



**Thermal, Mechanical and Neutronic Design
Considerations for a Graphite Structure Li_2O
Cooled Blanket**

D.K. Sze, I.N. Sviatoslavsky, E.T. Cheng, and H.I. Avci

October 1977

UWFDM-223

***FUSION TECHNOLOGY INSTITUTE
UNIVERSITY OF WISCONSIN
MADISON WISCONSIN***

**Thermal, Mechanical and Neutronic Design
Considerations for a Graphite Structure Li_2O
Cooled Blanket**

D.K. Sze, I.N. Sviatoslavsky, E.T. Cheng, and
H.I. Avci

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

<http://fti.neep.wisc.edu>

October 1977

UWFDM-223

THERMAL, MECHANICAL AND NEUTRONIC DESIGN CONSIDERATIONS FOR
A GRAPHITE STRUCTURE Li_2O COOLED BLANKET

D.K. Sze, I.N. Sviatoslavsky, E.T. Cheng and H.I. Avci
University of Wisconsin-Madison
Madison, Wisconsin 53706

The problems associated with the replacement and repair of the blanket of a fusion reactor are some of the most challenging ones in the field of fusion technology. Redundancy of the first wall may be needed to provide a viable reactor design if the first wall maintenance will be time consuming. Therefore, blanket repair and replacement schemes should be important input parameters to the system design, rather than added on as an afterthought of a design study. This paper presents a blanket design that provides built-in first-wall redundancy and requires minimum maintenance. The design is for a tokamak geometry, but is applicable to other plasma confinement concepts.

The first wall of a fusion reactor is the area which receives the highest radiation damage and heat flux. It is also the region that has the most severe thermal cycling, highest stress and reaches the highest temperature during most accidents. In addition, the first wall is the vacuum barrier between the plasma and the blanket. In most designs the blanket coolant pressure is on the order of $\sim 10^3$ psi, while the plasma chamber is at 10^{-7} psi. This 10 order of magnitude difference in pressure makes the plasma vulnerable to a first wall leak; and the design complication needed to prevent a first wall leak makes first wall replacement very difficult. The severe environment of the first wall, on the other hand, will require its frequent repair and scheduled replacement. The problem is further complicated by the tokamak geometry.

The blanket maintenance scheme can be considerably simplified if the pressure difference across the first wall can be eliminated. Various designs^{1,2} have suggested that the vacuum seal to the reactor room be moved to the back of the blanket. The first wall, however, is still the vacuum wall between the blanket and the plasma. The pressure in the blanket is mainly due to the pressure of the coolant. To really eliminate the vacuum requirement of the first wall, a very low pressure coolant has to be found.

A recent study by Sze^{3,4} has proposed the use of lithium oxide pellets flowing by gravity as the coolant. This coolant is a good heat-transport medium, but not a good heat-transfer medium. For this reason the first wall has to be separately cooled. If the first wall can be cooled by thermal radiation, the pressure of the blanket can be very low. To keep the first wall at a high operating temperature, graphite is used as the structural material. The advantages and problems associated with such a design are discussed in this paper.

System Description

Heat transfer problems associated with fusion reactor blankets are very different than that of fission reactors, especially if a lithium compound is used as the coolant. For tritium breeding reasons, 80 to 90% of the blanket is occupied by the lithium compound. In such a blanket, 80% of the energy is deposited within the coolant, with most of the rest deposited on the first wall. Therefore, with the exception of the first wall, there is really no heat

transfer problem but rather a heat transport problem. This idea was used by Sze^{3,4} to develop the concept of lithium oxide pellets flowing by gravity as the cooling-breeding material for a tokamak reactor. This concept is equally applicable to laser fusion reactors.⁵

The flow scheme in such a design is shown on Fig. 1. The lithium oxide introduced at the top of the reactor flows by gravity through the blanket and exits from the bottom of the reactor. The oxide pellets are transported upward by a mechanical transport system (i.e., a screw conveyor) and then again flow by gravity through the steam generator. A steam generator with moving solid particles on the shell side is being studied and initial heat transfer results are favorable⁶. Another mechanical transport system will lift the coolant up for re-entry into the fusion reactor. Such a circulating system can be operated at a low pressure. The vapor pressure of Li_2O at 600°C is on the order of 10^{-9} torr⁷. The pressure in the coolant will consist of outgassing and system leakage, as well as its partial pressure. It is our opinion that a pressure of 10^{-2} to 10^{-3} torr can be maintained in the blanket with proper vacuum pumping design.

The blanket structure is graphite in a modularized design. Each module is bolted and supported to an aluminum support plate located between the blanket and shield. This aluminum support plate also provides the vacuum seal between the reactor room and the plasma. The blanket sections fit into each other without providing a vacuum-tight seal, which is no longer needed since the blanket itself is evacuated. In such a design, the time consuming process of removing weld can be eliminated. First wall redundancy is built-in since there is no pressure difference across it. A blanket module is schematically shown in Fig. 2. It should be noted that the flow velocities through different channels are regulated to account for the non-uniform heating rate in the different zones.

Heat Transfer Considerations

The blanket has no heat transfer problems except in the first wall. The only serious problem is to assure that the coolant will exit from blanket at a nearly uniform high temperature. Solid particles flowing by gravity have little lateral movement so that close velocity control will be added to offset the effect of non-uniform nuclear heating. The lithium oxide at 400° is fed through an inlet tube, as shown on Fig. 3 which is connected to a blanket module. The blanket modules are divided by baffles, as indicated on Fig. 2. The velocities of the lithium oxide streams are controlled by louvers near the exit such that they are inversely proportional to the nuclear heating rate. The coolant exits at 600°C and is routed to the steam generator.

The first wall receives a high heat flux. In the previous design³, the first wall was separately cooled by boiling water. If graphite is used as the

structure, the first wall can be radiatively cooled. The first wall temperature facing the plasma is $\sim 1400^\circ\text{C}$ in such a system with a first wall thermal loading of 40 W/cm^2 . If the first wall thickness is 1 cm, the temperature gradient across it will be 90°C . This temperature gradient results in a low thermal stress. The level of thermal stress and its implication will be discussed later.

The steam generator design for such a system was done by D. C. Schluderberg as discussed in Reference 3. The steam conditions obtainable are 1000°F , at 2500 psi with 1000°F reheat. A gross thermal efficiency of 42.3% was indicated.

Vacuum Considerations

A blanket of this design would be difficult to operate in high vacuum. Vapor pressure, water leakage through the steam generator, outgassing and vacuum seal leakage all contribute to maintain a certain pressure in the coolant loop. The pressure level at which such a loop can be maintained is hard to decide at this stage. It is our opinion that a pressure of ~ 1 torr can be maintained, while a pressure of 10^{-5} torr may not be possible. A blanket pressure of 10^{-2} to 10^{-3} torr may be possible with proper vacuum pumping. A more definitive statement will require much more research.

It would be most attractive to maintain the blanket at the same pressure as the plasma. The advantage of applying this concept to a laser system is evident because of the relatively high pressure in the laser cavity. For a magnetically confined system, however, the gas leakage rate to the plasma as a function of the blanket pressure should be determined and compared to the vacuum pumping capacity of the reactor.

If the blanket pressure is $\sim 10^{-2}$ to 10^{-3} torr, the flow is still in the viscous regime and a leak rate can be calculated by the following equation:

$$V = \frac{P^2 g_c D^2}{64 \mu \pi L}$$

in which V is the leak rate per unit area

P is the blanket pressure

g_c is a conversion factor

D is the diameter of a pin hole in the blanket

μ is viscosity

P* is pressure where V is calculated

L is the length of a pin hole

If $P = 10^{-2}$ torr, $D = 10^{-2}$ cm, $L = 1$ cm, the leak rate is only 1.5 liter/ m^2 -sec, which is 3 orders of magnitude smaller than the pumping capacity of typical tokamak reactor. Therefore, it may be concluded that if the blanket can be maintained in the range of 10^{-2} - 10^{-3} torr a pressure in the plasma chamber of 10^{-5} torr is possible.

Mechanical Considerations

The idea of using graphite as the first wall and blanket material for a tokamak fusion reactor was initially proposed in 1974⁸. Although the idea is very attractive in many ways it poses some very challenging difficulties from a mechanical standpoint.

This paper proposes the use of a graphite structure blanket in conjunction with a gravity circulated Li_2O

bed concept. For a reference design, we have chosen the UWMK-III⁹ plasma geometry but without the diverter. The surface wall loading will be increased to account for this difference. Figure 3 shows a cross section of a blanket module.

The design assumes that Li_2O microspheres enter the blanket at the top, flow through it by gravity and exit on the bottom. The blanket is ~ 83 cm thick on the outer leg and about 50 cm thick on the inner leg. The outer leg has a 1 cm thick first wall, followed by a 50 cm Li_2O zone, a 20 cm thick graphite reflector zone, another 10 cm thick Li_2O zone and finally a 2 cm thick graphite back wall. The initial 50 cm Li_2O zone is divided into three compartments, such that the Li_2O velocity in each can be regulated to provide an approximately equal coolant temperature rise in each of the compartments. The inner leg of the blanket also has a 1 cm thick graphite wall followed by three 15 cm thick Li_2O zones with velocity regulation and finally, a 2 cm thick graphite back wall. The side walls are 2 cm thick. In the toroidal direction, the blanket is divided into 36 equal segments, which will be called modules. Each module is approximately 11.5 m high, weighing ~ 13 tonnes, of which ~ 8 tonnes is in the graphite reflector.

In UWMK-III the outer portion of the shield swings open on hinges to provide access to the blanket⁹. However, even such a large access port would be inadequate to allow the placement of a complete blanket module. For this reason it will be necessary to assemble the blanket module from several pieces, joining them together within the reactor, which might prove to be very difficult. The joints between blanket sections need not be vacuum tight, since the Li_2O will be in an environment of $\sim 10^{-2}$ torr. However, the Li_2O microspheres cannot be allowed to leak into the plasma chamber and thus the joints have to be good enough to prevent that. We have considered using a simple close fitting mechanical coupling but decided it would not work. Instead we have adopted the concept that the sections can be cemented together with a carbon cement and subsequently graphitized within the reactor.

The most likely candidate graphite structure for this application is the chopped fiber composites such as those used for ablation shields in re-entry vehicles. Such composites do not have the characteristic anisotropy with regards to such properties as thermal conductivity and coefficient of expansion which other bulk graphite structures display. The panels are produced by randomly orienting chopped graphite fibers, compacting them with a carbonaceous binder and then graphitizing the binder. This results in a fairly isotropic product with good mechanical properties. The panels are then cut to size and cemented together to form intricate shapes after which they again undergo a heat treatment to cure the cement binder. The very last step of joining the blanket sections into a full module can be performed in the reactor with a controlled hydrogen discharge similar to that used in present day experiments for baking out vacuum chambers.

For assembly within the reactor, the blanket module will be divided into four sections as shown in Fig. 4. The top and bottom sections are inserted first and attached to the inlet and outlet tubes. The transition between the graphite blanket and the stainless steel inlet and outlet tubes is made with a mechanical joint. A steel collar mounted on the graphite end flange is clamped to the supply or return tube by means of a

split yoke similar to the quick discount mechanical flange clamps in present day use. The intermediate sections can now be installed and Fig. 4 shows the direction from which they are inserted. Junctions between the sections will be made to overlap and, perhaps, even interlock in order to have a good mechanical as well as a cemented joint.

Obviously there are many questions that have to be answered with regards to this method of fabrication. Is it possible to graphitize the cemented joints in situ? Will the graphite at the joints have the same properties as the rest of the graphite? Can the stresses be kept low enough for the joints to survive?

Properties such as porosity for example, determine to a large degree the dimensional changes of graphite under neutron irradiation. Initial densification and subsequent swelling under irradiation is a strong function of temperature and neutron fluence. Regions of the blanket operating below 1000°C should be far enough back in the blanket where the neutron flux has already been attenuated.

In the proposed design, the blanket modules are contained within a water cooled aluminum vacuum chamber with an effective thickness of 5 cm. The reactor itself will be in a building evacuated to a pressure of 70 torr. The blanket and Li_2O environment will be evacuated to $\sim 10^{-2}$ torr and the plasma chamber will operate at about 10^{-5} torr. All the forces except those of gravity have been eliminated from the first wall and blanket. Where the gravity force is large, namely on the lower half of the blanket, the reinforced vacuum chamber wall will be used to support the blanket. This will be accomplished by building in insulated support points between the blanket modules and the vacuum chamber without producing a direct mechanical coupling. A direct coupling will introduce high stress points due to relative expansion and contraction of the two different materials. The gravity loading on the first wall itself can be minimized by properly designed radial structure within each blanket module. This structure can be optimized to give maximum support while minimizing thickness.

The thermal stresses in such a graphite structure appear to be less of a problem than in comparable metallic structures. This is primarily due to the fact that graphite has a relatively low coefficient of expansion and a low modulus of elasticity. It is estimated that with a neutron wall loading of 2 MW/m^2 , the temperature gradient across the first wall will be on the order of 100°C. Taking the following average values for the properties of graphite¹⁰.

Coefficient of expansion	$\alpha = 4 \times 10^{-6} \text{ }^\circ\text{C}^{-1}$
Young's Modulus	$E = 1.4 \times 10^{10} \text{ N/m}^2$
Poisson's Ratio	$\nu = 0.2$

the thermal stress is only $\pm 3.5 \times 10^6 \text{ N/m}^2$ ($\sim \pm 500 \text{ psi}$). This will certainly not be the dominant stress and consequently, fatigue does not appear to be a problem of concern in this case. Finally, radiation creep will tend to reduce the mechanical stresses in the blanket and further alleviate the problem of localized stresses.

Dimensional Instabilities and the Effect of High Helium Gas Content

Carbon or graphite goes through two stages during bombardment with neutrons. The first stage is

shrinkage by different amounts in volume and the second stage is expansion. In most of the data reported to date on the dimensional changes of carbon and graphite with neutron irradiation, the reversal from shrinkage to expansion takes place at fluences of about $1-2 \times 10^{22} \text{ n/cm}^2$ (5-10 dpa) at temperatures of 800-1400°C. Above 1400°C, the bulk of the data predicts a lower damage rate at the higher temperature¹¹.

Due to high $\text{C}^{12} (\text{n}, \text{n}') \text{ }^3\text{He}^4$ and $\text{C}^{12} (\text{n}, \alpha) \text{Be}^9$ reaction cross sections at high neutron energies the amount of helium generated in carbon first wall will exceed 2500 appm per year per 1 MW/m^2 wall loading. It was originally thought that this much helium could coalesce to form large bubbles and cause considerable dimensional changes and tearing. However, some recent experiments show that helium generated may in fact diffuse out with almost 100% efficiency¹².

Vacuum Properties of Graphite

Graphite vaporizes in a number of modes releasing molecules with 1 or more carbon atoms. The vapor pressures of these various species are shown on Fig. 5. The free vaporization rate of graphite is shown on Fig. 6. It is obvious that at the operating temperature of the first wall, neither the vapor pressure, nor the vaporization rate, present any serious problems.

Outgassing of graphite was considered to be a problem. However, recent work by Lang¹³ and Beitel^{13a} indicate that with proper outgassing and baking at high temperatures the residual gas content of carbon can be lowered drastically. The study by Lang¹³ also shows that the sticking probability of H or air on graphite fibers is $\sim 10^{-8} - 10^{-10}$. Therefore it is anticipated that from a vacuum properties point of view the use of carbon or graphite in tokamaks should not present any problem.

Sputtering of Graphite

Sputtering of carbon in a fusion reactor environment is a complicated problem. Both physical sputtering and chemical sputtering take place and they are important in different regions. However, there have been a few recent studies utilizing hydrogen isotopes to measure the sputtering behavior of carbon and some results from these studies are summarized in Figures 7^{14,15} and 8¹⁶. The following conclusions can be made from these results.

(1) The sputtering coefficients rise from a value of 10^{-2} atom/atom at room temperature to a maximum of $\sim 8 \times 10^{-2}$ at 600°C. This peak has been found to be associated with methane formation.

(2) As the temperature is increased above 600°C, the sputtering values start to return to their low initial values. This is accomplished by a reduced methane formation.

(3) The sputtering coefficient from tritium is reduced as its energy is increased from 2 to 6 keV. However, 6 keV tritium is more effective in producing methane than is the same energy deuterium.

(4) The low energy hydrogen bombardment depicted in Figure 8 shows smaller absolute levels of methane formation than for higher energy hydrogen. Furthermore, the low energy sputtering is quite sensitive to crystal orientation.

(5) Above 1000°C, the formation of acetylene becomes important although the absolute values are still low ($\ll 10^{-2}$ atom/atom).

(6) There appears to be a region between 800 and 1200°C where little gasification occurs.

Physical sputtering ratio of graphite by helium ions is practically independent of temperature and reported to be 0.06 for 0.4 KeV He ions¹⁷. The self sputtering yield of carbon in a tokamak environment is estimated to be 0.5 ~ 0.6. The best estimate of 14 MeV neutron sputtering of carbon is $\sim 10^{-4}$. A more detailed analysis of sputtering of graphite can be found in reference 21. The total sputtering calculation is summarized in Table 1. The total sputtering loss is 1.9×10^{11} atoms/cm²-sec, or 10^{-4} mm/year. Therefore the sputtering loss is acceptable.

NEUTRONICS

The neutronics calculations have been performed using the ANISN code,¹⁸ with P_3S_8 approximation, in cylinder geometry. The nuclear data used is the same as in Reference 19. Two design calculations with 2 and 10% graphite structure in the lithium oxide zones are completed. The neutronics results are tabulated in Table 2. The tritium breeding ratio in these designs is ~ 1.3 . The total nuclear heating in these systems is ~ 15 MeV per D-T neutron, among which $\sim 13\%$ is contributed by gamma-ray heating.

The radioactivity, afterheat and biological hazard potential (BHP) of this blanket are calculated, with either aluminum (Design I) or stainless steel supporting structure (Design I*), and shown on Figures 9, 10, 11. Those value are compared to that of UWMAK-I. The advantage of the graphite blanket is obvious.

Table 1

Sputtering From The Carbon First Wall Under
UWMAK-III⁽⁹⁾ Operating Conditions

Particle	Mean Energy keV	Incident Particle(*) Current-cm ² s ⁻¹	Sputt. Coeff.	Flux of Sputtered Atoms-cm ² s ⁻¹
D+	3	5.8×10^{12}	0.01	5.8×10^{10}
T+	3	5.8×10^{12}	0.01	5.8×10^{10}
He++	3	6.8×10^{11}	0.1	6.8×10^{10}
He++	100	6.8×10^{10}	0.05	3.4×10^9
n	14.1×10^3	4.43×10^{13}	10^{-4}	4.4×10^9
Total				1.9×10^{11}

(*) Adjusted to 1 MW/m² average neutronic wall loading.

Table 2

Summary of Neutronics Results in the Graphite
Structured Blanket Designs

Volume Percent Graphite in Breeding Zones	Design I 10%	Design II 2%
Tritium Production from ⁷ Li(n,n' α) Reactions	0.4512	0.4961
Tritium Production from ⁶ Li(n, α) Reactions	0.8497	0.8660
Tritium Breeding Ratio	1.3009	1.3621
Neutron Heating	12.9764	13.1801
Gamma-Ray Heating	1.9018	1.8474
Total Nuclear Heating	14.8782	15.0275
Energy Leakage to the Shield	0.0176	0.0478

Note: In units of tritons per D-T neutron or MeV per D-T neutron.

Conclusions

A graphite blanket cooled by gravitational flow of lithium oxide is introduced. Due to the low pressure of the blanket, the first wall is no longer required as the vacuum boundary. The maintenance and repairment of such a first wall will be much simplified. Redundancy of the first wall is a built in function. Many problems remain to be solved; among which the most critical ones are blanket vacuum problem, ceramic technology, vacuum heat transfer. Much further works are required to establish the feasibility of this concept.

Acknowledgements

This work is supported by Division of Magnetic Fusion Energy of Department of Energy and the Wisconsin Electric Research Foundation. Thanks are also due to Mrs. Bonnie Smart for the preparation of the manuscript.

References

1. S. Förster, et. al., "Conceptual Design of a Compact Tokamak Reactor", IAEA Conference and Workshop on Fusion Reactor Design, October 10-21, 1977, Madison, Wisconsin.
2. T. E. Shannon and D. Steiner, "ORNL Fusion Power Demonstration Study", IAEA Conference and Workshop on Fusion Reactor Design, October 10-21, 1977, Madison, Wisconsin.
3. D. K. Sze, D. C. Schluderberg and I. N. Sviatoslavsky, "Gravity Circulated Solid Blanket Design for a Tokamak Fusion Reactor", Second Topical Meeting on the Technology of Controlled Nuclear Fusion,
4. D. K. Sze, "A Low Maintenance, Low Pressure, Low Radioactivity Graphite Blanket Concepts for Fusion Reactor", Trans. Am. Nucl. Soc. 26, 35 (1977).

References

5. R. W. Conn, S. I. Abdel-Khalik and G. A. Moses, "The SOLASE Conceptual Laser Fusion Reactor", IAEA Conference and Workshop on Fusion Reactor Design, October 10-21, 1977, Madison, Wisconsin.
6. D. C. Schluderberg and T. A. Thornton, The B and W Company, Lynchburg, Virginia, private communication.
7. E. M. Larsen, R. G. Clemmer, and D. K. Sze, "Tritium Recovery and Handling in a Gas-Carried Li_2O CTR Blanket", Trans. Am. Nucl. Soc. 23,16 (1976).
8. G. R. Hopkins, "Fusion Reactor Applications of Silicon Carbide and Carbon", Proc. of the 1st Topical Meeting on the Technology of Controlled Nuclear Fusion, April 1974, San Diego, California.
9. R. Conn, et. al., "UWMAK-III, An Advanced Non-circular Tokamak Power Reactor Design", Nuclear Engineering Department Report UWFD-150, University of Wisconsin, Madison, 1976.
10. I. E. Campbell and E. M. Sherwood, "High Temperature Materials and Technology", John Wiley and Sons, Incorporated, 1967.
11. W. C. Morgan, "A Note on the Dimensional Change Rates of Graphite Irradiated Above 900°C", BNWL, 1976 (private communication).
12. J. B. Holt, et. al., "Helium Generation and Diffusion in Graphite", Lawrence Livermore Laboratory, Livermore, California, April 1976.
13. G. P. Lang and V. L. Holmes, "Gas Content and Sticking Probabilities for Fibrous Forms of Graphite as Applied to Fusion Reactors", Nuclear Fusion, 16, 162 (1976).
- 13a. G. A. Beitel, "The Use of Graphite in High and Ultra-High Vacuum - A Review" J. Vac. Sci. and Tech., Vol. 8, No. 5, p. 647 (1971).
14. H. Vernickel, EURATOM Association, Garching b. München, West Germany (private communication).
15. I. P. Busharov, E. A. Gorbatov, V. M. Gusev, M. I. Geseva and Y. V. Martynenko, "Chemical Atomization of Graphite by H^+ Ions", The I. V. Kurchatov Order of Lenin Institute of Atomic Energy, Moscow, 1975.
16. M. Balooch and D. R. Olander, J. Chem. Phys. 63, No. 11, December 1975.
17. N. Laegreid and G. K. Wehner, J. Applied Phys. 32, 365 (1961) and D. Rosenberg and G. K. Wehner, J. Appl. Phys. 33, 1842 (1962).
18. J. Koth, et. al., "Physical and Chemical Sputtering of Graphite and SiC by Hydrogen and Helium in the Energy Range of 300 to 600 eV", Phys. Rev., 1962.
19. W. W. Engle, Jr., "A User's Manual for ANISH", K-1693, Oak Ridge Gaseous Diffusion Plant (1967).
20. E. T. Cheng, T. Y. Sung and D. K. Sze, Nucl. Technol. 34, 362 (1977).
21. H. I. Avci et. al., "The Use and Performance of Graphite and Metal ISSECs in Tokamak Fusion Reactors", Report UWFD-202, Nucl. Engr. Dept., Univ. of Wisconsin, Madison, Wisconsin (1977) (Also submitted for publication in Nucl. Engr. and Design).

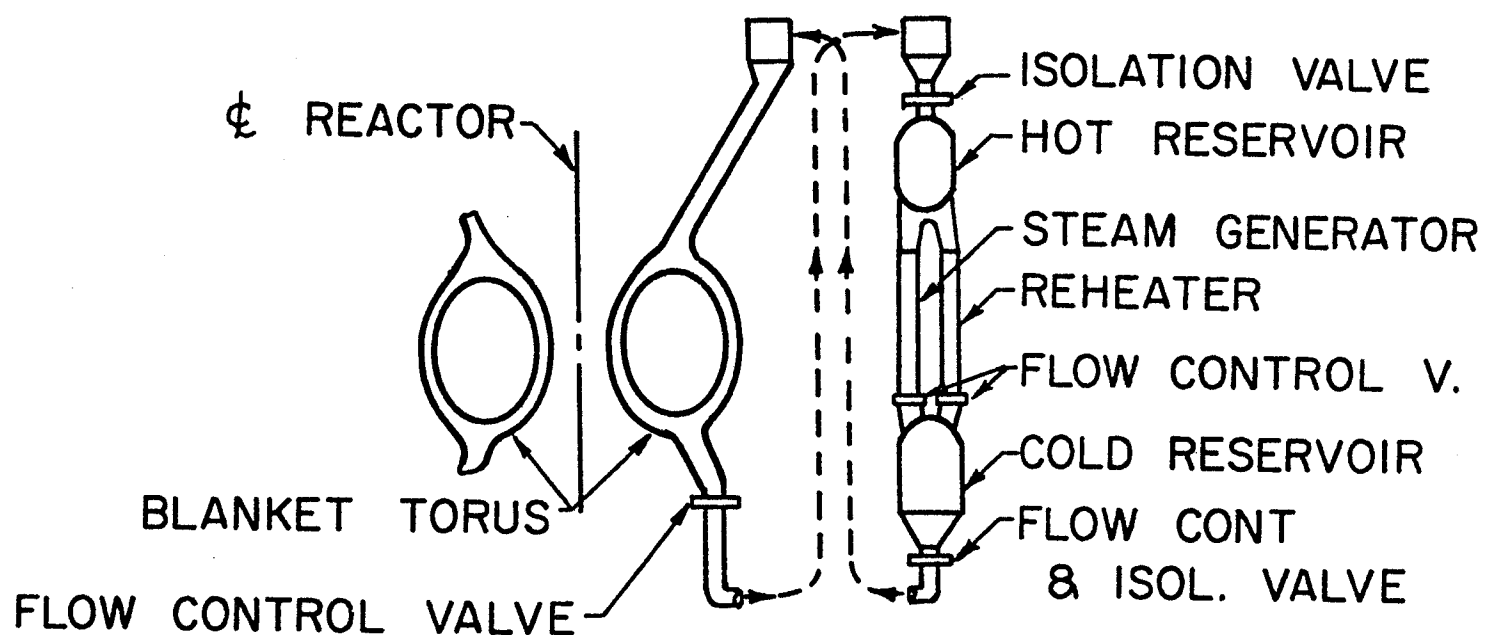


FIGURE I. SKETCH SHOWING GENERAL ARRANGEMENT
OF MOVING BED NSS LOOP FOR
TOKAMAK REACTOR

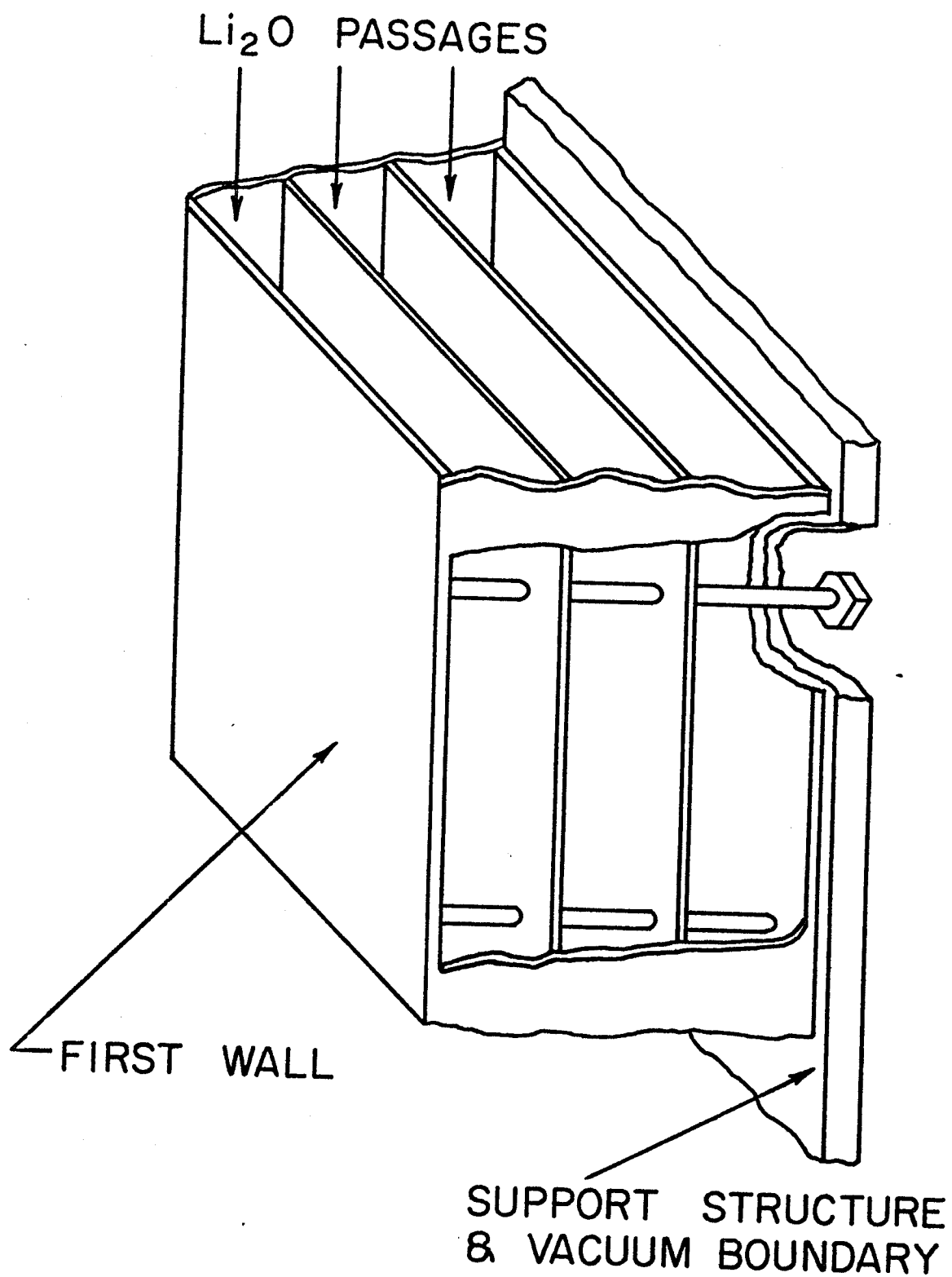


FIGURE 2 CROSS-SECTIONAL VIEW OF
THE BLANKET

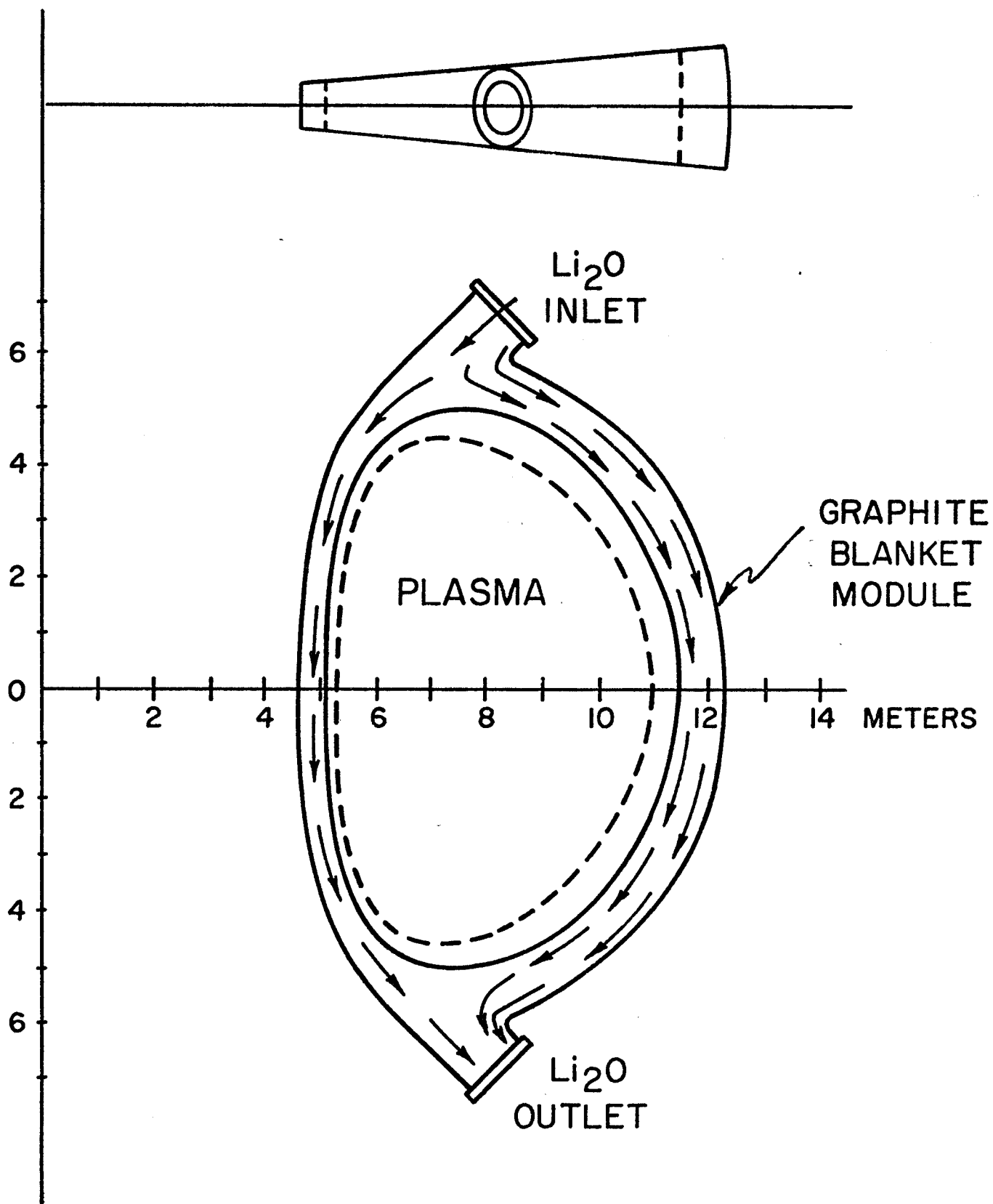


FIGURE 3 PLAN AND TOP VIEWS OF A GRAPHITE BLANKET MODULE

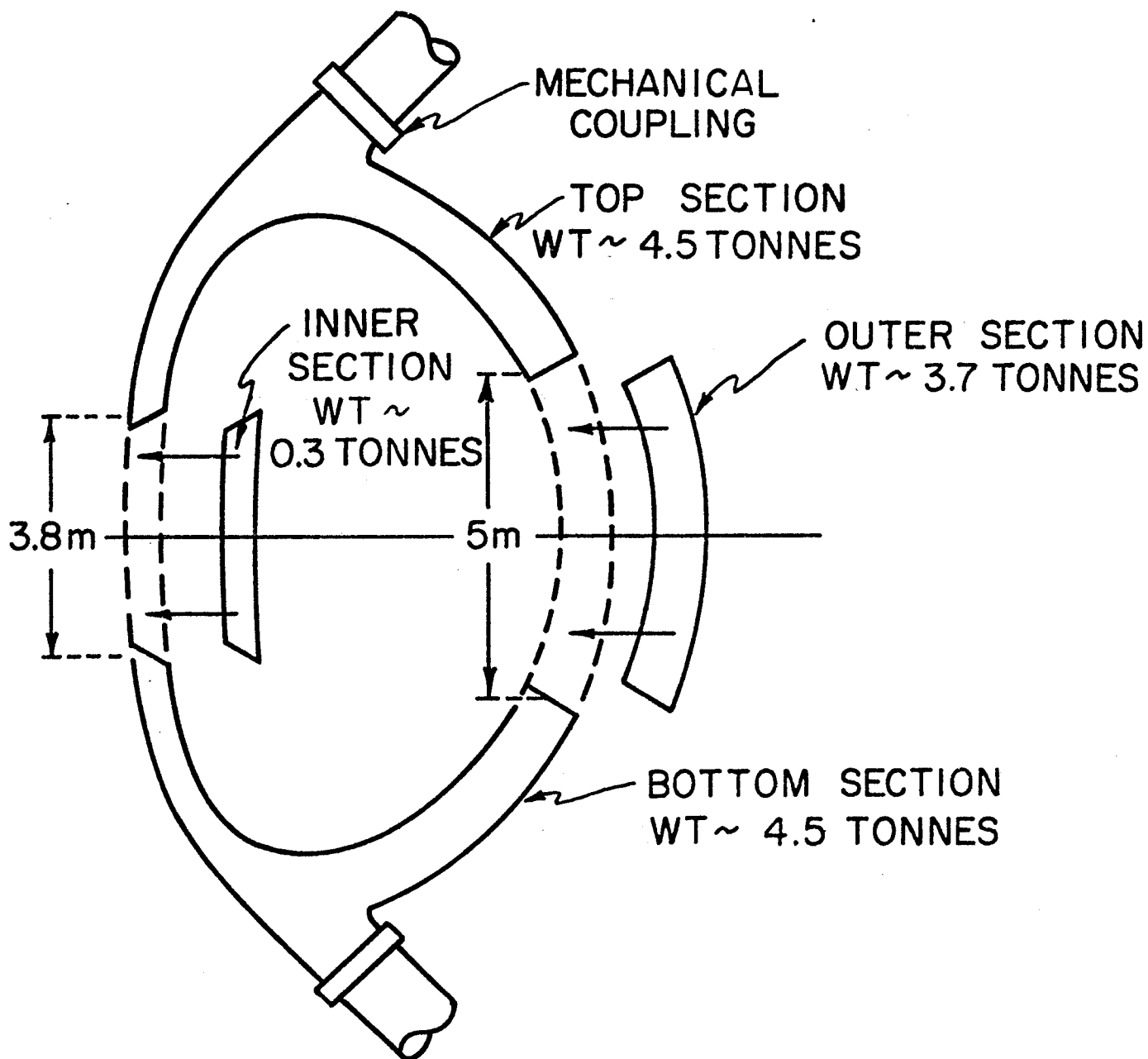


FIGURE 4 DIAGRAM SHOWING METHOD FOR ASSEMBLING A BLANKET MODULE

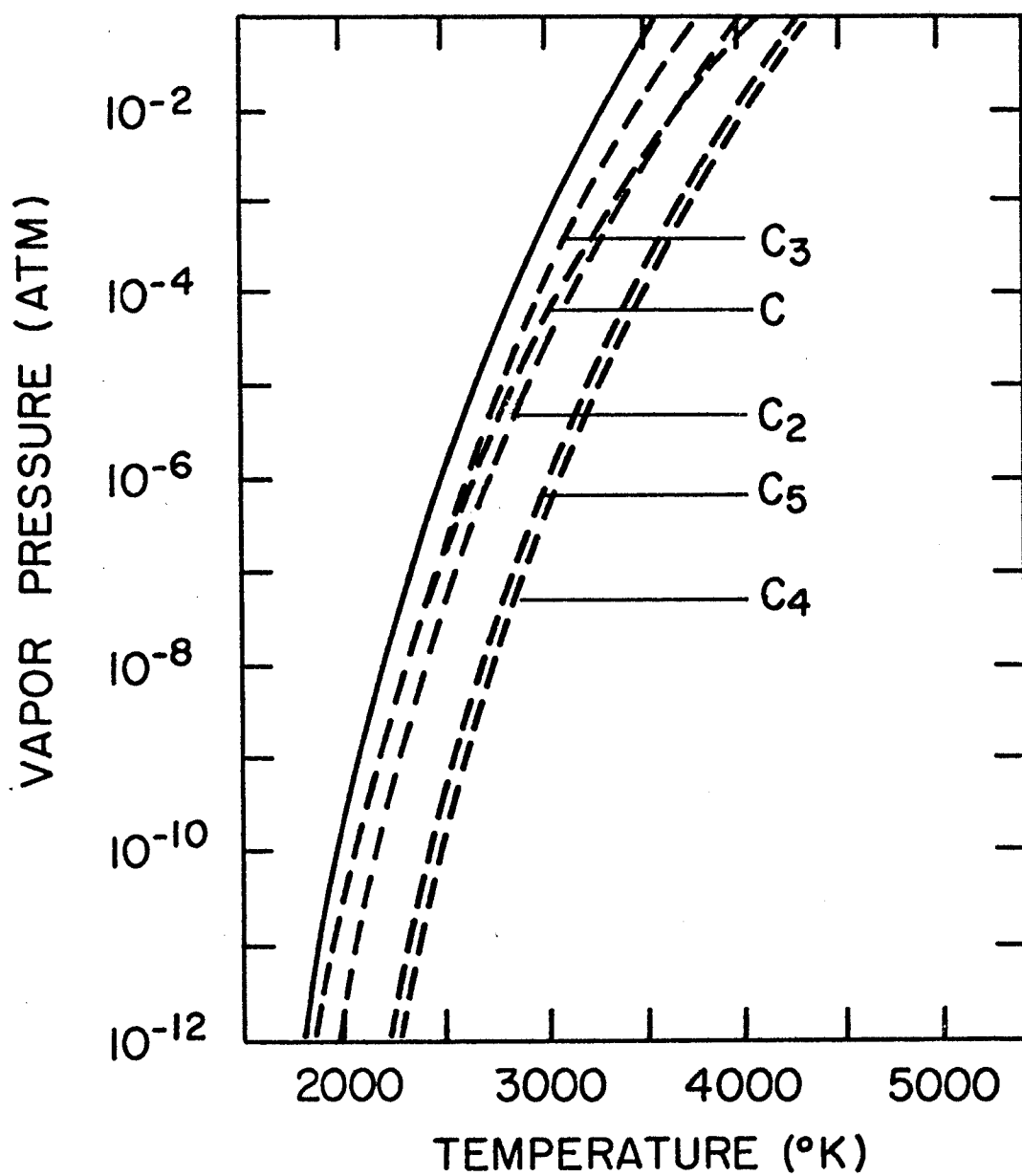


FIGURE 5 VAPOR PRESSURE OF GRAPHITE

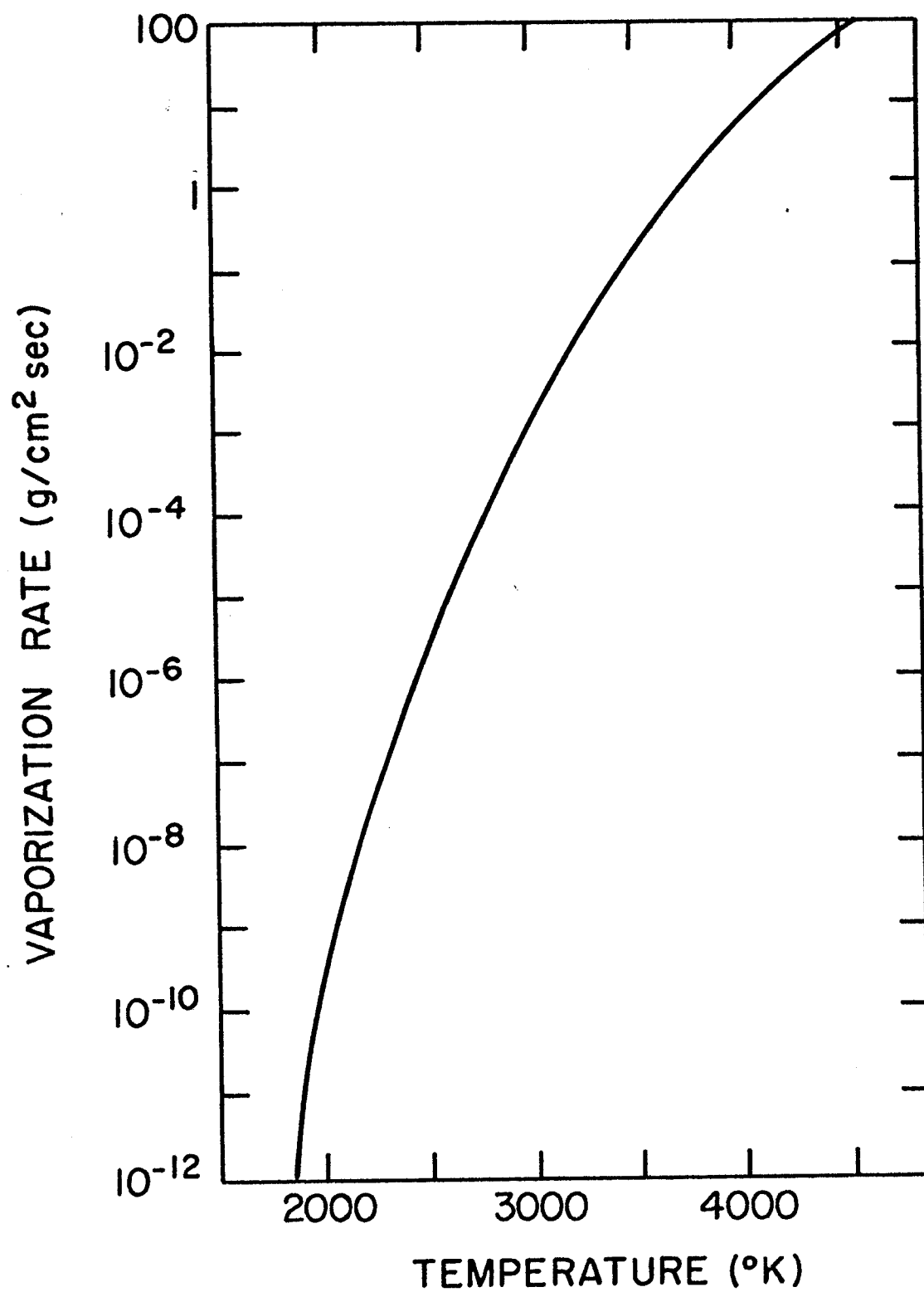


FIGURE 6 FREE VAPORIZATION RATE
OF GRAPHITE

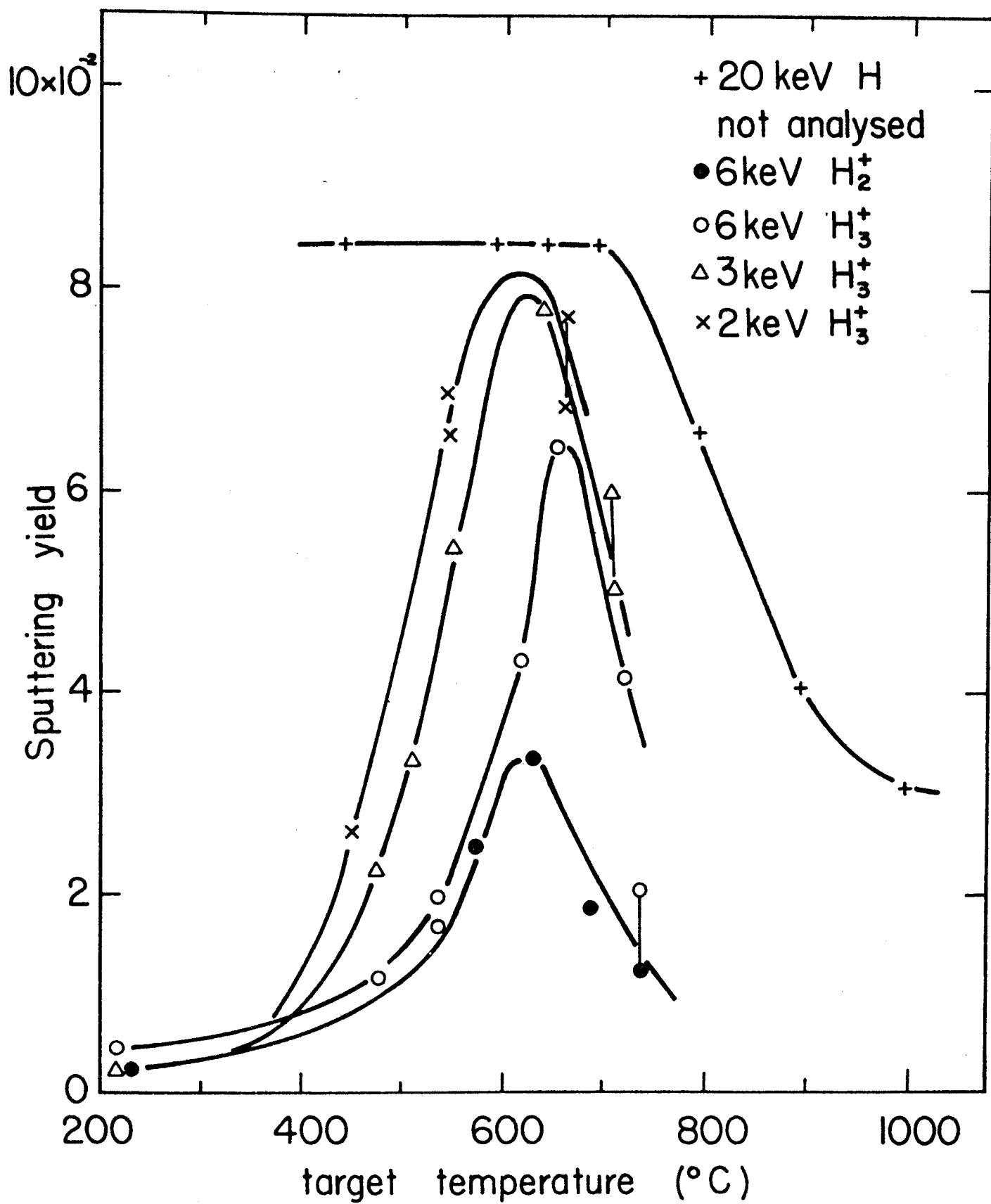


FIGURE 7 CHEMICAL SPUTTERING OF PYROLYTIC GRAPHITE

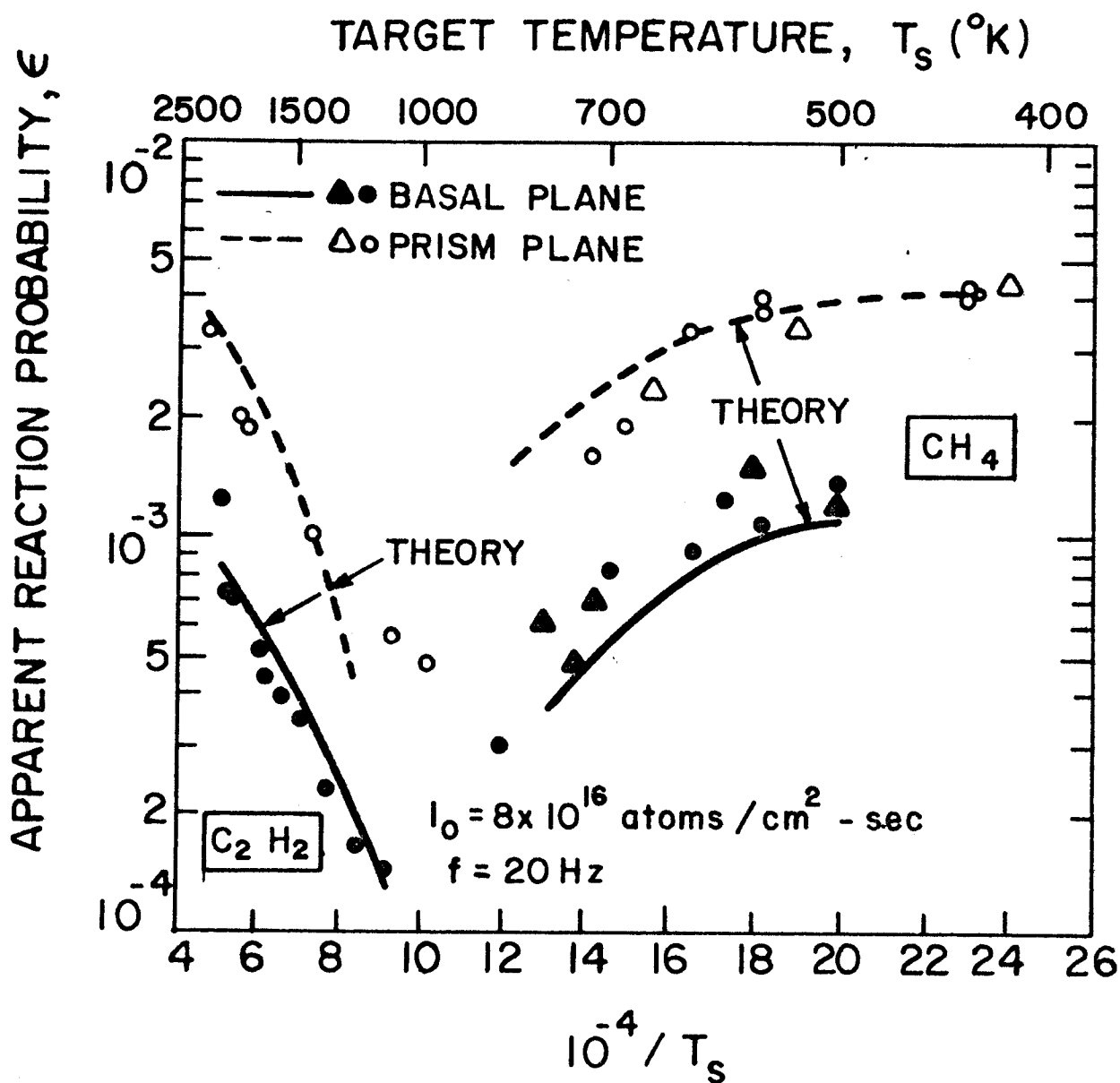


FIGURE 8 Effect of Temperature and Crystal Orientation on the Reaction Probability of Carbon

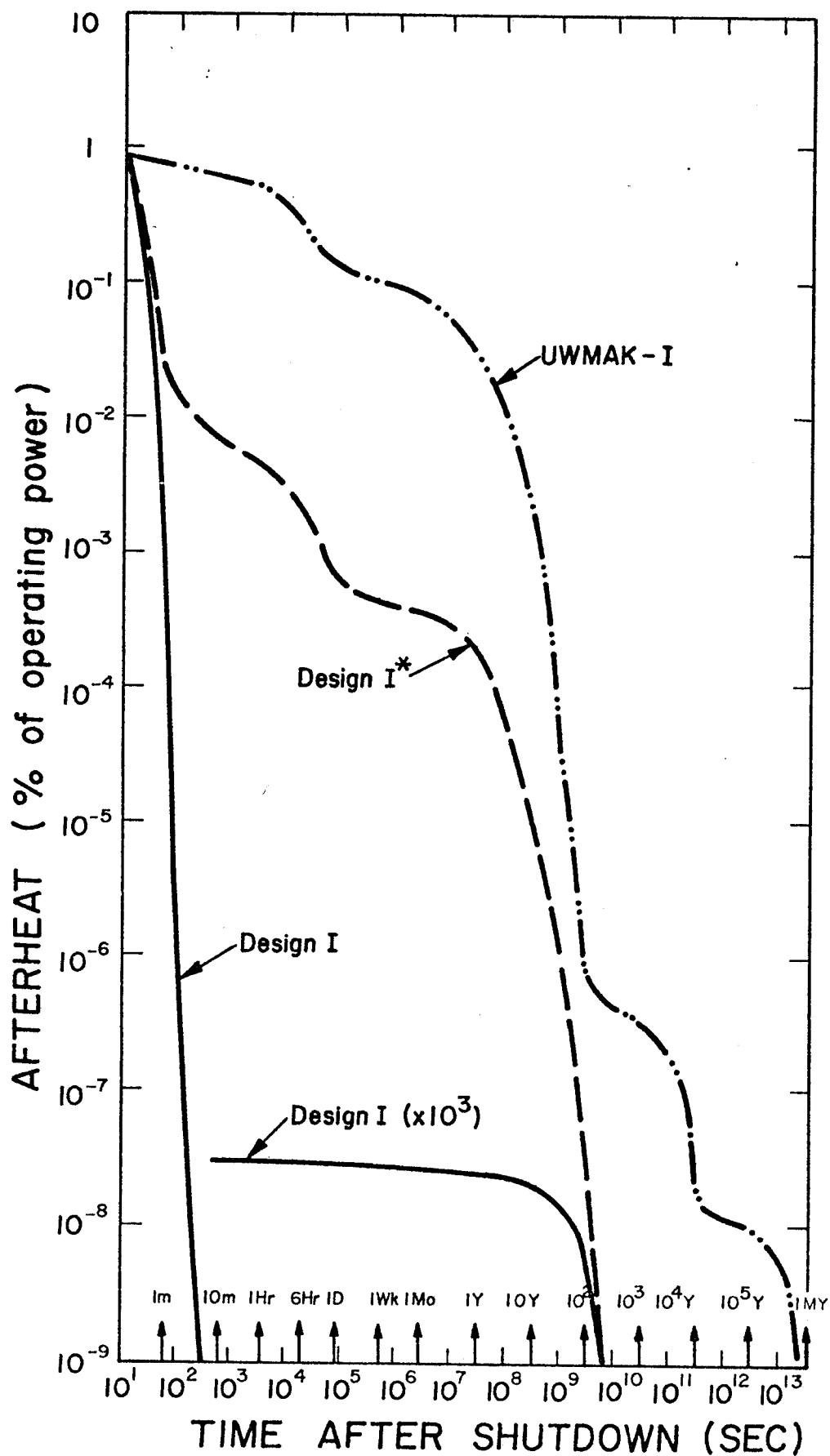


FIGURE 9 AFTERHEAT OF GRAPHITE BLANKET

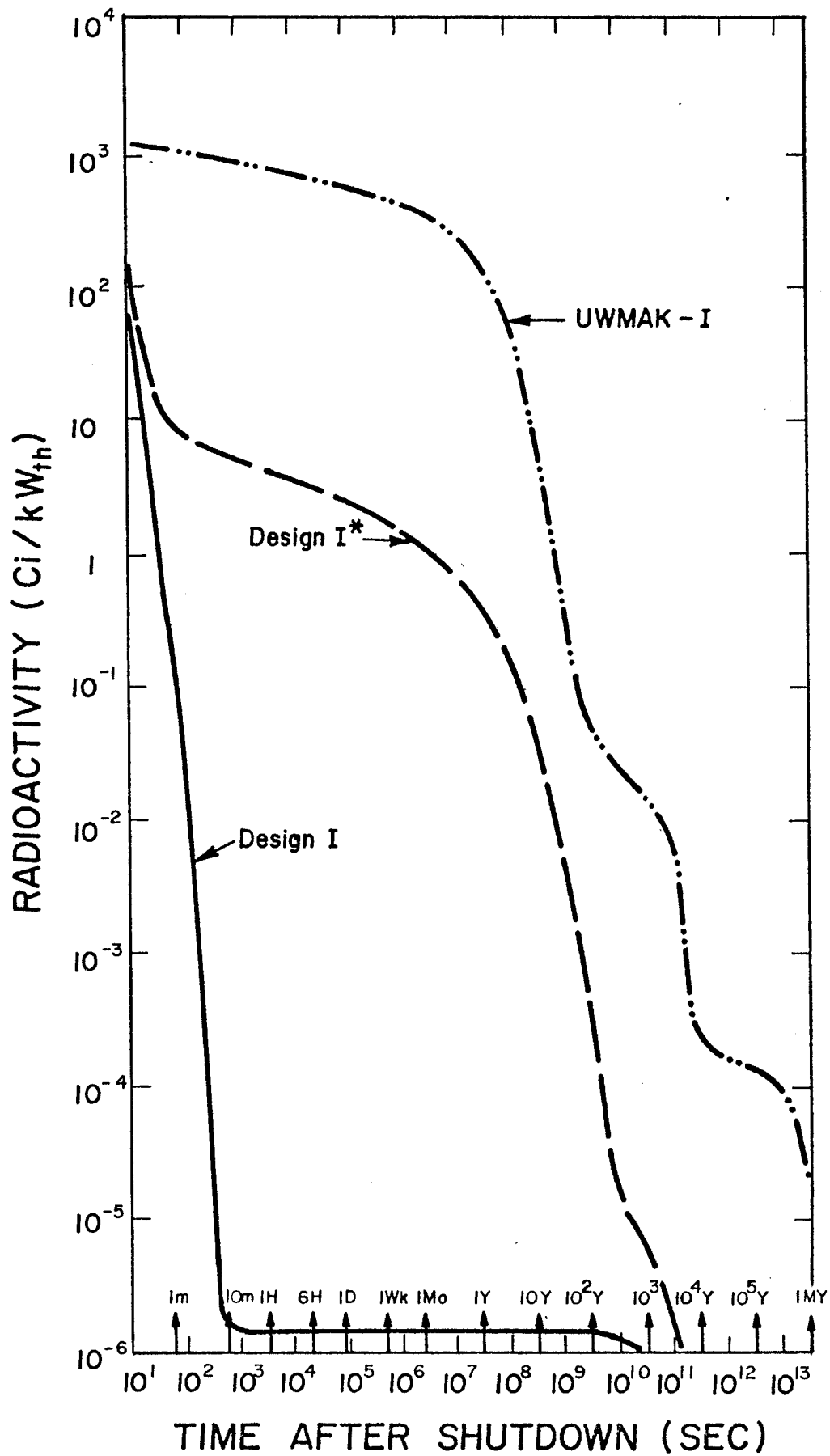
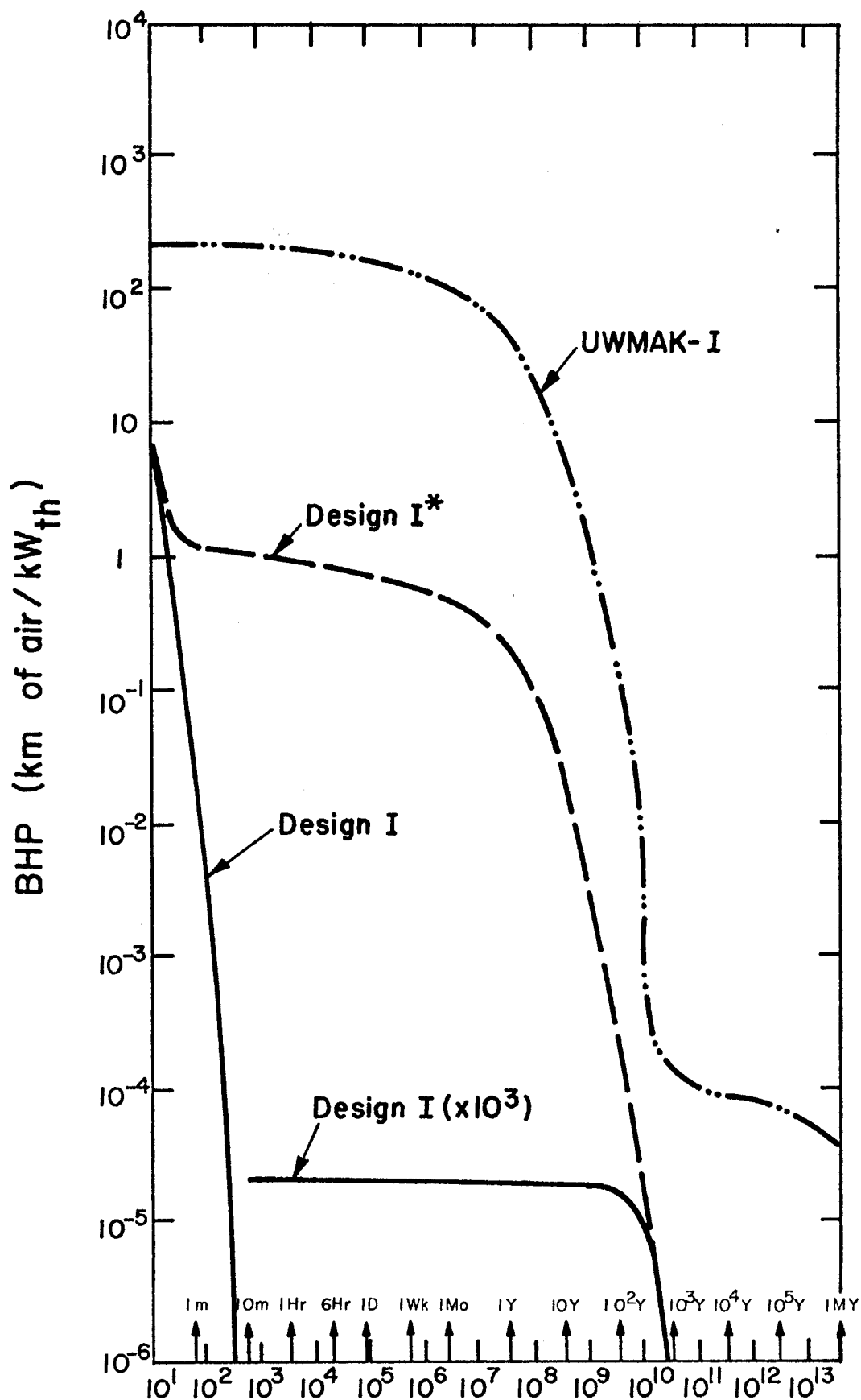


FIGURE 10 RADIOACTIVITY OF GRAPHITE BLANKET



TIME AFTER SHUTDOWN (SEC)
 FIGURE II BHP OF GRAPHITE BLANKET