.

Oxygen produces C<sup>14</sup> (5730 y) in the first generation via (n,α) reactions on O<sup>17</sup>. The inventory in a 1000 MW(e) reactor is approximately 30 curies after 30 years. The small inventory is a result of the low isotopic concentration of O<sup>17</sup> (0.039%) and the small (n,α) cross section. C<sup>14</sup> is produced from oxygen by several routes in the second generation, but the amount is small compared to that produced in the first generation. Be<sup>10</sup> ( $1.6 \times 10^6$  y) is also produced by a second generation route from oxygen



but the amount is negligible ( $\ll 10$  curies).

Beryllium produces Be<sup>10</sup> in the first generation by (n,γ) reactions, but the amount is small, e.g.,  $\sim 300$  curies after 30 years at 1000 MW(e).

Graphite produces C<sup>14</sup> in the first generation by (n,γ) reactions, but the inventory is small, about 100 curies after 30 years at 1000 MW(e). Graphite also produces Be<sup>10</sup> in the first generation but the inventory is negligible.

Titanium is a constructional material in the three types of modules described in this paper. It is strongly activated in the first generation by fast neutrons through (n,p) and (n,α) reactions, but is not activated by (n,γ) reactions. The activities resulting from neutron reactions with titanium are:

Activity	Inventory, Curies/1000 MW(e)		
	Design #1	Design #2	Design #3
Ca <sup>45</sup>	$10^6$	4000	400
Sc <sup>46</sup> , Sc <sup>47</sup>	$4 \times 10^7$	$1.2 \times 10^5$	$1.2 \times 10^4$

In design #1, titanium hydride and deuteride are used in the blanket and shield regions while in designs #2 and #3, TiC is used in the shield region. Titanium compounds probably should be ruled out for the blanket region, and only used for those parts of the shield region where activation is not significant. Titanium is not necessary for low activity blankets. In the rest of this section we do not consider any activation due to titanium.

For the blanket constructional materials, neutron reaction cross sections for the isotopes of interest are available in the ENDF/B-III library. Evaluated cross sections are not generally available, however, for isotopes of the impurities that are present in the various constructional materials. Fast neutron cross sections can be calculated with reasonable accuracy ( $\pm 20\%$ ) for isotopes of mass  $\geq 50$ , but (n,γ) cross sections cannot be calculated. Fortunately, (n,γ) reactions are not too significant in terms of long lived activation.

The impurities present in 1199 grade aluminum are summarized in Table 1. A measure of the (n,γ) activation of these impurities is given in Table 5, assuming 1/V cross sections for both Li<sup>6</sup> and the impurities present. An Al/Li<sup>6</sup> ratio of 10/1 is assumed, which corresponds to a Li<sup>6</sup> inventory of approximately three times that consumed during 30 years of operation at 1 MW(th)/m<sup>2</sup>. The Sc<sup>46</sup> and Co<sup>60</sup> activations correspond to concentrations of 0.02 ppm (atomic) in aluminum for Sc<sup>45</sup> and Co<sup>59</sup>. Both concentrations are derived from activation measurements by ALCOA on 1199 alloy. The Sc<sup>45</sup> and Co<sup>59</sup> concentrations are upper limits, and the

actual concentrations may be less.

Thus the total (n, $\gamma$ ) activation of aluminum impurities is small, on the order of a few hundred curies. This results from a relatively low Al/Li<sup>6</sup> ratio and the high capture cross section for Li<sup>6</sup>. Non i/v effects may somewhat increase impurity activation, but it seems clear that (n, $\gamma$ ) activation is small compared to Al<sup>26</sup> activation.

In terms of activation by fast neutrons, of impurities in aluminum, the potentially important isotopes are Mn<sup>55</sup>, Ti<sup>46</sup>, Ti<sup>47</sup>, Ti<sup>48</sup>, Sc<sup>45</sup>, Fe<sup>54</sup>, Fe<sup>56</sup>, Zn<sup>64</sup>, Zn<sup>66</sup>, Zr<sup>90</sup>, Zr<sup>91</sup>, Zr<sup>92</sup>, Zr<sup>94</sup>, Zr<sup>96</sup>, Sn<sup>116</sup>, Sn<sup>118</sup>, Sn<sup>120</sup>, Ni<sup>58</sup>, Ni<sup>60</sup>, Sb<sup>121</sup>, and Sb<sup>123</sup>. Although the concentrations of the Sb and Ni in 1199 aluminum are given as zero, the latter four isotopes are included to see if Sb and Ni could be important at 1ppm level.

Neutron cross sections for the above isotopes have been calculated as a function of energy by Bhat and Magurno<sup>11</sup> for the following reactions: (n,2n); (n,3n); (n,2 $\alpha$ ); (n,np); (n,p); (n,d); (n,He<sup>3</sup>); (n,T); and (n, $\alpha$ ). Activation inventories are calculated using these cross sections with the ANISN fluxes computed for the three different module designs. The various impurities are of such small concentration that they have no appreciable effect on flux distributions.

The significant activations resulting from these isotopes and reactions are shown in Table 6, along with Al<sup>26</sup> and Na<sup>24</sup> activations from (n,2n) and (n, $\alpha$ ) reactions on Al<sup>27</sup>. A 1000 MW(e) reactor is assumed with either type 1, 2, or 3 modules. Impurity concentrations are taken as the same as those for selected stock 1199 aluminum. The radioactive inventories are computed after 30 years of reactor operation. The three most significant impurity activations are Mn, Fe, and Zn. Concentration of these three elements is high, ~10 ppm, reaction cross sections are relatively great, and the important parent isotopes have high isotopic concentration.

Figure 6 shows the decay of total activity (including impurity, activation) in the aluminum structure and aluminum compounds (Al<sub>2</sub>O<sub>3</sub>, LiAl) of a 1000 MW(e) reactor, as a function of time after reactor shutdown. The total activity includes all activations with initial inventories greater than 30 curies. The decay curves for type 2 and 3 modules are virtually identical, since they use essentially the same amount of aluminum.

Design #1 has substantially more activity, because of the much greater amount of LiAl in the reactor. The total activity drops very rapidly since Na<sup>24</sup> predominates, and then levels out at ~10 days. The total activity then remains constant for several hundred days until Mn<sup>54</sup>, Fe<sup>55</sup> and Zn<sup>65</sup> decay. The activity then drops to a level plateau in which essentially only Al<sup>26</sup> remains. The Mn, Fe, and Zn contributions can be essentially eliminated by reducing Mn and Fe concentrations to 1ppm, and Zn to 3ppm. It should be possible to achieve these reductions by a variety of methods, including selection of suitable aluminum ore, more purification of alumina in the Bayer process, control of impurities in the graphite electrodes used in the electrolysis cells, recrystallization of the aluminum metal, and electrolytic refining (Hoopes cell), if needed. The present cost of 1199 aluminum is approximately \$0.50/lb; the extra purification could probably be accomplished with only a small raise in cost. The cost of this aluminum should be less than \$1.00/lb.<sup>3</sup>

Figure 6 also shows the total activity if isotopes with (E $\gamma$ )<sub>max</sub>  $\leq$  0.03 MeV (e.g., Fe<sup>55</sup>) are omitted. These low gamma energy isotopes do not require shielding if the blanket is being repaired or replaced. However, they are only a small part of the total activity.

Activation of the impurities in the other blanket materials (Be, BeO, C) will occur. However, since graphite impurity levels can be kept at a few ppm, and since graphite will be used in the part of the blanket where the fast neutron flux has been greatly reduced, the impurity activation in graphite will be much less than that for aluminum.

Impurity activations in Be and BeO have not yet been examined. It seems likely that both can be purified sufficiently that impurity activation will not be a problem.

The activation curves in Figure 6 are calculated on the basis of 30 years of operation. It is likely that blankets will have to be replaced at more frequent intervals because of radiation damage, and the long lived inventory in a given blanket will thus be considerably lower. If the blanket is replaced at three year intervals, the Al<sup>26</sup> inventory in a 1000 MW(e) reactor would then be ~700 curies. If the reactor is a tokamak segmented into 12 sectors, like the UWMAK-1 design, the Al<sup>26</sup> inventory in a given sector would then be only 60 curies. This low activity would be very attractive when it came time to replace the blanket. In order to take advantage of the low Al<sup>26</sup> inventory, however, it may be necessary to further reduce impurity levels in aluminum, so as to keep impurity activation comparable with Al<sup>26</sup> activation.

#### Tritium Recovery From Blanket

The tritium is bred in situ in solid lithium alloys (e.g., LiAl) or lithium ceramics (e.g., Li<sub>2</sub>Al<sub>2</sub>O<sub>4</sub>). The tritium can be recovered either by letting the tritium diffuse from the solid or by periodically removing the solid from the blanket and processing it. This latter approach is very costly and difficult. Leaving the solid in the blanket and recovering the bred tritium by diffusion seems quite feasible. The tritium holdup in the blanket is small, on the order of 1 days production. We only consider this method of tritium recovery.

The simplest form of tritium recovery (method A) is to let the bred tritium diffuse directly into the vacuum region between the plasma and the first wall. The LiAl alloy is outside of the SAP module pressure shell (but attached to it) and the evolved tritium diffuses between modules into the vacuum region.

Two modes of tritium recovery are possible:

(i) The tritium atom fraction in the LiAl alloy is at a steady state value with tritium leaving the alloy at the same rate that it is bred. Since the surface area is very large and reaction rates rapid, this steady state concentrations is given by the equilibrium between the LiAl and the partial pressure of tritium in the vacuum region.

(ii) The tritium inventory in the blanket using mode (i) can be reduced by scavenging. A high protium concentration is built up in the LiAl alloy by equilibrating with a high pressure of protium (i.e., H<sub>2</sub> gas) for a short time. The protium gas is then pumped away, carrying tritium. This cyclical process keeps the tritium inventory much smaller than that in mode (i).

No measurements have yet been made of the equilibrium constants for tritium gas in equilibrium with dilute solutions of tritium in LiAl. Sievert's Law will hold,

$$(P_{T_2}) = (K_T)^2 \text{LiAl} (X_T)^2 \text{LiAl} \quad (1)$$

where  $X_T$  = atom fraction of T in LiAl

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

The Sievert's constants for LiAl can be estimated from available measured data on Li and LiAl:

$$(K_T)^2 \text{LiAl} \approx (K_T)^2 \text{Li} \left( \frac{P_{H_2}}{P_{H_2}} \right)^* \text{LiAl} \quad (2)$$

where  $(P_{H_2})^* \text{LiAl}$  and  $(P_{H_2})^* \text{Li}$  are the plateau pressures (two phase region) for  $H_2$  in LiAl and Li.  $(K_T)_{\text{Li}}$  and  $(P_{H_2})^*_{\text{Li}}$  are given as a function of temperature by Maffoni<sup>12</sup>, and  $(P_{H_2})^*_{\text{LiAl}}$  by Aronson and Salzano.<sup>13</sup> At 500°C,  $(K_T)^2_{\text{LiAl}}$  is then  $2 \times 10^5$  torr, a factor of  $2 \times 10^3$  greater than  $(K_T)^2_{\text{Li}}$ .

If the vacuum region between the plasma and the first wall is at  $10^{-6}$  torr, and for simplicity is assumed to be 100% tritium, the equilibrium atom fraction of tritium in LiAl alloy is  $\sim 2 \times 10^{-6}$ . A 1000 MW(e) reactor would have a LiAl alloy inventory of  $75 \text{ m}^3$  (type 2 module). At a tritium atom fraction of  $2 \times 10^{-6}$ , the tritium inventory would be  $\sim 16$  gms or  $2 \times 10^5$  curies. At a thermal cycle efficiency of 40%,  $\sim 300$  gms of tritium will be burned in the plasma per day, and approximately the same amount bred in the blanket. A tritium atom fraction of  $2 \times 10^{-6}$  is much less than a days inventory. This will not affect doubling time, and inventory charges are negligible.

This tritium inventory seems acceptable from a hazards standpoint, and is less than the inventory for the refueling portion of the tritium cycle. Further reductions of the tritium blanket inventory do not seem likely by reducing vacuum pressure, since  $10^{-6}$  torr is practical lower limit. However, scavenging of tritium by additions of protium [mode (1)] can be used.

If the tritium concentrations in LiAl is  $(X_T)_0$ , and protium gas equilibrates at a pressure  $(P_{H_2})^*$  with the LiAl and is then pumped away and the normal operating pressure of  $10^{-6}$  torr resumed, the tritium concentration is reduced to

$$(X_T)_1 \approx (X_T)_0 \left[ \frac{10^{-6}}{(P_{H_2})^*} \right]^{\frac{1}{2}} \quad (3)$$

since the Sieverts constants for tritium and protium are essentially the same. The tritium is evolved as HT molecules when the excess protium is pumped away. With a scavenging protium pressure of  $10^{-4}$  torr, and a scavenging interval of 40 minutes, tritium blanket inventory is only  $10^4$  curies instead of  $2 \times 10^5$  curies. The extracted tritium is present at  $\sim 1\%$  in the evolved gas, and can be separated relatively easily from the scavenging protium either by metal hydrides, distillation of liquid hydrogen or by distillation of tritiated water after conversion of the evolved hydrogen to water. This example is illustrative and is not optimized.

The cyclical scavenging can be carried out during normal reactor operation by bleeding the protium into the spaces between the blanket modules, from where it diffuses into the vacuum region between

the plasma and first wall. The maximum pressure of  $10^{-4}$  torr should not adversely affect the plasma.

A small amount of tritium would diffuse through the first wall and module pressure shells into the helium coolant, but this can be easily removed either by reaction with metal hydrides, or by oxidation to  $T_2O$  with subsequent absorption.

For blanket designs in which LiAl or  $\text{Li}_2\text{Al}_2\text{O}_4$  is inside the SAP module shell, the bred tritium is released directly from the LiAl or  $\text{Li}_2\text{Al}_2\text{O}_4$  to the helium coolant stream (method B). Tritium recovery is still straightforward but the following factors have to be considered:

- Tritium must be released at a fast rate to the helium coolant to minimize blanket inventory.
- Tritium must be removed at a fast rate from the helium coolant.
- Tritium must not diffuse to the environment at a rate greater than the allowable limit, e.g., 1 curie per day.

Figure 7 gives a schematic view of a tritium removal flowsheet for an indirect cycle where the primary helium coolant stream transfers heat to a steam generator. Tritium can exist as  $T_2$ , HT,  $T_2O$ , or HTO molecules in the helium stream. About 300 gms of tritium is recovered from the blanket per day in a 1000 MW(e) reactor. The total helium inventory in the reactor is  $\sim 10,000$  lbs of helium, based on HTGR designs.<sup>4</sup> The tritium concentration in the helium coolant would increase by 100 ppm per day with processing. Unless oxygen is deliberately introduced or unless a significant amount of water vapor leaks in from the steam generator, only an insignificant amount of tritium can be present as  $T_2O$  or HTO. Initially  $O_2$  might be present in helium at a few ppm, but it would quickly be scavenged. The helium makeup rate is only  $\sim 2\%$  per year, so  $O_2$  introduced via makeup gas would be insignificant. Protium gas ( $H_2$ ) might be deliberately introduced to strip tritium out of the LiAl alloy at a faster rate. If it is not, some will still enter the helium coolant stream via (n,p) production on blanket materials. However, the amount will be small (on the order of a few percent) compared to the tritium production.

The tritium is removed from the part of the helium coolant stream that circulates through the absorbing trap (e.g., a metal hydride). If no protium is added to the helium coolant, the tritium concentration in the LiAl in the blanket is given by

$$(X_T) \text{LiAl} = \frac{1}{(K_T) \text{LiAl}} \left[ \frac{P_{He} \dot{N}_{T_2}}{\dot{N}_{He}} \right]^{\frac{1}{2}} \quad (4)$$

where  $P_{He}$  = Helium coolant pressure, torr  
 $\dot{N}_{T_2}$  = Tritium production rate in blanket, gm moles/second  
 $\dot{N}_{He}$  = Helium flow rate through trap, gm moles/second

If the tritium is completely removed from the helium leaving the trap, and if  $f$  = fraction of He flow through the trap, then

$$\dot{N}_{He} = \frac{N_{He} f}{\tau}$$

where  $N_{He}$  = Helium inventory in coolant circuit ( $\sim 10^4$  lbs)

$\tau$  = Time for He inventory flow through blanket ( $\sim 1$  second)

If protium ( $H_2$ ) gas is added to the coolant stream at a rate  $\dot{N}_{H_2}$ , where  $S = \dot{N}_{H_2}/\dot{N}_{T_2}$ , the tritium concentration in the blanket LiAl is reduced to:

$$(x_T)^*_{LiAl} = \frac{2}{S+1} (x_T)_{LiAl} \quad (6)$$

where  $(x_T)^*_{LiAl}$  is the tritium concentration with added protium.

Figure 8 shows the total tritium blanket inventory in a 1000 MW(e) reactor as a function of processing reaction (f), and LiAl temperature, for the case  $S=0$  (no protium addition) and  $S=400$ , with an inventory of  $75 m^3$  of LiAl in the blanket. One-third of the LiAl inventory (90%  $Li^6$ ) is consumed by neutron capture after 30 years. The tritium blanket inventory of  $10^5$  to  $10^6$  curies is comparable to the inventory with liquid lithium or flibe. The processing fraction f will probably be a few percent, and the LiAl temperature  $\sim 500^\circ C$ .

The recovered tritium can be separated from the added protium at a H/T ratio of 400/1.  $D_2O$  is separated from  $H_2O$  at a H/D ratio of  $\sim 8000/1$  at relatively low cost. The energy cost for recovery of bred tritium is less than that for obtaining D fuel for the reactor. The recovery equipment must handle a radioactive process stream, however. The separated protium is recycled to the helium coolant stream to prevent discharging tritium to the environment. At  $S=400$ ,  $\sim 60$  KW(th) is required to desorb hydrogen isotopes from the hydride bed, which is small compared to the  $\sim 10^6$  KW heat rate of the reactor.

The diffusion of hydrogen nuclei in the alloy and the surface reaction rate are sufficiently fast that the tritium should easily move from the interior of the alloy to the surface. The diffusion coefficient is  $\sim 10^{-5} cm^2 sec^{-1}$ . With a characteristic dimension for the LiAl alloy of 0.1 cm (the alloy would be made from sintered powder), the diffusion time is  $\sim 1000$  seconds compared to an inventory time of  $\sim 1$  day.

Tritium can also be removed from the helium coolant by oxidizing the tritium/protium to HTO/ $H_2O$  in the coolant entering the tritium trap, absorbing the water and, then separating tritium by distillation and electrolysis. This does not seem any better than the metal hydride bed.

The tritium blanket inventory can be reduced below the  $10^5$  to  $10^6$  curie range by several techniques: more protium addition, smaller lithium inventory in the blanket, and use of lithium compounds with higher equilibrium tritium pressures. Rates much greater than  $S=400$  require more elaborate tritium recovery units, however. A smaller lithium blanket inventory requires replacement of the blanket at intervals much shorter than 30 years. A shorter interval is likely anyway, because of radiation damage. The UWMAC-1 design assumes replacement of the entire blanket at two year intervals, for example. The most promising technique is to use lithium compounds with higher tritium equilibrium pressures, perhaps  $Li_2Al_2O_4$ . With a melting point of  $\sim 2000^\circ C$ , it could operate at very high temperatures (design 3) where the tritium equilibrium pressure should be very high.

With such techniques it should be possible to achieve a tritium blanket inventory of  $\sim 10^4$  curies.

A suitable permeation barrier (perhaps tungsten or copper clad tubes) for the helium-steam generator is needed to keep leakage into the steam circuit less than one curie/day. An intermediate helium coolant

circuit might be used, from which the small amount of tritium that leaks through could be scavenged. This would increase overall plant cost, and slightly lower thermal efficiency.

Another alternate approach is a direct cycle gas turbine. Permeation through the reject heat exchanger would be much smaller because of the lower temperature. The coolant exit temperature from the blanket would have to be  $800^\circ C$  or above for a practical cycle. This is possible with some of the blanket designs.

#### Implications of Low Activity Blankets for Hazards, Repair and Replacement and Waste Disposal

If part of the blanket is dispersed to the environment, an extremely unlikely event, the principal concern is tritium and  $Na^{24}$ . The latter is bound in aluminum (SAP and LiAl) and cannot be released unless the Al is destroyed. The 15 hour half life should make  $Na^{24}$  orders of magnitude less hazardous than the fission products in a fission reactor. Tritium has a relatively long half life (12 years) but its inventory is relatively small,  $10^5$  to  $10^6$  curies, and it is much less hazardous than fission products.

Low activity blankets should be much easier to repair and replace than high activity blankets. This can be extremely important if blankets have to be replaced every 2-3 years. The total inventory in a 1000 MW(e) reactor can be  $\sim 1000$  curies after several weeks of shutdown for a blanket irradiated for three years. Relatively little shielding would be required to work on such a blanket.

The only isotope with a half life long enough to survive in significant amounts in a permanent waste disposal burial area is  $Al^{26}$  ( $7.3 \times 10^5$  years). Contamination of ground water would be extremely small, much smaller than the MPC for unrestricted discharge, for three reasons:

1. Low aluminum concentration in natural ground water [ $\sim 10^{-7}$  gm/ml <sup>14</sup>].
2. Low specific activity of aluminum waste [ $\sim 10^{-5}$  curie/gm after 30 years].
3. The vast reservoirs of natural aluminum in the ground, which would greatly dilute any activity in ground water.

The calculated MPC for unrestricted discharge for  $Al^{26}$  is  $\sim 10^{-4}$   $\mu Ci/ml$ , which compares with the MPC of  $4 \times 10^{-5}$   $\mu Ci/ml$  established <sup>15</sup> for  $Sc^{46}$ , which is chemically similar to Al. If dilution by natural aluminum is not taken into account, the specific activity of ground water which has been equilibrated with the aluminum waste would be  $10^{-6}$   $\mu Ci/ml$ , which is approximately two orders of magnitude smaller than the calculated MPC.

This activity will be greatly diluted by natural aluminum. For example 1  $Km^3$  of ground has  $\sim 10^6$  times as much aluminum as the waste from a 1000 MW(e) reactor. In practice, the waste would be buried deeply enough that equilibration would occur in a much larger volume before the ground water ever emerged, and concentrations would be many order of magnitude, smaller than  $10^{-6}$   $\mu Ci/ml$ . Thus there appear to be no problems in permanent waste disposal of fusion blankets if the type described in this paper is used.

In summary, there appear to be many advantages to the type of low activity blankets examined in this paper and further study seems warranted.

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## References

1. Gough, W. C., AEC, personal communication (1973).
2. Heavy Water Organic Cooled Reactor - SAP Handbook - AI-CE-Memo-24 (March 23, 1966).
3. Kramer, R., ALCOA, personal communication (1973). Also, "The Residual Radioactivity in Pure Aluminum and Aluminum Alloys," R. Kramer, Report 1-73-3 ALCOA.
4. Hopkins, G., Gulf Atomics, personal communication (1973).
5. M. K. Drake, Editor, BNL-50274 (T-601) (ENDF 102, Vol. 1) (October 1970).
6. R. Q. Wright, N. M. Greene, J. L. Lucius and C. W. Craven, Jr., ORNL-TM-2679 (Sept. 1969).
7. "Abstracts of the Data Library Packages Assembled by the Radiation Shielding Information Center," ORNL-RSIC-30.
8. G. D. Joannou and J. S. Dudek, GA-4265 (1963).
9. D. E. Jusner and S. Kellman, WCAP-3845-1 (ENDF 114) (December 1969).
10. W. W. Engle, Jr., K-1693 (March 1967).
11. Bhat, M. and Magurno, B., BNL, personal communication, 1973.
12. Maroni, V. A., Cairns, E. J., and Cafasso, F. A., ANL-8001 (March 1973).
13. Aronson, A., and Salzano, F. J., Inorg. Chem. 8, 1541 (1969).
14. Bricker, O., and Garrels, R., "Mineralogic Factors in Natural Water Equilibria," in Principles and Application of Water Chemistry, ed., S. Fraus and J. Hunter, Wiley (1967).
15. Code of Federal Regulations, Title 10, Part 20.

## Figure Captions

- Figure 1 Residual Activation for Various Grades of Aluminum in a Thermal Neutron Flux
- Figure 2 Elevation View of Type 1 Blanket Module
- Figure 3 Elevation View of Type 2 Blanket Module
- Figure 4 Elevation View of Type 3 Blanket Module
- Figure 5 Stress Contours in Module Shell Due to 20 Atm Coolant Pressure and Thermal Stress
- Figure 6 Total Inventory of Radioactive Isotopes in 1000 MW(e) Reactor as a Function of Time After Reactor Shutdown
- Figure 7 Schematic of Processing Methods to Remove Tritium from Helium Blanket Coolant
- Figure 8 Tritium Inventory in 1000 MW(e) Reactor as a Function of LiAl Temperature, Processing Rate, and Scavenging Protium Rate

Table 1

Impurity Concentrations in ALCOA 1199 Aluminum  
Concentration, ppm (weight)

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

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

Table 2

Thermal Balance for Modules

<u>Parameter</u>	<u>Module Design</u>			
	<u>1A</u>	<u>2A</u>	<u>3A</u>	
			<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 <sub>He</sub> = 10 Atm)	2	2	2	2
(ΔT <sub>wall</sub> ) <sub>max</sub> , °C	25	25	50	-

\* 1 MW(th)/m<sup>2</sup> First Wall Flux (22.4 MeV/DT Fusion)

\*\* Includes Heat Leak from Hot to Cold Region

Table 3

Blanket Design # 2A

	Zone				
	3	4	5	6	7
<b>Compositions</b>					
<b>Material (%)</b>					
1. SAP	100	17	20	20	20
2. Al <sub>2</sub> O <sub>3</sub>	--	11.0	11.0	11.0	11.0
3. LiAl	--	10	10	10	--
4. Graphite	--	--	42	42	--
5. Be Metal	--	45	--	--	--
6. Ti(H) <sub>1.5</sub>	--	--	--	--	52
7. Helium Coolant	--	17	17	17	17
<b>Atomic Densities (x10<sup>22</sup>)</b>					
1. Al (natural)	5.88	1.53	1.70	1.70	1.70
2. O <sup>16</sup>	0.515	0.477	0.492	0.492	0.492
3. Be (natural)	--	5.4	--	--	--
4. C (natural)	--	--	4.41	4.41	--
5. Li <sup>6</sup>	--	0.243	0.243	0.243	--
6. Li <sup>7</sup>	--	0.027	0.027	0.027	--
7. Ti (natural)	--	--	--	--	1.955
8. H <sup>1</sup>	--	--	--	--	2.933
<b>Thickness, cm</b>	<b>2</b>	<b>30</b>	<b>25</b>	<b>25</b>	<b>80?</b>

Table 4

Neutron Balance, Blanket Design # 2A

Nuclide and reaction	Zone 3	Zone 4	Zones 5 & 6	Zone 7
<b>Al<sup>27</sup></b>				
n, 2n	.01070	.00713	.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
<b>O<sup>16</sup></b>				
n, γ	.00000	.00000	.00000	.00000
n, p	.0092	.00254	.00008	.00000
n, d	.00032	.00085	.00003	.00000
n, α	.00339	.01201	.00059	.00000
Total Abs	.00463	.01540	.00070	.00000
<b>Be</b>				
n, 2n	--	.71694	--	--
n, γ	--	.00105	--	--
n, p	--	.00005	--	--
n, d	--	.00000	--	--
n, t	--	.00000	--	--
n, α	--	.06794	--	--
<b>C</b>				
n, γ	--	--	.00000	--
n, α	--	--	.00398	--
<b>Li<sup>6</sup></b>				
n, 2n	--	.00242	.00008	--
n, t	--	1.3112	.17144	--
Total Abs	--	.00079	.00006	--
<b>Li<sup>7</sup></b>				
n, 2n	--	.00019	.00000	--
n, t	--	.00205	.00010	--
Total Abs	--	.00004	.00000	--
<b>Ti</b>				
n, 2n	--	--	--	.00000
n, γ	--	--	--	.00378
n, p	--	--	--	.00001
n, α	--	--	--	.00000
<b>H</b>				
n, γ	--	--	--	.00031

Breeding Ratio = 1.49 tritons/fusion neutron

TABLE 5

IMPORTANT ACTIVATIONS FROM THERMAL  
NEUTRONS (1/V ABSORPTION)

Basis: 1000 MW(e) Reactor

Al/Li<sup>6</sup> Ratio = 10/1

Impurities: Same as select 1199 stock

<u>Activation</u>	<u>Inventory (Curies)</u>	<u>Half- Life</u>	<u>(E<sub>γ</sub>)<sub>max</sub> (Mev)</u>
Co <sup>60</sup>	148	5.3y	1.3
Sc <sup>46</sup>	104	84d	1.1
Zn <sup>65</sup>	312	244d	1.1
Ca <sup>45</sup>	<2.4	163d	0.01
Zr <sup>95</sup>	2.2	66d	0.75
Po <sup>210</sup> (from Bi)	4.9	139d	0.8
Sn <sup>119</sup>	0.9	245d	0.02
Sn <sup>121</sup>	2.4	27h	0.04
Fe <sup>55</sup>	84	2.7y	0

TABLE 6  
IMPORTANT ACTIVATIONS FROM FAST  
NEUTRON ACTIVATIONS

	Daughter Isotope	Half- Life	$(E_{\gamma} \text{ max})$ (Mev)	Parent Isotope	Reaction	Inventory in 1000 MW(e) Reactor* (Curies)		
						Design 1	Design 2	Design 3
1.	Na <sup>24</sup>	15 h	2.75	Al <sup>27</sup>	(n,α)	1.9x10 <sup>9</sup>	1.2x10 <sup>9</sup>	1.2x10 <sup>9</sup>
2.	**Al <sup>26</sup>	7.5x10 <sup>5</sup> y	1.18	Al <sup>27</sup>	(n,2n)	1.1x10 <sup>4</sup>	7x10 <sup>3</sup>	7x10 <sup>3</sup>
3.	Fe <sup>55</sup>	2.7y	0	Fe <sup>56</sup>	(n,2n)	1.7x10 <sup>4</sup>	1.2x10 <sup>4</sup>	1.3x10 <sup>4</sup>
4.	Mn <sup>54</sup>	312d	0.83	Mn <sup>55</sup>	(n,2n)	4.9x10 <sup>4</sup>	3.3x10 <sup>4</sup>	3.1x10 <sup>4</sup>
				Fe <sup>54</sup>	(n,p)			
5.	Zn <sup>65</sup>	244d	1.1	Zn <sup>66</sup>	(n,2n)	1.1x10 <sup>4</sup>	6.9x10 <sup>3</sup>	7.3x10 <sup>3</sup>
6.	Zr <sup>89</sup>	3.3d	0.91	Zr <sup>90</sup>	(n,2n)	5.8x10 <sup>3</sup>	3.5x10 <sup>3</sup>	3.7x10 <sup>3</sup>
7.	Sn <sup>119</sup>	245d	0.024	Sn <sup>120</sup>	(n,2n)	2.0x10 <sup>3</sup>	1.2x10 <sup>3</sup>	1.3x10 <sup>3</sup>
8.	Sn <sup>117</sup>	14d	0.16	Sn <sup>118</sup>	(n,2n)	1.4x10 <sup>3</sup>	860	880
9.	Sc <sup>48</sup>	1.8d	1.3	Ti <sup>48</sup>	(n,p)	1.1x10 <sup>3</sup>	670	690
10.	Zr <sup>95</sup>	65d	0.76	Zr <sup>96</sup>	(n,2n)	860	520	530

\* Activations only from aluminum structure--impurity concentrations are for selected 1199 stock.

\*\* Yield of long lived Al<sup>26</sup> isomer assumed to be 50%.

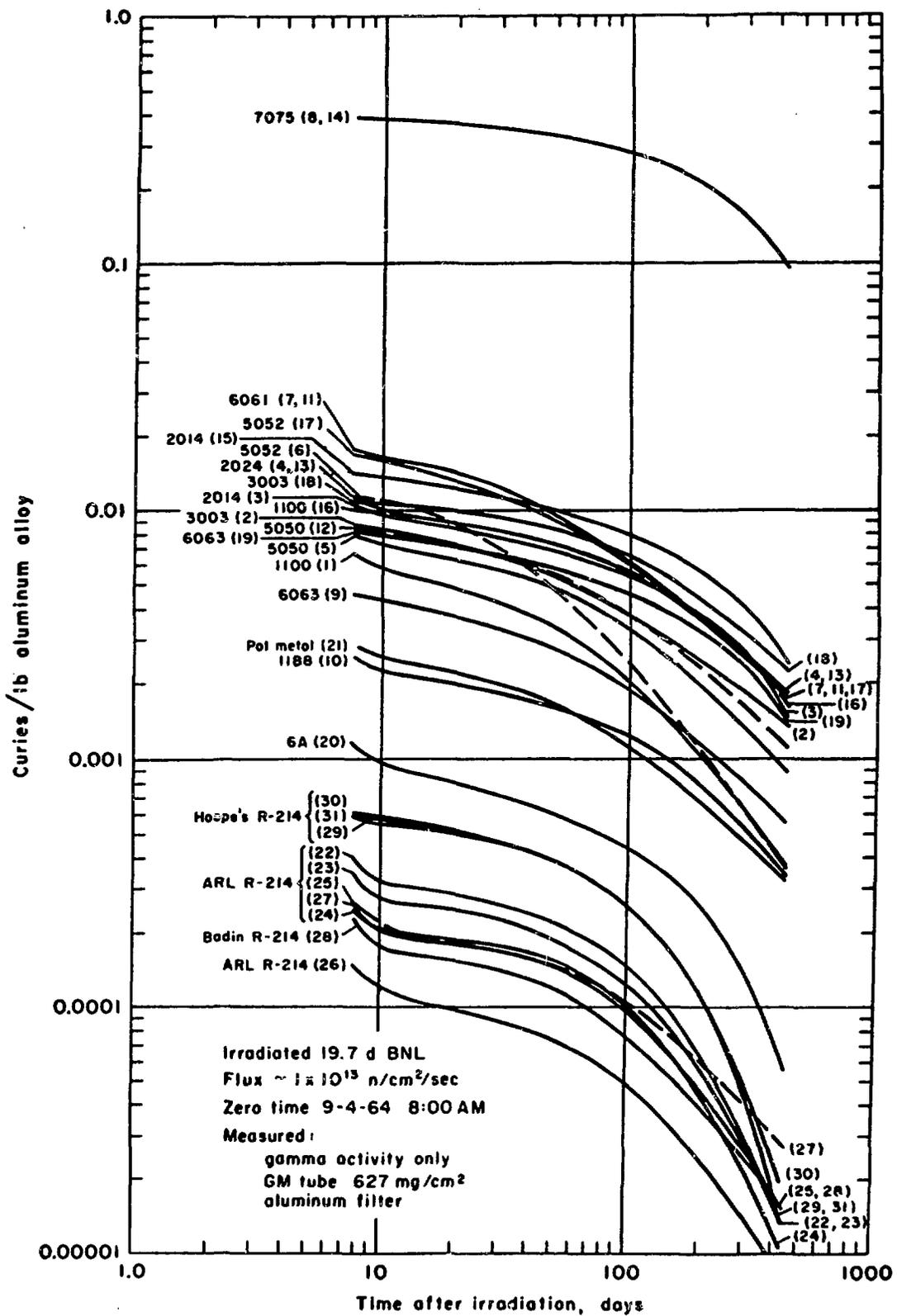
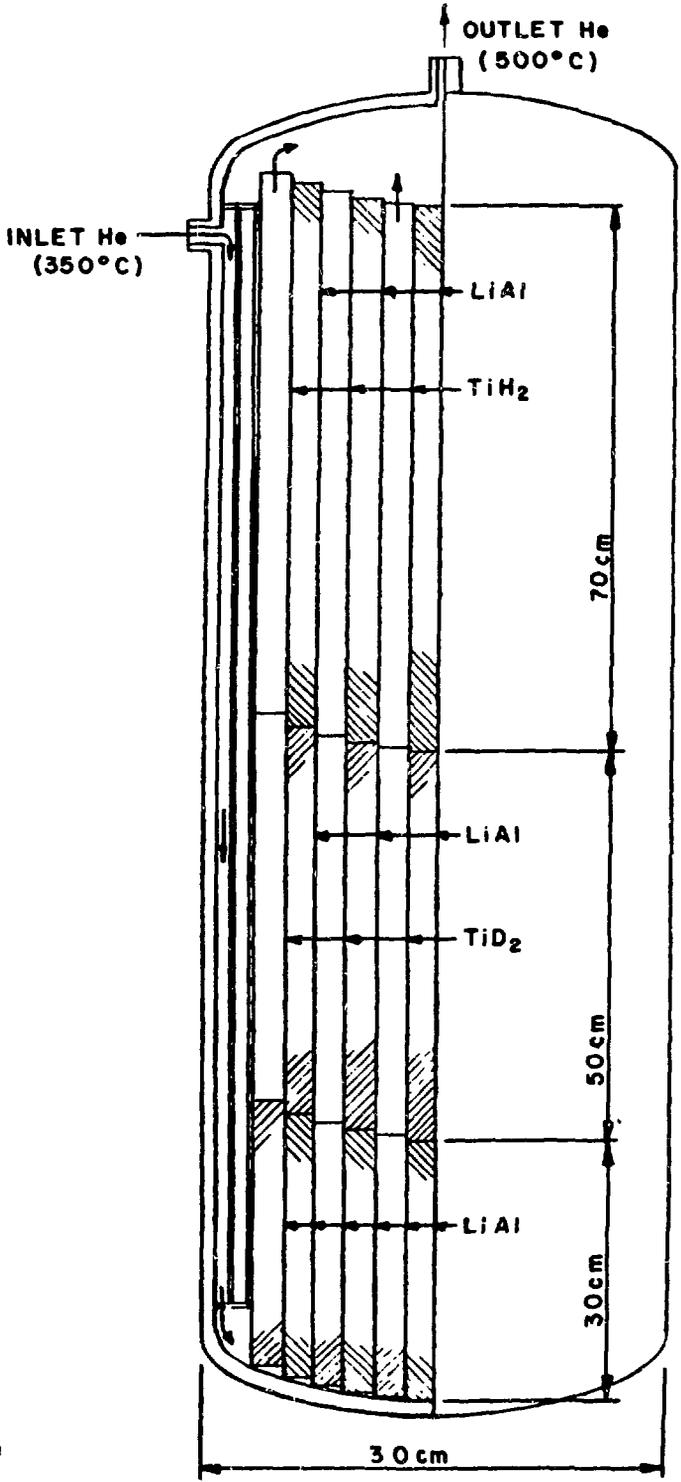


Figure 1



SCHEMATIC ELEVATION VIEW OF  
SOLID BLANKET MODULE #1

Figure 2

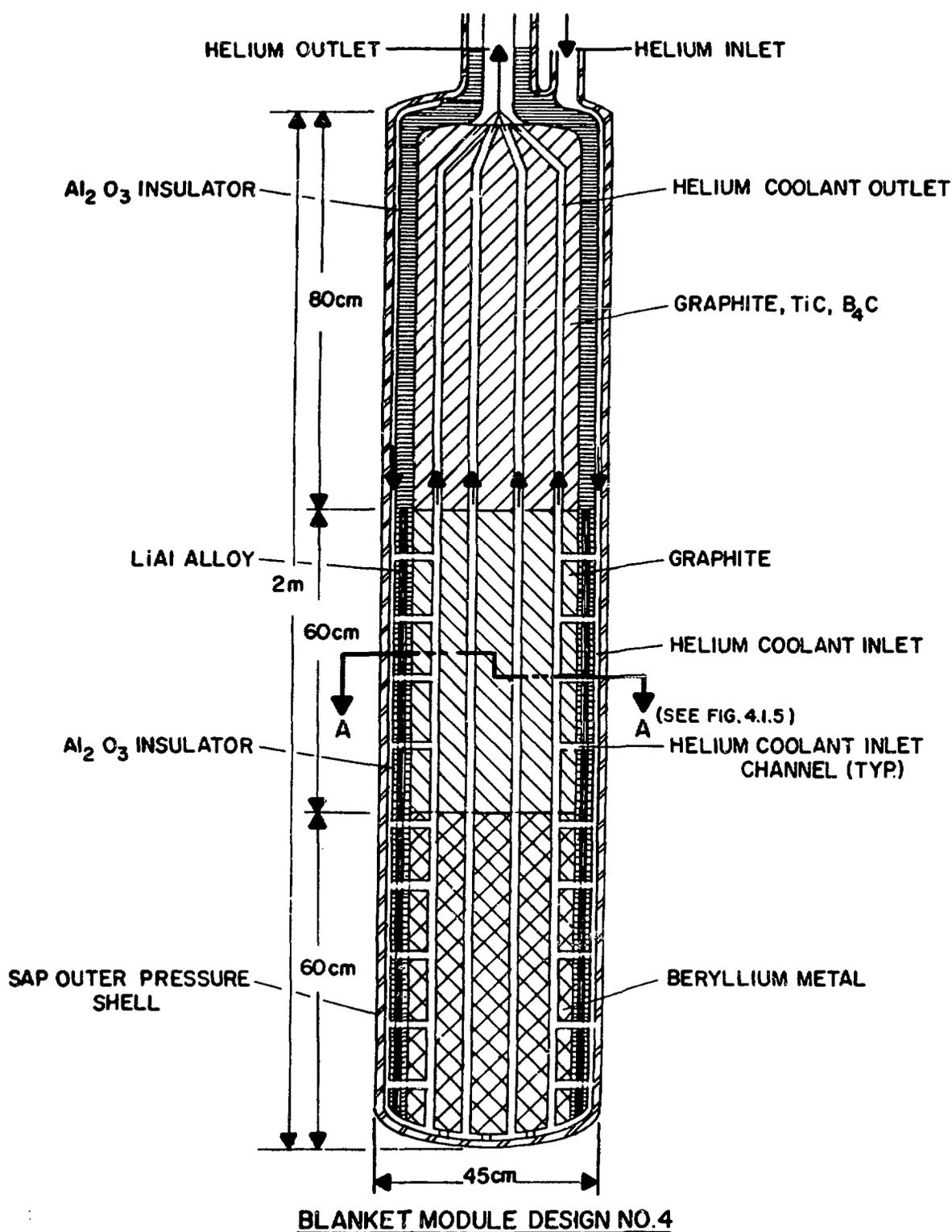
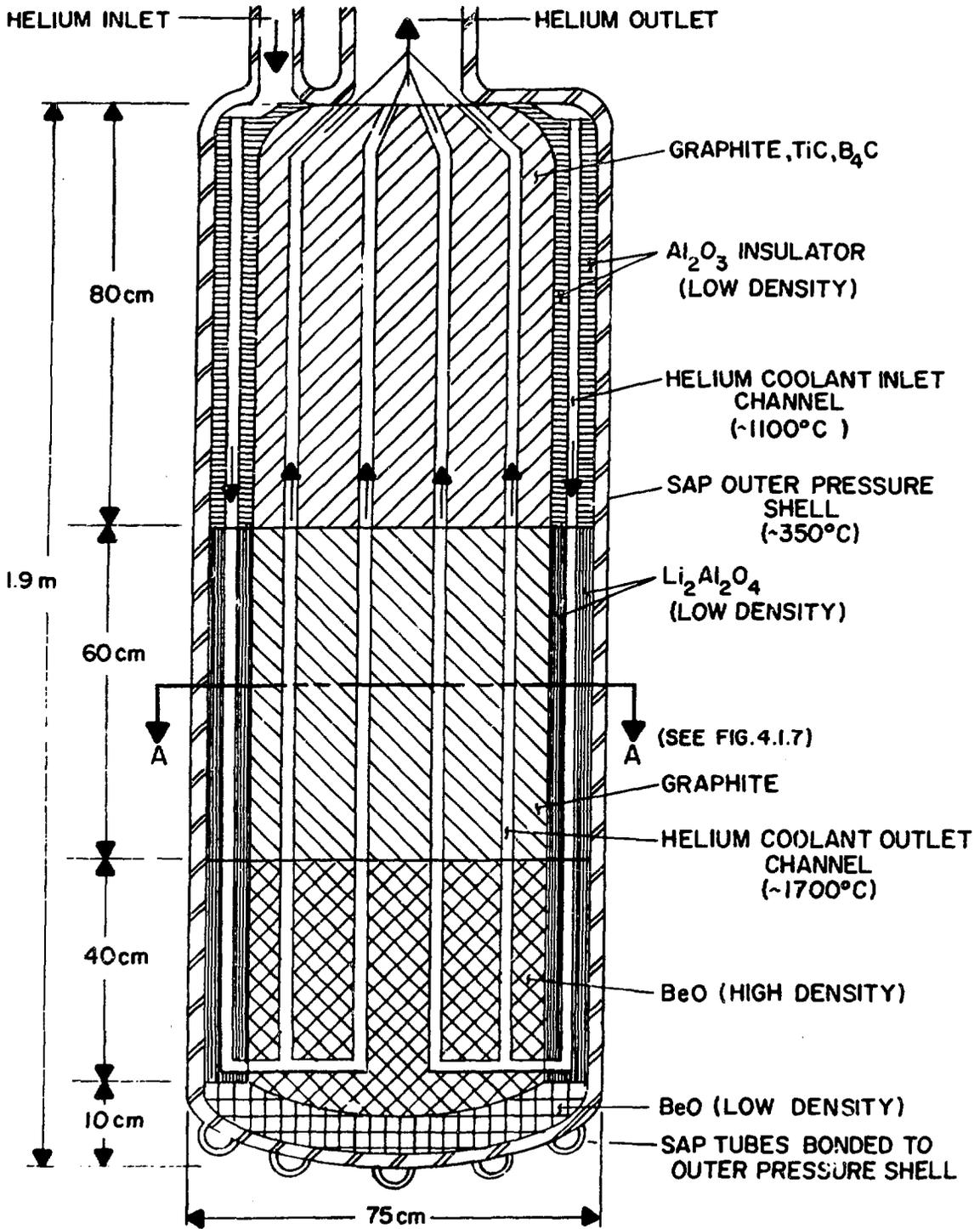


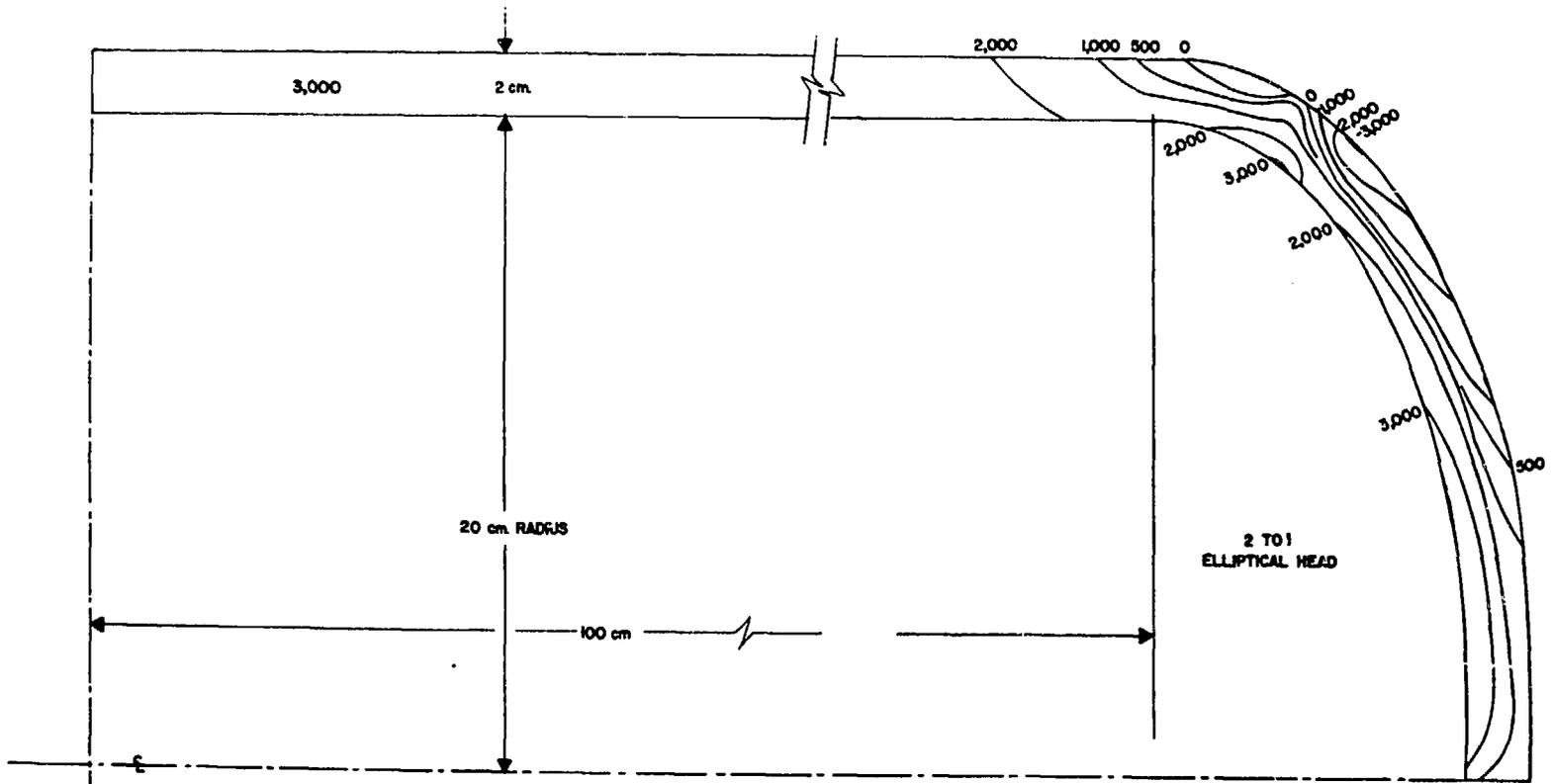
Figure 3



**BLANKET MODULE DESIGN NO.7**

Figure 4

Figure 5



THETA STRESS DISTRIBUTION IN SHELL OF LOW ACTIVITY MODULE - PRESSURE (20 ATM)  
AND FIRST WALL LOADING OF 1 MW (th)/m<sup>2</sup>

**RADIOACTIVE INVENTORY IN BLANKETS  
USING ALUMINUM STRUCTURE**

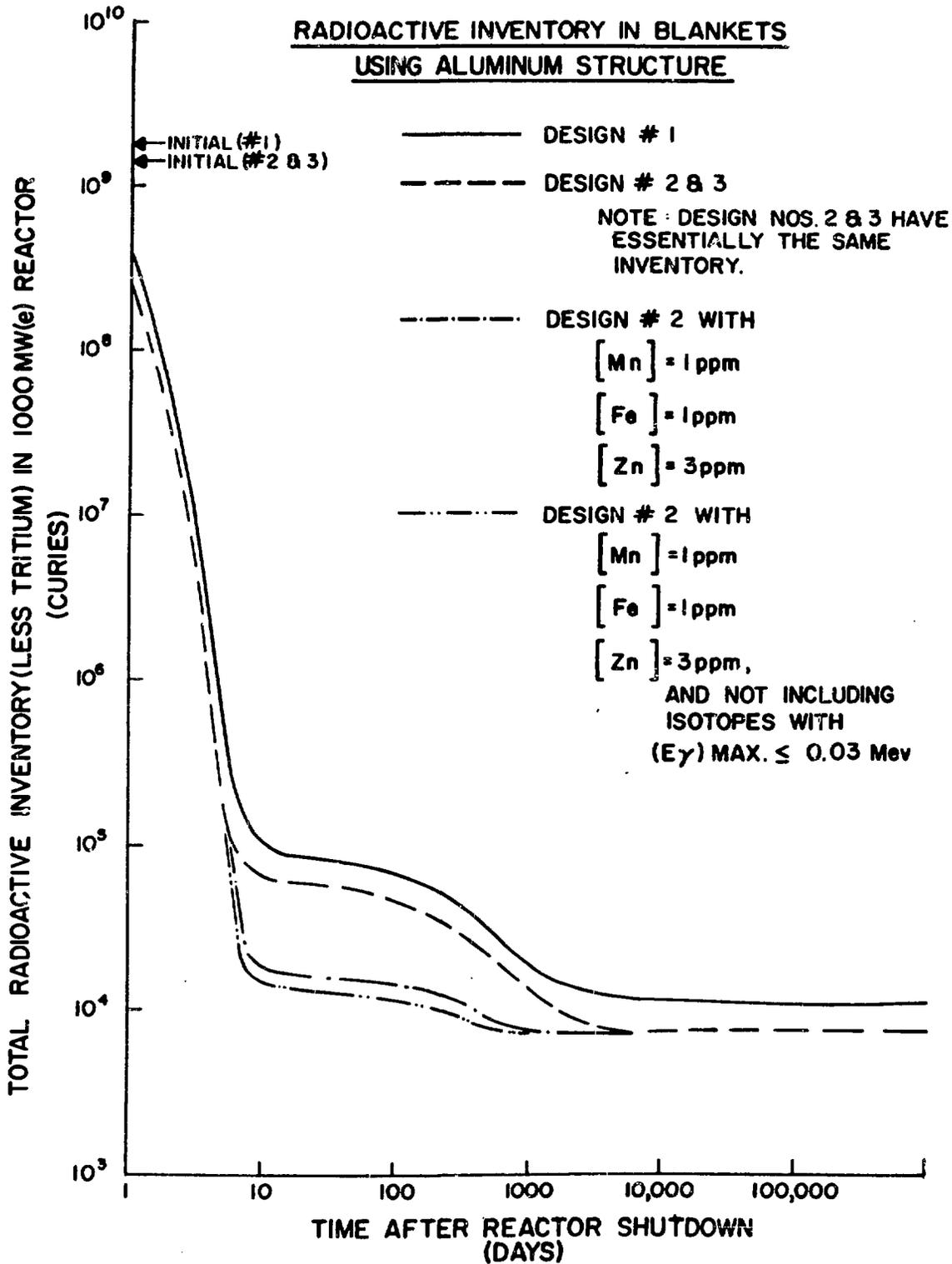
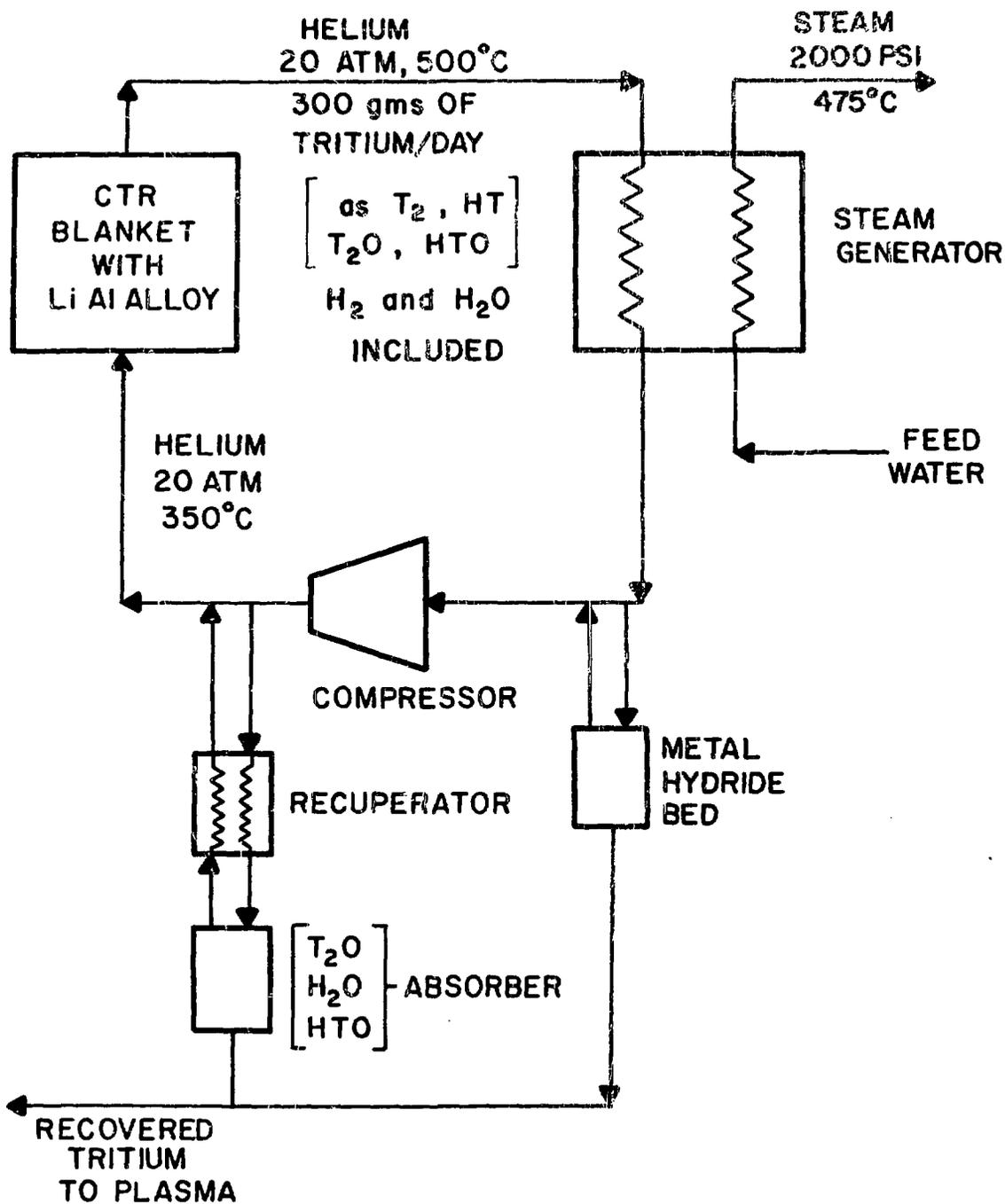


Figure 6



TRITIUM LEAKAGE TO STEAM GENERATOR, 1 CURIE / DAY

Figure 7

TRITIUM INVENTORY IN 1000 MW(e) REACTOR WITH BREEDING  
IN SOLID LIAI ALLOY

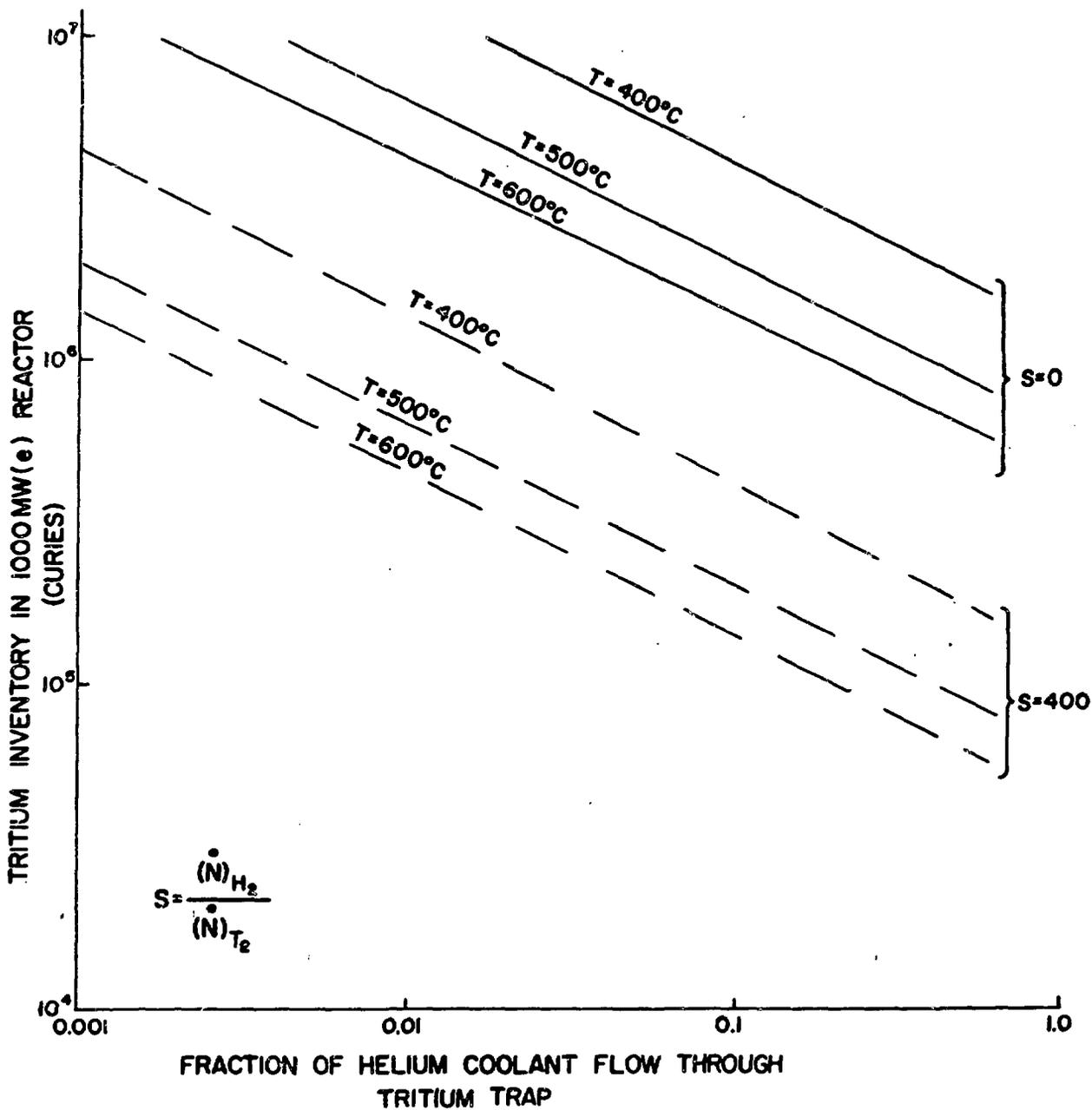


Figure 8