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MFTF-B+T Device**

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## I. Introduction

It has been proposed that the present MFTF-B axicell experiment<sup>(1)</sup> be upgraded to burn deuterium and tritium.<sup>(2)</sup> The purpose of such a change would be to provide information about the operation of fusion technology equipment in a neutron environment. The proposed experiment would use existing buildings, end plug magnets, beams and vacuum equipment while the central cell portion of the device would be replaced with a testing section and neutron shield. Of course, some modifications will also have to be made to address the problems of neutrons leaking out of the central cell into the end plug region.

The anticipated neutron wall loading would be in the range of  $\sim 1 \text{ MW/m}^2$  and it is anticipated that the MFTF-B+T experiment might run on the order of ten hours per month for about three years. This operating schedule precludes obtaining information of a fluence character ( $\sim 0.04 \text{ MW-y/m}^2$  maximum exposure in three years), but such a neutron wall loading should be sufficient to perform flux testing experiments. Typical examples of such experiments are nuclear heat removal from superconducting magnets, operation of cryogenic vacuum systems in a neutron environment, radiation effects on the insulators of neutral beams, performance of RF waveguides or antennas, and the operation of a fusion reactor blanket to name but a few.

The objective of this study is to present a possible test blanket design for liquid Pb-Li alloys or liquid Li that would be relevant to a power reactor and at the same time be consistent with the geometry and radiation environment of MFTF-B+T. In particular, we will present the neutronic, thermal and breeding response of a Li.17Pb.83 and a Li blanket. We will show that such blankets are compatible with the proposed operation schedule of the MFTF-B+T device.

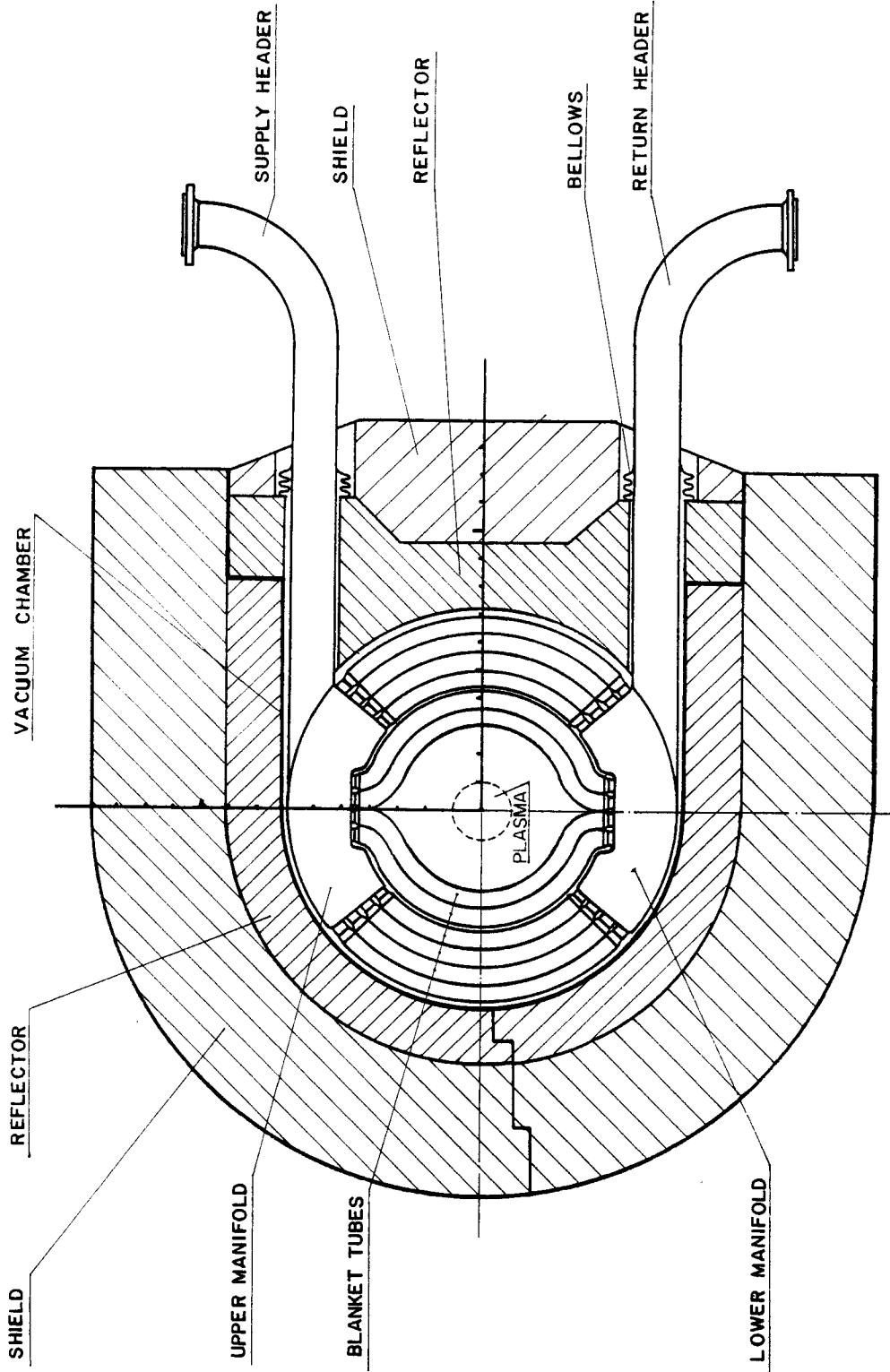
## II. Blanket Description

A proposed central cell blanket module design for MFTF-B+T is shown in Figs. 1 and 2. Fig. 1 is a vertical cross section through the center of a test module. The plasma is surrounded by rows of blanket tubes which are bent semi-circularly and attached to the upper and lower manifolds through short transition sections. The outer diameter of the tubes is 6.5 cm and the wall thickness is 0.15 cm. The transition tubes are 3.25 cm OD and have a wall thickness of 0.11 cm. The axial centerline distance between tubes is 6.7 cm and the radial centerline distance is 5.8 cm. Fig. 2 shows a horizontal cross section through a module which has 1 m axial length. Blanket thicknesses of 38 and 48 cm are used for the Li.17Pb.83 and Li coolants, respectively. The first wall radius is 30 cm.

The blanket module is located in a stainless steel vacuum chamber with rectangular flanged openings which allow the blanket modules to be inserted and removed. The part of the reflector and shield which is attached to the blanket module makes a seal with the vacuum chamber. Figure 2 shows that the blanket is surrounded by a 25 cm thick reflector. Shield thicknesses of 48 and 55 cm are used for the Li-Pb and Li blankets, respectively. There are three reflector and shield segments as shown in Fig. 1.

## III. Neutronics

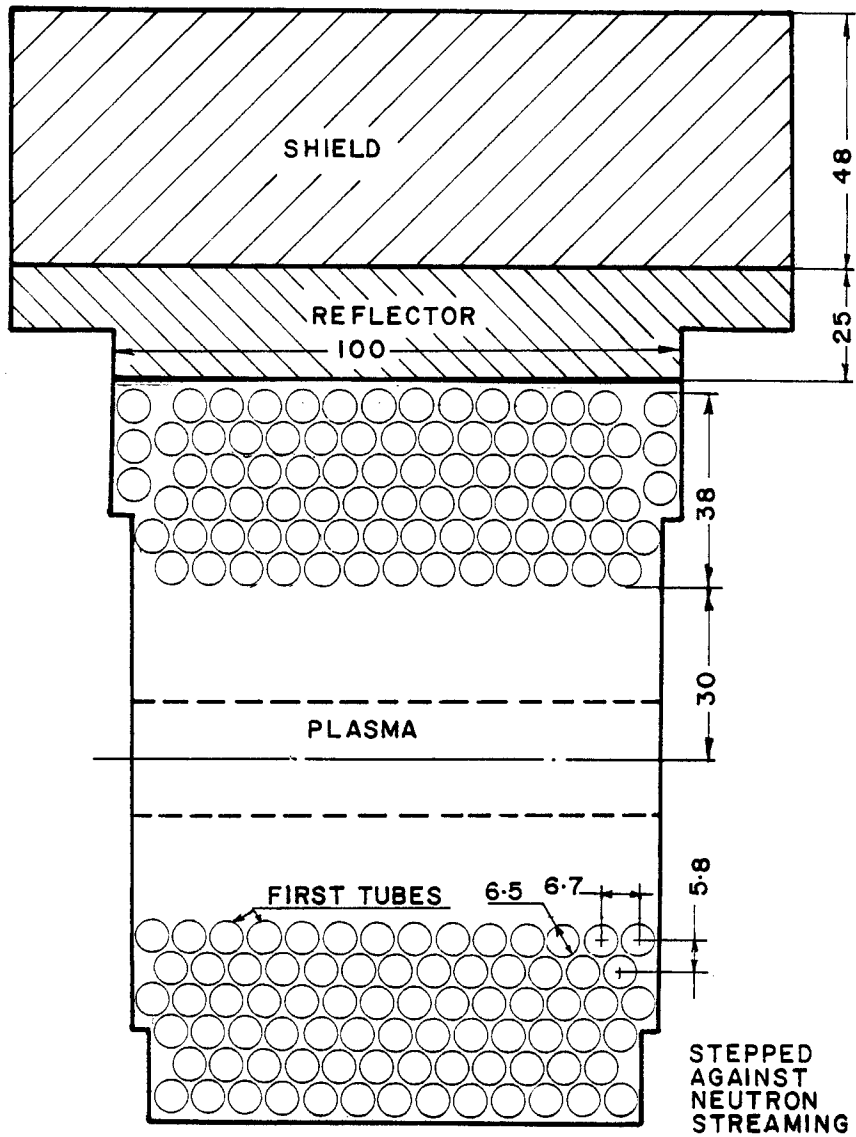
One-dimensional neutronics calculations have been performed for two blanket designs. One of the designs uses Li.17Pb.83 as a coolant and breeder enriched to 90%  $^6\text{Li}$ . Highly enriched lithium is used in this case to provide large tritium breeding with a small blanket thickness. The other design uses natural liquid lithium for cooling and breeding. In both designs the blanket consists of 73 v/o coolant, 7 v/o 316 stainless steel and 20 v/o void and is followed by a 25 cm thick 316 SS water cooled reflector. To achieve



20 cm

# CENTRAL CELL STRUCTURE

Figure 1



DIMENSIONS IN cm

**HORIZONTAL CROSS SECTION OF THE Li-Pb BLANKET**

Figure 2



a local tritium breeding ratio of  $\sim 1.1$ , blanket thicknesses of 38 and 48 cm are required for the Li-Pb and Li blankets, respectively.

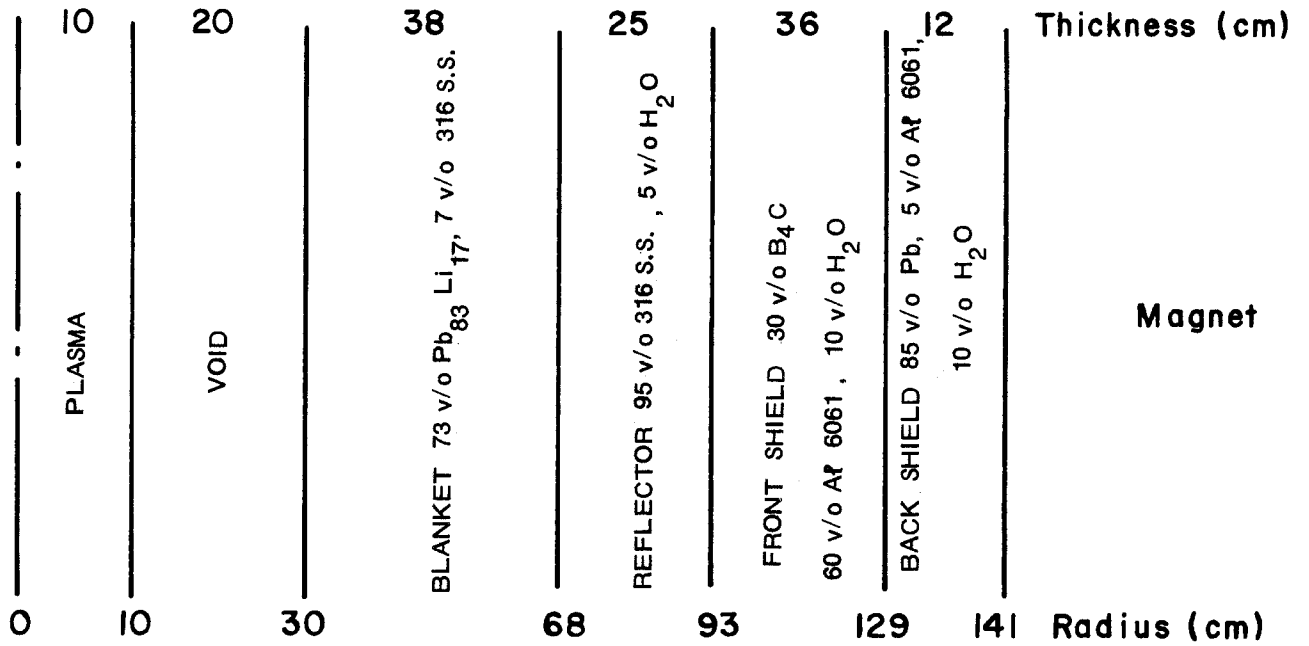
Adequate shielding is required to protect the superconducting magnets. The very small availability of MFTF-B+T implies that high dpa rates in the stabilizer and radiation dose rates in the insulator can be tolerated. Therefore, the shielding requirements are determined by the limit on the peak cryogenic heat load in the magnet. This limit is taken to be  $0.3 \text{ mW/cm}^3$ . The shield is considered to include two zones; a front zone consisting of 60 v/o Al 6061, 30 v/o  $\text{B}_4\text{C}$  and 10 v/o  $\text{H}_2\text{O}$  and a back zone consisting of 85 v/o Pb, 5 v/o Al 6061 and 10 v/o  $\text{H}_2\text{O}$ . The thickness of the back zone is fixed at 12 cm and is required primarily to attenuate the gamma rays resulting from radioactive decay of activated reactor materials. The required thicknesses of the front shield zone are 36 and 43 cm, for the Li-Pb and Li blankets, respectively.

Fig. 3 gives schematics of the blanket and shield for the two blanket designs. The neutronics calculations performed for these designs yield the spatial distribution of power density and tritium production which are used in the subsequent calculations. Some relevant neutronics results are listed in Table 1. The results are normalized to a wall loading of  $1 \text{ MW/m}^2$  at a first wall radius of 30 cm.

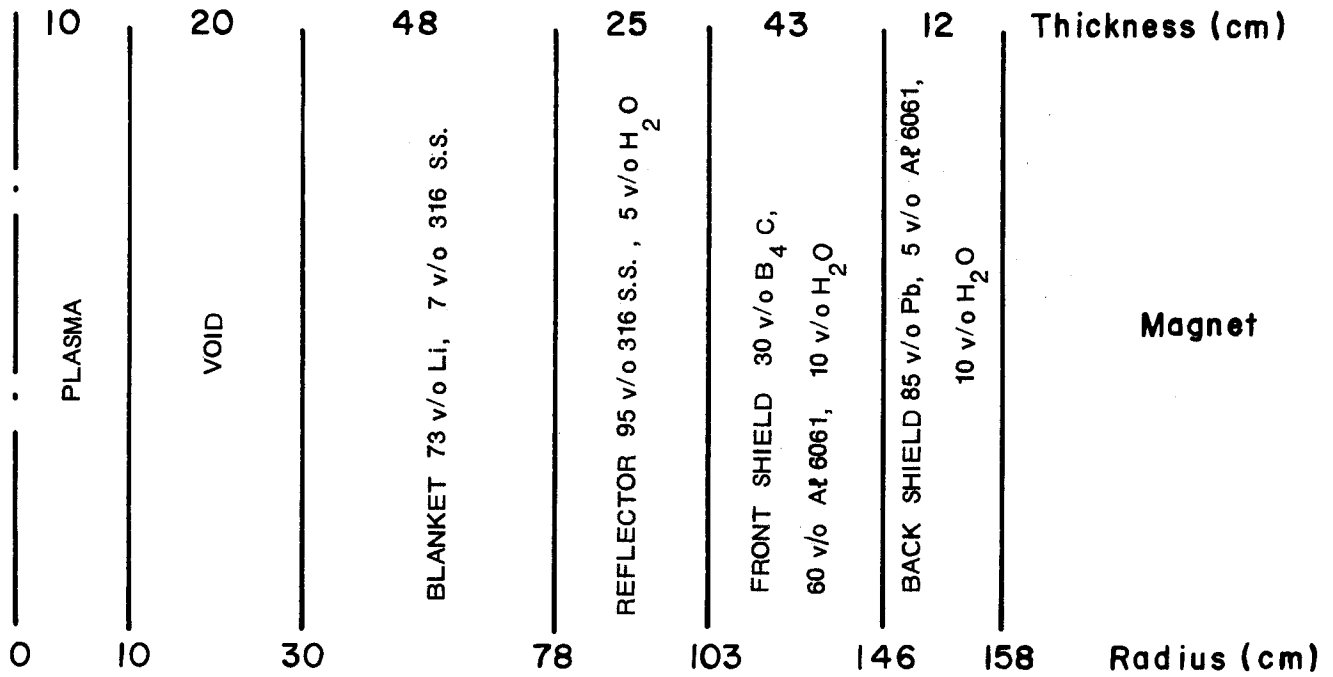
#### IV. Thermal Characteristics

The temperature response of the blanket and power cycle are calculated based on the following assumptions:

1. The blanket and the coolant loop are preheated to  $330^\circ\text{C}$  before the plasma startup.
2. The plasma is ramped to full power at  $t = 0$ .
3. The first wall receives a surface heating of  $5 \text{ W/cm}^2$  from the plasma.



a)  $Pb_{83}Li_{17}$  BLANKET



b) Li BLANKET

Figure 3 Schematic of blanket and shield configuration

Table 1  
Summary of Neutronics Results for the Proposed  
MFTF-B+T Liquid Metal Test Blanket

	<u>Pb.17Pb.83</u>	<u>Li</u>
Tritium breeding ratio	1.12	1.09
Peak heat load in magnet (mW/cm <sup>3</sup> )	0.30	0.28
Power (MW/m)		
Blanket	1.835	1.802
Reflector	0.643	0.711
Shield	0.062	0.052
Total	2.540	2.565

4. The coolant temperature rise is 100°C through the blanket.

The neutronic calculation of heat deposition in the blanket is shown in Fig. 4. The average volumetric heating in the first tube is 5 W/cm<sup>3</sup> for the Li.17Pb.83 blanket. The total energy deposited in each first row tube is 12 kW which requires a coolant velocity of 4.2 cm/sec. The temperature response of the first wall at various locations is shown in Fig. 5. Note that the time to reach steady state on the first tube is 23 seconds. The whole blanket will reach steady state in 146 seconds.

Fig. 6 gives the fraction of steady state power that reaches the heat exchanger as a function of time. It takes the first six seconds for the coolant exiting from the blanket to reach the heat exchanger. It takes another six seconds to heat up the coolant pipes from 330°C to 430°C and the time for the entire system to reach steady state is ~ 180 seconds.

A summary of the test blanket thermal characteristics is given in Table 2. It is clear that the time to reach thermal equilibrium for the Li.17Pb.83 is much smaller than a plasma burn of ten hours ( $3.6 \times 10^4$  sec). There is enough time to establish a steady state temperature profile and also conduct some transient experiments in a single run.

Similar calculations have been made for the lithium blanket. The first tube, the blanket and the loop reach steady state in 60 seconds, 260 seconds and 300 seconds, respectively.

#### V. Tritium Production and Extraction

The tritium concentrations in two liquid metal blanket modules, one containing lithium and the other containing the alloy Li.17Pb.83 as the coolant and breeding material, were investigated. The blanket modules were designed to have similar breeding ratios (~ 1.1) which correspond to a tritium production rate of  $5.0 \times 10^{-6}$  g/s. The mass of liquid metal in the

Fig.4 NUCLEAR HEATING RATES FOR Li AND Li-Pb BLANKETS

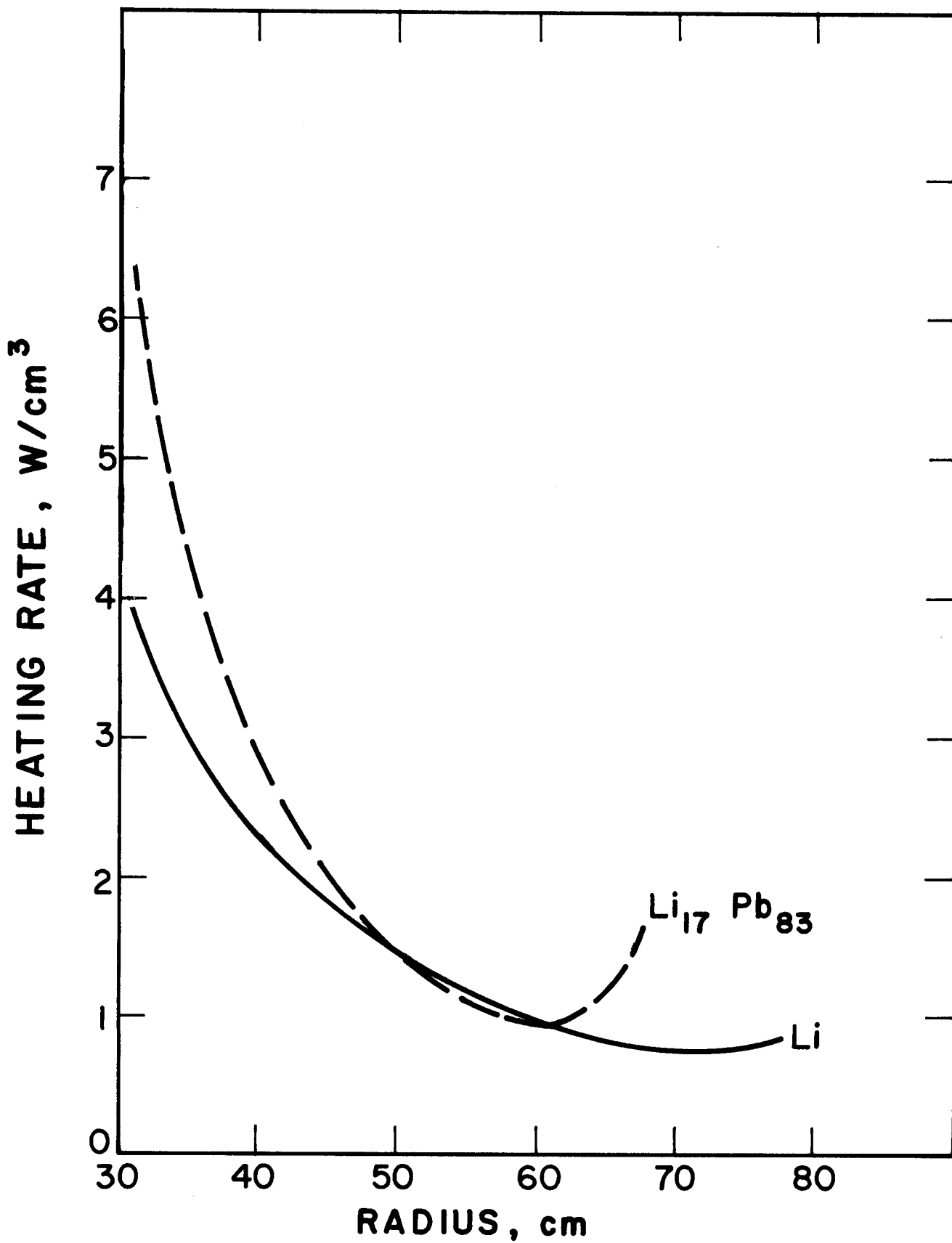
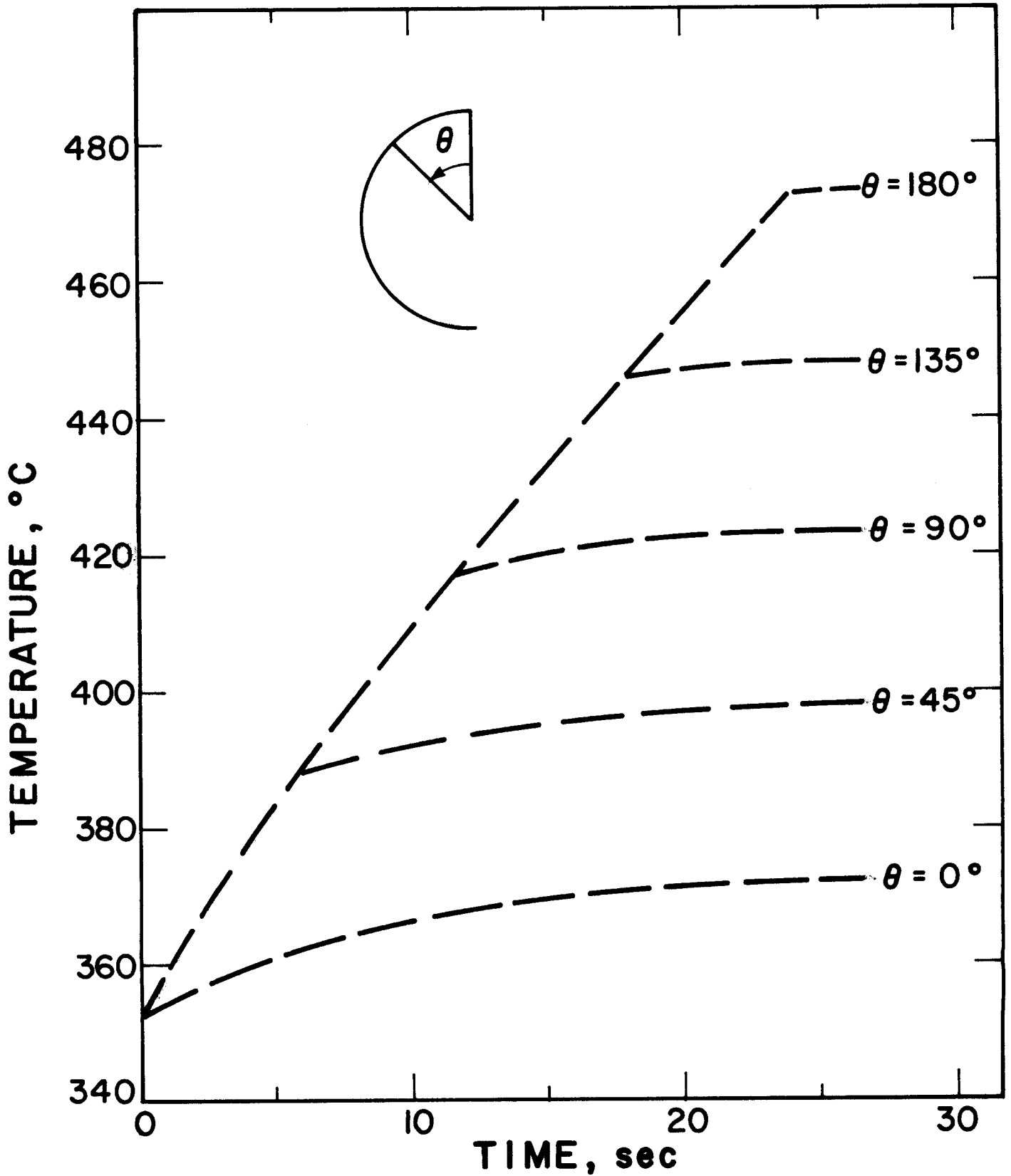


FIG.5 FIRST WALL TEMPERATURE RESPONSE AFTER PLASMA STARTUP



**FIG.6 FRACTION OF STEADY STATE POWER REACHING THE  
HX AS A FUNCTION OF TIME AFTER PLASMA STARTUP**

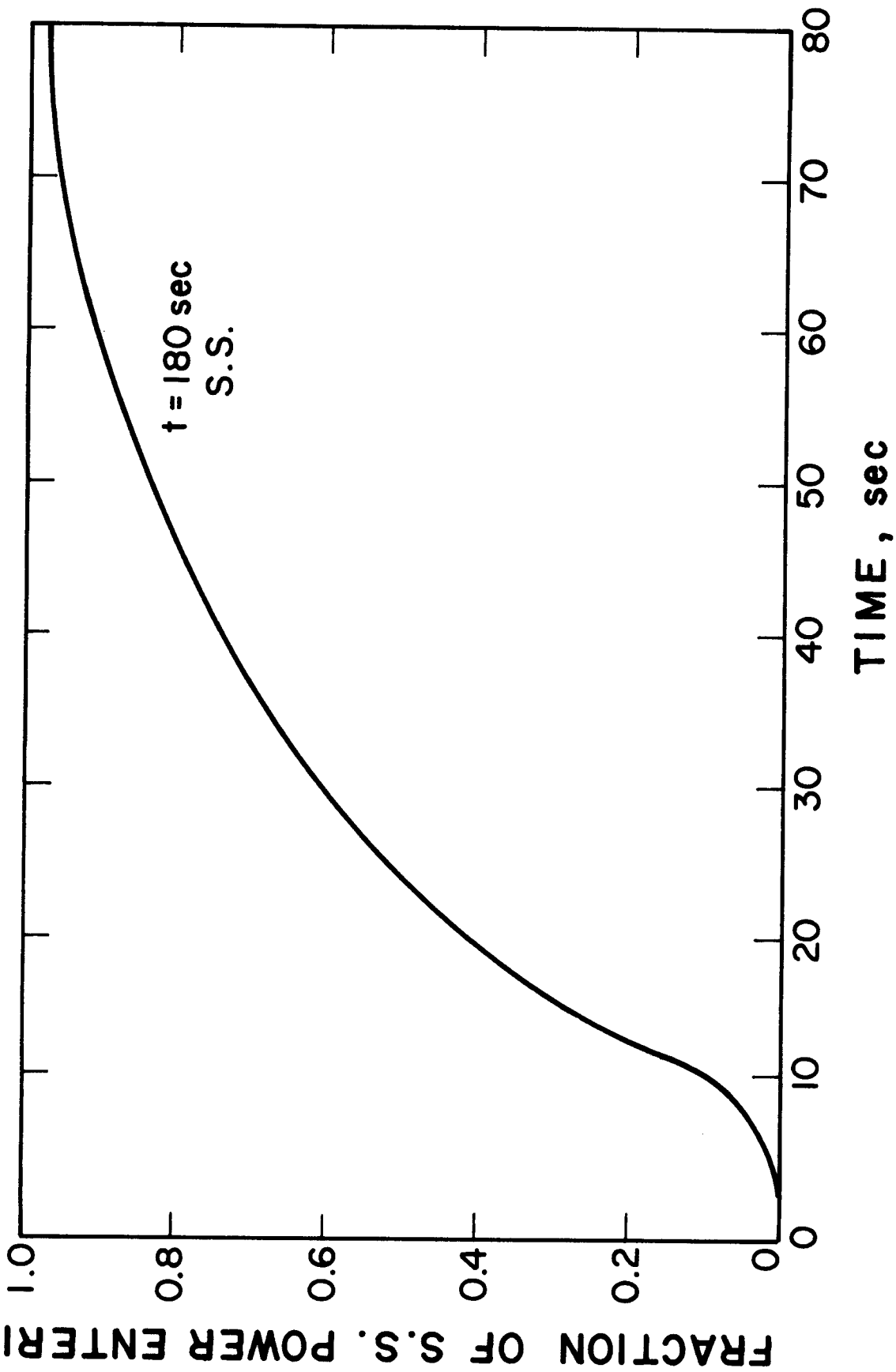


Table 2  
Summary of MFTF-B+T Li.17Pb.83  
Testing Blanket Module Characteristics

First Tube in First Row

Surface heat	5 W/cm <sup>2</sup>
Nuclear heating of first wall	7.3 W/cm <sup>3</sup>
First wall thickness	.15 cm
Average nuclear heating	5 W/cm <sup>3</sup>
Tube length	94 cm
Energy per tube	12 kW
Coolant velocity	4.2 cm/sec
Coolant residence time	23 sec

Minimum Heating Tube (Back of Test Module)

Nuclear heating	1 W/cm <sup>3</sup>
Radius	60 cm
Tube length	188 cm
Coolant velocity	1.28 cm/sec
Residence time	146 sec

Overall Parameters in Test Blanket

Blanket power	1.84 MW
Coolant flow rate	1.25x10 <sup>4</sup> cm <sup>3</sup> /sec
Blanket volume	8.5x10 <sup>5</sup> cm <sup>3</sup>
HX and tube volume	3.5 x 10 <sup>5</sup> cm <sup>3</sup>
Average coolant residence time in blanket	68 sec
Average coolant residence time in loop	95 sec
Maximum coolant residence time in loop	173 sec



total coolant loop was estimated as  $1.1 \times 10^7$  g in the lithium-lead loop and  $8.1 \times 10^5$  g in the lithium loop. The lithium lead contains  $7.6 \times 10^4$  g of lithium enriched to 90% lithium-6. The tritium concentration as a function of time of reactor operation is plotted in Fig. 7.

The tritium concentration increases linearly with time until the concentration reaches a level where the tritium can be extracted. At this point tritium is removed at the breeding rate and the concentration is maintained at a constant value. In the lithium-lead module extraction by vacuum degassing is assumed to occur when the tritium pressure reaches  $10^{-4}$  torr. This pressure corresponds to a very low tritium concentration ( $5.1 \times 10^{-4}$  wppm) due to the low solubility of tritium in the alloy. Tritium could conceivably be removed at even lower pressures; however, the  $10^{-4}$  torr pressure is assumed to be in the range of a reactor relevant pressure representing a compromise between the extraction and containment systems.

Tritium removal from liquid lithium would most likely be done by molten salt extraction. A concentration of about 1 wppm is the expected level where tritium removal would take place.

In Fig. 7 it can be seen that the tritium concentration increases more rapidly in the lithium blanket than in the lithium lead blanket because of the lower mass in the lithium loop. However, since tritium removal occurs at such a low concentration in the lithium lead system, the time it takes to reach the extraction level is only 0.31 hours compared to 45 hours for lithium. Tritium production and extraction can be demonstrated in the lithium lead module during the anticipated 10 hour shot per month reactor operation time. Extraction from lithium will require either a time period of 4.5 shots (4.5 months) or preloading the lithium with tritium. The total amount of tritium in the modules at the extraction point is 0.81 g in the lithium and 0.0056 g

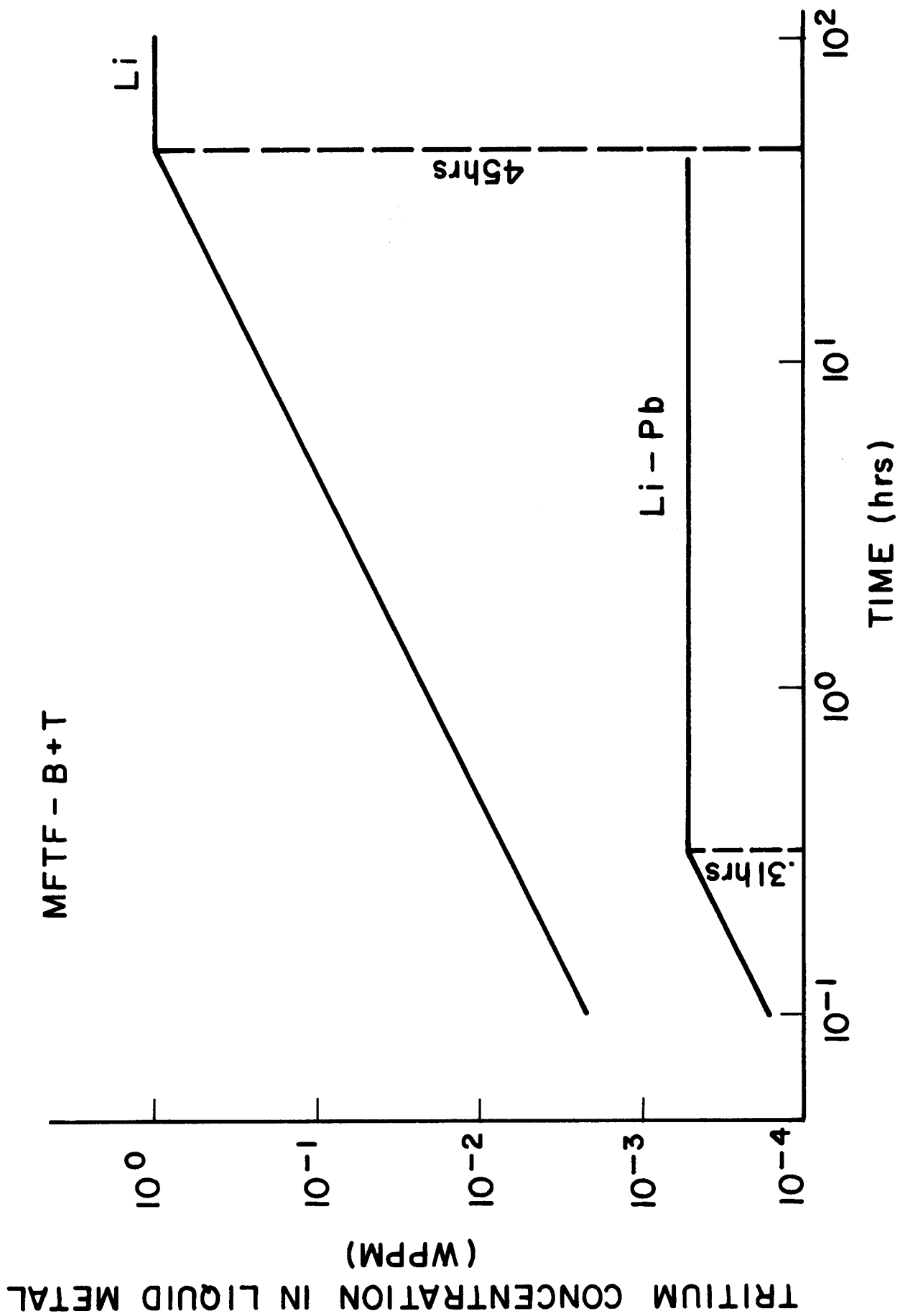


Figure 7 Tritium concentration as a function of time

in the lithium lead system.

It should also be noted that tritium extraction from both liquid metals can be tested and demonstrated outside of the reactor environment in a test loop where tritium is dissolved in the liquid metal. Possible reactor tests that could be performed include a study of the integration of the extraction and subsequent tritium purification into the reactor fuel cycle, the effects of radiation and corrosion products on the handling and extraction procedure, the effects of transporting impurities from the molten salt extractor into the reactor environment and tritium leakage in the steam generator and reactor cell.

## VI. Radioactivity

The radioactivity of the system was investigated in several ways. First, the buildup of activity with time was considered. This was done to show that the activity is approaching the saturation value characteristic of a two-year operating scenario. Fig. 8 shows the buildup with time of the activity for the Li system. It is seen from this figure that the activity has almost saturated after ten hours. The major portion of the final activity is due to  $^{56}\text{Mn}$  (2.6 hr half life). This is also the dominant isotope for longer operating machines (60% of the total activity in WITAMIR). Since the next most important isotope is  $^{55}\text{Fe}$  ( $T_{1/2} = 2.7 \text{ y}$ ), it is impossible to saturate this isotope unless approximately 14 full power years is achieved in any machine.

The other consideration was the dose at the back of the shield after shutdown. Fig. 9 illustrates the results for the two blanket designs. It is noted that the Li.17Pb.83 greatly attenuates the flux in the blanket resulting in a dose much lower than that in the Li blanket case. It can also be seen that the dose decreases with the half life of the  $^{24}\text{Na}$  (15 hours) which is

produced from the aluminum in the shield and that the radiation level reaches a low background level (due to the activated 316 SS in the blanket and reflector) after a week. Hands-on maintenance is possible which produces an integrated dose of no more than 100 mrem/40 hr work week if the maintenance is started 3.9 days after shutdown for the Li-Pb blanket and 4.6 days with the Li blanket.

Since the 15 hour Na-24 half life activity produces initially high radiation levels, a shield in which Al 6061 was replaced by 316 SS was investigated. This shield allowed hands-on maintenance after 2 days. However, the background activity level increases such that after several 10 hour runs, the regions of the back of the shield will have a dose rate that exceeds hands-on maintenance criteria. This would be long term activity and would require a period of several years to die down. Therefore, an Al structure in the final shield is recommended.

## VII. Conclusions

- The preliminary designs of two liquid metal blankets reveal that they can reach thermal, tritium and radioactivity conditions required for power reactor applications in the proposed operating schedule of MFTF-B+T. The Li-Pb system can reach all of these conditions in one 10 hr run while the Li system requires ~ 5, 10 hr runs to reach tritium equilibrium.
- Approximately 1.8 MW of thermal power will be generated in a 1 m wide liquid Li-Pb or Li test blanket for the present MFTF-B+T design.
- The time for the entire liquid metal systems to reach thermal equilibrium is ~ 3 minutes for the Li-Pb system and ~ 5 minutes for the Li system.
- If both breeding materials are free of  $T_2$  at the start of operation, it takes ~ 0.3 hr for the Li-Pb system and ~ 45 hr for the Li system to

reach concentrations typical of reactor relevant extraction systems.

- The buildup of radioactivity in the blanket, reflector and shield is ~ 90% of equilibrium within ~ 5 hours of operation.
- It was found that "hands-on" maintenance could be performed at the back of the shield following a decay period of 4 and 5 days for the Li-Pb and Li blanket designs, respectively.

#### Acknowledgement

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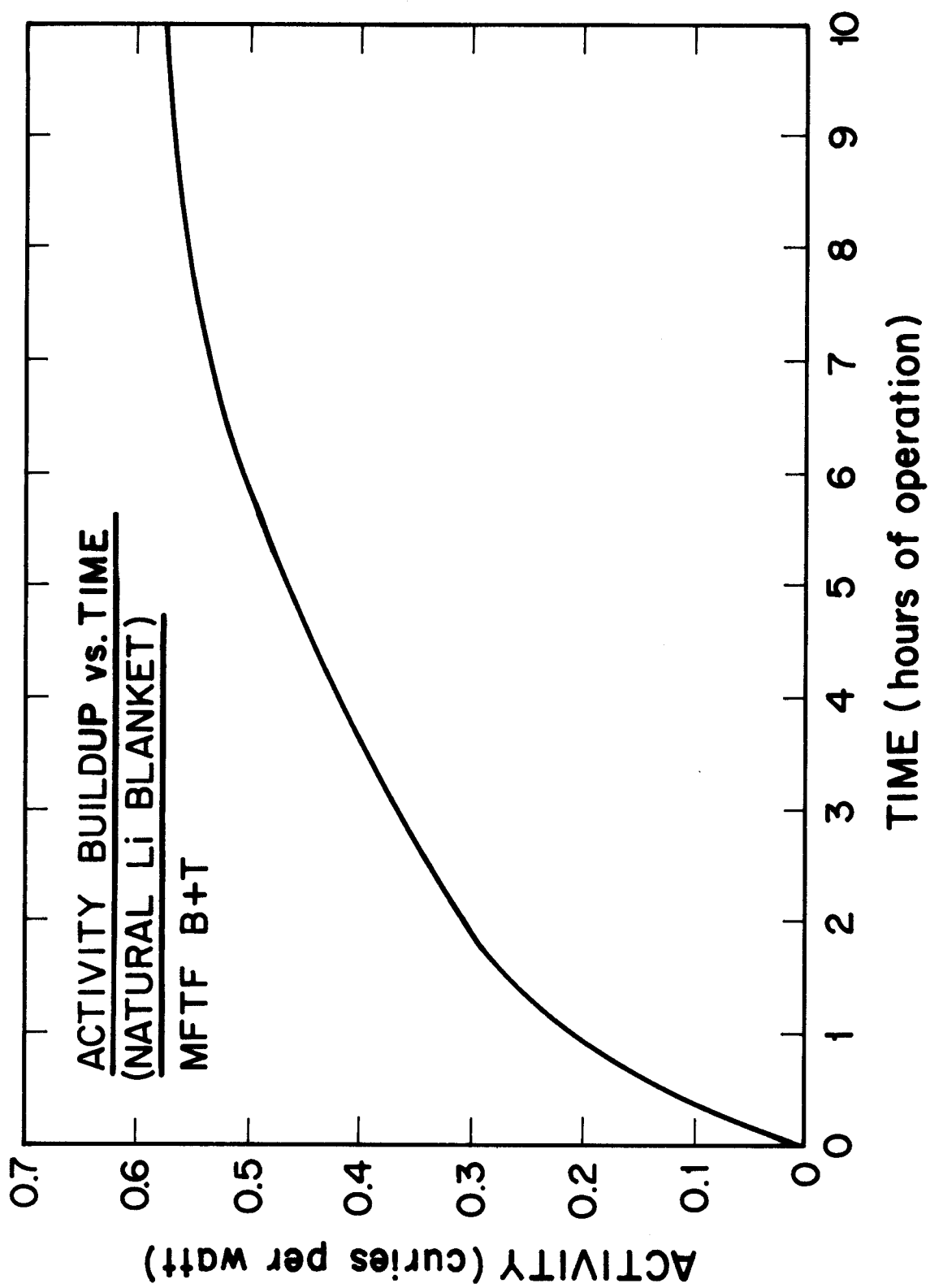


Figure 8 Activity buildup for the Li blanket design

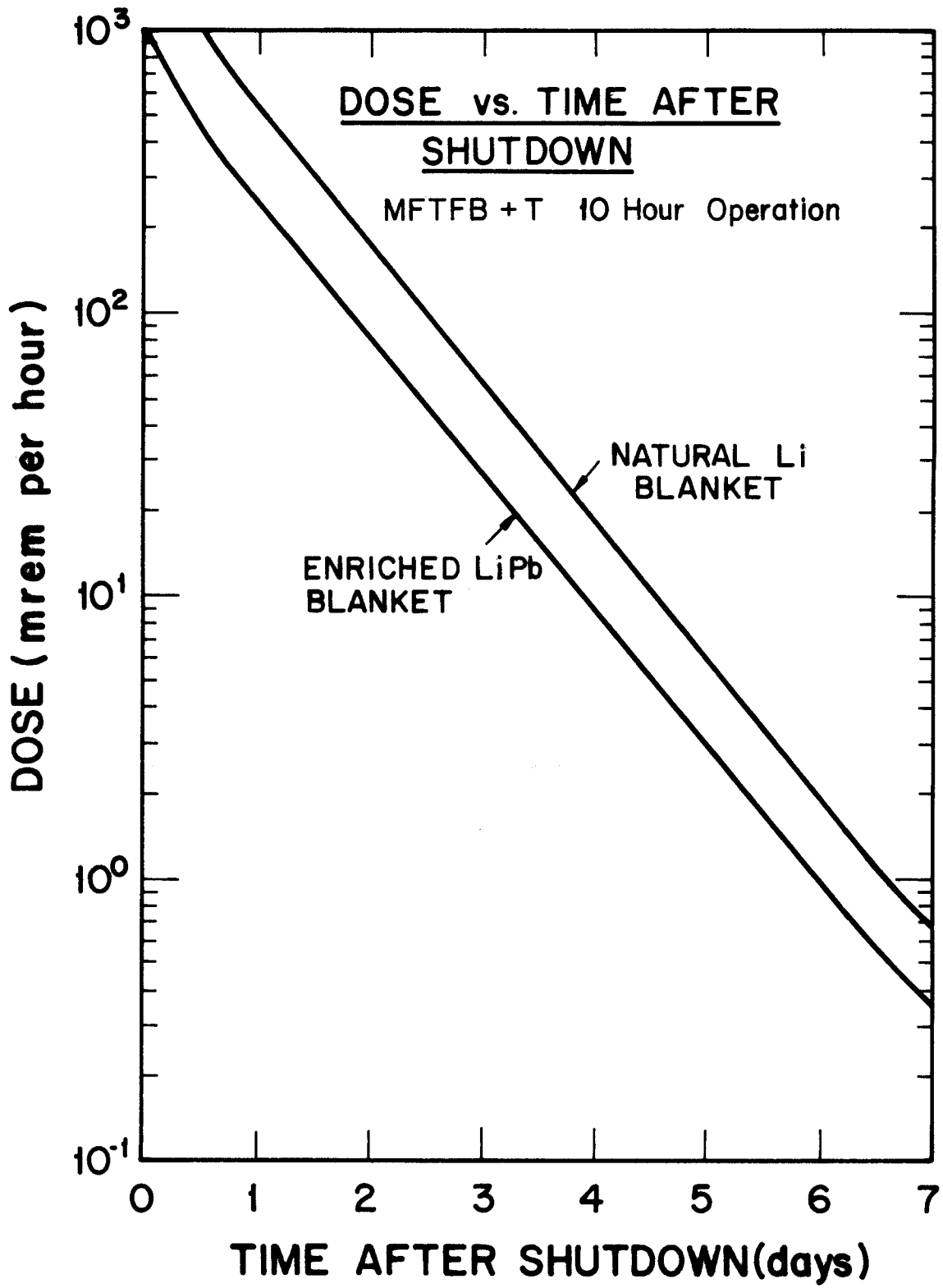


Figure 9 Dose at the back of the shield as a function of time after shutdown

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1. K. Thomassen and V. Karpenko, "An Axicell Design for the End Plugs of MFTF-B", UCID-19318, Lawrence Livermore National Laboratory, April 23, 1982.
2. K. Thomassen, to be published.