



## **TASKA - A Tandem Mirror Fusion Engineering Facility**

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TASKA  
Tandem Spiegelmaschine Karlsruhe  
A Tandem Mirror Fusion Engineering Facility  
A Preliminary Report  
(UWFDM/FPA Version)

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## PREFACE

TASKA (Tandem Spiegelmaschine Karlsruhe) is a near term engineering test facility based on the tandem mirror concept with thermal barriers. The study was essentially completed in December 1981 after approximately one year of study. The project received major financial support from the Kernforschungszentrum Karlsruhe (KfK) in the Federal Republic of Germany, with significant additional contributions from the University of Wisconsin, the Wisconsin Electric Utilities Research Foundation and Fusion Power Associates.

The main objectives of this study were to develop a pre-conceptual design of a facility that could provide engineering design information for a Demonstration Power Reactor. Another objective of this study was to design a facility that would be within financial reach of a single country's fusion budget. Trying to achieve both of these objectives was indeed a challenge but we feel that TASKA does come close to meeting both objectives. It must also be noted that this report was completed in only one year and normally, a second or even third year would be required to pass from the pre-conceptual to conceptual design phase. Therefore, the level of detail in this report is not consistent with that in the INTOR or US-ETF/FED activities which have been in progress for several years. However, we do think that the basic features of TASKA are so attractive as to warrant early exposure and early publication.

The present report is of a preliminary nature, designed to elicit comments from the fusion community before the final report is issued in mid-1982. While it is published by Fusion Power Associates, it contains both the U.S. and the FRG contributions to this study as of February 1982. As always in a study of this kind, the first draft copy will contain some inconsistencies and

there are a few calculations that were not completed by the publication of this copy. These will be included in the final version published by KfK in mid-1982.

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## I Introduction and Objectives of TASKA

There has been considerable progress in the plasma physics field during the last decade. It is expected that at least one of the major experiments to come into operation in the 1980's (e.g., TFTR, JET, JT-60, MFTF-B) will approach or even surpass the Lawson criterion. Some of these experiments may even reach ignition. If such positive expectations are realized, we will then be at a major turning point in fusion research where the main interest will switch from plasma physics to plasma engineering and fusion technology for a DT demonstration reactor.

There are a large number of scientists in the fusion community who feel that only one large device needs to be built before a DT demonstration reactor (DEMO) operates. This single device should be the maximum reasonable step beyond the experiments of this decade and considerable effort has been invested in defining the nature of that step.

The most advanced confinement concept at the present time is that of the tokamak and it is expected that tokamaks will demonstrate ignition first. Nevertheless, doubts continue to be expressed as to whether the tokamak line will lead to the most attractive commercial power reactor. Due to the complex and toroidally interlinked structure of the magnet system and its pulsed operation, intrinsic engineering difficulties are anticipated; e.g., restricted accessibility, material fatigue problems, pulsed power cycle. In principle, a linear device and steady state operation can avoid such problems. It will lead to a reactor with improved accessibility, thus easing exchange or repair of key components. Without the dynamic loads from pulsing, the mechanical behavior of the materials will be improved.

The present optimism for the successful design and operation of a major DT fusion experiment has initiated a considerable effort on studies of an engineering test facility with all the features mentioned above. A worldwide endeavor for such a conceptual design is the study of the International Tokamak Reactor, INTOR. Recent advances in the physics of tandem mirrors have caused us to examine the potential of a tandem mirror device with thermal barriers serving as such an engineering test facility. Therefore we limited our study, TASKA (Tandem Spiegelmaschine Karlsruhe), to be the maximum reasonable step beyond the next generation of large mirror machines (AMBAL, TMX-Upgrade, GAMMA-10, TARA, and MFTF-B). The objective of the study is to demonstrate that all key technologies required for this step (and later on for DEMO) can be integrated into one machine and adequately tested. Thus TASKA has to serve as test bed (e.g., it should have a reactor-relevant neutron flux, at least  $1 \text{ MW/m}^2$ ) for technologies of plasma engineering, superconducting magnets, existing materials, plasma heating (neutral beam and RF-heating), breeding and test blankets, tritium technology and remote handling. It has to prove that such a facility can operate safely and reliably as a DT burner. We have also limited the direct capital cost to 800 million dollars or less, and an operating cost of less than 80 million dollars per year.

The physics basis for TASKA has partly been verified on GAMMA-6 and TMX, and is partly expected to be proven by TMX-Upgrade during 1982. In addition MFTF-B, currently under construction, should demonstrate by 1986 long pulse operation ( $\approx 30$  seconds) in the TMR-thermal barrier mode.

The main objectives of the overall TASKA study are twofold: (1) the study of a feasible tandem mirror engineering test facility, and (2) the comparison of the technologies required for a tokamak based solution with a



solution based on the tandem mirror with thermal barriers. This report mainly addresses the first point and subsequent reports will concentrate more on the mirror/tokamak comparisons. It is felt that by such a "friendly" competition between the two major magnetic fusion concepts, we may arrive at the optimum facility to be built by the world fusion community.

## II. Overview of TASKA

### II.1 Introduction

The rapid progress in plasma physics during the past 10 years and the expectations of positive results from the major experiments to come into operation this decade (e.g., TFTR, JET, T-15, JT-60, MFTF-B) suggest that the next large device to be built before the Demonstration Reactor (DEMO) operates will be a DT Technology Test Facility. This point of view has initiated many studies directed towards the next step in the world fusion program which is envisaged to be an engineering test facility where all of the key technologies required for a Demonstration Reactor can be successfully integrated into, and tested in, one machine. One such worldwide endeavor for the conceptual design of a tokamak is the study of the International Tokamak Reactor, INTOR.<sup>(1)</sup> Indeed, tokamaks presently represent the most advanced fusion concept and are expected to demonstrate ignition and the behavior of an ignited DT-plasma in the late 1980's. Nevertheless there are some doubts that tokamaks may in fact lead to a commercial power reactor mainly because of their complex and toroidally interlinked magnet structure.<sup>(2)</sup>

In principle, a linear device could avoid such complexity. It can lead to a device which gives better access and the linear geometry allows for modular construction of the major components. This configuration is also a definite advantage for the exchange or repair of key components within the reactor vessel. The physics concept with the best chance for a linear geometry is the mirror and recent advances in the physics of mirrors give rise to the expectation that the difficulties in the plasma physics performance of minimum B mirror machines can be overcome by the tandem mirror concept with thermal barriers.<sup>(3)</sup> Conceptual designs of power plants have already shown

such configurations to be economically competitive with tokamaks.<sup>(4)</sup> In addition to the favorable possibilities of the tandem mirror, it was also evident that such a device might be a very attractive engineering test facility. Thus, the TASKA study was initiated in late 1980 to see if our optimistic expectations were justified from an engineering point of view.

## II.2 TASKA Objectives

The main objective of TASKA is to provide an engineering test bed to qualify and test materials and blanket concepts for a Demonstration Fusion Power Reactor. TASKA must also demonstrate that superconducting magnets, heating technologies, tritium handling equipment, remote handling equipment, etc., can all operate in an intense neutron environment with reliabilities that will allow ~ 50% availabilities to be achieved. It must do all of this in a timely, but yet cost effective manner which is consistent with the safe operation of a nuclear facility. To be more specific, TASKA must be the maximum reasonable physics and technology step beyond the next generation of large mirror machines (AMBAL,<sup>(5)</sup> TMX-Upgrade,<sup>(6)</sup> GAMMA-10,<sup>(7)</sup> TARA,<sup>(8)</sup> and MFTF-B<sup>(9)</sup>). It should be able to operate in the early 1990's (similar to INTOR<sup>(10)</sup>) and provide at least 5 MW-y/m<sup>2</sup> of large scale test volume in no more than 10 years of full power operation (exclusive of check-out and initial low power tests). Finally, the overall direct cost of TASKA should be  $\lesssim$  800 million dollars and the yearly operating cost (including personnel, power, T<sub>2</sub>, etc.) should be less than 50 million dollars.

### II.3 Physics Basics for TASKA

The tandem mirror physics concept has been previously verified on GAMMA-6<sup>(11)</sup> and TMX.<sup>(12)</sup> Some aspects of electron heating with ECRH in the end plugs of TMR's have been recently demonstrated on Phaedrus<sup>(13)</sup> and ICRH heating of plasmas has been demonstrated in PLT.<sup>(14)</sup> The key experiment to demonstrate operation with thermal barriers on a tandem mirror will be performed in 1982 on TMX-U.<sup>(6)</sup> This device will also advance the understanding of electron heating and the central cell region will be operated in the collisionless diffusion regime.

The MFTF-B<sup>(9)</sup> and its axicell upgrade<sup>(15)</sup>, currently under construction, should demonstrate long pulse operation (30 seconds) in the TMR-thermal barrier mode in the 1985-1986 time period. The limits on central cell beta will also be investigated in that device. High power, continuous (30 sec) 80 keV neutral beams will be used to achieve a D-T equivalent  $Q \sim 1$  in the same time period. MFTF-B will be the first tandem mirror machine to use all superconducting magnets, including high field barrier and yin-yang coils.

The physics relationship of TASKA to past, present, and future devices is shown in Fig. II.3-1. The anticipated performance of WITAMIR-I<sup>(4)</sup> is also included and the collection of parameters shows that TASKA lies on a reasonable projection of the aforementioned devices. More discussion on the physics basis of TASKA is given below and in Ref. 16. It is important to note that while the optimism about achieving the TASKA physics goals is high, the success of any device after MFTF-B depends on the degree to which current physics models can be verified. Progress through the TARA/AMBAL/TMX-U/GAMMA-10/MFTF-B series must therefore be closely monitored.

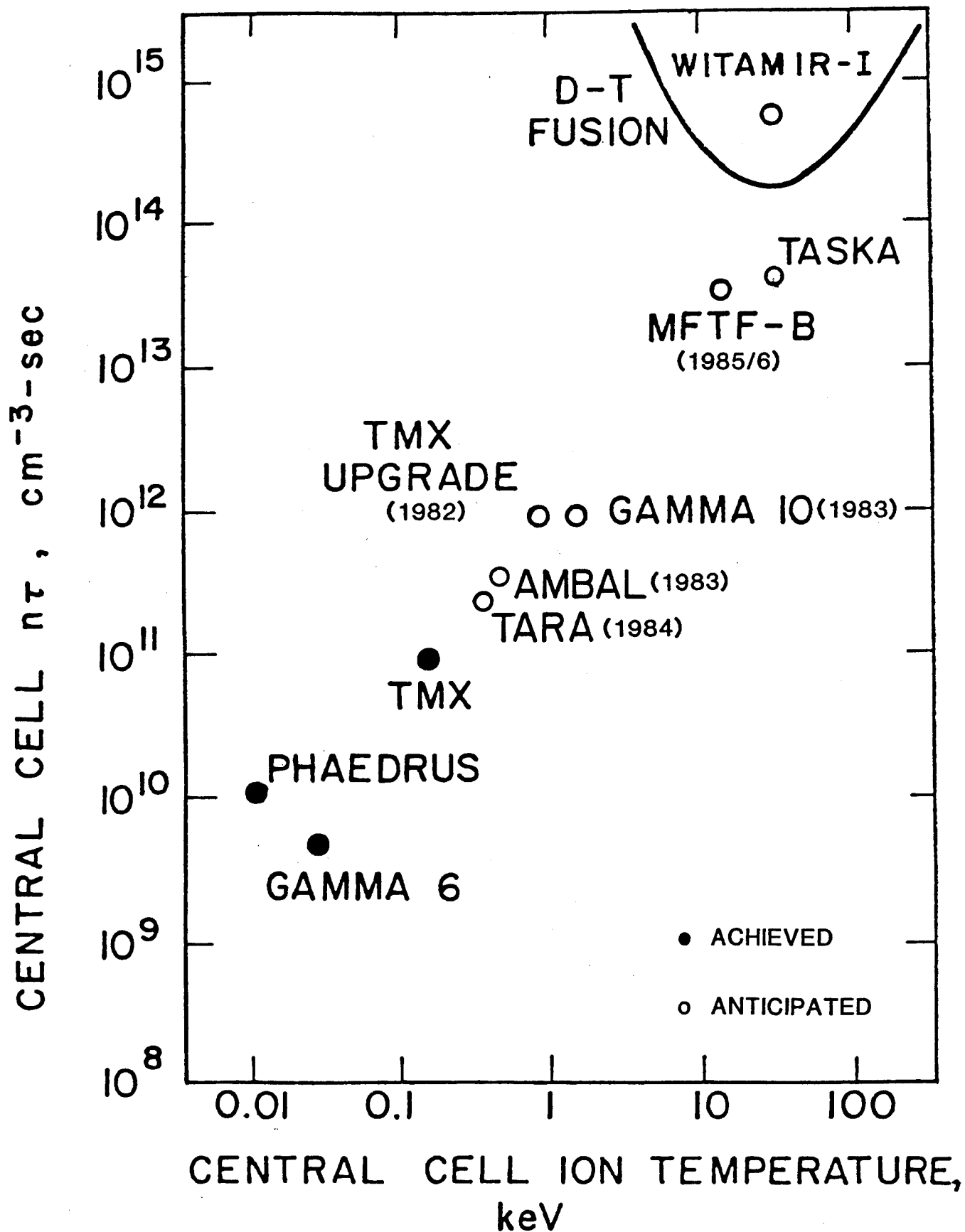


Fig. II.3-1 Critical physics parameters for present and future tandem mirrors.

## II.4 General Survey and Basic Reactor Parameters of TASKA

### II.4.1 Design Philosophy

TASKA is based on the concept of a tandem mirror with an inside thermal barrier. The design uses existing technologies or those which may be expected to be available in the near future. A net power gain is not required, therefore  $Q$  is not an important parameter and will be slightly less than unity. However, a main consideration is that the neutron wall loading should be high enough to achieve a significant neutron flux and fluence for materials and blanket tests. The central cell length is minimized because of the modest requirements of the test blankets, thus reducing costs without losing scalability. A tritium breeding ratio of 1.0 or more is foreseen so that there will be no net tritium consumption over the lifetime of the machine. Electric power production, direct energy conversion, or fission fuel breeding are options that can be examined during the life of TASKA.

### II.4.2 General Survey and Parameters

Schematic views of TASKA are shown in Figs. II.4-1 and II.4-2. The positions of the magnets, the test blankets, the neutral beam injectors and the generators for ECRH and ICRH are indicated along with the power and energy levels for NBI power and frequency for the RF heating systems. A selective list of TASKA general operating parameters is given in Table II.4-1. In Fig. II.4-3 the confining magnetic fields and electric potentials are shown. The end plugs consist of an inside thermal barrier, and a minimum-B yin-yang coil set. There are 3 central cell solenoids.

Some key features of TASKA are listed below:

- The DT power level is 86 MW, far below the 620 MW of INTOR<sup>(1)</sup> or the 180 MW of FED.<sup>(10)</sup> This lower power level greatly eases the tritium requirements

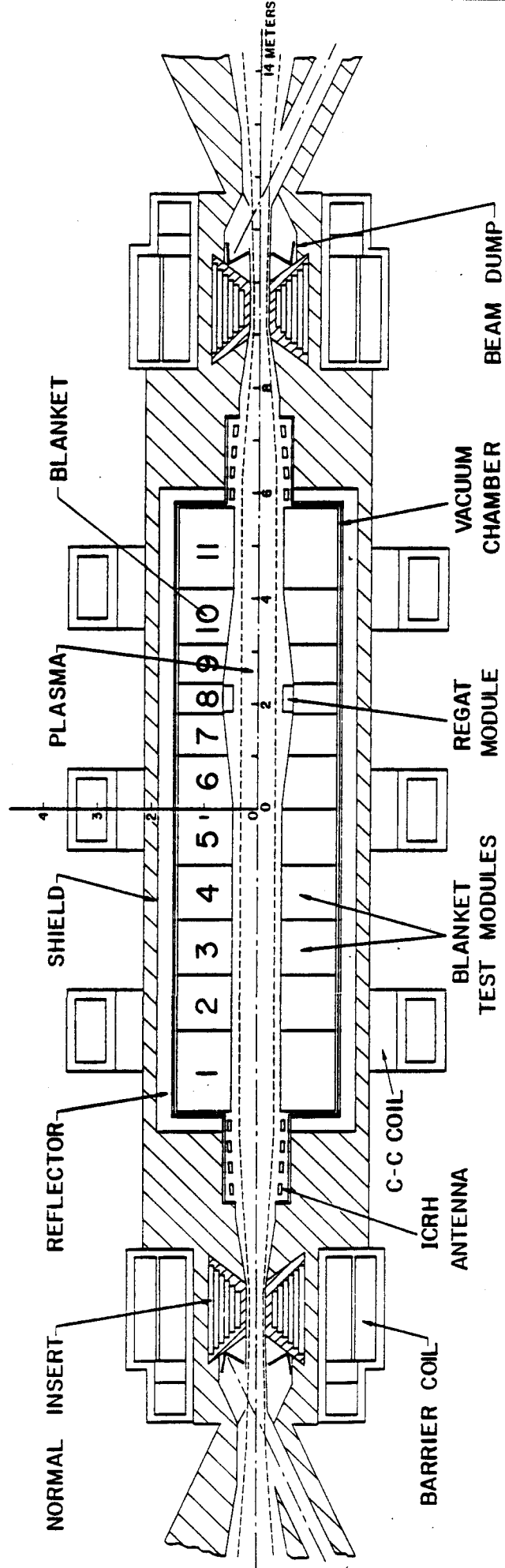
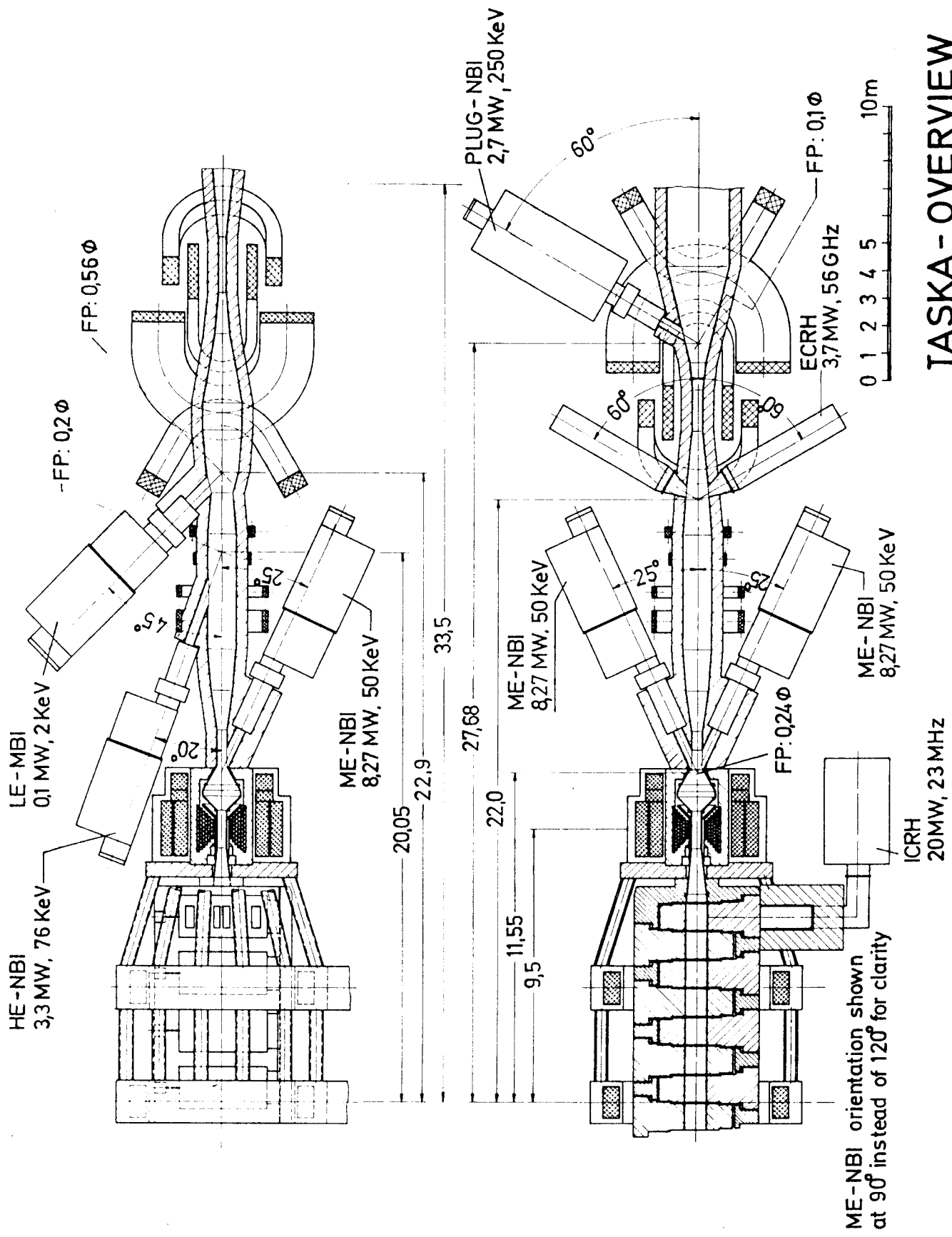


Fig. II.4-1 TASKA central cell layout.





II.4-3

## TASKA - OVERVIEW

Fig. II.4-2

Table II.4-1. General Operating Parameters - TASKA

|                                       |                                                |
|---------------------------------------|------------------------------------------------|
| DT Power Level                        | 86 MW                                          |
| First Wall Neutron Wall Loading       | 1.5 MW/m <sup>2</sup>                          |
| Total Heating Power                   | 110 MW                                         |
| Central Cell Magnetic Field (On-Axis) | 2.7 T                                          |
| Central Cell Length                   | 21 m                                           |
| Structure/Breeder                     | HT-9/Pb <sub>83</sub> Li <sub>17</sub>         |
| Number/Total Volume of Test Modules   | 1-Material/493 liters<br>2-Blanket/5700 liters |
| Operating Lifetime                    | 15 years                                       |
| Operating Scenario                    | <u>Year</u> <u>% Availability</u>              |
| H <sub>2</sub> Check Out              | 1      10                                      |
| DT Check Out                          | 2      15                                      |
| Short Term Test Phase                 | 4      25                                      |
| Long Term Test Phase                  | 8      50                                      |

# MAGNET and ELECTROSTATIC CONFIGURATION of TASKA

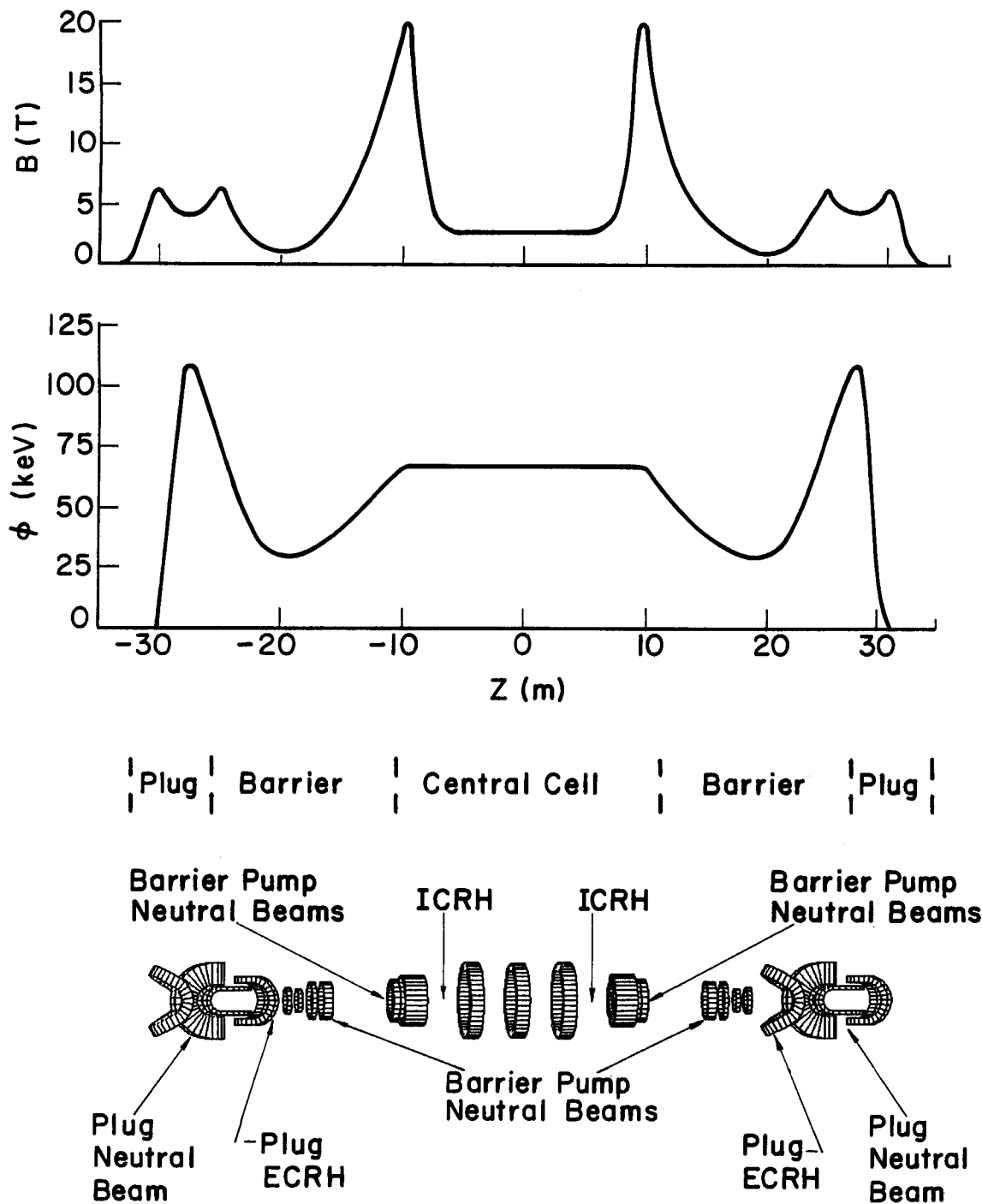


Fig. II.4-3 Magnet and electrostatic configuration of TASKA.

and reduces the overall costs compared to previous test devices.

- The neutron wall loading is  $1.5 \text{ MW/m}^2$ . This relatively high wall loading will allow reactor relevant testing to be performed in both blanket and materials modules. Coupled with reactor availabilities of 25 to 50%, this flux will also allow reactor relevant fluences to be obtained.
- The barrier region is pumped with neutral deuterium and tritium beams and the electrons in the end plug region are heated with ECRH at 56 GHz. The plug ions are sustained by  $\text{H}^0$  beams of 250 keV. The use of the very high barrier magnetic fields and ECRH requires selective ion pumping and the proposed design uses two stage charge exchange pumping<sup>(17)</sup> to reduce these requirements.
- One of the key features of the machine that allows such a favorable performance is the use of a high field, room temperature copper insert which raises the field in the barrier coil from 14 T produced by the superconducting coil to 20 T. The life limiting feature of this coil is radiation damage to the ceramic insulation conservatively assumed to be  $10^{12}$  rad. The projected life is then 10 years at 25% availability or 5 years at 50% availability.
- The tritium for this device is provided by circulating a  $\text{Pb}_{83}\text{Li}_{17}$  alloy in HT-9 ferritic steel tubes (see Fig. II.8-1). The localized breeding ratio is 1.65 and the overall breeding ratio of the machine (including test sections as well as leakage) is 1.0 so that no net tritium consumption is incurred over the life of the machine. The low solubility of  $\text{T}_2$  in  $\text{Pb}_{83}\text{Li}_{17}$ <sup>(18)</sup> results in only a 20 g inventory in the blanket.
- There are two modules devoted to blanket testing and one module devoted to materials testing. These test modules are placed between the central cell

coils for ease of access and maintenance.

- The secondary heat transfer loop contains an organic material, HB-40.

Normally, the approximately 60 MW from the central cell breeding blanket are dumped to cooling towers but provisions have been made to generate electricity with 350°C steam. Roughly 15 MW<sub>e</sub> can be generated in this manner.

- Normally, the energy of the plasma ions and electrons leaking from the plasma is deposited in heat dumps at either end of the machine but provisions have been made to test various direct convertor modules on one end. The operation of TASKA does not depend on the successful operation of the direct convertor.

## II.5 Plasma Physics

The end plug configuration used in TASKA consists of an inboard thermal barrier, generated by a high field hybrid solenoid, and the electrostatic plug located in a min-B yin-yang magnetic mirror (see Fig. II.4-3). The potential peak in the plug provides axial confinement of the central cell ions; the good magnetic curvature of the min-B mirror provides MHD stability of the entire system. The thermal barrier provides thermal isolation of the plug and central cell electrons and allows the plug electrons to be heated substantially without heating the central cell electrons. This also allows the positive confining potential of the plug to be achieved at lower density, which thereby reduces the required neutral beam power to sustain the plug. The central cell ion temperature is maintained by ICRF heating and the power input from the barrier pump beams. This reduces the power requirements for "pumping" of the thermal barriers.

The physics parameters of TASKA are given in Table II.5-1 and the heating parameters are given in Table II.5-2. The central cell beta of 50% is an assumed value, but is considered to be conservative in light of recent theoretical developments concerning finite Larmor radius stabilization of ballooning modes; this is discussed further in Section III.1. Microstability of the mirror-confined plug plasma is an unresolved issue with the TASKA configuration. The TASKA design utilizes "sloshing" ions in the plug to create a local electrostatic well and trap warm plasma. This can, in principle, provide microstability against the drift cyclotron loss cone (DCLC) mode and the Alfvén ion cyclotron mode, but stability against the axial loss cone mode (which has not yet been observed experimentally) is harder to obtain.

Table II.5-1. TASKA Plasma Physics Parameters

Central Cell

|                        |                                                |
|------------------------|------------------------------------------------|
| Magnetic field         | 2.7 T                                          |
| Density                | $1.94 \times 10^{14} \text{ cm}^{-3}$          |
| Alpha particle density | $1.6 \times 10^{12} \text{ cm}^{-3}$           |
| Ion temperature        | 30. keV                                        |
| Electron temperature   | 11.5 keV                                       |
| Beta                   | 0.5                                            |
| $(n\tau)_{ic}$         | $5.4 \times 10^{13} \text{ cm}^{-3} \text{ s}$ |
| $(n\tau)_{ec}$         | $5.3 \times 10^{13} \text{ cm}^{-3} \text{ s}$ |
| Potential, $\phi_c$    | 42.8 keV                                       |

Barrier

|                                |          |
|--------------------------------|----------|
| Barrier peak magnetic field    | 20 T     |
| Barrier minimum magnetic field | .8 T     |
| Potential, $\phi_b$            | 37.5 keV |
| Pumping parameter, $g_b$       | 2.       |

Plug

|                              |                                                |
|------------------------------|------------------------------------------------|
| Midplane magnetic field      | 4 T                                            |
| Vacuum mirror ratio          | 1.56                                           |
| Midplane density             | $6.3 \times 10^{13} \text{ cm}^{-3}$           |
| Mean ion energy              | 388. keV                                       |
| Electron temperature         | 59.3 keV                                       |
| Beta                         | 0.64                                           |
| $(n\tau)_{ip}$               | $2.9 \times 10^{13} \text{ cm}^{-3} \text{ s}$ |
| $(n\tau)_{ep}$               | $7.5 \times 10^{11} \text{ cm}^{-3} \text{ s}$ |
| Potential, $\phi_c + \phi_e$ | 109. keV                                       |
| Cohen parameter, $v_c$       | 0.5                                            |

Table II.5-2. TASKA Heating Parameters

Neutral Beams

|          |                   |               |
|----------|-------------------|---------------|
| Plug:    | power             | 5.4 MW        |
|          | energy            | 250. keV      |
|          | angle             | 60°           |
|          | species           | p             |
|          | trapping fraction | 0.21          |
| Barrier: | power             | 6.6 MW        |
|          | energy            | 76 keV        |
|          | angle             | 20°           |
|          | species           | d             |
|          | trapping fraction | 0.42          |
| Barrier: | power             | 49.7 MW       |
|          | energy            | 50 keV        |
|          | angle             | 25°           |
|          | species           | 0.44 d/0.56 t |
|          | trapping fraction | 0.95          |
| Barrier: | power             | 0.2 MW        |
|          | energy            | 2. keV        |
|          | angle             | 45°           |
|          | species           | d             |
|          | trapping fraction | 0.99          |

ECRF

|       |                       |         |
|-------|-----------------------|---------|
| Plug: | power                 | 14.9 MW |
|       | frequency             | 56. GHz |
|       | absorption efficiency | 1.      |

ICRF

|               |                       |        |
|---------------|-----------------------|--------|
| Central cell: | power                 | 40. MW |
|               | frequency             | 30 MHz |
|               | absorption efficiency | 0.8    |

|                             |         |
|-----------------------------|---------|
| <u>Total Injected Power</u> | 117. MW |
|-----------------------------|---------|

NB: Plug and barrier powers are total powers for both sides.



Maintaining the thermal barrier requires removal ("pumping") of the ions which become trapped in the barrier. TASKA utilizes neutral beam pumping for this purpose; this process occurs by charge exchange between the barrier trapped ions and the neutral atoms which have been injected with the appropriate energy and angle relative to the magnetic field. The required energies and power for neutral beam pumping of the barrier and for sustaining the plug are given in Table II.5-2.

Alpha particle accumulation in the central cell is a general problem with tandem mirror reactors. Accumulation in the central cell of TASKA is not a problem because the ion confining potential is relatively low compared with the ion temperature ( $\phi_c/T_{ic} \approx 1.4$ ); the steady state alpha concentration in the central cell is about 1%. Alpha accumulation in the thermal barriers is a potential problem that requires further study. Fueling of the central cell is done entirely by ionization of the neutral beams which pump the thermal barriers; a separate fueling mechanism is not required, except perhaps during startup.

## II.6 Magnet System of TASKA

The total magnet system of TASKA consists of:

- 3 solenoids in the central cell to give the required central cell field;
- 2 barrier mirror hybrid coils to provide the magnetic mirror field;
- 2 transition coils for plasma cross section shaping;
- 2 yin-yang systems to provide plasma stability;
- 2 recirculating coils for plasma cross section shaping; and
- 8 coils for field shaping in the thermal barrier region.

The (arrangement) position of the coils is given in Figs. II.4-1, II.4-2, and II.6-1, and the magnetic field generated by those magnets is shown in Fig. II.4-3. Table II.6-1 gives the general operating parameters of the magnet system.

The three central cell magnets have an inner radius of 2.8 m and a winding cross section of  $1.2 \times 0.58$  meters. The technology for these coils is available and is based on the experience with large coils already existing. The current density in the two outside coils is  $1440 \text{ A/cm}^2$  and it is  $1345 \text{ A/cm}^2$  in the center coil which is necessary to reduce the field ripple to 5% or less. The maximum field at the conductor is less than 6.0 T and the conductor consists of NbTi with copper and/or aluminum stabilizer and stainless steel reinforcement.

The most complicated coil is the barrier mirror hybrid coil. It consists of a superconducting outer part and a normal conducting inner part; each of these is built up by six solenoids with different current densities and different winding cross sections. The design goes to the limits of the technology which can be expected in the near future; the majority of the field is provided by the  $\text{Nb}_3\text{Sn}$  superconductor with a maximum field of  $\sim 15 \text{ T}$  at the

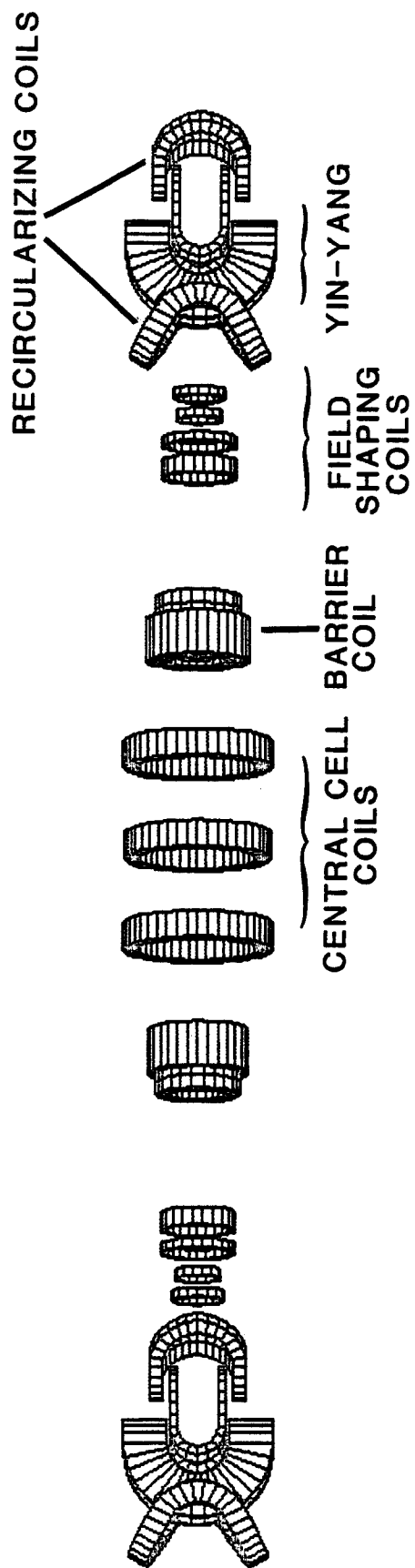


Fig. II.6-1 TASKA magnet configuration.

Table II.6-1. Main Parameters of the TASKA-Magnet System

| Conductor                                                         | Central Cell<br>Solenoids | Barrier Coil<br>S.C. Part                           | Barrier Coil<br>N.C. Part | Transition and<br>Recirculating<br>Coils | Yin-Yangs               | Field<br>Shaping<br>Coils |
|-------------------------------------------------------------------|---------------------------|-----------------------------------------------------|---------------------------|------------------------------------------|-------------------------|---------------------------|
|                                                                   | NbTi, Al-<br>stabilized   | NbTi/Nb <sub>3</sub> Sn<br>Cu and Al-<br>stabilized | Hard Cu                   | NbTi, Cu-<br>stabilized                  | NbTi, Cu-<br>stabilized | Hard Cu                   |
| Overall current<br>density in the<br>winding (A/cm <sup>2</sup> ) | 1350/1160                 | 1600/2400                                           | 418-3070                  | 1900                                     | 1630                    | 450-795                   |
| Maximum field at the<br>conductor (T)                             | 6                         | 15                                                  | 20.7                      | 7                                        | 7.9                     | 0.5                       |
| Stored energy (MJ)                                                | 368/202                   | 2600                                                | 12                        | 487                                      | 411                     | 0.013-2.25                |
| Weight of the winding<br>(tonnes)                                 | 68/72.5                   | 177                                                 | 20.5                      | 84                                       | 117                     | 1-12.9                    |
| Operating temperature<br>(K)                                      | 4.2                       | 4.2/1.8                                             | 300                       | 4.2                                      | 4.2                     | 300                       |

coil winding and a stored energy of  $\sim 2$  GJ. The design of the normal conducting part is made to be most power effective and is based on the experience of high field coils of the Bitter or polyhelix type already in operation at various High Magnetic Field Laboratories.

The general design of the normal Cu insert coils includes six nested cylinders, each 10 cm thick, but with a length which increases with increasing radius. Figure II.6-2 shows a cross section of the coil with the lines of constant magnetic field superimposed on it. The inner radius of the coil is 0.3 m to provide space for radiation shielding. The current density drops from  $3070 \text{ A/cm}^2$  in the inner cylinder to  $418 \text{ A/cm}^2$  in the outermost cylinder. The field contribution of the normal copper coil is about 6 T. The outer superconducting coil is limited by stress requirements and by the maximum allowable magnetic field at conductor of 15 T. The coil is graded in the NbTi region for conductor fields of about 8 T and in the  $\text{Nb}_3\text{Sn}$  region for higher fields. The current densities vary from  $1600 \text{ A/cm}^2$  to  $2200 \text{ A/cm}^2$  due to the stress limitation.

The end plug magnets (transition coil, yin-yang, recircularizer) are C-shaped with current densities less than  $1900 \text{ A/cm}^2$ . The design of these magnets was straightforward because they are similar in geometry and electrical data to those of the MFTF-B machine which has already operated under similar conditions. Characteristic dimensions for the yin-yang system are an overall height of 7.0 m and for a transition coil about 5 m. These are comparable with the MFTF-B dimensions of  $\sim 6$  m overall height for the yin-yang and  $\sim 12$  m for the outside A-cell coil. This coil was successfully tested in February 1982.

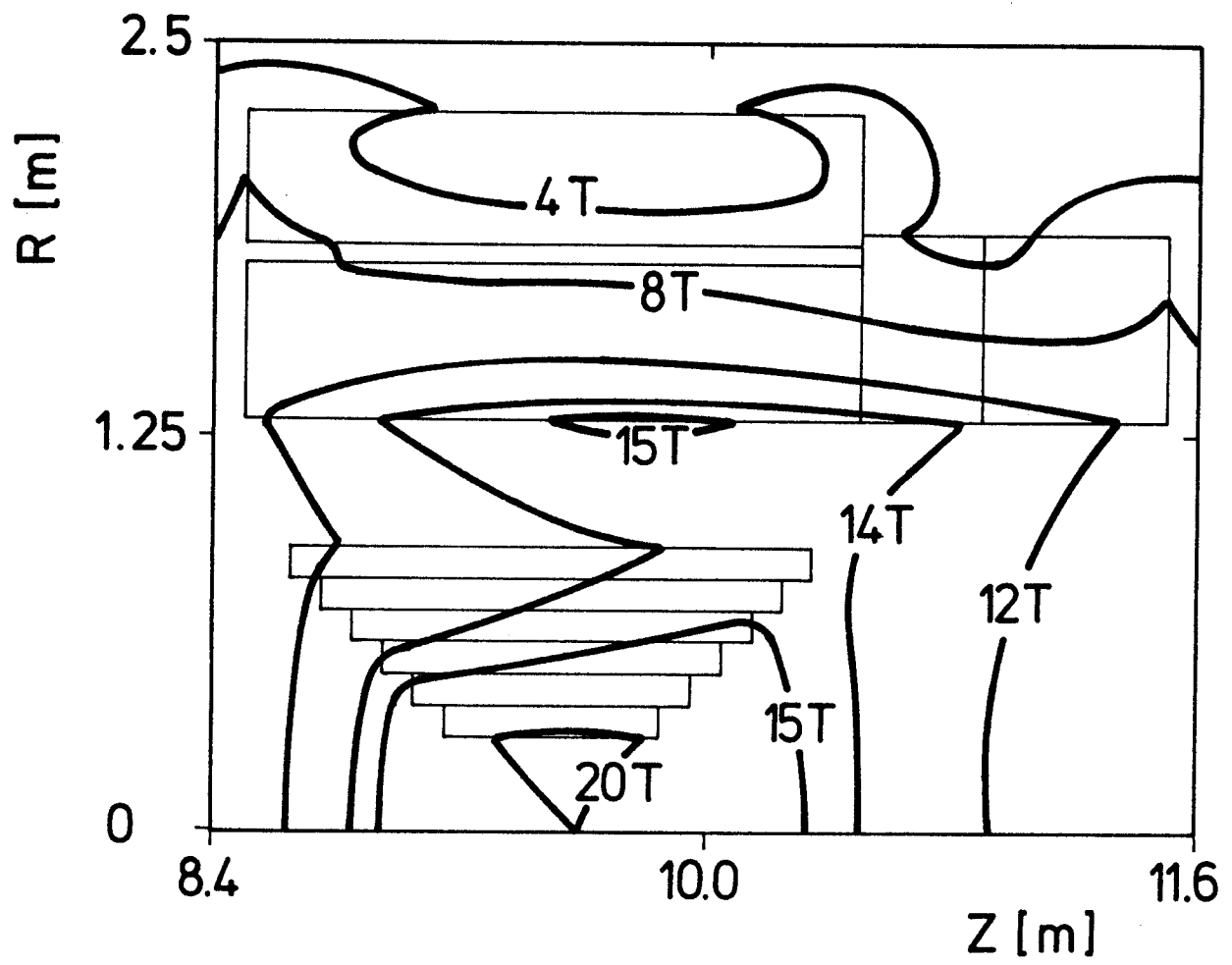


Fig. II.6-2 Cross section of the barrier coil with 4 T, 8 T, 12 T, 14 T, 15 T, and 20 T B-contours.

In addition, access for neutral beam injectors which provide the barrier pumping has to be included. The neutral beam injection angles, dictated by plasma physics requirements, are small which means that the present magnet design is the result of a trade-off between the magnetic fields from the plasma physics requirements and the access requirements for neutral beam injection. To fulfill these requirements, the barrier coil is built up by several parts with different thicknesses and different current densities. In addition, the end plug has been moved away from the barrier coil to enhance NBI access. However, the minimum field in the thermal barrier region is then lower than the required 0.8 T. This requires that field shaping coils (in this case normal conducting) be placed on each end of the device to increase the field to the required level. These coils are integrated in the shield as shown in Fig. II.4-2.

## II.7 Plasma Heating Technology

### II.7.1 Neutral Beam Injection

Neutral beam injection (NBI) is used as the primary means for plasma heating, plasma fueling, and selective ion charge exchange pumping in the TASKA materials test tandem mirror reactor design. Computational studies<sup>(19)</sup> of the equilibrium plasma parameters needed to attain the desired neutron wall loading of  $1.5 \text{ MW/m}^2$  in the central cell yield the preliminary NBI design and heating requirements (see Table II.7-1). For simplicity in present parameter studies, each neutral beam is assumed to consist only of the full energy component.

The NBI performance parameters (listed in Table II.7-1) are based upon a present capability or scaling of existing ion source technology. However, the application to steady state D-T operation for TASKA will require a further level of special technological development particularly in the following areas:

- The design of high power level steady state NBI/ion sources with high reliability and compatibility with remote NBI component changeout. Operation of NBI hardware in a high neutron fluence with a tritium loaded system will also require some form of remote handling for maintenance (the aim is for a minimum of 1 year normal lifetime before required maintenance).
- Special cooling technology will be required because of locally high beam power loading on the charged ion dump, on the remnant neutral beam dump, on associated diagnostics, on the ion source grids, and on the machine end walls. Solutions such as thin channel-cooled surfaces<sup>(20)</sup> or direct energy recovery for ion dump and end walls<sup>(21)</sup> are feasible.



Table II.7-1. TASKA Neutral Beam Injector and Heating Requirements

| Location                  | Injection Angle                            | $V_0$ (kV) | NBI Power Per Injector (MW) | # Ion Sources Per Injector | Total # Injectors | $I^0$ (Equiv.) Per Injector                 | $I^\pm$ (EST) Per Injector                   |
|---------------------------|--------------------------------------------|------------|-----------------------------|----------------------------|-------------------|---------------------------------------------|----------------------------------------------|
| P-NBI (plugs)             | 60°                                        | 250        | 2.7                         | 1/H <sup>-</sup> est.      | 2                 | 10.8 AH <sup>0</sup>                        | 29 AH <sup>-</sup>                           |
| HE-NBI (barriers)         | 20°                                        | 76         | 3.3                         | 1                          | 2                 | 43 AD <sup>0</sup>                          | 108.8 AD <sup>+</sup>                        |
| ME-NBI (barriers)         | 25°                                        | 50         | 8.3                         | 3                          | 6                 | 3 x 65 A<br>D <sup>0</sup> + T <sup>0</sup> | 3 x 362 A<br>D <sup>-</sup> + D <sup>+</sup> |
| LE-NBI (barriers)         | 45°                                        | 2          | .1                          | 1                          | 2                 | 50 AD <sup>0</sup>                          | 97.5 AD <sup>+</sup>                         |
| ECRH (plugs and barriers) | ~ 56 GHz 4 units each 3.7 MW to plasma     |            |                             |                            |                   |                                             |                                              |
| ICRH (central cell)       | ~ 21 MHz 1 unit yielding ~ 40 MW to plasma |            |                             |                            |                   |                                             |                                              |

- The development of steady state vacuum pumping schemes for the  $H_2$ ,  $D_2$  or  $D_2/T_2$  gas flow in the ion source and neutralizer is required. Sufficient local pumping speed is required to reduce the line density of gas beyond the ion dump to low enough values to limit re-ionization loss to less than 10% and eliminate duct choking. The low injection angles for the high energy and medium energy pump beams tend to require long ducts and careful baffling to avoid direct beam impingement on the walls. The cooling of duct walls (to avoid any thermal desorption) is also essential. Initial studies<sup>(22)</sup> with a long duct NBI system possessing a small thermal desorption term of .005 T- $\ell$ /s-kJ, revealed that we may have a duct choking problem. To overcome this, present plans call for  $H_2$ ,  $D_2$  and  $T_2$  gas pumping using panels of solid getter material arranged in a full surface folded panel configuration which allows cyclic regeneration. A short cold wall/hot exit neutralizer is being considered to reduce the neutralizer gas flow.<sup>(23)</sup>
- The relatively high plug-NBI energy requirements and the need to avoid neutron production in the plugs call for the use of  $H^0$  injection at 250 keV. This will require  $H^-$  ion source technology for a reasonable neutralization efficiency. Such a source is also desirable for other fusion devices.
- Magnetic shielding for the NBI's in the high fringe magnetic field from TASKA (particularly for beam lines at low injection angles) is required. While such shielding is difficult to include in the small space allowed, it is feasible.

It is obvious that TASKA represents a challenge to NBI technology, but no insurmountable problems have been identified. Testing of advanced neutral beam injectors and heating systems in an integrated system can be one of the key applications for TASKA.

### II.7.2 ECRH and ICRF Heating

The plug electrons are maintained at a temperature of 59 keV with 7.5 MW of ECRH power per plug. The frequency chosen is 56 GHz, which corresponds to resonance at a magnetic field of 2 T. The resonance surface is located between the minimum field point in the barrier and the mirror throat of the plug. Electrons reaching the resonance surface are primarily plug electrons; heating the plug electrons there heats the entire plug electron population because of their bounce motion along B, drift motion in the flux surface, and collisional transfer to electrons which don't reach the resonance surface. The 56 GHz frequency corresponds to the upper limit of high power cw gyrotron sources considered to be available on the TASKA time scale.

The ECRH power is delivered to the plasma using a quasi-optical offset Cassegrain beam waveguide transport system, as shown in Fig. II.7-1. Using a set of hyperbolic-parabolic mirrors, the microwave power is reflected and focussed onto the plasma at the desired angle. An array of gyrotrons feeds a single launcher system. Up to 4 MW can be focussed onto a reasonable spot size at the plasma with reasonable electric field intensity. Consequently, two launcher systems are required per plug.

Forty megawatts of ICRF heating of central cell ions is used to maintain the ion temperature at 30 keV; this allows a considerable reduction of the neutral beam energy and power required for pumping of the thermal barrier. The fundamental deuterium frequency at the beta-corrected magnetic field in the central cell is 15 MHz. In this frequency range, and because of the 32 cm hot plasma radius, we are constrained to using antennas to couple the ICRF power to the plasma. In order to improve antenna coupling, second harmonic heating (at 30 MHz) is used. To protect the antennas from alpha particle

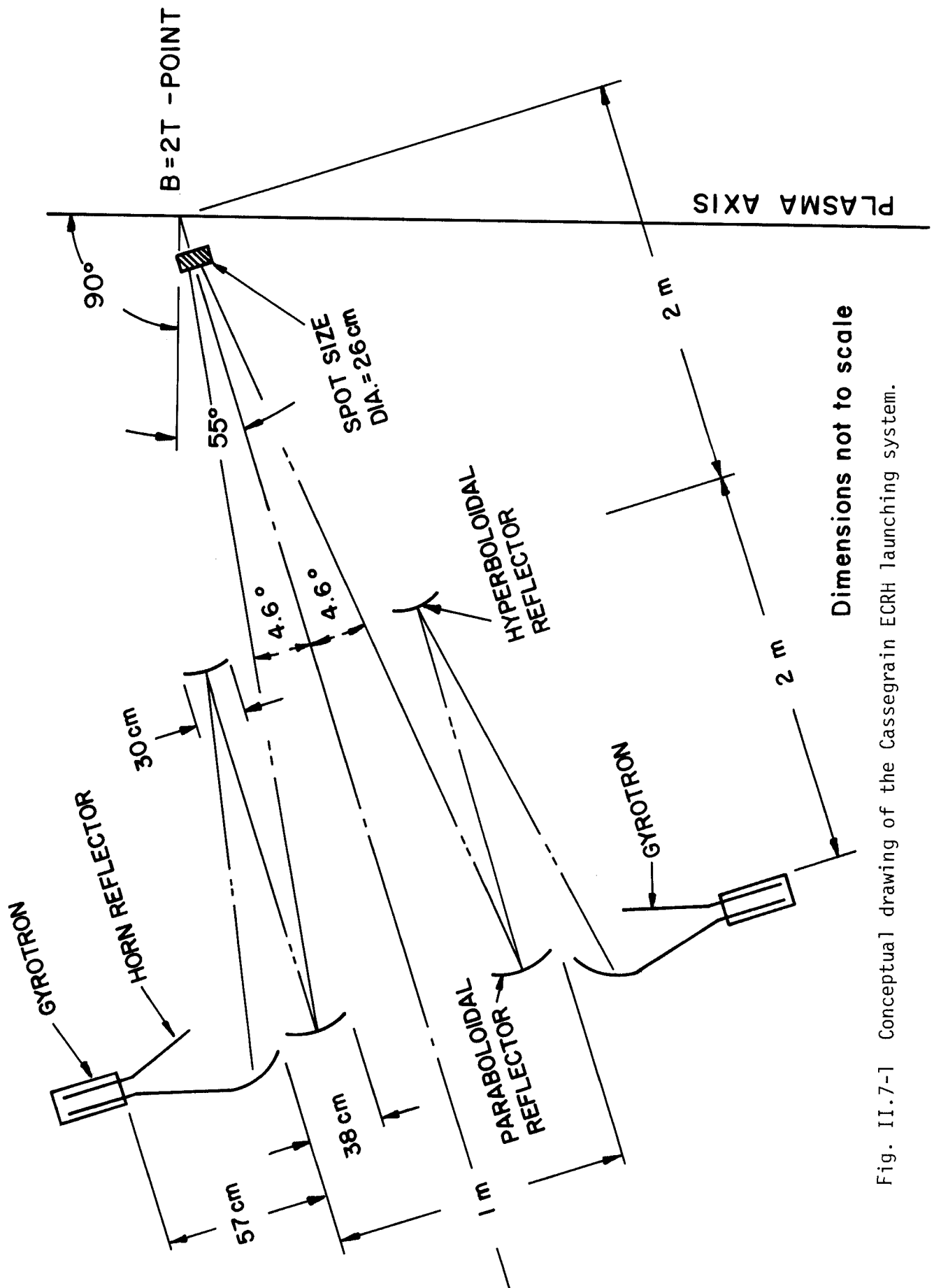
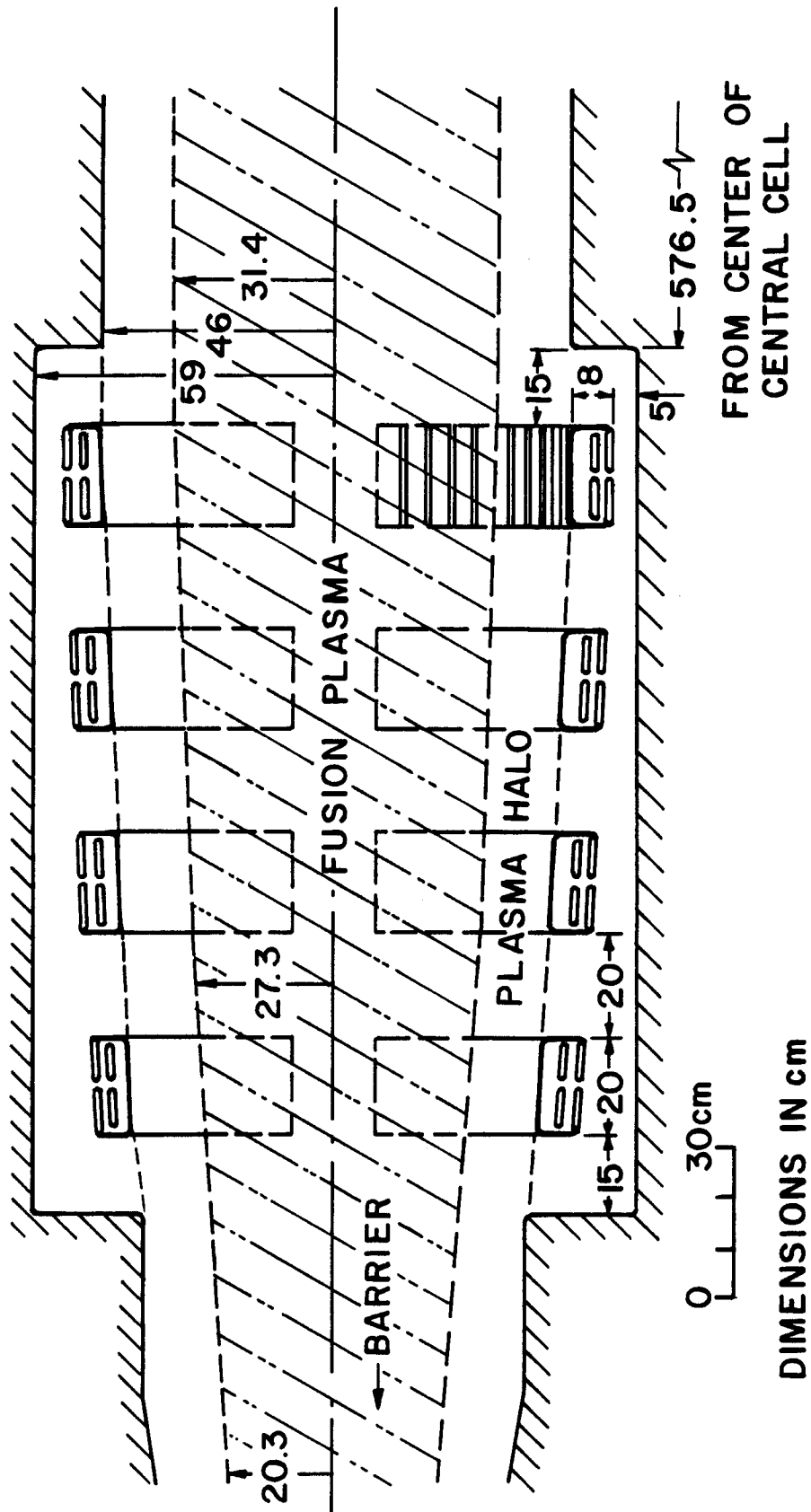


Fig. II.7-1 Conceptual drawing of the Cassegrain ECRH launching system.

bombardment and to improve the coupling to the hot plasma, a warm plasma halo between the hot plasma and the antennas is proposed. The antennas are located at each end of the central cell in order to leave the central region free for test modules. The four 2-coil sets (see Fig. II.7-2) take up 1.4 m of axial length at each end. The antennas are austenitic stainless steel with a high conductivity copper surface layer and cooled by water. The Faraday shields are made of molybdenum and radiation cooled.



II.7-7

## CENTRAL CELL ICRF ANTENNA LAYOUT

Figure II.7-2

## II.8 Blanket Design

The TASKA central cell region has two main types of blankets: (1) the permanent tritium breeding and heat removal blanket, and (2) test blanket modules. The first blanket system is designed to last the lifetime of TASKA (15 years), while the second blanket system is designed to be changed rather frequently to test a variety of breeding materials. There are completely separate coolant systems for each system, but they are based on the same design principles as shown in Fig. II.8-1.

### II.8.1 Permanent Breeding Blanket

The coolant-breeder is  $\text{Pb}_{83}\text{Li}_{17}$  which enters the U-shaped tubes at 300°C and leaves at 400°C. The tubes are made from a martensitic alloy, HT-9, which has displayed a great resistance to radiation damage in fission reactor tests. The design of the blanket is such that all welded joints are protected by roughly 1 meter of  $\text{Pb}_{83}\text{Li}_{17}$ /HT-9 material which reduces the radiation damage in the weld region.

The localized tritium breeding ratio is calculated to be  $\sim 1.65$  which more than compensates for the lack of breeding in the materials test modules and the loss of neutrons out the ends of the machine. When the overall tritium breeding ratio is calculated, it is  $\sim 1.0$  (including a nominal breeding ratio of 1 from the test blanket region). The low solubility of tritium in the  $\text{Pb}_{83}\text{Li}_{17}$  alloy gives an inventory of only  $\sim 20$  grams. This low inventory, coupled with the reduced chemical activity of this liquid metal, makes it very attractive from a safety standpoint.

### II.8.2 Test Blanket Module

A breeder test module with liquid lithium as coolant and breeding material has been designed (Fig. II.8-2). On the basis of neutronic calculations,

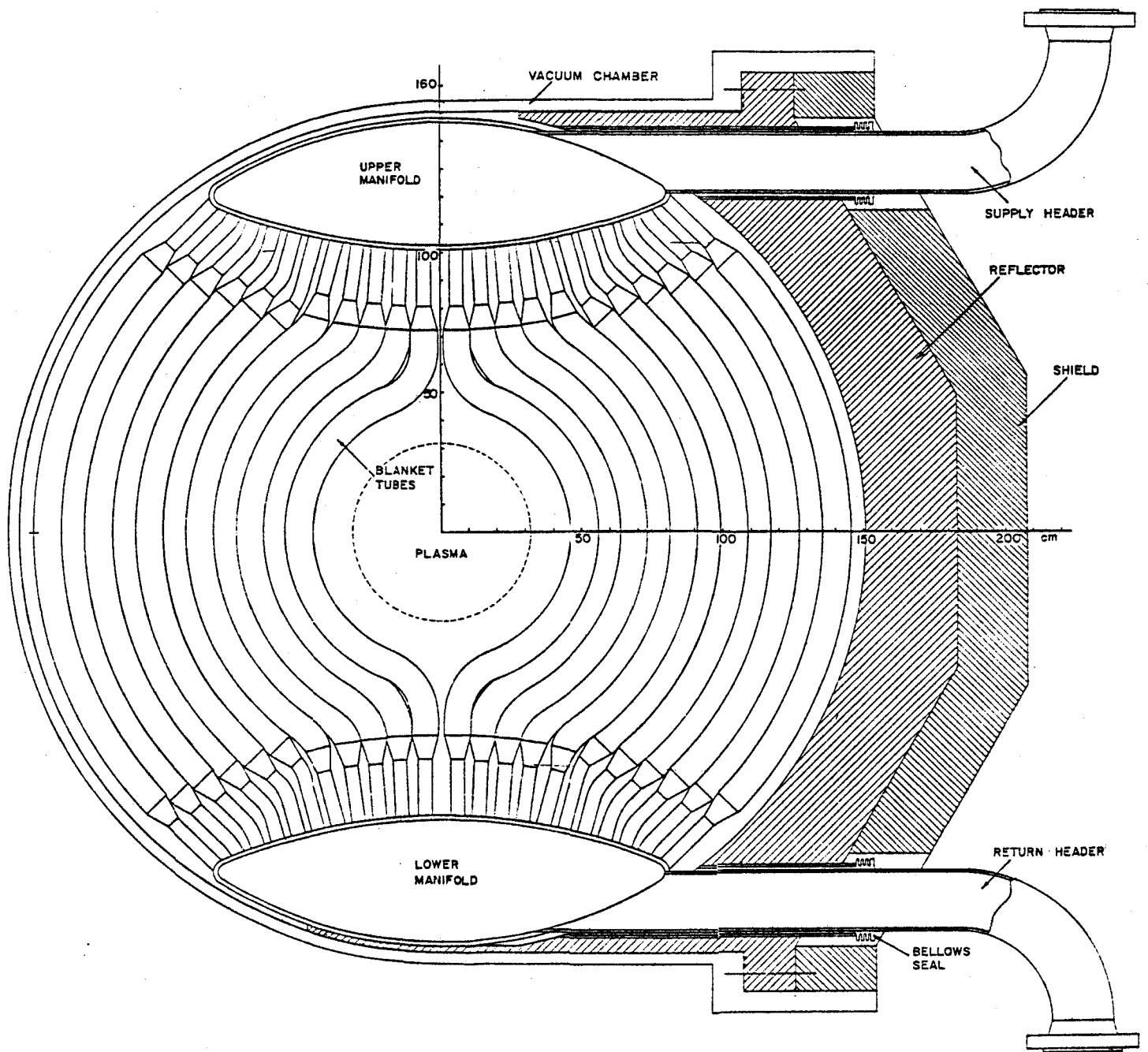


Fig. II.8-1 Vertical section of TASKA blanket



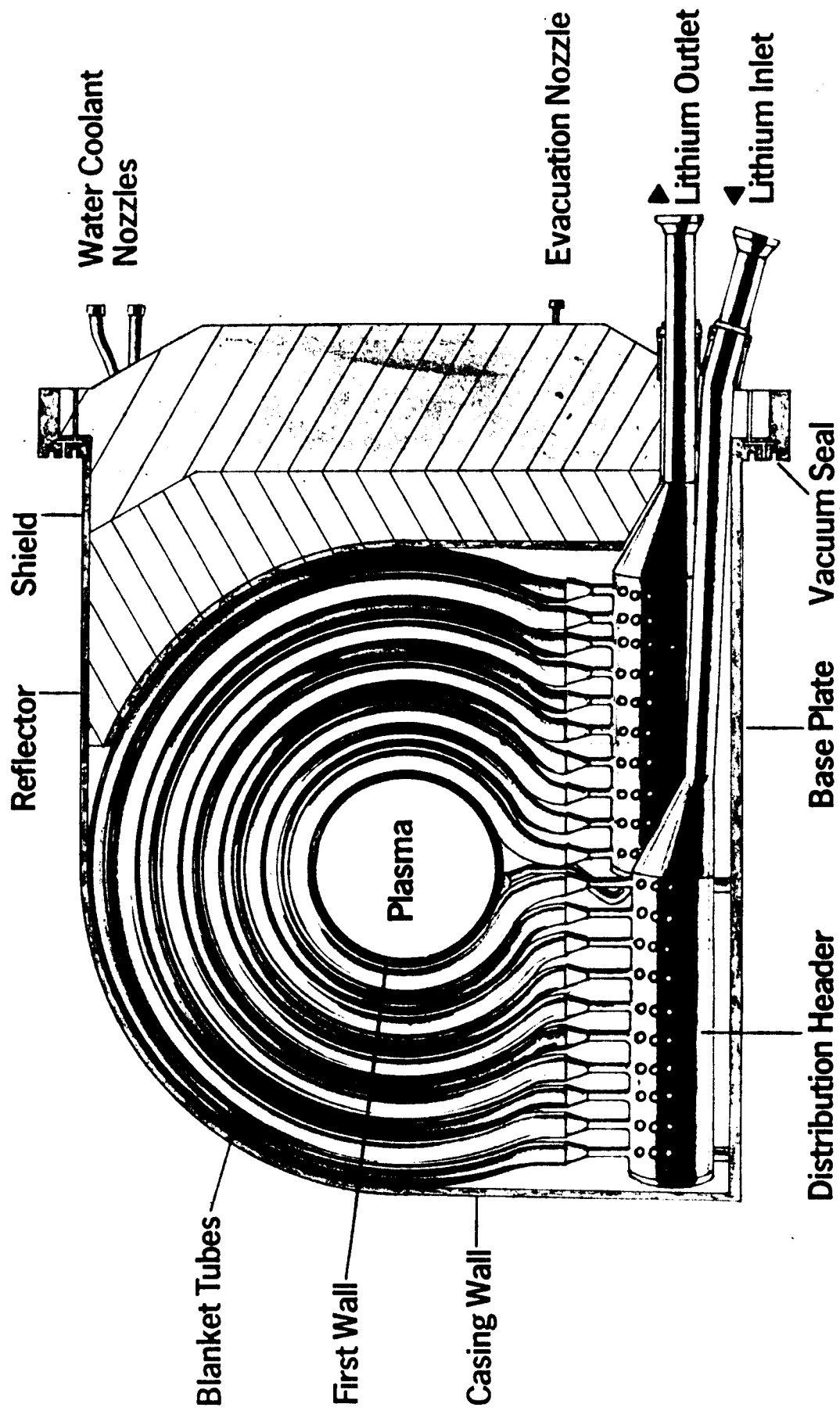


Fig. II.8-2

## Taska-Test Module

the test module will be provided with a liquid lithium blanket using natural lithium as breeding material. This will result in a blanket power of about 3 MW (design value) at a maximum neutron wall loading of  $1.5 \text{ MW/m}^2$ . The operation target of one test module is expected to be 4 years at a load factor of 25% or 2 years at 50%.

The test module consists of three main parts: casing, blanket with headers and coolant pipes, and reflector and shield. The blanket module is constructed such that the vacuum vessel wall is shielded from the direct 14 MeV neutron flux.

Liquid lithium is circulated in tubes bent around the first wall cylinder. The tubes are fed from headers connected by nozzles of the required size and length to connect all tube ends by one inlet and one outlet header. To keep the tubes a fixed distance from each other they are provided with wire or helical finned spacers. The main coolant pipes are stacked one above the other. Headers and main coolant pipes are heat isolated by ceramic materials. Reflector and shield are designed as water-cooled plug units.

## II.9 Materials Testing

One of the main functions of TASKA is to test structural materials for the fusion Demonstration Power Reactor which should operate shortly after the turn of the century. To satisfy this function, we require three main features of TASKA. The first is a high neutron wall loading to reduce the time needed to accumulate significant damage levels; TASKA is designed to give  $1.5 \text{ MW/m}^2$ . The second is a high reactor availability to make the most effective use of the irradiation time. In this respect we used the proposed operating sequence for INTOR <sup>(1)</sup> (see Table II.4-1), but because (1) the duty cycle of TASKA is 1 versus 0.7 to 0.8 for INTOR, (2) the wall loading is higher ( $1.5$  versus  $1.3 \text{ MW/m}^2$ ), and (3) the samples can be placed closer to the vacuum surface in TASKA than in INTOR, we obtain even higher cumulative damage levels in TASKA than in INTOR. It should also be noted that maintenance of TASKA is much simpler than for a tokamak so that if INTOR can attain an availability of  $\sim 50\%$ , TASKA should be able to attain an even larger value (however, we did not take credit for that in this study). The third requirement for the materials test program is for a large testing volume and in that respect we have designed for more than  $3 \times 10^5 \text{ cm}^3$  of high flux test volume in TASKA.

Analyses of the special test module REGAT (Reduced Damage Gradient Test) design as well as types and numbers of samples to be irradiated are given elsewhere<sup>(24,25)</sup> and we will only present the results of that discussion here. Figure II.9-1 shows the general design of the test module and one of the 354 test capsules that can be used to irradiate over 27,000 specimens in 10 years (see Table II.9-1) The REGAT modules are either He or  $\text{H}_2\text{O}$  cooled and provide test environments ranging from 300 to  $600^\circ\text{C}$  at damage levels up to  $\sim 100 \text{ dpa}$ .

# **TANDEM MIRROR SINGLE UNIT TEST MODULE**

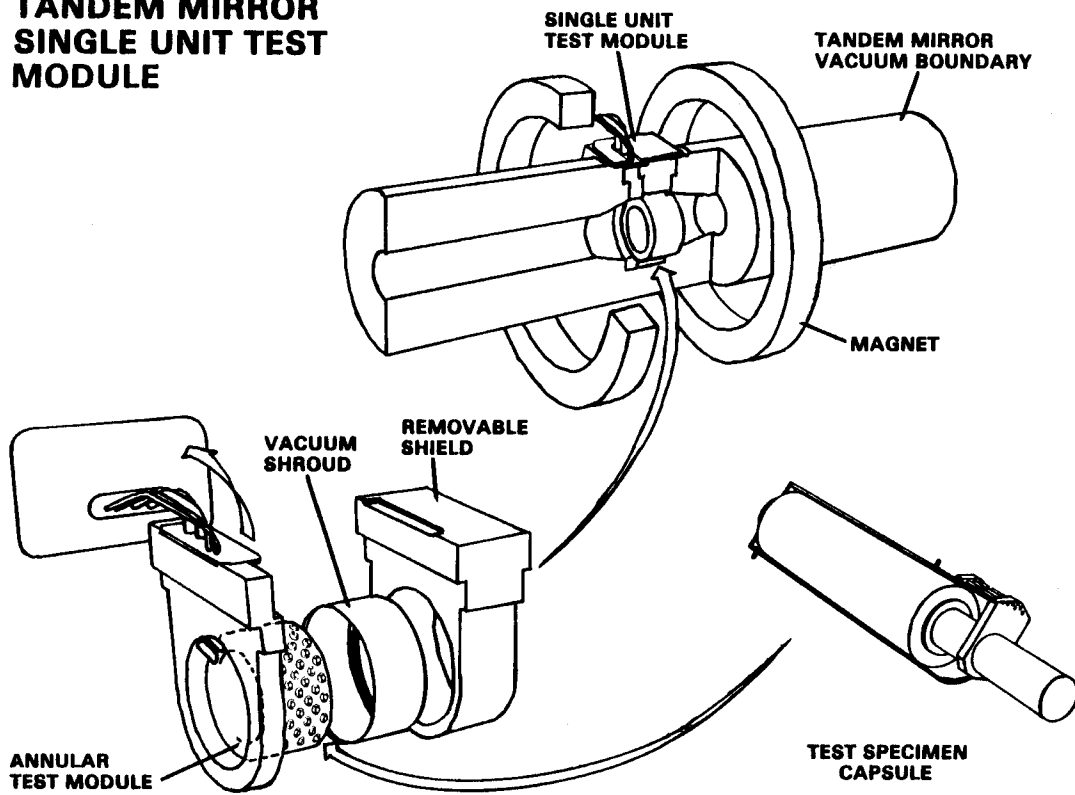


Fig. II.9-1

TABLE II.9-1

## Proposed Materials Test Matrix - TASKA

| Materials Surveillance       | Mat. Var. <sup>a</sup> | Dup. <sup>b</sup> | Temp. <sup>c</sup> | Fluence <sup>d</sup> | Conditions         | Total Specimens     |                      | Total Capsules <sup>g</sup> |                      |
|------------------------------|------------------------|-------------------|--------------------|----------------------|--------------------|---------------------|----------------------|-----------------------------|----------------------|
|                              |                        |                   |                    |                      |                    | Mirror <sup>e</sup> | Tokamak <sup>f</sup> | Mirror <sup>e</sup>         | Tokamak <sup>f</sup> |
| Materials Surveillance       |                        |                   |                    |                      |                    |                     |                      |                             |                      |
| Tensile                      | 3                      | 2                 | 2                  | 2                    | 2 rate/temp.       | 48                  | 48                   | 3                           | 3                    |
| Fatigue                      | 0(3)                   | 0(2)              | 0(2)               | 0(2)                 | 0(4) stress        | 0                   | 96                   | 0                           | 4                    |
| Crack growth                 | 1(2)                   | 2                 | 2                  | 3                    | 3 stress levels    | 36                  | 72                   | 1                           | 2                    |
| Fracture                     | 2                      | 2                 | 2                  | 2                    | 2 temp.            | 32                  | 32                   | 4                           | 4                    |
|                              |                        |                   |                    |                      | Subtotal           | 116                 | 248                  | 8                           | 13                   |
| Structural Materials         |                        |                   |                    |                      |                    |                     |                      |                             |                      |
| Tensile                      | 5                      | 3                 | 6                  | 5                    | 6 rate/temp.       | 2700                | 2700                 | 10                          | 10                   |
| Fatigue (High Cycle)         | 0(8)                   | 0(3)              | 0(2)               | 0(4)                 | 0(4) stress        | 0                   | 768                  | 0                           | 29                   |
| Crack growth                 | 3(6)                   | 2                 | 4                  | 4                    | 3 stress levels    | 288                 | 576                  | 3                           | 6                    |
| Fracture                     | 4                      | 3                 | 4                  | 2                    | 2 temp.            | 192                 | 192                  | 5                           | 5                    |
| Swelling                     | 10                     | 5                 | 10                 | 5                    | 4 post irr. tests  | 10,000              | 10,000               | 5                           | 5                    |
| Stress relax.                | 6                      | 4                 | 6                  | -                    | 5 stresses         | 720                 | 720                  | 3                           | 3                    |
| Creep-rupture                | 6                      | 2                 | 6                  | -                    | 5 stresses         | 360                 | 360                  | 2                           | 2                    |
| In situ-cyclic               | 0(4)                   | 0(2)              | 0(2)               | -                    | 0(4) stress levels | 0                   | 64                   | 0                           | 64                   |
|                              |                        |                   |                    |                      | Subtotal           | 14,260              | 15,380               | 28                          | 124                  |
| Other Materials <sup>h</sup> |                        |                   |                    |                      |                    |                     |                      |                             |                      |
| Fatigue                      | 5                      | 3                 | 4                  | 4                    | 2 strain range     | 480                 | 960                  | 11                          | 22                   |
| Tensile                      | 15                     | 3                 | 6                  | 4                    | 4 rate/temp.       | 4,320               | 4,320                | 15                          | 15                   |
| Dimensional stab.            | 15                     | 5                 | 6                  | 4                    | 4 post tests       | 7,200               | 7,200                | 4                           | 4                    |
| Creep                        | 10                     | 2                 | 6                  | -                    | 4 stresses         | 480                 | 480                  | 2                           | 2                    |
| In situ-cyclic               | 0(6)                   | 0(2)              | 0(2)               | -                    | 0(4) stresses      | 0                   | 96                   | 0                           | 96                   |
| Fracture                     | 3                      | 2                 | 2                  | 2                    | 6 temp.            | 144                 | 144                  | 7                           | 7                    |
| Elect. prop.                 | 6                      | 3                 | 2                  | 4                    | 3 tests            | 432                 | 432                  | 18                          | 18                   |
| Therm. Cond.                 | 6                      | 3                 | 4                  | 4                    | ---                | 288                 | 288                  | 12                          | 12                   |
|                              |                        |                   |                    |                      | Subtotal           | 13,344              | 13,920               | 69                          | 176                  |
|                              |                        |                   |                    |                      | Total              | 27,720              | 29,548               | 105                         | 313                  |

<sup>a</sup>Materials x variations<sup>b</sup>Duplication<sup>c</sup>Number of irradiation temperatures<sup>d</sup>Fluence levels<sup>e</sup>Specifically for the Mirror Program<sup>f</sup>Required for both Mirror and Tokamak Programs<sup>g</sup>Capsule volume-390 cm<sup>3</sup><sup>h</sup>Ceramics, electrical, and heat dump materials

( ) Addition for tokamak program

In order to compare the performance of TASKA to other test facilities we show in Fig. II.9-2 the cumulative damage in four devices of interest to fusion materials scientists: the RTNS-II at LLNL, the proposed FMIT device at HEDL, TASKA, and the INTOR test reactors. The main points from this figure are:

1. The accumulated damage rate and test volume is too small in RTNS to be of importance for the Demo.
2. The FMIT will produce high damage levels by the early 1990's, but the test volumes are rather small, 10-100's of cm<sup>3</sup>.
3. If TASKA and INTOR have exactly the same starting dates and operating schedules, reactor relevant damage levels (~ 50 dpa) can be achieved in TASKA by the year 2000, whereas it will take 5-10 years longer to achieve such levels in INTOR. The test volume in both devices is reasonably large (~ 10's of liters).

There is an even better way to represent the testing capabilities of fusion materials test devices and this is by multiplying the damage level times by the test volume at that damage level. Mathematically, this can be stated by:

$$\text{dpa-l} = \int \text{dpa}(v) \, dv \quad ,$$

where the volume,  $v$ , of test space that can give a dpa level in a given time is represented by  $\text{dpa}(v)$ .

A graphical representation of the  $\text{dpa-l}$  values for the test devices is given in Fig. II.9-3. It clearly shows, along with the previous figure, that if one wants to achieve the largest damage times volume product, then clearly

## CUMULATIVE DAMAGE IN FUSION MATERIALS TEST FACILITIES

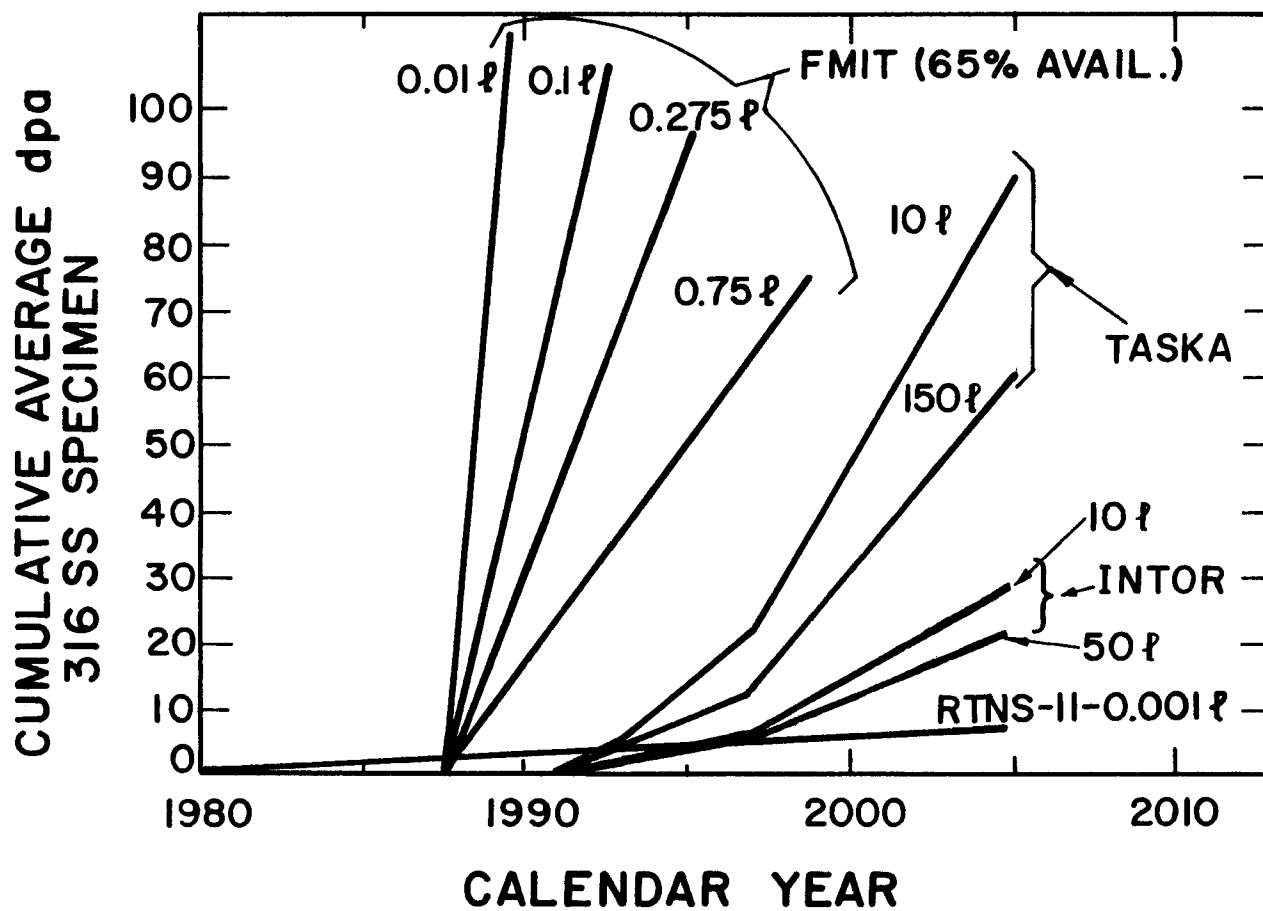


Fig. II.9-2 Cumulative damage in fusion materials test facilities.

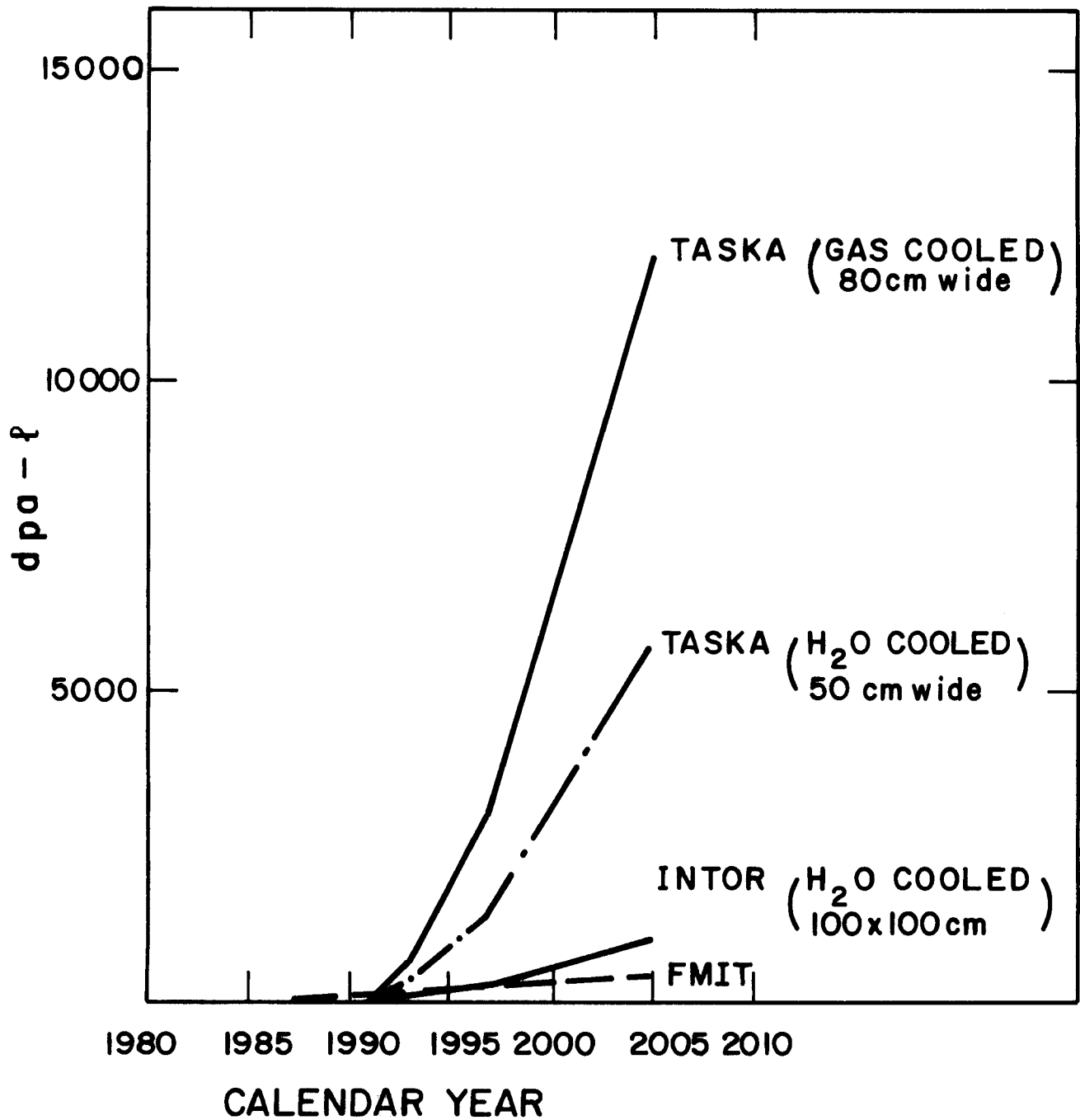


Fig. II.9-3 Cumulative damage times test volume for fusion material test facilities.



the TASKA device is superior to INTOR or FMIT (RTNS data is not discernible on this scale).

## II.10 Maintenance

Successful operation of TASKA can be achieved if the downtime can be maintained at a level consistent with the assumed availability. The TASKA reactor equipment is divided into four classes, depending on the anticipated replacement or repair frequency.

Class 1 - Designed for full lifetime such as buildings, support structure and parts of the vacuum vessel.

Class 2 - Lifetime  $> 5$  years, such as shield, magnets, etc.

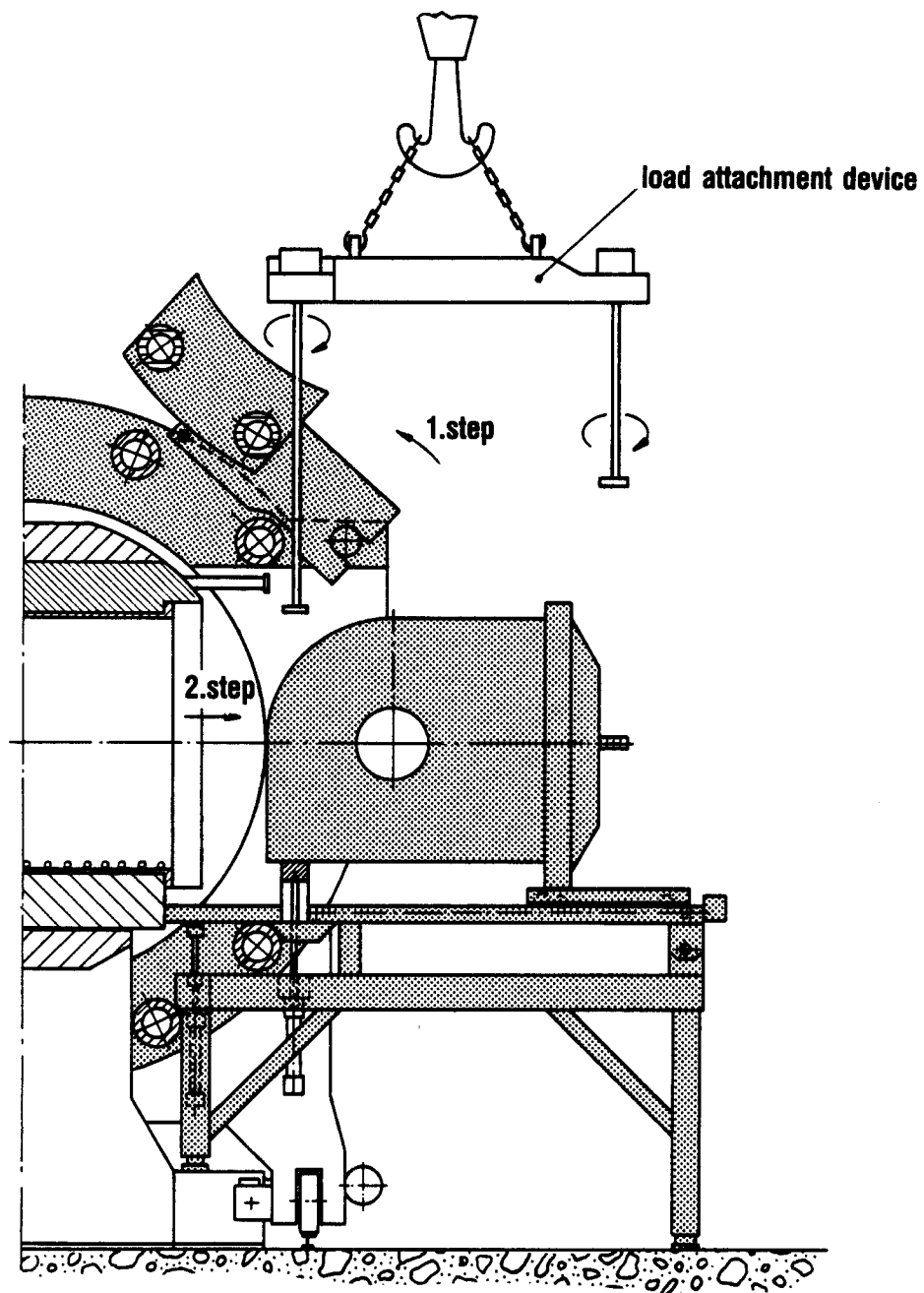
Class 3 - Average lifetime, such as blanket and material test modules, neutral beam components.

Class 4 - Short lifetime, such as handling and experimental equipment.

It is felt that at the minimum the following general purpose remote handling systems will be required:

1. Component handling machine and manipulator carrier attached to the over-head bridge crane.
2. Shielded cabin with manipulator.
3. Elevated work platform.
4. Stereo television viewing system.
5. Various tools adapted for remote handling use.
6. Moveable shields.

The maintenance concepts also address the changeout and repair of blanket modules (as shown in Fig. II.10-1), neutral beam injector components such as ion sources and getter panels, central cell coils and the barrier coils. Displacement of the barrier coil perpendicular to the axis of the reactor (as shown in Fig. II.10-2) is needed to provide access for maintaining the normal insert



### **Blanket exchange**

Fig. II.10-1

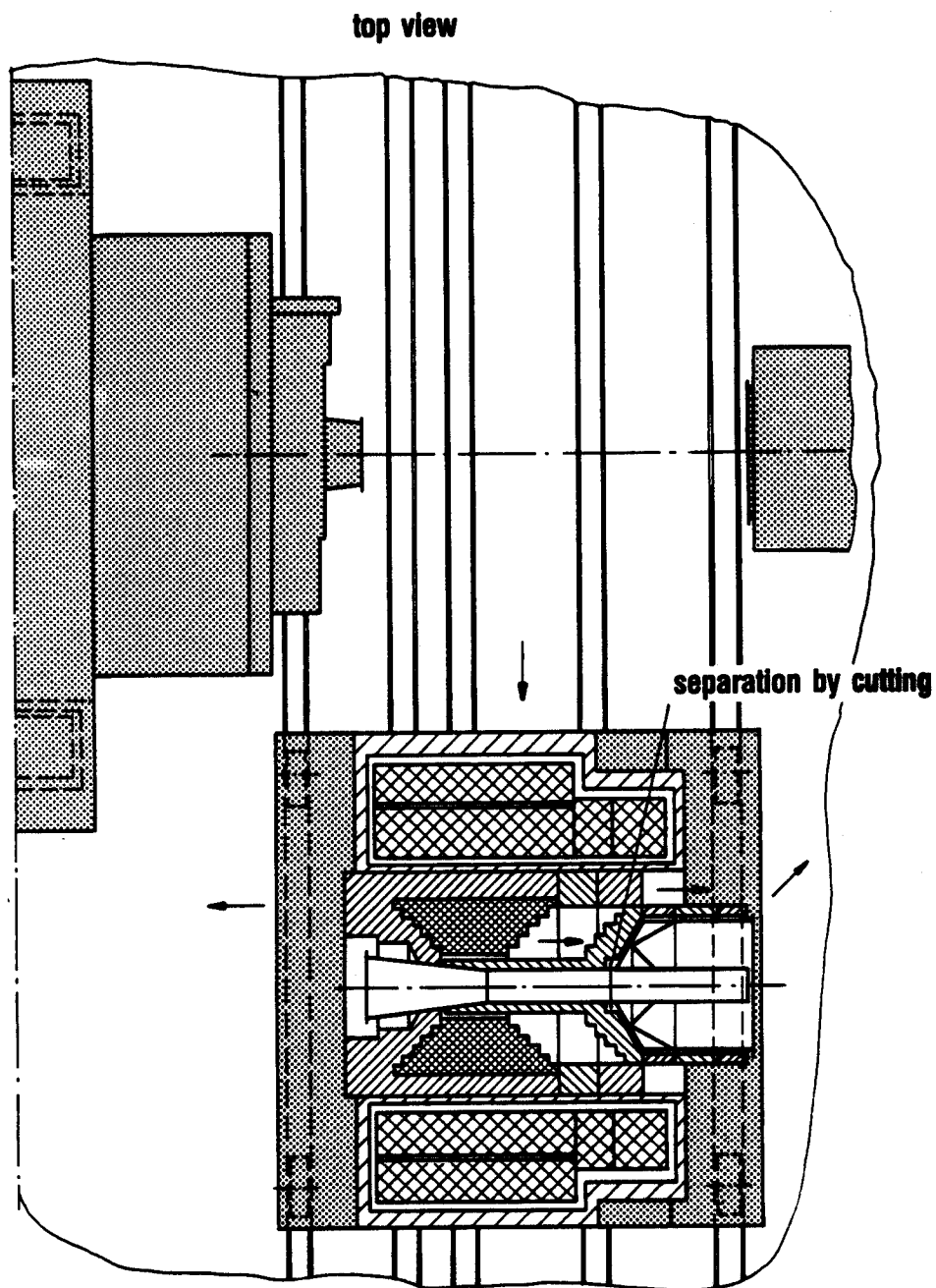


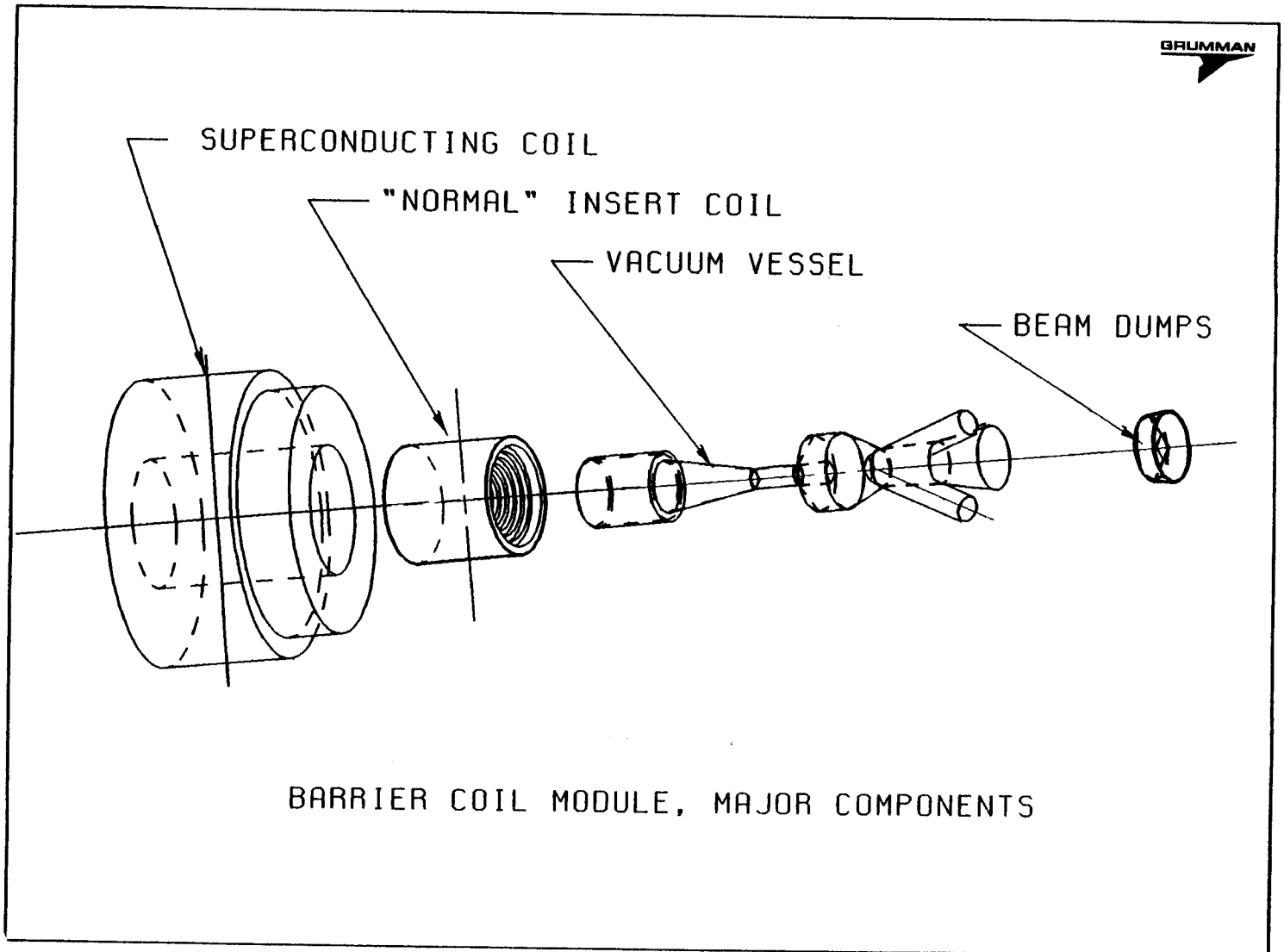
Fig. II.10-2 **Components exchange at barrier region-components dismantling**

and the thermal barrier dumps (an exploded view of the barrier coil and vacuum vessel is shown in Fig. II.10-3). This operation requires the following special purpose remote systems:

1. Special transportable carriage support structure with hydraulic adjustments and a lifting traverse.
2. Turnover device for use in maintaining getter panels.
3. A cutting and welding machine adapted for separating or joining vacuum chamber sections.
4. Auxiliary support structure used in removing the barrier coil.

Although only several critical maintenance operations have been addressed, it is felt that the remaining numerous maintenance tasks can be performed with the general purpose equipment provided and some additional special purpose fixtures.

Fig. II.10-3



## II.11 Cost Analysis

The costs developed for TASKA follow the format adopted for INTOR. The accounts are divided into blanket/shield, magnets, plasma heating, reactor support systems, and buildings. Indirect costs (ID) are engineering (45% of direct costs (DC)), installation/assembly (15% of DC), and contingency (30% of DC and IC). Unit costs and cost algorithms were taken from INTOR (FEDC-M-81-SE-062), PNL-2987, and ORNL (WFPS-TN-057) in that order of information availability. A summary of the costs for TASKA are given below in Table II.11-1:

Table II.11-1. Summary of Direct Costs for TASKA

|                             | <u>\$ x 10<sup>6</sup> (1981\$)</u> |
|-----------------------------|-------------------------------------|
| 1.1 Blanket/Shield          | 46.3                                |
| 1.2 Magnets                 | 227.9                               |
| 1.3 Plasma Heating          | 269.6                               |
| 2.0 Reactor Support Systems | 163.8                               |
| 3.0 Buildings               | <u>80.4</u>                         |
| Total Direct Costs          | 788.0                               |

It can be seen from Table II.11-1 that we were able to meet our objective of < 800 million dollars in direct costs. The main cost drivers are the high costs of ECRH and neutral beams. If some improvement in the ECRH costs can be made, the total cost of TASKA would be proportionately reduced. The other major cost driver is the magnets with the end plug and barrier coils accounting for roughly 1/3 each of the total cost. The intercoil support structure accounts for ~ 28% of the total coil costs. It is obvious that more intense effort to reduce the magnet costs would also be important to reducing the overall cost of TASKA.

In addition to the direct costs, certain percentages for indirect costs such as engineering, assembly and contingency have to be accounted for. As an example, in INTOR these consisted of 45% and 15% of the direct costs for engineering and assembly respectively, and 30% of the direct and indirect costs for contingency.

The operating costs for TASKA are composed of three main items:

1. Annual O & M (including salaries, administrative expenses, etc.).
2. Costs for continuously required power.
3. Additional electrical power costs needed during reactor operation.

The first item is usually taken as 3% of the direct costs, or 24 million dollars per year. It is important to note that this does not include the experimentalists or special equipment used in the blanket or test modules.

The second item amounts to 8.7 MW<sub>e</sub> continuously or ~ 3.4 million dollars per calendar year. The third item includes 189 MW<sub>e</sub> of power during the burning plasma phase and that is ~ 37 million dollars per year (at 50% availability).

The total O & M cost for 15 years is 360 million dollars and the total electrical power costs are 455 million dollars. The levelized annual costs are 54 million dollars per year and will be below our target of 80 million dollars per year.



## II.12 Conclusions

The purpose of the TASKA study was to identify the potential of a tandem mirror device as a technology test bed in a next generation of fusion experiments. This preliminary conceptual design has shown that TASKA can provide meaningful tests of heating technologies, superconducting magnets, remote maintenance equipment, etc., as well as blanket and material tests and that all these reactor relevant technologies could be integrated into one machine of moderate size and with relatively low costs. In particular, the materials testing capabilities are very attractive. A large volume (> 300 liter) of high damage level (up to ~ 100 dpa) testing space is available in TASKA and it can accommodate all the specimens needed to qualify alloys and non-metallic materials for a demonstration plant operating shortly after the turn of the century.

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### III Plasma Physics

#### III.1 Physics Background for TASKA

##### III.1.1 Overview of Tandem Mirror Concepts

The tandem mirror as a confinement scheme was invented in 1976 by Fowler and Logan,<sup>(1)</sup> and independently in the U.S.S.R. by Dimov.<sup>(2)</sup> The basic tandem mirror configuration is shown in Fig. III.1-1. A magnetic mirror cell, commonly called a "plug", is located at each end of a central cell in which the magnetic field is uniform. The fundamental concept is to use the positive ambipolar potential of the mirror-confined plasma to provide axial confinement of plasma in the central cell. The central cell ions are confined by the potential  $\phi_c$  and the electrons by the potential  $\phi_e$ . The idea of using mirrors at the ends of a central cell, but without the potential peak in the end mirrors to improve the axial confinement of the central cell plasma, was first suggested by Kelley<sup>(3)</sup> in 1967. The positive potential peak, however, is important for achieving high Q operation in a power reactor.

Maintaining the axial potential profile shown in Fig. III.1-1 implies, through the Boltzmann relation for electrons, i.e.

$$\phi(z) - \phi_e = T_e \ln \left( \frac{n(z)}{n_c} \right) , \quad (\text{III.1-1})$$

that there is also an axial (z) density variation. In Eq. (III.1-1) it is assumed that, in the central cell,  $n = n_c$ ,  $\phi = \phi_e$ .  $T_e$  is the electron temperature; it is assumed here to be constant along the magnetic field line. The plug density,  $n_p$ , is then related to the confining potential,  $\phi_c$ , by

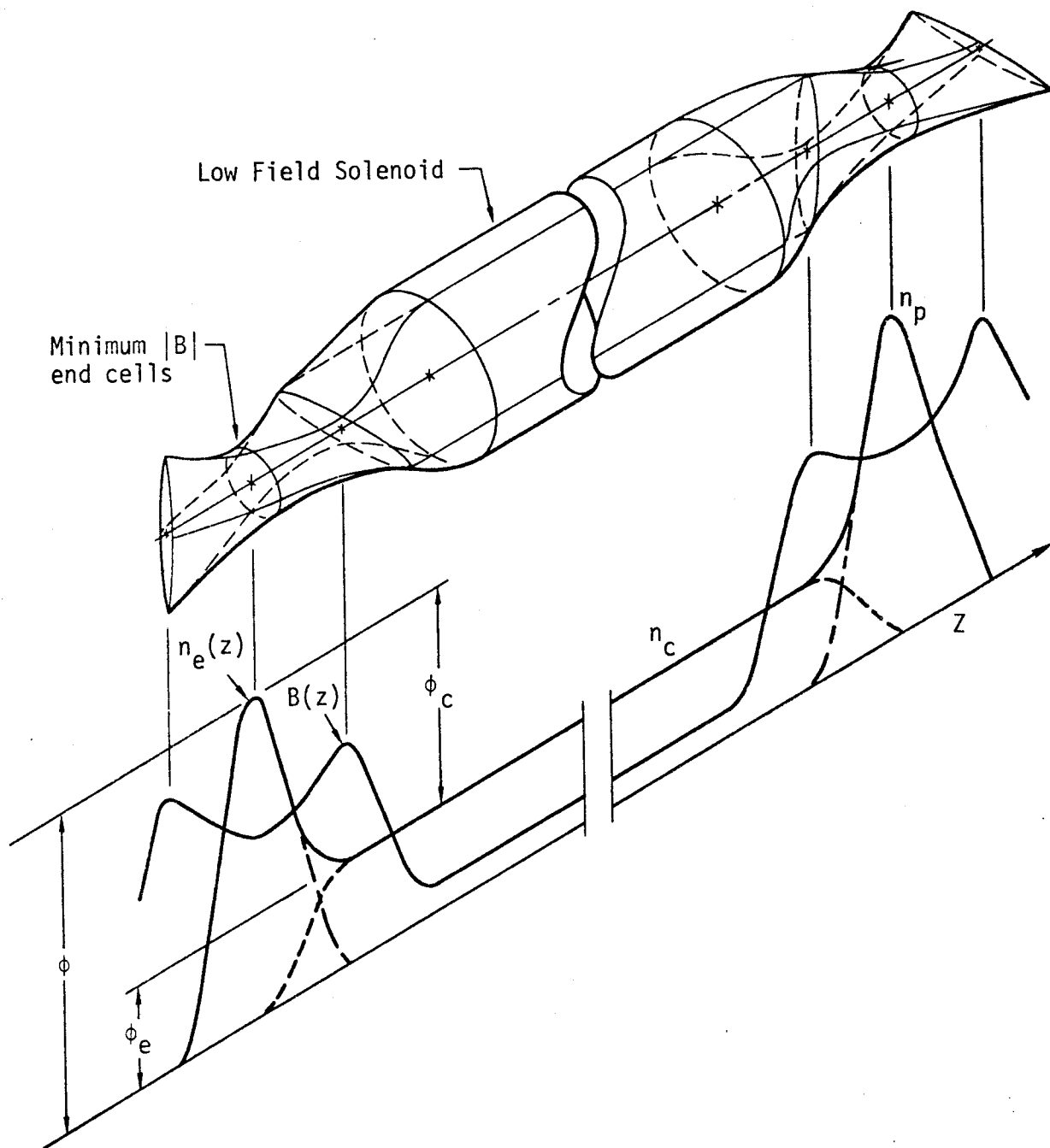


Fig. III.1-1. Tandem mirror with ambipolar plugs at the ends.

$$n_p = n_c \exp \left( \frac{\phi_c}{T_e} \right) . \quad (\text{III.1-2})$$

The axial confinement time of the central cell ions is related to  $\phi_c$  by an expression originally derived by Pastukhov<sup>(4)</sup> for electron confinement in mirrors; further corrections have been obtained by Cohen.<sup>(5)</sup> The confinement time is

$$\tau_i = \tau_{ii} \frac{g(R)}{I\left(\frac{\phi_c}{T_{ic}}\right)} \left(\frac{\phi_c}{T_{ic}}\right) \exp \left( \frac{\phi_c}{T_{ic}} \right) \quad (\text{III.1-3})$$

where

$$I(x) = 1 + \frac{\sqrt{\pi}x}{2} \exp \left( \frac{1}{x} \right) \operatorname{erfc} \left( \frac{1}{\sqrt{x}} \right) ,$$

$$g(R) = \sqrt{\pi} (2R + 1) \ln \frac{(4R + 2)}{4R} ,$$

$\tau_{ii}$  is the ion-ion collision time,  $T_{ic}$  is the central cell ion temperature, and  $R$  is the central cell mirror ratio. Equation (III.1-3) is valid in the collisionless regime, which is the TASKA regime. For current experiments, which are in a more collisional regime, there is an additional term which is not shown. Typically  $\phi_c \approx 2.5 T_{ic}$  for good axial ion confinement, so that  $n_p/n_c \approx 10$ . The high density in the end plug is sustained by neutral beam injection into the end plug.

The potential,  $\phi_e$ , of the central cell relative to the end wall is determined by equating the total ion loss rate to the total electron loss rate. If the radial transport is non-ambipolar, then the radial loss rate + axial loss rate is needed to determine  $\phi_e$ . Normally,  $\phi_e \approx 5 T_e$ .

The plug magnetic field has good curvature for MHD stability. The central cell has zero curvature, but the connection of the central cell to the minimum-B plug introduces bad curvature so that the total system is no longer a minimum-B system but minimum-average B, if the plugs provide enough good curvature. The contribution of the plugs to the overall stability is increased by operating the plugs at high  $\beta$ ; consequently stability against interchange modes is easy to obtain. Ballooning modes, which localize in regions of bad magnetic curvature, are a potential problem, but present indications are that finite gyroradius corrections to ideal MHD theory are sufficient to stabilize them. The MHD aspects of tandem mirrors are discussed in more detail in Section III.1.4.

The high density in the end plugs makes them rather collisional and thus increases the ion scattering rate into the loss-cone unless the mean ion energy is rather high. The high density coupled with the high mean energy implies high plasma pressure and high magnetic field strength in the plug to contain the end plug plasma. The high mean ion energy requires a high energy neutral beam to sustain the end plug plasma. Consequently the technological problems of the end plugs in a reactor based on the standard configuration can be severe. The reactor implications of the standard tandem mirror configuration have been studied at Livermore,<sup>(6)</sup> Wisconsin, and elsewhere. These studies found that very high magnetic field strengths in the plugs ( $\sim 15$ - $17$  T) and very high energy neutral beams ( $\sim 1$  MeV) were required for the reasons discussed above. Even with these requirements, the Q (ratio of fusion power to injected power) was marginal ( $\sim 5$ - $10$ ) and the neutron wall loading in the central cell was low ( $\sim .5$ - $1$  MW/m<sup>2</sup>). (A low neutron wall loading generally implies a high capital cost per unit power.) Using RF in the plugs to heat

the plug ions allowed the neutral beam energy to be reduced to  $\approx 500$  keV, but the other characteristics remained.

The difficulty with the standard tandem mirror configuration is easily seen from Eq. (III.1-1). Since  $T_e$  is assumed to be constant along a field line, the potential, which provides the confinement, varies only logarithmically with the density. Furthermore, the central cell has the lowest density, where one wants high density in order to get high fusion power density. Raising the electron temperature everywhere gives more potential for a given density variation, but increases the electron pressure in the central cell. For a given central cell  $\beta$  and field, the central cell ion density has to be reduced if the electrons are heated. This reduces the fusion power.

A solution to the difficulties with the tandem mirror is to heat the plug electrons to high temperature in order to get a high confining potential, but avoid heating the central cell electrons so that their contribution to the central cell  $\beta$  is not increased. The thermal barrier, introduced by Baldwin and Logan<sup>(7)</sup> in 1979, allows one to do this by introducing a potential dip between the central cell and the plug. This potential dip reflects most central cell electrons and does not allow them to enter the plug. This allows the plug electron temperature to be decoupled from the central cell electron temperature. The hotter plug electron temperature, achieved by ECRH or other forms of heating, allows one to reduce the plug density relative to the central cell density. This also reduces the collisionality of the plug ions and allows one to use lower energy neutral beam injection. The reduced plug pressure reduces the required magnetic field in the plug for a given  $\beta$ . The reactor implications are striking;<sup>(8,9)</sup> the net effect of the thermal barrier is substantially higher  $Q$  ( $\sim 20-30$ ) and neutron wall loading ( $\sim 2-3$  MW/m<sup>2</sup>),



with a peak plug magnetic field of the order of 9 T at the coil and neutral beam injection energy  $\sim 400\text{--}500$  keV.

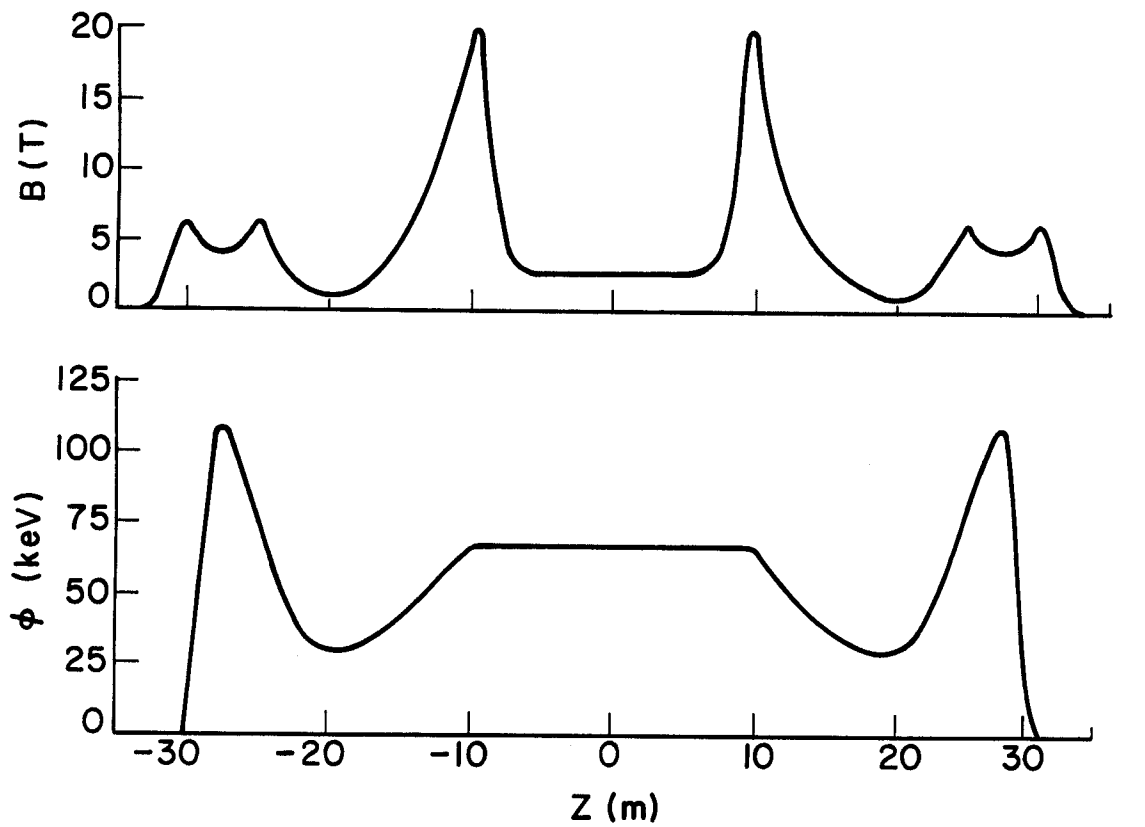
The original thermal barrier configuration, now called an "inboard" barrier, introduces an additional mirror between the plug and central cell, as shown in Fig. III.1-2. The density of ions streaming from the central cell through the barrier, where the magnetic field is low, is reduced by the expansion of the flux tube, and by the increase of the parallel velocity due to expanding magnetic field and falling potential. An approximate expression for the passing ion density at the minimum field of the barrier

$$n_b^{\text{passing}} = \frac{n_c}{R_b} \sqrt{\frac{T_{ic}}{\pi\phi_b + T_{ic}}} \quad (\text{III.1-4})$$

if  $\phi_b > T_{ic}$ . Here,  $R_b$  is the barrier mirror ratio,

$$R_b = \frac{B_{mb}}{B_b} .$$

In this expression the distribution function of ions in the central cell is assumed to be Maxwellian and these ions pass through the barrier without collisions. Particle sources localized in velocity space, e.g., due to ionization and charge exchange of the pump beams (discussed in Section III.4.1.4.2), will produce a distortion of the distribution function and therefore modify the passing ion density and the trapping rate in the barrier; these effects are ignored in this discussion. A factor  $g_b$  is introduced to phenomenologically account for ions trapped in the barrier;  $g_b$  is the total ion density/passing ion density and is usually assumed  $\approx 2$ . The electron density is assumed to be a Boltzmann relation with the central temperature,



| Plug | Barrier | Central Cell | Barrier | Plug |

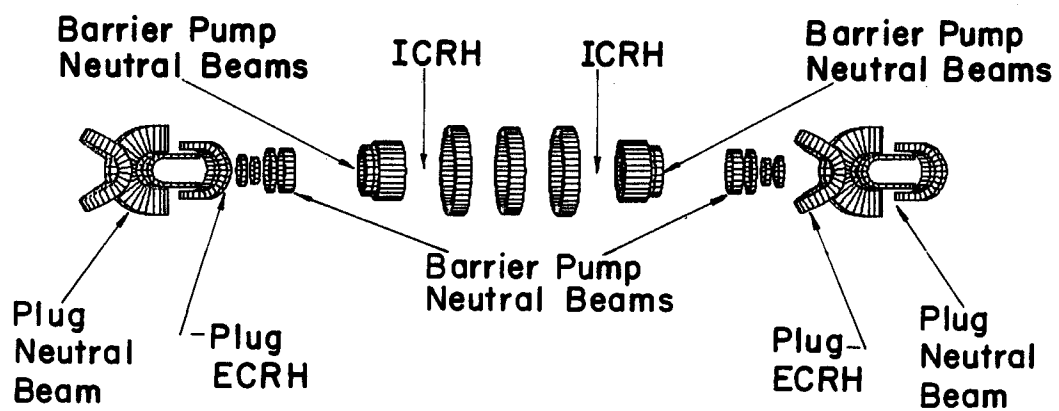


Fig. III.1-2 TASKA magnetic field and potential profiles.  
TASKA coil set also shown.

$T_{ec}$ , and density;

$$n_b = n_c \exp \left( - \frac{\phi_b}{T_{ec}} \right) . \quad (\text{III.1-5})$$

Equating the electron and ion densities determines the barrier potential  $\phi_b$ ,

$$\exp \left( - \phi_b / T_{ec} \right) = \frac{g_b}{R_b} \sqrt{\frac{T_{ic}}{\pi \phi_b + T_{ic}}} . \quad (\text{III.1-6})$$

In Ref. 7 it was assumed that a Boltzmann relation at the plug electron temperature,  $T_{ep}$ , also applied between the barrier magnetic field minimum and the plug. Cohen<sup>(10)</sup> has found from both analytical and Fokker-Planck calculations that a modified Boltzmann relationship applies,

$$\phi_b + \phi_c = T_{ep} \ln \left[ \frac{n_p}{n_b} \left( \frac{T_{ec}}{T_{ep}} \right)^\nu \right] , \quad (\text{III.1-7})$$

where the parameter  $\nu$  is approximately .5 when the plug electrons are confined inside a magnetic mirror but can be larger when the magnetic field is attempting to eject the electrostatically confined plug electrons. The latter case corresponds to the potential peaking at a magnetic hill or in a region of rising magnetic field. The effect of the temperature ratio,  $T_{ec}/T_{ep}$ , in the modified Boltzmann relation is to reduce the benefit of the thermal barrier by reducing  $\phi_c$  for a given density ratio,  $n_p/n_b$ . The modified relationship, Eq. (III.1-5), was used in the WITAMIR-I reactor design;<sup>(8)</sup> the benefits of the thermal barrier are still substantial relative to the standard tandem mirror configuration so that current reactor designs continue to incorporate thermal barriers.

The effectiveness of the thermal barrier can be increased by using microwave power at the minimum field point in the barrier to create a population of hot, magnetically trapped electrons<sup>(11)</sup> in the barrier. For a given total electron density, this reduces the density of Maxwellian electrons. Consequently in Eqs. (III.1-5) and (III.1-7)

$$n_b \rightarrow n_b - n_b^{\text{Hot}},$$

where  $n_b^{\text{Hot}}$  is the density of the magnetically confined electrons. Equations (III.1-6) and (III.1-7) are then modified to become

$$\exp\left(\frac{-\phi_b}{T_{ec}}\right) + \frac{n_b^{\text{Hot}}}{n_c} = \frac{g_b}{R_b} \sqrt{\frac{T_{ic}}{\pi\phi_b + T_{ic}}} \quad (\text{III.1-8})$$

and

$$\phi_b + \phi_c = T_{ep} \ln \left[ \frac{n_p}{n_b - n_b^{\text{Hot}}} \left(\frac{T_{ec}}{T_{ep}}\right)^{\nu} \right] \quad (\text{III.1-9})$$

This increases  $\phi_b$  (deepens the barrier) and increases the effective density ratio in Eq. (III.1-9) for a given plug density, which compensates for the temperature ratio in the argument of the logarithm. Producing a hot, magnetically trapped electron population requires more careful tailoring of the electron distribution function by the use of microwave power than simply heating a Maxwellian distribution. This is an as yet untested concept; there are also questions concerning the penetration and absorption of the microwave power.

Maintaining the thermal barrier requires that ions trapped in the barrier by collisional scattering be "pumped out" in some manner. The LLNL<sup>(9)</sup> and

WITAMIR-I<sup>(8)</sup> designs used neutral beam pumping to remove these ions. The neutral beams injected into the passing ion part of the velocity space "pump" trapped ions by charge exchange between the trapped ions and the neutral atoms. Ionization of these beams also provides refuelling of the central cell plasma. The neutral beams for barrier pumping also contribute to the heating of the central cell. Although the principles of neutral beam pumping of the barrier are based on atomic physics and appear straightforward, there are technological problems with this method of pumping. The neutral beams require considerable amounts of power, and providing space for beam lines at small angles relative to the magnetic field can be difficult. The ion sources need to be shielded from the magnetic field, but they cannot be too far away or choking of the beam duct becomes a problem.

Alternative schemes for pumping thermal barriers have been proposed. Kesner<sup>(12)</sup> suggested using nonaxisymmetric barriers and used the drift orbits of trapped ions to pump them from the barrier. Calculations by Braun and Emmert<sup>(13)</sup> indicate that the electric drift due to the ambipolar potential make the excursion of a trapped ion from a flux surface too small for this effect to be significant. Baldwin<sup>(14)</sup> proposed using AC magnetic fields in the barrier to enhance the radial diffusion of the trapped ions and thereby cause them to be pumped. Hamilton<sup>(15)</sup> proposed DC perturbation fields in the elliptical fan of the plug to cause radial drifts of the barely trapped ions; this could provide some pumping and reduce the requirement for neutral beam pumping. These latter schemes have not yet been sufficiently analyzed to assess their utility.

A variety of alternative plug and thermal barrier configurations have been proposed. The MFTF-B<sup>(11)</sup> configuration uses an outboard "A-cell", as

shown in Fig. III.1-3. The yin-yang minimum-B mirror provides good curvature for MHD stability, while the outboard mirror cell contains the thermal barrier and potential peak for central cell confinement. The barrier and plug can be placed in one mirror cell by injecting a neutral beam at an angle to the magnetic field. The trapping of this beam produces a "sloshing" ion distribution with density peaks near the turning points of the ions. Microwave power is applied at the outer density peak to heat the electrons there and produce the potential peak which confines the central cell ions. Microwave power is also used at the density minimum to produce a population of hot, magnetically confined electrons and thereby deepen the potential dip. This forms the thermal barrier. Ions which trap in the thermal barrier are removed by neutral beam pumping. The minimum-B mirror between the central cell and the A-cell provides the good curvature needed for MHD stability.

The barrier and potential peak can also be located in the minimum-B mirror cell, as shown in Fig. III.1-4, so that an additional mirror cell is not required; this is the TMX-Upgrade configuration.<sup>(16)</sup> In order to obtain good confinement of the sloshing ions, a deeper mirror ratio is required. Providing access for beam injection also complicates the coil configuration.

Both the outboard A-cell and TMX-Upgrade configuration, and the inboard barrier to a lesser extent, suffer from nonaxisymmetric magnetic fields in the transition region and end plug. Nonaxisymmetry causes radial drifts of central cell ions as they approach the plug; this enhances radial transport<sup>(17)</sup> in the central cell. One way of eliminating the radial drift and associated enhanced radial diffusion is to move the thermal barrier and potential peak to an axisymmetric mirror cell between the central cell and minimum-B cell (see Fig. III.1-5). Central cell ions are then reflected by

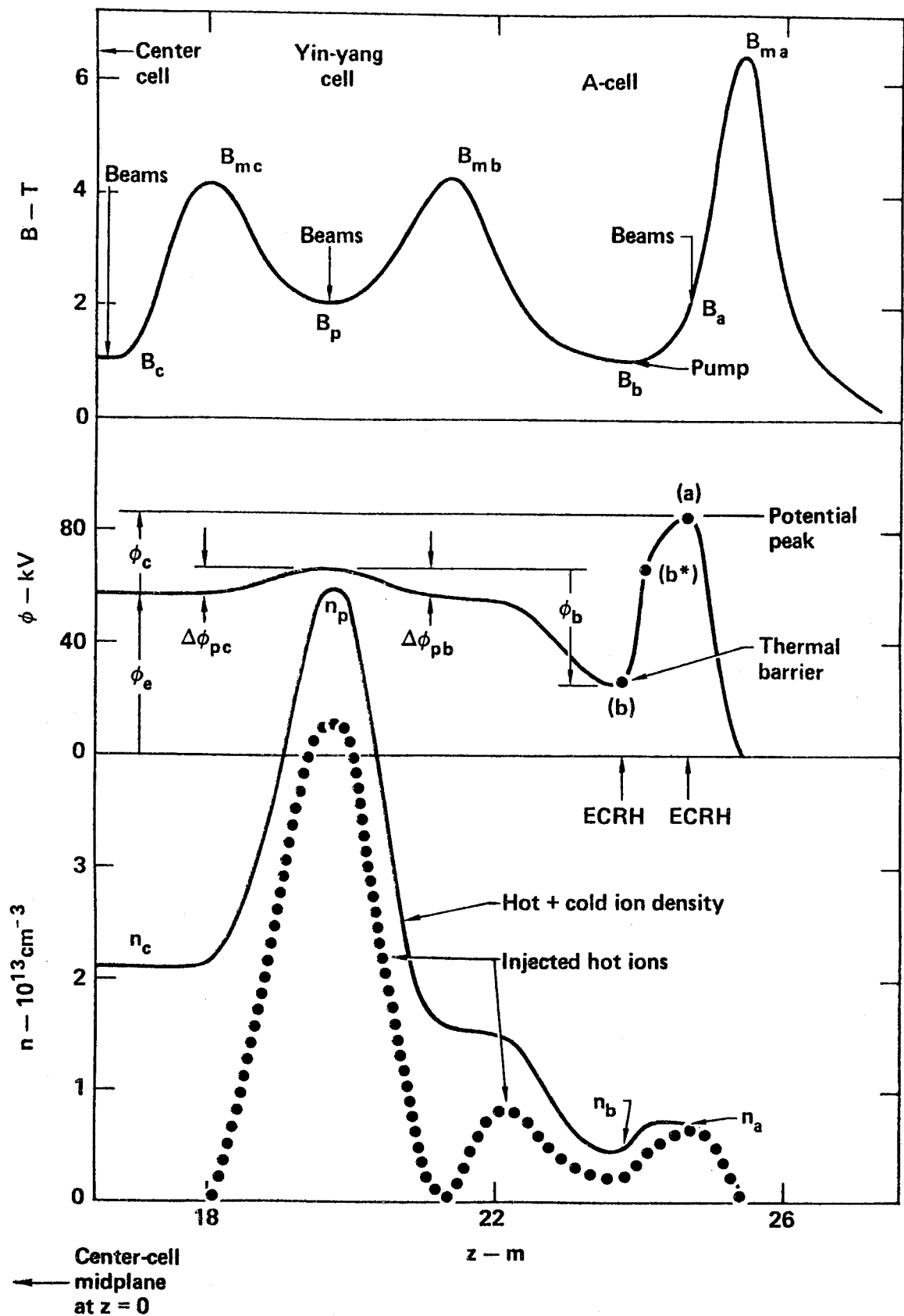


Fig. III.1-3. Profiles of field  $B$ , potential  $\phi$ , and density  $n$  vs. distance  $z$  at one end of MFTF-B operated with A-cell-type thermal barriers. (Taken from Ref. 11)

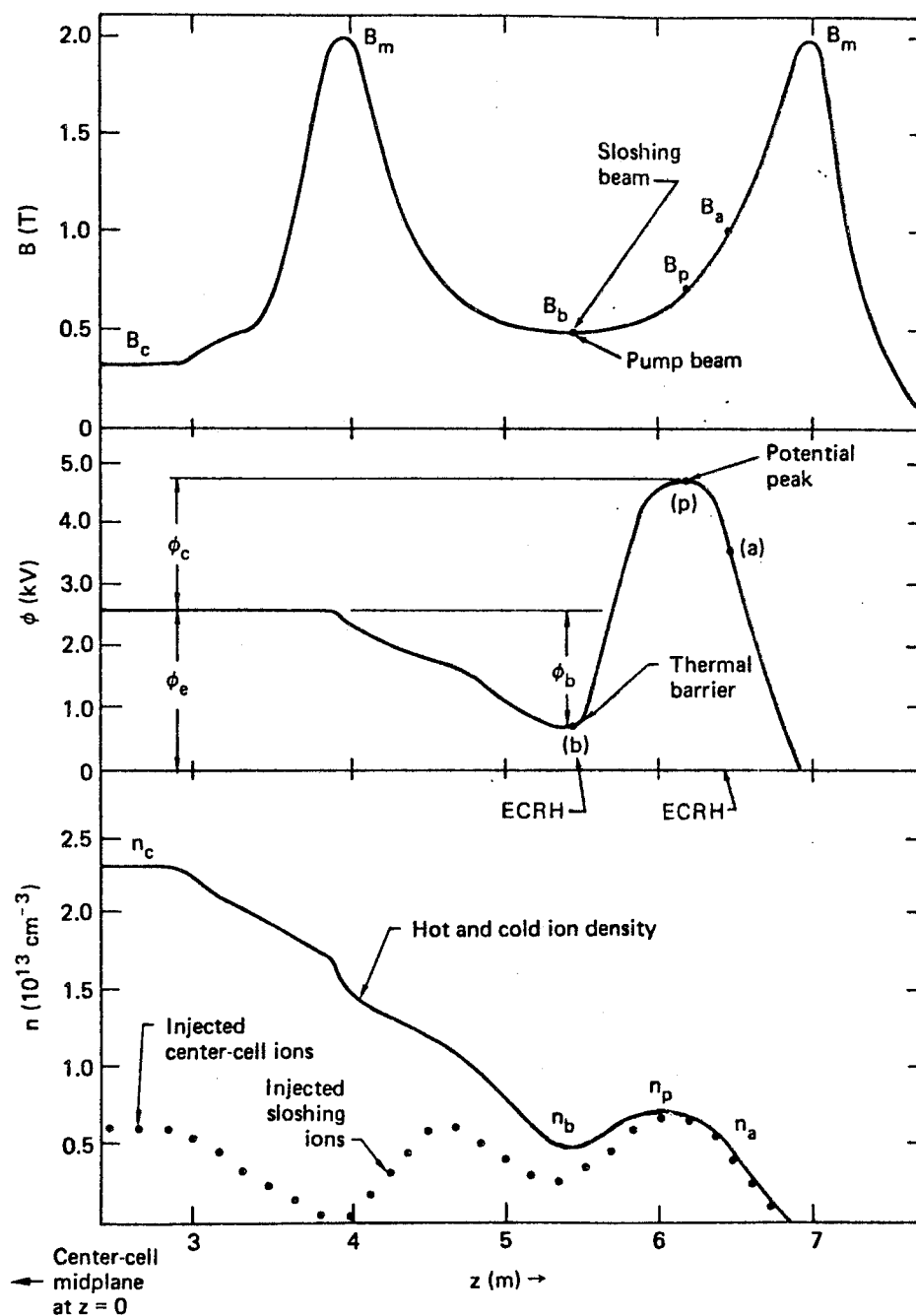


Fig. III.1-4. Profiles of magnetic field  $B$ , potential  $\phi$ , and density  $n$  versus distance  $z$  at one end of TMX Upgrade operated with thermal barriers (Reference case A). The subscripts refer to c, central cell; m, maximum; a and b, points where ECRH is applied; and p, peak. (Taken from Ref. 16)



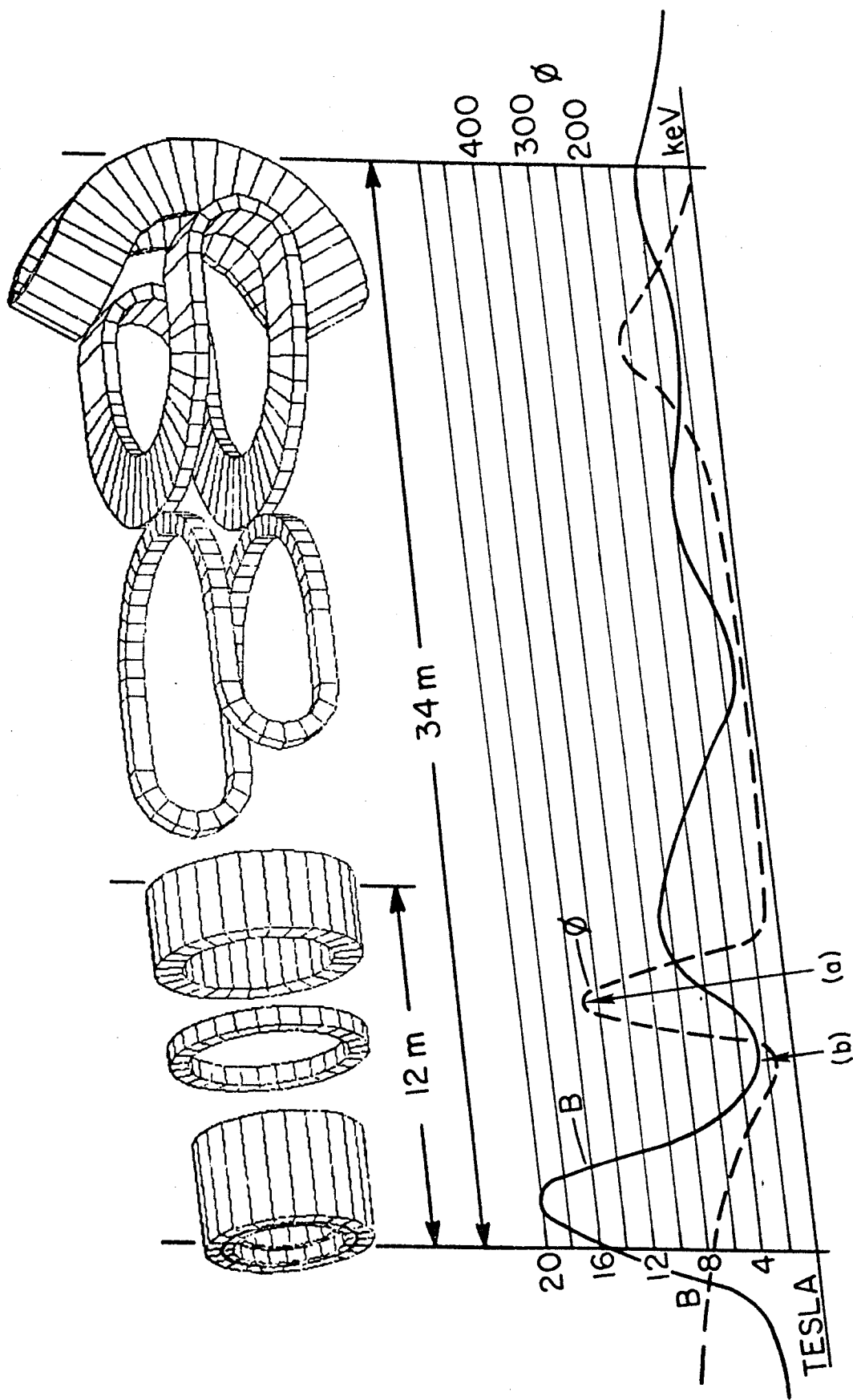


Fig. III.1-5. The magnetic field and potential profile and magnet set for the axi-cell end plug configuration. The central cell is to the left.

the potential peak before they can experience the nonaxisymmetric fields. This is the configuration to be used in the proposed TARA device.<sup>(18)</sup>

Another solution to the enhanced radial transport problem is to make the end plug purely axisymmetric. Good magnetic curvature can then be obtained by using a hot electron ring<sup>(19)</sup> in the end-cell as in EBT, using internal rings,<sup>(20)</sup> or a magnetic cusp configuration.<sup>(21)</sup>

All of these configurations are further removed from standard mirror and tandem mirror experience and their reactor implications are not yet well assessed. There is, however, considerable experience with hot electron rings in the EBT device and predecessor mirror experiments at Oak Ridge and Nagoya.

#### III.1.2 Experimental Results with Tandem Mirrors

Results from three different tandem mirror experiments have been reported to date. These are the GAMMA 6<sup>(22)</sup> device at the University of Tsukuba, Japan, TMX<sup>(23)</sup> at Lawrence Livermore National Laboratory, and Phaedrus<sup>(24)</sup> at the University of Wisconsin. These devices are all of the standard configuration without thermal barriers. Their major design parameters are listed in Table III.1-1.

The first test of the tandem mirror concept was in the GAMMA 6 device,<sup>(22,25)</sup> although the achieved parameters of the experiment are modest ( $T_e \approx 10\text{--}15\text{ eV}$ ,  $n_p \approx 1.7 \times 10^{13}\text{ cm}^{-3}$ ,  $n_c \approx 3 \times 10^{12}\text{ cm}^{-3}$ ), the fundamental concept of a potential peak in the plugs relative to the central cell was established. This potential peak was sustained by neutral beam injection into the plugs and found to be in agreement with the Boltzmann relation. The Pastukhov formula for the end-loss has also been tested and found to be in agreement.

Table III.1-1. Summary of Operating Design Parameters of Tandem Mirror  
Experiments in Operation or Construction (Taken from Ref. 30)

|                                | Gamma 6              | TMX                | Phaedrus           | AMBAL-1            |
|--------------------------------|----------------------|--------------------|--------------------|--------------------|
| Plug:                          |                      |                    |                    |                    |
| $B_0$ (T)                      | 0.4                  | 1.0                | .3                 | 1.2                |
| $B_{\text{mirror}}$ (T)        | 1.0                  | 2.0                | .57                | ---                |
| $R_p$ (cm)                     | 4                    | 10                 | 7                  | 12                 |
| $L_m$ mirror-mirror (cm)       | ---                  | 75                 | 105                | ---                |
| Heating method                 | Beam, RF, REB        | Beam               | ICRF, Beam         | Beam               |
| Heating Power (MW)             | 0.5                  | 7                  | 0.1                | 1.0                |
| Duration (ms)                  | 2.5                  | 25                 | 1                  | ---                |
| $n$ (cm <sup>-3</sup> )        | $5 \times 10^{13}$   | $4 \times 10^{13}$ | $5 \times 10^{12}$ | $3 \times 10^{13}$ |
| $W_i$ (keV)                    | 0.4-10               | 13                 | 2                  | 20                 |
| $T_e$ (eV)                     | 20-2000              | 260 max            | 40                 | 1000               |
| Solenoid:                      |                      |                    |                    |                    |
| $B$ (T)                        | 0.15                 | 0.05-0.2           | 0.05               | 0.2                |
| $L_{\text{plug-to-plug}}$ (cm) | 315                  | 640                | 390                | ---                |
| $R_p$ (cm)                     | $2 \times 20$        | 30                 | 30                 | 30                 |
| $n$ (cm <sup>-3</sup> )        | $1.0 \times 10^{13}$ | $1 \times 10^{13}$ | $2 \times 10^{12}$ | $1 \times 10^{13}$ |
| $W_i$ (eV)                     |                      | 250 max            | 15                 | 500-1000           |

The TMX device, shown schematically in Fig. III.1-6, began operation in July 1979. It is based upon favorable results found earlier with the 2XIIB<sup>(26)</sup> single mirror cell experiment. In 2XIIB, a plasma stream flowing through the mirror cell was found to be effective in stabilizing the drift cyclotron-loss-cone (DCLC) mode. This instability is driven by the loss-cone nature of the ion distribution function in the mirror and can seriously degrade mirror confinement if allowed to be present. TMX was designed so that the plasma stream leaving the central cell and flowing through the end plugs was sufficient to stabilize the DCLC mode.

TMX has also demonstrated the basic principles of tandem mirror confinement. The density in the plug has been maintained higher than in the central cell by neutral beam injection. The potential in the plug is also higher than in the central cell so that the central cell plasma is confined electrostatically by the potential generated by the end plug plasma. A confining potential,  $\phi_c$ , of up to 300 V and plug potentials exceeding 1 kV relative to the end wall have been measured.<sup>(27)</sup> Axial profiles of density and potential are shown in Fig. III.1-6. Axial confinement has also been demonstrated directly by turn-off of one end plug and observing the end-loss and decay. When both end plugs are present, the axial confinement is enhanced of up to a factor of 9 over what it would have been without the plug plasmas. The measured parameters are within a factor of 2 of theoretical calculations if the density ratio,  $n_p/n_c$ , does not exceed 3. In this case the end-loss from the central cell is sufficient to stabilize the ion cyclotron fluctuations in the plugs at a low level. At larger values of  $n_p/n_c$ , RF fluctuations in the plug increase due presumably to microinstabilities and central cell confinement is degraded. Depending on the mode of operation, RF fluctuations in the

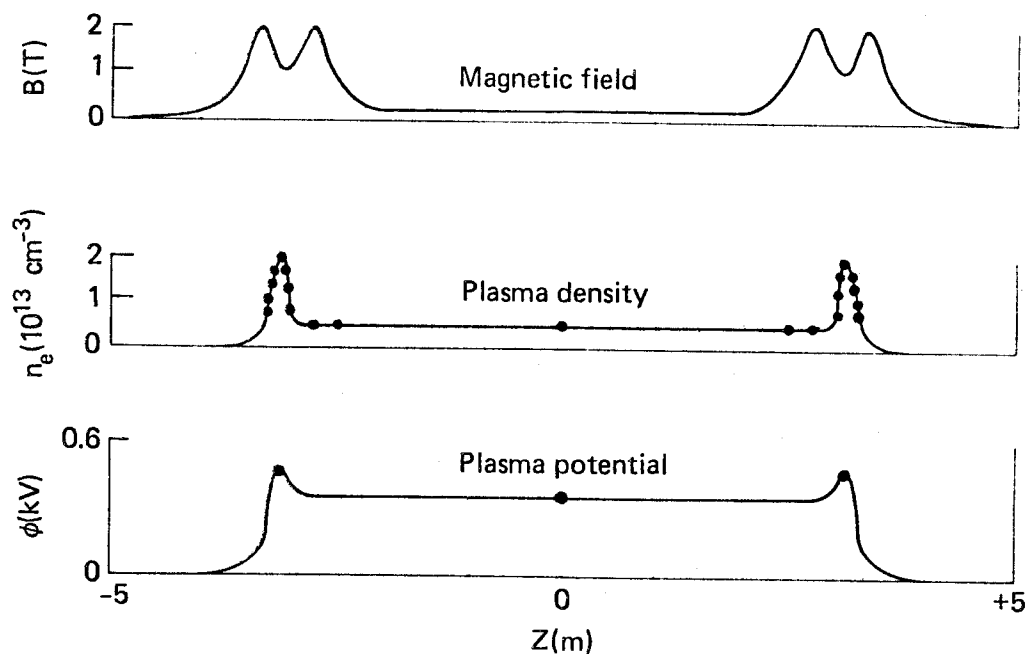
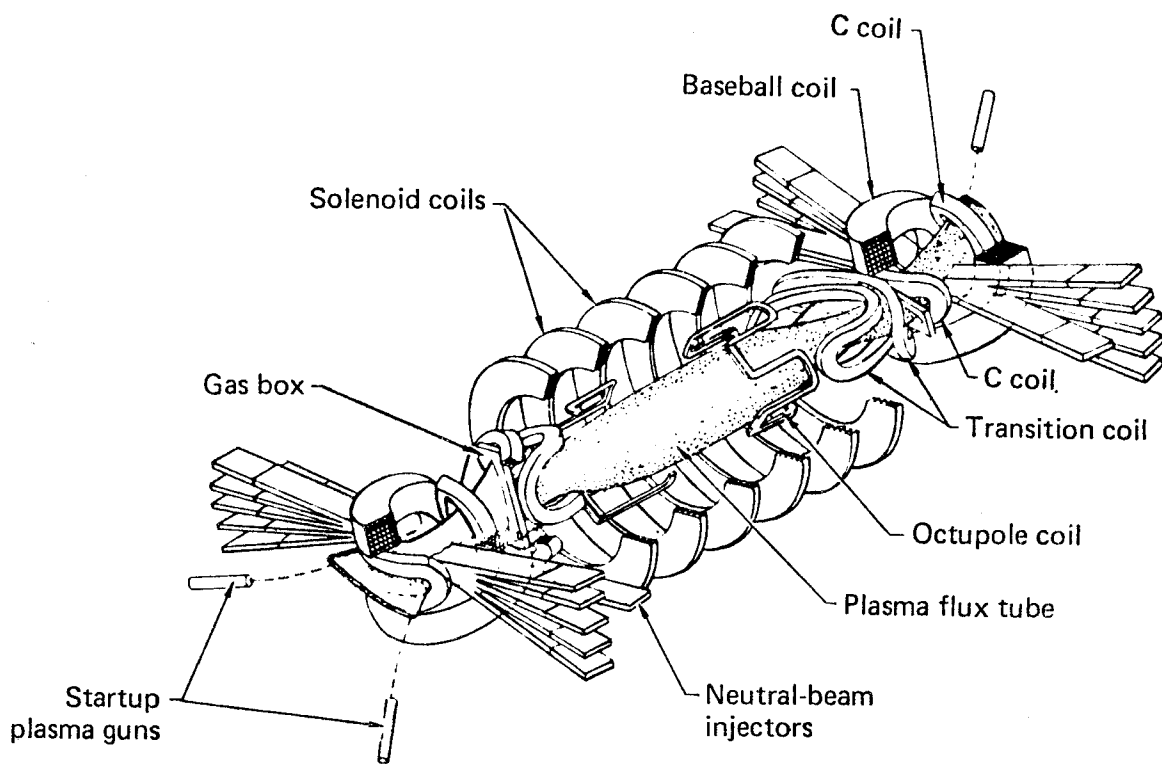


Fig. III.1-6. TMX magnet geometry and measured axial plasma profiles  
(Taken from Ref. 30)

plug have the characteristics of the DCLC or Alfvén-ion-cyclotron (AIC) instabilities. The latter instability is an electromagnetic mode driven by the pressure anisotropy in the plug.

A study of radial transport<sup>(28)</sup> in TMX indicated that the axial loss rate exceeds the radial loss rate in the core ( $r \lesssim 15\text{-}20$  cm) of the central cell, but at larger radii, radial losses exceed axial losses. The radial transport appears to be non-ambipolar with the ion radial flux exceeding the electron radial flux. This results in a net negative electrical current to the end walls; this was observed directly. Within the uncertainty of the data, the radial transport was equal to or greater than the transport predicted by resonant neoclassical theory.

Low frequency oscillations have been observed in the central cell of TMX. Both an  $m = 0$ , 13 kHz and an  $m = 1$ , 7 kHz oscillation have been found and studied. The  $m = 0$  mode appears to be an acoustic wave and is locked to RF "bursting" in the plugs. The  $m = 1$  mode is localized near the edge of the plasma and its amplitude is a function of the method and rate of gas fueling of the central cell. Gas box fueling in the elliptical fans of the plugs produces a larger mode amplitude than puffing into the central cell. For gas box fueling the mode is absent at low gas input rates, coherent at intermediate rates, and perhaps turbulent at high rates. There is some indirect evidence for radial transport caused by the  $m = 1$  mode. Its characteristics have been compared to rotational instabilities, drift waves, and Kelvin-Helmholtz instabilities. None of these adequately describe the observed oscillation. Consequently, it remains unidentified at this time.

The axial energy transport between central cell and plug electrons has been studied in TMX by attempting to assess all the power flows into and out

of the electrons.<sup>(29)</sup> Within the uncertainties of the measurements and the analysis, it was concluded that the energy transfer between the plug and central cell electrons is equal to or less than the classical rate,<sup>(10)</sup> depending on operating conditions. Measurement of the electron temperature,  $T_{ew}$ , near the end wall also indicates reduced energy transport to the end walls. The measured temperature was  $T_{ew} \approx 7-10$  eV when  $T_{ec} \approx 50$  eV. From Monte Carlo calculations and the Pastukhov analysis, one expects  $T_{ew} \approx T_{ec}$ . Good isolation of the confined plasma from the end wall is achieved when the startup guns are turned off. The external plasma density near the end wall is 3 to 4 orders of magnitude less than in the plug.

The maximum plasma parameters achieved in TMX are listed in Table III.1-2. Of particular interest to TASKA is the central cell  $\beta$  of 40%, achieved by neutral beam injection into the central cell. This result is discussed further in Section III.1.3. A summary of TMX results has been published.<sup>(30)</sup> The executive summary of that report is reprinted in Appendix III.1-A.

The Phaedrus experiment has concentrated on ICRF heating in the plug and in the central cell to produce hot ion plasmas.<sup>(31)</sup> Experiments with neutral beam injection into the plugs are in their initial phases. At the fundamental resonance, 50 kW of RF power was coupled to the plug with 30% efficiency. Increase of the mean ion energy to 700 eV at high density ( $10^{13} \text{ cm}^{-3}$ ) or 1 keV at low density ( $10^{12} \text{ cm}^{-3}$ ) was observed.<sup>(32)</sup> Heating at the first harmonic ( $2\omega_{ci}$ ) with 30 kW of power produced a bulk ion energy of 200 eV and a 700 eV hot ion tail. This tail could be localized in pitch angle so as to produce a "sloshing ion" tail distribution.

Experiments with ICRF heating of the central cell plasma have also demonstrated good heating. Central cell plasmas with  $n \approx 10^{12}-10^{13} \text{ cm}^{-3}$ ,

Table III.1-2 Maximum Plasma Parameters in TMX

(Achieved with deuterium and 0.1 T, central-cell magnetic field strength.  
These parameters were not all achieved simultaneously.) (Taken from Ref. 30.)

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|                                                                            |                                          |
|----------------------------------------------------------------------------|------------------------------------------|
| Plug density                                                               | $4 \times 10^{13} \text{ cm}^{-3}$       |
| Plug ion energy                                                            | 13 keV                                   |
| Plug electron temperature                                                  | 0.26 keV                                 |
| Plug radius                                                                | 10 cm                                    |
| Central-cell density                                                       | $3 \times 10^{13} \text{ cm}^{-3}$       |
| Central-cell ion energy                                                    | 0.25 keV                                 |
| Central-cell radius                                                        | 30 cm                                    |
| Plug plasma potential                                                      | 1 keV                                    |
| Central-cell confining potential                                           | 0.3 keV                                  |
| Central-cell axial confinement parameter                                   | $10^{11} \text{ cm}^{-3} \cdot \text{s}$ |
| Electrostatic enhancement in confinement                                   | 9                                        |
| Central-cell beta<br>(0.07 without central-cell<br>neutral beam injection) | 0.40                                     |



$T_i \approx 300\text{-}100\text{ eV}$  and  $\beta_c \approx 10\text{-}20\%$  have been produced. These ICRF heated plasmas have been useful for the investigation of the MHD stability characteristics of Phaedrus. Flute-like oscillations propagating in the ion diamagnetic direction with a dominant  $m = 1$  azimuthal mode number are observed at high beta. The threshold central cell beta,  $\beta_c$ , is 3-10 times the plug beta,  $\beta_p$ , and decreases with reduced plug quadrupole field. The observed threshold value of  $\beta_c/\beta_p$  is 2-5 times the theoretical value from ideal MHD theory. Line tying plays a role at the higher values of  $\beta_c/\beta_p$ , but is not believed to be significant at the lower values.

The rate of energy transfer between plug electrons and central cell electrons in a thermal barrier tandem mirror is an important element in determining the effectiveness of thermal barriers. This problem has been analyzed by Cohen.<sup>(10)</sup> The theoretical result was tested in Phaedrus by operating the central cell at low density relative to the plugs. The electron temperature in one plug was changed rapidly with a short high power microwave pulse. The rise of the electron temperature in the central cell and in the other plug was then monitored. The results are in reasonable agreement with point model theoretical calculations using the theoretical energy transfer ratio.<sup>(10)</sup> As indicated above, TMX showed less than the theoretical energy transfer under some conditions.

### III.1.3 Future Major Tandem Mirror Experiments

Three major tandem mirror experiments are under construction and should produce results relevant to TASKA. These are TMX-Upgrade,<sup>(16)</sup> which is expected to be operational at LLNL in early 1982, GAMMA 10,<sup>(33)</sup> which is to be operational in 1983 at the University of Tsukuba, Japan, and MFTF-B,<sup>(11)</sup> which

is to begin operation in 1985. The design parameters for these devices are shown in Table III.1-3.

The plug configuration of TMX-Upgrade was discussed in Section III.1.1 and is shown in Fig. III.1-4. This device is expected to provide a test of several physics concepts important to tandem mirrors. Among these are the thermal barriers and the use of ECRH to increase the barrier potential, ECRH heating of plug electrons, the creation of a sloshing ion distribution in the plug, and a test of the microstability of sloshing ion distributions. This concept is discussed further in Section III.1.4. In addition, TMX-Upgrade is expected to improve the performance of TMX by about an order of magnitude and to operate in the collisionless central cell regime appropriate to MFTF-B and reactors. Because of its lower central cell temperatures, TMX operated on the borderline between the collisional and collisionless regimes.

GAMMA 10 is a successor to the GAMMA 6 device in Japan. Current planning calls for it to operate in three different phases. Phase I utilizes a standard tandem mirror configuration with a minimum-B mirror as the end plug. Phase II is similar to MFTF-B in that it uses a mirror cell outboard of the minimum-B mirror. In GAMMA 10, this mirror is axisymmetric, however. The outboard simple mirror can be used to study sloshing ion distributions and thermal barriers with ECRH heating. Phase III will utilize an end plug which has only an axisymmetric simple mirror with hot electron rings to provide MHD stability. The parameters shown in Table III.1-3 are for Phase II operation.

MFTF-B started as a large single yin-yang mirror experiment, MFTF, which was approved and construction of the yin-yang magnet was begun. Following the successful results with TMX and the invention of the thermal barrier, it was modified to become a tandem mirror with an outboard A-cell thermal barrier.

Table III.1-3. Summary of Design Parameters of Major Future  
Tandem Mirror Experiments

|                                               | <u>TMX-Upgrade</u>     | <u>GAMMA 10</u>  | <u>MFTF-B</u>          |
|-----------------------------------------------|------------------------|------------------|------------------------|
| Begin Operation                               | 1982                   | 1983             | 1985                   |
| <u>Central Cell</u>                           |                        |                  |                        |
| Density (cm <sup>-3</sup> )                   | 2.3 x 10 <sup>13</sup> | 10 <sup>13</sup> | 2.1 x 10 <sup>13</sup> |
| Ion Temperature (keV)                         | .9                     | 1                | 15                     |
| Electron Temperature (keV)                    | .6                     | 1                | 9                      |
| Beta                                          | .25 <sup>(4)</sup>     | .05              | .2                     |
| Axial Confinement, nτ (sec/cm <sup>3</sup> )  | 10 <sup>12</sup>       | 10 <sup>12</sup> | 5 x 10 <sup>13</sup>   |
| Radial Confinement, nτ (sec/cm <sup>3</sup> ) | 5 x 10 <sup>11</sup>   | ---              | ---                    |
| Magnetic Field (T)                            | .3                     | .4               | 1                      |
| Radius (cm)                                   | 20                     | 26               | 54                     |
| Length (m)                                    | 8                      | 30               | 32                     |
| <u>Plug (or A-cell)</u>                       |                        |                  |                        |
| Density <sup>(1)</sup> (cm <sup>-3</sup> )    | 7 x 10 <sup>12</sup>   | 10 <sup>13</sup> | 7 x 10 <sup>12</sup>   |
| Mean Ion Energy <sup>(2)</sup> (keV)          | 10                     | 15               | 84                     |
| Electron Temperature <sup>(3)</sup> (keV)     | 1.4                    | 3                | 30                     |
| Magnetic Field (T)                            | 2 - 4                  | .5 - 3           | 1 - 6.5                |

1. In A-cell if used.

2. Hottest component present in yin-yang or A-cell.

3. Warm component if magnetically trapped barrier electrons are used.

4. Includes hot ion component.

It utilizes, however, the original MFTF yin-yang magnet; this placed substantial constraints on the design of the end plugs. As with other devices, this machine can be tested in various modes of operation. The parameters given in Table III.1-3 are the reference case with thermal barriers. For this case the primary research objectives are:

1. Generate central cell ion confining potentials,  $\phi_c$ , via electron heating and ion pumping in A-cells at densities lower than the central cell density.
2. Increase central cell electrostatic confinement ( $n\tau$ ) and ion temperature sufficient to test radial transport in the appropriate collisionless reactor regimes.
3. Confine sufficient plasma pressure in the central cell to test MHD beta limits at relatively high central cell magnetic fields ( $B_c \gtrsim B_p/2$ ).
4. Maintain steady-state density depression with sloshing ions in the A-cell midplane to test cold-ion stabilization of loss-cone modes at high potential.

Theoretical analysis of the expected sloshing-ion distribution function in the A-cell of MFTF-B indicates that all known mirror-related microinstabilities should be stabilized. This is discussed further in Section III.1.4. The relevant parameters for MFTF-B are expected to be about an order of magnitude more advanced than in TMX-Upgrade.

In addition to these devices, an "axisymmetric" tandem mirror, called TARA,<sup>(18)</sup> is under consideration for construction at MIT. As discussed in Section III.1.1, its end plug configuration places the potential peak and thermal barrier in an axisymmetric mirror cell between the central cell and

the MHD anchor, which is a minimum-B cell. This eliminates the enhanced ion radial diffusion associated with the quadrupole fields in the anchor. TARA will provide a test of this concept and of the microstability of sloshing ion distributions in the plug, along with a test of the thermal barrier concept and the use of ECRH to heat the electrons and increase the barrier potential.

The projected central cell ion temperature and ion confinement parameter,  $n\tau$ , are shown in Fig. III.1-7 for these three devices along with experimental results from GAMMA 6, Phaedrus, and TMX, and the projected values for TASKA. This graph is often used as a simplified way of presenting progress and distance to go to reach the objective of a thermonuclear fusion reactor; it should be remembered, however, that this type of graph ignores a lot of important physics questions and, if taken out of context, can give an erroneous impression of progress, or lack thereof, in fusion physics.

#### III.1.4 Physics Issues Related to TASKA

##### III.1.4.1 Plug Configuration

The TASKA plug configuration, shown in Fig. III.1-2, consists of an inside thermal barrier between the central cell and the minimum-B mirror, which provides good curvature for MHD stability and contains the ambipolar potential peak for confinement of central cell ions. In order to provide microstability in the hot, mirror-confined ion plug, a sloshing ion distribution with trapped warm plasma is used. A detailed schematic of the end plug is shown in Fig. II.4-2 and the plasma parameters are given in Table III.4-3.

This configuration has a number of advantages and disadvantages relative to possible alternatives. Of all the configurations proposed, it is closest to current tandem mirror experiments in the plug configuration. Consequently, there is a somewhat better experimental foundation for the physics analysis.

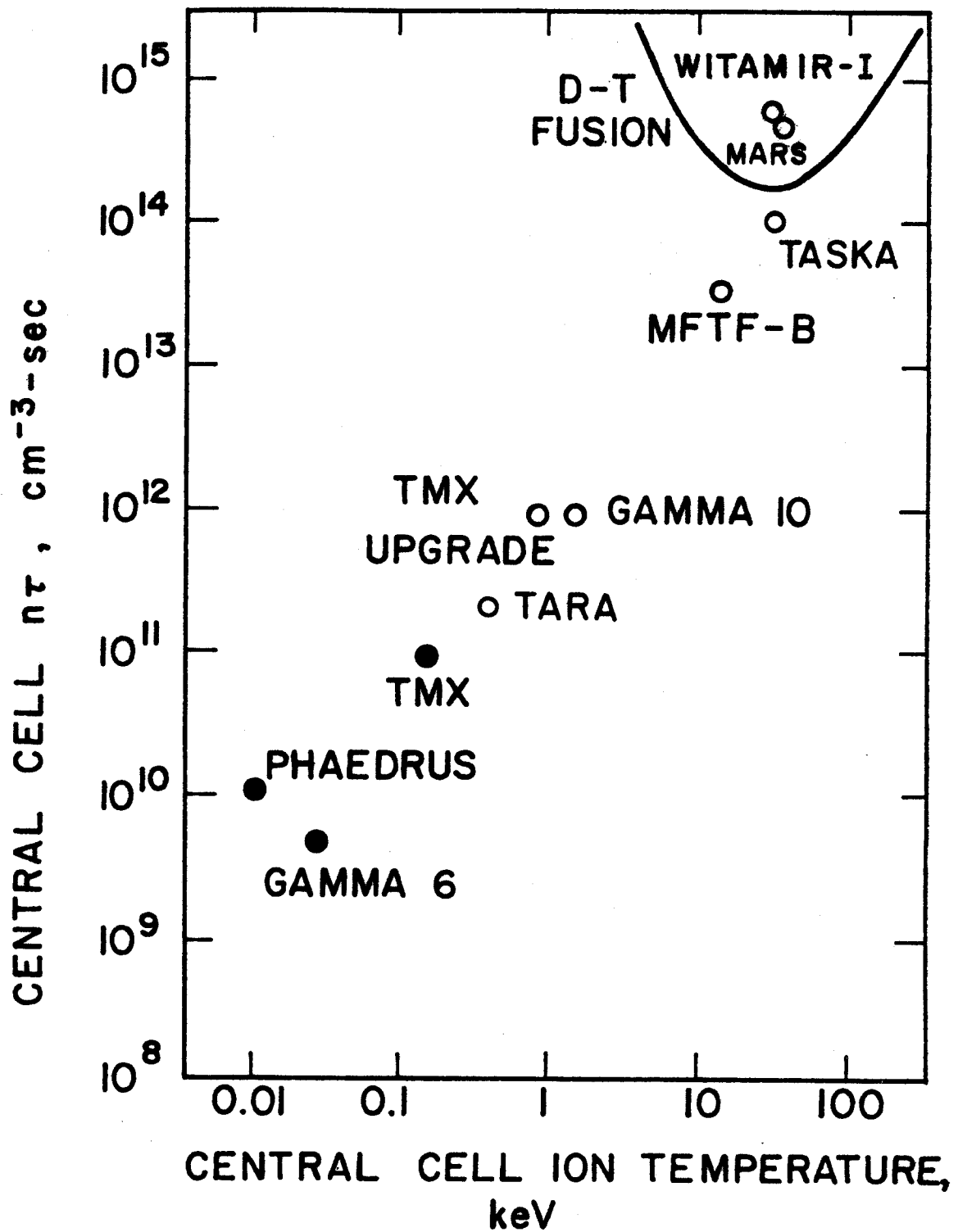


Fig. III.1-7.  $n\tau$  versus central cell ion temperature for present (●) and future (○) tandem mirrors.

In addition, less subtle plasma physics is required in this configuration than in the more advanced configurations, TMX-Upgrade and MFTF-B. As a result, the probability of the plug operating as planned should be increased. An example of simpler plasma physics is the decision to not incorporate hot, magnetically trapped electrons in the barrier to further depress the potential there. This would improve the performance of TASKA but is an as yet untested concept. Since the performance objectives of TASKA can be achieved without it, it was decided to design TASKA without this improvement. An experimental test of the use of hot, magnetically trapped electrons in the barrier will be made in TMX-Upgrade and MFTF-B. If successful, this concept could be incorporated into an upgraded version of TASKA.

The high magnetic field at the barrier throat in TASKA produces a number of desirable effects. The large central cell mirror ratio,  $B_{mb}/B_C$ , reflects a larger fraction of ions and electrons approaching the barrier. Consequently fewer central cell ions experience the nonaxisymmetric fields in the transition region and plug; this reduces the central cell radial diffusion caused by radial drifts in these nonaxisymmetric fields. The larger central cell mirror ratio also improves the confinement of unthermalized alpha particles in the central cell by decreasing their loss-cone for axial escape. The larger local barrier mirror ratio,  $B_{mb}/B$ , reduces the plasma pressure in the bad curvature regions in the barrier and transition region between the barrier and minimum-B plug. This reduced pressure decreases the driving term for MHD instabilities.

The axisymmetric plug-barrier configuration used in TARA<sup>(18)</sup> would have even better properties relative to radial transport but its MHD characteristics are not as good, since the high pressure associated with the potential

peak is in a region of bad curvature. It also requires the use of hot, magnetically trapped electrons to generate the thermal barrier. Completely axisymmetric configurations utilizing electron rings<sup>(19)</sup> seem too untested to be incorporated in TASKA at this time. Axisymmetric configurations with internal rings<sup>(20)</sup> present technological questions which have not yet been fully analyzed.

The inside barrier configuration with a high field barrier coil also has a neutronics advantage. The plug plasma can be nonfusion-reacting (e.g., protons). This, coupled with the low density of deuterium and tritium in the barrier and transition region, has the effect of reducing substantially the neutron production rate in the end plug and thereby reduces the shielding requirements for the yin-yang and transition coils and reduces neutron streaming into the direct convertor and down beam lines and other penetrations of the end plug and barrier shields.

#### III.1.4.2 Thermal Barrier Pumping

Pumping of the trapped ions in the thermal barrier is done in TASKA by charge exchange pumping using neutral beam injection, as discussed in Section III.1.1. This method is chosen because it is based on relatively straightforward atomic physics considerations. Although it appears to be manageable from a technological viewpoint, there are distinct disadvantages with charge exchange pumping. The associated hardware is expensive and is still in a stage of development. Present limits on beam current density and divergence dictate a minimum size for the plasma in the barrier, which then determines the plasma radius in the central cell. A larger neutron wall loading can be obtained, however, by operating with a smaller central cell radius for a given input power. Furthermore the space requirements for the beam sources,



magnetic shielding, neutralizers, beam lines, and vacuum pumping equipment conflict with other requirements.

It would be highly advantageous to utilize alternative barrier pumping schemes. Some alternatives have been proposed; these are discussed in Section III.1.1. They generally operate by removing trapped ions from the system, instead of "pumping" them back into the central cell. Rather than being an energy input, these schemes drain energy from the plasma so that they affect the stationary operating point in a thermal power balance sense. The hardware associated with them, however, ought to be minimal in comparison with neutral beam pumping. Since these alternative schemes are not yet thoroughly analyzed and are totally untested in experiments, we decided to design TASKA with neutral beam pumping. If these schemes prove successful, they could be implemented in TASKA with only a minor redesign. The converse, however, is not true. Designing for an alternative pumping scheme and having to shift to neutral beam pumping would require a major redesign of the machine.

#### III.1.4.3 MHD Stability

It has been recognized since about 1979 that ideal MHD stability theory places a severe constraint on the allowable beta in the central cell. The limiting instability is thought to be the ballooning mode driven by bad curvature. In the inboard thermal barrier configuration this occurs in the transition region between the thermal barrier and the minimum-B plug, in the thermal barrier itself, and in the connection between the thermal barrier and the central cell. Predicted ideal MHD beta limits are in the range of 10-25%.<sup>(11)</sup> The most unstable modes have high azimuthal mode number,  $n$ . The central cell, however, is a region with large volume, zero curvature, and relatively large gyroradius  $\rho$  in relation to the radial scale length,  $r_c$ . In TASKA, for

example,  $r_c = 32$  cm,  $\rho \approx 2$  cm. One might expect a substantial finite gyro-radius correction to ideal MHD stability theory in these circumstances. Recent analysis<sup>(34,35)</sup> confirms this result. A model analytical calculation gave an increase in the limiting central cell beta from 27%, at the zero gyro-radius limit at high  $n$ , to  $\beta > 1$  for all  $n > 5$ .<sup>(35)</sup> In this calculation, the finite gyroradius parameter  $\Gamma$ ,

$$\Gamma \approx n \left( \frac{L}{r_c} \right) \left( \frac{\rho}{r_c} \right)$$

was equal to .5. Since  $L$  (the parallel scale length)  $\gg r_c$ , this corresponds to rather modest values of  $\rho/r_c$ .

Finite gyroradius stabilization is most effective at high mode number,  $n$ , and vanishes for the  $n = 1$  mode if one uses a sharp boundary model for the plasma. Allowing for diffuse radial profiles will give some finite gyroradius stabilization at  $n = 1$ , but this may not be sufficient. Wall stabilization is most effective, however, for low  $n$ ,<sup>(36,37)</sup> so this may provide the needed stability for small values of  $n$ , including  $n = 1$ . In TASKA, the wall to plasma radius is only 1.4, so significant wall effects can be expected.

The calculations in Refs. 33 and 34 use model magnetic fields; the necessary revisions to MHD codes which calculate stability with real magnetic fields are under development at LLNL but not yet operational. Consequently, one has to assume a beta value for the design of TASKA. The choice of 50% for the central cell beta seems entirely reasonable in light of these calculations.<sup>(35,36)</sup>

Experience with tokamaks<sup>(38)</sup> and multipoles<sup>(39)</sup> also supports this conclusion. Both have operated at beta values in excess of theoretical ideal MHD

beta limits for ballooning modes. Experimental encouragement is also found in the Phaedrus tandem mirror experiment, which has exceeded the theoretical interchange stability limits.

The interchange instability and instabilities driven by plasma rotation give much higher beta limits<sup>(11)</sup> even without finite gyroradius corrections, so they are not a design consideration.

#### III.1.4.4 Plug Microstability

The minimum-B plug has to contain a hot ion, mirror-confined plasma. The distribution function of these hot ions will be definitely non-Maxwellian and hence will be a source of free energy to drive plasma instabilities. These instabilities will have the effect of seriously degrading the confinement time (by orders of magnitude) of the hot ions and hence must be avoided. The most dangerous of these modes are currently believed to be:<sup>(11)</sup>

1. The drift cyclotron loss-cone mode (DCLC).
2. The Alfvén ion cyclotron mode (AIC).
3. The axial loss-cone mode (ALC), which is the absolute version of the convective loss-cone mode.
4. The negative energy wave (NEW).

To date, the DCLC mode was identified as the dominant mode in the 2XIIB<sup>(26)</sup> experiment and believed to be one of the modes seen in TMX.<sup>(28)</sup> The AIC mode should have been unstable in 2XIIB but was not observed. It has been tentatively identified in TMX, however.<sup>(30)</sup> The ALC model should have been the dominant mode<sup>(11)</sup> in the PR-6 experiment<sup>(40)</sup> in the U.S.S.R., but the instability seen there was identified as the DCLC mode. The NEW mode has not been seen experimentally.

The linear theory of these modes, and the experimental evidence for them, is reviewed in Ref. 41.

The DCLC mode, which was the major consideration in the 2XIIB experiment and in the design of TMX can be stabilized by a sufficient amount of warm (unconfined) plasma in the mirror. In 2XIIB, this was produced by a plasma streaming through the mirror; this plasma stream was produced by external plasma guns or by ionization in a gas box. This method is undesirable in tandem mirrors because it keeps the electron temperature, and therefore the confining potential, low. TMX was designed so that the classical loss rate of central cell plasma was sufficient to provide enough plasma stream to stabilize the DCLC mode. This was only partially successful in actual practice.<sup>(30)</sup> For hotter plasmas, however, this method is inadequate since the loss rate goes down as the temperature goes up. Hence reactor relevant plasmas need to obtain stability in other ways.

The end plug in the WITAMIR-I reactor design<sup>(8)</sup> was calculated to be stable against the DCLC mode because of the combination of low end plug density, high beta, and large plasma radius. This, however, does not give stability against the other modes, which can exist in an infinite plasma or depend on the axial variation of the density and magnetic field.

TMX-Upgrade and MFTF-B are designed to avoid these instabilities by the use of a sloshing ion distribution in the end plug (or A-cell). This sloshing ion distribution, which is produced by injection of a neutral beam at an angle to the magnetic field (see Fig. III.1-8), causes the hot ion density to peak off the midplane, as shown in Fig. III.1-9. Corresponding to this density peak, the potential will also peak off the midplane. The local potential "valley" created in the plug can then trap low-energy ions which can partially

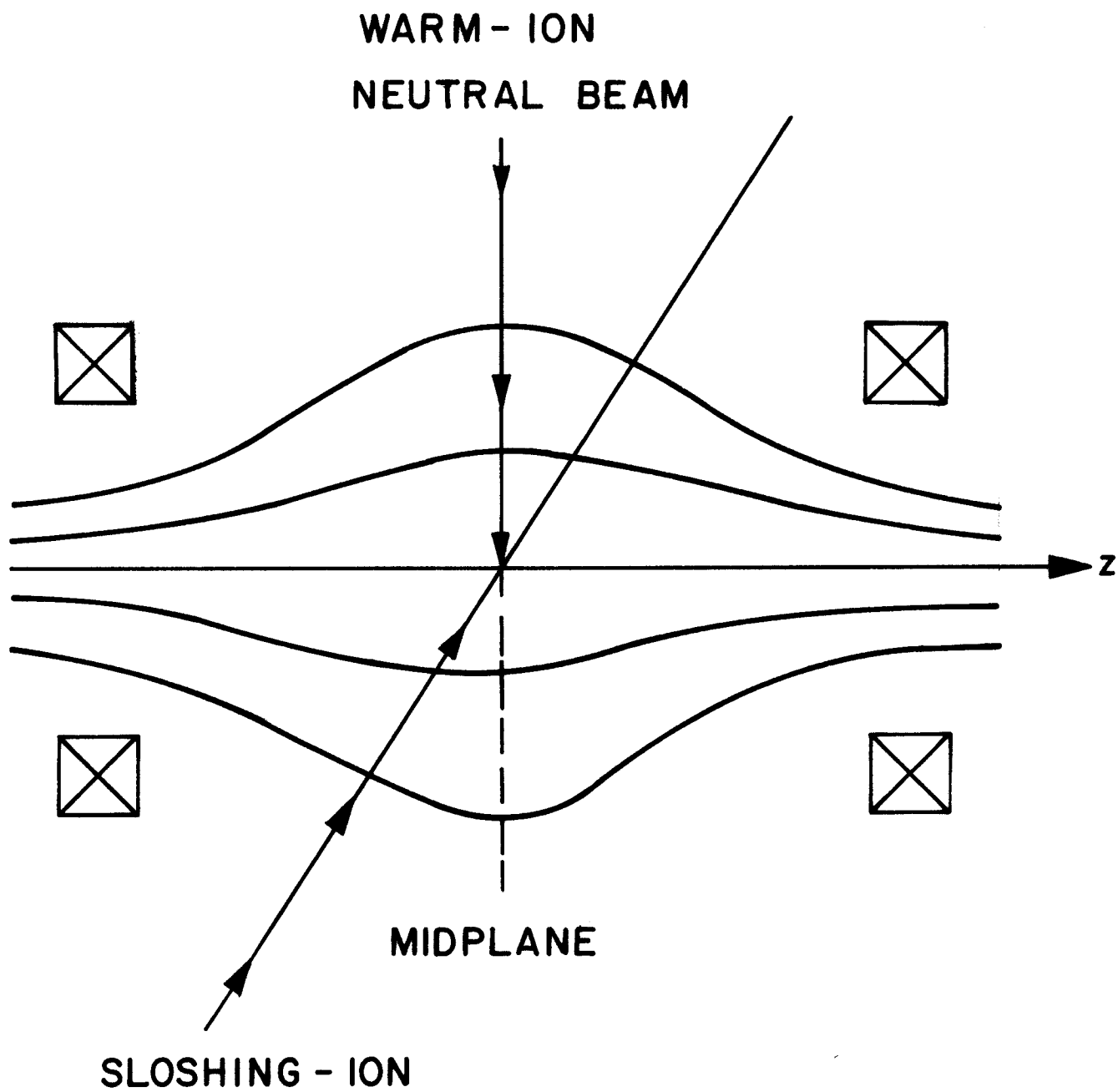


Fig. III.1-8. Neutral beam configuration for a sloshing ion mirror cell.

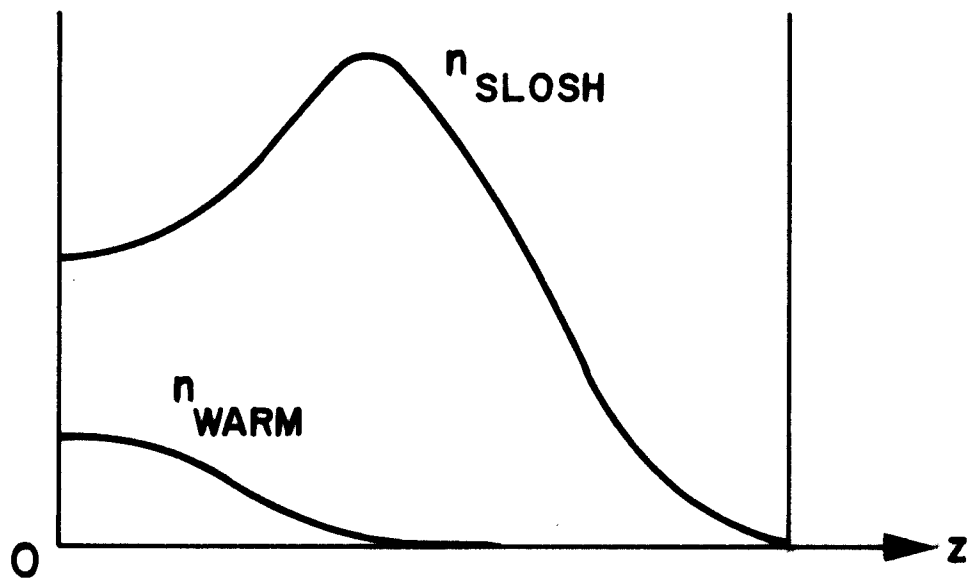
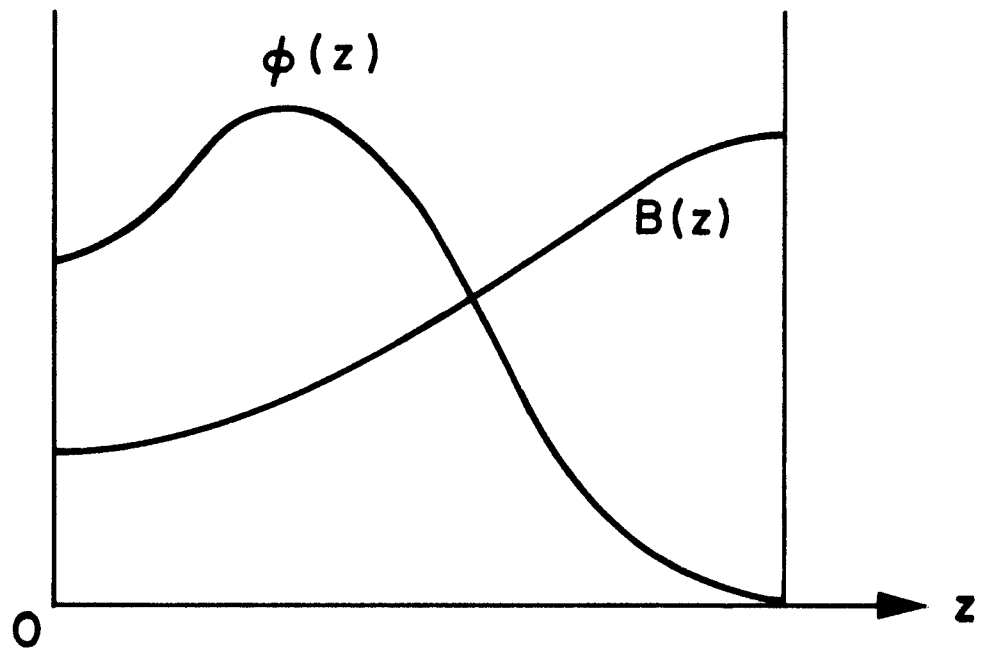


Fig. III.1-9. Axial profiles of density, potential, and magnetic field in a sloshing ion mirror cell.

fill in the hole in the ion distribution function at low energy. This is shown schematically in Fig. III.1-10. The central cell plasma streaming through the sloshing ion plasma also helps to fill in the ion distribution. In this manner, one obtains stability against the DCLC mode. The AIC mode is also stabilized by the increase in  $T_{\parallel}/T_{\perp}$  associated with the sloshing ion distribution and by reduced ion beta. The warm plasma coupled with the finite length along the magnetic field and the low density can also provide stability against the ALC and NEW modes, as well. As a result, both MFTF-B and TMX-Upgrade are expected to be stable against all of these modes.<sup>(42)</sup>

The design philosophy in TASKA is to utilize the sloshing ion beam concept in the end plug to provide stability against at least the DCLC and AIC modes, since these have been seen experimentally. Ignoring the other modes may be dangerous if they occur at a sufficient level to degrade plug ion confinement. On the other hand, designing TASKA to satisfy all existing theoretical stability criteria is difficult and may be overly restrictive if the modes turn out to be figments of the theorists' imagination. Given the weakness of the theory (both linear and non-linear) and the lack of experimental corroboration, we have chosen to put more priority on the DCLC and AIC modes. There is some precedence for this philosophy in the tokamak community. The trapped particle instabilities, and the dissipative trapped ion mode in particular, were used as design criteria for tokamak reactors until PLT<sup>(43)</sup> with beam heating produced plasmas which should have been unstable to the trapped ion mode, but only low level fluctuations were seen and plasma confinement was not significantly degraded. Currently, trapped particle instabilities are ignored in the design of tokamak reactors, although there is no theoretical justification for this, other than the realization that all theoretical models

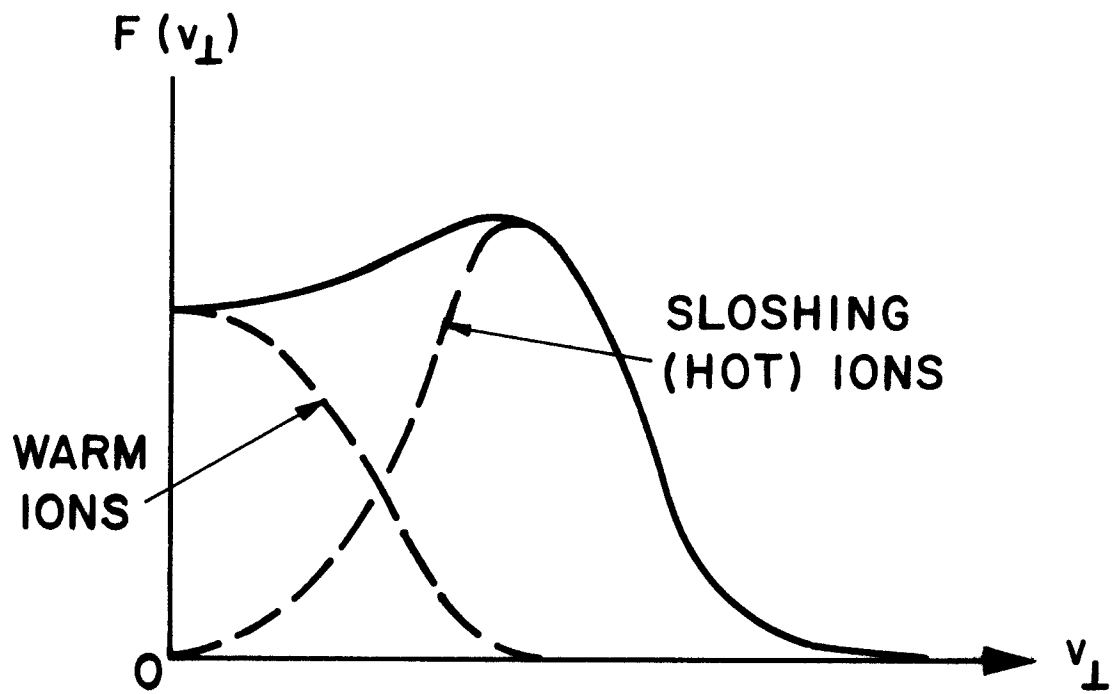


Fig. III.1-10. Ion distribution function;  $f(v_{\perp}) = \int f(\vec{v}) dv_{\parallel}$ .



are only an approximation to reality. On the other hand, it should be noted that near-classical plug ion confinement is required in tandem mirrors, while the radial transport in toroidal devices can be much larger in relation to the neo-classical rate and still be acceptable.

#### III.1.4.5 Radial Transport

In a tandem mirror central cell the problem of radial transport is of special importance because longitudinal losses are greatly reduced by the ambipolar potential of the end plugs. It was recognized by Ryutov and Stupakov<sup>(44)</sup> that the nonaxisymmetric fields of the end plugs can cause radial drifts of the central cell ions so that their drift surfaces are no longer circular. In most tandem mirror designs, the quadrupole fields at opposite ends are rotated by  $90^\circ$  relative to each other. This produces a lowest order cancellation<sup>(45)</sup> but higher order deviations of the drift surfaces from circular remain.

An important parameter determining the character of the radial diffusion is the azimuthal drift,  $\Delta\psi$ , of an ion in one axial transit of the central cell. This azimuthal drift arises from the grad-B drift in the central cell, the  $\vec{E} \times \vec{B}$  drift in the central cell, and azimuthal drifts in the end plugs. Three modes of diffusion have been identified, depending on the magnitude of  $\Delta\psi$ . If  $\Delta\psi \ll 1$ , one gets enhanced diffusion due to the step size per collision being determined by the ion stepping from one drift surface to another. This is called neoclassical diffusion and is analogous to neoclassical diffusion in toroidal devices. If  $\Delta\psi \approx 1$  then certain classes of particles in phase space experience a resonance with the radial drifts. Those particles for which  $\Delta\psi$  is an odd multiple of  $\pi/2$  experience the same sign of the radial drift at each end of the central cell and can have large radial excursions.

This is known as resonant diffusion. The third mode occurs when  $\Delta\psi \gg 1$  and  $a \frac{\partial}{\partial r} \Delta\psi \gtrsim 1$ . The parameter  $a$  is the maximum radial displacement of an ion due to a single reflection in the end plug. In this case the diffusion is called "stochastic", in analogy to nonlinear oscillation theory. Of course, the ions have different  $\Delta\psi$  corresponding to their velocity, so that a sum over these "modes" is needed to get the net particle transport rate.

Elaborate theoretical calculations of the transport coefficients resulting from these effects have been made.<sup>(44,46)</sup> They are highly non-linear, however, with a complicated dependence on the plasma properties, including the radial electric field. Furthermore, the radial transport is non-ambipolar and therefore plays a role in determining the electric field. Consequently, one needs self-consistent profiles of  $n$ ,  $T_i$ ,  $T_e$ , and  $\phi$  to get an accurate calculation of the radial transport.

A qualitative picture of the diffusion coefficient for these different modes is shown in Fig. III.1-11. Here  $r_c$  is the central cell radius and  $t_{dr}$  is the time required for an ion to drift azimuthally by  $2\pi$  in the central cell,

$$t_{dr} \approx \frac{r_c^2}{2 \rho_c^2 \Omega_c},$$

where  $\rho_c$  is the ion gyroradius and  $\Omega_c$  is the ion cyclotron frequency in the central cell. The small parameter  $\alpha$  is a measure of the noncircularity of the drift surfaces. An order of magnitude estimate<sup>(11)</sup> for  $\alpha$  is

$$\alpha \sim \frac{r_c^4}{L_c L_t^3},$$

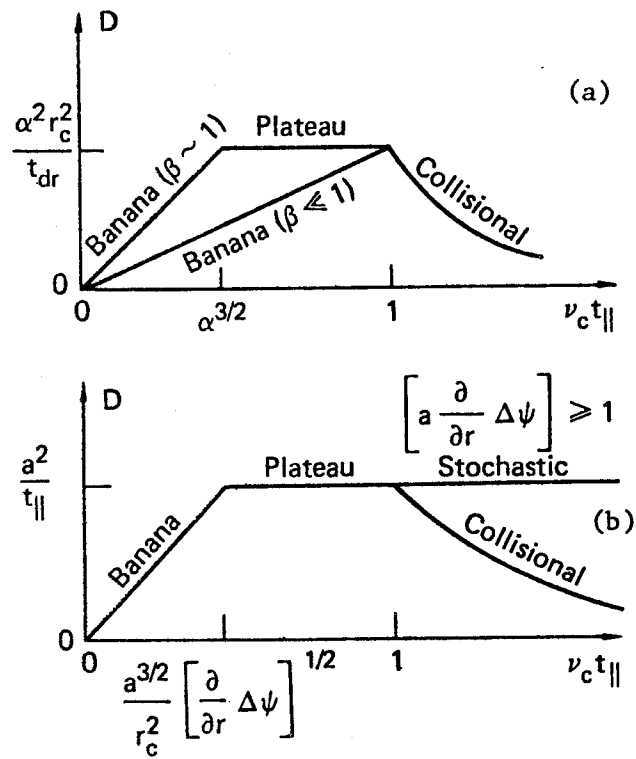


Fig. III.1-11. (a) Neoclassical diffusion and (b) Resonant diffusion.  
(Taken from Ref. 11)

where  $L_c$  is the central cell length and  $L_t$  is the transition length for the quadrupole magnetic field.. The longitudinal transit time is  $t_{\parallel}$ ,  $t_{\parallel} = L_c/v_{\parallel}$ , and  $\nu_c$  is the ion-ion collision frequency. An order of magnitude estimate of  $a$  is<sup>(11)</sup>

$$a \approx \frac{\rho_c r_c}{L_t} .$$

The significance of radial transport in TASKA can be determined using these order of magnitude estimates. For the TASKA design parameters, the appropriate diffusion regime is at the border between banana and plateau diffusion for resonant neoclassical diffusion. The estimated radial confinement time is  $\approx 1.4$  sec, which is to be compared with the axial confinement time of .3 sec. Furthermore, this is most likely a low estimate of the radial confinement time. The circular barrier magnet in TASKA produces an axisymmetric mirror field which reflects most central cell ions before they experience the nonaxisymmetric fields in the plug. This should reduce the radial transport (this effect has been seen in detailed MFTF-B calculations<sup>(11)</sup>) but no credit was given for this in the above estimate. We conclude from this, that radial transport in TASKA may play a role, but should not seriously affect the design operating point.

#### III.1.4.6 Alpha and Impurity Accumulation

The potential peak in the plugs confines higher-Z ions, such as fusion-born alpha particles and impurities, even better than the D-T fuel. Consequently one can get accumulation of alphas and impurities in the central cell. One of the advantages of single mirrors, namely that the positive potential expels impurities from the plasma, has been lost. The steady-state level of

these  $Z > 1$  ions is determined by a balance between sources and losses. In this section we consider only alpha particles, which are an irreducible minimum impurity source. The same considerations apply to other impurities as well.

The fusion produced alpha particles are born at 3.5 MeV energy. A small fraction of these are born inside the loss-cone of the high barrier field,  $B_{mb}$ , at the entrance to the barrier (see Fig. III.1-2) and escape immediately. The remaining alphas are trapped by the central cell mirror ratio and slow down in the plasma, first by electron drag, and later by ion-alpha collisions. Some of these alphas will scatter into the loss-cone during the slowing-down process and be lost. The remaining alphas scatter down in energy and become trapped by the electrostatic potential of the plugs. They can be lost by radial transport or by upscattering in energy to above the confining potential as they attempt to establish a Maxwell-Boltzmann distribution.

A rough estimate of the alpha accumulation level in the central cell can be obtained by using the Pastukhov formula to estimate the axial loss rate of alpha particles by upscattering in energy. The source rate is determined by the fusion power density. One gets

$$\frac{n_{\alpha}}{n_c} \approx \frac{f_b}{5} e^{\phi_c/T_{ic}} . \quad (\text{III.1-10})$$

In this result, the alphas have been assumed to thermalize to the central cell ion temperature,  $T_{ic}$ . The parameter  $f_b$  is the fractional burnup (ion "burned" per ion injected), the factor 5 arises from the higher collisionality of the alphas ( $Z = 2$ , mass = 4 amu). For TASKA, the fractional burnup,  $f_b$ , is low, and  $\phi_c/T_{ic} \approx 1.5$  so that  $n_{\alpha}/n_c$  is on the order of 1%. This steady-state alpha

accumulation level will have only a minor effect on the performance. For a reactor, however,  $f_b \approx 20\%$  and  $\phi_c/T_{ic} \sim 3$  so that alpha accumulation to the level given by Eq. (III.1-10) is unacceptable. Alpha accumulation in the thermal barrier has a serious effect at lower levels. If the alpha particles are not pumped out of the barrier, then the alpha density can be expected to satisfy a Boltzmann relation between the central cell and the barrier minimum;

$$n_{\alpha}^b = n_{\alpha}^{cc} e^{2\phi_b/T^*}$$

where  $T^*$  is some effective parallel temperature for alpha particles in the barrier and  $n_{\alpha}^b$ ,  $n_{\alpha}^{cc}$  are the alpha densities in the barrier and central cell, respectively. One might take  $T^* = T_{ic}$ , which is equivalent to assuming the alphas have established a Maxwellian distribution at the ion temperature,  $T_{ic}$ . This would occur primarily by alpha-ion collisions. The ions in the barrier, however, are essentially two interpenetrating streams with temperature  $T_{ic}$  and streaming energy  $\phi_b$ . The streaming motion should cause an increase in the mean alpha parallel energy in the barrier. Consequently, another possible choice is  $T^* = 2\phi_b$ . For both of these choices  $n_{\alpha}$  (barrier)  $\gtrsim n_{\alpha}$  (cc). The ion density in the barrier,  $n_b$ , is much less than  $n_c$ , and the total ion density, after summing over species, needs to be kept small in order to maintain the potential dip of the barrier. In thermal barrier tandem mirrors, the first effect of alpha particles is to cause "quenching" of the thermal barrier. To avoid quenching of the barrier, one wants  $n_{\alpha}^b \ll n_b$ , which is a more stringent requirement than  $n_{\alpha}^{cc} \ll n_c$ .

In TASKA the resulting alpha density in the barrier would be  $n_{\alpha}^b/n_b \approx 100$  or  $n_{\alpha}^b/n_b \approx 1$  depending on the choice for  $T^*$ . Either case is

unacceptable so some form of alpha particle removal from the barrier is needed. Since alpha particles get trapped in the barrier by pitch angle scattering, it would seem that classical cross-field diffusion in the barrier is too slow to keep  $n_{\alpha}^b$  small. An active means of removing alpha particles from the barrier is required. An early proposal<sup>(47)</sup> for removing alpha particles from standard tandem mirrors was to drop the potential in one end plug and allow some fraction of the central cell plasma to flow out; the potential peak is then re-established and the central cell plasma is brought back to its original operating point by fueling and heating. This technique will work in principle but the investment in energy in each cycle to re-establish the plasma is substantial and the degradation of Q is severe.<sup>(47)</sup> Using this procedure in a thermal barrier tandem mirror would be even more undesirable since it is only the barrier that needs to be purged, but this purging technique drains the entire central cell as well.

An alternative scheme for pumping alpha particles has been proposed by Hamilton and Logan.<sup>(48)</sup> This method uses the neutral beams which pump the barrier to convert  $\text{He}^{+2}$  to  $\text{He}^{+1}$  by charge exchange. If this occurs at the right point in the potential profile the kinetic energy acquired by the alpha particle is sufficient to allow it to escape through the end plug. In order that the alpha particle escape before re-ionization to  $\text{He}^{+2}$ , the neutral beam has to be directed towards the end plug. Estimates indicate that this can be a viable scheme for removing alpha particles in a reactor.

The same process can also be used to pump barrier trapped alpha particles back into the central cell. Figure III.1-12 shows the velocity space for alphas at the bottom of the barrier. Trapped  $\text{He}^{+2}$  ions in the cross-hatched region become untrapped and return to the central cell if they undergo charge

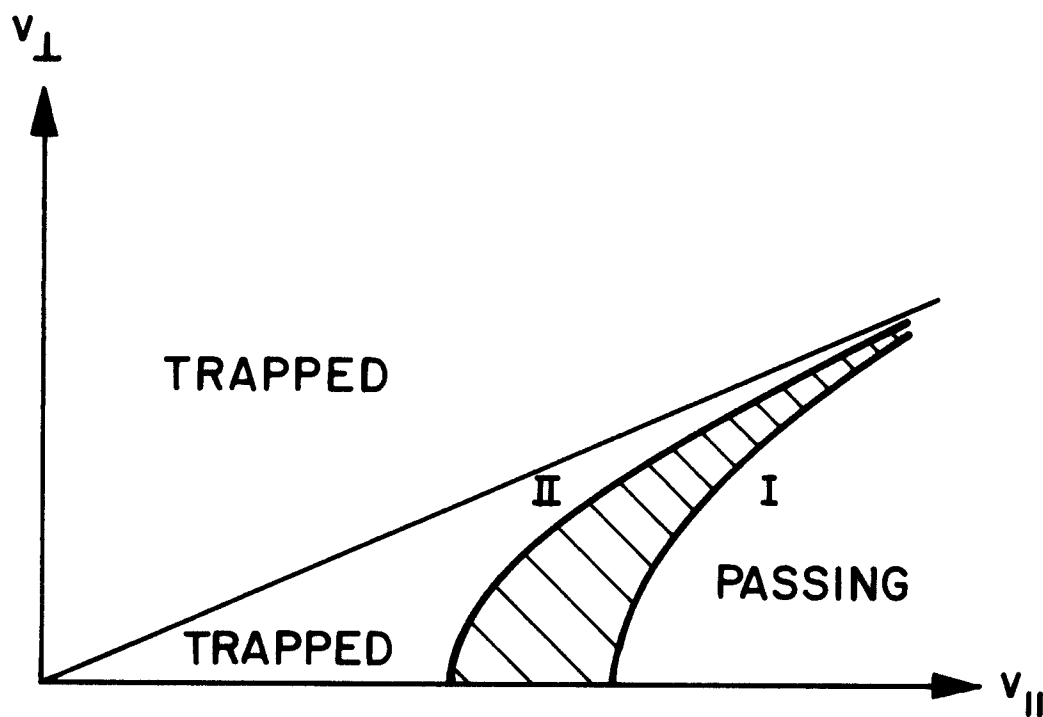


Fig. III.1-12. Velocity space for alpha particles at the barrier minimum. Curve I is the trapped-passing boundary for  $\text{He}^{+2}$ ; curve II is for  $\text{He}^{+1}$ . Charge exchange ( $\text{He}^{+2} \rightarrow \text{He}^{+1}$ ) causes alpha particles in the cross-hatched region to be pumped into the central cell. The diagram is symmetric in  $v_{||}$ .



exchange to the +1 state. Alpha particles more deeply trapped than the cross-hatched area cannot be pumped by a single charge exchange event. All alpha particles, in scattering from passing to trapped in the barrier, pass through the cross-hatched region in Fig. III.1-12 and hence have a finite probability for being pumped. Estimates of the magnitude of this effect for alpha particle removal from the thermal barrier have not yet been made.

The alternative barrier pumping schemes described in Section III.1.4.2 remove barrier trapped ions by enhanced radial loss; they would appear to be equally effective for removing alpha particles as well from the barrier. Consequently, if these methods prove suitable for barrier pumping in TASKA, then they may also solve the problem of alpha particle accumulation in the barrier.

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### Appendix III.1-A

Reprint of the executive summary of UCRL-53120, "Summary of TMX Results,"  
T.C. Simonen, editor.

# **Summary of TMX Results Executive Summary**

**TMX Group**  
**T. C. Simonen, Editor**

**Manuscript date: February 26, 1981**

**LAWRENCE LIVERMORE LABORATORY**  
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# Summary of TMX Results

## Executive Summary

### INTRODUCTION

This report summarizes results from the successful experimental operation of the Tandem Mirror Experiment (TMX) over the period October 1978 through September 1980. The experimental program, summarized by the DOE milestones given in Table 1, had three basic phases: (1) an 8-month checkout period, October 1978 through May 1979, (2) a 6-month initial period of operation, June through November 1979, during which the basic principles of the tandem configuration were demonstrated (i.e., plasma confinement was improved over that of a single-cell mirror), and (3) a 10-month period, December 1979 through September 1980, during which the initial TMX results were corroborated by additional diagnostic measurements and many detailed physics investigations

were carried out. This report summarizes the early results, presents results of recent data analysis, and outlines areas of ongoing research and data analysis which will be reported in future journal publications.

The TMX experiments demonstrated the fundamental tandem mirror principles, as summarized in Table 2. Table 3 lists the maximum plasma parameters achieved in TMX. The main result was that TMX generated electrostatic confining potentials that significantly improved central-cell plasma confinement. These data established a new scaling of ion confinement by ambipolar potential in magnetic mirror systems and provided the impetus for the initiation of both TMX Upgrade, a tandem mirror in which potential confinement is increased through the use of thermal barriers, and MFTF-B, a larger tandem mirror that will extend the TMX Upgrade results to thermonuclear temperatures.

TABLE 1. Summary of TMX milestones.

| Milestones                                                                                                        | Date achieved | Reference                                 |
|-------------------------------------------------------------------------------------------------------------------|---------------|-------------------------------------------|
| 1. Begin TMX checkout.                                                                                            | Oct 1978      | Direct communication to DOE               |
| 2. Begin plasma-buildup experiments.                                                                              | Jul 1979      | Direct communication to DOE               |
| 3. Determine density and beta of plug and solenoid.                                                               | Sep 1979      | Direct communication to DOE               |
| 4. Demonstrate electrostatic plugging of solenoid ions in a measured well.                                        | Sep 1979      | Direct communication to DOE               |
| 5. Submit draft report evaluating initial TMX performance.                                                        | Oct 1979      | <i>Phys. Rev. Letters</i> 44, 1132 (1980) |
| 6. Begin neutral-beam heating experiments in solenoid with two beams.                                             | Nov 1979      | Section III.E <sup>a</sup>                |
| 7. Submit final report evaluating initial TMX performance.                                                        | Jan 1980      | UCID-18496                                |
| 8. Submit a plan for modification of TMX, based on TMX data, that will address issues most appropriate to MFTF-B. | Jan 1980      | Mirror Senior Review Panel                |
| 9. Begin plug-optimization experiments.                                                                           | May 1980      | Section III.G <sup>a</sup>                |
| 10. Submit report on measurements of initial electron-beam experiments in TMX.                                    | May 1980      | Section III.G, <sup>a</sup> UCID-18725    |
| 11. Submit report evaluating impurities in TMX including types and origins of impurities.                         | Oct 1980      | Section III.I, <sup>a</sup> UCID-18883    |
| 12. Obtain Thomson-scattering measurements of TMX solenoid electron temperature.                                  | Sep 1980      | Section III.A <sup>a</sup>                |
| 13. Initiate procurement of hardware to improve TMX performance.                                                  | Aug 1980      | Initiation of TMX Upgrade construction    |
| 14. Submit report summarizing TMX results.                                                                        | Feb 1981      | UCRL-53120                                |
| 15. Submit report on radial transport of plasma in the solenoid of TMX.                                           | Feb 1981      | Section III.F <sup>a</sup>                |
| 16. Submit report on initial TMX central-cell ICRH experiments.                                                   | Dec 1980      | UCID-18866                                |

<sup>a</sup>Section of this report (UCRL-53120).



**TABLE 2. Summary of TMX results.**

|                                                                             |  |
|-----------------------------------------------------------------------------|--|
| ● <b>Generated tandem mirror configuration:</b>                             |  |
| —Configuration sustained for full 25-ms shot duration.                      |  |
| —Plug microstability maintained with solenoid outflow.                      |  |
| —Central-cell MHD stability (40% maximum beta with neutral-beam injection). |  |
| ● <b>Demonstrated central-cell electrostatic plugging:</b>                  |  |
| —Measured electrostatic potential well.                                     |  |
| —Direct evidence by measurements made when one end plug was turned off.     |  |
| —Measured factor-of-9 electrostatic enhancement.                            |  |
| —Radial confinement exceeds axial confinement.                              |  |
| ● <b>Improved electron confinement:</b>                                     |  |
| —Electron temperature higher than in 2XIIB.                                 |  |
| —Low density at end wall.                                                   |  |
| —Dominant power loss to end walls.                                          |  |
| —Low levels of impurities.                                                  |  |

**TABLE 3. Maximum plasma parameters achieved in TMX with deuterium and a central-cell magnetic field strength of 0.1 T. These parameters were not achieved simultaneously on the same shot.**

|                                                                      |                                          |
|----------------------------------------------------------------------|------------------------------------------|
| Plug density                                                         | $4 \times 10^{13} \text{ cm}^{-3}$       |
| Plug ion energy                                                      | 13 keV                                   |
| Plug electron temperature                                            | 0.26 keV                                 |
| Plug radius                                                          | 10 cm                                    |
| Central-cell density                                                 | $3 \times 10^{13} \text{ cm}^{-3}$       |
| Central-cell ion energy                                              | 0.25 keV                                 |
| Central-cell radius                                                  | 30 cm                                    |
| Plug plasma potential                                                | 1 kV                                     |
| Central-cell confining potential                                     | 0.3 kV                                   |
| Central-cell axial-confinement parameter                             | $10^{11} \text{ cm}^{-3} \cdot \text{s}$ |
| Electrostatic enhancement in confinement                             | 9                                        |
| Central-cell beta (0.07 without central-cell neutral-beam injection) | 0.40                                     |

## INITIAL TMX RESULTS

### Tandem Mirror Configuration

● One of the expected tandem mirror characteristics verified by TMX was that the density of the plasma in the end plug could be sustained at a higher level than that of the central-cell plasma, and that this produced higher electric potentials in the end plugs than in the central cell. These densities were controlled by varying the end-plug neutral-beam current and the central-cell gas-feed current. The density peaks generate potential peaks, as shown in Fig. 1, which also shows that relatively small end-plug plasmas can electrostatically confine a much larger central-cell plasma.

● The density and temperature of the TMX plasma are within a factor of 2 of those predicted by theoretical codes. Calorimeter measurements showed that most of the neutral-beam power deposited on the axis is carried to the end walls by ions.

● Gross MHD stability and microstability were achieved. Finite-beta plasma was confined in the central cell with minimum-B end-plugs. The outflow of central-cell plasma provided end-plug microstability.

### Improved Plasma Confinement

● The TMX end plugs improve confinement of the central-cell plasma by up to a factor of 9 over

that which would have been attained if the end-plug plasmas had not been present. Typical enhancements were in the range of 3 to 7 times. Central-cell axial confinement of ions is near that predicted by our theoretical models.

● TMX plasma confinement can be explained by classical Coulomb theoretical models, over a certain range of parameters. However, there must be sufficient warm plasma to stabilize the plugs. If insufficient low-energy plasma flows through the end plug, then fluctuating electric fields develop at the end-plug ion-cyclotron frequency and confinement of central-cell ions is reduced. Theoretical models describing TMX performance over a wide range of operation have been developed on the basis of these experiments.

● A second measure of improvement in the tandem mirror over the single mirror is the electron temperature that can be achieved with a given amount of input neutral-beam power. TMX end plugs achieved electron temperatures up to 260 eV, three to four times higher than the electron temperatures of the similar single-cell mirror machine 2XIIB when operated with comparable neutral-beam input power. Since TMX has such a large central-cell plasma, this electron temperature increase indicates a hundred-fold improvement in electron energy confinement. This improvement arises from the fact that the low-energy plasma required for end-plug microstability is supplied from

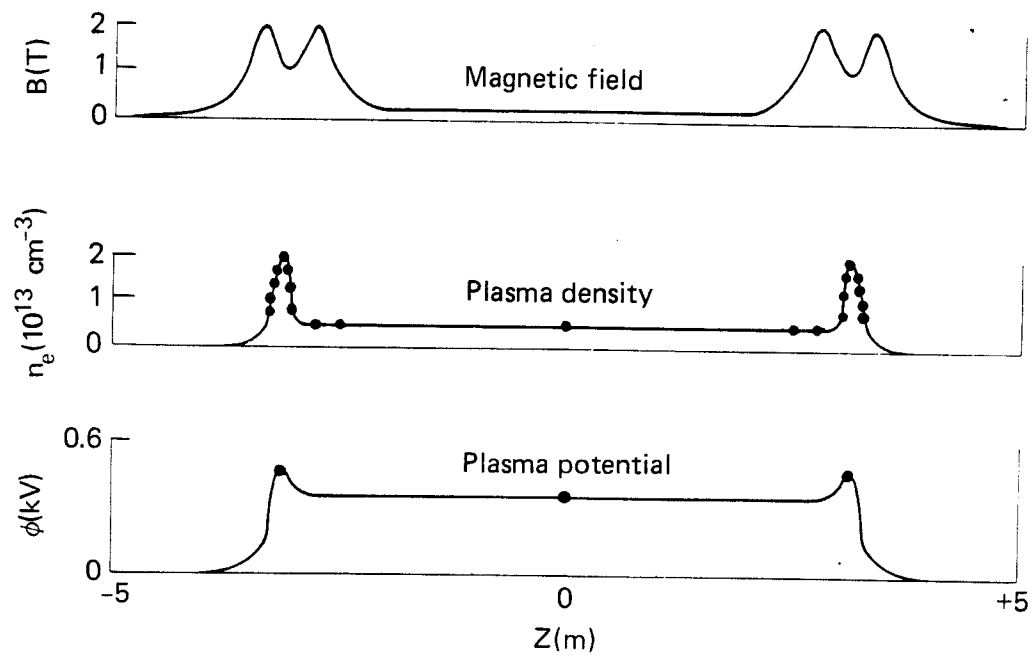
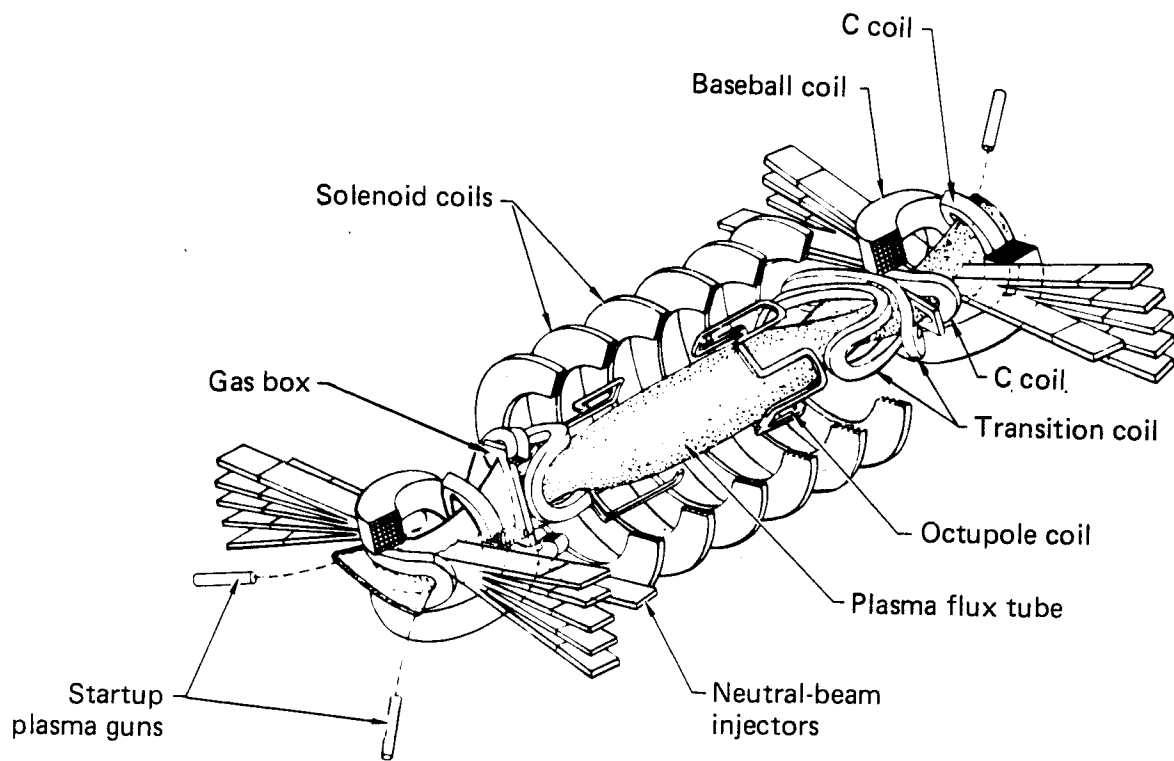


FIG. 1. TMX magnet geometry and measured axial plasma profiles.

the TMX central cell rather than from the ends, as in 2XIIB, thus reducing the electron energy loss to the end walls.

## NEW RESULTS PRESENTED IN THIS REPORT

During much of the last period of TMX operation (December 1979 through September 1980), poor vacuum conditions caused lower electron temperatures than had been achieved earlier (although sometimes temperatures above 200 eV were still observed). Therefore, our progress during the later phases of TMX operation was in physics understanding rather than in increasing plasma parameters. This increase in understanding resulted from more extensive diagnostic instrumentation, new data analysis, and improved theoretical understanding. Highlights of these recent results are summarized below.

### Tandem Configuration

- Plasma potential measurements carried out as a function of radius have shown that the electrostatic potential well is not just localized near the axis but extends across the central cell. The well-diagnosed plasmas had 150-V well depths, as expected for the measured electron temperature and densities. Other cases had well depths about twice as high.

- End-plug potentials exceeding 1 kV have been generated and maintained in TMX.

- Methods for controlling the radial profiles have been demonstrated. Operation of TMX over a wide range of central-cell gas feeds has shown that the radial density profiles can be changed from peaked on-axis to inverted profiles peaked off-axis. Additional diagnostic channels allowed us to measure these radial density profiles in more detail than was previously possible.

### Plasma Confinement

- Under proper operating conditions, the confinement of the central-cell plasma is in agreement with the theoretical Coulomb values. The highest confinement parameter achieved was  $n\tau = 10^{11} \text{ cm}^{-3} \cdot \text{s}$ . When end-plug fluctuation levels are significant, confinement is degraded in agreement with Monte-Carlo calculations. These fluctuations limited the range over which TMX could be

operated but did not prevent us from demonstrating the basic features of tandem mirrors.

### Power Balance

- We have been able to account for the neutral-beam power input by using multiple diagnostic arrays.

- Near the axis, most of the trapped neutral-beam power is lost axially, indicating good radial confinement. Near the edge, more power is lost radially. Radial arrays of calorimeters on the TMX end wall show that the power is more concentrated on the axis than was previously assumed. Radial end-loss analyzer measurements indicate that this concentration is due to the radial profiles of both the end-loss current and the plasma potential.

### Plasma Beta Measurements

- After TMX was shut down, an extensive calibration of the diamagnetic loops was carried out. With this new calibration, we determined that a maximum central-cell beta of 0.4 was achieved with neutral-beam injection.

- This calibration has also enabled us to conclude that the central-cell ion distribution has a non-Maxwellian component, as we had expected, because of ion-cyclotron heating by plug fluctuations.

### Radial Transport

- In TMX, radial particle confinement exceeds axial confinement near the axis. Near the edge, radial transport processes are more important.

- Resonant-neoclassical-ion-transport theory is consistent with the experimental measurements, but the measurements cannot resolve factor-of-3 uncertainties in the theory nor can we resolve comparable amounts of ambipolar radial transport.

### Radio-Frequency Measurements

- Wavelength and polarization measurements of the end-plug ion-cyclotron fluctuations indicate wave properties more similar to the Alfvén ion-cyclotron (AIC) mode than to the drift-cyclotron loss-cone (DCLC) mode. In comparison to 2XIIB, the AIC mode is theoretically more unstable in TMX, while the DCLC mode is less unstable. Theoretically, the AIC mode is expected to be much more stable in the low-ion-beta end plugs

of TMX Upgrade and MFTF-B, as is the DCLC mode.

- Turbulent noncoherent central-cell fluctuations extending up to 0.5 MHz, possibly associated with drift waves, have been detected in the central cell. No correlation with plasma confinement has been identified.

- Coherent low-frequency 7 kHz ( $m = 1$ ) and 12 kHz ( $m = 0$ ) oscillations have been observed in the central cell. The  $m = 0$  mode is correlated with bursting of end-plug ion-cyclotron fluctuations and thus would not be expected in TMX Upgrade with microstable sloshing-ion end plugs. The  $m = 1$  mode exists at large amplitude near the edge when sufficient central-cell gas input causes large density gradients near the edge. The  $m = 1$  mode can be controlled by modifying central-cell radial profiles by controlled central-cell fueling and heating.

#### **Impurity Studies**

- Further analysis continues to indicate remarkably low central-cell impurity levels (0.5%), resulting in less than 10% power loss by impurity radiation. Recent data analysis shows that the lower ionization states of the prevalent oxygen impurity are localized near the edge and the higher ionization states near the axis, as expected.

- High-resolution spectroscopy has provided Doppler-broadening measurements of impurity radiation that corroborate the 100-to-200-eV diamagnetic-loop measurements of central-cell ion temperature.

#### **Sandia Surface-Probe Studies**

- Surface probes have been employed to characterize plasma-wall interactions and to determine TMX plasma properties. Passive solid-state

probe measurements of particle fluxes and energies at the central-cell walls corroborate other central-cell diagnostic measurements.

- The probes collected the expected number of particles on the end walls. The major component of end losses comes from the central cell. The end-plug contribution is also in agreement with theoretical calculations.

#### **End-Wall Plasma Characteristics**

- We have succeeded in decoupling the TMX plasma from the end walls. This was a necessary accomplishment for future higher-temperature machines. A very low density ( $2 \times 10^9 \text{ cm}^{-3}$ ) and cool (5 eV) plasma exists near the end wall. This density is four orders of magnitude less and this electron temperature 40 times cooler than the density and electron temperature of the end plug plasmas.

- Secondary electrons emitted from the end wall are detected but the power losses are small, consistent with a model developed for MFTF-B end-wall processes.

#### **ONGOING RESEARCH**

The full report, UCRL-53120, describes many major new findings that will subsequently be reported in individual journal publications. Several questions have been raised and are the subject of ongoing research. Many of these questions will be answered by further data analysis and analytic modeling, while others can only be resolved experimentally in the TMX Upgrade, Phaedrus, or other tandem mirror machines. We expect to obtain more quantitative information about power balance, fluctuation studies, and radial transport.

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## III.2 Power Balance Model

### III.2.1 Introduction

The basic performance of TASKA has been calculated using a zero-dimensional model for the power and particle balance of the plasma. It treats three distinct regions of the machine (central cell, barrier, and plug) and includes power and particle flows between these regions as well as losses to the walls and absorption of injected power.

A zero-dimensional model was chosen for this study since, while tractable, it allows quick solution of the system of equations. It is, therefore, well-suited to parametric studies. This approach necessitates some approximations which will be discussed where appropriate. All formulas are written in the cgs system, except that energies and temperatures will be in keV.

The present work expands on the plasma physics analysis of Ref. 1, which was a major conceptual tandem mirror reactor study including engineering details. A rudimentary version of the analysis was given in Ref. 2. Most of the foundation of the physics of tandem mirrors with thermal barriers was laid at Lawrence Livermore National Laboratory<sup>(3,4,5)</sup>. Reference 3 is especially helpful, although it treats A-cell tandem mirrors, since much of the physics is the same or similar. Many of the equations and much of the notation in this paper is based on Ref. 3. For completeness, a full set of consistent equations will be given here.

There are two electron populations; their densities and temperatures or average energies are: 1) central cell ( $n_c, T_{ec}$ ) and 2) potential-trapped in the plug ( $n_{ew}, T_{ep}$ ). Electrons passing through both central cell and plug are counted in group 1. The three ion populations are: 1) central cell ( $n_c, T_{ic}$ ); 2) barrier, which includes both passing ( $n_{pass}^i, T_{ic}$ ) and trapped ( $n_{bt}, T_{ic}$ ); and

3) plug, which are mirror-trapped ( $n_p, E_p$ ). The central cell density  $n_c$  includes the density,  $n_\alpha$ , of thermalized alpha particles. These will be discussed in detail in the appropriate places. It has been assumed that alpha particles are removed from the barrier and do not contribute to the barrier density.

The exposition of the physics included here will begin with discussions of the details of the three main plasma regions. These details will then be combined into a set of power balance equations. First, some general considerations and essential formulas are given, perhaps most useful for reference.

### III.2.2 General Formulas

The plasma radius at a given  $B(z)$  follows from flux conservation:

$$r(z) = r_c \left\{ \frac{B_c}{B(z)} \left[ \frac{1-\beta_c}{1-\beta(z)} \right]^{1/2} \right\}^{1/2} \quad (\text{III.2-1})$$

where  $\beta(z) \equiv$  perpendicular plasma pressure/magnetic field pressure, values are referenced to the central cell, and the long-thin approximation has been used for beta correction of the magnetic fields.

Betas are given by

$$\beta_c = \frac{n_c (T_{ec} + T_{ic} + T_\alpha)}{2.5 \times 10^7 B_c^2} \quad (\text{III.2-2})$$

for the central cell,

$$\beta_b = \frac{n_b \left[ T_{ic} \frac{B_b}{B_{maxb}} + T_{ec} \right]}{2.5 \times 10^7 B_b^2} \quad (\text{III.2-3})$$

for the barrier, and

$$\beta_p = \frac{n_p (0.9 E_p + T_{ep})}{2.5 \times 10^7 B_p^2} \quad (\text{III.2-4})$$

for the plug, where  $B_b/B_{\max b}$  corrects for the constancy of  $u \equiv v_{\perp}^2/2B$  of barrier ions, the coefficient 0.9 in  $\beta_p$  approximately corrects for the average plug ion energy being mainly perpendicular, and alpha particles are assumed to contribute to  $\beta_c$  with an equivalent temperature of<sup>(5)</sup>

$$T_{\alpha} = 8.3 \times 10^{10} \langle \sigma v \rangle_{DT} E_{\alpha} f_{e\alpha} T_{ec}^{3/2} \quad (\text{III.2-5})$$

where  $\langle \sigma v \rangle_{DT}$  is the fusion reaction rate, and the alpha particle energy,  $E_{\alpha}$ , is assumed to be distributed to the central cell ions and electrons as follows: the fraction  $f_{e\alpha}$  goes to electrons,  $f_{i\alpha}$  goes to ions, and the remainder is lost out the ends. The values  $f_{e\alpha}$  and  $f_{i\alpha}$  depend on a variety of parameters, and the values used for TASKA are generated numerically by an analysis given in Ref. 6.

The collisional energy transfer rate  $v^{\gamma/\beta}$  from species  $\beta$  to species  $\gamma$  is approximated by assuming both species are Maxwellian and using standard formulas:<sup>(7)</sup>

$$v^{\gamma/\beta} = 5.7 \times 10^{-24} \frac{(m_{\gamma} m_{\beta})^{1/2} Z_{\gamma}^2 Z_{\beta}^2 n_{\beta} \Lambda_{\gamma\beta}}{(m_{\gamma} T_{\beta} + m_{\beta} T_{\gamma})^{1.5}} \quad (\text{III.2-6})$$

where  $m_j$  is the mass,  $Z_j$  is the atomic number,  $n_j$  is the density, and  $T_j$  is the temperature of species  $j$ ;  $\Lambda_{\gamma\beta}$  is a Coulomb logarithm, also given in Ref.

7. The assumption of Maxwellian distribution functions is inaccurate for



streaming components, but collisional terms tend to be small, and no serious consequences of this approximation are anticipated.

The Logan-Rensink model is used for  $(n\tau)_{ip}$  of ions in the plug<sup>(8)</sup>, giving

$$(n\tau)_{ip} = \left[ \frac{\Lambda_{iip} (m_D/m_{ip})^{1/2}}{3.9 \times 10^{12} E_{inj}^{1.5} \log_{10} R_{eff}} + \frac{\Lambda_{eip} (m_H/m_{ip})}{10^{13} T_{ep}^{1.5} \ln(E_{inj}/E_{out})} \right]^{-1} \quad (III.2-7)$$

where  $m_D$  is the mass of deuterium,  $m_H$  is the mass of hydrogen,  $m_{ip}$  is the plug ion mass,  $\Lambda_{iip}$  and  $\Lambda_{eip}$  are Coulomb logarithms given by  $\Lambda_{iip} = 28 + 1/2 (\ln T_{ep} + \ln E_p - \ln n_p)$  and  $\Lambda_{eip} = 24 + \ln T_{ep} - 1/2 \ln n_p$ ,  $E_{inj}$  is the neutral beam injection energy. The mean ion escape energy,  $E_{out}$ , is given by

$$E_{out} = \frac{E_{inj}}{1 + \frac{\tau_i}{\tau_{dr}}} + \frac{(\phi_c + \phi_e)}{R_p \sin^2 \theta} \frac{\tau_i / \tau_{dr}}{[(1 - \beta_p)^{1/2} - 1] (1 + \frac{\tau_i}{\tau_{dr}})} \quad (III.2-8)$$

where  $R_p$  is the plug vacuum mirror ratio,  $\theta$  is the beam injection angle,  $\beta_p$  is the plug beta, and  $\tau_i / \tau_{dr}$ , the ratio of ion collision time to electron drag time, is given by

$$\frac{\tau_i}{\tau_{dr}} = .11 \left( \frac{E_{inj}}{T_{ep}} \right)^{1.5} \frac{\Lambda_{eip}}{\Lambda_{iip}} \frac{\log_{10} R_{eff}}{\ln \left( \frac{E_{inj}}{E_{out}} \right)} \left( \frac{m_D}{m_{ip}} \right)^{1/2} \quad (III.2-9)$$

where  $T_{ep}$  is the plug electron temperature. The effective mirror ratio,  $R_{eff}$ , is

$$R_{eff} = \frac{R_p / (1 - \beta_p)^{1/2}}{1 + \frac{(\phi_e + \phi_c)}{E_{inj}}} \sin^2 \theta. \quad (III.2-10)$$

For ions in the central cell,  $(n\tau)_{ic}$  is given by Pastukhov's analysis as corrected and expanded by Cohen, et al.<sup>(9,10)</sup>

$$(n\tau)_{ic} = \frac{(m_i/2)^{1/2}}{Z_{ic} \pi e^4 \Lambda_{iic}} \frac{G(Z_{ic} R_c)}{I(\frac{T_{ic}}{\phi_c})} T_{ic}^{1/2} \phi_c e^{\phi_c/T_{ic}} \quad (\text{III.2-11})$$

where  $m_i$  is the ion mass,  $\Lambda_{iic}$  is the ion Coulomb logarithm given by  $\Lambda_{iic} = 24 + \ln T_{ic} - 1/2 \ln n_c$ ,  $Z_{ic} = 0.5$  for ions because they scatter only slightly off of electrons,  $R_c = B_{\max}/B_c$  is the central cell mirror ratio,

$$G(x) \equiv \frac{\pi^{1/2}}{4} (1 + \frac{1}{x})^{1/2} \ln \left[ \frac{(1 + \frac{1}{x})^{1/2} + 1}{(1 + \frac{1}{x})^{1/2} - 1} \right] \quad (\text{III.2-12})$$

and

$$I(x) \equiv \frac{1 + \frac{x}{2}}{1 + \frac{x^2}{4}}. \quad (\text{III.2-13})$$

For electrons in the central cell, similar analysis gives

$$(n\tau)_{ec} = \frac{(m_e/2)^{1/2}}{Z_{ec} \pi e^4 \Lambda_{eec}} \frac{G(Z_{ec} R_c)}{I(\frac{T_{ec}}{\phi_e})} T_{ec}^{1/2} \phi_e e^{\phi_e/T_{ec}} \quad (\text{III.2-14})$$

where  $m_e$  is the electron mass,  $\Lambda_{eec}$  is the electron Coulomb logarithm, given by  $\Lambda_{eec} = 24 + \ln T_{ec} - 1/2 \ln n_c$ , and  $Z_{ec} = 1$  for electrons because they scatter off both ions and electrons.

Synchrotron power for a given density  $n$ , electron temperature  $T_e$ , plasma radius  $r$ , magnetic field  $B$ , volume  $V$ , and reflection coefficient  $R_s$  is given by<sup>(11)</sup>

$$P_S = 4.2 \times 10^{-23} \left(\frac{n}{T}\right)^{1/2} (BT_e)^{5/2} (1 - R_S)^{1/2} \text{ V watts (III.2-15)}$$

and Bremsstrahlung power is given by<sup>(7)</sup>

$$P_B = 5.34 \times 10^{-31} n^2 T_e^{1/2} \text{ V watts . (III.2-16)}$$

### III.2.3 Barrier Physics

The barrier potential,  $\phi_b$ , arises from a dip in ion density (and therefore electron density) through the equation

$$e\phi_b = T_{ec} \ln \frac{n_c}{n_b} \quad (\text{III.2-17})$$

assuming Maxwellian electrons.

This density decrease is caused by the expansion of the magnetic field flux tube as the B field drops and by acceleration of the ions as they fall down the potential hill. To maintain the density dip, passing central cell ions which collisionally scatter into trapped orbits must be pumped out of the barrier region. The pumping mechanism considered here is charge exchange pumping. Neutral beam particles injected into the barrier loss cone become central cell ions when they charge exchange. The resulting neutrals tend to have fairly large perpendicular velocities and should quickly leave the machine; thus, reionization of these particles is neglected. For a trapped barrier ion with zero parallel energy at a point where the potential is  $\phi_{in}$ , relative to the central cell, the injection energy necessary to create a passing ion by overcoming that potential and the  $\nabla B$  forces is

$$E_{in} = \frac{\phi_{in}}{1 - \frac{B_{maxb}}{B_{in}} \sin^2 \theta_{in}} \quad (\text{III.2-18})$$

where  $B_{in}$  is the magnetic field at the injection point and  $\theta_{in}$  is the injection angle. When the potential  $\phi_{in}$  is positive relative to the central cell, the effective injection angle is actually smaller than  $\theta_{in}$  because  $\phi_{in}$  increases  $v_{\parallel}$  at the  $\phi = 0$  point. The result is

$$\theta_{eff} = \arctan \left[ \frac{\sin \theta_{in}}{\cos \theta_{in} + (\phi_{in}/E_{in})^{1/2}} \right] \quad (\text{III.1-19})$$

Charge exchange with passing ions is neglected since the resulting neutral can still pump the barrier. Since each charge exchange event, which does the pumping, is accompanied by  $\langle \sigma v \rangle_{ion} / \langle \sigma v \rangle_{cx}$  ionization events, the total energy change per pumped ion is

$$(E_{in} - \phi_{in}) \left( 1 + \frac{\langle \sigma v \rangle_{ion}}{\langle \sigma v \rangle_{cx}} \right) - \frac{3}{2} T_{ic} \quad (\text{III.2-20})$$

where  $\langle \sigma v \rangle_{ion}$  includes ionization on both trapped and passing ions. If the electron from an ionization event enters the central cell, it will add energy  $\phi_{in}$  to the central cell electrons. If the electron enters the plug, it will add energy  $\phi_{in} + \phi_c$ , but a plug electron must then become a passing electron in order to maintain quasi-neutrality, carrying an energy  $\phi_b + \phi_c + T_{ep}$  from the plug and adding  $\phi_b + T_{ep}$  to the central cell.

Achieving a high barrier potential while keeping the total barrier density to a minimum is important. The lower limit on barrier density is the density of central cell ions passing through the barrier,

$$n_{\text{pass}}^i(z) = n_c \frac{B(z)}{B_{\text{maxb}}} \left[ \frac{T_{ic}}{\pi \phi_b(z) + T_{ic}} \right]^{1/2} . \quad (\text{III.2-21})$$

The total barrier density is normalized to  $n_{\text{pass}}^i$  through the relation

$$n_b = g_b n_{\text{pass}}^i . \quad (\text{III.2-22})$$

The calculation of  $g_b$  is difficult; a reasonable value of  $g_b = 2$  is generally used in this report.

The current per unit volume of passing ions which trap in the barrier at  $z$  is given by a formula fitted to Fokker-Planck studies<sup>(5)</sup>:

$$J_{\text{trap}}(z) = \frac{[n_{\text{pass}}^i(z)]^2}{5.5 \times 10^9 T_{ic}^{3/2}} \left[ 1 + 0.185 \frac{B_{\text{maxb}}}{B(z)} \right] . \quad (\text{III.2-23})$$

Average  $B$  and  $\phi$  values for the barrier are found assuming parabolic profiles,

$$\langle B \rangle_b = \frac{2}{3} B_b + \frac{1}{6} (B_{\text{maxb}} + B_{\text{maxp}}) \quad (\text{III.2-24})$$

and

$$\langle \phi \rangle_b = \frac{2}{3} \phi_b , \quad (\text{III.2-25})$$

then these are inserted into  $J_{\text{trap}}$  and also used to find the effective barrier volume  $V_b = \pi \langle r \rangle_b^2 L_b$ .  $\langle r \rangle_b$  is calculated using  $\langle B \rangle_b$  in Eq. (III.2-1). The total current of ions scattering into the barrier is then

$$I_{\text{trap}} \approx \frac{2V_b n_c^2}{5.5 \times 10^9 T_{ic}^{1/2} B_{\text{maxb}}^2} \left[ \frac{\langle B \rangle_b^2 + 0.185 B_{\text{maxb}} \langle B \rangle_b}{\pi \langle \phi \rangle_b + T_{ic}} \right] . \quad (\text{III.2-26})$$

This approach agrees well with the values from Ref. 12, Table 1, after compensating for the use in Ref. 12 of 0.55 instead of the more accurate 0.185 in  $J_{\text{trap}}$ . Total currents tend to be a factor of one to two higher than the Ref. 12 currents, in harmony with the philosophy used here of being slightly pessimistic when estimating uncertain effects.

#### III.2.4 Central Cell Physics

The calculation of the potential  $\phi_c$ , which confines central cell ions, has been examined in Ref. 13. Passing electrons scattering into the plug and electrons from neutral beam ionization balance the loss of electrons by collisional scattering to parallel energies above the potential  $\phi_b + \phi_c$  to give

$$\begin{aligned} \frac{n_{\text{ew}}^2(p)}{(n\tau)_{\text{ep}}} \left[ 1 - \frac{n_{\text{ec}}(b)}{n_{\text{ew}}(p)} \left( \frac{T_{\text{ep}}}{T_{\text{ec}}} \right)^{v_c} \exp \left( \frac{\phi_b + \phi_c}{T_{\text{ep}}} \right) \right] &= \frac{n_p^2}{(n\tau)_{\text{ip}}} \\ &+ I_{\text{trap}} \sum_{k=1}^3 f_{\text{nk}} f_{\text{pk}} \frac{\langle \sigma v \rangle_{\text{ionk}}}{\langle \sigma v \rangle_{\text{cxk}}} \end{aligned} \quad (\text{III.2-27})$$

where  $n_{\text{ew}}(p)$  is the density of warm (plug) electrons at point p,  $n_{\text{ec}}(b)$  is the density of passing central cell electrons at point b, the subscript  $k = 1, 2, 3$  labels the barrier pumping neutral beams in order of descending energy,  $f_{\text{ck}}$  and  $f_{\text{pk}}$  are the fractions of beam k lying to central cell or plug side of the barrier,  $\langle \sigma v \rangle_{\text{ionk}}$  and  $\langle \sigma v \rangle_{\text{cxk}}$  are ionization and charge exchange rates, and  $f_{\text{nk}}$  is the fraction of trapping current which beam k must pump. When the ions resulting from one beam are pumped into a region where another beam can pump them into the central cell (two stage pumping),  $f_{\text{nk}}$  for the second region increases and  $f_{\text{n1}} + f_{\text{n2}} + f_{\text{n3}} > 1$ . The amount of increase is (for beam 1 pumping to the beam 3 region)

$$\Delta f_{n3} = f_{n1} \left( \frac{g_b}{g_b - 1} + \frac{\langle \sigma v \rangle_{ion1}}{\langle \sigma v \rangle_{cx1}} \right) . \quad (\text{III.2-28})$$

Plug electron confinement time is given by

$$(n\tau)_{ep} = \frac{8.2 \times 10^9}{\Lambda_{ep}} \frac{G\left(\frac{B_{maxp}}{B_p}\right)}{I\left(\frac{T_{ep}}{\phi_b + \phi_c}\right)} T_{ep}^{3/2} \exp\left(\frac{\phi_b + \phi_c}{T_{ep}}\right) , \quad (\text{III.2-29})$$

where  $\Lambda_{ep}$  is the plug electron Coulomb logarithm. The first term on the RHS of Eq. (III.2-27) gives the source of plug electrons due to ionization of the plug neutral beam. The second RHS term gives the plug electron source due to pump beam ionization events occurring on the plug side of the barrier potential. The parameter  $\nu_c$  is 0.5. Unfortunately,  $\nu_c = 0.5$  requires, as can be seen in Eq. (III.2-27), a higher value of  $n_{ew}(p)$  for a given  $n_{ec}(b)$  than  $\nu_c = 0$  does, and plug power varies approximately as  $n_{ew}^2(p)$ .

The main contributors to central cell power are the collisional slowing of fusion alpha particles on ions and electrons, and power from pump beam particles which ionize and enter the central cell. These are chiefly balanced by end loss. The central cell electron power balance is given by

$$\begin{aligned}
& \frac{n_c^2}{4} \langle \sigma v \rangle_{DT} f_{e\alpha} E_\alpha \left(1 + \frac{E_r}{E_{fus}}\right) V_c + \frac{3}{2} n_c v^{ec/ic} (T_{ic} - T_{ec}) V_c \\
& + n_c v^{ec/ip} \left(E_p - \frac{3}{2} T_{ec}\right) V_c + \frac{n_p^2}{(n\tau)_{ip}} (\phi_b + T_{ep}) 2V_p \\
& + 3 \frac{n_{ec}(b) n_{ew}(p)}{(n\tau)_{ep}} \left(\frac{T_{ep}}{T_{ec}}\right)^{V_c} (T_{ep} - T_{ec}) \exp\left(-\frac{\phi_b + \phi_c}{T_{ep}}\right) V_p \\
& + 2 I_{trap} \sum_{k=1}^3 f_{nk} [f_{ck} \phi_{ink} + f_{pk} (\phi_b + T_{ep})] \frac{\langle \sigma v \rangle_{ionk}}{\langle \sigma v \rangle_{cxk}} \\
& = \left[ \frac{n_c^2}{(n\tau)_{ec}} + \frac{n_c^2}{(n\tau)_{xfe}} \right] (\phi_e + T_{ec}) V_c + P_{sc} + P_{br}
\end{aligned} \tag{III.2-30}$$

where  $\phi_{ink}$  is the potential at which the beam  $k$  is aimed, and  $(n\tau)_{xfe}$  is the  $(n\tau)$  product for electron cross-field diffusion. The equilibrium central cell alpha particle density,  $n_\alpha$ , is found by equating alpha particle production to end loss. That is,

$$\frac{(n_c - 2n_\alpha)^2}{4} \langle \sigma v \rangle_{DT} V_c = \frac{n_\alpha^2}{(n\tau)_{\alpha c}} V_c \tag{III.2-31}$$

where, similar to Eq. (III.2-11),

$$(n\tau)_{\alpha c} = \frac{(m_\alpha/2)^{1/2}}{Z_{\alpha c} \pi e^4 \Lambda_\alpha} \frac{G(R_c/2)}{I(T_{ic}/z_\alpha \phi_c)} T_{ic}^{1/2} z_\alpha \phi_c e^{z_\alpha \phi_c / T_{ic}} \tag{III.2-32}$$

with  $m_\alpha$  the alpha particle mass,  $\Lambda_\alpha = 20$  the alpha particle Coulomb logarithm,  $Z_\alpha \approx .33$ , and  $z_\alpha = 2$ .  $E_{fus}$  is the energy output per fusion, and  $E_r$  is the average reacting ion energy, taken to be (5)

$$E_r = 45 + \frac{3}{2} T_{ic} . \tag{III.2-33}$$



The first term in Eq. (III.2-30) is power from fusion; the second and third terms come from collisional energy transfer; the fourth and fifth terms from conduction and convection of energy between the plug and central cell<sup>(3)</sup>; the sixth term from ionization of barrier pumping neutral beams; the seventh term from end loss and cross-field transport; the eighth term from synchrotron radiation; and the last term from Bremsstrahlung.

Central cell ion power balance is given by

$$\begin{aligned}
 & \frac{(n_c - n_\alpha)^2}{4} \langle \sigma v \rangle_{DT} [f_{i\alpha} E_\alpha (1 + \frac{E_r}{E_{fus}}) - E_r] V_c + P_{ICRH} \\
 & + 2 I_{trap} \sum_{k=1}^3 f_{nk} [(1 + \frac{\langle \sigma v \rangle_{ionk}}{\langle \sigma v \rangle_{cxk}}) (E_{ink} - \phi_{ink}) - \frac{3}{2} T_{ic}] \\
 & = [\frac{n_c^2}{(n\tau)_{ic}} + \frac{n_c^2}{(n\tau)_{xfi}}] (\phi_c + T_{ic}) V_c + \frac{3}{2} n_c v^{ic/ec} (T_{ic} - T_{ec}) V_c
 \end{aligned} \tag{III.2-34}$$

where  $(n\tau)_{xfi}$  is the ion cross-field  $(n\tau)$ .

The first term gives power distributed to the ions from alpha particles; the second term gives heating due to ion cyclotron range of frequencies (ICRF) power; the third term comes from neutral beam barrier pumping; the fourth term gives end loss and cross-field transport; and the fifth term gives collisional energy transfer.

Fueling of the central cell must compensate for end loss, cross-field transport, and loss of ions to fusion events. Since TASKA is a relatively low power device, operating points were chosen where ionization events of the barrier pumping neutral beams supply the entire fueling needs of the central cell. This alleviates the requirement of fueling by gas puffing or pellet injection, which are difficult and not well understood. The ion particle

balance is governed by

$$\left[ \frac{n_c^2}{(n\tau)_{ic}} + \frac{n_c^2}{(n\tau)_{xfi}} + \frac{n_c^2}{2} \langle \sigma v \rangle_{DT} \right] V_c = 2 I_{\text{trap}} \sum_{k=1}^3 f_{nk} \frac{\langle \sigma v \rangle_{\text{ionk}}}{\langle \sigma v \rangle_{\text{cxk}}} . \quad (\text{III.2-35})$$

### III.2.5 Plug Physics

Plug electron power balance is given by

$$\begin{aligned} \frac{P_{\text{eaux}}}{2} + n_{\text{ew}}(p) v^{\text{ep/ip}} (E_p - \frac{3}{2} T_{\text{ep}}) V_p &= I_{\text{trap}} \sum_{k=1}^3 f_{nk} f_{pk} \frac{\langle \sigma v \rangle_{\text{ionk}}}{\langle \sigma v \rangle_{\text{cxk}}} (\phi_b - \phi_{\text{ink}} \\ &+ T_{\text{ep}}) + \frac{3}{2} \frac{n_{\text{ec}}(b) n_{\text{ew}}(p)}{(n\tau)_{\text{ep}}} \left( \frac{T_{\text{ep}}}{T_{\text{ec}}} \right)^{v_c} (T_{\text{ep}} - T_{\text{ec}}) \exp \left( \frac{\phi_b + \phi_c}{T_{\text{ep}}} \right) V_p \\ &+ \frac{n_p^2}{(n\tau)_{ip}} (\phi_b + \phi_c + T_{\text{ep}}) V_p + P_{\text{sp}} + P_{\text{bp}} . \end{aligned} \quad (\text{III.2-36})$$

$P_{\text{eaux}}$  is the total auxiliary heating of plug electrons for both plugs; the second term gives collisional energy transfer; the third term comes from ionization events of the barrier pumping neutral beams; the fourth and fifth terms arise from conduction and convection of energy between the plug and the central cell;  $P_{\text{sp}}$  is synchrotron power, given by Eq. (III.2-15); and  $P_{\text{bp}}$  is Bremsstrahlung power, give by Eq. (III.2-16).

Plug ion power balance is given by

$$\begin{aligned} \frac{n_p^2}{(n\tau)_{ip}} (E_{\text{inj}} - E_{\text{out}}) V_p &= \frac{n_p^2}{(n\tau)_{ip}} \frac{\langle \sigma v \rangle_{\text{cxp}}}{\langle \sigma v \rangle_{\text{ionp}}} (E_p - E_{\text{inj}}) V_p \\ &+ n_p v^{\text{ip/ep}} (E_p - \frac{3}{2} T_{\text{ep}}) V_p + n_p v^{\text{ip/ec}} (E_p - \frac{3}{2} T_{\text{ec}}) V_p \end{aligned} \quad (\text{III.2-37})$$

where  $\langle\sigma v\rangle_{cxp}$  and  $\langle\sigma v\rangle_{ionp}$  are the plug charge exchange and ionization rates. The first term is the power deposited by neutral beams; the second term represents energy lost to charge exchange; and the third and fourth terms give collisional energy transfer.

### III.2.6 Input Power

The power required to sustain a single plug is

$$P_{nb}^p = \frac{n_p^2}{(n\tau)_{ip}} E_{inj} V_p \quad (III.2-38)$$

while barrier pumping powers are given by

$$P_{nk}^b = f_{nk} I_{trap} \left(1 + \frac{\langle\sigma v\rangle_{ionk}}{\langle\sigma v\rangle_{cxnk}}\right) E_{ink} \quad (III.2-39)$$

where  $k = 1, 2, 3$  is the beam index. These give the absorbed beam power. Beam trapping fractions are then found using

$$f_{tr} = 1 - \exp \left[ \frac{-4.52 \times 10^{-8} n \ell (\langle\sigma v\rangle_{ion} + \langle\sigma v\rangle_{cx})}{(E_{in}/m_i)^{1/2}} \right]. \quad (III.2-40)$$

The expression in square brackets is the ratio of beam path length in the plasma to the mean free path.  $n$  is density,  $\ell$  is path length,  $\langle\sigma v\rangle_{ion}$  is ionization rate,  $\langle\sigma v\rangle_{cx}$  is charge exchange rate,  $E_{in}$  is injection energy, and  $m_i$  is the ratio of ion mass to proton mass. Injected powers are then given by

$$P_{nk}^{inj} = \frac{P_{nk}}{f_{trk}} \quad (III.2-41)$$

where  $k$  is again the beam index.

### III.2.7 Numerical Solution of the System of Equations

The equations are solved by conventional iterative methods.<sup>(14)</sup> The power balance equations, Eqs. (III.2-30), (III.2-36), and (III.2-37) are written as time rate of change equations for temperature or energy. For a given set of input guesses, standard methods are used to simultaneously iterate equations until  $dT_{ec}/dt = 0 = dE_p/dt$ . Then, the other quantities allowed to vary are updated and the process is repeated until a relative accuracy of about  $10^{-4}$  is reached. Time is thus used as a dummy variable; more care must be exercised if the truly time-dependent behavior is desired. A time-dependent code for solution of this system is presented in Sec. IX.1.

#### References for Section III.2

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### III.3 Special Cases and Parametric Analysis

This section will discuss some of the modes of operation which were considered as alternatives to the TASKA reference design presented in Section III.4. Also, the dependence of selected important parameters on a variety of variables will be examined.

For the reference case, in this section only, the parameters of Tables III.3-1 and III.3-2 are used. This case is very close to the TASKA design point given in Section III.4, but reflects some minor mistakes corrected in the computer program since the plasma physics parameters were frozen on July 31, 1981. The changes were: 1) the central cell ion power balance coefficients for two stage pumping were corrected so as not to subtract the two stage pumped ion twice from the central cell power; and 2) the neutral beam trapping fraction was corrected by dividing the injection energy by particle mass in order to get the correct particle velocity. Parameters not given here are equal to those given in Section III.4 for the TASKA reference case.

The methodology of the parametric variation was that as few parameters as possible were varied when investigating a given variable. When necessary, the ICRF power and occasionally the medium energy neutral beam energy were altered to give beam fueling equal to particle loss. In a few instances, central cell magnetic field had to be lowered slightly in order to reach an equilibrium state.

One of the most difficult problems faced in the TASKA study was finding a suitable means of pumping out ions which become trapped in the barrier region. The most straightforward means is to inject neutral beams into the barrier loss cone so that charge exchange events will give ions which travel into the central cell, and this was chosen for TASKA. Some alternatives are discussed

Table III.3-1. Section III.3 Reference Case Power and Machine Parameters

|                        |                        |
|------------------------|------------------------|
| CC neutral wall load   | 1.50 MW/m <sup>2</sup> |
| Fusion power           | 86. MW                 |
| CC plasma radius       | 0.32 m                 |
| Average barrier radius | 0.20 m                 |
| Midplane plug radius   | 0.29 m                 |
| CC magnetic field      | 2.7 T                  |
| <u>Neutral Beams</u>   |                        |
| Plug: power            | 5.2 MW                 |
| energy                 | 75 keV                 |
| angle                  | 20°                    |
| Barrier: power         | 48.6 MW                |
| energy                 | 50 keV                 |
| angle                  | 25°                    |
| Barrier: power         | 48.6 MW                |
| energy                 | 50 keV                 |
| angle                  | 25°                    |
| Barrier: power         | 0.2 MW                 |
| energy                 | 2 keV                  |
| angle                  | 45°                    |
| ECRF power             | 15.3 MW                |
| ICRF power             | 38.5 MW                |
| Total injected power   | 113 MW                 |

Table III.3-2. Section III.3 Reference Case Plasma Parameters

Central Cell

|                        |                                                  |
|------------------------|--------------------------------------------------|
| Density                | $1.95 \times 10^{14} \text{ cm}^{-3}$            |
| Alpha particle density | $2.78 \times 10^{12} \text{ cm}^{-3}$            |
| Ion temperature        | 30. keV                                          |
| Electron temperature   | 11.4 keV                                         |
| $(n\tau)_{ic}$         | $5.3 \times 10^{13} \text{ cm}^{-3} \text{ sec}$ |
| $(n\tau)_{ec}$         | $5.3 \times 10^{13} \text{ cm}^{-3} \text{ sec}$ |
| Potential, $\phi_c$    | 43 keV                                           |

Barrier

|                                         |                                      |
|-----------------------------------------|--------------------------------------|
| Minimum density                         | $6.8 \times 10^{12} \text{ cm}^{-3}$ |
| Potential, $\phi_b$                     | 37.3 keV                             |
| High energy pumping fraction            | 0.025                                |
| Medium energy pumping fraction          | 0.975                                |
| Low energy pumping fraction (two stage) | 0.109                                |

Plug

|                              |                                                  |
|------------------------------|--------------------------------------------------|
| Midplane density             | $6.3 \times 10^{13} \text{ cm}^{-3}$             |
| Mean ion energy              | 388 keV                                          |
| Electron temperature         | 59.4 keV                                         |
| $(n\tau)_{ip}$               | $2.9 \times 10^{13} \text{ cm}^{-3} \text{ sec}$ |
| $(n\tau)_{ep}$               | $7.4 \times 10^{11} \text{ cm}^{-3} \text{ sec}$ |
| Potential, $\phi_c + \phi_e$ | 109 keV                                          |



in Section III.1. However, a number of different neutral beam configurations in the barrier still had to be evaluated. The reference case uses three barrier beams, given in Table III.3-1. The distinctive feature of this method is the use of two stage pumping whereby trapped ions are pumped from the bottom of the barrier to a region where a low energy beam can pump them into the central cell. Parameters for three other possible configurations are given in Tables III.3-3 and III.3-4.

Case A describes a situation where the high energy neutral beam pumping the bottom of the barrier is at a sufficiently small angle with respect to the magnetic field so that charge exchange ions are in the barrier loss cone and reach the central cell in a single step (cf. Eq. III.2-18). Although the parameters for case A are attractive in some respects, engineering considerations effectively prohibit reaching the small angle required for pumping the bottom of the barrier, especially at the required high power. This one stage scheme was used in the WITAMIR-I tandem mirror reactor study<sup>(1)</sup>, but beam lines were very long and central cell blanket access was hampered for over ten meters on each end; this is unacceptable for a test device such as TASKA.

In case B, the low energy beam is eliminated and its function in two stage pumping is assumed by the medium energy beam. Required input power rises dramatically and, since the low energy beam needs only relatively easy technology, the trade-off greatly favors the reference case. Also, the high energy beam now needs a  $16^\circ$  injection angle.

Case C investigates the possibility of aiming the high energy beam into the loss cone from barrier bottom to plug. The mirror ratio involved is then 7.8 instead of 25, although the potential to be overcome is approximately doubled. The charge exchange ions then leave the machine entirely for

Table III.3-3. Power and Machine Parameters

A. HEPB Pumps Directly Into the Central Cell

B. HEPB Pumps to MEPB Region

C. HEPB Pumps Out Over Plug Potential

|                                         | <u>Case A</u> | <u>Case B</u> | <u>Case C</u> |
|-----------------------------------------|---------------|---------------|---------------|
| CC neutron wall load, MW/m <sup>2</sup> | 1.50          | 1.53          | 1.50          |
| Fusion power, MW                        | 87.           | 86.           | 92.           |
| CC plasma radius, m                     | 0.34          | 0.31          | 0.36          |
| Average barrier radius, m               | 0.21          | 0.19          | 0.22          |
| Midplane plug radius, m                 | 0.31          | 0.27          | 0.32          |
| CC magnetic field, T                    | 2.8           | 2.6           | 2.7           |
| <u>Neutral Beams</u>                    |               |               |               |
| Plug: power, MW                         | 7.3           | 4.8           | 7.8           |
| energy, keV                             | 250.          | 250.          | 250.          |
| angle                                   | 60°           | 60°           | 60°           |
| Barrier: power, MW                      | 51.6          | 92.2          | 58.4          |
| energy, keV                             | 200.          | 207.          | 190.          |
| angle                                   | 10°           | 16°           | 15°           |
| Barrier: power, MW                      | 14.4          | 80.3          | 48.4          |
| energy, keV                             | 20.           | 50.           | 50.           |
| angle                                   | 25°           | 25°           | 25°           |
| ECRF power, MW                          | 11.7          | 14.2          | 13.6          |
| ICRF power, MW                          | 35.           | 45.           | 32.5          |
| Total injected power, MW                | 120.          | 237.          | 161.          |

Table III.3-4. Plasma Parameters

A. HEPB Pumps Directly Into the Central Cell

B. HEPB Pumps to MEPB Region

C. HEPB Pumps Out Over Plug Potential

|                                               | <u>Case A</u>         | <u>Case B</u>         | <u>Case C</u>         |
|-----------------------------------------------|-----------------------|-----------------------|-----------------------|
| <u>Central Cell</u>                           |                       |                       |                       |
| Density, $\text{cm}^{-3}$                     | $1.93 \times 10^{14}$ | $2.00 \times 10^{14}$ | $1.85 \times 10^{14}$ |
| Alpha density, $\text{cm}^{-3}$               | $4.8 \times 10^{12}$  | $1.1 \times 10^{12}$  | $3.9 \times 10^{12}$  |
| Ion temperature, $\text{cm}^{-3}$             | 30.                   | 30.                   | 30.                   |
| Electron temperature, $\text{cm}^{-3}$        | 13.8                  | 8.7                   | 12.9                  |
| $(n\tau)_{ic}$ , $\text{cm}^{-3} \text{ sec}$ | $7.7 \times 10^{13}$  | $3.0 \times 10^{13}$  | $6.9 \times 10^{13}$  |
| $(n\tau)_{ec}$ , $\text{cm, sec}$             | $7.5 \times 10^{13}$  | $2.9 \times 10^{13}$  | $6.4 \times 10^{13}$  |
| Potential, $\phi_c$ , keV                     | 49.1                  | 32.8                  | 47.1                  |
| <u>Barrier</u>                                |                       |                       |                       |
| Minimum density, $\text{cm}^{-3}$             | $6.2 \times 10^{12}$  | $8.0 \times 10^{12}$  | $6.2 \times 10^{12}$  |
| Potential, $\phi_b$ , keV                     | 46.2                  | 27.2                  | 42.7                  |
| HEPB fraction                                 | 0.025                 | 0.025                 | .025                  |
| MEPB fraction                                 | 0.975                 | 1.37                  | .975                  |
|                                               |                       | (two stage)           |                       |
| <u>Plug</u>                                   |                       |                       |                       |
| Midplane density, $\text{cm}^{-3}$            | $6.2 \times 10^{13}$  | $7.0 \times 10^{13}$  | $6.4 \times 10^{13}$  |
| Mean ion energy, keV                          | 388.                  | 357.                  | 383.                  |
| Electron temperature, keV                     | 63.5                  | 45.1                  | 58.1                  |
| $(n\tau)_{ip}$                                | $2.5 \times 10^{13}$  | $3.2 \times 10^{13}$  | $2.6 \times 10^{13}$  |
| $(n\tau)_{ep}$                                | $9.6 \times 10^{11}$  | $4.9 \times 10^{11}$  | $8.9 \times 10^{11}$  |
| Potential, $\phi_c + \phi_e$ , keV            | 130                   | 81.3                  | 121.7                 |

sufficiently energetic ions. Unfortunately, as seen in Table III.3-3, the required total power is very large, primarily because the high energy beam pumps energy from the barrier and thus from the central cell while giving no energy to the central cell. The injection angle of  $15^\circ$  is even more of a problem, as it is very difficult to reach.

A disadvantage shared by cases A, B, and C is that the high energy pump beam is both high energy and high power, in contrast to the moderate energy, low power beam of the reference case two stage pumping scheme. Construction of the reference case is thus far easier.

The low energy pump beam is very efficient, and some tandem mirror designs do a large fraction of the barrier pumping by this beam.<sup>(2)</sup> There is a disadvantage associated with the low energy beam, however: electrons resulting from ionization events are born on the plug side of the barrier bottom. These electrons eventually must be heated to the ambient plug electron temperature, resulting in an increase in required plug electron cyclotron range of frequencies (ECRF) heating. This is shown in Fig. III.3-1, where total input power and plug ECRF power are plotted vs. fraction of pumping done by the low energy pump beam. Two stage ions are not counted in this graph. For the same reason, although the powers involved are relatively small for the reference case, it is advantageous to aim the footprint of the beam pumping the barrier bottom to overlap the central cell side of the potential to as great a degree as possible.

The medium energy pump beam, which does the majority of the barrier pumping, plays a large role in central cell ion energy and particle balance. If, as in the case where there is no auxiliary heating of central cell ions, it functions as the primary source controlling each of those processes, there

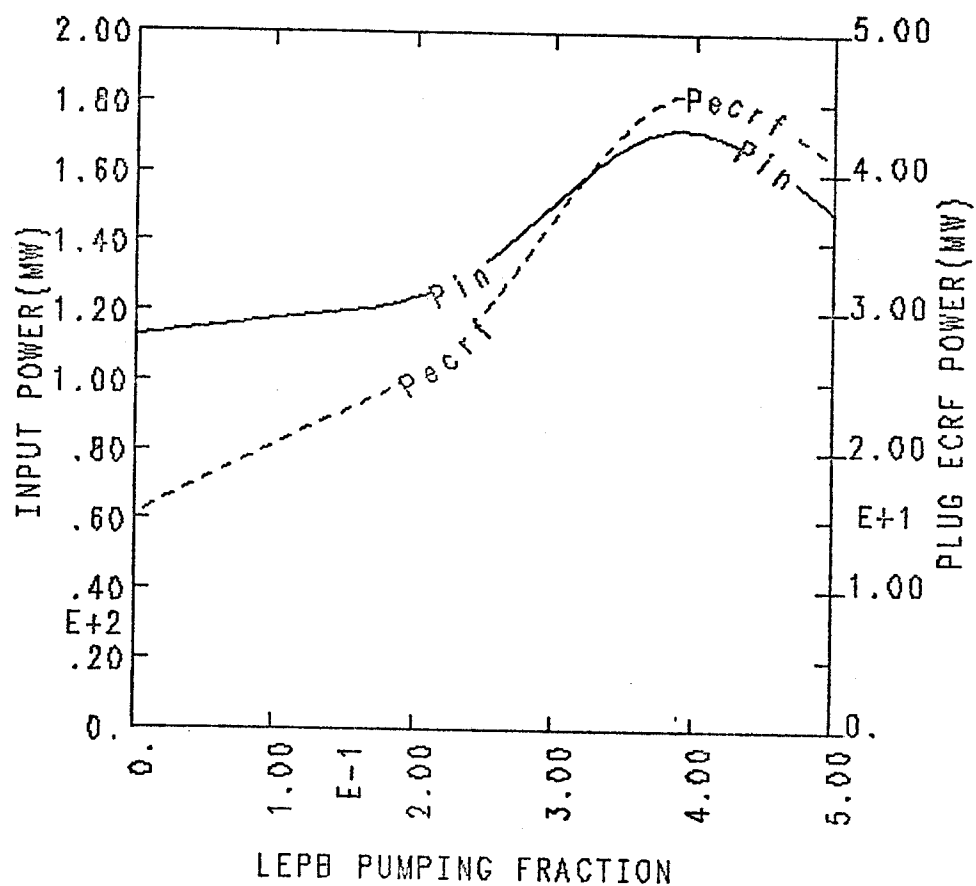


Fig. III.3-1. Total input power and plug ECRF power vs. fraction of pumping done by the low energy pump beam.

exists little flexibility in choosing an operating point and it is difficult to find an attractive reference case. Figure III.3-2 shows the trade-off between total input power, medium energy pump beam energy, and central cell ion cyclotron range of frequencies (ICRF) heating power. In addition, there is a tradeoff between the complexity of adding an ICRF launching system and the difficulties of building a very high power neutral beam system in a region where space is limited.

The high energy beam which pumps the bottom of the barrier is very difficult to fit physically into the machine, as discussed in Section IV.3. This is chiefly due to the small angle with respect to the magnetic field which the beam must achieve. In order to reach the low energy pumping point at a magnetic field of 3 T, the beam must fit within a loss cone of  $31^\circ$ . Since the energy required for the beam is governed by Eq. (III.2-18), values approaching the loss cone are difficult to achieve. Figure III.3-3 shows the dependence of input power on high energy beam angle.

Injection of neutral beams into the plug is somewhat easier in terms of access, but achieving the required energy is very difficult without invoking negative ion technology. High energy is required in order to overcome the effective potential  $(\phi_c + \phi_e)/(R_p \sin^2 \theta_{in} - 1) = 114 \text{ keV}$ .  $R_p$  is the beta-corrected plug mirror ratio. For reasonable plug beam input power, neutral beam energies greater than 200 keV are required. The dependence of total input power and plug power on injection energy for injection at a  $90^\circ$  angle is shown in Fig. III.3-4. Injection angle is also critical since a sloshing ion distribution is necessary for plug microstability as discussed in Section III.1. Powers do not vary greatly until the loss cone is approached; they then rise sharply. Total input power and plug neutral beam power are given as a

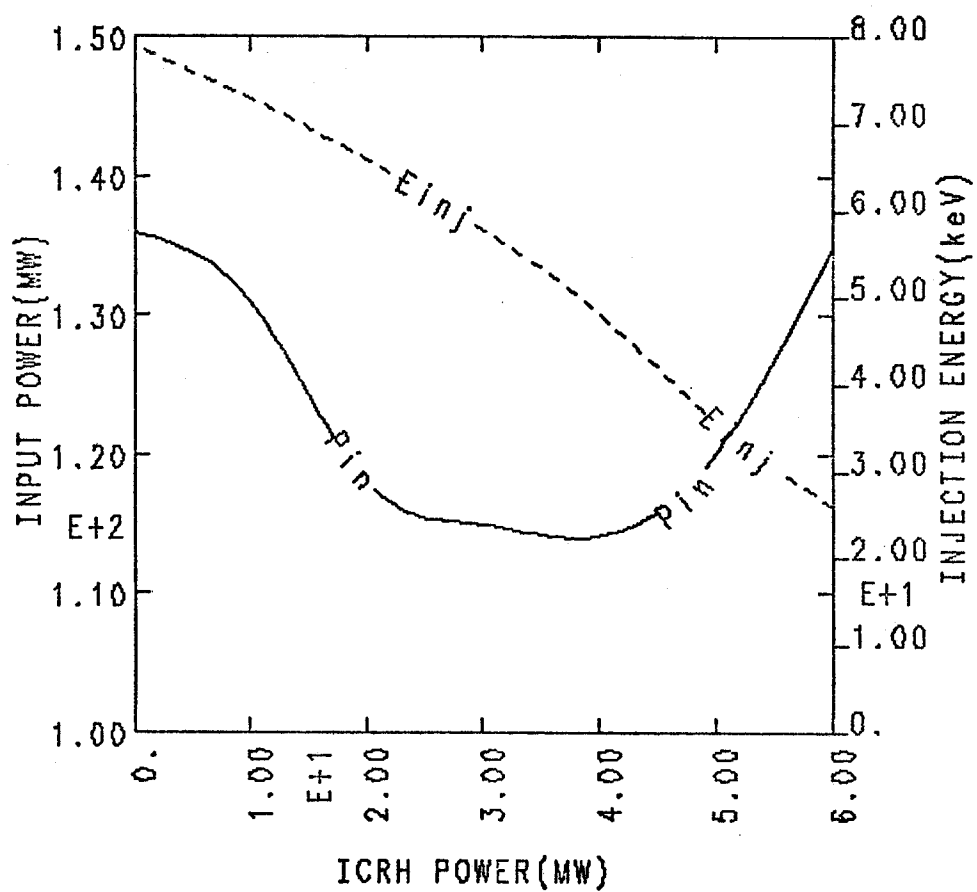


Fig. III.3-2. Total input power and medium energy pump beam energy vs. ICRF power.

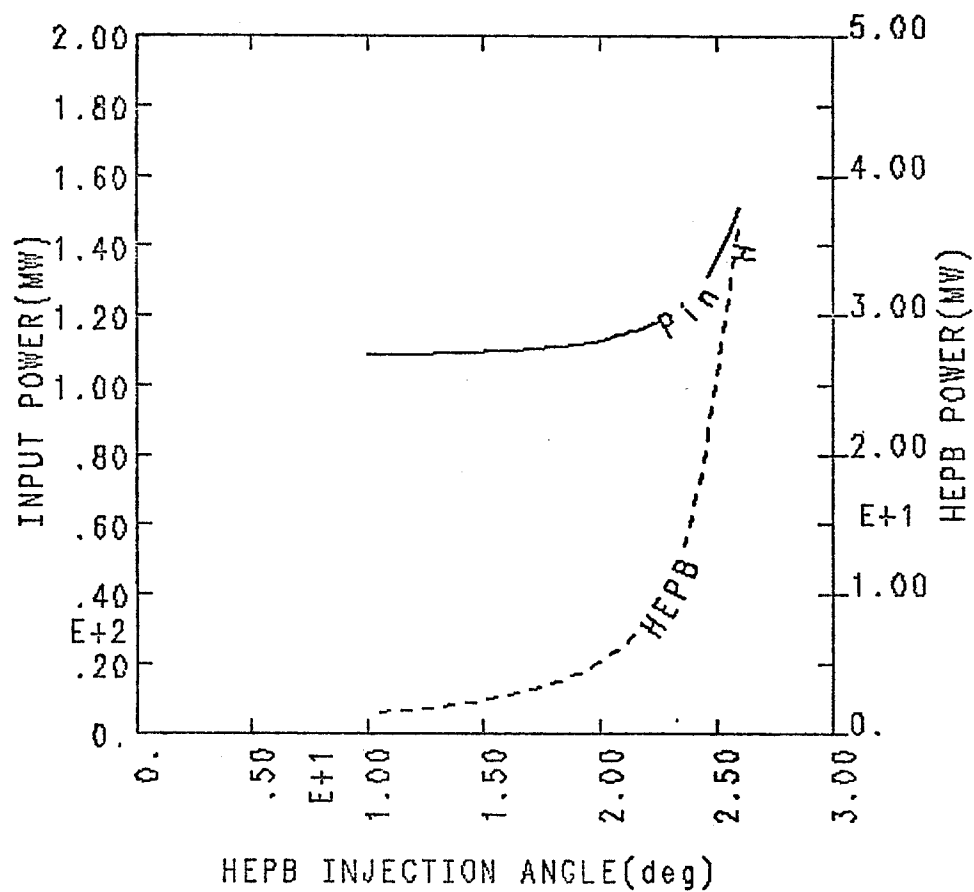


Fig. III.3-3. Total input power and high energy pump beam power vs. high energy beam injection angle.



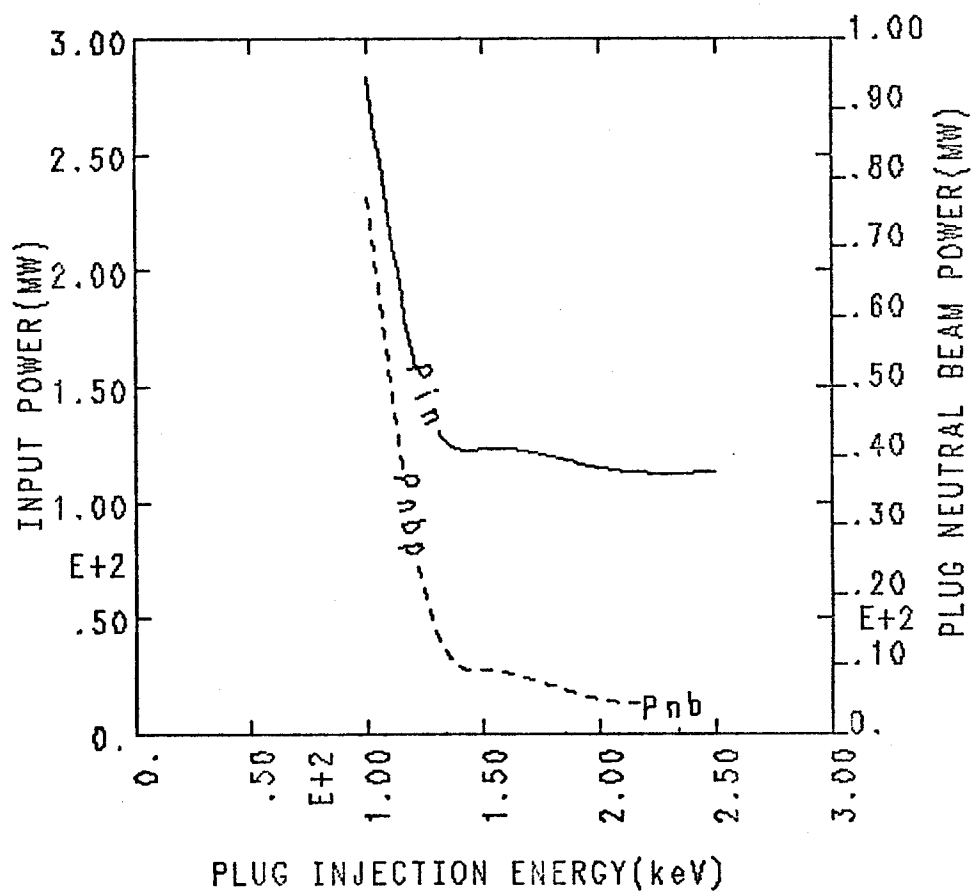


Fig. III.3-4. Total input power and plug beam power vs. plug injection energy for injection at a 90° angle.

function of plug beam injection angle in Fig. III.3-5. A possible means of achieving high plug beam energy without invoking negative ion technology is the use of non-hydrogen beams. Since approximately 100 keV/amu is attainable with positive ion technology, even helium could reach the desired energies. Suitably modifying the plug physics,<sup>(3)</sup> helium beams lead to a total machine input power of about 135 MW. As discussed in Section III.1 and Ref. 3, there is little chance for the He atoms to contaminate the central cell since the plug ions are well-separated from the rest of the machine in phase space.

Flexibility exists in choosing an operating point in terms of total input power and wall loading. The maximum wall loading attainable depends, to a great extent, on how high an input power level is allowed by economic or engineering constraints. The trade-off is shown in Fig. III.3-6 for barrier mirror ratios of 32.5, 25, and 17.5. The minimum barrier field is kept constant at 0.8 T here. To achieve the reference case mirror ratio of 25, a copper insert has been added to the superconducting barrier coil as discussed in Chapter V. Thus, the penalties for using a high barrier mirror ratio are that the technology becomes more complicated and difficult, and that power must now be expended to run the normal-conducting copper insert. Figure III.3-6 shows that there is a net gain in going from the  $R_b = 17.5$  case where no copper insert is needed to the  $R_b = 25$  case where approximately 60 MW of extra line power is needed, since the power needed to run the copper inserts is much lower grade and less expensive than that needed for neutral beams or RF sources.

Two other variations of barrier fields have been investigated. First,  $B_{\text{maxb}}$  was held constant while  $B_b$  was lowered. The resulting higher mirror ratios give a smaller trapping rate into the potential well since barrier

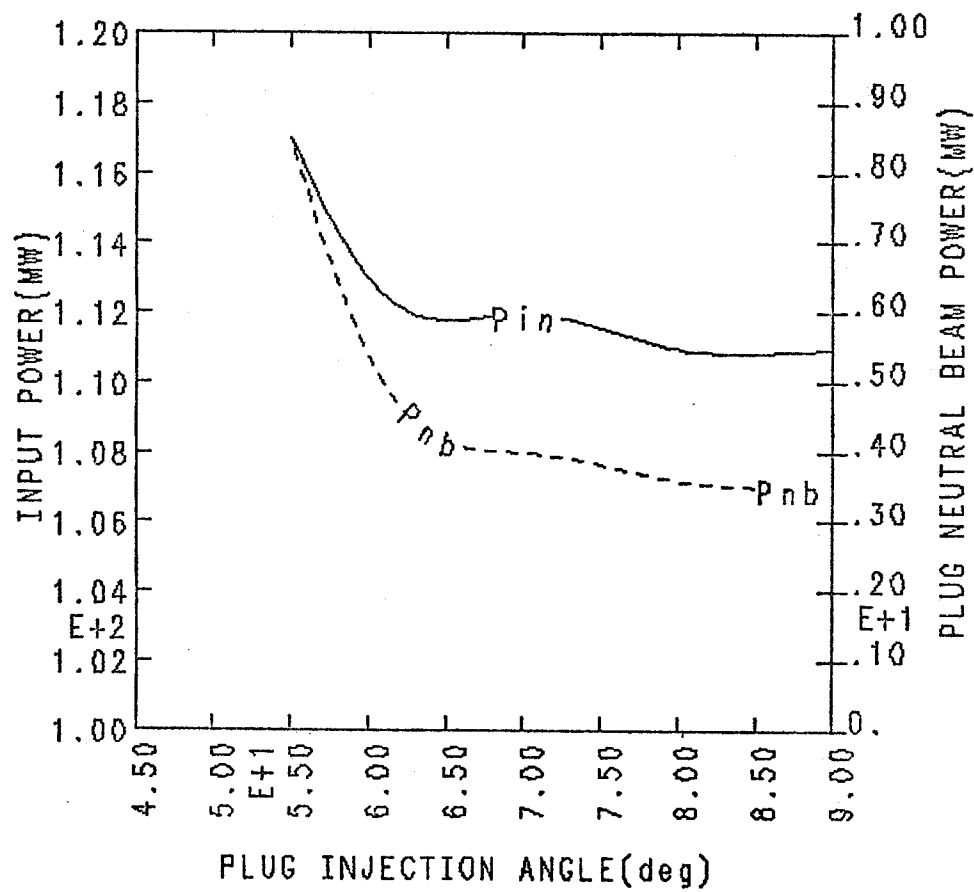


Fig. III.3-5. Total input power and plug beam power vs. plug injection angle.

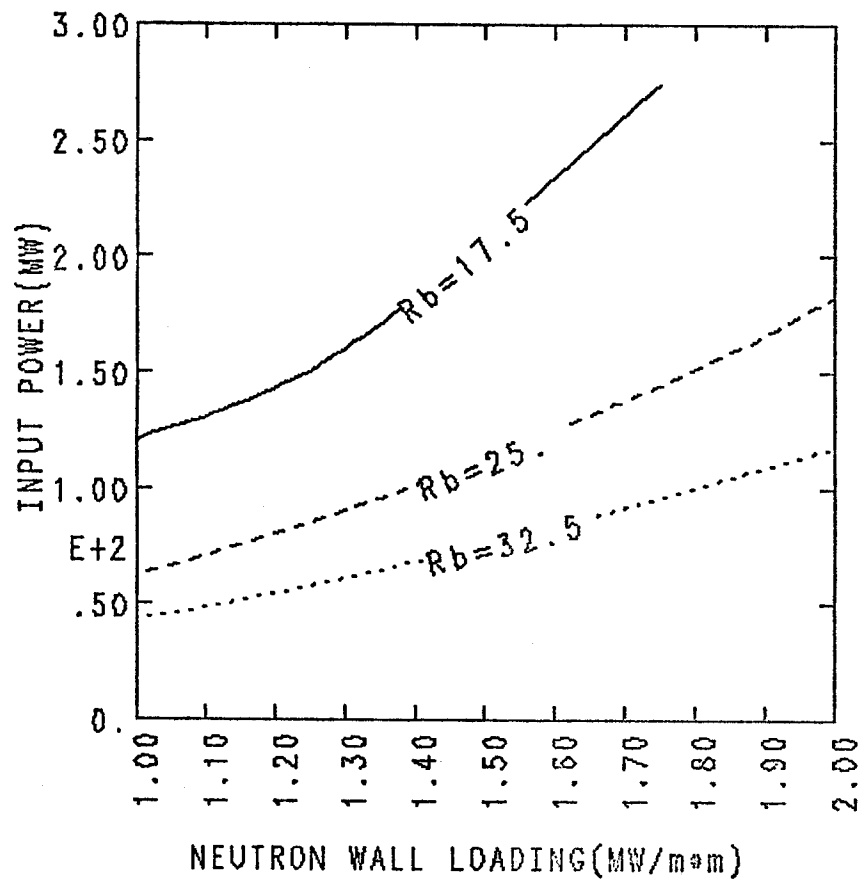


Fig. III.3-6. Total input power vs. neutron wall loading for barrier mirror ratios of 17.5, 25, and 32.5

density is reduced but, unfortunately, the beam trapping fraction is likewise reduced and no total power savings is achieved. Second, the barrier mirror ratio was held constant while varying the magnitudes of  $B_{\text{maxb}}$  and  $B_b$ . Changing these from 20 T and 0.8 T to 17.5 T and 0.7 T led to an increase in total input power of 11 MW. Lower field cases required even more power. This result is due to slightly poorer mirror confinement of central cell particles and to the larger mirror ratio which the two-stage pumping process must then overcome.

Central cell beta is important because high beta allows efficient utilization of the magnetic field. As central cell beta rises, total input power drops, as shown in Fig. III.3-7. Similar behavior occurs as  $B_c$  varies, a low  $B_c$  implying low density by Eq. (III.2-2). Input power and central cell density vs.  $B_c$  are plotted in Fig. III.3-8.

Central cell alpha particle density is allowed to reach an equilibrium level as described in Section III.2. Figure III.3-9, where the alpha particle fraction is legislated to various values, shows that only a few percent of alpha particles are tolerable before machine performance is seriously degraded.

The reference point design assumes that essentially all central cell particle loss will be end loss. A very small amount of classical, collisional radial transport will also occur. Some resonant transport<sup>(4)</sup> may take place but, since this is driven by magnetic flux tube azimuthal asymmetry in particle bounce regions and most central cell ions turn near the barrier solenoids, little effect is expected. Another possible cause of transport is that the use of ferritic steel in central cell blanket regions of finite length gives non-circular flux tubes, as discussed in Section V.9. When mapped into the

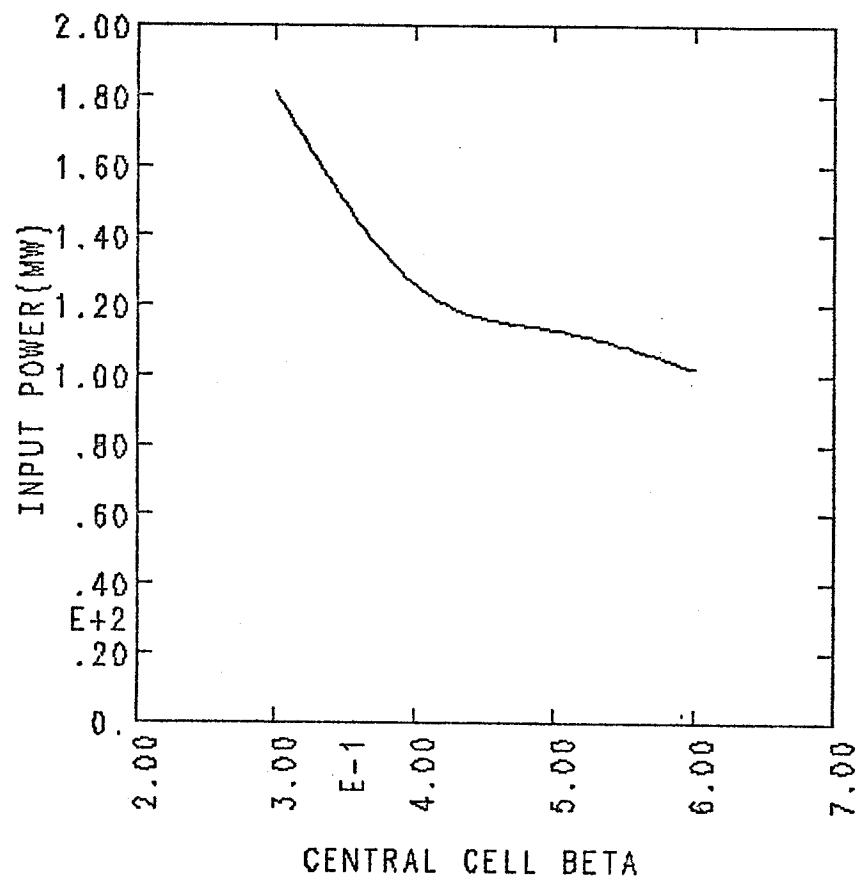


Fig. III.3-7. Total input power vs. central cell beta.

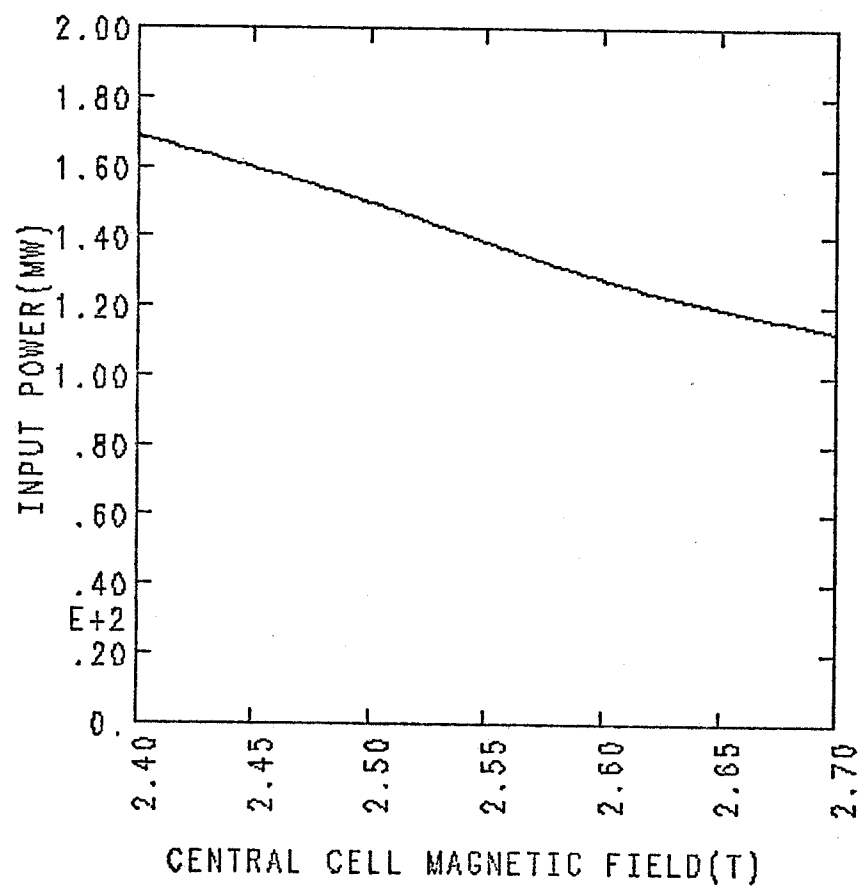


Fig. III.3-8. Total input power vs. central cell magnetic field.

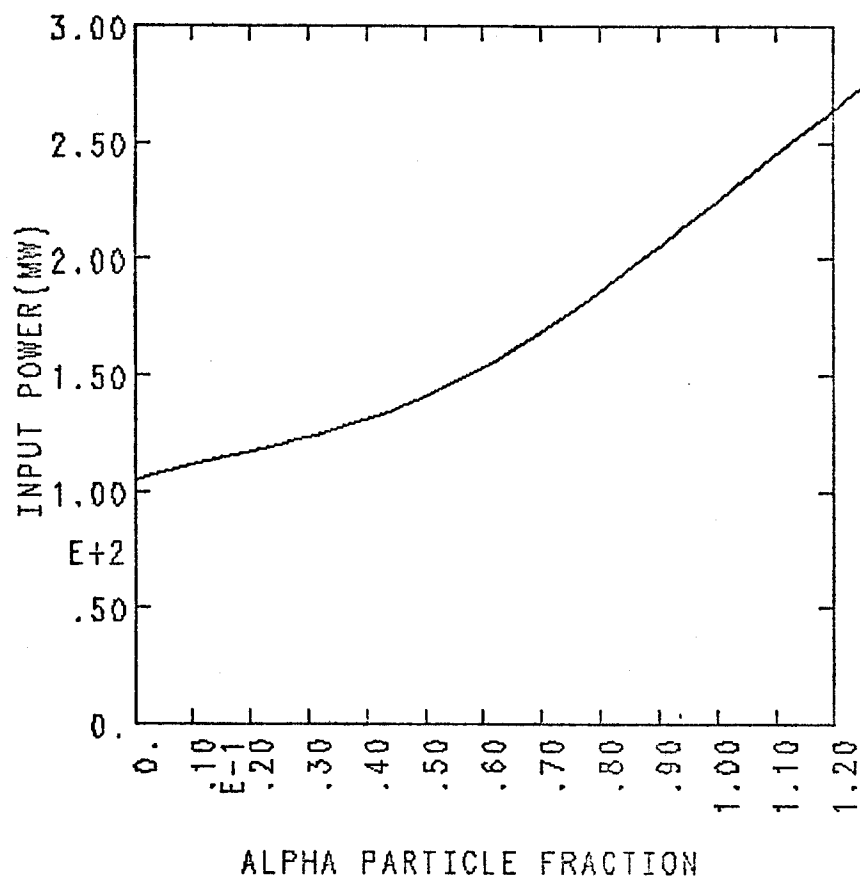


Fig. III.3-9. Total input power versus central cell alpha particle fraction.



bounce regions, this asymmetry is a small perturbation of the magnetic field and thus should have a small effect on resonant transport.<sup>(5)</sup> However, the large radial excursions due to flux tube asymmetry can lead to enhanced diffusion since the random walk step size is proportional to the excursion. Estimates of the diffusion and allowed asymmetry are discussed in Appendix III.3-A. The effect of cross field transport on total input power is shown in Fig. III.3-10. Surprisingly, a cross field loss time equal to only four times the end loss time does not lead to a large increase in input power.

The central cell ion temperature has a broad optimum at about 30 keV. Two major effects are involved: 1) For low temperatures, the trapping rate into the barrier is high and barrier pumping power becomes excessive; and 2) for high temperatures, required neutral beam injection energies become large since potentials vary approximately as temperatures. The dependence of total input power and plug injection energy on central cell ion temperature is shown in Fig. III.3-11.

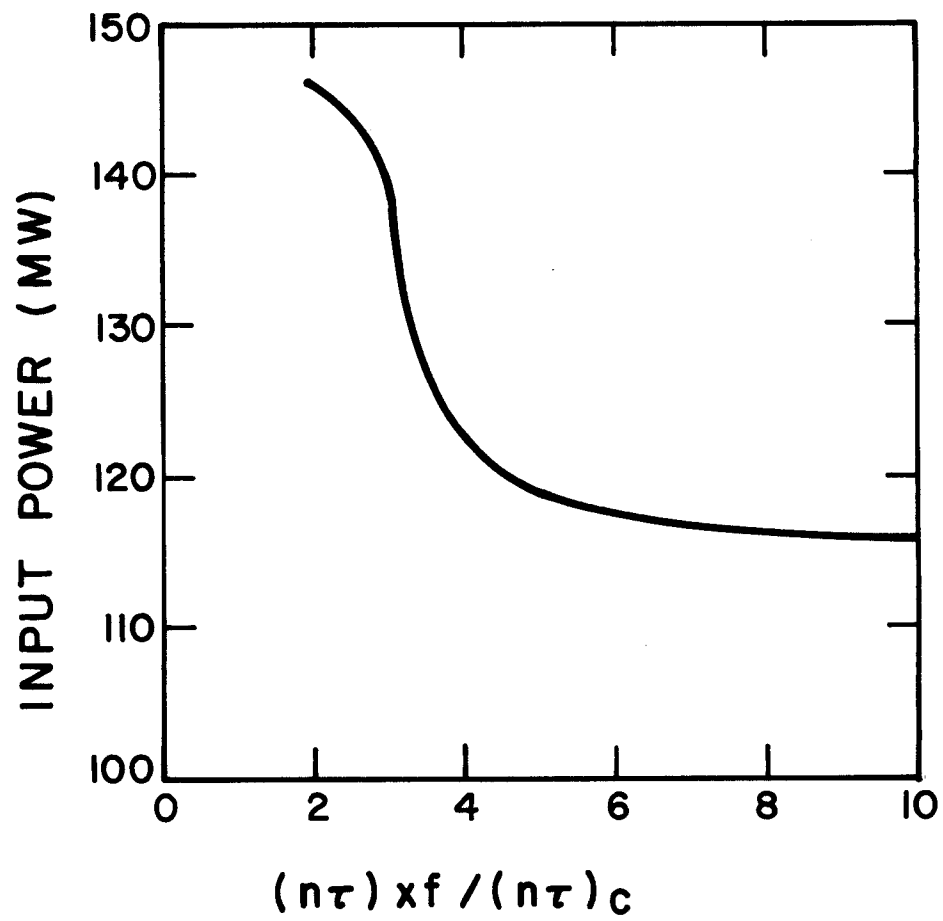


Fig. III.3-10. Total input power versus  $(n\tau)_{xf} / (n\tau)_c$  ( $\equiv$  cross field diffusion time/end loss time).

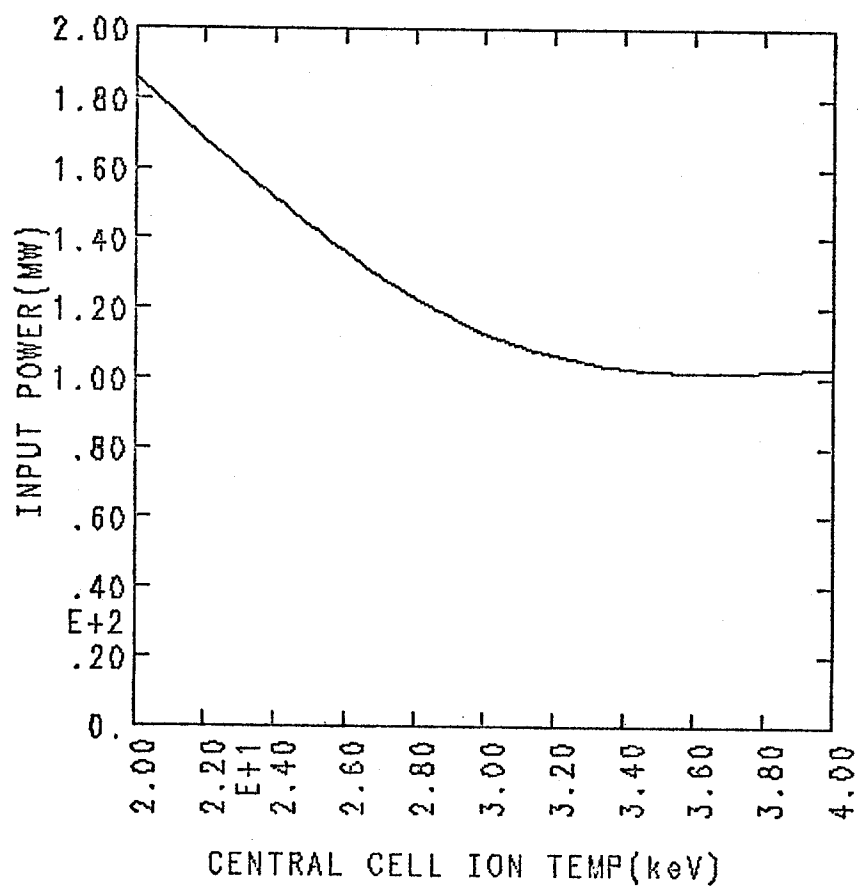


Fig. III.3-11. Total input power and plug injection energy vs. central cell ion temperature.

### References for Section III.3

1. B. Badger, et al., "WITAMIR-I, A Tandem Mirror Reactor Design Study," University of Wisconsin Fusion Engineering Program Report UWFD-400 (1980).
2. D.E. Baldwin, B.G. Logan, and T.C. Simonen, "Physics Basis for MFTF-B," Lawrence Livermore National Laboratory Report UCID-18496 (1980).
3. D.E. Post, T.R. Grisham, J.F. Santarius, and G.A. Emmert, "Heavy Atom Neutral Beams for Tandem Mirror End Plugs," Princeton Plasma Physics Laboratory Report PPPL-1786 (1981) (submitted to Nuclear Fusion).
4. R.H. Cohen, "Analytic Approximation to Resonant Plateau Transport Coefficients for Tandem Mirrors," Nucl. Fus. 19, 1579 (1979).
5. B.G. Logan, private communication.

### Appendix III.3-A

#### Radial Transport Due to Central Cell Flux Tube Asymmetry

Some central cell asymmetry is expected due to ferritic steel in the blanket module, as discussed in Section V.9. This can enhance collisional diffusion since, while following a flux surface, a particle may experience an appreciable radial excursion between collisions. A rough estimate for diffusion due to a general asymmetry, plus a calculation for a TASKA blanket will be presented here.

For the reference case,

$$\tau_{\theta} \lesssim \tau_b \ll \tau_i \quad (1)$$

where  $\tau_{\theta}$  is the azimuthal drift time,  $\tau_i$  is the central cell ion collision time,  $\tau_b$  is the ion bounce time. A gradient of the magnetic field,  $B$ , then gives a radial drift velocity of

$$V_{dr} = \frac{T_{ic} c}{q} \left( \frac{\vec{B} \times \nabla B}{B^3} \right)_r \sim \frac{\mu c}{q r_c} \frac{\tilde{B}}{B} \quad (2)$$

where  $\mu$  is the ion magnetic moment,  $q$  is the ion charge,  $c$  is the speed of light, and  $\tilde{B}$  is the magnetic field perturbation over an angle of  $\theta \sim 1$ . For TASKA parameters,

$$V_{dr} \sim 4.8 \times 10^6 \frac{\tilde{B}}{B} \text{ cm/sec} . \quad (3)$$

Since  $\tau_{\theta} \lesssim \tau_b$ , the random walk radial diffusion process can be looked at as occurring with step size

$$\delta r \sim \frac{V_{dr}}{4} \tau_{\theta} \sim 7.7 \frac{\tilde{B}}{B} \text{ cm} \quad (4)$$

where the factor of 4 is a crude attempt at averaging. Then, assuming the randomization occurs on the collisional time scale, the diffusion coefficient due to  $\tilde{B}$  may roughly be taken to be

$$D \sim \frac{(\delta r)^2}{\tau_i} \sim 2.1 \times 10^3 \left(\frac{\tilde{B}}{B}\right)^2 \frac{\text{cm}^2}{\text{sec}} \quad (5)$$

and the radial loss time is

$$\tau_r \sim \frac{r_c^2}{2D} \sim 0.24 \left(\frac{B}{\tilde{B}}\right)^2 \text{ sec} . \quad (6)$$

The axial loss time is

$$\tau_{ic} = \frac{(n\tau)_{ic}}{n_c} = 0.28 \text{ sec} . \quad (7)$$

For good confinement, a rough estimate is  $\tau_r \sim 10 \tau_{ic}$  which leads to an allowed value of

$$\frac{\tilde{B}}{B} \lesssim 0.29 .$$

This should be easy to satisfy, although a more detailed analysis is needed before definite conclusions are drawn.

In any case, the TASKA blanket designs lead to much lower values of  $V_{dr}$ , as shown by  $\tilde{B} \times \nabla B/B^3$  values presented in Section V.9. The reference case,

almost-symmetric blanket module leads to a maximum  $D \sim 10^{-5} \text{ cm}^2/\text{sec}$ . This is less than the estimate given in Eq. (5) unless  $\tilde{B}/B$  is negligibly small.

Thus, radial diffusion due to central cell magnetic flux tube asymmetry induced by ferritic steel blanket modules seems to be of little concern for TASKA or similar devices.

### III.4 TASKA Design Point

The plasma physics reference point parameters were set on July 31, 1981 so that work on related engineering systems could proceed. TASKA design point parameters are given in Tables III.4-1, III.4-2, and III.4-3. Note that the physics model is based on present theoretical knowledge of mirror plasmas, some of which has not yet been completely experimentally verified. Experiments currently under construction will address these questions. Major changes in theoretical understanding would, of course, alter the design point significantly.

The basic philosophy behind the choice of these parameters was to find the simplest, least expensive system which would achieve the technological goals. Since one of the primary functions of TASKA is materials testing, a central cell neutron wall loading of at least  $1 \text{ MW/m}^2$  was sought while an attempt was made to minimize input power, magnet technology, physical size, beam technology, and ECRF technology. Because TASKA reaches the reactor regime for temperature and density, achieving a high plasma Q value is not a concern; large Q's are obtained primarily by using a long central cell.

Thus, the search for a TASKA operating point was constrained within fairly well-defined parametric boundaries by a number of variables. The chief boundaries were defined by the following considerations:

- 1) For economic reasons, the neutron wall loading of  $> 1 \text{ MW/m}^2$  had to be achieved by using on the order of or less than 100 MW of input power. Minimizing ECRF power is particularly important, since its cost seems likely to be higher than that for other sources.

- 2) For magnetic field coils on the 1990's time scale (see Chap. V) a maximum of 6.25 T on the yin-yang axis was deemed possible. The minimum of



Table III.4-1. TASKA Power and Machine Parameters

|                                 |                        |
|---------------------------------|------------------------|
| CC neutron wall load            | 1.52 MW/m <sup>2</sup> |
| Fusion power                    | 86. MW                 |
| CC plasma power density         | 17. MW/m <sup>3</sup>  |
| Q (fusion power/injected power) | 0.74                   |

Central Cell

|                         |        |
|-------------------------|--------|
| Length (peak to peak B) | 19.2 m |
| Plasma radius           | 0.32 m |
| Minimum wall radius     | 0.46 m |

Barrier

|                       |         |
|-----------------------|---------|
| Physical length       | 15.6 m  |
| Effective length      | 9.0 m   |
| Minimum plasma radius | 0.099 m |
| Average plasma radius | 0.20 m  |
| Maximum plasma radius | 0.50 m  |

Plug

|                 |        |
|-----------------|--------|
| Length          | 5.0 m  |
| Midplane radius | 0.28 m |

Magnetic Fields

|                 |        |
|-----------------|--------|
| Central cell    | 2.7 T  |
| Barrier maximum | 20. T  |
| Barrier minimum | 0.8 T  |
| Plug maximum    | 6.25 T |
| Plug minimum    | 4. T   |

Table III.4-2. TASKA Heating Parameters

Neutral Beams

|          |                   |               |
|----------|-------------------|---------------|
| Plug:    | power             | 5.4 MW        |
|          | energy            | 250. keV      |
|          | angle             | 60°           |
|          | species           | p             |
|          | trapping fraction | 0.21          |
| Barrier: | power             | 6.6 MW        |
|          | energy            | 76 keV        |
|          | angle             | 20°           |
|          | species           | d             |
|          | trapping fraction | 0.42          |
| Barrier: | power             | 49.7 MW       |
|          | energy            | 50 keV        |
|          | angle             | 25°           |
|          | species           | 0.44 d/0.56 t |
|          | trapping fraction | 0.95          |
| Barrier: | power             | 0.2 MW        |
|          | energy            | 2. keV        |
|          | angle             | 45°           |
|          | species           | d             |
|          | trapping fraction | 0.99          |

ECRF

|       |                       |         |
|-------|-----------------------|---------|
| Plug: | power                 | 14.9 MW |
|       | frequency             | 56. GHz |
|       | absorption efficiency | 1.      |

ICRF

|               |                       |        |
|---------------|-----------------------|--------|
| Central cell: | power                 | 40. MW |
|               | frequency             | 30 MHz |
|               | absorption efficiency | 0.8    |

|                             |         |
|-----------------------------|---------|
| <u>Total Injected Power</u> | 117. MW |
|-----------------------------|---------|

NB: Plug and barrier powers are total powers for both sides.

Table III.4-3. TASKA Plasma Physics Parameters

Central Cell

|                        |                                                  |
|------------------------|--------------------------------------------------|
| Density                | $1.94 \times 10^{14} \text{ cm}^{-3}$            |
| Alpha particle density | $1.6 \times 10^{12} \text{ cm}^{-3}$             |
| Ion temperature        | 30. keV                                          |
| Electron temperature   | 11.5 keV                                         |
| Beta                   | 0.5                                              |
| $(n\tau)_{ic}$         | $5.4 \times 10^{13} \text{ cm}^{-3} \text{ sec}$ |
| $(n\tau)_{ec}$         | $5.3 \times 10^{13} \text{ cm}^{-3} \text{ sec}$ |
| Potential, $\phi_c$    | 42.8 keV                                         |

Barrier

|                                              |                                      |
|----------------------------------------------|--------------------------------------|
| Minimum density                              | $6.8 \times 10^{12} \text{ cm}^{-3}$ |
| Beta                                         | 0.054                                |
| Potential, $\phi_b$                          | 37.5 keV                             |
| Pumping parameter, $g_b$                     | 2.                                   |
| High energy pumping fraction                 | 0.025                                |
| Medium energy pumping fraction               | 0.975                                |
| Low energy pumping fraction (two stage ions) | 0.109                                |

Plug

|                              |                                                  |
|------------------------------|--------------------------------------------------|
| Midplane density             | $6.3 \times 10^{13} \text{ cm}^{-3}$             |
| Mean ion energy              | 388. keV                                         |
| Electron temperature         | 59.3 keV                                         |
| Beta                         | 0.64                                             |
| $(n\tau)_{ip}$               | $2.9 \times 10^{13} \text{ cm}^{-3} \text{ sec}$ |
| $(n\tau)_{ep}$               | $7.5 \times 10^{11} \text{ cm}^{-3} \text{ sec}$ |
| Potential, $\phi_c + \phi_e$ | 109. keV                                         |
| Cohen parameter, $v_c$       | 0.5                                              |

4 T was chosen to give a reasonably high mirror ratio at which to aim the sloshing ion neutral beam, while allowing a fairly high plug density which allows a high central cell density. On the solenoidal barrier coil, 14 T on axis is created by the superconductor; the total possible barrier maximum field then depends on trade-offs between the field desired, power costs in the normal conducting insert coil, and neutron shielding needed. An operating point of 20 T was chosen as a suitable compromise, for reasons discussed in Chapter V and Section III.3.

3) Constraints are also associated with the neutral beam systems. As mentioned in Sections III.3 and IV.3, injection angles are extremely critical. The medium energy pump beam requires a 10 cm plasma radius for all of the beam to intersect the plasma; when magnetic flux tube conservation is taken into account, this translates into a central cell plasma radius of at least 32 cm for  $B_c = 2.7$  T. Injection energy is also a problem, particularly for the plug beam where energies greater than 200 keV are required (cf. Fig. III.3-4). The use of helium plug beams was considered because positive ion neutral beam technology could be used, as mentioned in Section III.3, but was rejected, primarily because of vacuum pumping difficulties.

4) Fueling the plasma also creates something of a dilemma, since the high plasma densities and temperatures make source penetration very difficult. For TASKA, since neutral beam ionization does a large amount of fueling, plasma and power parameters are adjusted to give particle loss equal to neutral beam fueling. This somewhat limits flexibility in choosing operating points, but more than balances difficulties with gas puff or pellet fueling. Since the low energy and high energy pump beams do only a small amount of the fueling, only the medium energy beam contains significant amounts of tritium,

with the percentage of tritium to deuterium set to give 50% total fueling of each species. This obviates the necessity of handling large amounts of tritium in the low and high energy pump beam systems.

A central cell neutron wall loading of  $1.5 \text{ MW/m}^2$  was chosen since the total input power requirement of 117 MW is not prohibitive. The central cell length was set by consideration of space requirements for materials testing and tritium breeding.

Because of magnetic flux tube conservation, the plasma radius decreases in the central cell end regions, thus decreasing fusion power output and neutron wall loading there. This is modelled here by using an effective central cell length of 15.7 m to compute central cell volume.

Central cell wall radius was defined by the requirement that there be a distance of two 3.5 MeV alpha particle Larmor radii between the plasma and the wall; this insures that alpha particle orbits do not intersect the wall. Unfortunately, after the plasma physics parameters were set, it was discovered that the Larmor radius used was too small by a factor of  $\sqrt{2}$ . For a flat density profile, the resultant flux of alpha particles can seriously damage the first wall. However, increasing the first wall radius by 4 cm removes the problem without greatly modifying the design.

Central cell ion cyclotron range of frequencies heating (ICRF) is used because, as shown in Fig. III.3-2, it reduces total power and beam injection energy. Also, ICRF technology needed for TASKA is state-of-the-art and would require very little development (see Section IV.2). Reducing power requirements for the medium energy pump beam is a significant advantage, too, since even the reference case beams for TASKA are non-trivial.

The central cell ion temperature was chosen as a compromise between low temperature, where the barrier ion trapping rate becomes formidable, and high temperature, where required pumping energies become very high.

Plug beta was chosen as a typical value for this type of magnetic field configuration. Central cell beta was chosen somewhat conservatively, since finite Larmor radius MHD theory is still in a state of flux.

The barrier pumping parameter,  $g_b$ , was chosen equal to two as a reasonable value, but detailed calculations of  $g_b$  are very difficult and have not been attempted here.

In summary, the TASKA design point is a major step beyond present devices, but is an achievable and reasonable one. The inside-barrier tandem mirror configuration thus appears to possess good potential as an engineering and materials test facility.

### III.5 The DT Fuel Ion Density in the Thermal Barrier and Plug

The TASKA tandem mirror configuration utilizes an inboard thermal barrier with a yin-yang plug. The plug ions were chosen to be protons in order to avoid fusion reactions in the plug and thereby reduce the neutron shielding requirements for the yin-yang magnets and the degree of neutron streaming through penetrations, such as neutral beam lines, and into the direct converter. The fuel ions, deuterium and tritium, from the central cell pass through the thermal barrier and are reflected by the rising potential in the plug. This will produce some rate of DT fusion in the barrier and plug; the 14 MeV neutron source in the plug cannot be entirely eliminated by choosing a non-reacting plug plasma. In this section we estimate the density of deuterium and tritium in the barrier and plug in order to provide a neutron source for neutronics calculations.

In a previous report,<sup>(1)</sup> we calculated the potential profile in the thermal barrier. Implicit in this was the calculation of the ion density profile, although explicit results were not obtained. Here we utilize the same formalism to calculate explicitly the fuel density (deuterium plus tritium) in the barrier and plug.

The barrier and plug region is divided axially into three regions, as shown in Fig. III.5-1. Region I spans from the barrier throat (peak magnetic field) to the point of minimum field. In this region the electrons are assumed to be Maxwellian with temperature  $T_{ec}$ , which is the same as in the central cell. The passing ions are assumed to be Maxwellian in the appropriate region of velocity space. The trapped ions in the barrier will not be calculated explicitly, although their effect is discussed later.

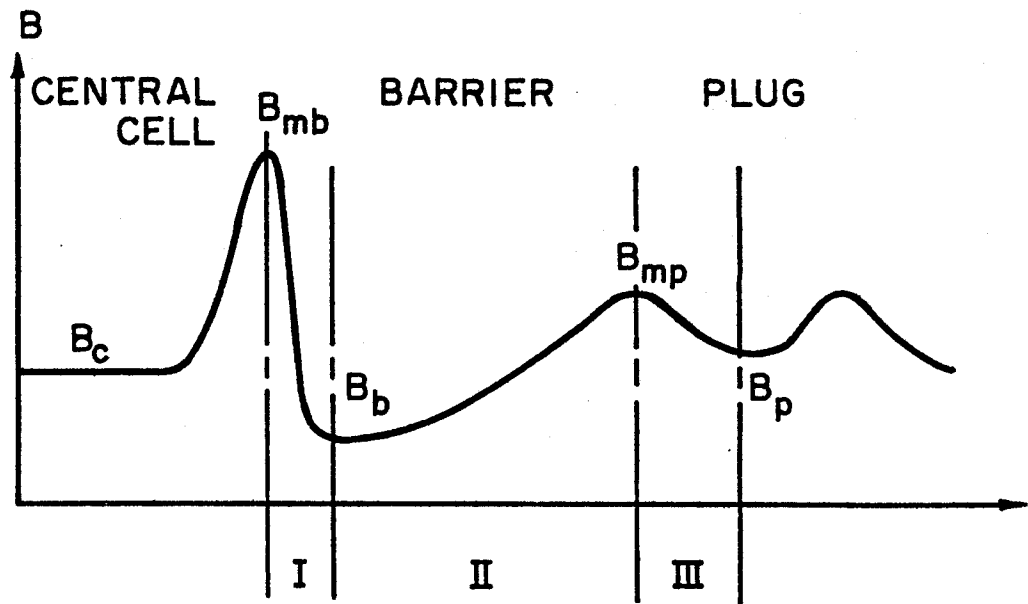


Fig. III.5-1. Axial profile of the magnetic field in the inboard thermal barrier and plug.



Region II spans from the barrier minimum to the throat of the plug. The ion distribution function in this region is determined by the choice in Region I; it cannot be chosen independently. The electrons in this region consist of three "classes": electrons coming from the central cell, from the plug, and electrons trapped by the barrier potential at one end and by the magnetic field,  $B_{mp}$ , at the other end. The latter class is analogous to Yushmanov<sup>(2)</sup> trapped ions in mirrors, and are referred to here as Yushmanov electrons. The phase space for electrons in Region II is shown in Fig. III.5-2. The electron distribution function is taken to be piece-wise Maxwellian at different temperatures. Electrons in zones M and N of Fig. III.5-2 come from Region I and are therefore assigned the temperature,  $T_{ec}$ . Electrons in zone K come from the plug and are therefore assigned the plug temperature,  $T_{ep}$ . Electrons in zone L are the Yushmanov electrons; they are assigned the temperature  $T_{ec}$  under the assumption that their energy exchange is more rapid with central cell electrons than with plug electrons.

Region III is between the plug mirror throat and the plug midplane. As long as the potential profile in Region III is not calculated, we need not specify the hot ion distribution function in the plug. The midplane electron density and potential are needed, however, to calculate the contribution of plug electrons to Region II. These are taken as input data.

The calculational procedure is to integrate over the ion and electron distribution functions to obtain the local electron and ion density as functions of the local magnetic field and potential. These are shown in Figs. III.5-3 and III.5-4. The assumption of quasi-neutrality ( $n_i(\phi, B) = n_e(\phi, B)$ ) then determines explicitly the potential and ion density as functions of  $B$ .

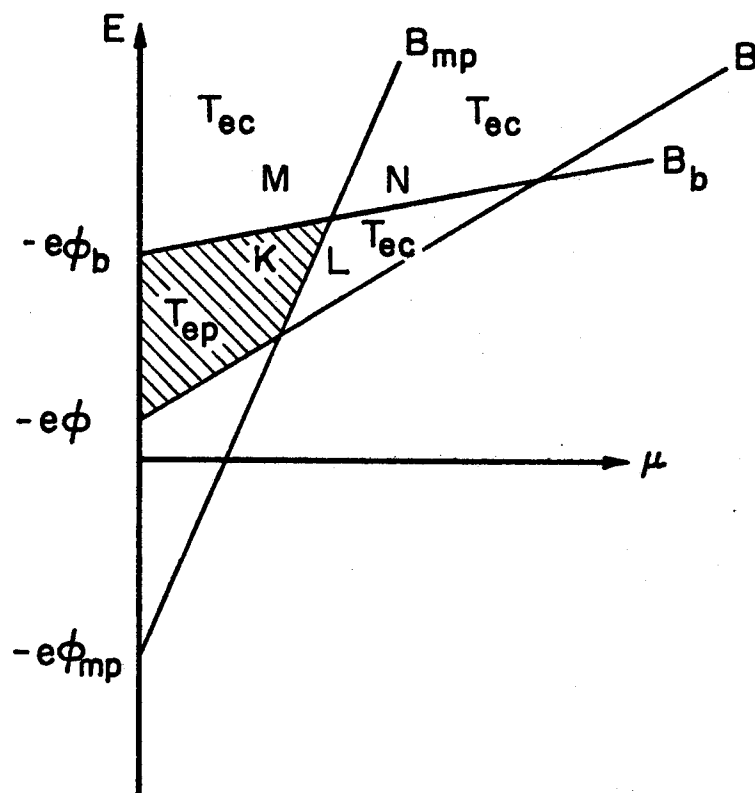


Fig. III.5-2. Phase space for electrons in Region II.

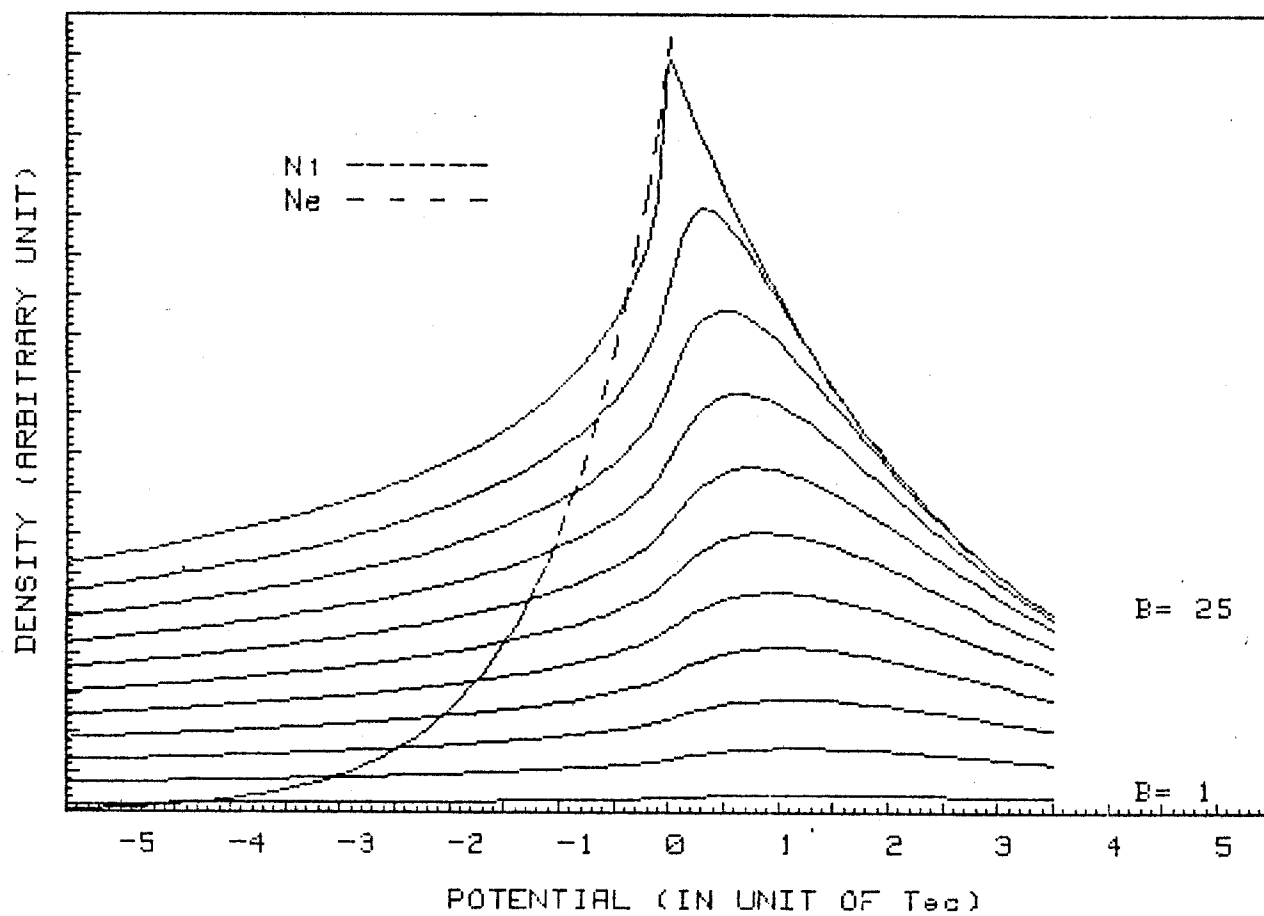


Fig. III.5-3. Ion and electron density versus potential and magnetic field strength (normalized to  $B_0$ ; increment in  $B = 2.4$ ) in Region I.

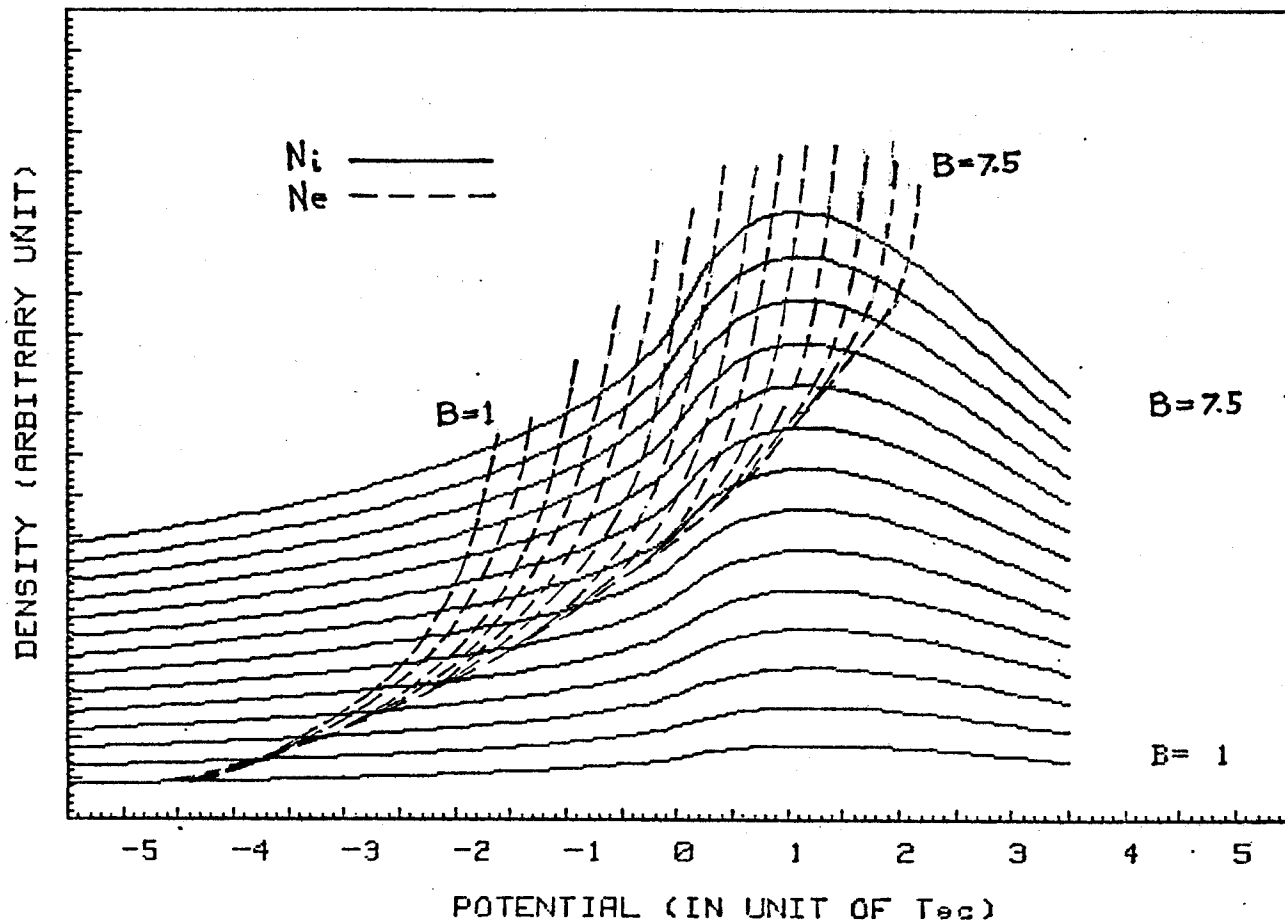


Fig. III.5-4. Ion and electron density versus potential and magnetic field strength (normalized to  $B_0$ ; increment in  $B = .5$ ) in Region II.

Note that the spatial coordinate does not enter explicitly, but only implicitly through  $B(z)$ .

The total ion density (deuterium plus tritium) and electrostatic potential profile are shown in Fig. III.5-5 for TASKA for the two choices for the ion distribution function. The ion density drops in Region I because of the expanding magnetic field and the falling potential, as expected. In Region II the ion density first rises, because of the increasing magnetic field and potential, but then peaks and starts to fall while the potential continues to rise. When the potential becomes positive ( $\phi = 0$  in the central cell is the reference value) passing ions from the central cell are reflected; this causes the density to fall. Since  $n_i = n_e$ , it is clear that the potential and electron density in Region II do not satisfy the Boltzmann relation. This is because the electron distribution function is non-Maxwellian; it is only piece-wise Maxwellian with the "colder" (i.e. temperature equal to  $T_{ec}$ ) electrons at the higher kinetic energy than the "hotter" electrons at temperature  $T_{ep}$ .

Ions trapped in the thermal barrier will add to the ion density shown in Fig. III.5-5. They will contribute most near the bottom of the barrier but should not contribute much where  $\phi > 0$ , since the trapped ions are mostly trapped by the potential. Consequently the trapped ions should not affect the magnitude or location of the density peak near the throat of the yin-yang magnet. Since the neutron source rate is proportional to the square of the density, the peak value and location are more important for neutronics calculations. The peak density in the barrier is about one-fifth the central cell density. The density at the barrier minimum is a factor of 70 lower than the central cell density. The significance of this density peak as a neutron

# AXIAL DT DENSITY PROFILE TASKA

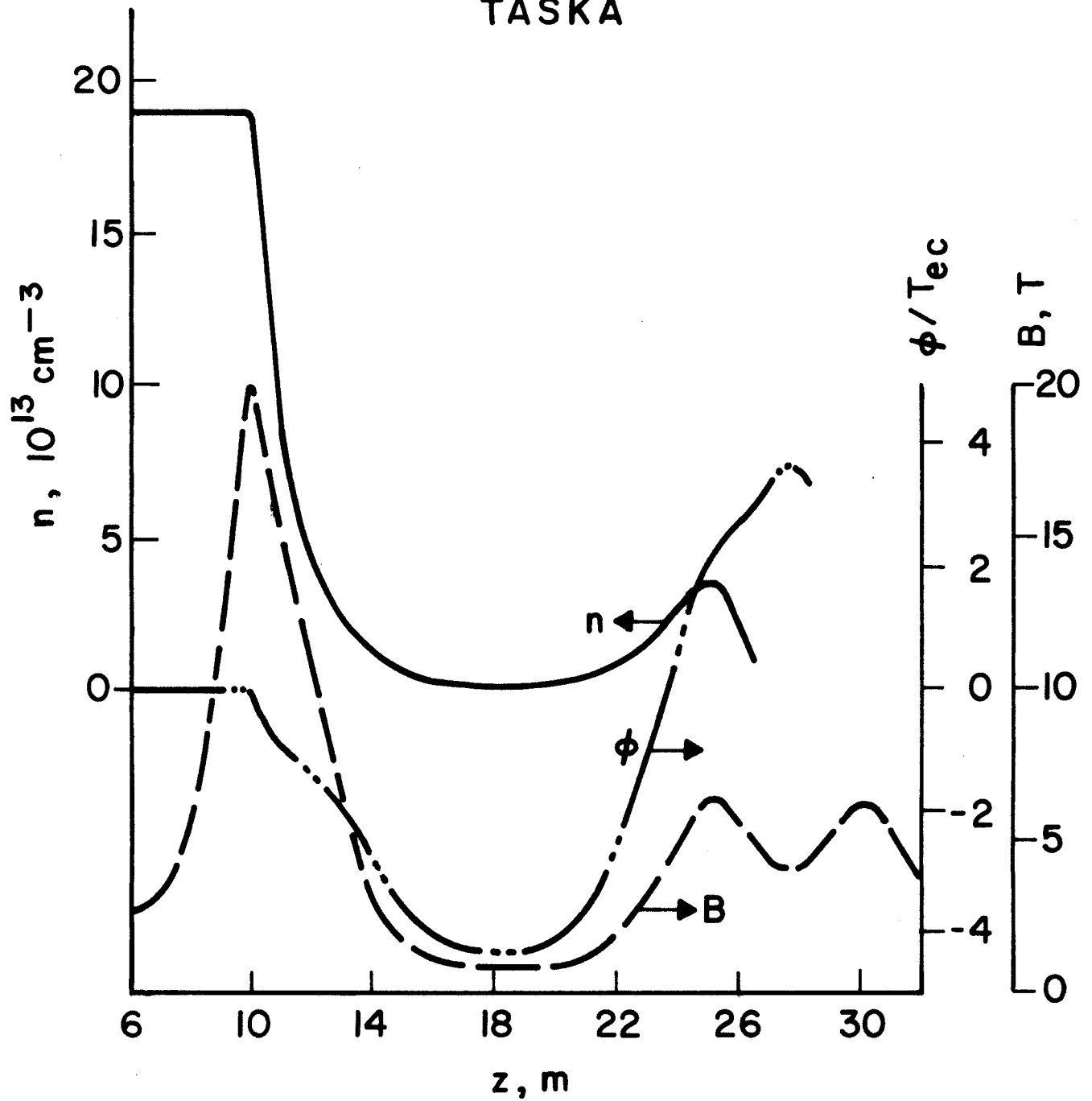


Fig. III.5-5. Axial profile of the potential and passing ion density in the thermal barrier and plug of TASKA.

source depends on its location relative to beam lines and other penetrations and is determined by detailed three-dimensional neutronic calculations.

A similar calculation<sup>(3)</sup> has been done for the WITAMIR-I<sup>(4)</sup> tandem mirror power reactor, which utilizes the same basic end-plug configuration. The density profile in TASKA is similar to that in WITAMIR-I except that the density peak in TASKA is shifted towards the throat of the yin-yang mirror. This is because the ion confining potential,  $\phi_c$ , in TASKA is smaller relative to the central cell ion temperature, and consequently the central cell ions can penetrate further into the plug.

In addition to the neutron production in the barrier/plug due to fusion reactions between passing central cell deuterium and tritium ions, there is also neutron production due to fusion between the non-thermalized deuterium and tritium ions produced by the barrier pump beams and the plasma ions in the barrier. One can also get neutron production in the beam dump because of buildup of D and T in the surface of the beam dump. This is evaluated in Section VIII.3.

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### III.6 Plasma Wall Interactions and the "Halo" Plasma

In open-ended devices, such as mirrors and tandem mirrors, the primary plasma loss is parallel to the magnetic field (i.e., out the ends). Consequently the primary location for plasma-wall interaction problems is at the ends where the plasma is neutralized at thermal dumps or in a direct converter. With sufficient expansion of the magnetic field between the plugs and the thermal dumps and with adequate pumping near the thermal dumps, the thermal contact between the main plasma and the wall can be kept small, so that cooling of the hot plug electrons by parallel heat flow to the thermal dumps is not a problem. This has been achieved in TMX;<sup>(1)</sup> theoretical calculations<sup>(2)</sup> also support this conclusion.

Impurities generated by plasma bombardment at the end walls are unable to enter the main plasma because of their relatively low energy and the positive potential of the plasma. This assumes they are ionized in the plasma stream before reaching the peak potential in the plug plasma. Impurities generated at the walls of the central cell (by charge exchange neutral atom bombardment or a residual ion flux hitting the central cell walls) have only to cross a short distance before entering the electrostatic well of the central cell plasma. If ionized there, the impurities are well-confined by the electrostatic potential; accumulation of such impurities in the central cell is a potential problem that needs to be addressed.

Fortunately, the tandem mirror has a natural divertor action which can, in principle, alleviate the problem of impurity accumulation in the central cell. The radius of the plug plasma is determined by the footprint of the plug neutral beam and the gyroradius of the hot plug ions. This radius determines the radius over which there is a sufficiently positive confining



potential for the central cell. Following the field lines from the edge of the plug plasma back to the central cell determines the central cell radius for which there is adequate axial confinement. In TASKA, this is 32 cm, which is what we define to be the plasma radius in the central cell. Plasma on magnetic field lines outside this radius is not axially confined and can flow to the end walls. This region is analogous to the scrape-off layer due to divertors or limiters in tokamaks. The plasma between the central cell wall and the axially confined plasma radius is sometimes called a "halo" plasma. We will use that terminology here.

The halo plasma can play an important role in preventing impurity accumulation in the main confined plasma in the central cell. Impurity atoms released from the wall have to traverse the halo plasma before reaching the main plasma. If the line density of the halo plasma is sufficiently dense ( $\int n dr \gtrsim 10^{13} \text{ cm}^{-2}$ ) then the impurity atoms are ionized in the halo plasma and swept out the ends before they can diffuse into the main plasma and become confined. The halo plasma is maintained by cross-field transport from the main plasma and by ionization of gas in the halo. An upper limit for the loss rate of the halo plasma can be estimated<sup>(3)</sup> by the transit time of the ions (at temperature  $T_i$ ) modified by the mirror ratio,  $R_c$ , between the central cell and the peak field of the barrier;

$$\tau_{\parallel} \approx \frac{R_c L}{2v_{\parallel}}$$

$$v_{\parallel} \approx \sqrt{\frac{2kT_i}{\pi m_i}} .$$

Using a "slab" model for the halo region, the particle conservation equation

is

$$\frac{\partial \Gamma}{\partial r} = - \frac{n}{\tau_{\parallel}} ,$$

where  $\Gamma$  is the cross-field flux and the parallel loss is treated as an absorption process. Assuming  $\tau_{\parallel} \approx \text{constant}$ , one gets an expression for the line density of the halo plasma,

$$\int n dr = \tau_{\parallel} \Gamma_r .$$

Here  $\Gamma_r$  is the radial flux at the boundary between the halo and the main confined plasma.

The value of  $\Gamma_r$  is unknown since it depends sensitively on details concerning radial transport in the central cell. For estimation purposes, we can normalize  $\Gamma_r$  to the total loss  $\dot{N}_T$  of the main plasma with a factor  $f$ ;

$$\Gamma_r = \frac{f \dot{N}_T}{2\pi r_c L} = \frac{f}{1-f} \frac{\dot{N}_a}{2\pi r_c L}$$

where  $\dot{N}_a$  is the total axial end loss (particle/sec) from the main central cell plasma. Putting these results together, one gets an estimate for the line density in the halo plasma

$$\int n dr \approx \frac{R_c \dot{N}_a}{\pi r_c} \sqrt{\frac{\pi m_i}{2kT_i}} \frac{f}{1-f} .$$

For TASKA,  $\dot{N}_a = 3.5 \times 10^{21}$  ions/sec,  $R_c = 7.4$ , and  $r_c = .32$  m. If we assume  $f = .2$  and  $T_i = 100$  eV, then

$$\int n dr \approx 1.3 \times 10^{13} \text{ cm}^{-2}$$

which is a marginally adequate line density for the halo plasma. This line density is comparable to that sought in the scrape-off layer of divertor tokamaks. This value, however, depends on the assumption for  $f$  and  $T_i$  in the halo; both of these are unknown at this time. The factor  $f$  depends on the details of radial transport, and  $T_i$  is determined by a power balance for the halo plasma, which remains an unsolved problem (in the tokamak scrape-off layer,  $f = 1$ , but the value of  $T_i$  is very poorly known).

One contribution to the power balance for the halo plasma is line radiation from partially stripped atoms. These could be helium or impurity atoms entering the plasma from the outside. This problem has been well studied for impurity radiation in tokamaks<sup>(4-6)</sup> and for the case of high density z-pinch plasmas.<sup>(7)</sup> In this section, the populations of the ionization states and the line radiation are calculated for helium and a possible impurity, oxygen, in a halo plasma with an electron density of  $10^{13} \text{ cm}^{-3}$  and an electron temperature of 100 eV. In both cases, the losses from line radiation are found to be insignificant in the power balance of the halo plasma if the impurity density is  $\lesssim 1\%$  of the hydrogen ion density in the halo.

In order to estimate the power from line radiation, the populations of atoms with each of the ionization states must be calculated. Since the density of electrons is  $10^{13} \text{ cm}^{-3}$  and the temperature is 100 eV, the equilibrium ionization populations may be calculated in the Coronal model.<sup>(8)</sup> In this model, if the recombination rate for Z-times ionized atoms,  $\alpha_Z$ , and the ionization rate,  $S_Z$ , obey the relation

$$\alpha_z N_z = S_{z-1} N_{z-1} ,$$

where  $N_z$  is the density of ions in ionization state  $z$ , then

$$\frac{\alpha_{z+1}}{S_z} = 7.87 \times 10^{-9} \frac{I_z^2}{n_z} \left(\frac{I_z}{T}\right)^{.75} \exp\left(-\frac{I_z}{T}\right) .$$

In this expression,  $I_z$  is the ionization potential in eV for ions moving from state  $Z$  to  $Z+1$ ,  $n_z$  is the degeneracy of electrons doing this ionization and  $T$  is the electron temperature in eV. Using the ionization potentials calculated by Carlson,<sup>(9)</sup> the densities of atoms in each possible ionization state for helium and oxygen are determined for total impurity densities of  $10^{11} \text{ cm}^{-3}$  and are shown in Table III.6-1. It may be noted that the helium is almost fully stripped and should not emit much line radiation while most oxygen ions still have electrons bound to them and may emit significant amounts of line radiation.

The power of line radiation may be estimated with the analysis of Mosher,<sup>(7)</sup> where the power is given as

$$P_L \approx \frac{3.5 \times 10^{-25}}{T^{1/2}} N_z N_e \langle \exp\left(-\frac{E_{N1}^2}{T}\right) \rangle \text{ (W/cm}^3\text{)}$$

where

$$E_{N1}^2 = \left(\frac{Z}{137}\right)^2 \frac{m_e c^2}{2} \left[ \frac{1}{N^2} - \frac{1}{(N+1)^2} \right]$$

is a typical energy of a line if  $N$  is the principal quantum number of the electrons involved in the transition. The electron and ion densities are in

Table III.6-1. Ionization State Densities (cm<sup>-3</sup>)

| <u>Z</u> | <u>He</u>               | <u>0</u>               |
|----------|-------------------------|------------------------|
| 0        | 2.82                    | $1.71 \times 10^{-17}$ |
| 1        | $.53 \times 10^6$       | $1.07 \times 10^{-10}$ |
| 2        | $.99997 \times 10^{11}$ | $3.91 \times 10^{-5}$  |
| 3        | ---                     | 2.14                   |
| 3        | ---                     | $1.99 \times 10^4$     |
| 5        | ---                     | $8.64 \times 10^7$     |
| 6        | ---                     | $8.98 \times 10^{10}$  |
| 7        | ---                     | $9.97 \times 10^9$     |
| 8        | ---                     | $8.63 \times 10^7$     |

The plasma temperature in the "halo" is 100 eV.

The plasma density in the "halo" is  $10^{13} \text{ cm}^{-3}$ .

The total density of He is  $10^{11} \text{ cm}^{-3}$ .

The total density of 0 is  $10^{11} \text{ cm}^{-3}$

units of  $\text{cm}^{-3}$  while the temperature and energies are in eV. Using the densities in Table III.6-1, the line radiation powers are found to be

$$P_L \approx 5.89 \times 10^{-7} \text{ W/cm}^3$$

for helium and

$$P_L \approx 1.64 \times 10^{-2} \text{ W/cm}^3$$

for oxygen. For the halo plasma, the power loss due to parallel flow is  $.2 \text{ W/cm}^3$  at  $n = 10^{13} \text{ cm}^{-3}$  and 100 eV. Both values for the line radiation are small compared to the total power of the system and should not be important to the power balance.

Post et al.<sup>(4)</sup> have done more detailed calculations which lead to an even lower power for oxygen ( $3 \times 10^{-4} \text{ W/cm}^3$ ) and approximately the same value for helium ( $2 \times 10^{-6} \text{ W/cm}^3$ ). Though the resulting powers are somewhat different than those calculated herein, the general statement that line radiation is not an important effect for oxygen and helium impurities is supported.

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## IV Heating, Fueling and Exhaust

### IV.1 Electron Cyclotron Resonance Heating

#### IV.1.1 Introduction

Continuous direct heating of the potentially trapped electron population in the TASKA end plugs is necessary to maintain the excess potential  $\phi_c$  required to confine the central cell ions.<sup>(1)</sup> This excess plug potential is a nearly linear function of the plug electron temperature  $T_{ep}$ :

$$\phi_c = T_{ep} \ln \left( \frac{n_e}{n_b} \sqrt{\frac{T_{ec}}{T_{ep}}} \right) - \phi_b . \quad (IV.1-1)$$

Here  $T_{ec}$  is the central cell electron temperature,  $\phi_b$  is the barrier potential dip, and  $n_e$  and  $n_b$  are the plug and barrier electron densities, respectively. The desired excess potential of 43 keV requires a plug electron temperature of 59 keV.

A detailed study of the barrier/plug plasma physics reveals that good confinement can be achieved if the heating zone is centered around the 2 tesla-point, i.e., at an axial distance of 2200 cm outwards from the midplane of the central cell. This is because the plug electron population is electrostatically confined by the total potential  $\phi_b + \phi_c$ , and heating anywhere within the electrostatic well should heat the bulk electron population.

Following criteria were used to define a feasible solution of the heating problem:

- a. No variation of the signal frequency should be required during the startup phase of the reactor.
- b. The launching direction should be constant, too, and should not be close to tangential to the plasma surface.



- c. Penetration of the plasma should be good, i.e., most of the wave energy should be absorbed by the bulk of the plasma (as opposed to surface absorption, which approaches 100% when the plasma appears as a blackbody at the local electron cyclotron resonance frequency).
- d. Absorption should be near 100% for a single pass through the plasma. To a certain extent this criterion is in contradiction to c), and one has to expect a fairly narrow window in parameter space where both requirements are satisfied - if such a window exists at all.
- e. The transport system between wave source and plasma should have low losses and show a low susceptibility to neutron damage.
- f. The wave source should require only a slight extrapolation of existing technology. We will discuss in detail the solutions we propose, starting from the plasma requirements.

#### IV.1.2 The Plasma

Propagation and absorption of electromagnetic waves close to the electron cyclotron frequency of the plasma was studied numerically using the general purpose code ECNAG.<sup>(2)</sup> This code is characterized by:

- a. Numerical solution of the geometric optics equations

$$\frac{d\vec{r}}{dt} = - \frac{\partial D / \partial \vec{k}}{\partial D / \partial \omega} ; \frac{\partial \vec{k}}{\partial t} = \frac{\partial D / \partial \vec{r}}{\partial D / \partial \omega} \quad (\text{IV.1-2})$$

using a cold-plasma dispersion relation

$$D(\vec{k}, \omega) = 0 . \quad (\text{IV.1-3})$$

Here  $\bar{r}$  and  $\bar{k}$  represent the position of the wave front along the ray and the wave vector, respectively, while  $t$  and  $\omega$  are time and frequency. Because we are interested in spatial absorption in a stable plasma  $\omega$  is held constant while  $\bar{k}$  is allowed to vary according to the requirements of the dispersion relation.

It has been shown that using a hot plasma dispersion relation does not affect the calculated wave trajectories<sup>(3)</sup> as long as the characteristic distance of the absorption is significantly larger than the wavelengths involved.

- b. An analytic absorption model. We would have preferred to use a more generally valid numerical model, but at present such a model is not available for absorption around the fundamental frequency. The choice of the resonance at  $2 T$  excludes the use of harmonics (the cyclotron frequency  $\omega_c$  at  $B = 2 T$  is 56 GHz, i.e., at the limit of available power sources).

The model consists of the following expressions for the absorption coefficient  $\alpha$ , correct to lowest significant order in  $v_{th}/c$ :

\*oblique propagation, low density,<sup>(4)</sup>

$$\alpha^{X,0} = \frac{\sqrt{\pi}}{2} \frac{\omega_p^2}{\omega^2} \frac{\omega}{c} (1 + \cos^2 \theta) \frac{c}{v_{th}} \frac{e^{-\left(\frac{c}{v_{th}} \frac{\omega - \omega_c}{ck \cos \theta}\right)^2}}{|\cos \theta|} \quad (IV.1-4)$$

\*oblique propagation, high density,<sup>(5)</sup>

$$\alpha^{X,0} = 2\sqrt{2} R^{X,0} \frac{v_{th}}{c} \frac{\omega_c}{c} \operatorname{Im}\left[-\frac{1}{Z(\zeta)}\right] \quad (IV.1-5)$$

where  $R^{x,0}$  represents a complicated function of  $\omega_p$ ,  $\omega_c$ ,  $\frac{ck}{\omega}$  and  $\theta$  (which scales as  $\frac{\omega_p^2}{\omega_c^2} \sim \frac{1}{n_e}$  !),  $\zeta = \frac{\sqrt{2}}{2} \frac{c}{v_{th}} \frac{\omega - \omega_c}{ck \cos \theta}$  and  $Z(\zeta)$  represents the familiar plasma dispersion function.

\*quasi-perpendicular propagation, low density<sup>(5,6)</sup>

$$\alpha^0 = \frac{1}{\sqrt{2}} \left(1 - \frac{\omega_p^2}{\omega_c^2}\right)^{1/2} \frac{\omega_p^2}{\omega_c^2} \frac{\omega}{c} [-F_{7/2}''(z)] \quad (\text{IV.1-6})$$

$$\alpha^x = \frac{\omega_p^2}{\omega_c^2} \left(\frac{c}{v_{th}}\right)^2 \frac{\omega}{c} [-\frac{1}{2} F_{5/2}''(z)] \quad (\text{IV.1-7})$$

\*quasi-perpendicular propagation, high density<sup>(5)</sup>

$$\alpha^0 = \frac{1}{\sqrt{2}} \left(1 - \frac{\omega_p^2}{\omega_c^2}\right)^{1/2} \frac{\omega_p^2}{\omega_c^2} \frac{\omega}{c} \frac{-F_{7/2}''(z)}{|G_{7/2}| + \sqrt{G_{7/2}' + |G_{7/2}|}} \quad (\text{IV.1-8})$$

$$\alpha^x = \sqrt{2} \left(1 - \frac{\omega_p^2}{2\omega_c^2}\right)^{3/2} \frac{\omega_c^2}{\omega_p^2} \left(\frac{v_{th}}{c}\right) \frac{\omega}{c} \frac{-F_{5/2}''(z)}{|F_{5/2}(z)|^2} \cdot \quad (\text{IV.1-9})$$

The function  $F_q(z)$  is the relativistic counterpart of the plasma dispersion function and is defined by<sup>(7)</sup>

$$F_q(z) = F_q'(z) + iF_q''(z) = -i \int_0^\infty \frac{e^{izt}}{(1-it)^q} dt \quad (\text{IV.1-10})$$

$$\text{with } z = \frac{c^2}{v_{th}^2} \frac{\omega - \omega_c}{\omega_c} . \quad G_q(z) \text{ is a shorthand for } 1 + \frac{1}{2} \frac{\omega_p^2}{\omega_c^2} F_q(z) .$$

Remarkable is the inverse dependence on the squared plasma frequency  $\omega_p^2$  (i.e. on the electron density  $n_p$ ) of the absorption coefficient for the extraordinary mode in a high density plasma. The ordinary mode does not show this anomalous behavior, and Eq. (IV.1-6) is simply the low density limit of Eq. (IV.1-8). The code selects between oblique (where Doppler broadening prevails) and quasi-perpendicular (where a relativistic mechanism is dominant) regimes by comparing the quantities  $\frac{ck}{\omega} |\cos\theta|$  and  $v_{th}/c$ . Both high and low-density absorptivities are calculated, and the lowest value is used. Finally, the choice between X and O-mode depends on the user and is dictated by the polarization of the incoming signal. The code is completed by user supplied models for the confining magnetic field and the plasma equilibrium.

#### IV.1.2.1 Magnetic Field and Plasma Model

We use a field model based on quadratic regression of data from an exact field calculation with EFFI:

$$B_x = [.0085 \, z^2 - 4.35 \, z + 512] \, x$$

$$B_y = [-.0028 \, z^2 + 1.22 \, z - 198] \, y \quad (\text{IV.1-11})$$

$$B_z = [B_p + 100 \, (z - 300)] \, (1 + 1.8 \times 10^{-5} \, x^2) \, (1 - 1.2 \times 10^{-5} \, y^2) .$$

This model does not satisfy  $\nabla \cdot \vec{B} = 0$  at every point  $(x,y,z)$  where the model holds, but  $\nabla \cdot \vec{B} \ll |\nabla B|$ . The point  $z = 300$  cm where  $B_z = B_p = 2$  T on axis is an arbitrary chosen coordinate, corresponding to  $z = 2200$  cm in the regular system. Distances and fields are in cm and gauss, respectively, and the model holds for  $|x,y| \leq 30$  cm and  $250 \text{ cm} < z < 350$  cm. The plasma electron density and temperature are represented by

$$n_e = n_{e0} e^{-\left(\frac{r^2}{a^2}\right)^5} \quad (\text{IV.1-12})$$

$$T_e = T_{e0} e^{-\left(\frac{r^2}{a^2}\right)^5} \quad (\text{IV.1-13})$$

where  $T_{e0}$  and  $n_{e0}$  are the central electron temperature and density, respectively,  $r^2 = x^2 + y^2$  and  $a$  is the plasma radius. The  $\beta$ -factor, i.e., the ratio between kinetic and magnetic pressure, is calculated for every point along the ray trajectories and the vacuum magnetic field calculated by means of Eq. IV.1-11 is corrected accordingly.

#### IV.1.2.2 Results

Preliminary calculations showed that the extraordinary mode (wave electric vector in a plane perpendicular to the confining magnetic field at perpendicular prop.) cannot be used for the following reasons.

A complication typical of linear confinement systems is that the resonant layer roughly has the shape of a disc perpendicular to the machine axis. As a consequence the ray trajectory needs to have a substantial axial component if one is to achieve absorption in the central plasma area. For an X-polarized wave the plasma acts as a dielectric with a refractive index larger than unity, i.e., with a high permittivity. This implies that a ray is bent towards

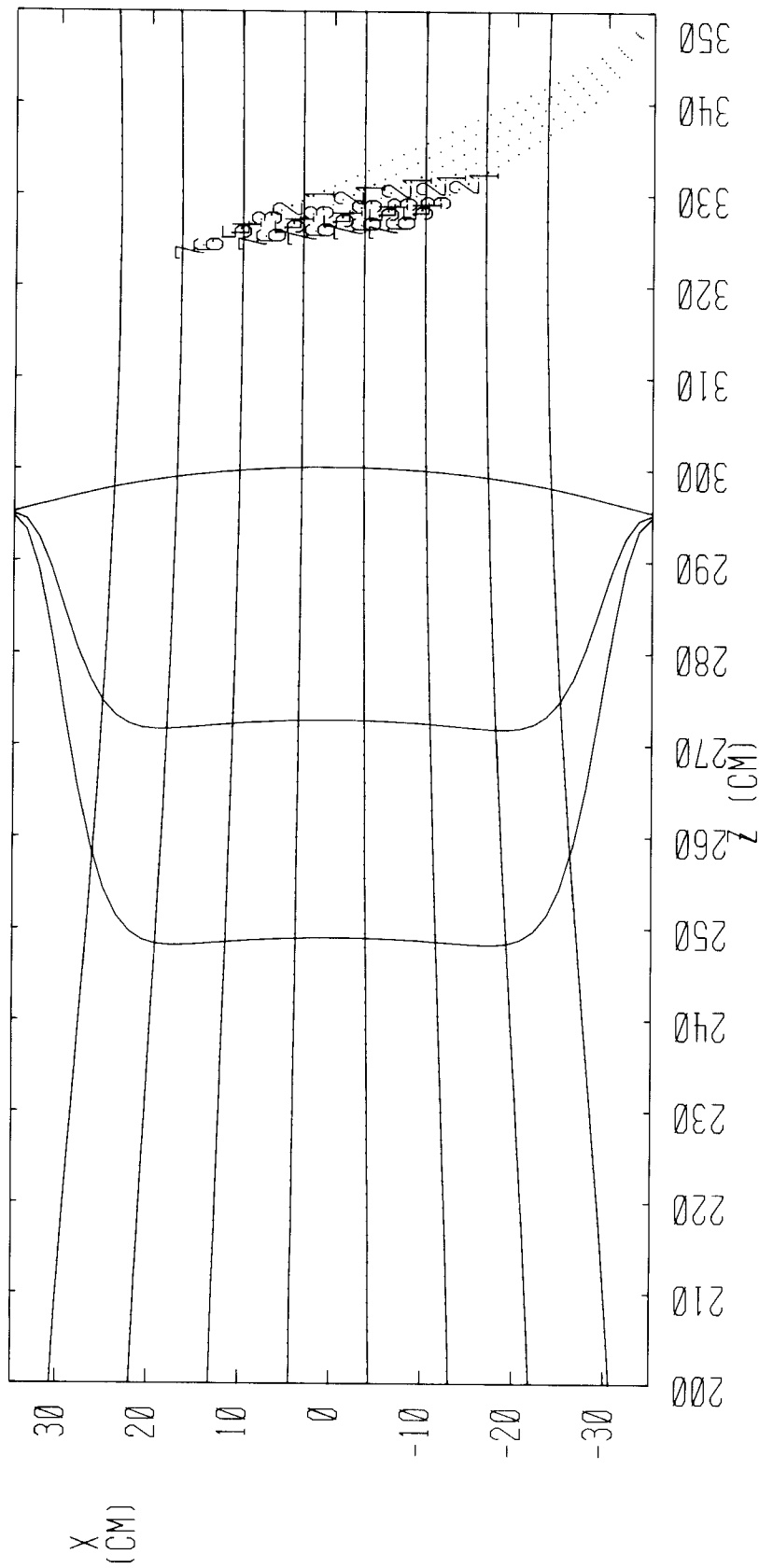
the normal and will have to enter the plasma at a grazing angle to allow for any penetration in the axial direction. There simply is not enough room to mount the launching structure parallel to the axis of the reactor. Furthermore, such a mounting would make the structure very vulnerable to neutron damage.

The extraordinary mode as a rule has a very high absorption coefficient. While this is an attractive feature during the early startup phase and, in general, for devices with a low operating temperature, it turns the plasma into a black body which totally absorbs the wave at the plasma surface. Therefore, X-mode heating of a mirror plug can only be useful during the earliest startup phase. This would, however, require separate sources for startup and steady-state operation. For this reason the use of waves with X-polarization has been rejected altogether. Figure IV.1-1 shows typical ray trajectories and absorption patterns for the X-mode.

The ordinary mode has its wave electric vector oriented parallel to the direction of the confining magnetic field (for propagation perpendicular to the field) and, therefore, displays little or no refraction. One expects no penetration problems with the ordinary mode until high electron densities or very high temperatures are reached. If the density is sufficiently high, then the condition

$$\omega = \omega_p \quad (\text{IV.1-13})$$

is satisfied on a closed surface in the plasma. Because the plasma frequency  $\omega_p$  is a cut-off for the ordinary mode the wave cannot penetrate inside this closed surface. A total electron density of  $1.1 \times 10^{13} \text{ cm}^{-3}$  in the center



EXTRAORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0,0,300.0) 1.9970E+04 GAUSS  
 DENSITY AT (0,0,300.0) 1.0000E+13 1/CM\*\*3  
 EL. TEMP AT (0,0,300.0) 2.00 KEV  
 INCIDENCE : FROM -65. TO -45. DEGREES ( 5.)

Fig. IV.1-1 Typical ray trajectories for the X-mode.

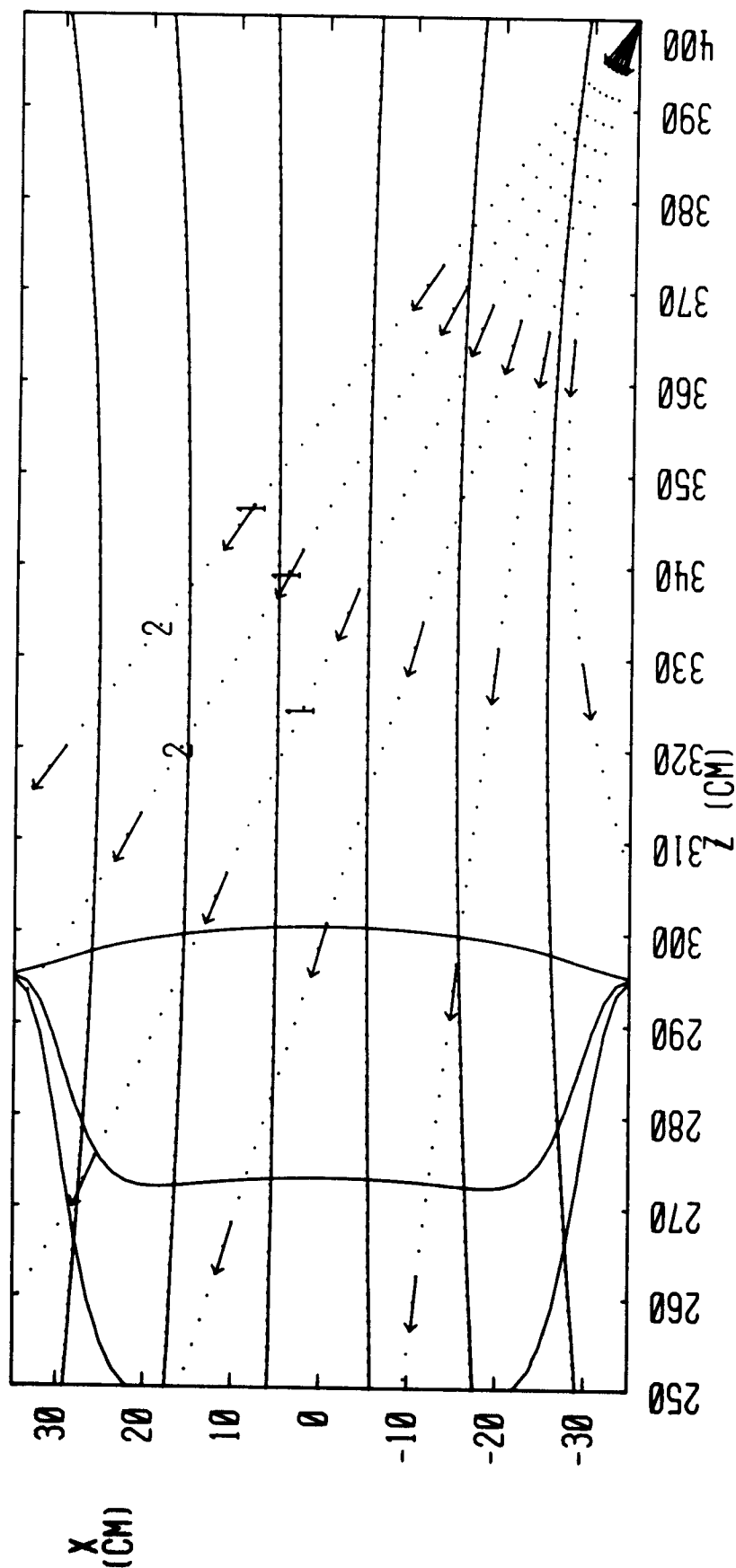
implies a lower limit for the wave frequency of 29.8 GHz, which is well below the electron cyclotron frequency of 56 GHz. The flat density profile (Eq. IV.1-12) causes the  $\omega = \omega_p$  surface to be ill defined in the bulk of the plasma and prevents the use of reflection at this surface to enhance the effective optical depth of the plasma.

Figures IV.1-2a through 2g show ray trajectories and absorption levels for various temperatures representative for both startup and steady-state. The launching point is chosen .5 m outwards from the  $B_0 = 2$  T point and the frequency is kept at 56 GHz. Although it is certain that a different frequency may lead to improved results, it was felt that the commercial availability of a 56 GHz tube was too important an asset to be sacrificed for slightly better plasma physical performance. The dots representing the ray trajectories are plotted at fixed time intervals (of  $10^{-10}$  s) and, therefore, their separation can be used as a measure for the phase velocity. The numbers 1-7 represent the percentage of incident energy absorbed up to the point where they are printed; they stand for 25, 50, 75, 90, 95, 99 and 100% absorption, respectively.

Other features of the plot are the quasi-axial field lines and, from right to left, the electron cyclotron resonance, the right-hand cyclotron cut-off and the cold plasma resonance surfaces; the latter two only affect the extraordinary mode. The angles of incidence are measured from the upwards vertical, in clockwise direction. Linear scales are the same in both directions.

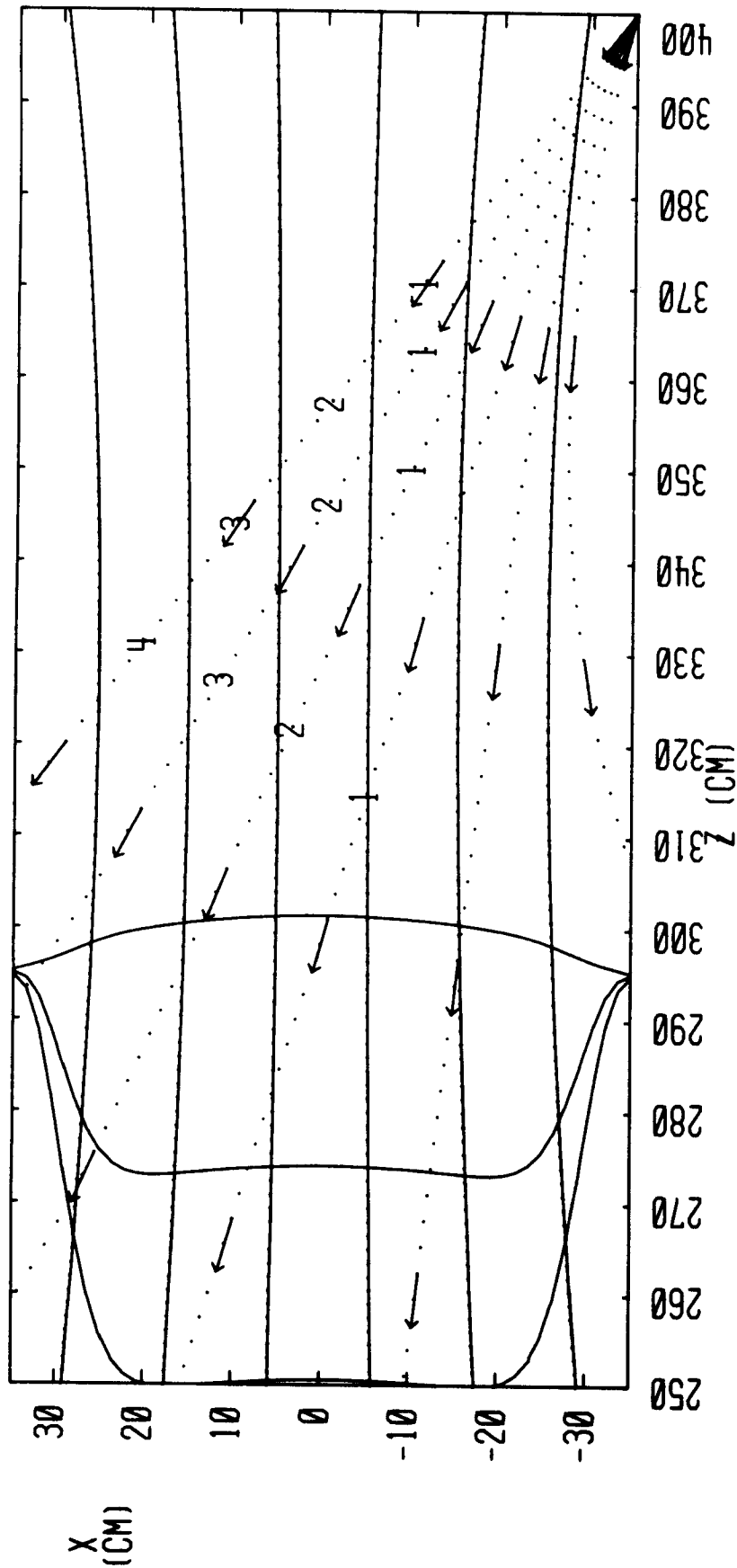
At 5 keV absorption of the ordinary mode is already considerable: the rays with a  $60^\circ$ ,  $55^\circ$ , and  $50^\circ$  incidence reach 46, 66 and 68 percent absorption, respectively. At 10 keV the corresponding percentages are 68, 88 and





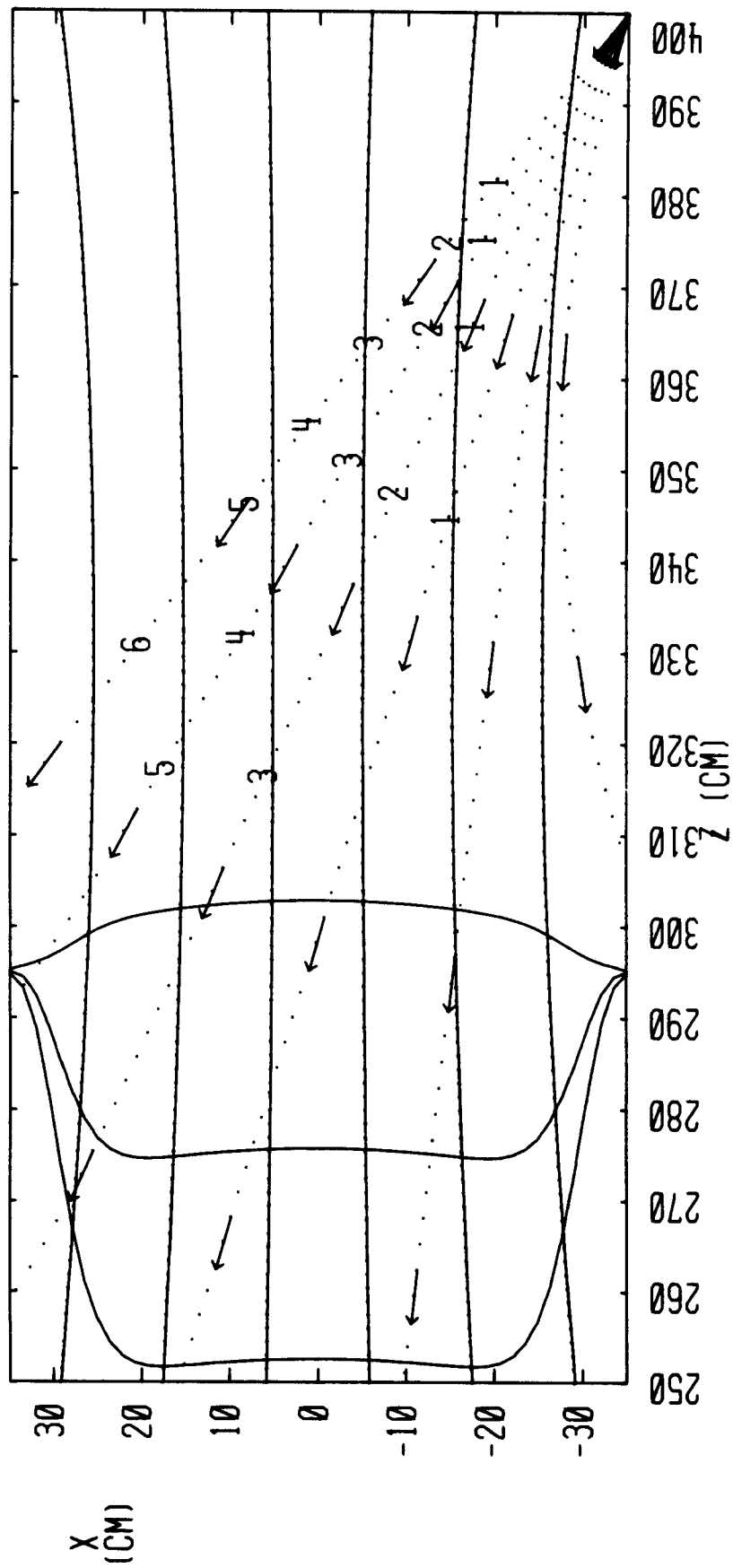
ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0, 300.0) 1.9925E+04 GAUSS  
 DENSITY AT (0.0, 300.0) 1.0000E+13 1/CM.<sup>3</sup>  
 EL. TEMP AT (0.0, 300.0) 5.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2a. Ray trajectories and absorption profiles for the O-mode at 5 keV.



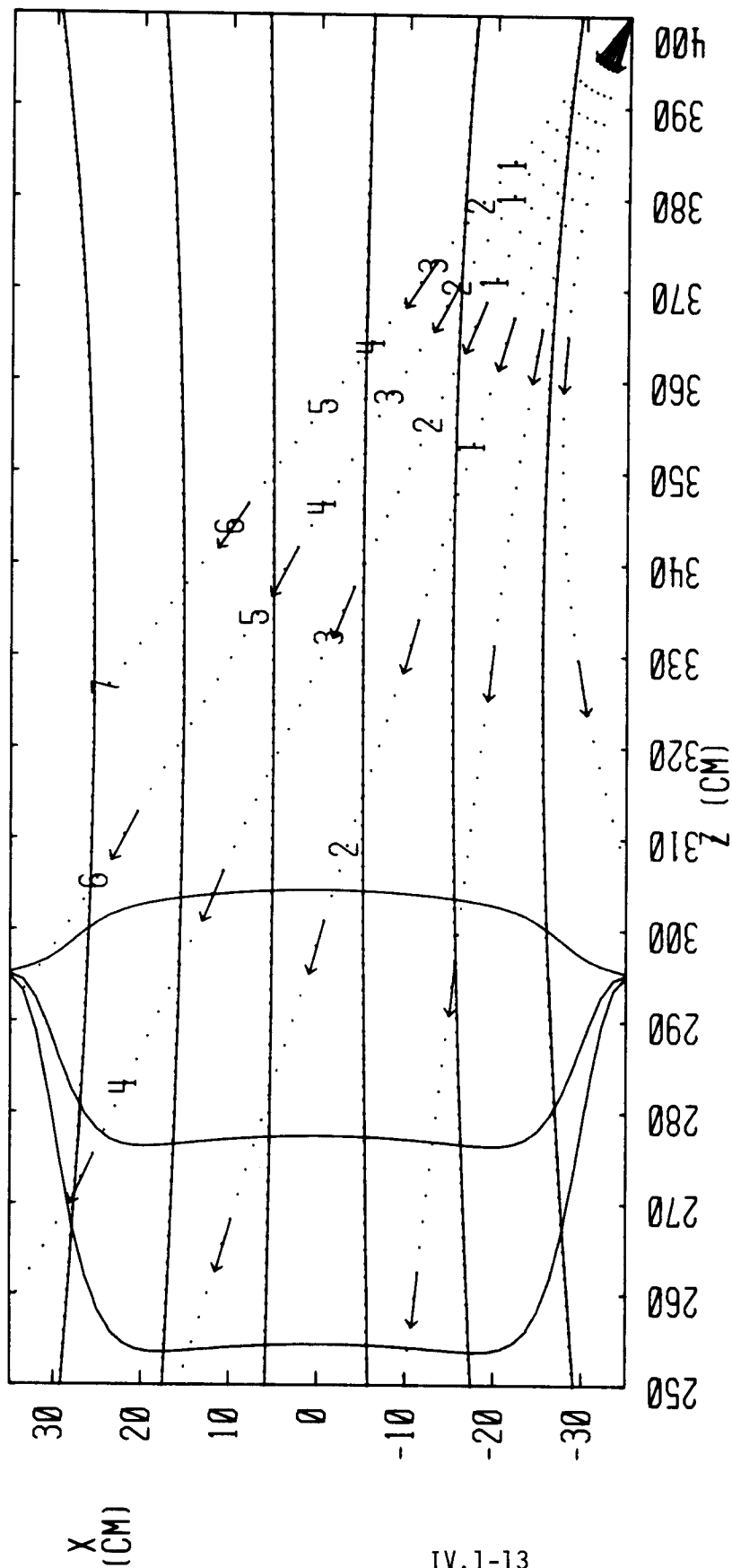
ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0, 300.0) 1.9849E+04 GAUSS  
 DENSITY AT (0.0, 300.0) 1.0000E+13 1/CM<sup>3</sup>  
 EL. TEMP AT (0.0, 300.0) 10.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2b Ray trajectories and absorption profiles for the O-mode at 10 keV.



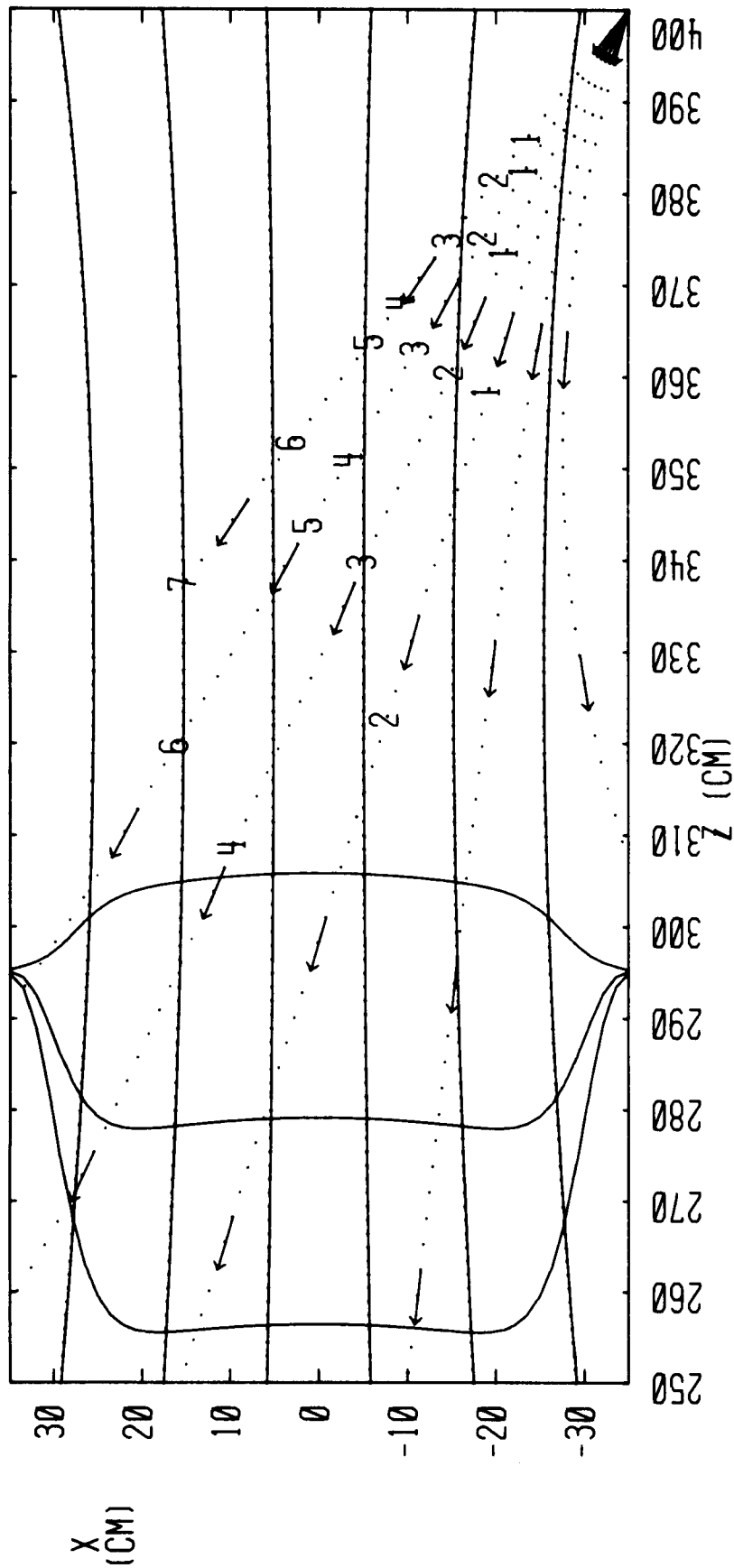
ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0, 300.0) 1.9698E+04 GAUSS  
 DENSITY AT (0.0, 300.0) 1.0000E+13 1/CM<sup>3</sup>  
 EL. TEMP AT (0.0, 300.0) 20.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2c Ray trajectories and absorption profiles for the 0-mode at 20 keV.



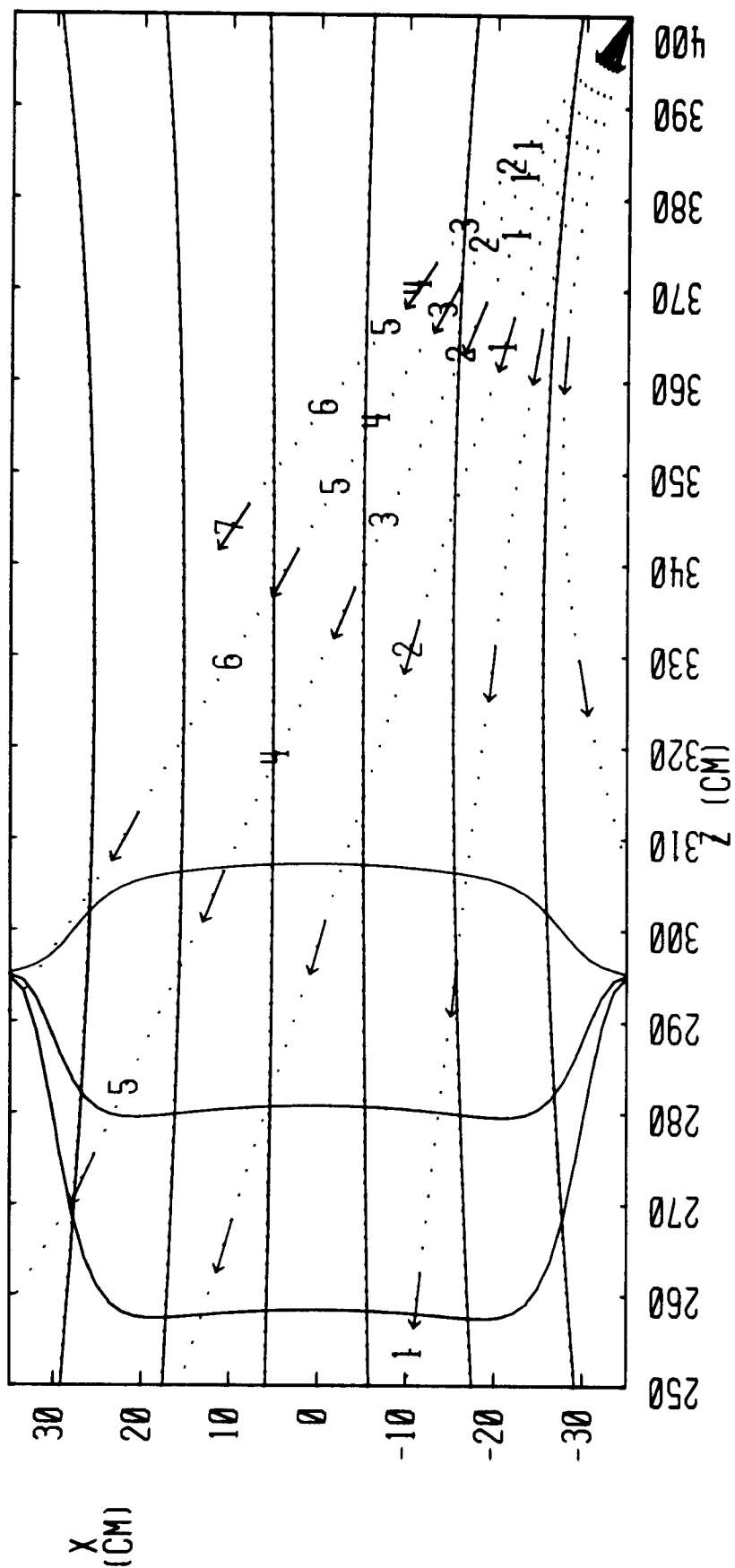
ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0.300.0) 1.9545E+04 GAUSS  
 DENSITY AT (0.0.300.0) 1.0000E+13 1/CM<sup>3</sup>  
 EL. TEMP AT (0.0.300.0) 30.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2d Ray trajectories and absorption profiles for the 0-mode at 30 keV.



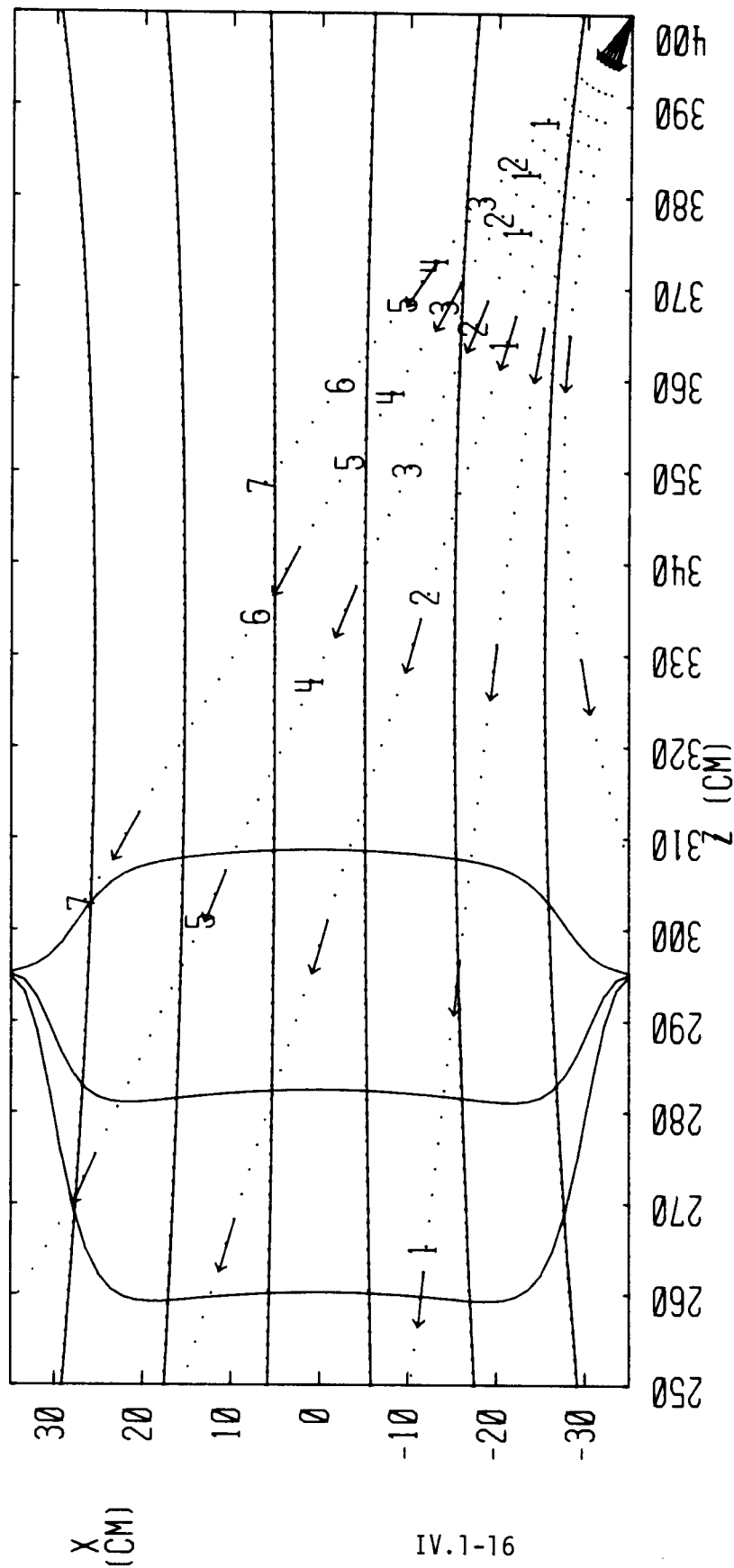
ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0, 300.0) 1.9391E+04 GAUSS  
 DENSITY AT (0.0, 300.0) 1.0000E+13 1/CM<sup>3</sup>  
 EL. TEMP AT (0.0, 300.0) 40.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2e. Ray trajectories and absorption profiles for the 0-mode at 40 keV.



ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0.300.0) 1.9235E+04 GAUSS  
 DENSITY AT (0.0.300.0) 1.0000E+13 1/CM\*\*3  
 EL. TEMP AT (0.0.300.0) 50.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2f Ray trajectories and absorption profiles for the 0-mode at 50 keV.



IV.1-16

ORDINARY MODE  
 FREQUENCY 56. GHZ  
 B FIELD AT (0.0, 300.0) 1.9079E+04 GAUSS  
 DENSITY AT (0.0, 300.0) 1.0000E+13 1/CM<sup>3</sup>  
 EL. TEMP AT (0.0, 300.0) 60.00 KEV  
 INCIDENCE : FROM -75. TO -50. DEGREES ( 5.)

Fig. IV.1-2g Ray trajectories and absorption profiles for the 0-mode at 60 keV.

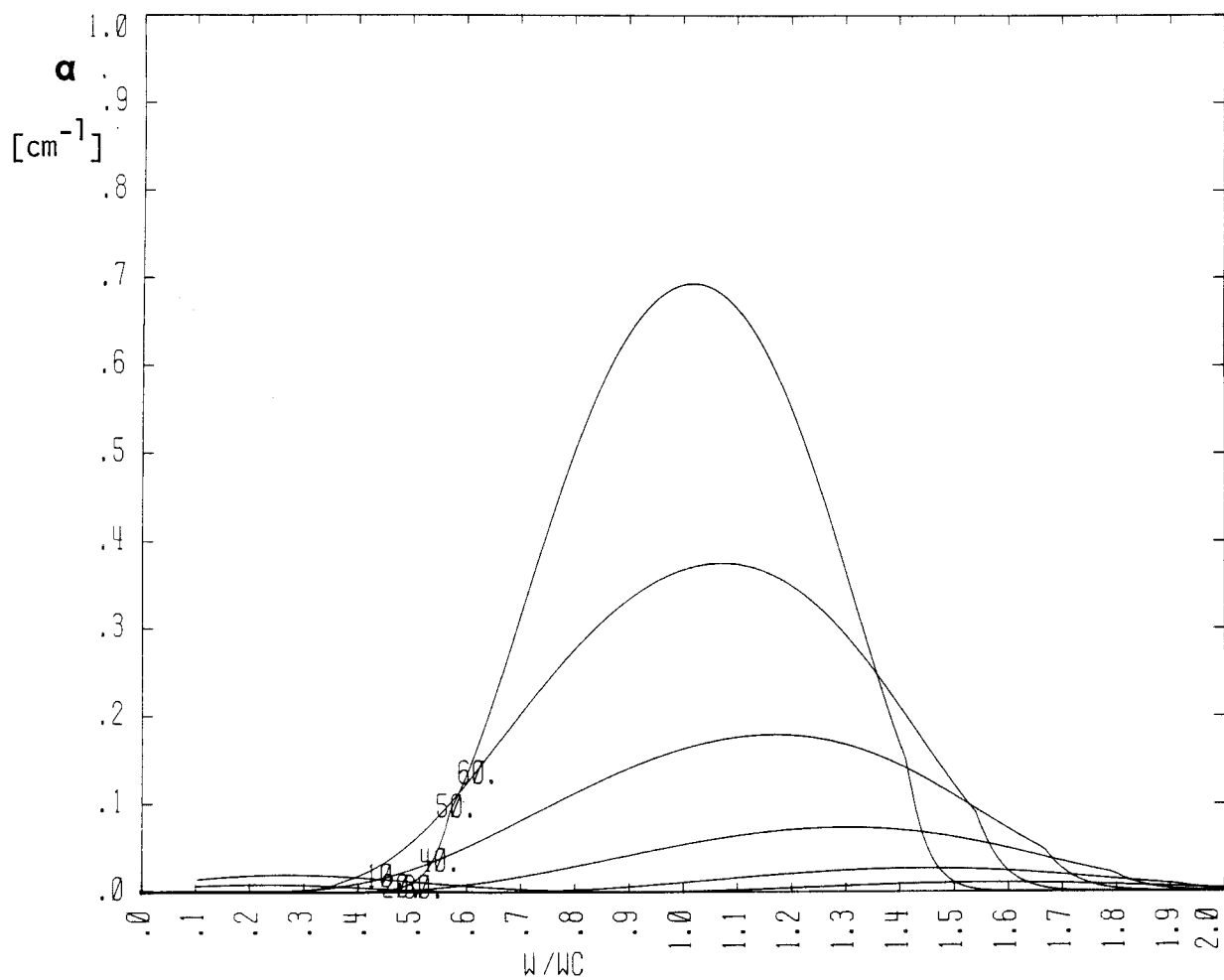
95. Within the accuracy of the model this probably means that total absorption of the ordinary mode can be expected for electron temperatures from ca. 10 keV on. For higher temperatures all rays in the 50°-60° range yield good absorption and penetration, with the 55°-60° launching direction giving the best centering around the plasma axis.

Careful examination reveals that the absorption is not symmetric about the cyclotron resonance surface. This is explained by cold plasma dispersion effects which vary quite strongly in that frequency range, especially when the angle  $\theta$  between the wave propagation vector  $\vec{k}$  and the confining magnetic field  $\vec{B}_0$  becomes small. This effect is clearly illustrated in Fig. IV.1-3 which represents the absorption coefficient  $\alpha$  as a function of  $\omega_c/\omega$  with  $\theta$  varying from 10° to 60° with 10° intervals. At 30 keV the dominant absorption mechanism is Doppler broadening for these directions, and the absorption profile is obviously asymmetric for the lower values of  $\theta$ .

Very recently, the ray tracing and absorption code has been coupled to a one-dimensional time dependent transport code. Only preliminary results were obtained, but they contribute very positively to our understanding of the heating during the initial startup phase.<sup>(8)</sup>

At a temperature of 2.8 keV a 56 GHz wave was launched at 60° with respect to the normal. The cumulative absorption was 86% when the ray left the plasma which means that an unexpectedly low fraction of the energy will be reflected at the wall and, in all likelihood, be absorbed somewhere in or close to the central cell. The transport code results show that (Fig. IV.1-4) energy deposition is strongly peaked in the plasma core. Figure IV.1-4 shows the absorbed energy as it is distributed over annuli of constant width and





ORDINARY MODE  
 PARAMETER : THETA  
 VTH/C = .350 TE = 31.3 KEV  
 CYCLOTRON FREQUENCY : 56. GHZ  
 PLASMA FREQUENCY : 30. GHZ

Fig. IV.1-3 Absorption coefficient vs. frequency.

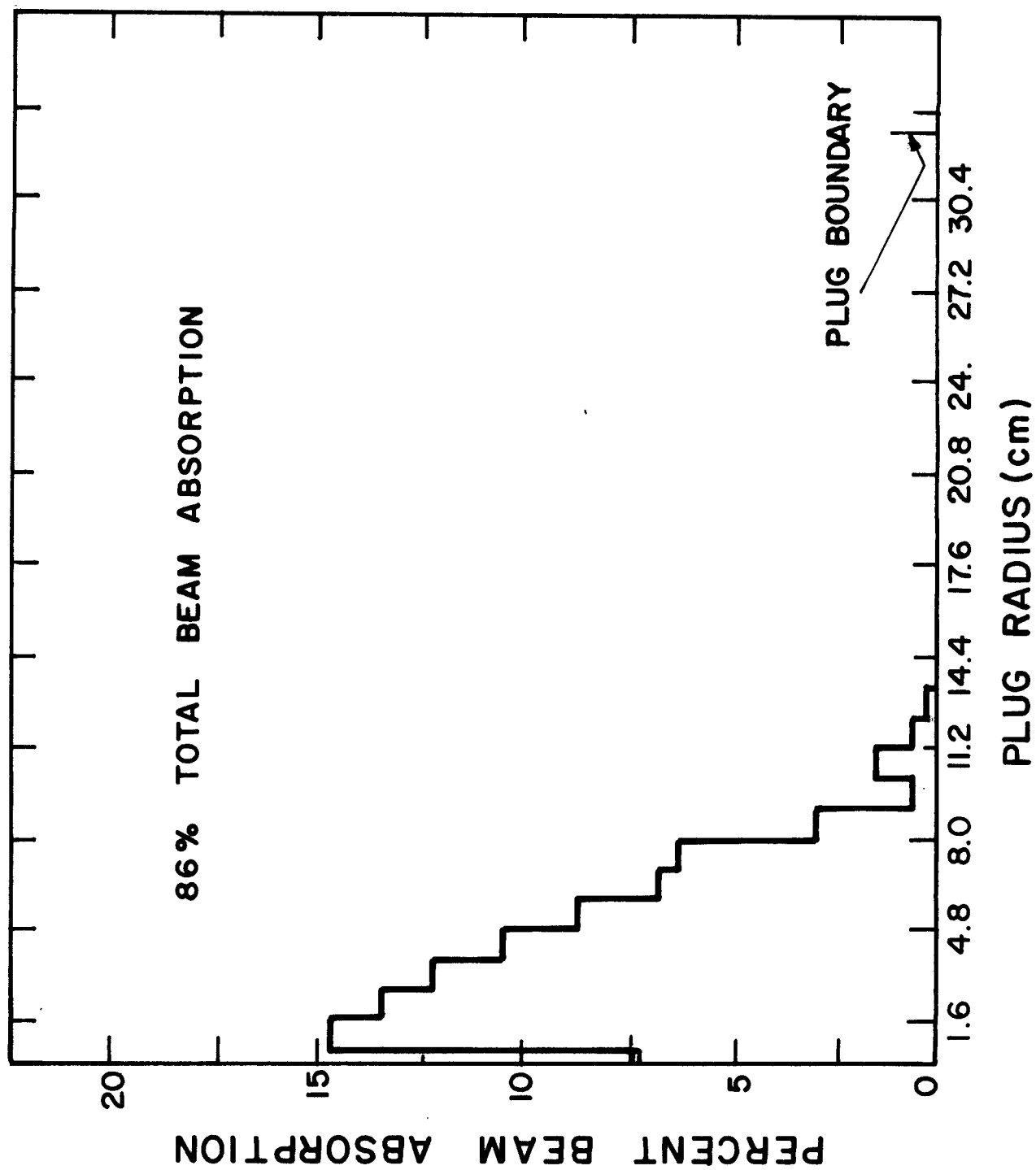


Fig. IV.1-4 Radial energy deposition profile.

hence the distribution per unit volume is even more peaked than suggested by the figure.

Concluding, it can be stated that a wave

- launched with ordinary polarization
- at 56 GHz
- at  $55^\circ$  with respect to the normal

will be totally absorbed in the electron temperature range between 5 and 60 keV, while contributing strongly to the plasma electron heating during the early startup phase.

#### IV.1.3 The Gyrotron Microwave Power Source

For the RF power source we chose a 56 GHz gyrotron because a suitable tube in that frequency range will be commercially available from at least two companies (Hughes Aircraft Co. and Varian Associates, Inc.) at power levels from 200 to 250 kW. It is expected that power levels between 350 kW and 1 MW<sup>(9,10)</sup> for continuous wave (CW) operation will become available by 1988. We opted for a conservative extrapolation of present-day technology to the 350 kW level.

The wave produced by these tubes has a circular (azimuthal) electric field. This can be converted to a linear polarization by using a stepped reflector of which one-half of the surface is displaced half a wavelength with respect to the other half. The desired polarization direction can be obtained by rotating the reflector.

Both Varian and Hughes have 28 GHz, 200 kW CW tubes available and expect to produce CW, 200 kW tubes at 56/60 GHz within the next two years. Experts at both companies rate the cost of the tubes at \$1. to \$2. per watt of microwave output with another \$.50 to be added for the superconducting magnets.

The power supplies that are currently delivered for use with experimental gyrotrons are 80 kV, 8 A (i.e., 640 kW) devices costing  $\$1.2 \times 10^6$  for 200 kW of RF output, or \$6. per watt.<sup>(11)</sup> This makes, however, a very inefficient use of some of the power supply equipment. The whole installation is typically to be fed from a 12 kV, 3 phase-grid and consists of a phase disconnecter, a step-start contactor, and a voltage regulator feeding a transformer-rectifier. The latter's cost scales as 1.3 times its KVA rating. The next and most costly device in the power supply is a series tetrode which is capable of feeding five gyrotrons in parallel. This, of course, would require upgrading the power rating of the transformer accordingly. It has been estimated that feeding five gyrotrons in parallel would reduce the cost of the power supply to about \$2.5 per watt RF power delivered. Therefore, it seems realistic to adopt a global figure of \$5/watt for a combined unit of gyrotrons, magnets and power supplies for TASKA.

#### IV.1.4 Beam Waveguide Launcher

Transport and energy balance studies have shown that about 7.5 MW of RF power for each plug at the electron cyclotron frequency are necessary to maintain the required electron temperature (59 keV) in the plug regions. As shown in section IV.1.2.2, a launching direction of  $55^\circ$  with respect to the normal ensures good absorption over a wide temperature range.

Since gyrotron sources at present are limited to power levels of about 350 kW, and it is not anticipated that the output of individual tubes is likely to rise above 500 kW to 1 MW in the future, power combining of several individual sources will be necessary. If delivering the total required power to the plasma is accomplished by multiple antennas, each feeding a separate region, then many ports and considerable space will be needed. This may very

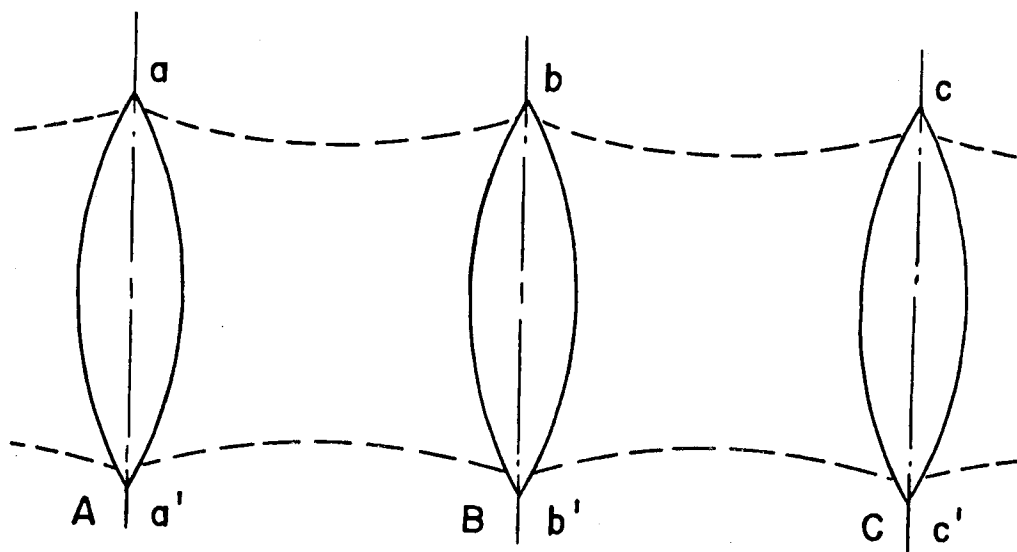
well conflict with magnet requirements. Passive combining results in an intolerable 3 dB loss and nonreciprocal combiners are also impractical at high power levels as well as in the presence of high magnetic fields.

To overcome these difficulties it is proposed to utilize a quasi-optical beam waveguide system. Such a system not only affords low loss spatial combining of gyrotron sources to multimewatt levels but also directs the radiation onto the plasma region from specific angles.

The beam waveguide shown in Fig. IV.1-5 is an open (non-metallic) waveguide composed of periodically spaced phase transformers (lenses or conducting reflectors in the optical sense) which redirect the beam and thus maintain high energy concentration along its axis. Such phase correction can be obtained on reflection from parabolic mirrors as will be seen shortly. The lowest loss beam waveguide mode designated  $TEM_{00}$  is characterized by a Gaussian radial amplitude distribution and linear polarization. Beam modes in general are characterized by the product of a Gaussian and the generalized Laguerre polynomials.<sup>(12)</sup> The lowest order Laguerre polynomial,  $L_0^0$ , is simply equal to one, however, and hence the  $TEM_{00}$  mode transverse fields are given by

$$E_{trans} = A_{00} e^{-a^2 r^2 / R^2} \quad (IV.1-14)$$

where  $R$  = guide radius and  $a = \sqrt{k_0 / 2z_0}$  plays the role of the Fresnel number of geometrical optics with  $2z_0$  designating the phase transformer spacing. Diffraction loss on such a waveguide is shown in Fig. IV.1-6 vs. the dimensionless parameter "a". Suppose, for example, a diffraction loss of 0.01 dB/iteration (or 0.2%) is chosen, leading to an "a" value at 56 GHz of 2.3 from Fig. IV.1-6. This "a" yields a "lens" spacing of 2 meters and a radius



BEAM WAVEGUIDE

$$E_{trans} = A_{mn} L_n^m \left( \frac{a^2 \rho^2}{R^2} \right) e^{-\frac{a^2 \rho^2}{R^2}}$$

$$a = \sqrt{\frac{k_o}{2z_o}} R$$

$R$  = guide radius

$2z_o$  = iteration length

Fig. IV.1-5 Beam waveguide.

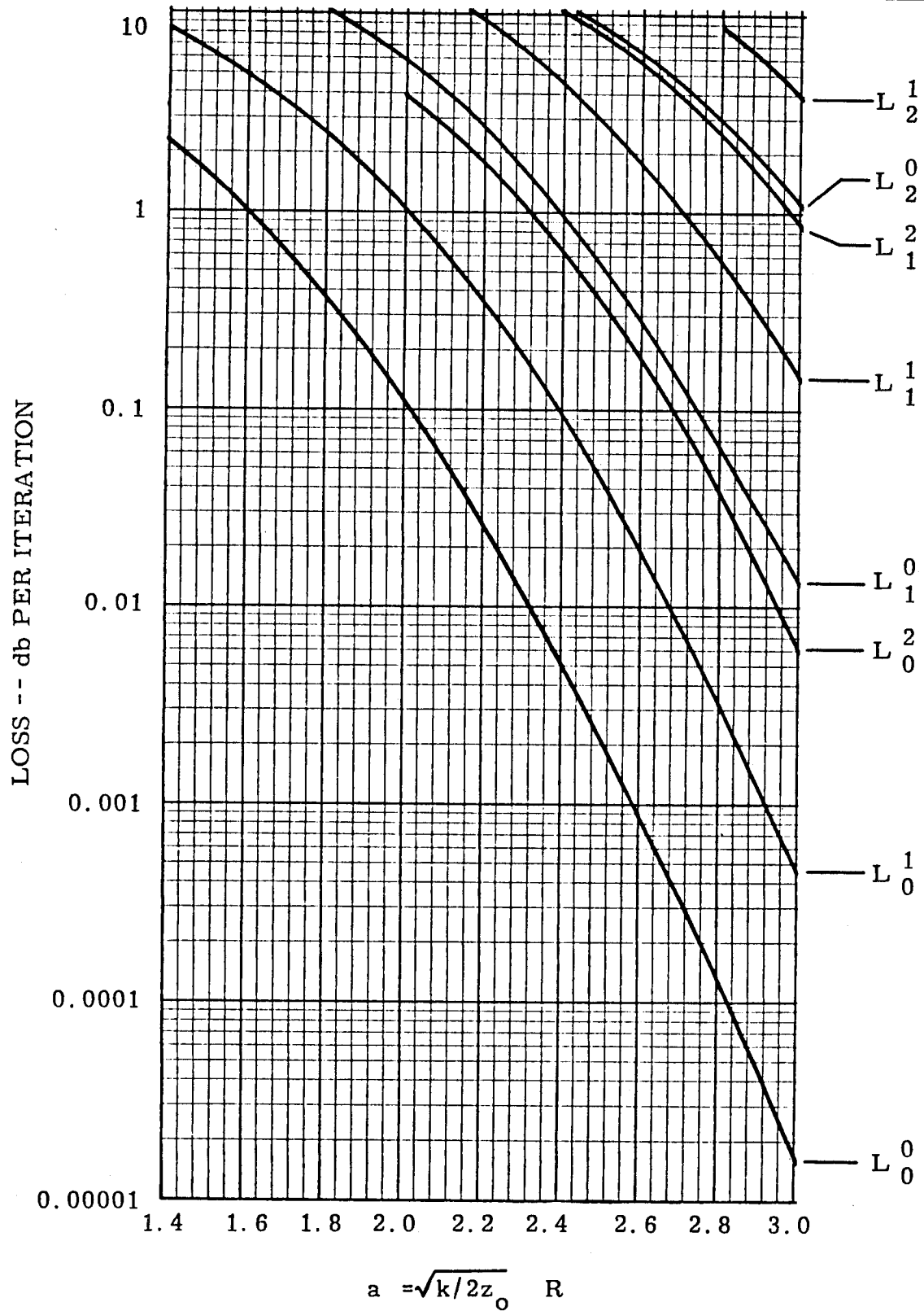


Fig. IV.1-6 Diffraction losses for beam modes.

of 9.5 cm. It is evident that at a given frequency it is possible to design the lens spacing and radius for any given diffraction loss.

It can be seen from Fig. IV.1-5 that the beam mode fields are reproduced at the centers of succeeding phase transformers, i.e. the fields of  $bb'$  equal the fields at  $aa'$ , etc. In addition, these fields must have uniform phase as must the fields midway between each transformer. In fact if a short circuiting metal plate were placed at the center of a phase transformer and another midway between two transformers a resonator would result. Such a resonator is also shown in Fig. IV.1-7 where the required phase correction is obtained by reflection from a parabolic mirror instead of from a half lens backed by a flat plate. That is, a Gaussian amplitude uniform phase beam mode at point "a" in Fig. IV.1-7 is exactly reproduced upon itself after reflection from the mirror.

Consider now the familiar Cassegrain system of geometrical optics shown in Fig. IV.1-8. In the absence of the hyperboloidal mirror, excitation at the focal point,  $a'$ , would result in parallel rays, i.e., a uniform phase front, at  $b$ . In the offset Cassegrain system shown, adding the hyperboloidal section merely switches the excitation point to "a" and aperture blocking is prevented by the offset. If now this system were excited at "a" by a beam mode (Gaussian amplitude, uniform phase) instead of a point source then that excitation would be exactly reproduced at "b" just as it was in Fig. IV.1-7 because the Cassegrainian tandem hyperboloid-paraboloid acts identically to a single paraboloid of focal length  $f$ .

Such an arrangement has considerable practical utility for plasma heating. If one imagines the plasma at point "b" and a proper beam mode excitation device at "a" then uniform phase radiation can be concentrated onto the



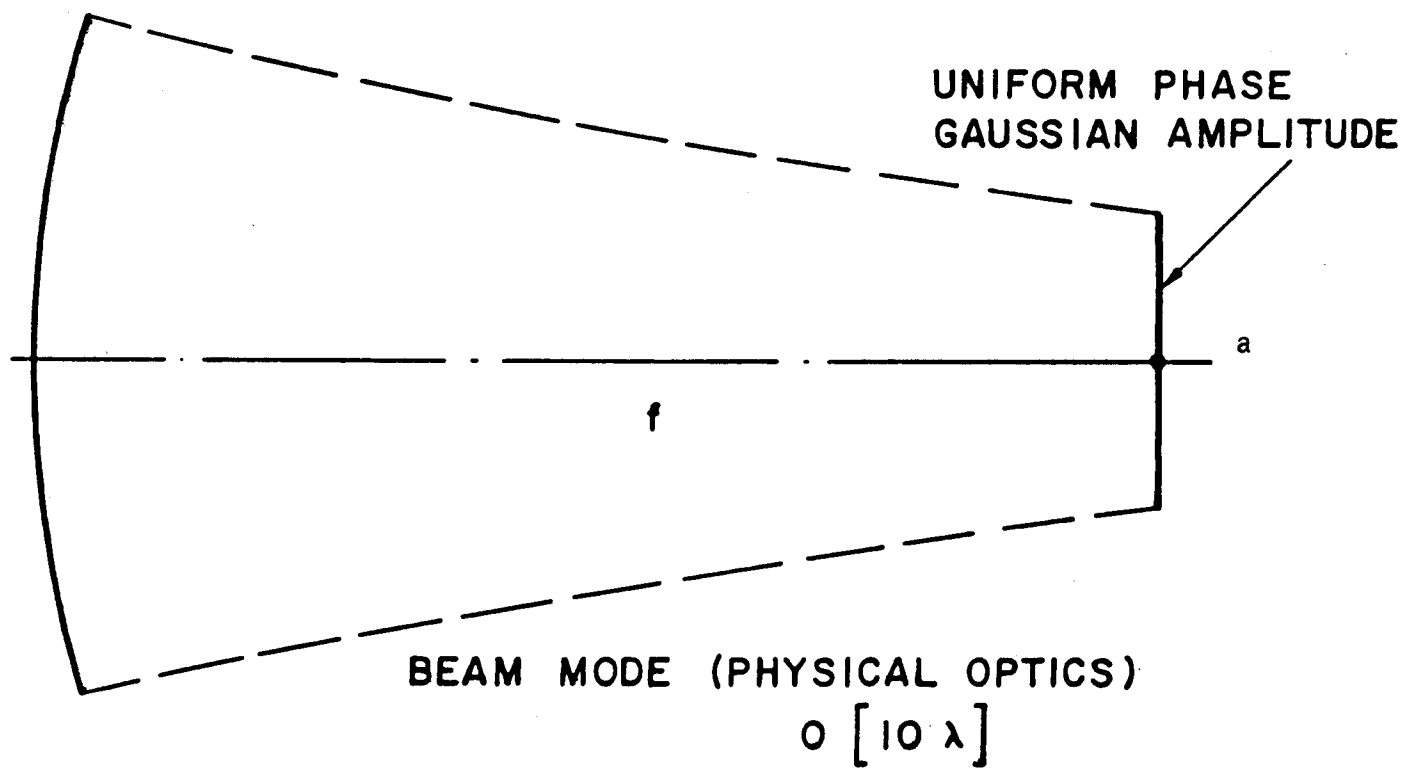


Fig. IV.1-7 Beam mode (physical optics)

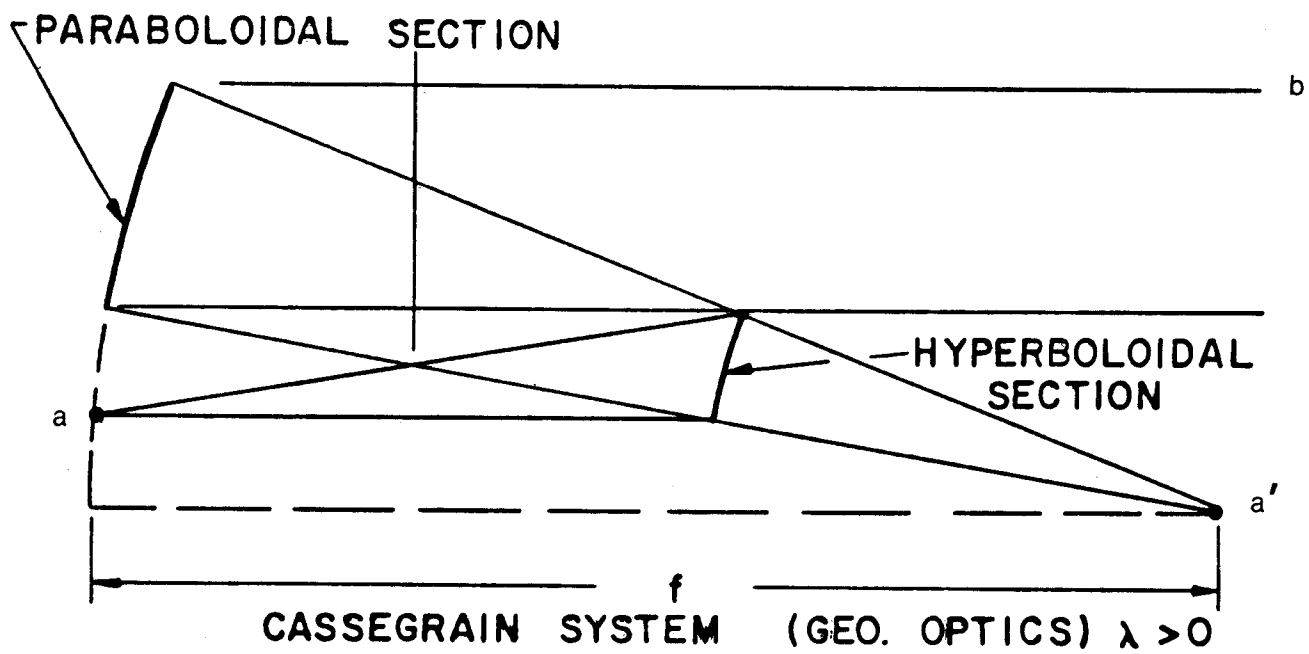


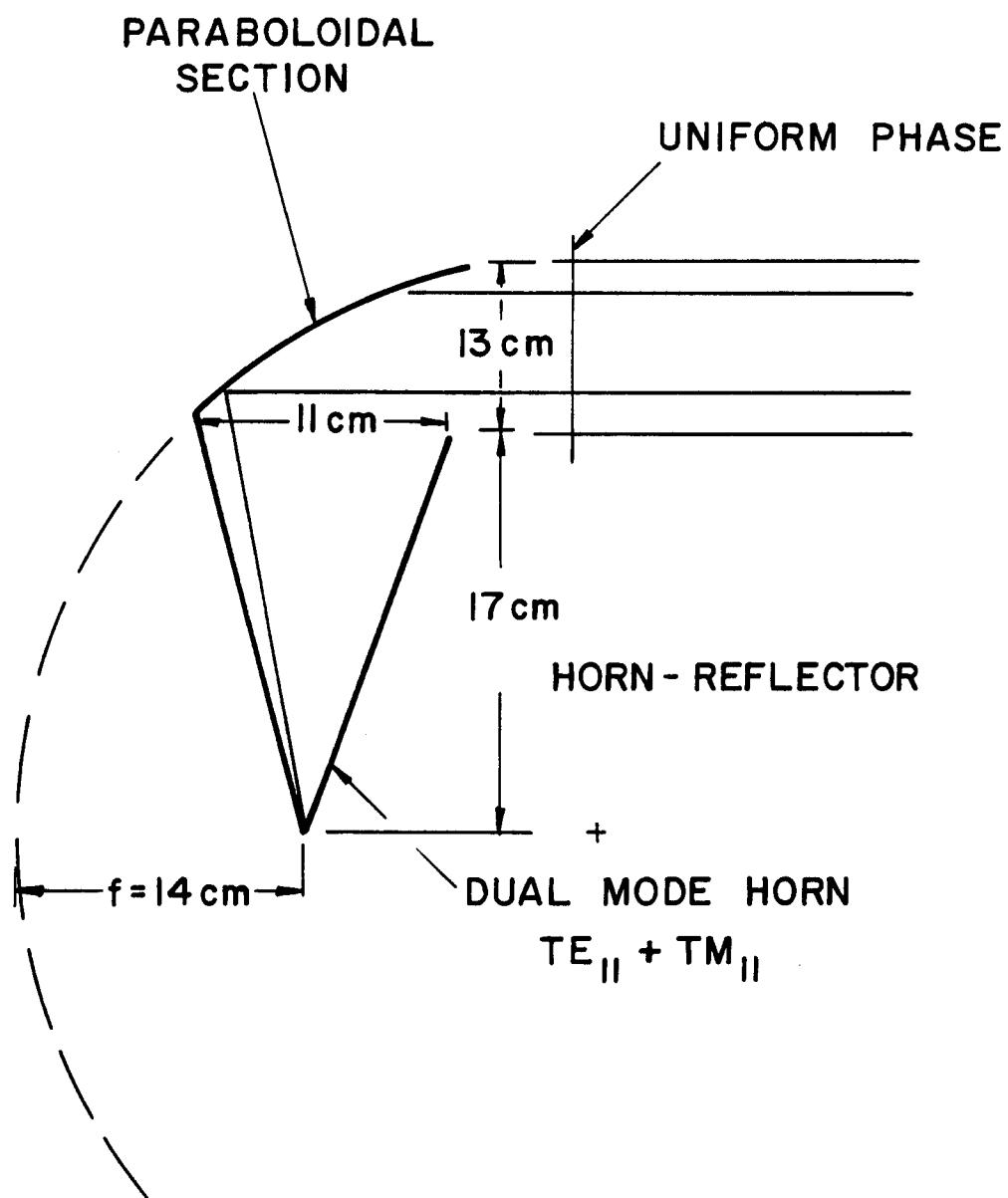
Fig. IV.1-8 Offset Cassegrain antenna

plasma from any desired angle and the source is shielded from the neutron radiation flux by the intervening hyperboloid which can be backed by neutron absorber. A suitable launcher for the fundamental beam waveguide mode is the familiar horn-reflector antenna.

Horn-reflector antennas have been employed for many years in communication systems.<sup>(13)</sup> Such an antenna shown in Fig. IV.1-9 is also an offset paraboloid. Here, however, the apex of a cylindrical horn is placed at the focal point of the paraboloid. The horn has a mode converter in its throat which when fed with the  $TE_{11}$  mode of cylindrical waveguide produces both the  $TE_{11}$  and  $TM_{11}$  modes in the horn. These modes propagating together in the horn produce by interference a near Gaussian radial amplitude distribution at the horn mouth.<sup>(14)</sup> Upon reflection from the paraboloid a uniform phase front is created and hence the resultant aperture field is just that required to launch a beam mode.

Since the dual mode horn must be fed with the  $TE_{11}$  cylindrical waveguide mode and because the gyrotron output is typically  $TE_{0n}$  an intervening mode transducer is required. Suppose, for example, a  $TE_{01}$  gyrotron is used. In that case a stepped twist-reflector will provide the necessary conversion to the  $TE_{11}$  mode. A quarter wave step over one half of the reflector converts the circular symmetric mode electric field into an anti-symmetric field. Now the quarter wave grating of the twist-reflector reverses the polarization of only one of the rectangular components of the electric field with the result that the fields now conform to those of the  $TE_{11}$  mode.

A complete delivery system incorporating each of the principles discussed above is shown in Fig. IV.1-10. This system designed for TASKA could consist of up to 13 gyrotrons distributed around an axis which in turn makes an angle



DIMENSIONS NOT  
TO SCALE

Fig. IV.1-9 Offset paraboloid launcher

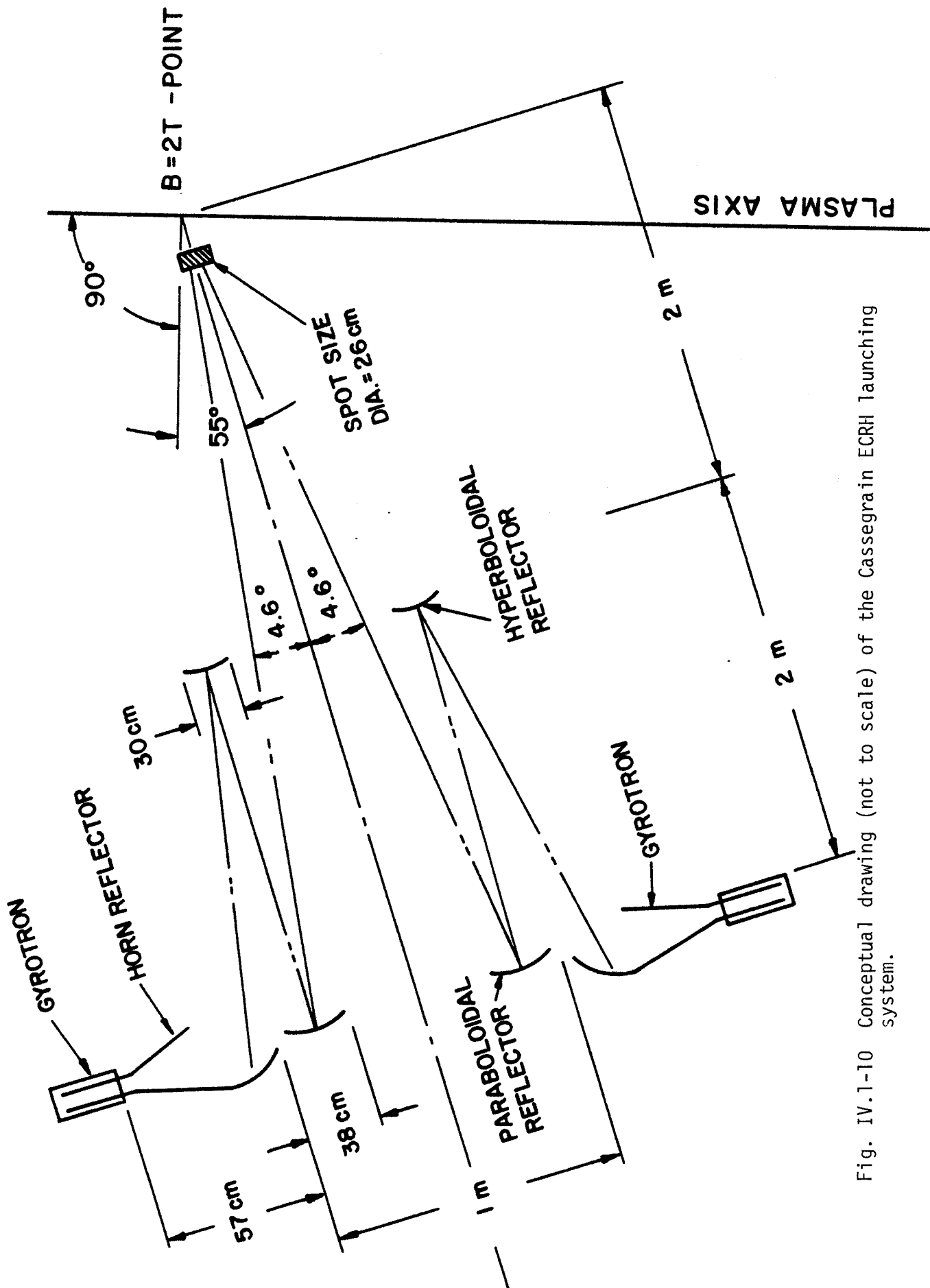


Fig. IV.1-10 Conceptual drawing (not to scale) of the Cassegrain ECRH launching system.

of 35° with the machine axis. For the design given, all beams incident on the plasma lie within an angle of 30° to 40° as required from our ray tracing calculations. Hence power levels per port of up to 4 megawatts are possible with this system if each individual gyrotron is limited to 350 kW. Each gyrotron feeds a stepped twist reflector mode converter which in turn excites the dual mode horn. Since individual sources are limited to 350 kW, breakdown fields in the metallic waveguides governed by

$$E_{\max}^2 = \frac{P_{\max}}{3.97 \times 10^{-3} \sqrt{1 - \left(\frac{\lambda}{3.41 r_0}\right)^2 r_0^2}}$$

are limited to about  $4 \times 10^6$  V/m which pose no breakdown problems in the absence of plasma at  $p_0 < 10^{-6}$  Torr. When the powers are combined, the Gaussian beam's maximum amplitude is dictated by

$$E_{\max}^2 = \frac{2z_0 P_{\text{avg}}}{\pi R^2} \left( \frac{4a^2}{1 - e^{-2a^2}} \right)$$

where "a" is the parameter of Fig. IV.1-6. This results in a field intensity of 113 kV/m for  $P_{\text{avg}} = 4$  MW. This is a worst case calculation since it assumes all fields combining in phase, a highly unlikely event. Even in this worst case the field intensity is low because of the relatively large area over which the energy is distributed, and since no metal boundaries are in proximity to the fields there should be no breakdown problems.

It has been shown that a quasi-optical feed system for plasma heating in fusion reactors is totally compatible with the reactor constraints. Such a

system allows multiple medium power gyrotron sources to be combined into intense beams which can be focused onto the plasma in a well defined cone. The antenna system is derived from an offset Cassegrain and employs Gaussian beams which give good energy concentrations without breakdown and are characterized by an absence of side lobes. The antenna system also isolates the sources from neutron flux and allows the gyrotrons to be located well away from strong magnetic fields as well as providing for good vacuum pumping the intervening region.

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## IV.2 ICRF Heating in Central Cell

### IV.2.1 Introduction

The overall power balance, heating system layout and heating efficiency of TASKA can be greatly improved by considering direct ion cyclotron frequency range (ICRF) heating of the central cell ions. As discussed in detail in Sections III.1-III.4, the potential use of direct central cell heating by an ICRF heating system provides an easier layout for access of high energy neutral beams in the plug and barrier regions with lower total power and energy requirements on them. In addition, the anticipated high heating efficiency of the ICRF system and relatively low cost per watt of RF power coupled to the ions make this an important design consideration. The central cell ICRF system will greatly improve the startup scenario and can add to the control of the plasma burn. The main concern of a central cell ICRF heating system is not in the heating efficiencies or cost factors which have been demonstrated at above the megawatt level in tokamaks such as PLT<sup>(1)</sup> and TFR,<sup>(2)</sup> but in the viability of the antenna structure in the hot central cell region of the test reactor. Since very little work has been done on this subject, we will propose a design based on physics and machine requirements and give a preliminary evaluation of the antenna viability in a reactor environment.

### IV.2.2 ICRF Physics and Design Considerations

The power balance calculations discussed in section III show that a central cell ion cyclotron heating requirement of 40 MW of power coupled to the central cell ions is required for the design operating point. The central cell operates at low magnetic field ( $B_0 \approx 2.7$  T) and the relatively high  $\beta = 50\%$  at full operating regime provides a fundamental deuterium ion cyclotron resonance frequency of  $f_{cd} = 15$  MHz. In this frequency range and given



the radial constraint of the central cell reacting hot plasma edge at  $r = 32$  cm, we are constrained to coil antenna coupling. The high density core plasma with  $\bar{n} = 2 \times 10^{14}/\text{cm}^3$  will provide good coupling in the ICRF range.

In order to improve the antenna coupling to the plasma the second harmonic near  $f = 2f_{cd} = 3f_{ct} = 30$  MHz is proposed as the frequency for the heating system.<sup>(3-5)</sup> This will allow the wave evanescence in the plasma edge to be decreased and allow the wave mode, including the 14 cm thick plasma halo annulus, to be better coupled to the plasma core. Recent  $f = 2f_{ch} = 42$  MHz heating experiments at above the megawatt level on the PLT tokamak have proven quite successful.<sup>(6)</sup> It is expected that a detailed analysis of the scaling of this heating scheme to reactor parameters will also prove attractive. It also allows the possibility of minority fundamental hydrogen heating  $f = 2f_{cd} = f_{ch}$  at lower ion temperatures as part of the startup scenario.

Another consideration for the ICRF system is its location to ensure heating of the central cell ions. The ideal location would be in the central region of the central cell with the magnetic field decreasing slightly towards the ends of the central cell where wave particle resonance and heating occurs. This would ensure that the wave would be fully absorbed in the central cell region and not allowed to propagate into the barrier regions where it could be absorbed and have a deleterious effect on the barrier performance. However, since the REGAT and blanket test modules are located in the central region of the central cell, we consider ICRF coupling regions towards the ends of the central cell.

To provide a directivity of radiated power, we first considered a phased array of four coil sets of two each on each end of the central cell. They would be excited in an  $m = \pm 1$  configuration by 20.3 cm inner diameter coaxial

feeds which will accommodate the 5 MW/two coil set power coupled with a sufficient safety factor to eliminate RF arcing. To provide directivity towards the central part of the central cell a  $\lambda_{\parallel}/4$  array with a  $\phi = 90^\circ$  electrical phase advancement for each successive coil away from the central region could be employed to provide a maximum radiation towards the central region and a null towards the barrier region. However, the parallel wavelength values for the fast magnetosonic wave mode in the plasma region corresponding to  $k_{\parallel} = 5-15 \text{ m}^{-1}$  require that  $\lambda_{\parallel} = 1.26-0.42 \text{ m}$ . At these lower frequencies, the substantial mutual inductance between the closely spaced adjacent coil sets in the vacuum region makes the RF circuit drive problem more difficult. It would be hard to achieve the  $90^\circ$  electrical phase difference required for directivity of the radiation pattern. The adjacent antennas are also very closely spaced (20 cm center to center) for this level of coupled RF power.

We then considered a  $\lambda_{\parallel}/2$  spacing with a  $180^\circ$  electrical phase shift between adjacent coils. This provides a symmetric pattern of radiation in the axial direction but is easier to excite since the mutual inductance term is minimized due to the  $180^\circ$  electrical phase shift between adjacent coils. We must then invoke the property of reflection of the portion of coupled RF power propagating towards the barrier zone before it reaches the ion cyclotron resonant zone. This is feasible since the plasma density decreases drastically as one moves from the central cell to the barrier zone to  $n_B = 6.8 \times 10^{12}/\text{cm}^3$ . The plasma radius is also much smaller than in the central cell so that the  $na^2$  criteria for fast mode propagation is not met and the mode can be localized in the central cell region.

Finally, to protect the antennas from the 3.5 MeV  $\alpha$  particles resulting from the D-T fusions in the low field central cell region we have introduced a

14 cm thick plasma halo around the reacting plasma to allow the antenna to be located some distance from the fusion plasma. This allows the antenna to be located more than an  $\alpha$ -particle gyroradius away to protect it from  $\alpha$ -particle bombardment. The intervening region  $32 \text{ cm} < r < 46 \text{ cm}$  cannot be made a vacuum region since the reduced plasma loading on the antennas would decrease the coupling efficiency and the wave fields would be evanescent from the antenna to the fusion plasma surface. Thus, the halo plasma with high density and lower temperature allows the fast wave to propagate while locating the antennas at a safer distance from the reacting fusion plasma.

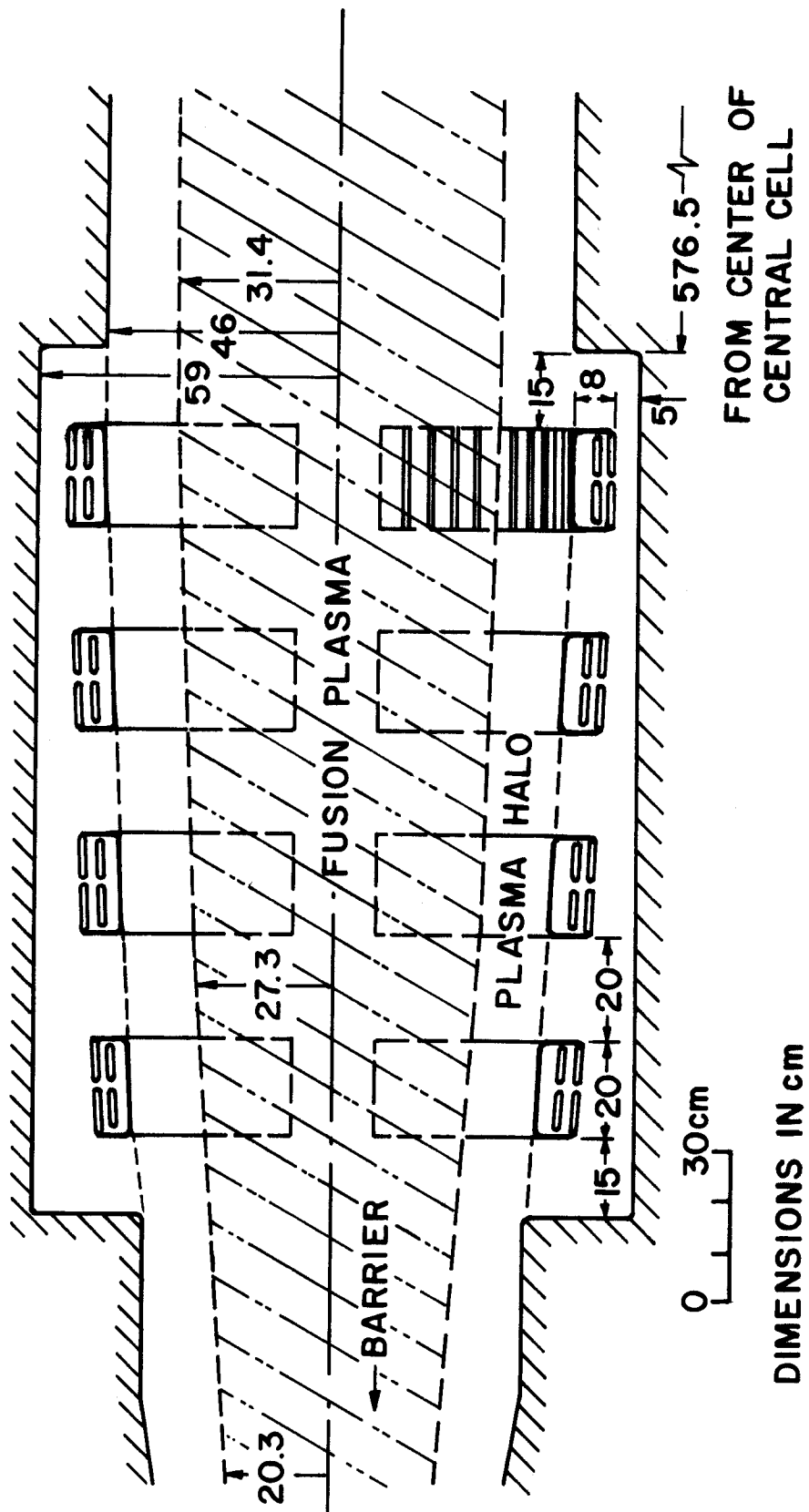
Another consideration involves the launching and absorption of the RF power with the deuterium ions. An axial magnetic beach is proposed. This is desired since we wish to launch the RF power in a region where the wave frequency is well below ( $\approx 10 - 15\%$ ) the second harmonic deuterium resonance and avoid local harmonic resonance of ions which could strike the antennas and interfere with an efficient coupling process. The wave then propagates from the ends of the central cell towards the central region where the magnetic field decreases by 10% and deuterium absorption occurs. For the 30 keV central cell ion temperatures and a dominant  $k_{\parallel}$  spectrum component of  $k_{\parallel} = 7.85 \text{ m}^{-1}$ , the  $k_{\parallel} v_i$  Doppler width corresponds to 5% of the second harmonic resonance at  $f = 30 \text{ MHz}$ . At these higher densities,  $n = 2 \times 10^{14} / \text{cm}^3$ , the deuterium-tritium equilibrium and ion isotropization times will be sufficiently rapid so that both ion species will be heated to the same temperature and instabilities arising from a high degree of ion thermal anisotropy will not affect confinement. If additional flexibility is desired in determining the axial heating profile, the antennas can be driven at slightly different

frequencies or a more broad band source excitation can be considered to tailor the axial power deposition.

#### IV.2.3 Antenna Layout

A more detailed drawing of the ICRF coil antenna launching region is shown in Fig. IV.2-1. The fusion reacting plasma producing  $\alpha$  particles is given by  $r \leq 32$  cm. A 14 cm halo plasma of sufficiently high density to allow fast wave coupling and propagation ( $n > 8 \times 10^{12}/\text{cm}^3$ ) and sufficiently low temperature to minimize antenna Faraday shield erosion is produced between the core plasma and antenna. The antennas and Faraday shield are of 8 cm radial by 20 cm axial extent and are of an all conductor, no dielectric design scaled from recent designs used on TFR and PLT. Finally, a 5 cm gap between the back plate of the antenna Faraday shield and the first wall is required to minimize wall image currents affecting coupling efficiency.

Each antenna is of 20 cm axial extent with a 20 cm space between each Faraday shield. The electrical phase alternates by  $180^\circ$  between each antenna to minimize the mutual inductance term for the driving circuit and provide the dominant  $k_{\parallel} = 7.85 \text{ m}^{-1}$  spectrum component. The total RF coil system (8-2 coil sets) then take up 2.8 m of axial extent of the central cell. The antennas can be end fed by a 20 cm diameter coax with 5 MW/two coil set power handling capacity. The 20 cm diameter coaxial feed through was scaled to provide an adequate breakdown protection based on current PLT experiments at 42 MHz. The two antenna set on PLT is fed by 10 cm diameter coax and has a coupled power capability to the plasma of 1.5 MW. The proposed 20 cm diameter coax should provide an adequate safety margin for the 5 MW/two coil set on TASKA. The Faraday shields surrounding the antenna coils will suppress axial electronic fields between the antennas. However, an RF test facility would be useful to



IV.2-6

## CENTRAL CELL ICRF ANTENNA LAYOUT

Figure IV.2-1

test breakdown and material limitations for the high ICRF power density proposed for TASKA. At a frequency of 30 MHz and the wall diameter of 92 cm, a center fed resonant  $\lambda/2$  antenna of 50 cm circumferential extent can also be considered to optimize the plasma load impedance to the RF drive and matching circuit to minimize high voltage breakdown. The feed system must be optimized for impedance matching and modular design for rapid replacement of antenna modules. By recessing the first wall to accommodate the antennas, the potential axial heat flux on the antenna array is eliminated.

#### IV.2.3 Materials Considerations

The antenna coils and Faraday shields must be made of good conductors which will withstand the neutron bombardment, radiation damage, plasma heat flux and surface erosion. If metallic material or copper clad stainless steel were used, cooling would be required. Water cooling is readily introduced to the central coaxial conductor by means of quarter-wave coaxial stubs outside the reactor vessel. The higher operating temperatures can anneal the neutron induced loss of high conductivity in these materials, keeping conductor RF power losses to an acceptable level. Another material which was considered is pyrolytic graphite. It has a high conductivity, high temperature capability, and could be considered to radiatively cool without the need for water cooling. It is also a low Z material and would help to lessen the impurity radiation problem. Its structural integrity and resistance to cracking and fracture will have to be further evaluated for use in fusion plasmas.

Furthermore, graphite is known to be very susceptible to neutron radiation damage. It is subject to large dimensional changes<sup>(7)</sup> which are accompanied by the formation of microcracks throughout the material. As a

result, it is expected that the electrical conductivity of pyrolytic graphite will rapidly degrade with neutron irradiation damage.

It is, therefore, proposed to employ metallic materials for the antenna coils and the Faraday shields. In order to obtain good electrical conductivity, the antenna coils are made of austenitic stainless steels plated with high-conductivity copper. As demonstrated by Haubenberger<sup>(8)</sup> the plating of copper on austenitic steels can be accomplished by electrolytic or explosive methods. The advantage of this composite metal is that the high-conductivity copper is needed only where the RF current is carried, namely in the skin depth layer. The surface temperature for the copper layer is chosen to be in the range from 270°C to 330°C where most of the neutron damage does not significantly contribute to a degradation of electrical conductivity. On the other hand, the temperature is sufficiently low for austenitic stainless steel so that swelling poses no severe problem.

The Faraday shields are made of molybdenum sections, which are loosely attached by appropriately machined tongues and grooves to the rear antennas. In contrast to the antennas which are cooled by water, the Faraday shields are cooled by radiation. Therefore, they reach temperatures estimated to be above 700°C. These temperatures are sufficiently high for molybdenum to avoid the radiation-induced embrittlement and a large shift of the DBTT.

The Faraday shield acts to ensure the coupling of the fast magnetosonic mode, minimizes electrostatic mode coupling and protects the coil conductors from particle bombardment and plasma flux which could cause internal arcing. As such, some erosion of the outer Faraday shield surface will not cause RF power loss in the antenna and will protect it. Further materials testing and

an RF test facility are required to develop materials for antenna coils for fusion reactor plasmas.

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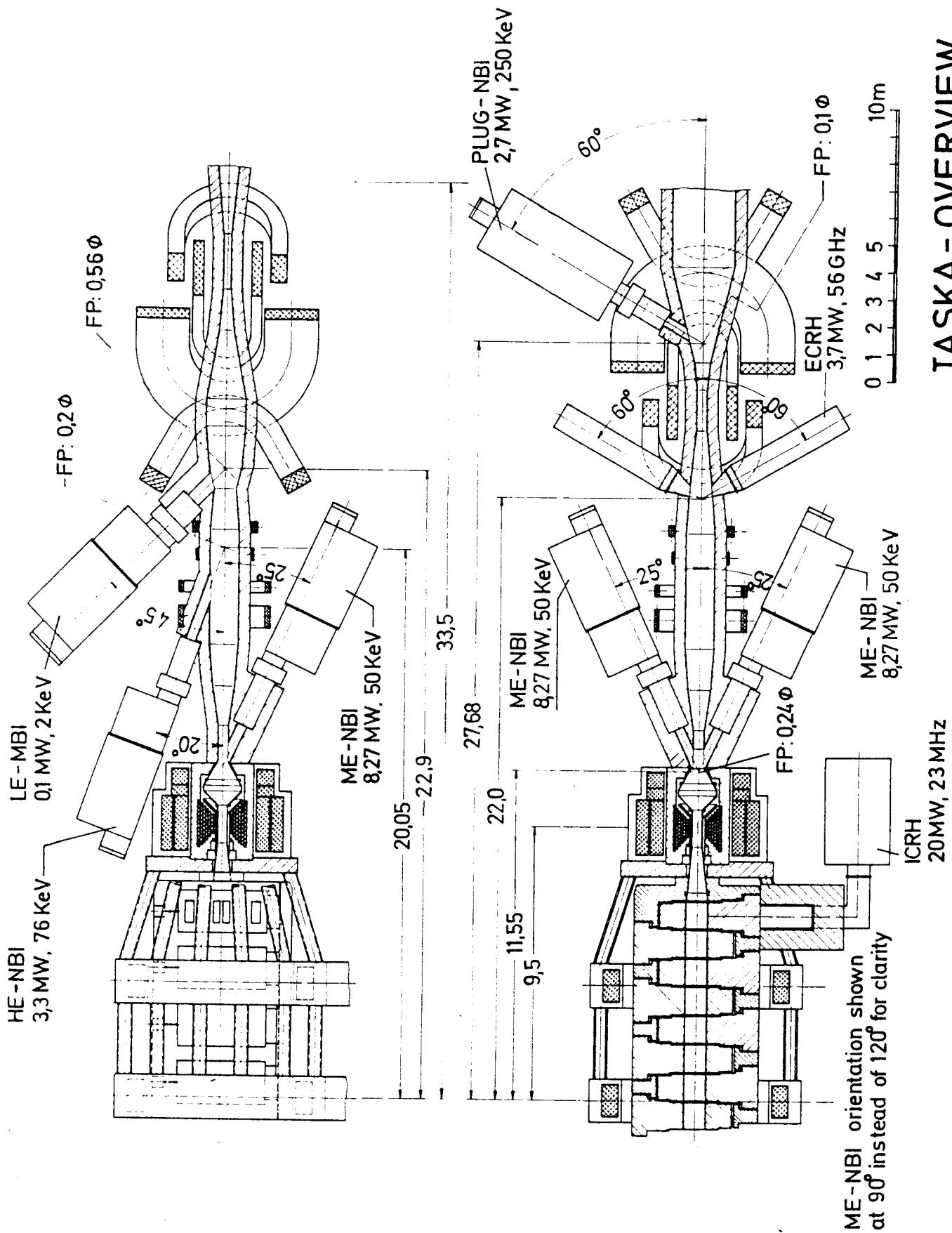


### IV.3 Neutral Beam Injection

#### IV.3.1 General

Neutral beam injection is the primary means for initial hot ion buildup in the plug, plasma heating, plasma fueling, and selective charge exchange barrier pumping in the TASKA tandem mirror device. The overall neutral beam injection requirement is generally divided into convenient modular (practical sized) neutral beam injector (NBI) units each producing an output atom current at a designated energy suitable for injection into the appropriate region of the plasma. The steady state NBI system for TASKA includes a pair of very high energy plug neutral beams (plug NBI) to help buildup and to maintain the plasma in the high potential end plugs, a pair of high energy pump beams (HE-NBI) to pump trapped ions out of the barrier minima, a set of six medium energy pump beams (ME-NBI) arranged for both barrier cell pumping and central cell fueling, and finally a pair of low energy neutral beams (LE-NBI) located to inject into a region of the thermal barriers where the plasma potential is such as to allow a low energy NBI's to pump out the higher energy trapped ions. The general layout of these NBI on the TASKA confinement vessel is shown in Fig. IV.3-1. The specific NBI equivalent current and energy requirements for each region and each function in TASKA are the result of a set of parametric code studies of plasma equilibrium needed for the neutron wall loading goal of  $1.5 \text{ MW/m}^2$  (described in Section III.3). The specific NBI parameters needed to produce the specified incident (on plasma) neutral beam current and power for TASKA are given in Table IV.3-1, including the estimated ion source high voltage power supply current requirements (with a 20% margin).

For simplicity in initial wide ranging parameter studies, the equilibrium code calculations and the resultant NBI parameter listing in Table IV.3-1 are



# TASKA - OVERVIEW

Fig. IV.3-1

Table IV.3-1. TASKA Neutral Beam Injector Requirements

|                                                  | <u>Units</u>      | <u>Plug Nb</u> | <u>HE-NBI</u>   | <u>ME-NBI</u>              | <u>LE-NBI</u>                     |
|--------------------------------------------------|-------------------|----------------|-----------------|----------------------------|-----------------------------------|
| Location                                         |                   | Plugs          | Barriers        | Barriers                   | Barriers                          |
| Injection angle                                  | Degrees           | 60             | 20              | 25                         | 45                                |
| $V_0$                                            | kV                | 250            | 76              | 50                         | 2 to 5                            |
| $I_0$ /NBI                                       | A                 | 10.8 ( $H^0$ ) | 43.4 ( $D^0$ )  | 3 x 55<br>( $D^0 + T^0$ )  | 50 ( $D^0$ )                      |
| $P_0$ /NBI                                       | MW                | 2.7            | 3.3             | 8.3                        | .1 to .25                         |
| # Ion sources per<br>NBI                         |                   | 1 est.         | 1               | 3                          | 1                                 |
| Total # NBI                                      |                   | 2              | 2               | 6                          | 2                                 |
| $I^\pm$ estimate (for<br>$I_0$ w/ 20% margin)    | A                 | 23 ( $H^-$ )   | 108.8 ( $D^+$ ) | 3 x 121<br>( $D^+ + T^+$ ) | 97.5 ( $D^+$ )                    |
| $j$ (s.c.l.)                                     | A/cm <sup>2</sup> | .05            | .27             | .3                         | .25                               |
| Extraction area<br>( $f_t = .45$ )               | cm <sup>2</sup>   | ~ 20 x 50      | 20 x 45         | 20 x 45                    | 20 x 45                           |
| Beam focal length                                | m                 | 7.5            | 12              | 7.5                        | 6.3                               |
| Beam divergence                                  | Degrees           | ≤ .3*          | .5*             | .7*                        | 3.5* to 2.2*                      |
| Collimation (at<br>TASKA wall;<br>single source) | cm <sup>2</sup>   | ~ 16 x 21      | 38 x 47         | 34 x 39                    | 81 x 110<br>(5 kV un-<br>focused) |
| Min. footprint<br>(radius)                       | cm                | 8**            | 12              | 9.3                        | ~ 38 (5 kV)                       |
| Gas flow (total)                                 | Torr-l/s          | ~ 20           | 15              | 50                         | 13                                |

\*Divergences quoted are conservative 1/e values with reduction of up to a factor of two likely within the planning time for construction.

\*\*Minimum footprint quoted for slightly defocused beam to reduce beam loading on the remnant neutral beam dump.

both based upon only a single full energy species output (e.g.,  $D^0$  at an energy  $eV_0$ ) from each NBI. Therefore, some revisions are to be expected in the NBI requirements of Table IV.3-1 when more refined equilibrium and self-consistent NBI calculations are made using the actual detailed output species composition from the ion sources. The best present  $D_2$  fed ion sources yield an ion beam of 80%  $D^+$ ,  $\sim 15\%$   $D_2^+$ , and  $\sim 5\%$   $D_3^+$  (instead of 100%  $D^+$ ), with the resultant neutral beam output including a mixture of full energy ( $eV_0$ ), one-half energy, and one-third energy  $D^0$  components after dissociation and charge exchange in the neutralizer cell. The result for this typical species mix is effectively 1.25 A atomic ( $D^+$ ) current per A of electrical current, but with an average  $D^+$  energy =  $.8 eV_0$ .

The NBI's listed in Table IV.3-1 and their associated electrical power supplies and gas pumping vacuum system represent a significant portion of both the capital facility and operating costs of the machine. The machine performance (neutron wall loading) is predicted by code studies to be strongly dependent on the maximum performance of the NBI system (and also of the required ECRH and ICRH systems). Therefore part of the NBI task is to get the maximum reliable performance from the NBI system.

This chapter will therefore augment the simple NBI parameter listing of Table IV.3-1 by including a brief discussion of a typical positive-ion source neutral beam lines in Section IV.3.2 (to illustrate the factors limiting NBI performance), then describing a specific TASKA NBI prototype design selection in Section IV.3.3, and adding a few ion source details for ion source design selection in Section IV.3.4 for reference purposes. A negative ion source based NBI appears essential for efficient high energy NB production (e.g., for the 250 kV  $H^0$  plug NBI), but is described only briefly in Section IV.3.5 due to

uncertainty as to which type of negative ion source will be found most practical and most efficient for high energy neutral beam formation.<sup>(1)</sup> Note, however, that the hot ion heating function of the 250 kV  $H^0$  plug NBI can also be attained through use of a heavier mass 375 kV  $He^0$  NBI using existing positive ion technology, if  $H^-$  source technology does not develop as rapidly as expected. Helium gas, however, is difficult to vacuum pump, and therefore this less desirable approach was not used for the TASKA reference design.

#### IV.3.2 Typical Positive Ion Source/Beam Line System

Each NBI unit includes one or more close coupled ion source, neutralizer cell and remnant ion sweep set(s) (as listed in Table IV.3-1), plus associated gas pumping panels, beam collimation apertures, and provision for beam diagnostics. The ion source tank section containing this equipment is in turn connected via a bellows (with provision for adjusting the ion source tank orientation for beam steering), a vacuum isolation valve, and ducting to the appropriate position on the TASKA confinement vessel. Extensive magnetic shielding and appropriate neutron shielding are required for the ion source, neutralizer cell and remnant ion sweep set(s). Finally, each NBI line requires a remnant neutral beam power (and gas) dump on the far wall of the confinement chamber. A sketch of a typical NBI system is shown in Fig. IV.3-2.

The ion sources, neutralizer cells, and associated gas pumping system are expected to largely determine the limits on the NBI performance. In particular, each ion source requires a relatively high amount of support technology for operation and represents a very high power instrument. As such, the ion source designs and power supply specifications are required to be within conservative specifications to allow long-lived and relatively trouble free service. The beam heat loads incident upon the ion sweep dump, collimation

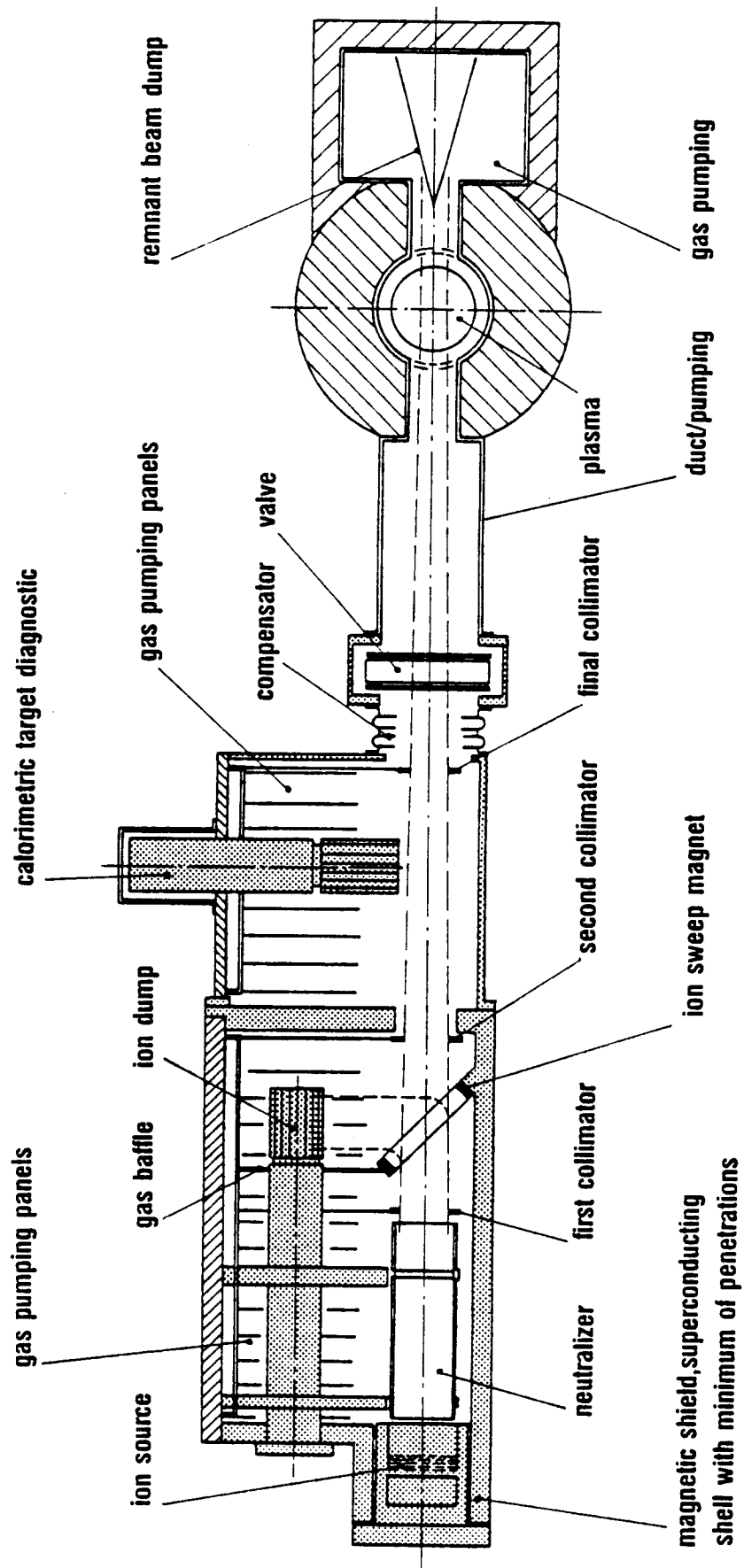


Fig. IV.3-2 **Neutral beam injector**

apertures, diagnostic calorimeter (when inserted into the beam), and the remnant neutral beam dump are often  $\geq 2 \text{ kW/cm}^2$ , so special design is required for adequate cooling and long lifetime. The LBL/IPP conceptual design study for the Zephyr NBI system is a good reference base for NBI details.<sup>(2)</sup>

#### IV.3.2.1 Primary Limitation to Ion Source Output

The primary limitations to the ion source output result from (1) the space charge limit on the extracted ion density, (2) the limited transparency of the electrode set needed to form the ion extraction and acceleration field, (3) the need for a reasonably small sized electrode area to allow for cooling and to maintain accurate inter-electrode gap and extraction aperture set alignment (between electrodes), and (4) the intrinsic divergence of the ion beam. A typical ion source<sup>(3)</sup> is shown in Fig. IV.3-3.

##### IV.3.2.1.1 Space Charge Limited Current Density $j(\text{s.c.l.})$

The fundamental limit to the maximum extracted ion current density arises through the Childs-Langmuir equation for the space charge limited current density. The generalized<sup>(4)</sup> form for a mixed ion species beam is:

$$\sum_i j_i M_i = \frac{1.72 V^{3/2}}{d^2}$$

where:  $j_i \equiv$  current density of  $i^{\text{th}}$  species ( $\text{mA/cm}^2$ ),

$M_i \equiv$  mass of  $i^{\text{th}}$  species (a.m.u.),

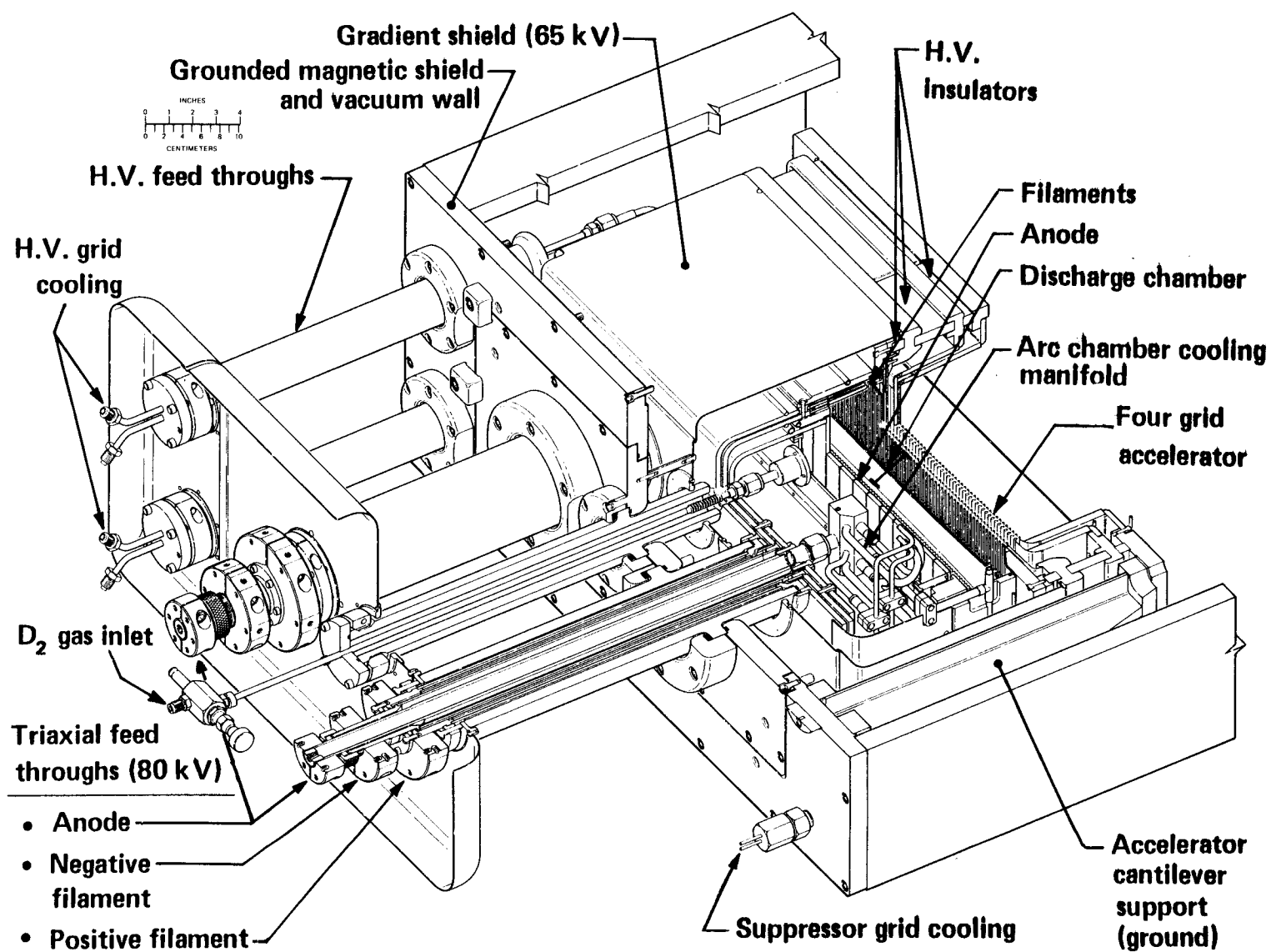
$V \equiv$  total voltage applied across the extraction gap (kV),

$d \equiv$  effective extraction gap (cm).

For a three electrode system  $V \approx V_0 + |V_S|$ , where  $eV_0$  is the beam ion energy of interest and  $V_S$  is the (negative) electron suppressor electrode voltage. For good optics  $V_S$  is generally kept to the minimum needed (a few kV) to suppress

Fig. IV.3-3

# MFTF 80 keV -- 80 "AMP" NEUTRAL BEAM





electron backstreaming (from the neutralizer cell region back into the ion source), so generally  $V \sim V_0$ . The extraction gap is limited by the need for a sufficient inter-electrode gap to avoid voltage breakdown. In general, the effective gap  $d$  is equal to the physical electrode gap plus a fraction of the characteristic aperture dimension. For a typical polished set of plane electrodes, the limiting breakdown voltage<sup>(5)</sup> over a physical gap  $d$  is  $V_{BD}(kV) \sim 60 \sqrt{d} \text{ (cm)}$ . The above space charge considerations form an upper limit on the total  $j(s.c.l.) < 0.5 \text{ A/cm}^2$  (typical).

#### IV.3.2.1.2 Extraction Electrode Transparency $f_t$

Formation of a multi-ampere total ion beam then requires extraction from a substantial plasma area. Maintaining the required electrostatic field gradient (for a short gap over a large area) and shaping the plasma meniscus boundary for ion trajectories that balance the defocusing space charge forces against the focusing field from the aperture for minimum angular divergence output requires use of a set of multi-aperture electrodes for extraction. The first electrode of such a grid set reduces the effective plasma area available by a transparency factor  $f_t$ ; typically  $f_t \leq 0.45$  for circular apertures and  $f_t \leq 0.6$  for slot apertures. The average space charge limited current density for apertures then is  $\sim f_t \times j(s.c.l.)$ .

#### IV.3.2.1.3 Extraction Electrode Area

The extraction electrodes for steady state multi-aperture ion sources are generally water cooled by flow through many capillary tubes. The required cooling flow through the small capillary tubes tends to limit the extractor grid dimension in the corresponding flow ("height") direction to  $\leq 20 \text{ cm}$ . The need for highly accurate dimensional stability in both the extraction gap and aperture alignment similarly limits the orthogonal ("width") dimension. Dimen-

sions of up to approximately 20 cm x 50 cm appear feasible, though from an ion source failure standpoint even smaller modular units would be desirable if they can be packed closely together. Electrode cooling limitations, the need for a long electrode lifetime (minimum erosion loss from sputtering), and wall cooling requirements in the plasma source section of the ion sources combine to further limit extraction power densities. Therefore, average extraction current densities for high voltage steady state ion sources are expected to range from 0.1 to 0.3 A/cm<sup>2</sup>, with total extraction areas < 10<sup>3</sup> cm<sup>2</sup>.

#### IV.3.2.1.4 Intrinsic Ion Beam Divergence $\theta(\text{div.})$

Another basic factor limiting ion source performance is the intrinsic angular divergence  $\theta(\text{div.})$  of the ion beam resulting from a combination of initial transverse ion motion in the plasma chamber, plus any unbalanced transverse ion beam motion from distortions in the plasma meniscus formed for each aperture at the beam forming extraction electrode, or from any aperture grid to grid misalignment. The value of  $\theta(\text{div.}) \approx \tan^{-1}(v_{\perp}/v_z)$  where  $v_{\perp}$  is the total transverse ion velocity and  $v_z$  is the final z component of the velocity after acceleration along the z axis. In practice  $\theta(\text{div.})$  is experimentally measured for the total ion beam resulting from the merging of an array of beamlets, while code studies involve minimizing the divergence of a single typical beamlet for the selected operating parameters.

For scaling and planning purposes in this study a conservative empirical fit for much of the multiple circular aperture 1/e divergence data<sup>(4)</sup> is adopted:

$$\theta(\text{div.}) \approx \frac{5}{\sqrt{V_0}(\text{kV})} \text{ degrees} \quad .$$

A similar approximate scaling can be used for slot apertures, with  $\theta(\text{div.})$  parallel to the slots having a slightly lower coefficient and the divergence perpendicular to the slots higher by at least  $\sqrt{2}$ . Recent data for ion sources now under test show this empirical divergence reduced by  $\sim 0.7$ . The initial ion transverse motion implied for these improved sources is still relatively large ( $T_{i\perp} \sim 3.7$  eV if thermal) so it likely represents an intermediate limit with a lower  $T_{i\perp} < 1$  eV possible with expected further plasma source development. However the  $D^+$  ions from  $D_2$  or  $D_2^+$  dissociation in the plasma sources have  $\sim$  Frank-Condon energies of a few eV, so additional ion-wall accommodation is needed for cooling  $T_{i\perp}$  (without again leading to molecular recombination).

#### IV.3.2.2 Ion Source Output Composition

Positive ion sources operating on a molecular gas such as  $D_2$  (etc.) generally yield a mixture of primary atomic ions ( $D^+$ ), diatomic ions ( $D_2^+$ ), and triatomic ions ( $D_3^+$ ). The exact mixture depends largely upon details of the arc operation, and upon the plasma chamber walls. For most applications, a 100%  $D^+$  beam would be most desirable but present large sources appear to be limited to  $\sim 80\%$   $D^+$ ,  $\sim 15\%$   $D_2^+$ , and  $\sim 5\%$   $D_3^+$ .

#### IV.3.2.3 Impurity Limitation

Tandem mirror machines with substantial electrostatic potential end plugs or barriers, and closed magnetic field (e.g., tokamak type) confinement configurations are sensitive to accumulation of high Z impurities because of the expected relatively long radial diffusion loss times. Initial estimates for impurity plasma buildup in a tandem barrier indicate an ion source limitation of  $< 0.1\%$  high Z ( $Z > 2$ ) impurities is required.<sup>(6)</sup> Several means of selectively ejecting impurities appear possible if necessary.

#### IV.3.2.4 Ion Current Loss During Extraction

In operation, the extraction electrode set (three or four electrode systems) contributes to an accelerated ion current loss and excess power supply drain. The major portion of this loss appears due to the ion beam ionizing some of the background gas flowing through the electrode apertures. The resultant ion and electron trajectories then impinge upon an electrode or wall and contribute to the electrical power supply current drain. The primary ion can also be lost through charge exchange on this gas, impact one of the electrodes, and possibly eject secondary electrons and thereby increase further the electrical current drain. The total acceleration power supply current loss for an early LBL 20 kV ion source design has been measured by Berkner<sup>(7)</sup> to be  $\sim 20\%$ ; this yields an effective acceleration efficiency  $\eta_{acc} \approx 80\%$ . A higher  $\eta_{acc}$  is expected for the higher gas efficiency ion source design adopted for the TASKA NBI due to lower gas flow and density.

#### IV.3.2.5 Neutralization Efficiency

The beam line charge exchange neutralizer efficiency for conversion of a positive ion beam to a neutral beam is limited because the rate of electron pickup by the ion (cross section  $\sigma_{10}$ ) can come into equilibrium with the rate for electron loss (cross section  $\sigma_{01}$ ). The equilibrium fraction  $F_{\infty}^0 = \sigma_{10}/(\sigma_{10} + \sigma_{01})$  is a rapidly decreasing function of energy, limiting reasonable positive ion beam conversion to ion beams with energies  $< 100$  keV/nucleon. In practice, gas scattering and excess gas flow and pumping problems limit the gas cell to a thickness  $\int n \, dl \sim 3/(\sigma_{10} + \sigma_{01}) \sim 10^{16}/\text{cm}^2$  for a conversion value  $\sim 0.95$  of the infinitely thick value. The neutralizer efficiency for molecular ions is similar (for a sufficient cell thickness for nearly complete dissociation), except the cross sections are evaluated at the

reduced fragment energy (1/2 or 1/3 energy). However, the molecular ion dissociation process also adds approximately the Frank-Condon energy to the resultant fractional energy neutral components increasing their angular divergence (over that of the incident ion beam). In principle the neutralizer gas need not be the same as that of the ion source, but in practice it generally is, because the cell must be closely coupled to the ion source (ion beam transport without good space charge neutralization is impractical) and the need to avoid high Z impurities (that would be expected from any higher Z neutralizer gas back flow into the ion source though carefully designed metallic vapor curtain designs might be considered). The only useful exceptions are that  $H_2$  or  $D_2$  could be used as the charge exchange mediums for a  $T^+$  NBI. The best low gas flow designs for the required  $\int n \, dl \sim 10^{16}/\text{cm}^2$  thickness neutralizer cell appear to use cooled (77°K) neutralizer walls for the main length and warm walls near the beam exit. Any additional neutralizer gas beyond that required with the minimum gas flow from the ion source should be added in the neutralizer cold wall section to reduce the neutralizer gas temperature (and flow velocity).

#### IV.3.2.6 Ion Sweep and Dump

The removal of the remaining ion equilibrium fraction of the beam at the exit of the neutralizer cell requires an ion sweep system, a high power density beam dump (or a direct energy collector), and appropriate gas pumping. The reason for this requirement is that the ions in the beam path beyond the end of the magnetic shielding would be bent by any transverse magnetic fringe field to strike the wall and possibly cause undesirable intense local heating and gas desorption. The ion sweep section is intended to remove these ions at a selected location with appropriate cooling design capability. The ion sweep

can be either magnetic or electrostatic in action with a  $\sim 90^\circ$  deflection magnetic geometry being the most straightforward. The presence of fractional energy ions after molecular ion dissociation in the charge exchange neutralizer cell must also be considered in the magnetic ion beam dump (or direct energy recovery unit) design.

#### IV.3.2.7 Neutral Beam Line Transport Efficiency

The neutral beam line transport efficiency  $\eta_t$  involves two general types of losses. The first fundamental one being from the finite "Gaussian" beam density distribution  $j = j_0 \exp[-(\theta/\theta_{div})^2]$  for a beam with an angular divergence  $\theta_{div}$  (from the ion source or added in the neutralizer from scattering and molecular ion dissociation) contributing to beam losses on the beam line defining collimation. The second loss process is due to neutral beam ionization on any background gas with a beam attenuation  $\propto e^{-\int \sigma_T n dl}$ . An energetic ion formed from neutral beam ionization beyond the magnetic shielding would be deflected in any magnetic fringe field and likely impinge upon a wall and be lost. A total transport efficiency  $\eta_{transp.} \sim 0.9$  is typical,<sup>(8)</sup> but improvement is possible with a lower beam divergence and better gas pumping for a lower gas density ( $\int n dl$ ) from the ion sweep to the TASKA plasma. Use of a focused ion beam input to the neutralizer can allow smaller collimation aperture dimensions (and hence reduce gas flow beyond the ion sweep section), and reduce the NBI footprint size on the plasma.

#### IV.3.2.8 Neutral Beam Support System

##### IV.3.2.8.1 Electrical Power Supplies

For operation, each ion source requires several types of programmed power. Each ion source involves a plasma source requiring substantial cathode and arc power (at high voltage if in the standard positive ion source configu-

ration, or near ground if the neutralizer sweep sections are arranged at negative high voltage - see Section IV.3.4.3). The main ion source power required is the high voltage acceleration supply, with associated current limiting controls in case of ion source sparking or breakdown. An intermediate voltage tap is also required for the gradient electrode if used with a four electrode system (desirable for  $V_0 \gtrsim 50$  kV). Finally, a smaller few kV negative power supply is required to maintain the electron suppressor grid negative to prevent electron acceleration from the neutralizer back into the ion source.

#### IV.3.2.8.2 Gas Pumping System

Each ion source unit requires a minimum gas flow into the arc chamber for plasma formation and to support the extracted ion beam flow. In addition, sufficient gas must be added in the neutralizer for a total  $\int n \, dl \gtrsim 3/(\sigma_{10} + \sigma_{01}) \sim 10^{16}/\text{cm}^2$ . A high gas efficiency nominally 50 A (atomic ion equivalent current yield) 50 kV NBI is expected to require a total  $D_2$  (or  $T_2$ ) gas flow of  $\sim 15$  torr-l/s. Preliminary calculations indicate that at least three pumping stages are desirable for a duct pressure  $< 2 \times 10^{-5}$  torr and that the final duct must be short ( $< \text{few m}$ ), well cooled, and shielded from any direct energetic beam impingement to avoid thermal "Stewart" desorption and possible duct choking.<sup>(9,10)</sup>

The computer code studies require a 44%  $D_2$  and 56%  $T_2$  fueling mixture of the ME-NBI beam, which can be obtained either through the use of separate  $T^0$  and  $D^0$  beam lines or by using a mixed  $D_2$ ,  $T_2$  and DT input in each of the ME-NBI. The mixed gas approach appears the most desirable, as it would allow most of the ME-NBI gas effluent to simply be purified and recompressed for recycling, with just enough makeup gas added from the cryo-distillation facility to maintain the needed flow and gas composition. A relatively smaller cryo-

distillation plant could then be used to separate the  $H_2$ ,  $D_2$  and  $T_2$  fractions from the vessel end wall pumps for use with the appropriate NBI as well as remove the accumulated fusion product He.

#### IV.3.2.8.3 Magnetic Shielding

Operation of the NBI with short ducts and small injection angles in the relatively intense fringe magnetic field of a few tesla from TASKA will require extensive magnetic shielding. The ion source is the most sensitive portion of a NBI with a perpendicular field requirement of  $< .0001$  T. The beam path through the neutralizer section to the low particle density sweep region must also have  $\int B_{\perp} dl$  fields  $\ll 1.50 \times 10^{-2}$  T-cm to avoid excessive ion beam deflection prior to charge exchange conversion from an ion to a neutral. The task appears to require either superconducting shielding, or an extensive set of bucking coils to reduce the fields to a range handled by Fe and  $\mu$ -metal. Details of the magnetic shielding for the TASKA NBI are described in Section V.6.

#### IV.3.2.8.4 Remnant Neutral Beam Dump

A remnant neutral beam power dump and associated gas pumping system is required at the far vessel wall for each of the high power NBI's (all except the LE-NBI). The specific arrangement for TASKA is given in Section VIII.3. During TASKA startup these dumps must be capable of handling a major fraction of the NBI power (typically just beyond the design beam focus waist and therefore over a relatively small footprint area) with power levels often  $> 2 \text{ kW/cm}^2$  for short periods ( $\lesssim 20$  msec) and a small fraction of the NBI power during steady state equilibrium operation, with minimal evolved gas return to the hot plasma region.



### IV.3.3 Specific NBI Design Parameters for TASKA

The specific NBI design choices indicated in Table IV.3-1 for TASKA represent a compromise between high output performance and a reduced current density conservative design for reliable long lived operation. The prototype ion source design adopted for this study was based upon the Jülich (KFA) version of the Culham/JET ion source.<sup>(11)</sup> The Culham type ion source uses a multipole cusp wall arc chamber to achieve up to ~ 50% gas efficiency, a plasma source power requirement of  $\lesssim 2$  kW/A extracted beam, an atomic ion fraction (e.g.,  $D^+$ )  $\gtrsim 80\%$ , and a plasma ion temperature  $\lesssim 0.4$  eV for a minimum contribution to the ion beam divergence.<sup>(12)</sup> The prototype ion source design is also based upon use of a set of four water cooled electrodes, with shaped circular apertures for extraction. A focused ion beam can be achieved either by using spherically shaped electrodes, programmed aperture offset,<sup>(13)</sup> or subdividing the extraction area into a set of tilted subsections (as will be used with JET<sup>(14)</sup>). The resultant NBI design parameters for the TASKA beam lines are listed in Table IV.3-1. Note that the gas flow requirements are the total flow for the ion source and neutralizer cell to attain  $0.95 F_{\infty}^0$  conversion of the primary (full energy) ions to neutrals. The beam collimation dimensions listed at 8 m as an example are scaled to allow  $2\theta(\text{div.})$  clearance around both the height and width of the focused beam from the rectangular source extraction area indicated. The scaling for  $2\theta(\text{div.})$  beam clearance "width" for collimation has the form  $W = |W_0 - 2Z\theta_f| + 4Z\theta_D$  for small angles where  $W_0$  = the initial ion beam width,  $\theta_f = W_0/2L$  where  $L$  is the beam focal length, and  $\theta_D$  is the divergence angle in radians. A similar expression applies for the beam height, with appropriate substitution of the initial beam height  $H_0$ . The beam footprint including the beam edge out only to  $\theta_D$  has a minimum "radius" at the

beam focus  $Z = L$  of  $L\theta_D$  (or  $\sim$  elliptical axis  $L\theta_{Dx}$  and  $L\theta_{Dy}$  if from a bi-gaussian distribution with different  $\theta_{Dx}$  and  $\theta_{Dy}$  divergences along the beam x and y axis).

A prototype ion source tank section for the HE-NBI is shown in Fig. IV.3-4. Steady state high speed gas pumping is attained using an articulated "two position" folded wall system of getter pumping panels as described in Section IV.4. The ion sweep beam dump and the insertable beam calorimeter shown are of the water cooled (high speed) rotating design developed by Hemmerich and Kupschuss<sup>(11)</sup> at KFA Jülich. Detailed optical Doppler shift measurements of the line spectra from the neutral beam at the neutralizer exit are expected to be the main on-line beam diagnostic for measuring the beam current density distribution, divergence, and species composition.<sup>(15)</sup>

The power supplies for a typical 50 kV 50 A equivalent NBI (e.g., one of the ion source-neutralizer-ion sweep sets of the ME-NBI) are listed in Table IV.3-2. NBI operation will also require fast automated power supply control and an extensive interlock system. Diagnostic sensors will be required to assure proper cooling, proper beam aiming, presence of a suitable target plasma, and a suitable ion sweep field before the NBI can be turned on to prevent possible damage to some wall section (without adequate power handling capability). Further considerations on this are given in Section VIII.3.3.

The use of  $T_2$  gas, the high neutron fluences during operation, induced radioactivity in components, and the high fringe-field magnetic shielding problem with the TASKA coils energized combine to prohibit short term NBI maintenance. This study therefore envisions continued research and development on NBI components and a conservative design to result in NBI with mean time before failure  $> 1$  year. The ion source tank design shown in Fig. IV.3-4

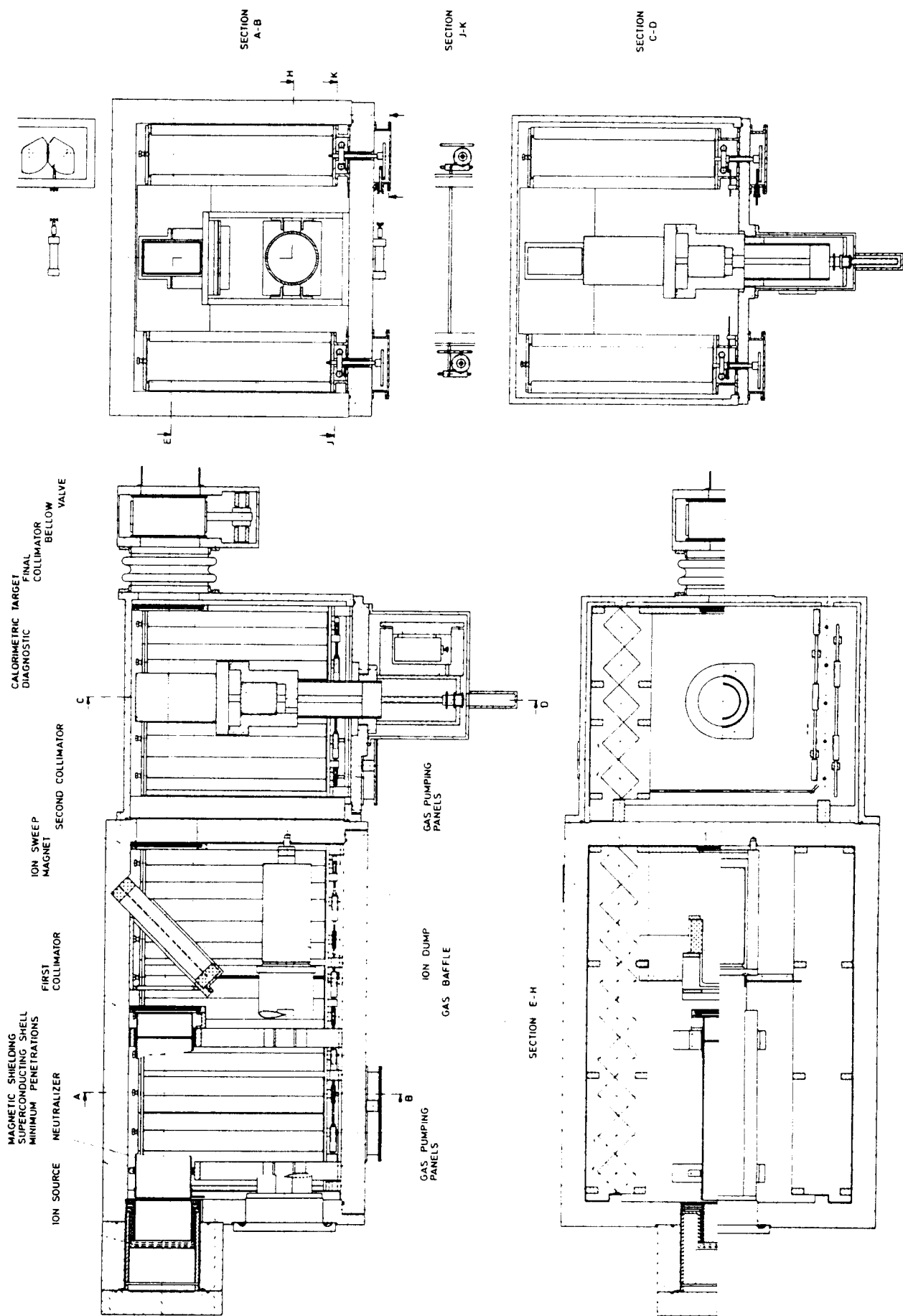


Fig. IV.3-4 High energy neutral beam injector

Table IV.3-2. Typical Ion Source Power Supply Requirements

(e.g., for Ion Source for 50 kV 50 A Equivalent D<sup>0</sup> NBI)

|                     |             |                            |
|---------------------|-------------|----------------------------|
| Filaments           | 15 V        | 2,000 A                    |
| Arc power supply    | 50 to 150 V | to 4,000 A                 |
| Accel. power supply | 50 kV       | to 110 A (with 20% margin) |
| Gradient grid       | ~ 40 kV     | ≤ 5 A                      |
| Suppressor grid     | -1 to -5 kV | ≤ 10 A                     |

includes planned access during major maintenance shutdown periods for remote maintenance to the ion source components from the rear of the tank and access to the pumping panels from the top of the tank. An excess current output capacity of 20% is included with the parameter listing of Table IV.3-1, to cover expected degradation of performance and losses from component failures over the operating period before required major maintenance. If possible this excess NBI capacity should be in the form of "spare" installed ion source, neutralizer, and ion sweep units, since equipment failures in an ion source beam line tend to disable the entire unit and not just fractionally degrade performance.

#### IV.3.4 Design Considerations for a Positive Ion Source (Options)

An ion source is generally considered to consist of two distinct units, (1) a plasma source of appropriate plasma density and area, and (2) a multi-apertured extraction system.

##### IV.3.4.1 The Plasma Source (Arc Chamber)

The plasma source for the present large area ion source designs is generally either of the  $B \approx 0$  type (developed at LBL)<sup>(16)</sup> or of the magnetic multipole wall ( $B(\text{central}) \approx 0$ ) type in use at Culham,<sup>(17)</sup> ORNL,<sup>(18)</sup> and under test at LBL,<sup>(19)</sup> or of the periplasmatron design by Fontenay-aux-Roses.<sup>(20)</sup> In any case, the primary need is for a source of uniform density ( $\delta n/n < 0.05$ ) plasma with adjustable density ( $n \lesssim 5 \times 10^{12}/\text{cm}^3$ ) to match the extraction current density  $j(\text{s.c.l.})$  over the needed area (up to 20 cm x 50 cm). The multipole cusp magnetic wall type of plasma source is quoted as having the highest gas efficiency, highest power efficiency, and highest primary ion fraction ( $D^+$ ,  $D_2^+$ ,  $D_3^+$  fractions of at least  $\sim 80\%$ ,  $15\%$ ,  $5\%$ ).<sup>(12)</sup>

#### IV.3.4.1.1 Plasma Source Cathode

Cathode electrode construction is now generally based upon use of a large number of pure tungsten hairpin filaments arranged as a window frame or fence around the outside edge of the extraction area of interest, with a typical filament lifetime of about 100 hr.<sup>(2)</sup> Future designs should consider larger area, lower temperature tungsten operation (to reduce tungsten evaporation and give longer lifetime) or use of a hollow thoriated tungsten or sintered LaO/Mo cathode for long lived good emission.<sup>(21)</sup> The usual cathode design requires a total emission electron current larger than the expected arc current. Cathode lifetime is a limiting maintenance problem, but one year lifetime appears a reasonable goal. Filament circuits are generally designed in parallel with one leg common (to avoid large electric field differences in the plasma) and with a stabilizing arc voltage dropping resistor connection to each hairpin circuit to help distribute the arc current. The orientation of the hairpin filament and input leads are generally arranged for a minimum resultant B field. Cathode temperatures are found to be arc current dependent and must be readjustable for maximum filament lifetime.

#### IV.3.4.1.2 Plasma Source Anode

The outer walls of the plasma chamber are generally arranged to function as a large area anode surface for electron collection. However, in the multipole field ion source the wall is built to include narrow strips of high field permanent magnets so the walls appear as a set of line cusps with strong local (short range) magnetic fields tending to reduce the electron mobility and increase their ionizing efficiency. This feature appears to improve gas efficiency, power efficiency and the atomic ion fraction.<sup>(12)</sup> Wall anode heating by the arc discharge electrons is concentrated along these strips; this

requires careful cooling design for steady state operation. Normally the chamber walls are of copper for good conductivity and quite thin walled at the line cusp magnets for maximum magnetic field utilization. This suggests use of special thin capillary cooled wall construction designed such that the walls do not experience the vacuum load and a thin protective covering layer of chrome or molybdenum to reduce metallic sputtering or evaporation.

#### IV.3.4.1.3 Ion Source Gas Feed

The back wall of the arc chamber is generally used as a gas manifold or gas distribution system. If thermionic dissociation of  $D_2$  appears useful in increasing the source output  $D^+$  fraction the ion source feed gas can be arranged to enter through a heated tungsten diffuser. The gas flow directly into the ion source should be the minimum required for arc operation (with extraction). The remainder of the gas required to produce a neutralizer thickness  $\int n \, dl \sim 10^{16}/\text{cm}^2$  should be fed directly into the neutralizer (colder gas to reduce the gas flow).

#### IV.3.4.1.4 Arc Associated Electrode Feedthrough Insulator

Ceramic to metal bonded vacuum tight feed-throughs are required for each cathode lead into the arc chamber (relatively high temperature environment service). Ceramic standoff of the first extraction electrode from the anode wall is also desirable, though pulsed ion sources in a low neutron fluence environment have used simple Kapton plastic sheets for needed insulation. The cathode back wall plate should be designed to be readily removable for cathode replacement (the item with the shortest expected lifetime). All ceramic seals and cooling circuits should include flexible (e.g., bellows) connections to avoid mechanical stress on ceramic insulators under varying temperature conditions.

#### IV.3.4.2 Extraction Electrode System

The multi-aperture ion extraction electrode system must serve the functions of defining the shape and location of the plasma meniscus for each aperture, forming the extraction field to achieve optimum ion beamlet extraction optics and acceleration, and orienting the total array of beamlets for the desired total beam focal length. The first "accel" or beam forming electrode serves as an interface between the plasma chamber and the first acceleration gap, and defines the location and area of each extraction meniscus surface. The auxiliary gradient or second electrode (in a four electrode system - not present in a three electrode system) allows optimum field gradient control near each plasma meniscus surface, somewhat independent of the main extraction voltage  $V_0$ . The next electrode is the negative voltage "decel" electron suppression grid; this is made relatively thick to reduce field penetration through each of its aperture openings. Finally, the last electrode of the extraction set is used to establish the ground plane for each beamlet of the neutralizer ion beam system. The extraction electrode system is best designed for a specific voltage and current density. Design optimization involves a parameter study of the ion optics of an individual beamlet to minimize its beam divergence. Several ion extraction system codes are available including "WOLF" at LBL,<sup>(22)</sup> a version of the "BATE" code at Culham,<sup>(12)</sup> and the "SNOW" at LASL.<sup>(23)</sup> The choices in extraction system include:

- Number of electrodes: three is simpler, four allows wider range of current adjustment and likely is best for good beam optics for  $V_0 \gtrsim 50$  kV.
- Multi-aperture type: circular apertures appear to have slightly lower average divergence, lower acceleration losses and heat load, but also lower



transparency; slot electrodes have the best 1-D divergence, are easier to fabricate and align, but appear more susceptible to plasma variations causing electrical (accel) breakdown.

- Shaped aperture rims: a semi-Pierce shaped<sup>(24)</sup> edge is required for the apertures in the first beam forming electrode to obtain the best optics. Simpler aperture forms are often used for higher transparency or for easier machining, cooling and alignment. The design (appropriate to either circular apertures or the cylindrical symmetry of slots) should be chosen from code optimization studies.
- Electrode materials: choose for needed cooling and structural stability. Water cooled copper or molybdenum is the most common, but one could consider composites such as stainless steel covered with electrodeposited copper with milled channels and a copper covering (with a thin chrome or molybdenum surface film to reduce sputtering).

#### IV.3.4.3 Electrical Configuration

The standard power supply arrangement for powering a positive ion source is to run the plasma source at the positive high voltage  $V_0$ , the gradient grid at a slightly reduced positive high voltage, the electron suppressor grid a few kV negative and the final grid and neutralizer at ground potential. This configuration involves the minimum volume at high voltage potential, though power supplies for the ion source arc chamber are then at high voltage, and are quite susceptible to problems from high voltage sparking. Direct energy recovery of the remnant ion sweep portion would require use of another neutralizer exit electron suppression grid or transverse magnetic field at the entrance to the ion collector region to prevent unwanted electron acceleration to the positive ion collector plate.

An attractive alternate configuration would place the ion source at or near ground potential, the gradient grid at a slight negative voltage, the electron suppressor at the maximum negative high voltage and the neutralizer cell at negative high voltage,  $-V_0$ . This configuration also requires a second electron suppressor grid or transverse magnetic field at the exit of the neutralizer to prevent unwanted electron acceleration from the plasma produced by ionization occurring within the neutralizer gas. However, the use of a sweep magnetic field and direct energy recovery of the swept ion flux fulfill this requirement.

#### IV.3.5 Negative Ion Source Based NBI

Negative ion beams appear essential for efficient production of neutral beams with an energy  $\geq 100$  keV/nucleon, because of the decrease of  $F_{\infty}^0$  with energy. In the case of the plug NBI, the equilibrium fraction of  $H^+$  conversion to  $H^0$  at 250 keV has fallen so  $F_{\infty}^0 \sim 0.01$ , while conversion of  $H^-$  to  $H^0$  remains a high 0.8 (for the plasma stripping cell) and is almost independent of energy. The factor limiting wide use of negative ions for all high energy NBI systems is the present lack of good, high current, steady state negative ion sources. However, recent progress appears promising at least for the following two types of negative ion sources: (1) using double charge exchange of a low energy ion beam in a suitable alkali metal vapor, or (2) using direct negative ion extraction from a volume or surface negative ion production source.

- (1) The characteristics of double charge exchange sources are largely determined by requirements for maximum positive ion to negative ion (two stage) conversion; cesium or strontium have conversion efficiencies of 0.25 to 0.5 for  $H^-$  energies  $\lesssim 1$  kV. Typical system requirements include:

- A low energy, high current source of positive ions; limited by  $j(\text{s.c.l.}) \propto v^{3/2}/d^2$ , with the minimum gap  $d$  limited by mechanical tolerances, and the  $\theta(\text{div.}) \propto 1/\sqrt{V}$  is poor. Use of a source of molecular ions at higher energies may be desirable. A suitable source of charged pellets, with acceleration to the required velocity range, is also a possible alternative.<sup>(25)</sup>
- A double charge exchange cell filled with a suitable alkali metal vapor such as cesium or strontium. This must be designed for the specific application in order to minimize high  $Z$  impurity flow into the ion source or down the beam line; this will likely require a special vapor jet curtain design.
- A large area (or large array of small areas) accelerator section is required to accelerate the negative ions to the final desired energy. The use of grids is a problem for high beam current densities, and optics of large area short gap accelerators tend to be difficult without grids. Pumping must be good enough to achieve a low enough vacuum pressure to prevent excessive beam drain and negative ion loss, but not so low as to prevent production of positive ions for needed space charge neutralization during beam transit from the double charge exchange cell to the accelerator section. Use of time averaged current from an electrostatically focused RF acceleration system (array)<sup>(26)</sup> has some attractive features.
- A stripping cell is required to remove the loosely bound electron. Gas cells have  $\text{H}^-$  to  $\text{H}^0$  conversion efficiencies of  $\sim 0.65$ , plasma filled cells an efficiency of  $\sim 0.83$ , and photostripping with an intense laser beam can have an even higher efficiency.

- The beam line also requires a remnant ion sweep, appropriate gas pumping (gas efficiency is relatively poor as only part of the low energy positive beam can be used), and other hardware standard to an NBI.
- (2) The characteristics of a direct extraction negative ion source depend upon details of the volume or surface source. Typical system requirements include:
- The Ehlers<sup>(27)</sup> surface source uses  $H^+$  arc plasma bombardment of a shaped cesium covered tungsten negatively biased surface to yield a suitable flux of "backscattered"  $H^-$  ions available for direct extraction. A major problem of this approach is separation of the plasma electrons from the  $H^-$  before extraction and acceleration to a significant energy.
  - A version of the Bacal<sup>(28)</sup> type of volume source is under study by Holmes.<sup>(12)</sup> A standard multipole arc chamber is operated in a lower arc current density range favoring molecular ion and negative ion production. The major problem is again the means for separating out the electron portion of the current before extraction or acceleration to a significant energy. This source requires no cesium, but the current density is expected to be limited to  $\leq 0.05 \text{ A/cm}^2$ .
  - The two above approaches will likely be arranged for two stage acceleration - first extraction of a suitable  $H^-$  rich beam and then the major accelerator stage (only negative ions) to a full energy  $eV_0$ . Acceleration requirements are similar to (1) above.
  - A stripping cell is required as in (1) above.

- The beam line also requires a remnant ion sweep, appropriate pumping, and other hardware appropriate for a standard NBI.

Recent results with both of these negative ion approaches appear sufficiently promising to incorporate this approach in the plug - NBI. Pulsed yields of over one ampere have been achieved from both the surface source and double charge exchange sources. The general characteristics of the volume sources without cesium appear very desirable if the output current can be increased.

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## IV.4 Vacuum System

### IV.4.1 Introduction

The vacuum pumping system for TASKA has been designed to meet the following requirements:

- Pumping hydrogen, hydrogen deuteride (HD), deuterium, tritium, hydrogen tritride (HT), deuterium tritride (DT), and helium from the vacuum vessel to a total pressure of  $10^{-5}$  torr between experimental shots.
- Pumping hydrogen, hydrogen deuteride, deuterium, tritium, hydrogen tritride, deuterium tritride, and helium from the vacuum vessel and regeneration of cryopanel.
- Largely oil-free and contamination-free pumping system.
- Backfilling the vacuum vessel with He to 20 torr.
- The pumping system should be attached to the vacuum vessel to form a compact modular unit. It must not occupy too much space for pumping leads and pipe connections.
- Regeneration of the volume getter panels in the neutral beam injectors.

The consideration of these requirements led to the design of two pumping units, one of which is shown schematically in Fig. IV.4-1. Cryopumps, combined with turbomolecular pumps and backed by roughing pumps (at some distance from the vacuum vessel), offer an optimum solution to evacuate the vacuum vessel and the neutral injection beam dumps.

### IV.4.2 Vacuum Pumping of the Discharge Chamber

The vacuum system for the discharge chamber must provide for evacuation from atmospheric pressure and maintaining the chamber at  $10^{-5}$  mbar at the operating temperature of the wall. In order to achieve this low pressure, we anticipate performing a series of operations to clean and degas the chamber



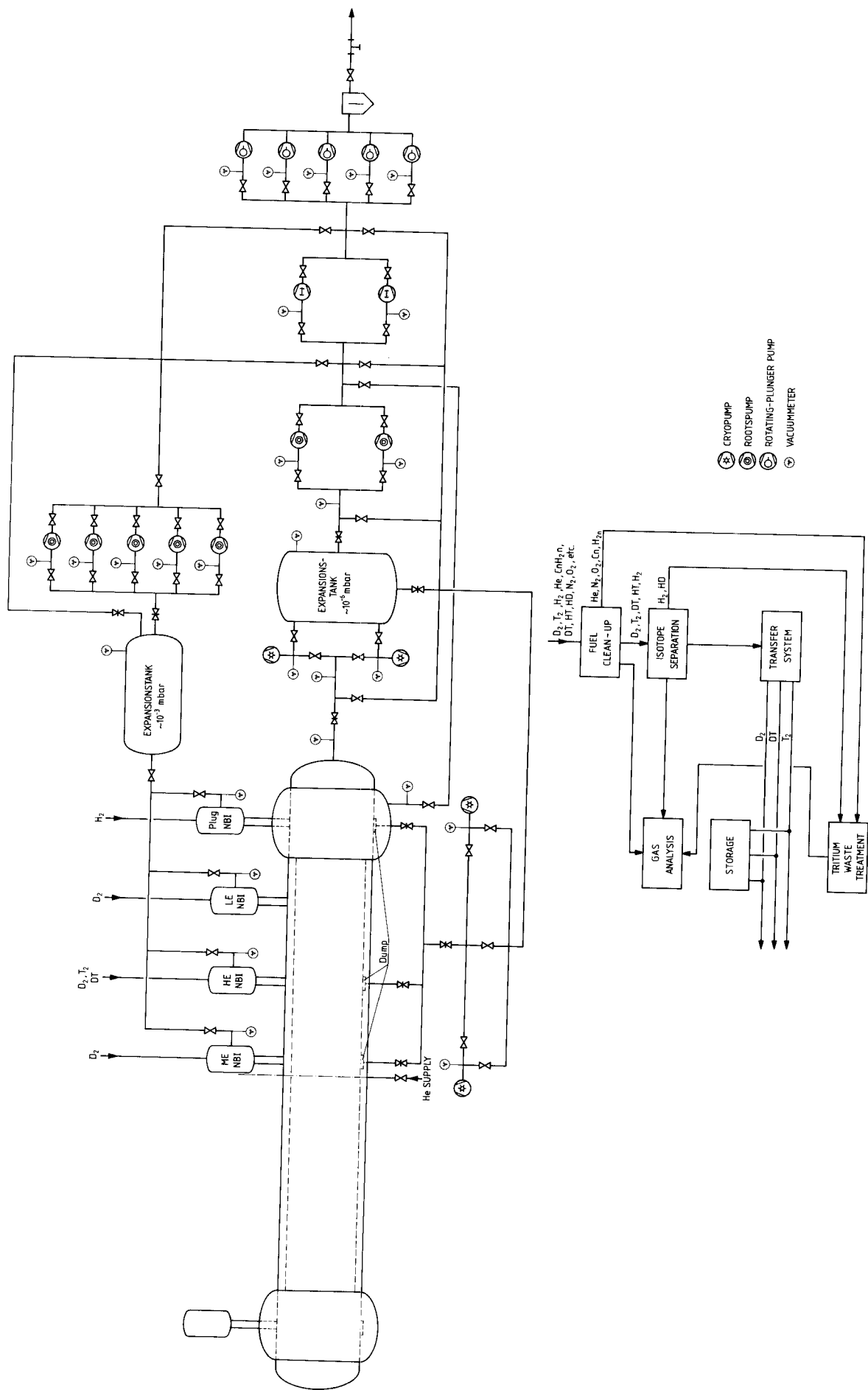


Fig. IV.4-1 Vacuum system.

wall. During the construction of the chamber, the components must be electropolished and cleaned ultrasonically. Lowering the specific outgassing rate from the chamber wall to  $10^{-11}$  torr- $\ell$ /cm<sup>2</sup>-s at room temperature and to  $10^{-9}$  torr- $\ell$ /cm<sup>2</sup>-s at the operating temperature<sup>(1)</sup> will be accomplished by conditioning the chamber wall by means of a discharge through inert gases. To obtain the design pressure in the degassed chamber, an effective pumping speed of  $2 \times 10^3$   $\ell$ /s will be required.

For evacuation in the range between  $10^{-3}$  and  $10^{-6}$  torr under conditions which exclude a large inventory of tritium, cryopumps are used. The cryopumps are backed by turbomolecular pumps needed for the pumping of helium and for cleaning the cryopumps. As a primary evacuating system, a scheme consisting of two pumping stations, both at the ends of the vacuum chamber, is used. These stations will have cryopumps each with a pumping speed of  $12 \times 10^6$   $\ell$ /s for H<sub>2</sub>. This will provide an oil free vacuum which is needed in this pressure range.

The pumping lines will have gate valves which close on the order of one second and seal hermetically in 10 seconds.

An inert gas (He) is admitted into the discharge chamber at the end of a burn cycle to reduce the possibility of localized overheating of the chamber wall.

#### IV.4.2.1 Calculations

Gas load with consideration of:

- leaks
- permeation
- outgassing
- vapor pressure of the materials

- particle flow from the central cell.

The total gas load pumped from the end cell is about 100 torr-ℓ/sec, of which 55 torr-ℓ/sec is the gas load from the direct convertor.

Pressure in the vacuum vessel  $p < 10^{-5}$  torr.

$$S = \frac{100 \text{ torr } \ell}{s \times 10^{-5} \text{ torr}} \sim 1.0 \times 10^7 \frac{\ell}{s}$$

Pumping speed

$$S_{\text{eff}} \sim 15 \times 10^6 \frac{\ell}{s} .$$

Pumping speed of the cryopanel  $S \sim 1.2 \times 10^5 \ell/s \cdot m^2$ .

Cryo-area  $\sim 125 m^2$ .

Maximum cycle time:  $t_c = \frac{\rho A_K R T_n}{S M \times p} \times d$

$$R : 8314 \text{ J/K kmol}$$

$$\rho : \sim 80 \text{ kg/m}^3$$

$$S/A_K : 120 m^3/s \cdot m^2$$

$$p : 10^{-3}/s \cdot m^2$$

$$T_n : 293 \text{ K}$$

$d$  : 0.025 mm thickness of the condensate

$$t_c \sim 25 \text{ h} .$$

Regeneration system for the cryopumps.

Maximum gas load from the cryopumps 2 h regeneration time  $\sim 400$  torr- $\ell$ /s.

In order to keep the tritium inventory low the cycle time should also be 2 h.

Maximum pressure in the expansion tank by using a turbopump with a pumping speed of 50000  $\ell$ /s.

$$p \sim \frac{400 \text{ torr-}\ell/\text{s}}{s \ 50000 \ \ell} = 8 \times 10^{-3} \text{ torr} .$$

#### IV.4.3 Evacuation System for the Dumps

The purpose of the evacuation system for the dumps is to avoid the diffusion of the scattered and sputtered particles from the dumps. The pumping load is determined by the particle flux from the neutral beam injectors. In the outline design, it is proposed to use 4 cryopumps with a pumping speed of  $1.0 \times 10^6$   $\ell$ /s (for  $H_2$ ). Each cryopump needs a cryopanel area of  $\sim 95 \text{ m}^2$ .

##### IV.4.3.1 Calculations

Gas load  $Q \sim 6$  torr- $\ell$ /s (all dumps in the right or left part), pressure in the dumps  $p \sim 10^{-5}$  torr.

$$S \sim \frac{6 \text{ torr-}\ell}{s \times 10^{-5} \text{ torr}} \sim 6 \times 10^5 \frac{\ell}{s}$$

Pumping speed

$$S_{\text{eff}} \sim 1.0 \times 10^6 \frac{\ell}{s} .$$

Pumping speed of cryopanel<sup>s</sup>(1)

$$S \sim 120000 \frac{\ell}{s \cdot \text{m}^2} .$$

Cryo-area  $\sim 9 \text{ m}^2$ .

The cycle time should be 2 hrs.

#### IV.4.4 Evacuation System for the Neutral Beam Injectors

One of the main characteristics for the design of the neutral beam injectors is the need of DC-operation (see Section IV.3). This condition can be realized by large volume getter panels inside the boxes of the neutral beam injectors. Using this pumping system, one has a weak dependence of the pumping speed on the gas load.<sup>(2)</sup>

Since  $H_2$  isotopes form a solid solution with the getter material the amount of gas which can be adsorbed depends on the temperature of the getter material and on the acceptable equilibrium pressure in the gas phase established by  $H_2$  isotopes at a given concentration of the getter material.

Regeneration of the volume getter panels, saturated with hydrogen isotopes, may be obtained by raising its temperature and by removing the evolved hydrogen isotopes by means of turbomolecular pumps.

##### IV.4.4.1 Calculations

###### IV.4.4.1.1 He Neutral Beam Injection System

Within the practical geometry (see Section IV.3) of the beamline we have a gas flow of 7.1 torr-ℓ/s into the first stage of the box, ~ 2 torr-ℓ/s in the second stage of the box.

Each of the rotating flat getter panels is about 875 x 1200 mm and has a  $D_2$  pumping speed of 18000 ℓ/s (at 400-200°C). The reversible sorbtion capacity at constant pumping speed (at 200°C and  $10^{-5}$  torr) is 21000 torr-ℓ for each panel.

The power required for activation or regeneration of each getter panel at 700°C is about 7500 watt (760 A). The duration of the treatment is 45 minutes. Operation at 400°C and 200°C would require about 24% and 4.5% of the activation power.<sup>(3)</sup>

First stage: 12 panels

Gas load  $Q \sim 7.1 \text{ torr-}\ell/\text{s}$

Attainable pressure

$$p \sim \frac{7.1 \text{ torr-}\ell/\text{s}}{\text{s} \cdot 120 \times 18000 \ell} \sim 3.2 \times 10^{-5} \text{ torr}$$

Maximum cycle time

$$t_c = \frac{21000 \text{ torr-}\ell \cdot 12 \text{ s} \cdot \text{h}}{7.1 \text{ torr-}\ell \cdot 3600 \text{ s}} \sim 9.9 \text{ h}$$

Sweep: 6 panels

Gas load  $Q \sim 2 \text{ torr-}\ell/\text{s}$

Attainable pressure  $p \sim 1.9 \times 10^{-5} \text{ torr}$

Maximum cycle time  $t_c \sim 18 \text{ h}$

Second stage: 10 panels

Gas load  $Q \sim 2 \text{ torr-}\ell/\text{s}$

Attainable pressure  $p \sim 1.2 \times 10^{-5} \text{ torr}$

Maximum cycle time  $t_c \sim 28 \text{ h}$ .

#### IV.4.4.1.2 ME Neutral Beam Injection Systems

The gas flow in the beam line is (see Section IV.3) 27 torr- $\ell$ /s into the first stage of the box, 5 torr- $\ell$ /s into the sweep, and 3 torr- $\ell$ /s into the second stage.

The panel geometry is 875 x 2300 mm. The pumping speed for the  $D_2$ ,  $T_2$  mixture is about 24000  $\ell$ /s and the reversible sorbtion capacity is 28000 torr- $\ell$ .

First stage: 14 panels

Gas load  $Q \sim 27 \text{ torr-l/s}$

Attainable pressure

$$p \sim \frac{27 \text{ torr-l s}}{s \cdot 14 \times 24000 \text{ l}} \sim 8 \times 10^{-5} \text{ torr}$$

Maximum cycle time

$$t_c \sim \frac{28000 \text{ torr-l} \cdot 14 \text{ s h}}{27 \text{ torr-l} \cdot 3600 \text{ s}} \sim 4 \text{ h}$$

Sweep: 10 panels

Gas load  $Q \sim 5 \text{ torr-l/s}$

Attainable pressure  $p \sim 2.1 \times 10^{-5} \text{ torr}$

Maximum cycle time  $t_c \sim 15.6 \text{ h}$

Second stage: 14 panels

Gas load  $Q \sim 3 \text{ torr-l/s}$

Attainable pressure  $p \sim 8 \times 10^{-6} \text{ torr}$

Maximum cycle time  $t_c \sim 37 \text{ h}$ .

Since the ME Neutral Beam injects deuterium and tritium, the cycle time should be minimized in order to keep the tritium inventory low. The minimum cycle time is equal to the regeneration time which is 45 minutes.

#### IV.4.4.1.3 Regeneration System for the Volume Getter Panels

When the pumping speed for the hydrogen isotopes has decreased to an unacceptable limit due to the increased isotope concentration in the getter

material, a regeneration process is carried out. More frequent regenerations are performed when pumping tritium. An indefinite number of pumping cycles are possible.

For regeneration of the volume getter panels within the neutral beam boxes we assumed an average gas load of  $Q \sim 200 \text{ torr-}\ell/\text{s}$ . The total pumping speed for the expansion tank must be in the order of  $2 \times 10^5 \ell/\text{s}$  and we will maintain a pressure of  $10^{-3} \text{ torr}$ :

$$p \sim \frac{200 \text{ torr } \ell/\text{s}}{2 \times 10^5 \ell/\text{s}} \sim 10^{-3} \text{ torr} .$$

The pumping system consists of five turbomolecular pumps; four in action, one for reserve.

#### IV.4.5 Roughing Pump System

For evacuation in the range  $760\text{--}10^{-1} \text{ torr}$  a combination between roots pumps and rotating-plunger pumps are used.

##### IV.4.4.5.1 Calculations

Roots pumps

Maximum gas load  $Q \sim 400 \text{ torr-}\ell/\text{s}$

Pressure after the turbomolecular pumps  $< 0.1 \text{ torr}$

Pumping speed of the roots pumps

$$S \sim \frac{400 \text{ torr } \ell/\text{s}}{0.1 \text{ torr}} \sim 4000 \ell/\text{s}$$

$$S_{\text{eff}} \sim 2000 \text{ m}^3/\text{h} .$$



We need two roots pumps in the evacuating system; one in action, one for reserve.

Rotating plunger pumps

Pressure after the roots pump < 1 torr

Pumping speed of the rotating plunger pumps

$$S \sim \frac{400 \text{ torr } \ell}{\text{s } 1 \text{ torr}} \sim 400 \ell/\text{s}$$

$$S_{\text{eff}} \sim 2000 \text{ m}^3/\text{h} .$$

We need five rotating plunger pumps; four in action, one for reserve.

#### IV.4.6 Control of Vacuum Parameters

For measuring the neutral gas pressure in the chamber during a burn cycle, a pulsed noise free ionization gauge will be used, which will make pressure measurements of the neutral gas in the vicinity of the plasma in the range  $5 \times 10^{-3}$  torr -  $1 \times 10^{-9}$  torr.

Controlling the background gases during a burn cycle will be done by a mass spectrometer capable of operating in a radiation environment.

The pressure and partial pressure of the background gases in the range  $1\text{-}10^{-8}$  torr in the discharge chamber, in the neutral beam boxes, in the pumps, vacuum lines and other elements of the vacuum system will be measured with commercial vacuum gauges and mass spectrometers with analog outputs.

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## IV.5 Tritium Fuel Cycle

### IV.5.1 Overview

This section describes the hydrogen isotope pathways in the fueling of TASKA and discusses the tritium handling, containment and inventory associated with the fuel cycle. (The tritium inventory and extraction from the breeding blanket and the fuel purification systems are described in Section VI.5.)

Three specific areas associated with the tritium fuel cycle are discussed:

(1) In TASKA the tritium fueling is accomplished by injection of a mixed deuterium-tritium neutral beam. The inefficiency of the neutral beam injection (NBI) system requires large amounts of fuel to be pumped, purified and recycled. (2) The use of Zr-Al getter panels to pump uninjected tritium from the neutral beam lines. (3) The requirement of tritium barriers in the beam, ion and thermal dumps to prevent tritium diffusion into the water coolant cycle. The tritium fueling and exhaust flow is summarized in Fig. IV.5-1 and the tritium inventories associated with the fuel cycle are given in Table IV.5-1.

#### IV.5.1.1 Neutral Beam Fueling

In TASKA the primary fueling beam injects 49.7 MW of a DT mixture (44% D and 56% T) at an energy of 50 keV. There are 6 total DT NBI guns; three are located in the barrier region at each end of the central cell. Additional deuterium fueling (and heating) is provided by two low energy (2 keV) and two high energy (76 keV) deuterium neutral beams that deliver 0.2 MW and 6.6 MW respectively into the barrier region. In the plug at both ends of the machine, a 250 keV beam injects 5.4 MW of hydrogen. The NBI parameters are given in Tables IV.5-2 and IV.5-3.

Fig. IV.5-1

# TASKA FUEL AND EXHAUST CYCLE

FLOW RATES IN g/d

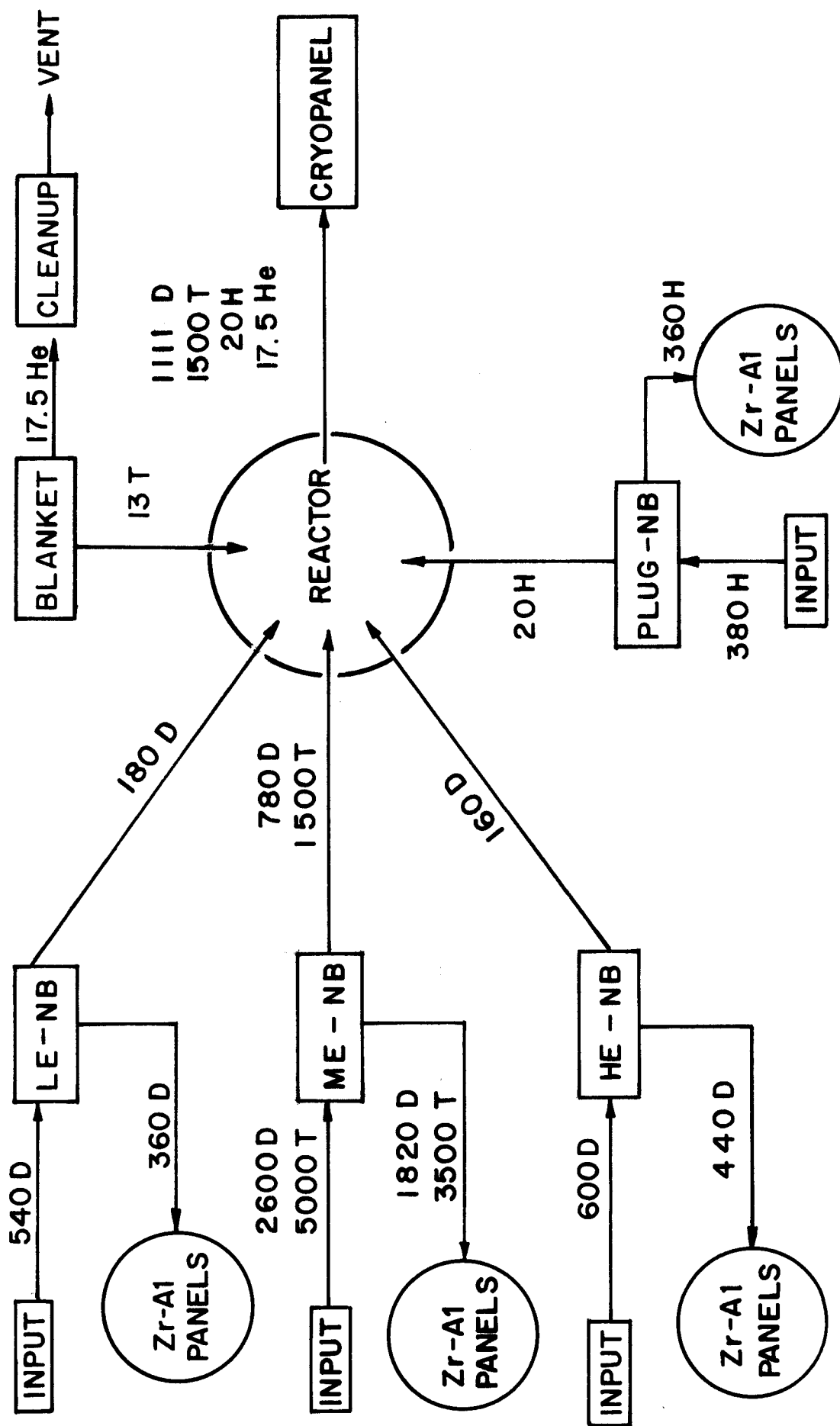


Fig. IV.5-1

Table IV.5-1. Tritium Inventory in TASKA Fuel and Exhaust Cycle

|                                   |      | <u>Subtotals (g)</u> |
|-----------------------------------|------|----------------------|
| NBI Getter Panels (g):            |      | 259                  |
| LE-NBI                            | 2.6  |                      |
| ME-NBI                            | 256  |                      |
| HE-NBI                            | 0.5  |                      |
| PLUG-NBI                          | 0.09 |                      |
| Cryopanel (g):                    |      | 125                  |
| Beam Dump                         | 6    |                      |
| End Cell                          | 119  |                      |
| Dump Areas (g):                   |      | 0.52                 |
| DT-Beam Dump                      | 0.50 |                      |
| DT-Ion Dump                       | ?    |                      |
| Thermal Dumps                     | 0.02 |                      |
| Fueling and Exhaust Subtotal (g): |      | 385                  |
| Storage (g):                      |      | 5,000                |

Table IV.5-2. Injection Parameters for TASKA Neutral Beams

| <u>Beam<br/>Designation</u> | <u>Energy<br/>(keV)</u> | <u>Power<br/>(MW)</u> | <u>Species</u> | <u>Number<br/>of NBI</u> | <u>Location</u> |
|-----------------------------|-------------------------|-----------------------|----------------|--------------------------|-----------------|
| LE-NB                       | 2                       | 0.2                   | D              | 2                        | Barrier         |
| ME-NB                       | 50                      | 49.7                  | .44 D<br>.56 T | 6                        | Barrier         |
| HE-NB                       | 76                      | 6.6                   | D              | 2                        | Barrier         |
| PLUG-NB                     | 250                     | 5.4                   | H              | 2                        | Plug            |

Table IV.5-3. Injection Rates for TASKA Neutral Beams

| <u>Beam<br/>Designation</u> | <u>Species</u> | <u>Particles/sec<br/>per NBI</u> | <u>g/d<br/>Total</u> |
|-----------------------------|----------------|----------------------------------|----------------------|
| LE-NB                       | D              | $3.1 \times 10^{20}$             | 180                  |
| ME-NB                       | D              | $4.5 \times 10^{20}$             | 780                  |
|                             | T              | $5.8 \times 10^{20}$             | 1,500                |
| HE-NB                       | D              | $2.7 \times 10^{20}$             | 160                  |
| PLUG                        | H              | $6.7 \times 10^{19}$             | 20                   |

A disadvantage of using neutral beams to fuel fusion devices is the fact that NBI is an inefficient method; the amount of fuel circulated in the beam lines is usually much greater than the amount injected into the reactor chamber.<sup>(1)</sup> The gas injection efficiency ( $\epsilon$ ) for a neutral beam injector can be defined as:

$$\epsilon = \frac{\text{gas injected into the reactor chamber}^*/\text{NB}}{\text{total gas input/NB}} .$$

[The total gas flow into a beam line is approximated as

$$\tau = 15 \times \frac{I^0(A)}{50(A)} \times \frac{.8}{f_{00}^0}$$

where  $\tau$  is the gas flow ( $\frac{\text{torr} \cdot \ell}{\text{sec}}$ ) for a 150 cm neutralizer of efficiency  $f_{00}^0$  injecting  $I^0$  amps<sup>(2)</sup>.] The gas injection efficiencies are approximately 30% (Table IV.5-4) for the D and DT beams. The plug-NB is of much higher energy (requiring negative ion technology) and has a gas efficiency of about 5%. The overall mass balance for the NBI system is given in Table IV.5-5.

For tritium NBI, low gas efficiencies result in large quantities of tritium that must be handled by the NB vacuum system, which typically uses cryopanel or getter panels for pumping. The inventory in the pumping panels as a function of the gas injection efficiency (based on 1500 g T/d injected) is shown in Fig. IV.5-2 for a 0.75 hr and a 2 hr pump on-line time. Low injection efficiencies can result in intolerably high inventories. In TASKA tritium is injected at a rate of 1500 g/d. The gas efficiency is about 30%

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\* includes gas not trapped by plasma.

Table IV.5-4. Approximate Gas Efficiencies for TASKA Neutral Beams

| <u>Beam Designation</u> | <u>Gas Injected/NBI*<br/>(torr·ℓ/sec)</u> | <u>Gas Efficiency (%)</u> | <u>Total Gas Input/NBI<br/>(torr·ℓ/sec)</u> |
|-------------------------|-------------------------------------------|---------------------------|---------------------------------------------|
| LE-NB                   | 4.4 (50 A)                                | 33                        | 13                                          |
| ME-NB                   | 15. (165 A)                               | 30                        | 50                                          |
| HE-NB                   | 3.9 (44 A)                                | 26                        | 15                                          |
| PLUG-NB                 | 0.98 (11 A)                               | 5                         | 20                                          |

Table IV.5-5. Mass Balance for Neutral Beams

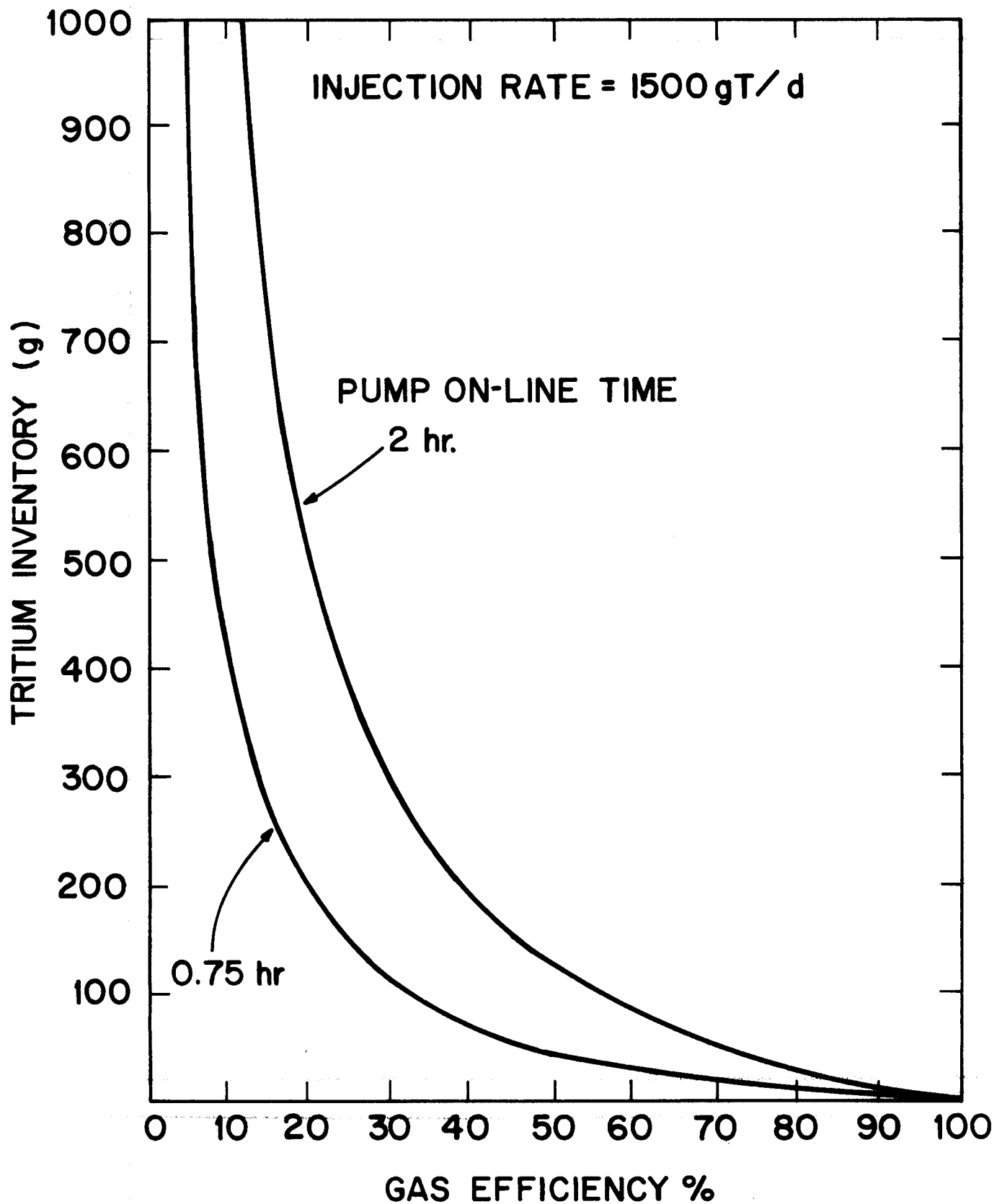
| <u>Species</u> | <u>Source</u> | <u>Total<br/>(g/d)</u> | <u>Recycled<br/>(g/d)</u> | <u>Injected<br/>(g/d)</u> |
|----------------|---------------|------------------------|---------------------------|---------------------------|
| T              | ME-NB         | 5,000                  | 3,500                     | 1,500                     |
| D              | LE-NB         | 540                    | 360                       | 180                       |
|                | ME-NB         | 2,600                  | 1,820                     | 780                       |
|                | <u>HE-NB</u>  | <u>600</u>             | <u>440</u>                | <u>160</u>                |
|                | TOTAL D       | 3,740                  | 2,620                     | 1,120                     |
| H              | PLUG          | 380                    | 360                       | 20                        |

\*Based on 11.28 amps atomic = 1 torr·ℓ/sec molecular gas flow.



Fig. IV.5-2

# TRITIUM INVENTORY IN NEUTRAL BEAM PUMPING SYSTEM



resulting in 5000 g T/d being supplied to the injection system and 3500 g T/d must be pumped by the NB vacuum system and recycled. The tritium inventory in the NB vacuum system (45 min on-line time) is 259 g, a large component of the total active tritium inventory.

#### IV.5.1.2 Tritium Inventory in Zr-Al Getter Panels

The NB gases are pumped by Zr-Al getter panels<sup>(3)</sup> with turbomolecular backing pumps. The getter material<sup>(4)</sup> is capable of pumping hydrogen and its isotopes as well as other active gases, such as O<sub>2</sub>, CO and N<sub>2</sub>. The hydrogen isotopes are absorbed reversibly while the active gases react chemically to form stable products and are permanently fixed into the getter.<sup>(5)</sup> The recommended operation temperature is 200-400°C when impurities are present, whereas pure hydrogen ( $\lesssim$  10 ppm impurities) can be pumped even at room temperature.<sup>(6)</sup> Inert gases are not pumped by the getters. This may be important where tritium is concerned, because of the unavoidable presence of the <sup>3</sup>He decay product. Compared to hydrogen, water and methane are pumped slowly at the higher temperature ( $\sim$  400°C) by a mechanism which involves dissociations or cracking at the surface followed by absorption of the subsequent components.<sup>(7)</sup> The use of Zr-Al alloys as getters has been proposed to control plasma density and impurities in tokamak chambers such as TFTR<sup>(8)</sup> and divertor channels.<sup>(9)</sup>

For NB pumping, both cryopanel and getter panels can provide the necessary vacuum requirements. Getter panels are thought to be more suitable for tritium pumping in NB lines than cryopanel for the following reasons:

- (1) The thermal loading from condensation and gas conduction is a more severe problem for cryopanel (4°K) than getter panels ( $\sim$  673°K).<sup>(10)</sup>
- (2) Similarly, the tritium beta decay heat is more critical in the case of cryopanel.<sup>(11)</sup>
- (3) Operating costs are higher ( $\sim$  40X) for cryopanel due mainly to liquid

helium consumption.<sup>(12)</sup> (4) The getter surface holds gases chemically at room temperature and thus, an accident causing large releases of gas is less likely.<sup>(11)</sup> Also, the manufacturer of Zr-Al getters states that the danger of fire in the pumping system caused by an air leak is only possible at or above 500°C.<sup>(13)</sup> The advantages of Zr-Al alloys over Ti as a getter material has been reviewed elsewhere.<sup>(8)</sup>

There is some actual experience in the pumping of tritium with Zr-Al getters. Research at Lawrence Livermore National Laboratories (LLNL) was conducted to evaluate the pumping speed and holdup capabilities of commercially available getter pumps for tritium handling in TFTR.<sup>(13,14)</sup> Results of the study indicated that the ratio of pumping speeds of deuterium to tritium follows the square root mass ratio. The pumps were shown to be satisfactory getters for tritium at temperatures between 250-400°C and on a short term basis tritium decay did not appear to adversely affect the pumps.<sup>(13)</sup> Additional research was performed to measure the amount and chemical form of tritium released when a commercial getter cartridge was loaded with tritium and then exposed to moist air at ambient temperatures. Pumps loaded with 750 Ci of tritium released 0.04 Ci after 66 min and a total of 0.1 Ci (~ 0.01%) was released after 1200 hrs. The majority of gas is released upon initial exposure (> 72% in 4 min) and the rate decreases rapidly with time. Of the total released gas 80% was in the form of tritiated water.<sup>(14)</sup> Recommended areas for further research include investigation of the effects of radiolysis and <sup>3</sup>He production on the holdup and pumping capabilities of the getters over a time scale representative of what would be experienced in future fusion reactors and evaluation of tritium holdup under normal and off-normal conditions after a number of recycles have been performed.<sup>(15)</sup>

An important parameter in determining the tritium inventory in the getter material is the amount of time the panels are on-line pumping (Fig. IV.5-3). In TASKA, the getter panels are arranged so that one surface is absorbing gas, while the back surface is being degassed. With this arrangement the panels can be periodically rotated to control the tritium inventory. The minimum pump on-line time is equal to the time required to regenerate the pump surface. The pump regeneration time can be estimated from Eq. (IV.5-1):<sup>(16)</sup>

$$t = \frac{M}{F} \left( \frac{1}{q_F} - \frac{1}{q_0} \right) 10^{-\left(4.4 - \frac{7000}{T}\right)} \quad (\text{IV.5-1})$$

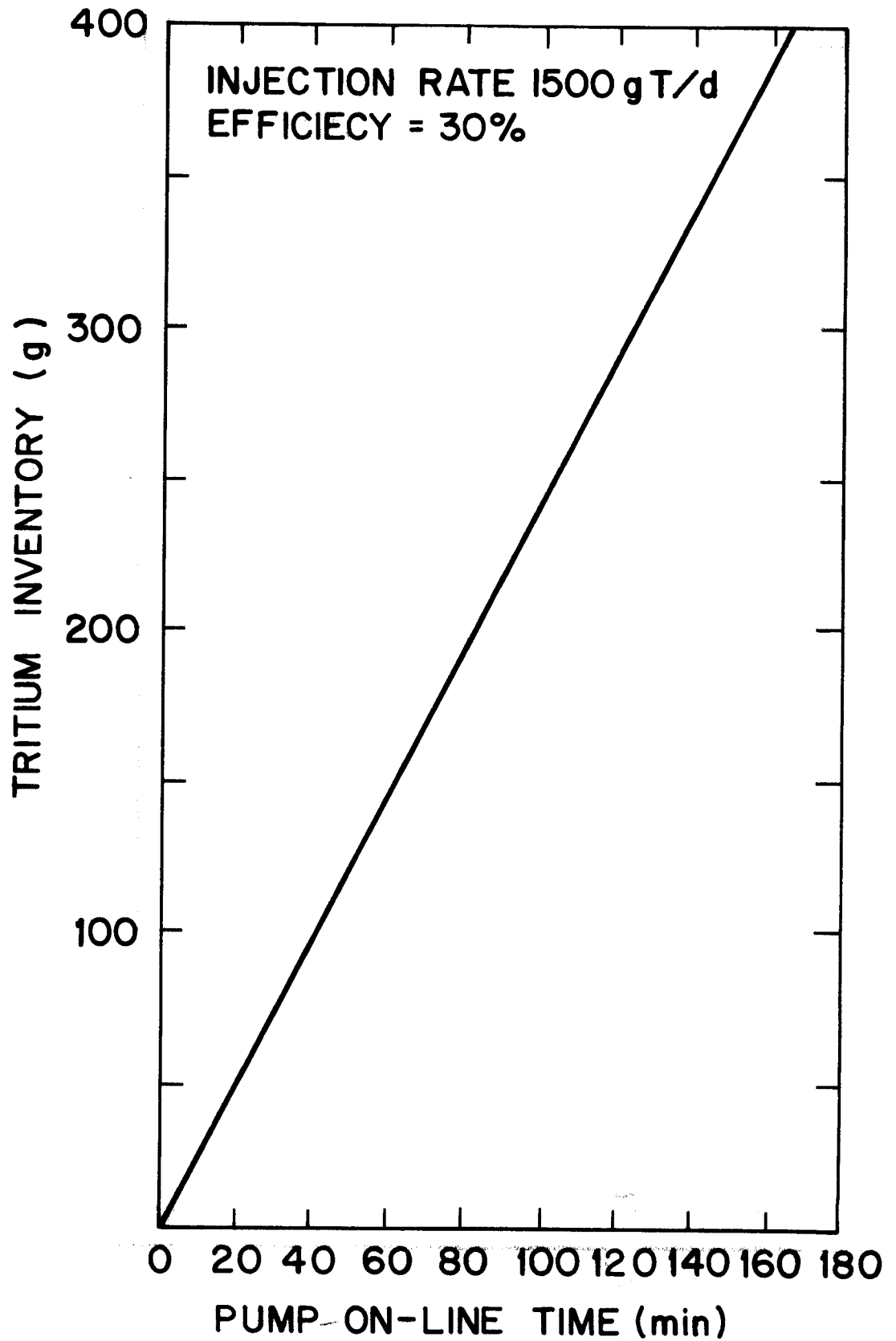
where:  $t$  = regeneration time (sec)  
 $M$  = mass of Zr-Al getter (g)  
 $F$  = back-pumping speed (ℓ/s)  
 $q_F, q_0$  = final and initial concentration (torr·ℓ/g(Zr-Al))  
 $T$  = regeneration temperature (K)

and the exponential term is the solubility constant for hydrogen in Zr-Al in terms of  $\text{torr} \cdot \ell^2 / \text{g}(\text{Zr-Al})^2$ . The following assumptions were made in evaluating Eq. (IV.5-1):

- The entire gas load is absorbed onto the getter panels.
- The panels pump uniformly at a constant rate.
- The temperature is constant during regeneration.
- The hydrogen pumping rate is  $1\text{--}10 \text{ } \ell\text{s}^{-1}\text{cm}^{-2}$  depending on activation and operating conditions.<sup>(6)</sup> Correcting for isotope effects, the getter pumping speed for DT is approximated as  $3 \text{ } \ell\text{s}^{-1}\text{cm}^{-2}$ .
- Average required pressure  $\sim 5 \times 10^{-5}$  torr.
- Total Zr-Al mass  $\sim 5 \times 10^4$  g.

Fig. IV.5-3

# TRITIUM INVENTORY IN NEUTRAL BEAM PUMPING SYSTEM



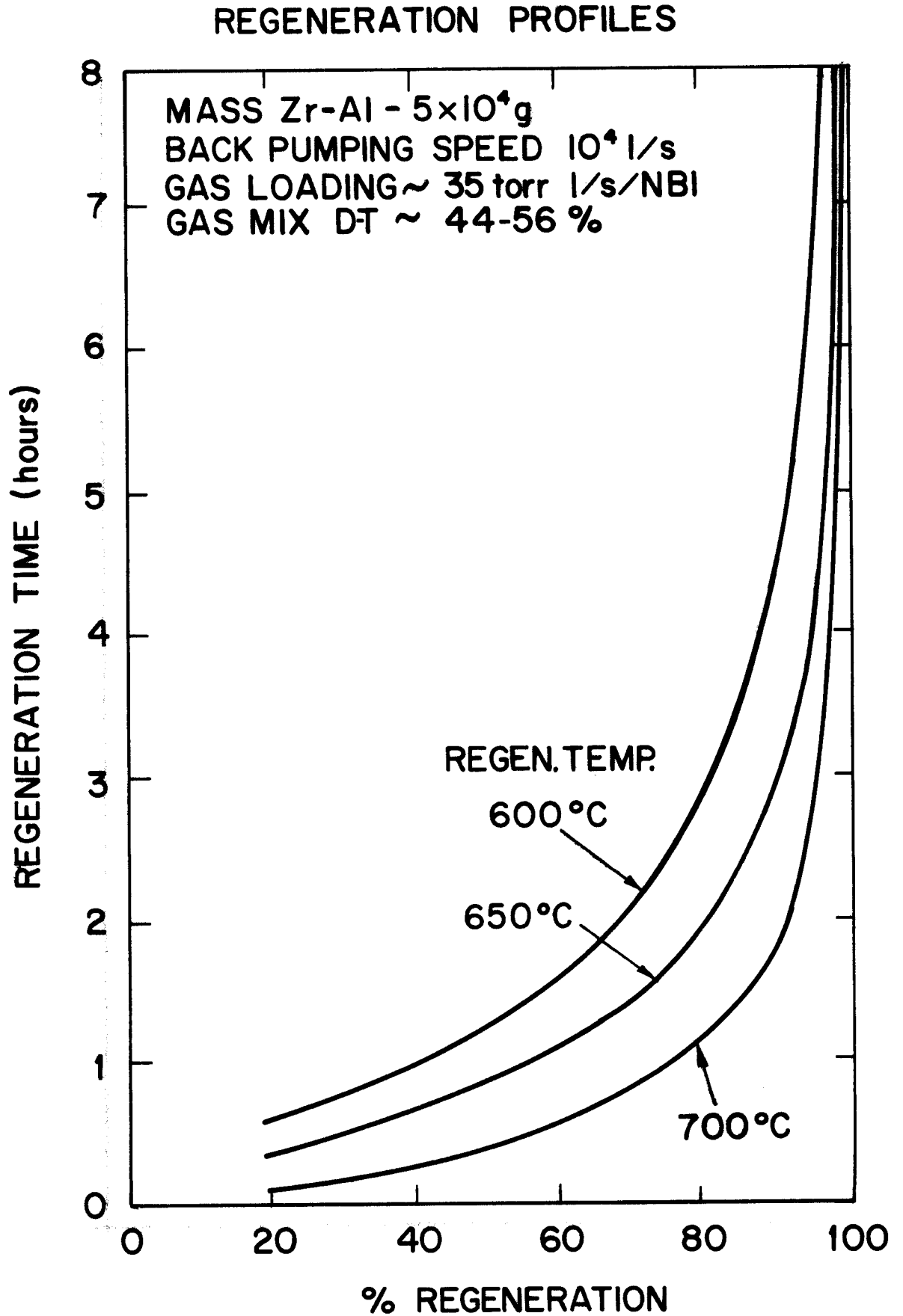
For a DT gas loading of 35 torr·ℓ/sec, the regeneration profiles at 600, 650, and 700°C are shown in Fig. IV.5-4. The regeneration temperature is kept below 750°C to avoid any adverse effects of copper evaporation from amagnetic constantan (Cu-45% Ni) backing plates.<sup>(16)</sup> It is desirable to choose a minimum regeneration time at the maximum allowable temperature. The thermal effects of continuous frequent recycling may be critical;<sup>(17)</sup> however, the getter panel lifetime is usually limited by impurity absorption.<sup>(9)</sup> For 700°C regeneration, values as short as 30-45 min<sup>(18,10)</sup> have been reported. The maximum value is determined by the H<sub>2</sub> embrittlement limit of 20 torr·ℓ/g<sup>(16)</sup> which corresponds to ~ 8 hrs for the given gas loading. For TASKA a 45 min regeneration time was chosen at 700°C. This results in regenerating about 60% of the absorbed gases. The pump inventory after 45 min on-line is 110 g T (18.3 g T/NBI) and the initial tritium hold-up after degassing is 44 g T (7.3 g T/NBI).

At steady state the hold up will be 73 g (44 g T/0.6). This gives a tritium inventory of 183 g on the pumping face and 73 g on the degassing face resulting in a 256 g T total inventory in the Zr/Al getter panels in the DT beam lines.

It is noted that the above analysis is preliminary and contains many approximations and assumptions that can be evaluated only when the complete details of the vacuum system are available. The general principles illustrated, however, are important and lay the ground work for a more in-depth future study.

For the neutral beams which inject pure deuterium, the regeneration time is much less critical. The major source of tritium contamination into the deuterium NB is particle backstreaming from the reactor vessel. The

Fig. IV.5-4



backstreaming rate depends on the gas pressure near the reactor wall, the local particle loss rate from the reactor plasma, and the area of the duct opening connecting the beam line to the reactor vessel.<sup>(11)</sup> The tritium backstreaming is approximated in a simple fashion from Eq. (IV.5-2) by assuming random molecular flow onto a unit area (Eq. IV.5-2):

$$F = \frac{NP}{(2\pi MRT)^{1/2}} \quad (\text{IV.5-2})$$

where:  $F$  = molecules  $\text{cm}^{-2}\text{sec}^{-1}$

$N$  =  $6.02 \times 10^{23}$  mole $^{-1}$

$R$  =  $8.314 \times 10^7$  erg  $^{\circ}\text{K}^{-1}\text{mole}^{-1}$

$M$  = molecular weight (g mole $^{-1}$ )

$P$  = gas pressure at the wall (dyne  $\text{cm}^{-2}$ )

$T$  = wall temperature ( $^{\circ}\text{K}$ ).

Assuming a pressure of  $\sim 10^{-6}$  torr of gas (50%  $\text{D}_2$  and 50%  $\text{T}_2$ ) and a wall temperature of  $400^{\circ}\text{C}$  results in a flow rate of  $3 \times 10^{14} \text{ T}_2 \text{ cm}^{-2}\text{sec}^{-1}$ . This is probably a maximum value as the neutral gas pressure at the wall is estimated to be in the range of  $10^{-7}$  to  $10^{-8}$  torr. However, gases from the beam dumps may contribute to the equilibrium gas pressure. Therefore,  $10^{-6}$  torr is taken as an upper limit. For one LE-NB with duct size 81 cm x 110 cm the backstreaming rate is  $3 \times 10^{18} \text{ T}_2 \text{ sec}^{-1}$ , which is  $\sim 2\%$  of the injection rate. A similar calculation for the HE-NB and PLUG-NB results in tritium backstreaming of  $5 \times 10^{17} \text{ T}_2 \text{ sec}^{-1}$  (0.4% injection rate) and  $1 \times 10^{17} \text{ T}_2 \text{ sec}^{-1}$  (0.3% injection rate) respectively. The actual backstreaming rate is unknown and will have to be investigated by experiment and in-depth calculations.



The tritium pump inventory in the pure deuterium and proton NB lines is determined by the tritium impurity level that is allowed to build up from backstreaming. The estimated backstreaming is small and thus the getter panels can be regenerated more slowly than the minimum 45 min time used for the DT beam line. For a 12 hr regeneration time, the following total inventories are obtained: LE-NB ~ 2.6 g T, HE-NB ~ 0.5 g T, and PLUG-NB ~ 0.09 g T.

The recycled gases from the pure deuterium and proton beams are continuously sent to the isotopic separation unit (Section VI.5) so that tritium impurities are not allowed to build up in the beams. This minimizes the pump tritium inventory and also reduces tritium losses in the beam and ion dumps of the mono-isotopic neutral beams.

#### IV.5.1.3 Reactor Exhaust

The DT fusion power in TASKA is 86 MW. At a particle energy of 17.6 MeV, this corresponds to a burn rate of  $3.05 \times 10^{19}$  particles/sec. The tritium fractional burnup in TASKA can be defined in two ways:

1. Overall burn fraction ( $f_o$ ):

$$\% f_o = \frac{T_{\text{burned}}}{T_{\text{total input to beam lines}}} = \frac{13 \text{ g/d}}{5000 \text{ g/d}} \times 100 = .26\%$$

2. Injected burn fraction ( $f_I$ ):

$$\% f_I = \frac{T_{\text{burned}}}{T_{\text{injected into the reactor chamber}}} = \frac{13 \text{ g/d}}{1500 \text{ g/d}} \times 100 = .87\%$$

The low burn fraction has a negative impact on the tritium inventory and recycling (Fig. IV.5-5).

Fig. IV.5-5

## TRITIUM INVENTORY IN EXHAUST PUMPS

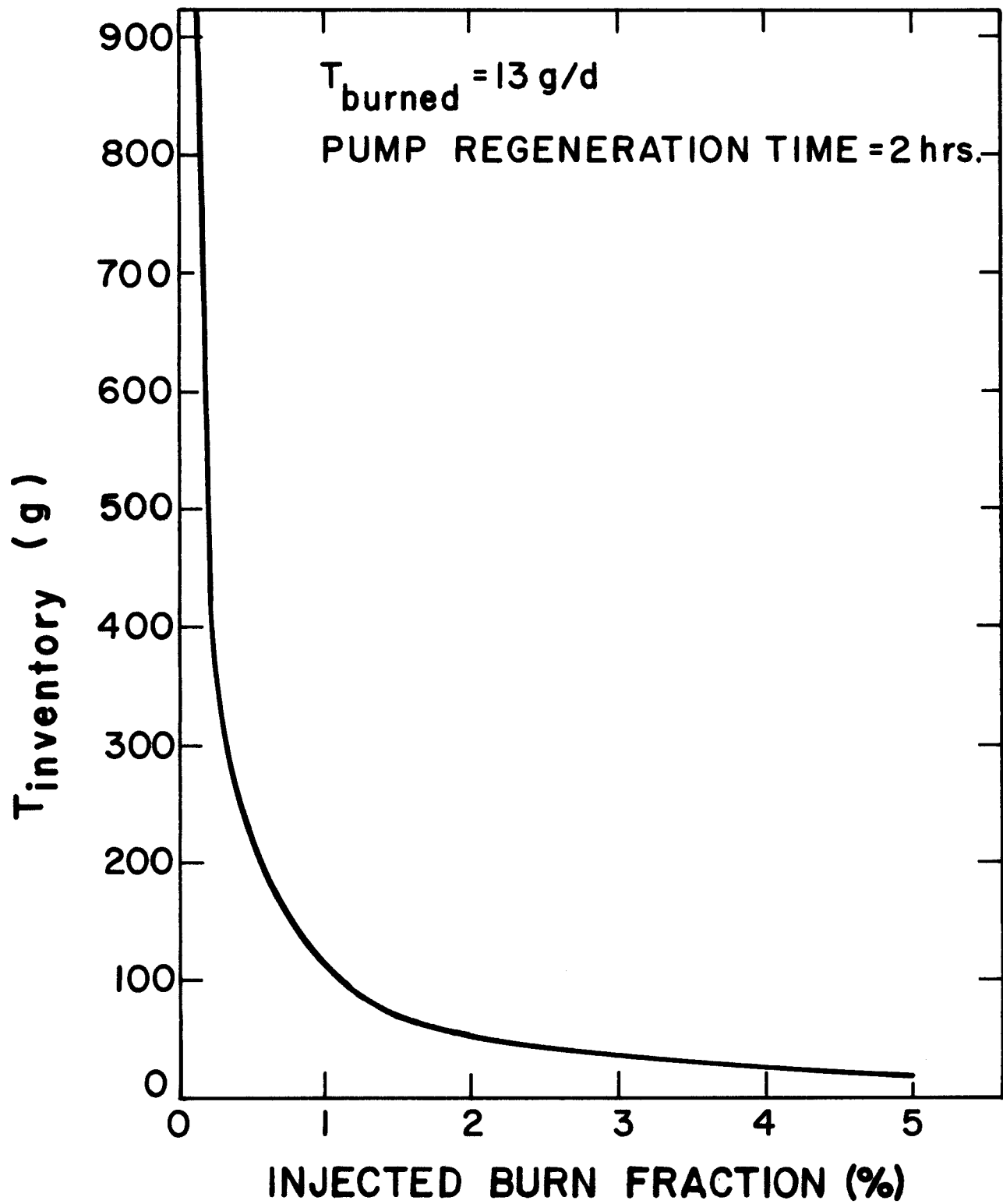


Table IV.5-6. Reactor Exhaust Characteristics

|                      | <u>T</u> | <u>D</u> | <u>H</u> | <u>He</u> |
|----------------------|----------|----------|----------|-----------|
| Fuel Injected (g/d)  | 1,500    | 1,120    | 20       | --        |
| Fuel Burned (g/d)    | 13       | 9        | --       | --        |
| Fusion Ash (g/d)     | --       | --       | --       | 17.5      |
| Breeding (g/d)       | 13       | --       | --       | 17.5      |
| Exhaust Pumped (g/d) | 1,500    | 1,111    | 20       | 17.5      |

The components in the exhaust are given in Table IV.5-6. Note that the tritium from the breeding blanket is included as it will be allowed to permeate through the blanket coolant tubes into the central cell (Section VI.5) and will be pumped with the exhaust gases. The helium ash must also be evacuated with the exhaust.

Of the 1500 g T/d that is injected into the central cell, 5% of this gas load (75 g T/d) is not absorbed by the plasma and is incident on a beam dump. The majority of this gas is pumped by cryopanel located in the beam dump area. For an on-line time of 2 hrs the inventory in the beam dump cryopanel is 6 g.

The remaining 95% of the injected tritium will flow axially through the end cells and be deposited in a thermal dump. It is assumed that all of the ions and half of the alphas flow to one end of the central cell and the electrons and remaining alphas flow to the opposite end. Periodically, an area of the thermal dump will be used to test a direct convertor module. For the tritium analysis we will assume only a thermal dump at both ends of the reactor. Particles will impact on the thermal dump and then be pumped by the

end cell cryopanel. For a 2 hr on-line time the inventory in the end cell cryopanel is 119 g T.

#### IV.5.1.4 Tritium Considerations in Beam, Ion, and Thermal Dump Areas

In TASKA there are three main areas where high energy tritium particles impact metal surfaces; the beam and ion dumps associated with the DT-NBI and the thermal dump in the end cells. In these areas tritium particles implant at a certain depth and then diffuse in both directions. Since the implantation depth is small with respect to the thickness of the dump and the front face temperature is much higher than the back face, the majority of particles will diffuse out the front face into the vacuum and only a small fraction diffuses out the back into the dump coolant.

A simplified method of estimating the flow of tritium into the coolant assumes that the front and back diffusion rates are equal; then, the ratio of the tritium flux to the water is related to the flux to the vacuum by the ratio  $\delta/t$  where  $\delta$  is the implantation depth and  $t$  is the plate thickness.<sup>(19)</sup> For the DT beam dump a flux of 37.5 g T/d hits a 6 mm thick molybdenum surface and implants at a depth of  $5.4 \times 10^{-5}$  cm. This results in a loss of about 34 Ci/d per dump using the simple model.

A more sophisticated model which takes the temperature dependence of the diffusion process into account is outlined in Section VIII.5. This model gives much lower values for the diffusion of tritium into the coolant as the temperature on the back face is much lower. The estimated losses to the dump coolant using the temperature dependent model are summarized in Table IV.5-7. These estimates appear more realistic than those obtained from the simplified method. For example, a value of 16 Ci/d is calculated for the diffusion into the DT beam dump coolant (Section VIII.3). Tritium barriers can also be used

Table IV.5-7. Tritium Implantation Parameters

|                                                  | End Cell             |                      |
|--------------------------------------------------|----------------------|----------------------|
|                                                  | <u>DT Beam Dump*</u> | <u>Thermal Dump</u>  |
| Material                                         | Mo                   | Stainless Steel      |
| Area (cm <sup>2</sup> )                          | $1.2 \times 10^4$    | $5.4 \times 10^5$    |
| Thickness (cm)                                   | 0.6                  | 0.4                  |
| Temperature (°C)                                 |                      |                      |
| Front Face                                       | 277                  | 546                  |
| Back Face                                        | 190                  | 200                  |
| Tritium Flux (g/cm <sup>2</sup> /sec)            | $7 \times 10^{-8}$   | $1.6 \times 10^{-8}$ |
| T Implantation Depth (cm)                        | $5.4 \times 10^{-5}$ | $4 \times 10^{-5}$   |
| C <sub>max</sub> (g T/cm <sup>3</sup> )          | $6.6 \times 10^{-5}$ | $1 \times 10^{-7}$   |
| T <sub>inventory</sub> (g)                       | 0.50                 | 0.02                 |
| T <sub>loss-to-coolant</sub> (Ci/d) (no barrier) | 16                   | 7                    |

\*Totals for all DT beam dumps

to minimize the permeation even further (< 1 Ci/day). The maximum tritium inventory (Table IV.5-7) in the dumps is calculated by multiplying the maximum concentration at the implantation depth by the total dump volume. (There is insufficient data to do a similar calculation for the DT ion dump at this time.)

#### IV.5.1.5 Storage

A certain quantity of tritium must be maintained in storage in the event of a temporary malfunction of any of the tritium handling equipment. Typical-

ly the storage supply is equivalent to a one day's fuel supply. For TASKA this corresponds to 5000 g T stored on uranium beds.

#### IV.5.1.6 Comments on the Tritium Fuel Cycle

The tritium fuel cycle has been described with particular emphasis on the use of tritium neutral beams, Zr-Al pumping and losses through dump areas. The tritium inventory has also been estimated.

The following areas need further work in the TASKA tritium fuel cycle:

- Integration of the analysis of tritium in the Zr-Al getter panels with the overall vacuum system designed by FRG.
- Reassessment of the regeneration profiles with the vacuum system data obtained from FRG.
- Indepth calculations or experiments are needed to assess the problem of backstreaming.
- Effects of tritium decay and the formation of  $^3\text{He}$  in the neutral beam pumping system.
- Assessment of total impurities in NB recycle gases.
- Incorporation of tritium barriers into beam, ion, and thermal dump areas.
- Further details about the ion dump are needed to calculate the tritium losses.
- More detailed examination of loss mechanisms from the fuel cycle.

It is difficult to compare the tritium aspects of neutral beam fueling with other proposed methods of fueling magnetic confinement devices such as pellet injection or gas puffing. Improvements in neutral beam technology may increase the efficiency reducing the tritium inventory and recycling. The efficiencies and inventories associated with the formation of simple, solid DT fuel pellets are difficult to assess. However, based on the TASKA analysis,

pellet fueling would decrease the large amount of fuel recycling, eliminate the problems associated with tritium losses in the DT ion and beam dumps and reduce the tritium storage inventory. Finally, the tritium aspects cannot be compared without also comparing the plasma physics fueling considerations.

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## V The TASKA Magnet System

### V.1 Introduction

#### V.1.1 Magnet Design Philosophy

The objective of the TASKA study is the conceptual design of a mirror-based fusion engineering test facility using technologies that should be available in a few years. In the field of magnet technology, axisymmetric coils of 15 T peak field ( $\text{Nb}_3\text{Sn}$ -conductors) and 8 T peak field in C-shaped coils ( $\text{NbTi}$ -conductors) at 4.2 K can probably be provided in the near future. These data must be taken as constraints for the plasma physics model of TASKA. On the other hand, the basic geometry of the magnet system is determined by both the plasma physics requirements and space requirements for heating and handling.

Firstly, the field magnitude on the machine axis is important, as well as the ripple in the central cell region. Special field shapes have to be taken into account, e.g. the minimum B field configuration in the end plug or multipole components. The shape of the magnetic field lines must not incorporate any large excursions or runaways.

The second requirement is to provide space for the plasma, the blanket and the radiation shield. The neutral beam injectors must be oriented with the correct injection angles. Space for microwave heating waveguides and vacuum pumps has to be provided. The coil supports and the reinforcement also need space as well as the supply lines for current and coolant. Diagnostic devices for plasma measurement and control need space in low field regions.

The reaction of the electromagnetic forces in the structure must be compatible with other space requirements as mentioned above. In particular, the reaction of a pulsed load, e.g. in case of failure, has to be taken into account. However, in TASKA fatigue problems are of minor importance compared

with tokamaks. Additional structural requirements are the maximum allowable strain and stress in the windings and the arrangement and size of cooling channels.

Conductor-related requirements taken into account are the design current density in the superconducting material at the designed magnetic field strength and operating temperature. These parameters, together with the filling factor and the amount of stabilization and structural material, give the overall current density in the winding block. For the stabilizing material, the magnetoresistance has to be known, especially for high magnetic fields. The cooling mode and the arrangement (radial or axial or both) have to be chosen. An absolute limitation is the maximum field at superconductor, which is of the order of 8 T for NbTi and 15 T for Nb<sub>3</sub>Sn, at 4.2 K. Conductor movement must be largely restrained. For pulsed operation the AC losses must be accommodated.

Other requirements for the magnet system are low cost and a reliable safe discharge. Magnetic pollution has to be as small as possible and, in certain critical areas, magnetic shielding is necessary, e.g. for neutral beam sources.

#### V.1.2 General Description of the Magnet System

The magnet system of TASKA consists of:

- 3 solenoids in the central cell to give the required central cell field and ripple (CC1, CC2, CC3),
- 2 sets of hybrid solenoids to provide the magnetic mirror field between central cell and barrier (BC1, BC2),
- 2 transition coils for plasma cross section shaping (RCN1, RCN2),
- 2 yin-yang systems to provide gross MHD-stability with a minimum B

- region (CEEN1 and CEEP1, CEEN2 and CEEP2),
- 2 recircularizing coils for plasma cross section shaping (RCP1, RCP2) and
  - 8 coils for field shaping in the thermal barrier region (FSC1, ..., FSC8).

The arrangement and the winding cross sections of the magnets are shown in Fig. V.1-1; the names are also indicated. Only the right half of the machine is shown, the left half being a mirror image but rotated by 90° around the z-axis. Fig. V.1-2 shows a computer drawing of the TASKA magnet system and Fig. V.1-3 shows the types and specifications of the coils.

Table V.1-1 gives the z-positions of the magnet centers, where the center of the barrier coil is defined as the center of the normal conducting insert coil. The main magnetic characteristics are given in Table V.1-2. Fig. V.1-4 shows the magnetic field on axis, while Fig. V.1-1 shows also the field lines in the x-plane (upper half) and y-plane (lower half). The magnet system parameters are the result of a trade-off between plasma physics requirements, magnet engineering limitations and access requirements for neutral beam injection and blanket handling.

For the central cell, a four solenoid design was compared with a three solenoid design. With the three solenoid design, the field ripple requirement along the axis  $\Delta B/\bar{B}$  ( $\bar{B}$  = average magnetic field in the central cell) of < 5% could be fulfilled with a coil length of 1.2 m. Additionally, access for shield and blanket handling was sufficient and better than in the four-solenoid-design.

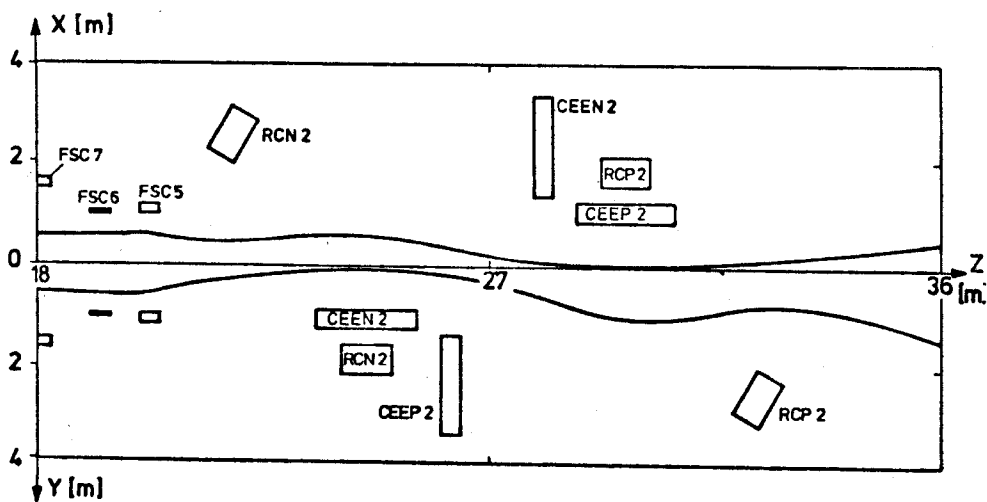
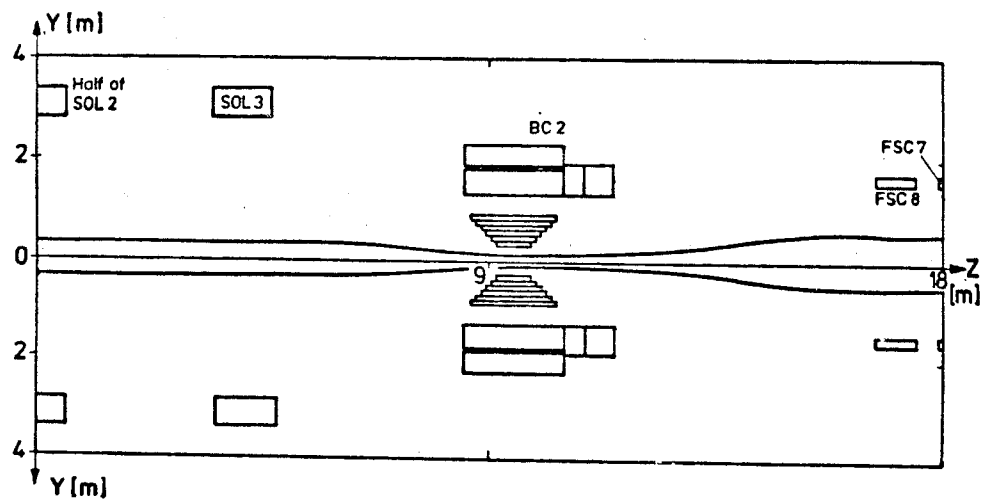


Fig. V.1-1 Winding cross section of the TASKA magnets in x- and y-plane. The right half of the machine is shown.

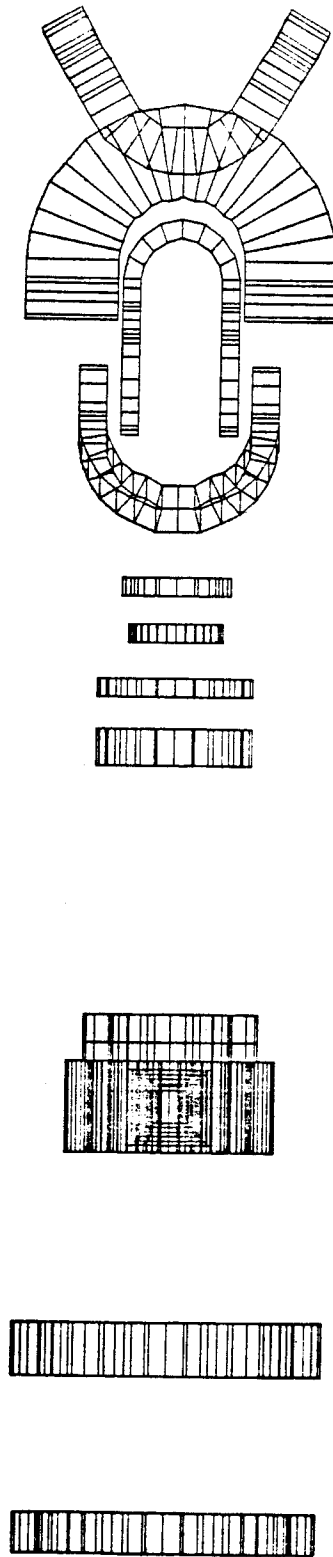


Fig. V.1-2 Computer drawing of the TASKA magnet system (right half).

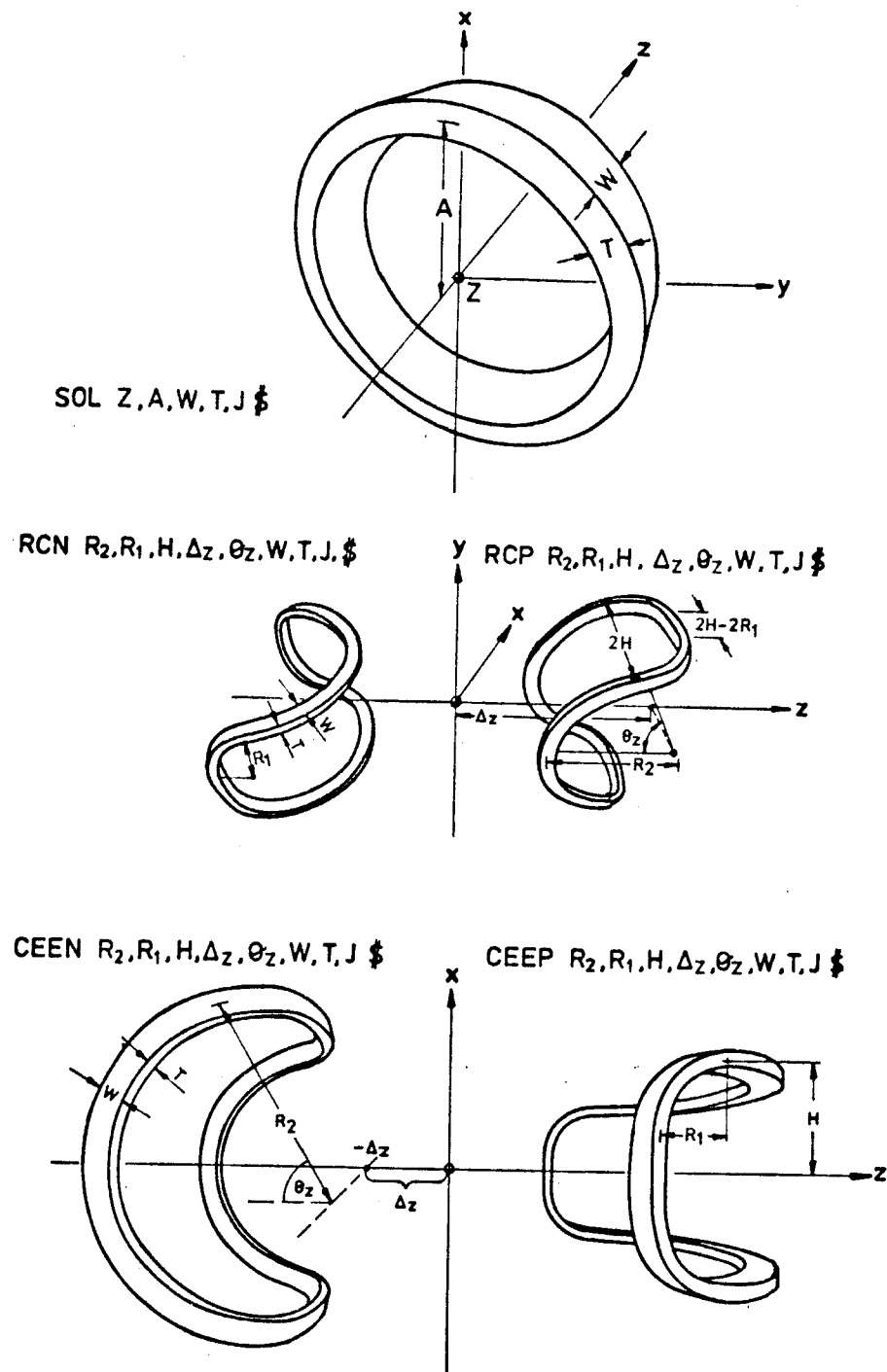


Fig. V.1-3 Magnet types in TASKA, together with the necessary input parameter set to specify each coil in the computer code EFFI(1).

Table V.1-1 z Positions of the Magnet Centers

|            | <u>m</u> |
|------------|----------|
| RCN1       | - 32.16  |
| CEEN1      | - 27.86  |
| CEEP1      | - 27.50  |
| RCP1       | - 23.20  |
| FSC1       | - 20.80  |
| FSC2       | - 19.80  |
| FSC3       | - 18.60  |
| FSC4       | - 17.35  |
| BC1        | - 9.5    |
| SOL1 (CC1) | - 4.2    |
| SOL2 (CC2) | 0.0      |
| SOL3 (CC3) | + 4.2    |
| BC2        | + 9.5    |
| FSC8       | 17.35    |
| FSC7       | 18.60    |
| FSC6       | 19.80    |
| FSC5       | 20.80    |
| RCN2       | 23.20    |
| CEEN2      | 27.50    |
| CEEP2      | 27.86    |
| RCP2       | 32.16    |

Table V.1-2 Main Magnetic Characteristics of TASKA

|                                |                             |
|--------------------------------|-----------------------------|
| Magnetic field in central cell | 2.7 T                       |
| Maximum field in barrier coil  | 20 T at $z = \pm 9.6$ m     |
| Minimum barrier field          | 0.8 T at $z = \pm 19.8$ m   |
| First maximum in plug          | 6.25 T at $z = \pm 24.70$ m |
| Minimum in plug                | 4 T at $z = \pm 27.20$ m    |
| Second maximum in plug         | 6.25 T at $z = \pm 29.70$ m |

All quoted fields are vacuum fields.



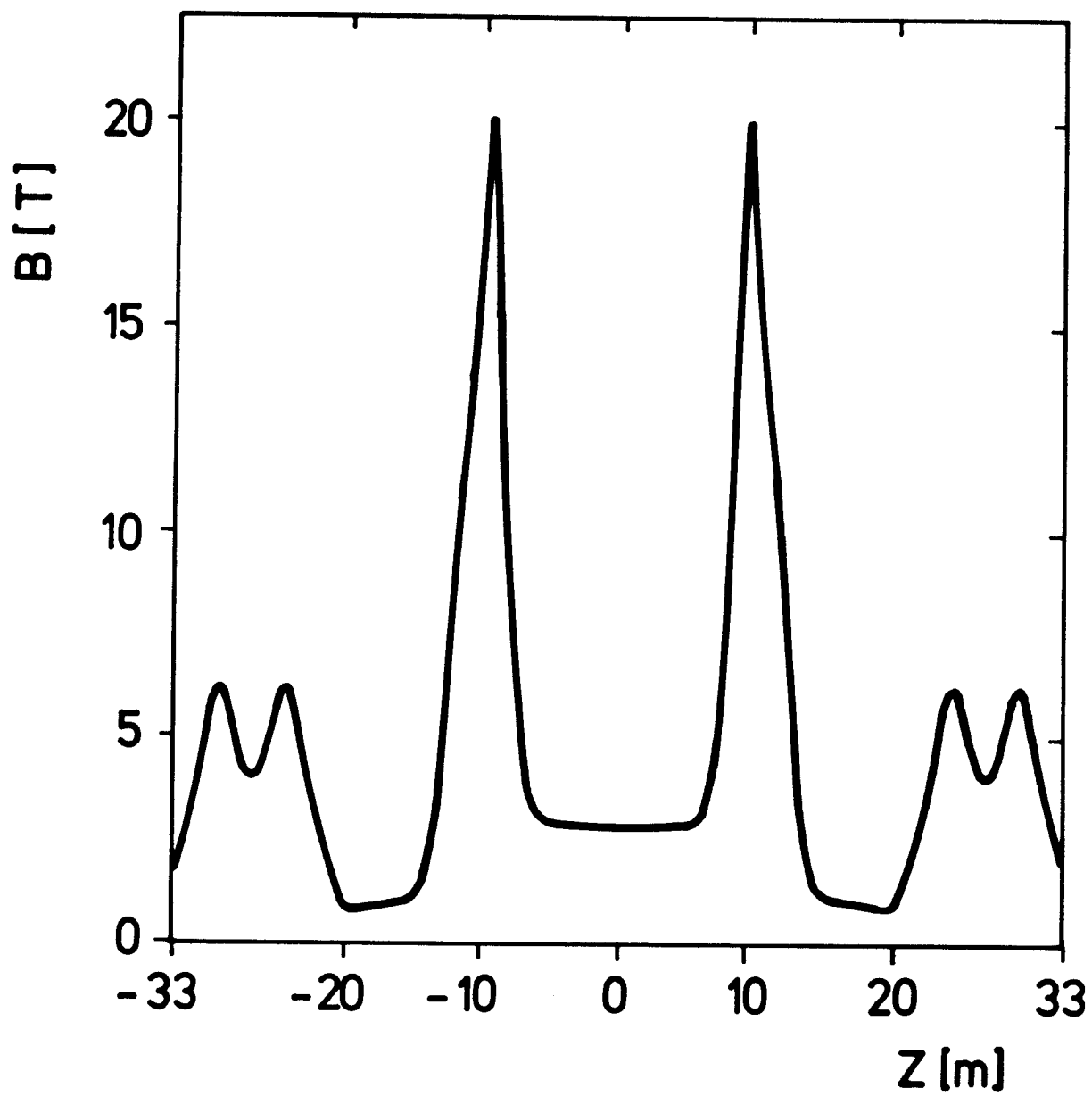


Fig. V.1-4 Magnetic field on axis.

The barrier coil shape takes into account the low injection angle of  $20^\circ$  of the high energy neutral beam injector. This coil is built up by a set of superconducting solenoids consisting of four winding packages with different current designs and different thicknesses of the winding cross sections to provide space for the neutral beam injector. In order to obtain a B-field as high as possible and to provide space for the neutral beam dumps, a normal conducting insert coil is placed concentrically within the superconducting coil set and located axially with respect to the center of largest radial thickness of the superconducting packages. The injection field region is defined by the plasma physics requirements and in particular by the footprint of the medium energy neutral beam. The field shape in the injection region was adjusted by using different current densities in the smaller parts of the barrier coil.

The positioning of the high energy neutral beam injector required an increase of the barrier length over the value defined by physics and magnet technology considerations. It was achieved by an axial translation of the end plug magnets outside. However, as a result, the minimum field in the thermal barrier region is lower than 0.8 T required in the plasma model. Therefore, field shaping coils (in this case normal conducting) on each side are used to enhance the field to the required level; these coils will be integrated into the shield. The boundary position of these coils is dictated again by the beam ducts of the high energy, medium energy and low energy neutral beam injectors.

The C-shaped end plug magnets are more compact than the end plug magnets in WITAMIR-I<sup>(2,3)</sup> and are comparable to the MFTF-B end plug magnets.<sup>(4)</sup> This is because of a smaller plasma radius and less shield thickness. Note that the maximum field in the plug is 6.25 T and the minimum field is 4 T, while

the maximum conductor field is less than 8 T. Unique to this design in the end plug is that the yin and the yang magnets have no common center; in fact the centers of the magnets are separated by 0.36 m. The reason is that for a given minimum plug field (4 T), the maximum plug field can be increased without raising the conductor field. Thus, a plug mirror ratio as high as possible can be provided.

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## V.2 Central Cell Magnets

The three central cell magnets have an inner diameter of 5.6 m and winding cross sections of 1.2 m x 0.5 m (central solenoid CC2) and 1.2 m x 0.58 m (outer solenoids CC1 and CC3), respectively. These magnets are comparable to big bubble chamber magnets already existing, e.g., at ANL<sup>(1)</sup> and CERN.<sup>(2)</sup> Therefore, the technology for the central cell magnets is currently available and is based on the experience with these large coils.

The axial spacing of the three magnets allows sufficient access to the interior of the central cell for exchange of blanket modules or portions of the shield. In Table V.2-1 the main parameters of the central cell magnets are given. The current density of the center coil is 1160 A/cm<sup>2</sup>, while the outside coils have 1350 A/cm<sup>2</sup> to fulfill the ICRH requirements. The maximum field at the conductor is less than 6 T.

Fig. V.2-1 shows a cross section of the center coil and one of the outside coils with the B-field contours of 5 T, 4 T, 3 T and 2 T. It turns out that the current density could be graded for the most cost-effective design.

The conductor for these coils consists of NbTi with copper and/or aluminum stabilizer and stainless steel reinforcement. The conductor is described in section V.7.

Table V.2-1 Main Central Cell Magnet Parameters

|                                    |                                   | CC1      | / | CC2      | / | CC3      |
|------------------------------------|-----------------------------------|----------|---|----------|---|----------|
| <u>Geometric Features</u>          |                                   |          |   |          |   |          |
| Center of magnet, z-position       | Unit<br>m                         | -4.2     |   | 0.       |   | 4.2      |
| Minor radius                       | m                                 | 2.8      |   | 2.8      |   | 2.8      |
| Major radius                       | m                                 | 3.38     |   | 3.3      |   | 3.38     |
| Average radius                     | m                                 | 3.09     |   | 3.05     |   | 3.09     |
| Bundle cross section               | mxm                               | 1.2x0.58 |   | 1.2x0.50 |   | 1.2x0.58 |
| Mean turn length                   | m                                 | 19.42    |   | 19.17    |   | 19.42    |
| Volume of winding                  | m <sup>3</sup>                    | 13.5     |   | 11.5     |   | 13.5     |
| Mean density                       | t/m <sup>3</sup>                  | ~ 6.3    |   | ~ 6.3    |   | ~ 6.3    |
| Weight of winding                  | t                                 | ~ 86     |   | ~ 72.5   |   | ~ 86     |
| <u>Electromagnetic Features</u>    |                                   |          |   |          |   |          |
| Block current density              | A/cm <sup>2</sup>                 | 1350     |   | 1160     |   | 1350     |
| Ampere-turns                       | 10 <sup>6</sup> A-turns           | 9.4      |   | 6.96     |   | 9.4      |
| Self-inductance per N <sup>2</sup> | 10 <sup>-6</sup> H/N <sup>2</sup> | 8.325    |   | 8.343    |   | 8.325    |
| Stored self-energy                 | MJ                                | 368      |   | 202      |   | 368      |
| Max. conductor field               | T                                 | < 6.0    |   | < 6.0    |   | < 6.0    |
| <u>Operating Features</u>          |                                   |          |   |          |   |          |
| Operating temperature              | K                                 | 4.2      |   | 4.2      |   | 4.2      |
| Max. working strain                | %                                 | < 0.25   |   | < 0.25   |   | < 0.25   |
| Max. working stress                | MPa                               | < 500    |   | < 500    |   | < 500    |
| <u>Conductor Properties</u>        |                                   |          |   |          |   |          |
| Superconductor                     |                                   |          |   | NbTi     |   |          |
| Stabilizing material               |                                   |          |   | Al       |   |          |
| Structure material                 |                                   |          |   | SS       |   |          |

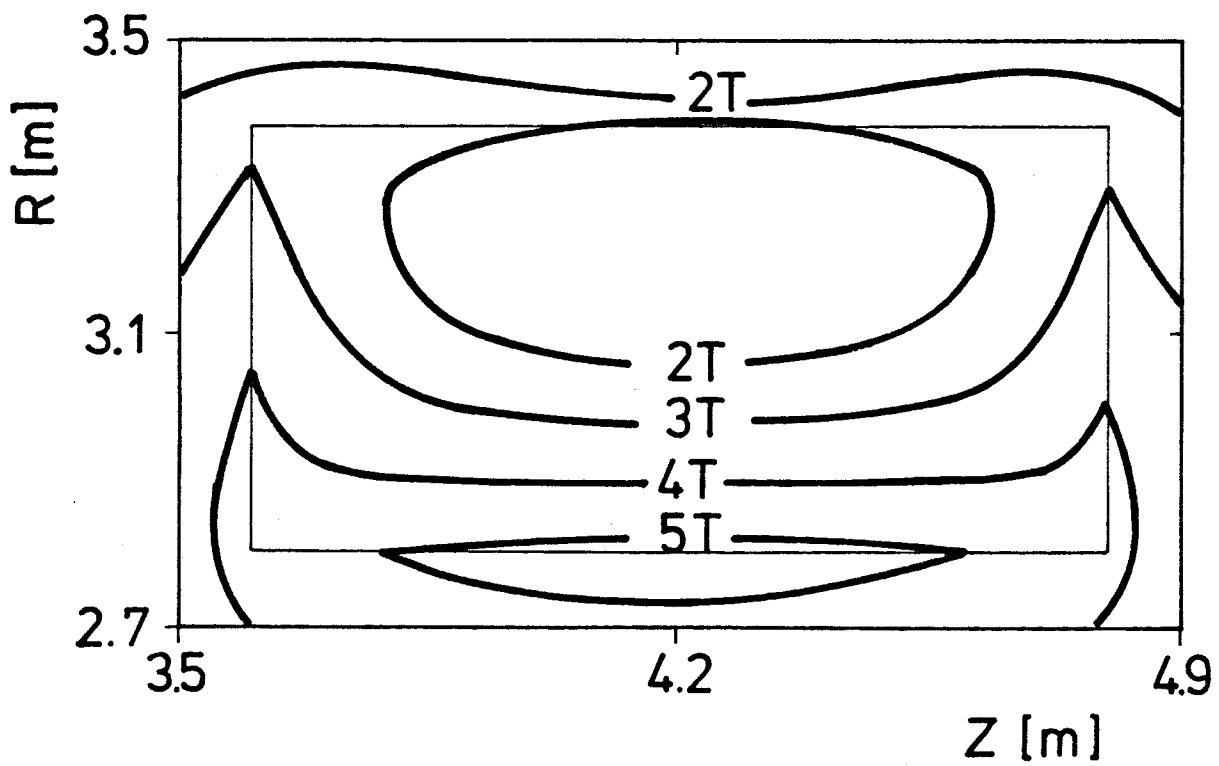
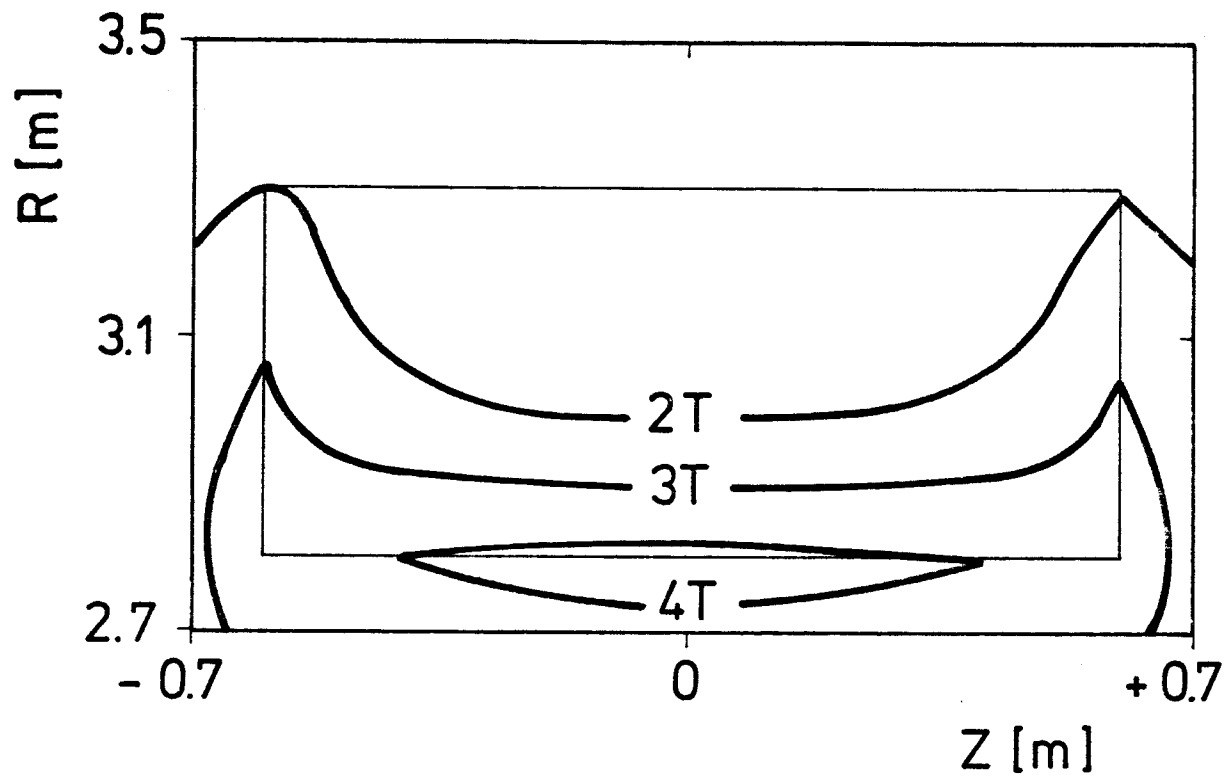


Fig. V.2-1 Cross section of central cell coils with 2 T, 3 T, 4 T and 5 T B-contours. Top: center coil in central cell; bottom: neighbor coil.

References for Section V.2

1. R.E. Jones, et al., Advances in Cryogenic Engineering, 15, 141 (1970).
2. F. Wittgenstein, Rev. Industries Atomique 5/6, 23 (1970).

### V.3 The Barrier Mirror Coil

The barrier mirror magnet is a hybrid consisting of a superconducting outer solenoid set and a normal conducting insert coil. The aim of this magnet is to produce a field in the mirror throat as high as possible. A maximum field of 20 T on axis is achieved with this design. The superconducting coil gives 13.7 T on axis - the rest is delivered by the normal conducting insert coil. The thickness of the superconducting coil varies axially; the reason is to provide access for one of the barrier pump beams as discussed previously in Section V.1.

#### V.3.1 The Superconducting Solenoid Set

The superconducting part consists of four winding packages; a cross section is shown in Fig. V.3-1. Some field lines are given to show the region where NbTi (up to 8 T on conductor) and Nb<sub>3</sub>Sn (up to 15 T) can be used. Figures V.3-2 and V.3-3 give additional information about the radial and axial field variations in the barrier coil. The radial variation shows that the maximum field at the superconductor is 15 T. The radial variation is calculated at the z-position where the maximum on axis field occurs. Figure V.3-3 shows that the field at the superconductor ( $R = 1.3$  m) varies from about 10 T to 14 T.

In Table V.3-1 the main parameters of the superconducting coils are given with Fig. V.3-4 showing the locations and names of the various coils. The design of this coil set approaches the limits of the technology which can be expected in the near future: Nb<sub>3</sub>Sn superconductor with about 15 T at the conductor and a stored energy of about 2.6 GJ in the coil set.



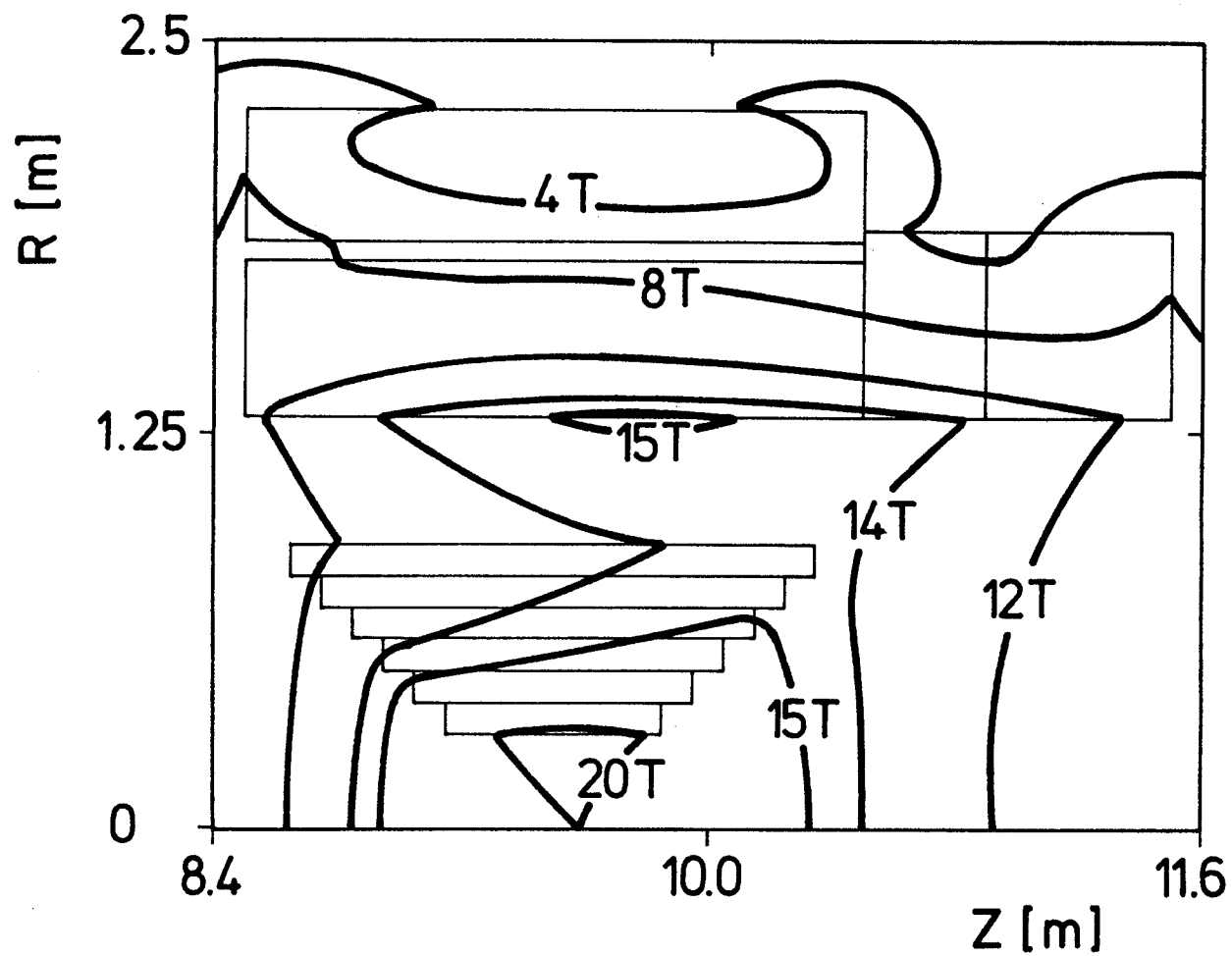


Fig. V.3-1 Cross section of the barrier coil with 4 T, 8 T, 12 T, 15 T, and 20 T B-contours.

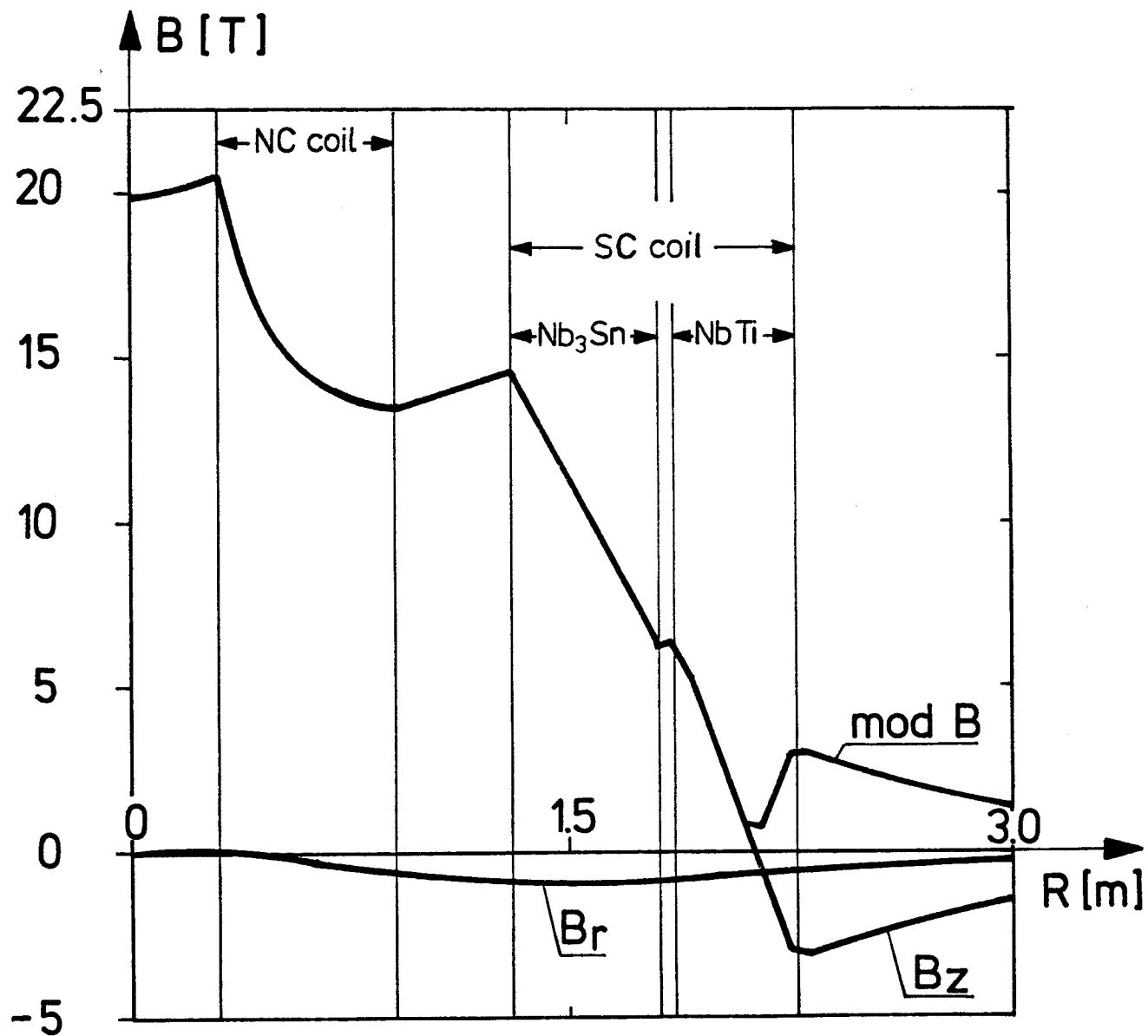


Fig. V.3-2 Radial variation of magnetic field in the barrier coil at  $z = 9.6$  m (maximum field point on axis).

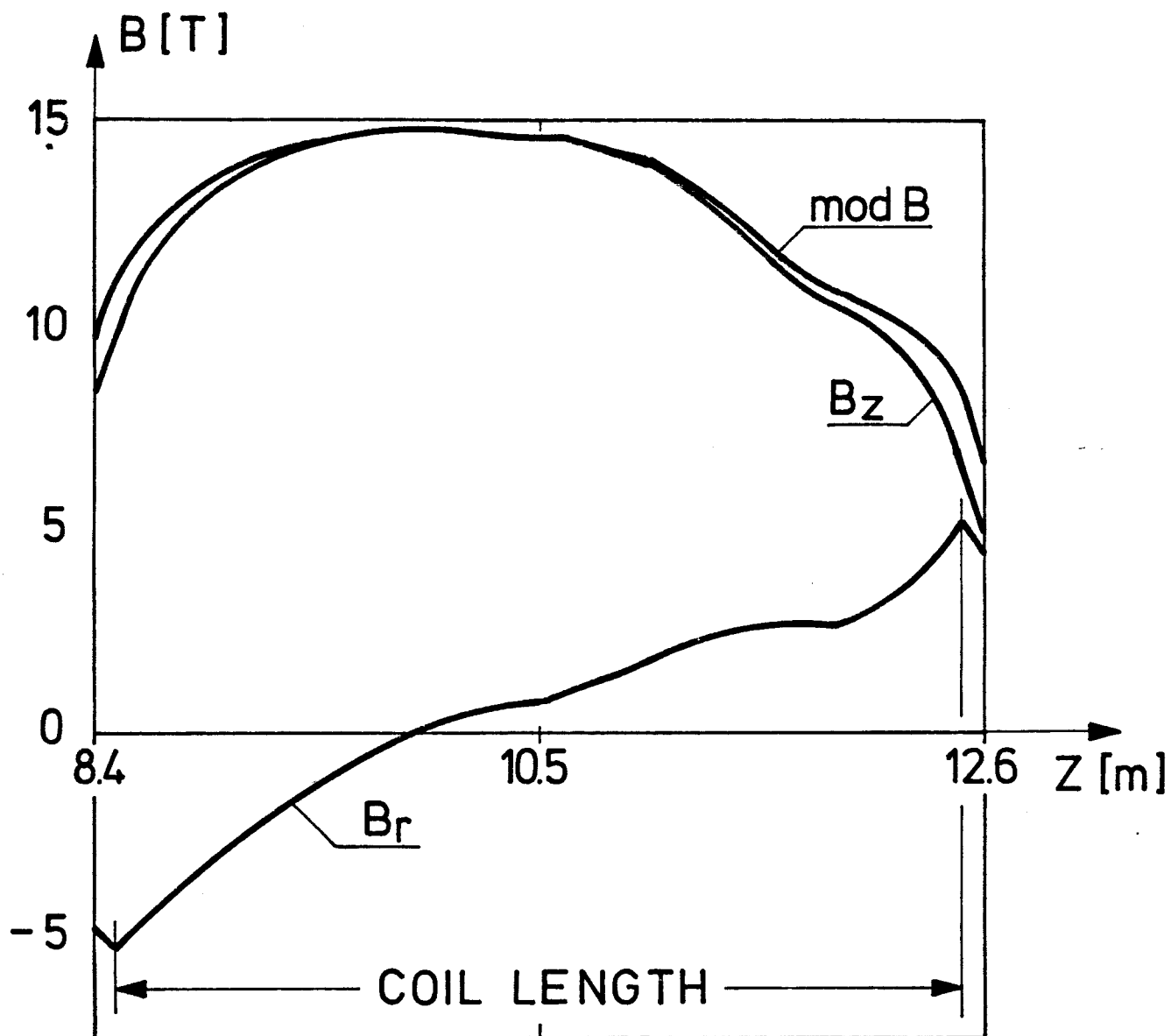


Fig. V.3-3 Axial variation of magnetic field in the barrier coil at  $r = 1.3$  m (along the inner edge of the superconducting winding block).

Table V.3-1. Main Magnet Parameters of the Superconducting Solenoid Set  
of the Barrier Coil

|                                                                                         |                                   | LI                 | LO         | ML                 | MR                 | Total |
|-----------------------------------------------------------------------------------------|-----------------------------------|--------------------|------------|--------------------|--------------------|-------|
| Center of the magnet:<br>z-position (m) relative<br>to center cell plane of<br>symmetry |                                   | ± 9.5              | ± 9.5      | ±10.7              | ±11.2              |       |
| <u>Geometric Features</u>                                                               | <u>Unit</u>                       |                    |            |                    |                    |       |
| Minor radius                                                                            | m                                 | 1.3                | 1.86       | 1.3                | 1.3                |       |
| Major radius                                                                            | m                                 | 1.8                | 2.28       | 1.9                | 1.9                |       |
| Average radius                                                                          | m                                 | 1.55               | 2.07       | 1.6                | 1.6                |       |
| Bundle cross section                                                                    | m x m                             | 0.5 x 2.0          | 0.42 x 2.0 | 0.6 x 0.4          | 0.6 x 0.6          |       |
| Mean turn length                                                                        | m                                 | 9.74               | 13.0       | 10.05              | 10.05              |       |
| Volume of winding                                                                       | m <sup>3</sup>                    | 9.74               | 10.93      | 2.413              | 3.62               |       |
| Mean density                                                                            | t/m <sup>3</sup>                  | ~ 6.3              | ~ 7        | ~ 6.3              | ~ 6.3              |       |
| Weight of winding                                                                       | t                                 | 62                 | 77         | 15.2               | 22.8               | 177   |
| <u>Electromagnetic Features</u>                                                         |                                   |                    |            |                    |                    |       |
| Block current density                                                                   | A/cm <sup>2</sup>                 | 1600               | 2400       | 2000               | 2200               |       |
| Ampere-turns                                                                            | 10 <sup>6</sup> A-turns           | 16                 | 20.16      | 4.8                | 7.92               | 49    |
| Self-inductance per N <sup>2</sup>                                                      | 10 <sup>-6</sup> H/N <sup>2</sup> | 2.354              | 3.876      | 4.145              | 3.792              |       |
| Stored self-energy                                                                      | MJ                                | 301                | 788        | 48                 | 119                |       |
| Max. conductor field                                                                    | T                                 | < 15               | < 8.6      | < 15               | < 14.5             |       |
| <u>Operating Features</u>                                                               |                                   |                    |            |                    |                    |       |
| Operating temperatures                                                                  | K                                 | 4.2                | 4.2        | 4.2                | 4.2                |       |
| Max. working strain                                                                     | %                                 | 0.24               | 0.24       | 0.24               | 0.24               |       |
| Max. working stress                                                                     | MPa                               | 500                | 500        | 500                | 500                |       |
| <u>Conductor Properties</u>                                                             |                                   |                    |            |                    |                    |       |
| Superconductor                                                                          |                                   | Nb <sub>3</sub> Sn | NbTi       | Nb <sub>3</sub> Sn | Nb <sub>3</sub> Sn |       |
| Stabilizing material                                                                    |                                   | Al                 | Cu         | Al                 | Al                 |       |
| Structure material                                                                      |                                   | SS                 | SS         | SS                 | SS                 |       |

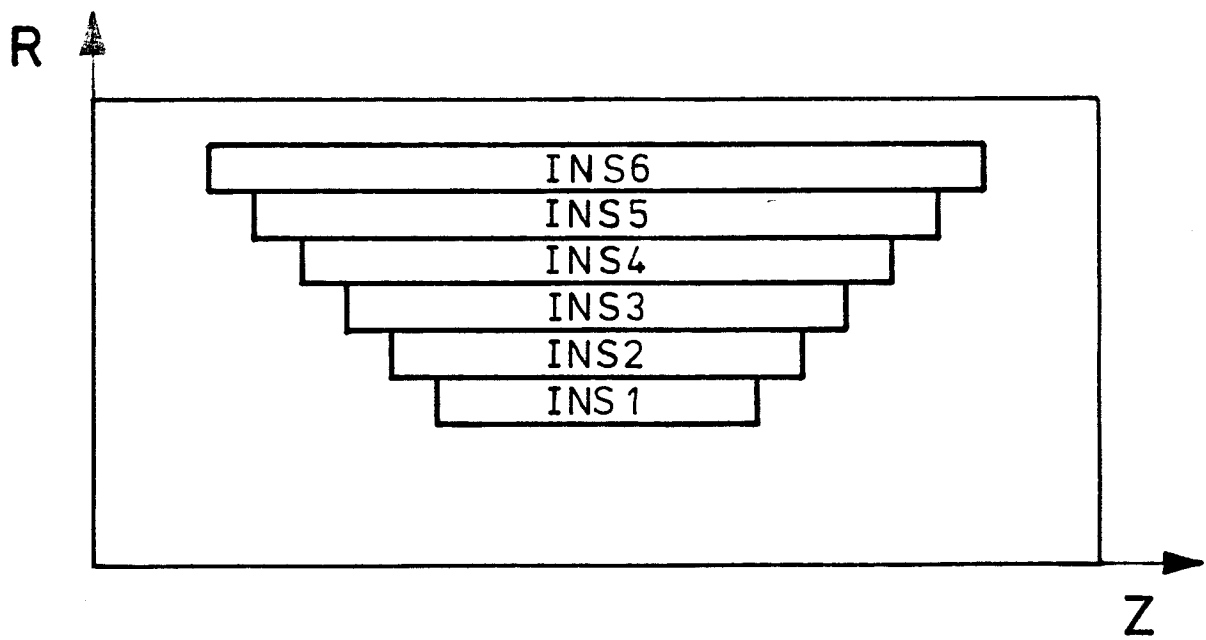
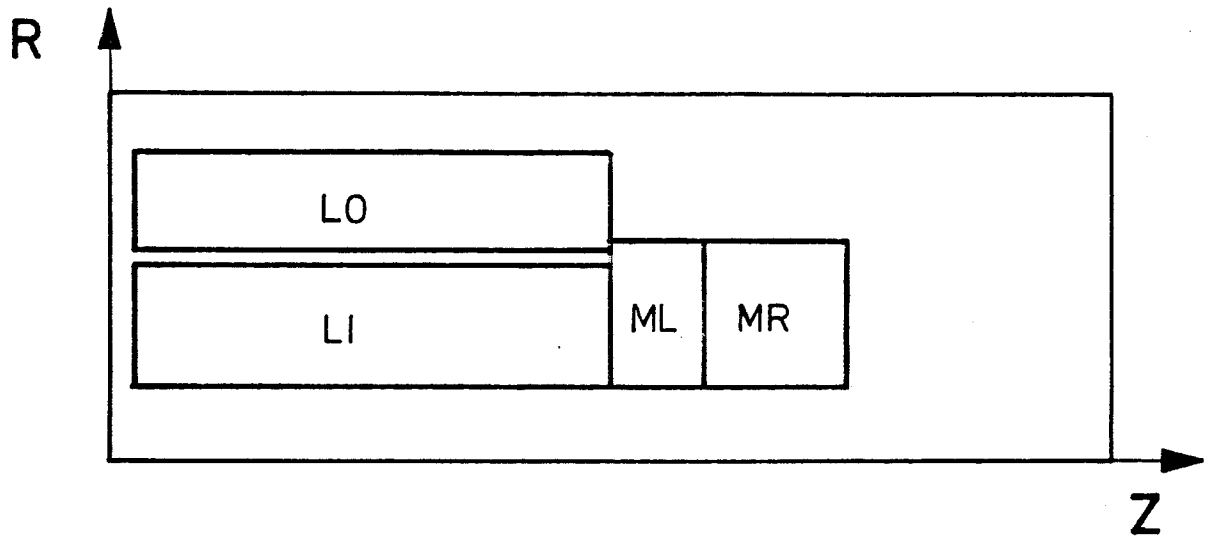


Fig. V.3-4 Names of the magnet parts.

## V.3.2 The Normal Conducting Insert Coil

### V.3.2.1 General Principles

The primary objective of the high field barrier coil in TASKA is to establish a magnetic mirror for particles in the central cell. Increasing the barrier field reduces the particle current passing into the barrier region and, therefore, the requirements for pumping in the barrier. As pumping is done by power-consuming neutral beams into the barrier, a power trade-off was studied (see Section III.3) between pumping requirements and enhancement of the barrier field with normal conducting coils inserted into the radiation shield of the superconducting coils. The sample case chosen for the study showed a decrease in neutral beam power from 200 MW to 150 MW when the peak field on axis in the barrier coil was increased from 13.7 T to 20 T.<sup>(1)</sup> The resulting power needed for operation of the insert coils is less than the 50 MW of beam power saved, and it can be provided more economically via rectifiers. Without further refined optimization studies it was decided to design the barrier coil set for a peak field of 20 T in which 6.7 T are provided by the insert, 13.3 T by the superconducting coil.

The design of the insert coil is governed by the following aspects which are discussed in the following paragraphs:

- Geometric constraints,
- Minimization of dissipated power,
- Stress and heat transfer limits,
- Conductor design and insulation.

Geometric constraints limit the radial region in which a normal conducting insert can be positioned. The radius of the vacuum chamber at the 20 T point is 0.15 m, increasing along the axis towards lower field values. On the

central cell side of the peak field point, the fusion power density in the plasma is about the same as in the central cell, subjecting the normal conducting insert to neutron and gamma radiation. As long as little operational experience with magnet coils in high radiation fields is available, conservative assumptions with respect to the lifetime of conductor and insulation, and to the operating features at high fluxes must be taken. It was therefore decided to allow radial space for a radiation shield of 0.15 m which is sufficient to reduce the radiation flux by more than one order of magnitude. The inner radius of the insert is thus fixed at 0.3 m.

The outside radius of 0.9 m is governed by the position and radiation shielding requirements of the superconducting coil. With an inner coil radius of 1.3 m and 0.1 m of radial space needed for coil case, superinsulation and vacuum tank, 0.3 m are available for a final effective radiation shield in addition to the shielding established by the normal conducting insert. 30 cm of additional shielding is certainly on the safe side and may appear over-conservative but a detailed neutronics analysis must consider voids in the shielding for hydraulic and power leads to operate the insert. Should the detailed analysis show that the amount of shielding can be reduced, decrease of the inner radius of the superconducting coil will be far more economical than increase of the insert outer radius. Finally, the axial length of the insert can be made as long as the superconducting coil if desired.

In a solenoidal magnet, the least power is dissipated for generation of a given central field  $B_0$  if the current density in the cross section varies as the "Kelvin"-distribution<sup>(2)</sup>  $j_k \sim r(r^2 + z^2)^{-3/2}$ . In a real magnet, it is of course impractical to realize this distribution but it should be approached if possible. With the radial variation being most important, it was decided to

subdivide the insert coil set into 6 nested cylinders of 0.1 m thickness each. The length of each cylinder is determined by power optimization for axially constant current density. Let  $\alpha_k = r_{out,k}/r_{in,k}$  and  $\beta_k = l_k/2r_{in,k}$  denote the geometry parameters of the k-th cylinder, then according to Ref. 2 we have:

$$B_{0,k} \sim j_k F(\alpha_k, \beta_k) \quad \text{and} \quad P_k \sim j_k^2 \beta_k (\alpha_k^2 - 1)$$

for field contribution and power dissipation, with  $F(\alpha, \beta) = \beta(\sinh^{-1} \alpha/\beta - \sinh^{-1} 1/\beta)$ . The parameter  $\alpha_k$  is fixed for each coil by its thickness of 0.1 m, giving  $P_k \sim \beta_k/F^2(\alpha_k, \beta_k)$ . From  $dP_k/d\beta_k = 0$  we have  $F(\alpha, \beta) = 2\beta [dF(\alpha, \beta)/d\beta]$  as a condition for  $\beta$  of a solenoid of constant current density, fixed  $\alpha$ , and minimum power dissipation. Evaluation of this condition shows that optimum  $\beta$  is very accurately approximated if the length of the solenoid equals its average diameter (see Table V.3-2).

A similar power minimization is possible for adjustment of current density in the radially nested coils (for a general procedure, see Chapter 4 of Ref. 2). For two concentric solenoids which have to generate a central  $B_0$ , one has  $B_0 = c_k j_k + c_1 j_1$  and  $P = d_k j_k^2 + d_1 j_1^2$  with  $c = \mu_0 r_{in} F(\alpha, \beta)$  and  $d = \rho 2\pi\beta(\alpha^2 - 1)r_{in}^3$ , with  $\rho$  the material resistivity. Loss power  $P$  is minimized if current densities  $j$  are graded such that

$$\frac{j_1}{j_k} = \frac{(c_1 d_k)}{(c_k d_1)} .$$

The procedure just described offers a straightforward method for the design of a set of nested solenoids with constant current density in each individual solenoid. However, experience with small insert coils in high field



Table V.3-2. Main Parameters of the Normal Conducting Insert Coils of the Barrier Coil

|                                                        |                                                                                                                                     | INS 1     | INS 2     | INS 3     | INS 4     | INS 5     | INS 6     | Total |
|--------------------------------------------------------|-------------------------------------------------------------------------------------------------------------------------------------|-----------|-----------|-----------|-----------|-----------|-----------|-------|
| <u>Geometric Features</u>                              |                                                                                                                                     |           |           |           |           |           |           |       |
| Center of the magnet:<br>z-position <sup>+</sup> )     | m                                                                                                                                   | ± 9.5     | ± 9.5     | ± 9.5     | ± 9.5     | ± 9.5     | ± 9.5     |       |
| Minor radius                                           | m                                                                                                                                   | 0.3       | 0.4       | 0.5       | 0.6       | 0.7       | 0.8       |       |
| Major radius                                           | m                                                                                                                                   | 0.4       | 0.5       | 0.6       | 0.7       | 0.8       | 0.9       |       |
| Average radius                                         | m                                                                                                                                   | 0.35      | 0.45      | 0.55      | 0.65      | 0.75      | 0.85      |       |
| Bundle cross section                                   | m x m                                                                                                                               | 0.1 x 0.7 | 0.1 x 0.9 | 0.1 x 1.1 | 0.1 x 1.3 | 0.1 x 1.5 | 0.1 x 1.7 |       |
| Mean turn length                                       | m                                                                                                                                   | 2.2       | 2.83      | 3.46      | 4.084     | 4.712     | 5.341     |       |
| Volume of winding                                      | m <sup>3</sup>                                                                                                                      | 0.15      | 0.26      | 0.38      | 0.53      | 0.71      | 0.91      | 2.9   |
| Weight of winding<br>(Mean density 7t/m <sup>3</sup> ) | t                                                                                                                                   | 1.1       | 1.8       | 2.7       | 3.7       | 4.9       | 6.3       | 20.5  |
| <u>Electromagnetic Features</u>                        |                                                                                                                                     |           |           |           |           |           |           |       |
| Block current density                                  | A/cm <sup>2</sup>                                                                                                                   | 2800      | 1683      | 1137      | 814       | 613       | 480       |       |
| Ampere turns                                           | 10 <sup>6</sup> A-turns                                                                                                             | 1.96      | 1.51      | 1.25      | 1.06      | 0.92      | 0.82      | 7.52  |
| Self-inductance per N <sup>2</sup>                     | 10 <sup>-6</sup> H/N <sup>2</sup>                                                                                                   | 0.4169    | 0.5508    | 0.6877    | 0.8249    | 0.9597    | 1.096     |       |
| Stored self-energy                                     | MJ                                                                                                                                  | 0.97      | 0.64      | 0.48      | 0.4       | 0.32      | 0.28      | 12.0  |
| Loss power                                             | MW                                                                                                                                  | 4.3       | 2.6       | 1.8       | 1.3       | 0.9       | 0.7       | 11.6  |
| <u>Conductor Properties</u>                            |                                                                                                                                     |           |           |           |           |           |           |       |
| Conductor                                              | Hardened copper with $\bar{\rho} = 2.5 \times 10^{-8} \Omega\text{m}$ at average operating temperature and 300 MPa tensile strength |           |           |           |           |           |           |       |
|                                                        | Height > 3.3 cm, Width > 10.7 cm, Current < 100 kA, Pressure drop in water < 15 bar, $\Delta T = 60^\circ\text{C}$                  |           |           |           |           |           |           |       |

<sup>+</sup> ± 9.5 means, one is at 9.5 m on the positive side of the z-axis and one is at -9.5 m on the negative side of the z-axis.

hybrid magnets<sup>(3)</sup> shows that two additional technical constraints must be considered; the upper limit of stress  $\sigma = j r B$  in the coil of the order of 200 to 500 MPa depending on the conductor alloy chosen, and the upper limit on heat removal from the conductor which limits the current density in copper to about 25 kA/cm<sup>2</sup>. While this latter limit is of no concern for large radius magnets, the stress limit will become essential for mirror fusion applications. The TASKA-insert is not yet governed by the stress limit if a copper alloy is chosen that can operate up to 300 MPa (e.g., SE-Cu with temper F 30 as offered by Kabelmetal, Osnabrück/W. Germany).

Heavily radiation loaded magnets were operated in the past mainly in accelerator centers, e.g., LAMPF and SIN.<sup>(4)</sup> The major difference with respect to conventional magnets is that a ceramic insulation must be used if doses of more than 10<sup>9</sup> rad are expected. MgO and Al<sub>2</sub>O<sub>3</sub> as compacted powders will withstand doses in excess of 10<sup>12</sup> rad, making resistivity-increasing transmutations in the conductor material the prime lifetime limiting effect. Recently, a detailed analysis<sup>(5)</sup> was published in which an MgO-insulated copper conductor, jacketed by a steel sheet for chemical and mechanical protection of the insulation, was optimized for use in an unshielded bundle divertor magnet for a tokamak-based FED. The analysis concluded that even with conservative assumptions, the required magnet can be built and operated.

In TASKA, coil INS 1 is simultaneously subjected to the highest ohmic power dissipation and radiation load. Several key parameters are therefore given and justified for this coil; they may be applied to the other insert coils as well. An average resistivity in the copper at the average operating temperature of  $\rho = 2.5 \times 10^{-8} \Omega m$ , is taken, even considering transmutations after 10<sup>12</sup> rad.

### V.3.2.2 Conductor Cross Section

The conductor must be bent at a radius of 0.3 m. Combining LANL experience of ratios (radius of bending to height of 6) and US manufacturer recommendations (a factor of 12 for ceramically insulated conductors) into a factor of 9, a conductor height of  $h = 3.33$  cm results, giving three layers for the coil thickness of 10 cm. Contrary to bundle divertor coils or yin-yang coils, the solenoidal conductor is bent only in one plane, allowing an almost arbitrary width of the conductor. Operational safety and desire for thin insulators call for low voltage, the water cooling system calls for large hydraulic diameters of cooling channels and short channel lengths. Both requirements favor a wide conductor which should be limited only by practical considerations on the operating current. If a reasonably manageable peak current of  $I_0 = 100$  kA is assumed (from considerations of busbar dimensions, losses and cooling, etc.), the conductor width as given by  $w = I_0 / h \bar{J}$ ,  $\bar{J}$  being the average current density in the conductor block of  $2800 \text{ A/cm}^2$  for INS 1, becomes  $w = 10.7$  cm.

### V.3.1.3 Layout of Conductor

According to Table V.3-2, a power of 4.3 MW is ohmically dissipated in INS 1 for a copper filling factor of  $\lambda = 0.7$ . An additional 0.2 MW of nuclear heating must be considered if a peak power deposition of  $2 \text{ MW/m}^3$  is used, as indicated by the neutronics calculations. Power dissipation in the MgO insulation,<sup>(5)</sup> the resistivity of which decreases significantly in a radiation environment, is negligible in INS 1 due to the low terminal voltages and to the 0.15 m shield between plasma and coil. The water cooling system must finally handle a heat load of 4.5 MW, mainly caused by ohmic heating. 30% of the conductor cross section is available for steel jacket, insulator, and

cooling channels. Using a steel jacket of 0.6 mm and an MgO thickness of 1.4 mm, about 3.6 cm<sup>2</sup> of cross section area (~ 10%) remains for cooling channels. In order to have a hydraulic diameter as large as possible and at the same time a large surface area of the channels for good heat transfer, three parallel channels of 1.2 cm diameter each are proposed, spaced along the width of the conductor. Hollow copper conductors with 2 or more parallel channels are produced by Kabelmetal, Osnabrück. With this design and hydraulic operation of the three layers of INS 1 in parallel, the cooling system will require 1.5 MPa of (ideal) pressure drop and 15 m/s of water velocity. If these values are too high for reliable operation in view of pumping power needed, a reduction of current density in INS 1 and an increase in INS 2 may be considered (minimization of the sum of pump power and ohmic power).

In summary, operation of the normal conducting insert coils of TASKA appears feasible even with conservative assumptions. With the remote handling scheme designed for their exchange, an operating failure can be tolerated. Some development needs for construction of the insert are:

- ceramic insulation and jacketing of large rectangular conductors,
- economic provision of a cladding of cooling channels which is corrosion-resistant against radiolysis products of water,
- design of electrical (100 kA) and hydraulic (15 bar) manifolds.

### References for Section V.3

1. R.P. Drake, private communication, August 1981.
2. D.B. Montgomery, Solenoid Magnet Design, New York, Wiley Interscience, 1969.
3. G. Landwehr, "Problems Connected with the Generation of Continuous Magnetic Fields above 15 T," IEEE Trans. Magn. 17(5), (1981) 1768.
4. A. Harvey, "High Current Density Coils for High Radiation Environments," IEEE Trans. Magn. 17(5), (1981) 1717.
5. J.H. Schulta, "A Costing and Sizing Code for Highly Irradiated Normal Magnets," M.I.T. Res. Rep. PFC/RR-81-13, Cambridge, Mass.

#### V.4 The End Plug Coils

Each end plug is formed by a yin-yang pair and two transition coils. The main features of these coils are given in Table V.4-1 with Fig. V.4-1 showing the locations and names of the end plug magnets. The current densities of these C-shaped coils are less than  $2000 \text{ A/cm}^2$ . Their design is straightforward, because they are similar in geometry and electrical parameters to those of the MFTF-B machine. Characteristic dimensions are an overall height of about 7 m for the yin-yang system and about 5 m for a transition coil. These are comparable to the MFTF-B dimensions of about 6 m overall height for the yin-yang and about 12 m for the outside A-cell coil. The successful test of a MFTF-B yin-yang in February of 1982 encourages us to state that the C-shaped and plug magnets of TASKA can be built with current technology.<sup>(1)</sup>

Fig. V.4-2 shows B-contours for the region of closest approach between a transition coil and a yin coil. Fig. V.4-3 demonstrates the magnetic well of the yin-yang.

Table V.4-1 Main Parameters of the End Plug Magnets

|                                    |                                   | TC/RC        | Yin/Yang     |
|------------------------------------|-----------------------------------|--------------|--------------|
| <u>Geometric Features</u>          |                                   |              |              |
| Center of the magnets              | Unit<br>m                         | ±22.7/±31.66 | ±27.0/±27.36 |
| "Minor" radius A 1                 | m                                 | 1.6          | 0.9          |
| "Major" radius A 2                 | m                                 | 1.4          | 1.4          |
| Average "major" radius R2          | m                                 | 1.9          | 2.4          |
| Average "major" radius R2          | m                                 | 1.9          | 1.09         |
| Sweep angle                        | degree                            | 60           | 90           |
| Bundle cross section               | mxm                               | 1.0x0.6      | 2.0x0.38     |
| Mean turn length                   | m                                 | 19.90        | 21.93        |
| Volume of winding                  | m <sup>3</sup>                    | 11.94        | 16.67        |
| Mean density                       | t/m <sup>3</sup>                  | ~ 7          | ~ 7          |
| Weight of winding                  | t                                 | ~ 84         | ~ 117        |
| <u>Electromagnetic Features</u>    |                                   |              |              |
| Block current density              | A/cm <sup>2</sup>                 | 1900         | 1630         |
| Ampere-turns                       | 10 <sup>6</sup> A-turns           | 11.4         | 12.4         |
| Self-inductance per N <sup>2</sup> | 10 <sup>-6</sup> H/N <sup>2</sup> | 7.497        | 5.349        |
| Stored self-energy                 | MJ                                | 487          | 411          |
| Max. conductor field               | T                                 | < 7          | < 7.9        |
| <u>Operating Features</u>          |                                   |              |              |
| Operating temperature              | K                                 | 4.2          | 4.2          |
| Max. working strain                | %                                 | 0.25         | 0.25         |
| Max. working stress                | MPa                               | 500          | 500          |
| <u>Conductor Properties</u>        |                                   |              |              |
| Superconductor                     |                                   | NbTi         | NbTi         |
| Stabilizing material               |                                   |              | Cu(3/4 hard) |
| Structure material                 |                                   |              |              |

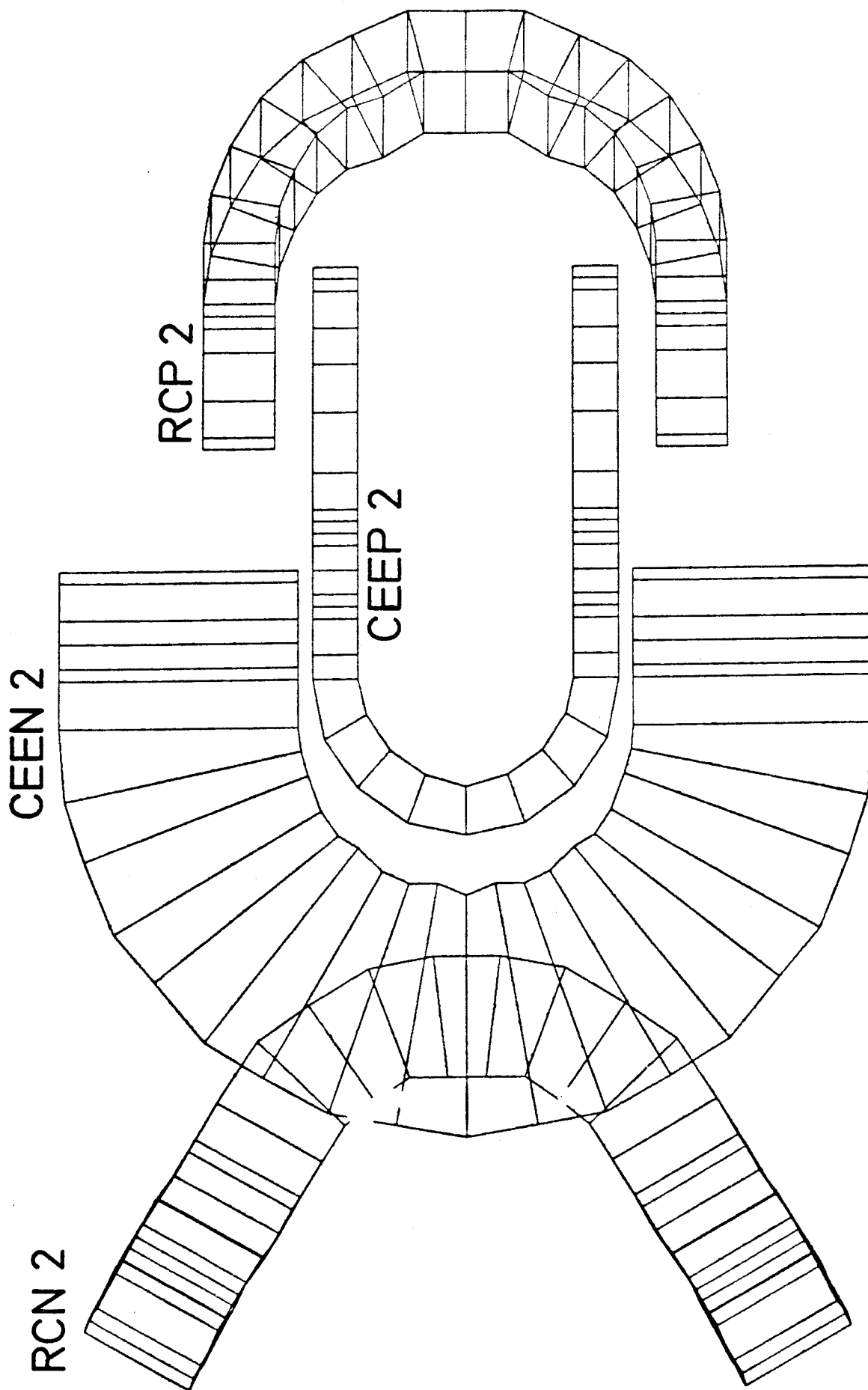


Fig. V.4-1 Identification of end plug magnets.



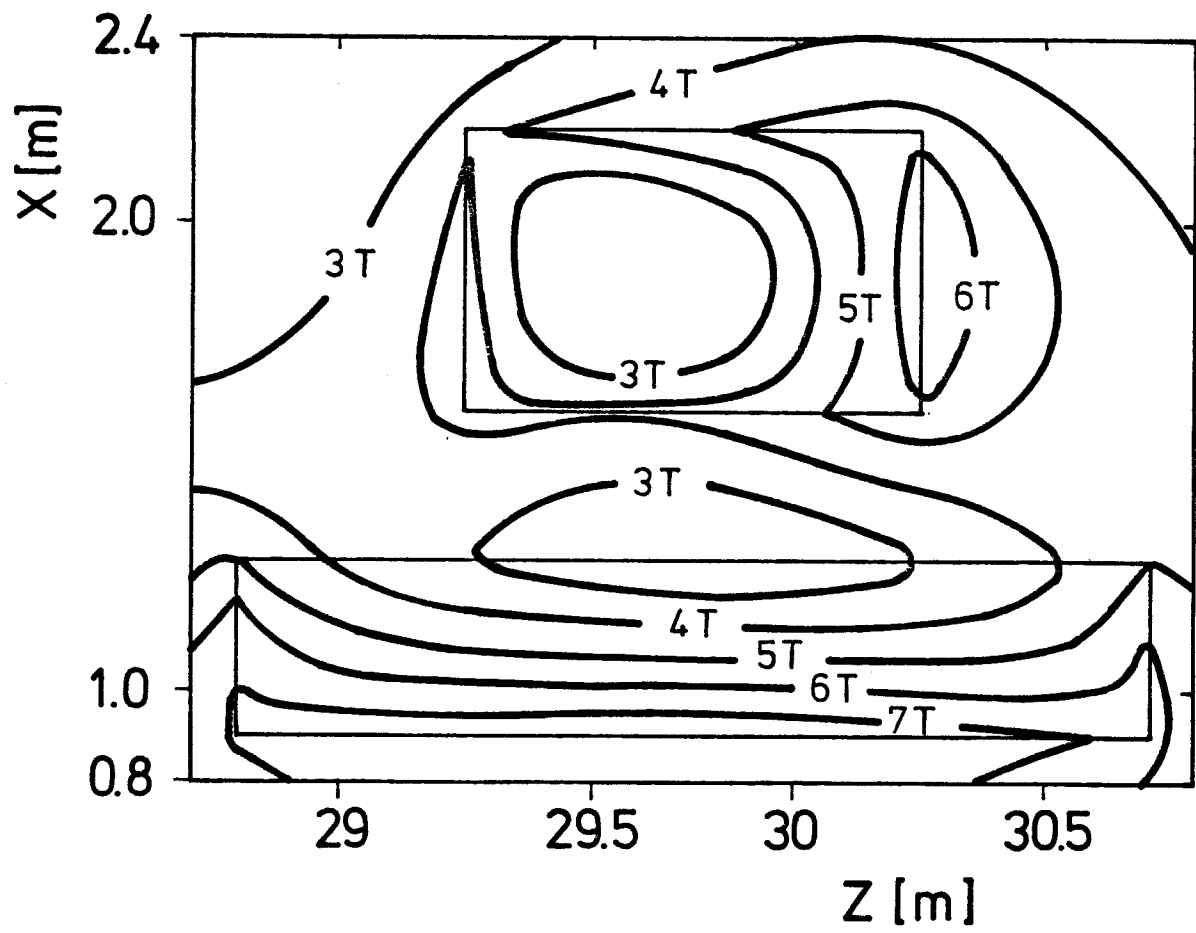


Fig. V.4-2 Field at conductor for the region of closest approach between a transition and yin coil.

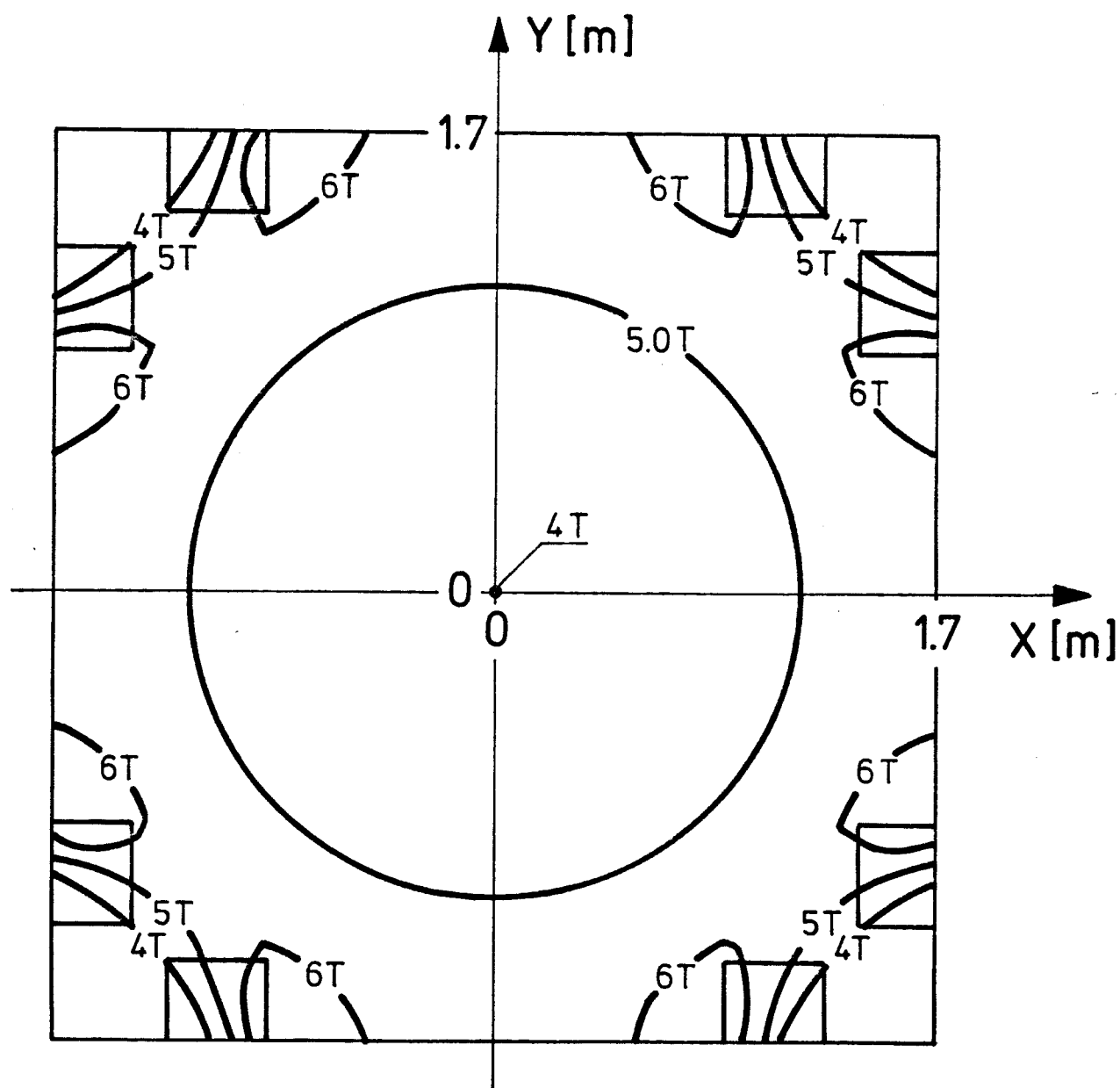


Fig. V.4-3 Magnetic well in the yin-yang coil set ( $z = 27.18$  m corresponds to the center of the coil set).

#### References for Section V.4

1. T.K. Fowler, private communication.

## V.5 Field Shaping Coils

As explained in section V.1, field shaping coils had to be introduced in the barrier region to shape the field such that the high energy pump beam could be injected at the point of minimum barrier field. These coils are partly integrated with the shield. They are normal conducting and will be built with the same hardened copper material as the barrier insert coil. Table V.5-1 gives the main parameters of these coils with Fig. V.5-1 showing their locations and names. The ohmic losses of the eight field shaping coils are  
9 MW.

Table V.5-1 Main Parameters of the Field Shaping Coils

|                                           |                                   |         |   |         |   |          |   |                   |
|-------------------------------------------|-----------------------------------|---------|---|---------|---|----------|---|-------------------|
| Magnet:                                   |                                   | FSC8    | / | FSC7    | / | FSC6     | / | FSC5              |
| <u>Geometric Features</u>                 | <u>Unit</u>                       |         |   |         |   |          |   |                   |
| Center of magnet: z-position <sup>+</sup> | m                                 | ±17.35  |   | ±18.6   |   | ±19.8    |   | ±20.8             |
| Minor radius                              | m                                 | 1.5     |   | 1.5     |   | 1.0      |   | 1.0               |
| Major radius                              | m                                 | 1.7     |   | 1.7     |   | 1.05     |   | 1.2               |
| Average radius                            | m                                 | 1.6     |   | 1.6     |   | 1.025    |   | 1.1               |
| Bundle cross section                      | mxm                               | 0.8x0.2 |   | 0.4x0.2 |   | 0.4x0.05 |   | 0.4x0.2           |
| Mean turn length                          | m                                 | 10.06   |   | 10.06   |   | 6.44     |   | 6.92              |
| Volume of winding                         | m <sup>3</sup>                    | 1.61    |   | 0.81    |   | 0.13     |   | 0.553             |
| Mean density                              | t/m <sup>3</sup>                  | ~ 8     |   | ~ 8     |   | ~ 8      |   | ~ 8               |
| Weight of winding                         | t                                 | ~ 12.87 |   | ~ 6.44  |   | ~ 1.03   |   | ~ 4.43            |
| <u>Electromagnetic Features</u>           |                                   |         |   |         |   |          |   |                   |
| Bloc current density                      | A/cm <sup>2</sup>                 | 650     |   | 480     |   | 450      |   | -795 <sup>*</sup> |
| Ampere-turns                              | 10 <sup>-6</sup> A-turns          | 1.04    |   | 0.384   |   | 0.09     |   | 0.636             |
| Self-inductance per N <sup>2</sup>        | 10 <sup>-6</sup> H/N <sup>2</sup> | 4.161   |   | 5.154   |   | 3.111    |   | 3.037             |
| Stored self-energy                        | MJ                                | 2.25    |   | 0.38    |   | 0.013    |   | 0.614             |
| Consumed poer                             | MW                                | 2.43    |   | 0.666   |   | 0.094    |   | 1.25              |
| <u>Operating Features</u>                 |                                   |         |   |         |   |          |   |                   |
| Operating temperature                     | K                                 | room    |   | room    |   | room     |   | room              |
| Max. working strain                       | %                                 |         |   |         |   |          |   |                   |
| Max. working stress                       | MPa                               | < 210   |   | < 210   |   | < 210    |   | < 210             |
| <u>Conductor Properties</u>               |                                   |         |   |         |   |          |   |                   |
| Conductor                                 | Hardened copper                   |         |   |         |   |          |   |                   |
|                                           | with                              |         |   |         |   |          |   |                   |
| Structure material                        | $\rho \sim 2.5 \times 10^{-8}$ cm |         |   |         |   |          |   |                   |

\* Inverse current direction

+ ± means one magnet on the positive z-axis and one on the negative z-axis

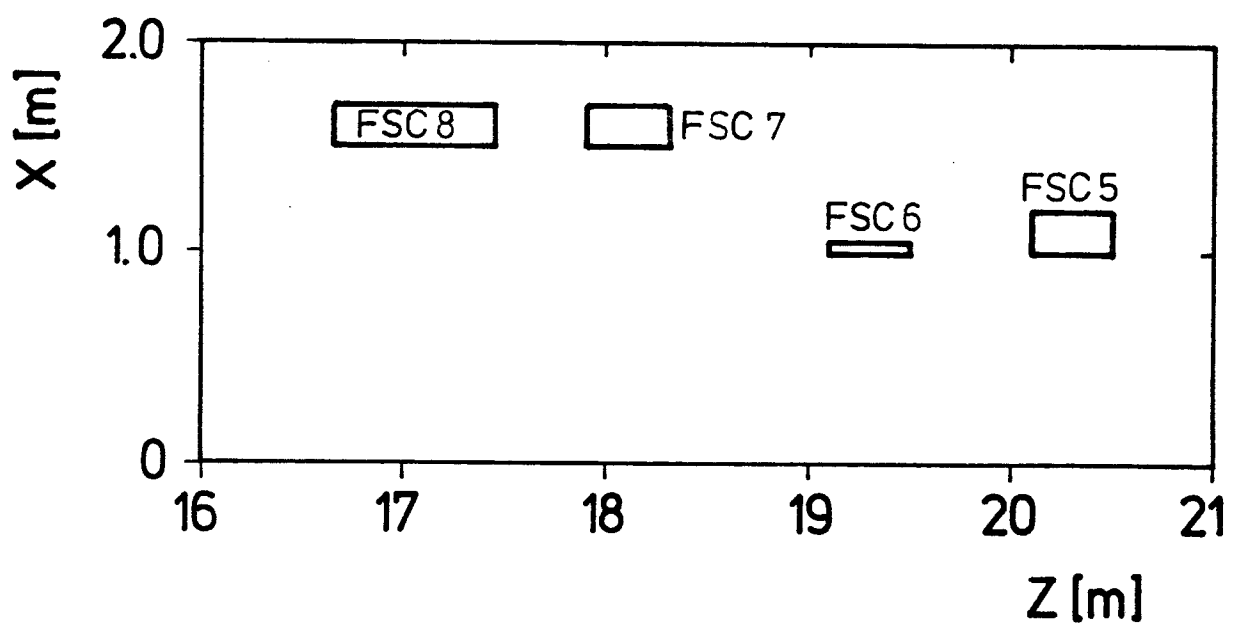


Fig. V.5-1 Position of field shaping coils.

## V.6 Magnetic Shielding

Shielding of the intense magnetic fields in TASKA must be considered in view of a reduction of magnetic pollution of the environment and in view of provision of regions of reduced magnetic field for the operation of machine components, in particular of ion sources and ion beam paths in the neutral beam injectors (see Section IV.3.2.8). Reduction of magnetic pollution, a problem common to all devices of magnetic fusion,<sup>(1)</sup> is not considered here because no conclusive information on exposure levels is available and the means for exposure reduction (distance, bucking coils, ferromagnetic shields) are known and developed.

Magnetic shielding of vital TASKA components in intense stray fields is exemplified by the ion source for the high energy pump beams, situated just outside the 20 T-barrier coils. As shown in Fig. V.3-1, fields in excess of 2 T must be shielded from a volume of about  $10 \text{ m}^3$ . The beam sources are designed to allow a thickness of 0.2 m around for shielding. Ferromagnetic materials of saturation flux densities of the order of 2 T are not sufficient for this task, although they may be used around the most sensitive parts, i.e. ion source and beam line to the neutralizer, if the field intensity is reduced at these points by bulk shielding. The only conceivable method at present appears to be bulk shielding by superconductors, using their perfect diamagnetic properties. Some experimental evidence<sup>(2,3)</sup> with shielding tubes of about 2 m length and several cm bore is available, indicating that fields of about 2 T can be shielded by stacks of superconductors and stabilizing normal conductors of about 1 cm thickness. In order to permit a free flow of the shielding currents, wide superconductor foils are generally used. This technique is also proposed for superconducting magnetic shields studied in the field of magnetic levitation.<sup>(4)</sup> In experiments on the inverse problem, i.e.,

trapping of magnetic fields by superconductors,<sup>(5)</sup> good results were also achieved with helically wound Nb<sub>3</sub>Sn ribbons. However, homogeneous fields were trapped in this case, and the results may not be transferrable to the inhomogeneous shielding problem in TASKA. Several computer programs, based on iteration and the definition of current elements in a superconducting shield, were developed to predict the shielding capability and current distribution in conjunction with the critical current properties of the superconductor.<sup>(6,7)</sup>

In summary, available results on shielding properties of superconductors suggest their applicability for shielding the neutral beam sources in TASKA. However, experiments are needed to demonstrate efficient large volume shielding and the dependence of such shielding on superconductor thickness. Field leakage into the source region will depend on the width of foils of superconductors that can be manufactured. However, the leakage fields will be shieldable by conventional methods at the most sensitive places.

Local shielding of neutral beam generators will distort the overall magnetic flux pattern. Again, the region of the barrier coils around the high energy pump beam generators is most sensitive because the shielding requirements are high and the axisymmetry of the vacuum magnetic field on axis is distorted in a plasma region in which the central cell particles are reflected. A plasma physics analysis is needed to determine the level of non-axisymmetry tolerable in this region. One cure for this problem might be provision of axisymmetric, ferromagnetic material around the barrier coil. It would reduce the flux density at the neutral beam generator and, conversely, shield the plasma from flux distortions due to the neutral beam shield. Magnetic pollution from TASKA would be reduced. Accurate magnetostatic calculations are complicated by the high level of saturation at which the iron would operate.



At present, no final solution to the shielding problem for the high energy pump beam can be given within the spatial limitations assumed. If experimental results can be scaled-up from small dimensions, stacked superconducting foils can be used. If no solution appears at hand, the spatial limitations must be relaxed, e.g., by extending the barrier length, thus displacing the neutral beam generator further from the barrier coil and, therefore, decreasing the stray fields. Even now, the design barrier length of TASKA with the need for bucking coils and the two-stage pumping scheme proposed for the barrier are consequences of a credible design for the high energy pump beam with short duct length.

#### References for Section V.6

1. INTOR-Report, Conceptual Design Phase I, Chapter XVI.5.3, July 1981, IAEA, Vienna.
2. F. Martin and S.J. St. Lorant, J. Appl. Phys. 44 (1973), 460.
3. E.U. Haebel and W. Witzeling, Proc. 6th Int. Conf. Magn. Technol., Bratislava, Aug. 30 - Sept. 2, 1977, Vol. II, p. 349.
4. H. Joyce, et al., IEEE Trans. Magn. 17 (1981), 2154.
5. M. Rabinowitz, et al., Appl. Phys. Lett. 30 (1977), 607.
6. W. Witzeling, Cryogenics 16 (1976), 29.
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## V.7 Superconductors for TASKA

### V.7.1 Introduction

To guarantee reliable and safe operation of the TASKA magnets, a number of design criteria for the conductors have to be met:

- The conductor must sustain the mechanical stresses due to magnetic forces. Each turn should sustain its hoop stresses by suitable addition of reinforcing material.
- The conductor must recover from small disturbances like conductor movement, plastic deformation of material, etc., i.e., cryogenic stabilization is essential for magnets of this size.
- In the case of a nondisappearing normal zone, the magnet has to be discharged safely without irreversible damage by excess voltage or temperature.

These requirements have been combined in a design procedure which is outlined in Fig. V.7-1. The components of this procedure are now described.

### V.7.2 Stresses

The hoop stresses in a self supporting winding of a solenoid are given by

$$\sigma_r = \frac{I_0 (B \cdot r)_{\max}}{A_r} \quad (V.7-1)$$

where

|            |                                       |
|------------|---------------------------------------|
| $A_r$      | Cross sectional area of reinforcement |
| $\sigma_r$ | Stress in reinforcing material        |
| $B$        | Magnetic field in axial direction     |
| $I_0$      | Operating current                     |

## CONDUCTORS FOR TASKA

### DESIGN PROCEDURE

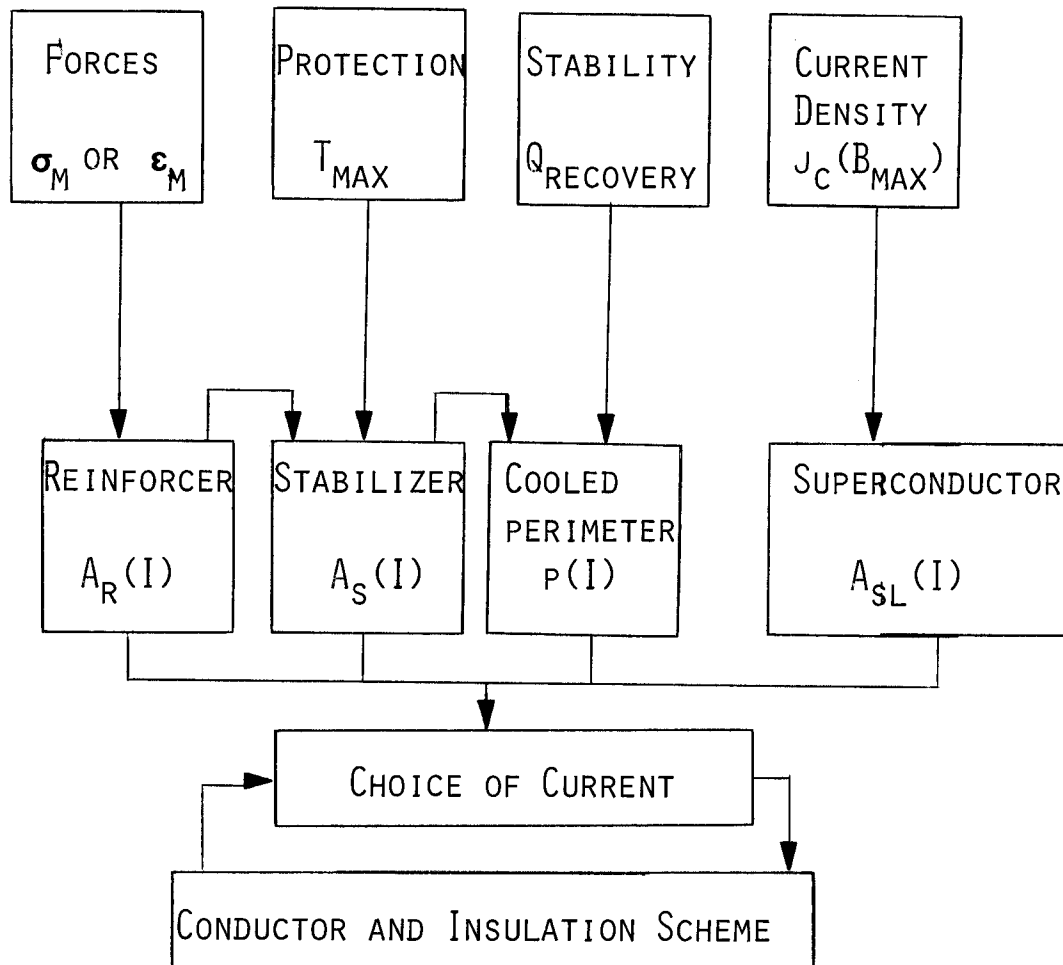


Fig. V.7-1 Design procedure for TASKA conductors.

r Radius

$(B \cdot r)_{\max}$  The maximum value in the coil. In the TASKA solenoids the maximum is attained with the field  $B_{\max}$  at the inner turn with radius  $r_{\min}$ .

With stainless steel chosen as reinforcing material and a maximum allowable strain of .24% in accordance with the strain capability of  $Nb_3Sn$ , a stress

$$\sigma_{\max} = 500 \text{ MPa}$$

is used as a design parameter.

### V.7.3 Fast Discharge

The maximum hot spot definition temperature is fixed at

$$T_{\max} = 200 \text{ K} .$$

It is given by the hot spot integral

$$A_s \int_{4.2 \text{ K}}^{T_{\max}} \frac{\gamma_s C_{ps}}{\rho_s} dT + A_r \int_{4.2 \text{ K}}^{T_{\max}} \frac{\gamma_r C_{pr}}{\rho_s} dT = \frac{Q_0}{U_0} \cdot \frac{I_0}{A_s} \quad (V.7-2)$$

$Q_0$  Stored energy

$U_0$  Maximum discharge voltage

$I_0$  Operating current

$\gamma$  Density of material

$C_p$  Specific heat

$\rho$  Specific resistivity

|       |                                               |
|-------|-----------------------------------------------|
| $A_s$ | Cross sectional area of stabilizer            |
| $A_r$ | Cross sectional area of reinforcing material. |

In this treatment the amount of stabilizing material is primarily determined by the hot spot temperature. The enthalpy of the reinforcing steel is utilized. This requires good thermal contact between these two components, but gives a higher mean current density. Aluminum is well suited because of its high electric conductivity in high magnetic fields.

The hot spot integrals  $\int \gamma_s (C_p / \rho) dT$  for the combinations stainless steel-aluminum, aluminum - aluminum and copper - copper have been calculated:

$$J(200 \text{ K}) = 6.5 \times 10^{16} \frac{\text{A}^2 \text{s}}{\text{m}^4} \text{ for Al-Al}$$

$$J(200 \text{ K}) = 13.3 \times 10^{16} \frac{\text{A}^2 \text{s}}{\text{m}^4} \text{ for Al-SS}$$

$$J(200 \text{ K}) = 9.7 \times 10^{16} \frac{\text{A}^2 \text{s}}{\text{m}^4} \text{ for Cu-Cu.}$$

The maximum discharge voltage  $U_0$  is set to 2.5 kV in the barrier coil and to 1 kV in the other coils. After a quench is detected in one coil, a fast discharge is initiated in this coil alone while the others are discharged slowly or remain at the nominal current.

#### V.7.4 Cryogenic Stability

With the amount of stabilizer fixed by coil protection requirements, the condition of cryogenic stability must be fulfilled by a proper choice of cooled surface and cooling mode.

Pool boiling at 4.2 K is chosen as it combines simple cryogenics (no

pumps and pumping losses) with relatively high heat transfer.

For the cold end recovery heat flux from the conductor to helium, the measured value for samples of the MFTF yin-yang conductor was chosen

$$q_{\text{rec}} = 2000 \text{ W/m}^2 .$$

The resistivity of high purity aluminum is

$$\rho = 10^{-10} \text{ } \Omega\text{m} ,$$

which takes account of magnetoresistivity, strain and radiation induced resistivity up to  $10^{17} \text{ n/cm}^2$ .

For copper at 8 T,  $\rho = 5.3 \times 10^{-10} \text{ } \Omega\text{m}$  was used.

The cooled perimeter  $p$  is then determined by

$$\frac{I_0^2 \rho}{A_s} \lesssim p \cdot q_{\text{rec}} . \quad (\text{V.7-3})$$

#### V.7.5 Design Procedure

Equations V.7-1,2,3 constitute a system to determine in an unique manner the parameters  $A_s$ ,  $A_r$  and  $\rho$  as a function of the nominal current  $I_0$ . This dependence is shown in Fig. V.7-2 for the conductor employed for the central cell solenoids.

The overall current density in the conductor is only weakly dependent on the current level as the main fraction of the cross section in coils of this size is occupied by steel.

The proper choice of  $I_0$  is finally determined by an iterative procedure with the constraint that the required values of  $A_r$ ,  $A_s$  and  $\rho$  are compatible

# CC - CONDUCTOR GRADE I

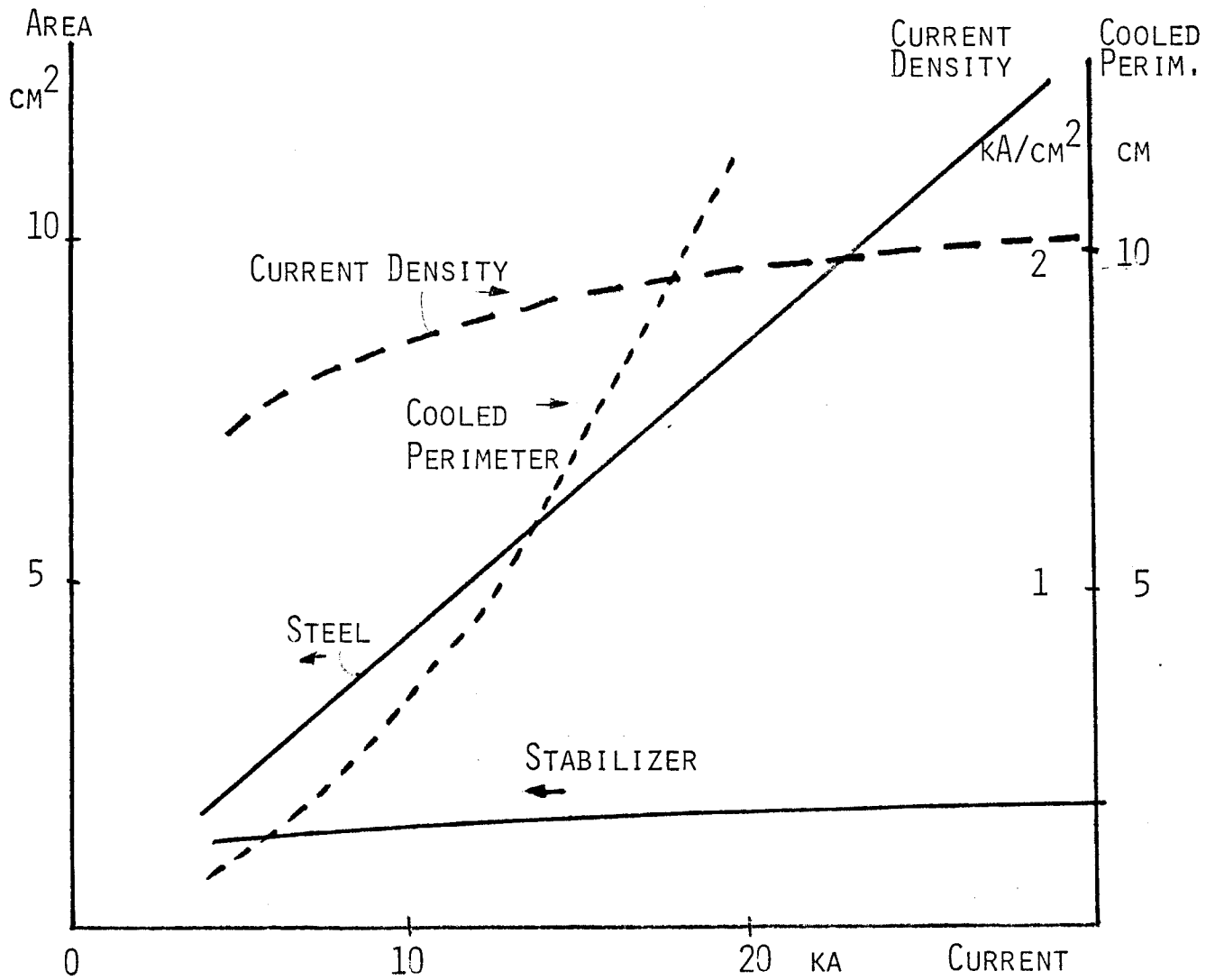


Fig. V.7-2 Conductor parameters as a function of current.

with requirements of fabrication, winding and insulation of the conductor.

A number of features have served as guidelines in the choice of a conductor concept for the solenoids in TASKA. These are listed below:

- Cooling on the sides of conductor
  - heat transfer rate independent of position in the coil
  - radial interturn forces easily transmitted
- No interturn forces transmitted to superconductor or stabilizer
  - source of degradation eliminated
  - no plastic deformation in stabilizer
- Metallurgical bond (solder) between superconductor, stabilizer and steel
  - source of frictional heating eliminated
  - enthalpy of steel utilized in hot spot protection (essential for large coils)
- Standard NbTi or Nb<sub>3</sub>Sn multifilamentary conductors as the basic unit.
- Margin for critical current  $j_{s_c} = \frac{2}{3} j_c (B_{\max})$ .

#### V.7.6 Conductor Parameters

This design procedure had led to the conductor principle shown in Fig. V.7-3. This is an example of the 10 kA conductor for the central cell solenoid. The design parameters for the solenoids are listed in Table V.7-1 and Table V.7-2. Table V.7-3 gives yin-yang and recircularizing coil parameters.

The same conductor scheme is applied to all solenoids. Aluminum stabilizer turns out to be essential in the barrier coil (copper would require



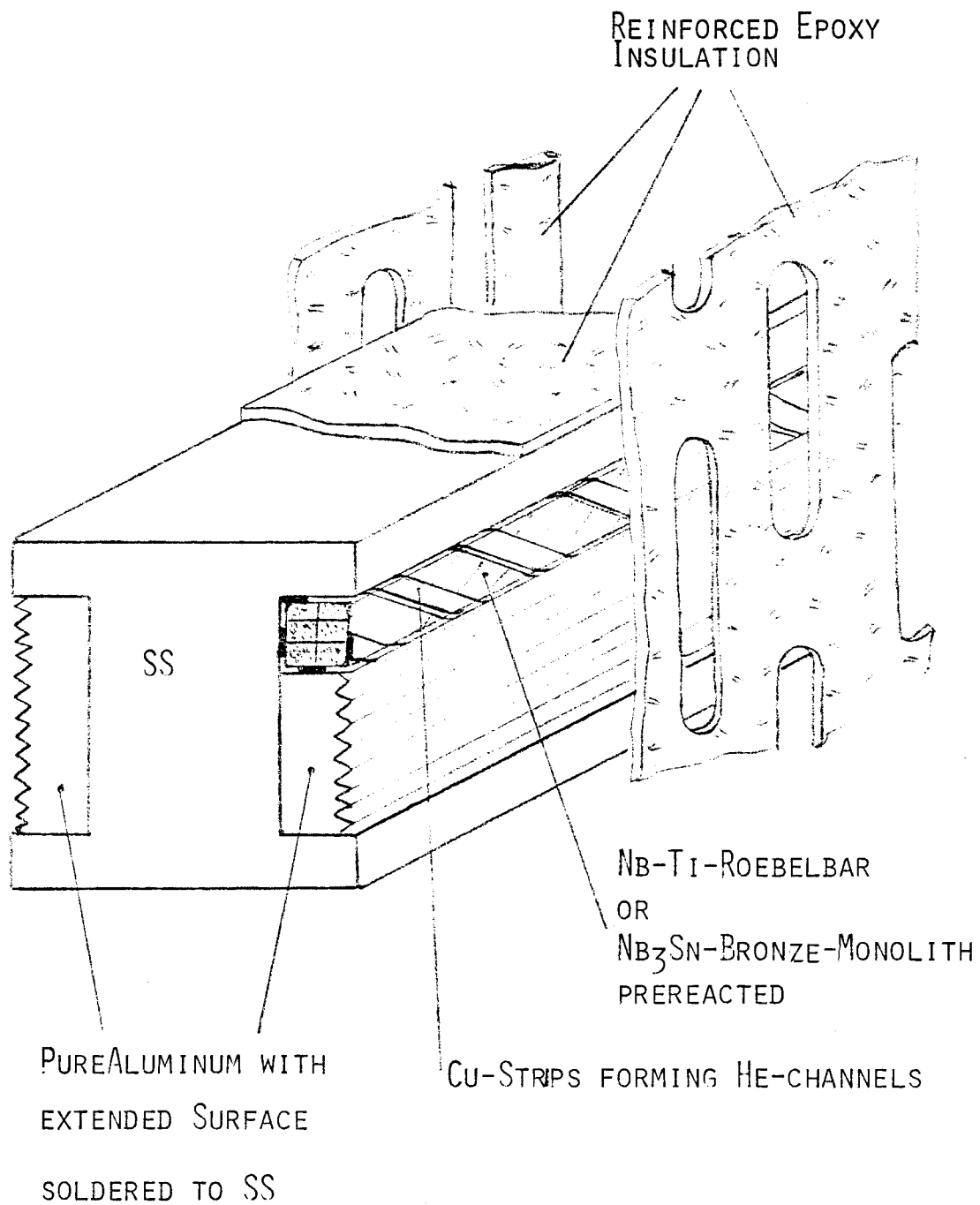


Fig. V.7-3 Schematic of Al-stabilized conductor for solenoids.

Table V.7-1 Conductor Parameters of Central Cell Solenoids

| Coil/Grade                      |                      | CC/1                | CC/2                |
|---------------------------------|----------------------|---------------------|---------------------|
| Maximum field                   | (T)                  | 6.0                 | 4.0                 |
| Minimum radius                  | (m)                  | 2.8                 | 3.1                 |
| Stored energy                   | (J)                  | 0.6x10 <sup>9</sup> |                     |
| Discharge voltage               | (kV)                 | 1.0                 |                     |
| Reinforcer/stabilizer           |                      | SS/Al               |                     |
| Superconductor                  |                      | NbTi                |                     |
| Critical current density in s/c | (A/cm <sup>2</sup> ) | 1.2x10 <sup>5</sup> | 1.8x10 <sup>5</sup> |
| Filament diameter               | ( $\mu$ m)           | 30-60               |                     |
| Operational current             | (A)                  | 10000               |                     |
| Overall j <sub>c</sub> in coil  | (A/cm <sup>2</sup> ) | 1500                | 1800                |
| Cross sectional areas           |                      |                     |                     |
| - Steel                         | (cm <sup>2</sup> )   | 3.4                 | 2.50                |
| - Aluminum                      | (cm <sup>2</sup> )   | 1.5                 | 1.6                 |
| - Superconductor                | (cm <sup>2</sup> )   | 0.125               | 0.08                |
| Cooled perimeter                | (cm)                 | 4.0                 | 4.0                 |

Table V.7-2 Conductor Parameters of Barrier Coil

| Coil/Grade                         |                      | LI1 | ML1                 | MR1                | LI2 | ML2               | MR2                 | L01                 | L02                 |
|------------------------------------|----------------------|-----|---------------------|--------------------|-----|-------------------|---------------------|---------------------|---------------------|
| Maximum field                      | (T)                  |     | 15                  |                    |     | 12.5              |                     | 8.5                 | 5.0                 |
| Minimum radius                     | (m)                  |     | 1.3                 |                    |     | 1.50              |                     | 1.86                | 2.0                 |
| Stored energy                      | (J)                  |     |                     |                    |     |                   | 2.0x10 <sup>9</sup> |                     |                     |
| Discharge voltage                  | (kV)                 |     |                     |                    |     |                   | 2.5                 |                     |                     |
| Reinforcer/<br>stabilizer          |                      |     |                     | SS/A1              |     |                   |                     | SS/A1               |                     |
| Superconductor                     |                      |     |                     | Nb <sub>3</sub> Sn |     |                   |                     | NbTi                |                     |
| Critical current<br>density in s/c | (A/cm <sup>2</sup> ) |     | 1.5x10 <sup>4</sup> |                    |     | 5x10 <sup>4</sup> |                     | 0.6x10 <sup>5</sup> | 1.5x10 <sup>5</sup> |
|                                    |                      |     |                     | (s/c + bronze)     |     |                   |                     |                     |                     |
| Filament diameter                  | (μm)                 |     |                     | 1-3                |     |                   |                     |                     | 30-60               |
| Operational current                | (A)                  |     |                     |                    |     |                   | 10000               |                     |                     |
| Overall j <sub>c</sub> in coil     | (A/cm <sup>2</sup> ) |     | 1200                |                    |     | 1400              |                     | 1600                | 2000                |
| Cross sectional areas              |                      |     |                     |                    |     |                   |                     |                     |                     |
| - Steel                            | (cm <sup>2</sup> )   |     | 3.9                 |                    |     | 3.9               |                     | 3.2                 | 2.0                 |
| - Aluminum                         | (cm <sup>2</sup> )   |     | 1.70                |                    |     | 1.7               |                     | 1.8                 | 2.0                 |
| - Superconductor                   | (cm <sup>2</sup> )   |     | 1.0                 |                    |     | 0.3               |                     | 0.25                | 0.1                 |
|                                    |                      |     |                     | (sc + bronze)      |     |                   |                     |                     |                     |
| Cooled perimeter                   | (cm)                 |     | 4.0                 |                    |     | 4.0               |                     | 3.0                 | 3.0                 |

Table V.7-3 Yin-Yang and Recircularizing Coils

| Coil                     |                      | Yin-Yang          | TC/RC             |
|--------------------------|----------------------|-------------------|-------------------|
| Maximum field            | (T)                  | 7.9               | 7.0               |
| Minimum radius           | (m)                  | 0.9               | 1.4               |
| Stored energy            | (J)                  | $1.3 \times 10^9$ | $1.0 \times 10^9$ |
| Discharge voltage        | (kV)                 | 1.0               |                   |
| Reinforcer/stabilizer    |                      | Cu (3/4 hard)     |                   |
| Superconductor           |                      | NbTi              |                   |
| Critical current density | (A/cm <sup>2</sup> ) | $5 \times 10^4$   |                   |
| Filament diameter        | ( $\mu$ m)           | $\approx 200$     |                   |
| Operational current      | (A)                  | 4000              |                   |
| Cross sectional areas:   |                      |                   |                   |
| - Steel                  | (cm <sup>2</sup> )   | -                 |                   |
| - Copper                 | (cm <sup>2</sup> )   | 2.7               | 2.7               |
| - Superconductor         | (cm <sup>2</sup> )   | 0.12              | 0.12              |
| Max. hoop stres          | (MPa)                | 105               | 145               |
| Cooled perimeter         | (cm)                 | 2.3               | 2.3               |

a cooled perimeter ten times larger, a requirement not compatible with a simple compact conductor). The technology of Al-stabilization is also used with profit in the NbTi coils.

For the recircularizing coils and yin-yang coils, the principle of the MFTF-B conductor was adopted due to lower stress levels and the necessity to bend the conductor on two axes. Stabilization and reinforcement is combined here in the conductor jacket of 3/4 hard copper.

To increase the overall current density and to lower costs, a twofold graded conductor is used in the Nb<sub>3</sub>Sn- as well as NbTi-part of the barrier coil. Grading of the CC-coils has only a minor impact on the overall current density according to Table V.7-1. Using the grade 1 conductor throughout will save splices in the coil so this simpler version is recommended.

The overall current densities in Tables V.7-1 and 2 are obtained with a filling factor of 0.8 to allow for insulation and helium passages.

In order to manufacture the TASKA magnets, development is still required in two areas:

- Aluminum stabilization; soldering to SS and superconducting strands
- Nb<sub>3</sub>Sn technology for large coils and high fields.

## V.8 Coil Support

With regard to magnetic forces, the TASKA coil system can be divided into three independent regions: the central cell (including the barrier coils) and the two plug regions. The forces between these regions are negligible compared to those which appear within each one. Therefore, each region can be considered as an isolated system. The forces which act on the field-shaping coils located between the plug region and the central cell are relatively small. These coils can be supported by the shielding structure and don't need a special support system.

### V.8.1 Central Cell

As all coils of the central cell and barrier are solenoids with a common axis - the z-axis - the body forces acting on each coil are axisymmetric to the z-axis and the resulting intercoil forces lie on the z-axis. Under normal operation, this means that, if all coils are completely energized, the resulting forces are also symmetric with respect to the center of the central cell. Fig. V.8-1 shows the resulting forces in MN acting on the coils under normal operating conditions. Because of the symmetry, the resulting force on the central coil is zero. The maximum forces which have to be supported are between the barriers and the outer central cell coils; these forces are 51 MN each. As the support structure must be laid out for the maximum forces likely to be encountered, abnormal operation must be investigated, i.e., when one or more coil(s) are not energized. The maximum force between the coil in the center and one of the outer central cell coils appears when the other outer central cell coil is not energized. The resulting force is 45 MN. This is shown in Fig. V.8-2. All these forces have been calculated with the "HEDO" code.<sup>(1)</sup>

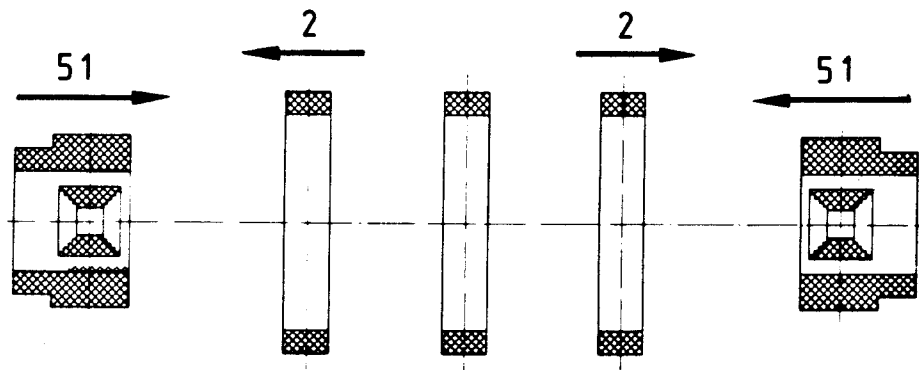


Fig. V.8-1 Forces on central cell coils and barrier coil under normal operating conditions (forces in MN).

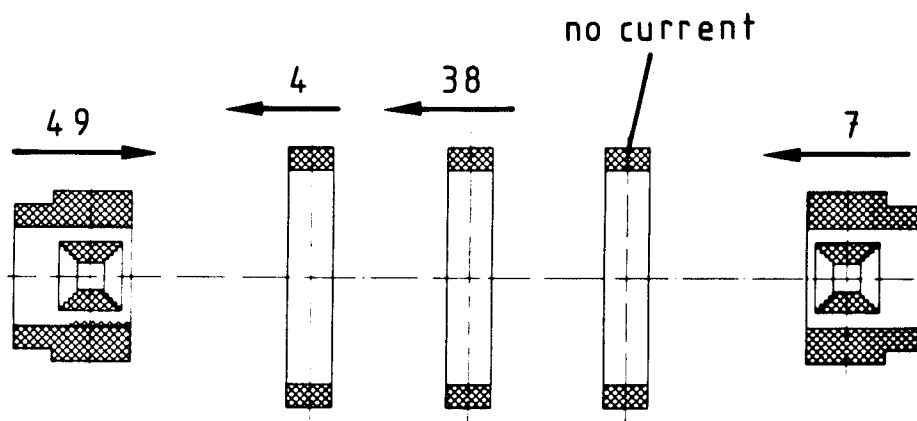


Fig. V.8-2 Forces on central cell coils and barrier coil with one coil not energized (forces in MN).

Fig. V.8-3 shows the main parts of the coil support structure. They consist of:

- the annular elements A which bear the load from the whole barrier region,
- twelve inclined supporting struts B between each barrier and the outer central cell coils,
- twelve horizontal supporting struts D between each outer central cell coil and the coil in the center,
- the casing C for the three central cell coils. The reverse of each casing consists of thick cylindrical rings which protect the conductor of the coil against the load exerted by the struts from both sides.

The supporting struts are exposed only to compressive load. The maximum compressive stress is only 36 MPa. This is a very low value which guarantees a good stiffness of the whole supporting structure.

#### V.8.2. Plug Region

Fig. V.8-4 shows the plug region which consists of the two outer transition coils and the two yin-yang coils. The whole plug region has mirror symmetry with respect to the x-z plane and to the y-z plane. It also has axisymmetry with respect to the axis w which is normal to the z-axis and has an angle of 45 degrees with respect to the x- or y-axis.

The forces which appear are all mirror-symmetrical to the x-z plane and to the y-z plane. That means that the arcs which are designated in Fig. V.8-4 with index "2" are mirror-symmetrically loaded to those which are designated with index "1". Thus there are no resulting moments on the coils and the resulting forces lie on the z-axis. If all coils are completely energized



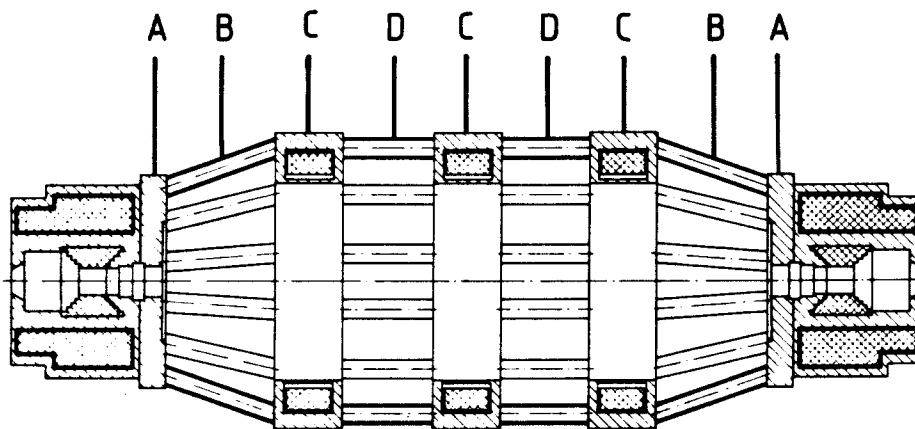


Fig. V.8-3 Main components of central cell support structure.

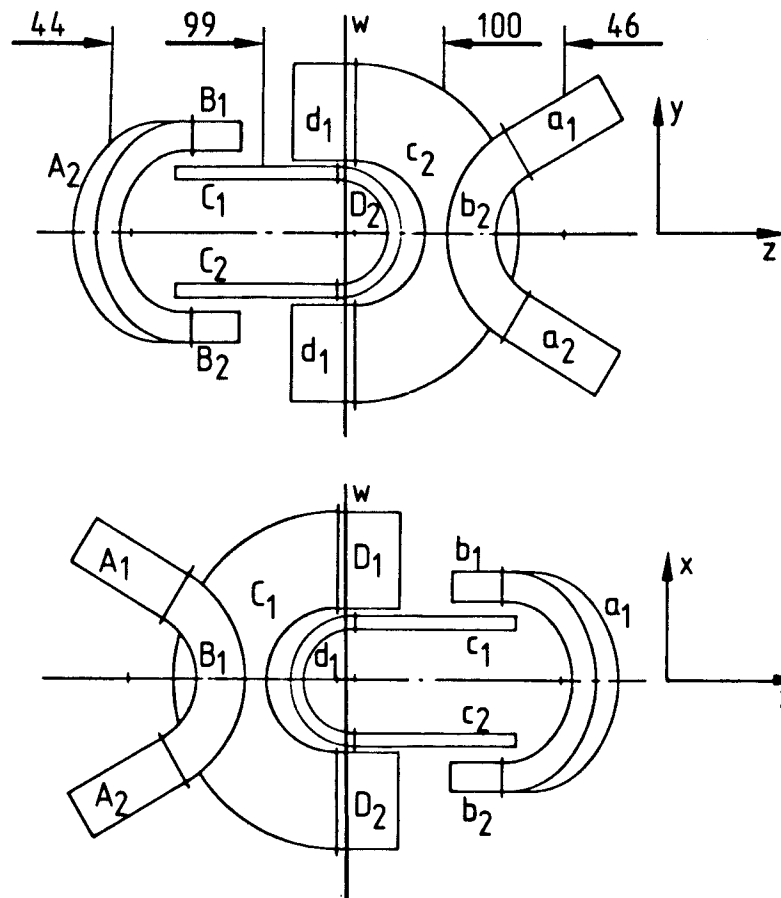


Fig. V.8-4 Subdivision of plug coils into single arcs and resulting forces on the coils under normal operation.

then the forces are also symmetric about the w-axis. That means that the arcs which are designated in Fig. V.8-4 with capital letters have the same mechanical load as the arcs designated with small letters.

Fig. V.8-4 shows also the resulting forces (calculated) with the "EFFI"-code<sup>(2)</sup> for each coil under normal operation. The maximum force which must be supported is between the two yin-yang coils with an amount of 99 MN; this is about a factor of two higher than in the central cell. The forces are nearly symmetrical with respect to the w-axis. Only a small resulting force of 3 MN, due to the influence of the central cell coils and the field shaping coils on the plug region, makes a small distortion of symmetry with a maximum of 5%. Therefore, for further investigations the plug region can be treated as symmetrically mechanically loaded.

Figs. V.8-5 to V.8-8 show the body forces calculated by the "EFFI"-code for each arc under normal working conditions. Each arc has been divided every 10 degrees into 12 to 18 sections and for any section the resulting force has been calculated. These forces are drawn as histograms over the angle of the arcs in Figs. V.8-5 to V.8-8. The highest force on one element is 22 MN, which results in a load of 52 MN/m along the mean length of the arc.

The support structure of the end plug coils is shown in Fig. V.8-9. In order to obtain a high stiffness, the coil casings are supplied with a large number of ribs. All the coils are supported by each other. The two yin-yang coils are fixed together by their outer ribs (A). Each transition coil is fixed to both yin-yang coils by many cover plates (B). The large arcs of the yin-yang coils are held together by 6 large tie rods (C) at each coil. Two of them (D) are also fixed to the transition coils. These tie rods bear the main part of the forces which try to bend apart the coil arcs.

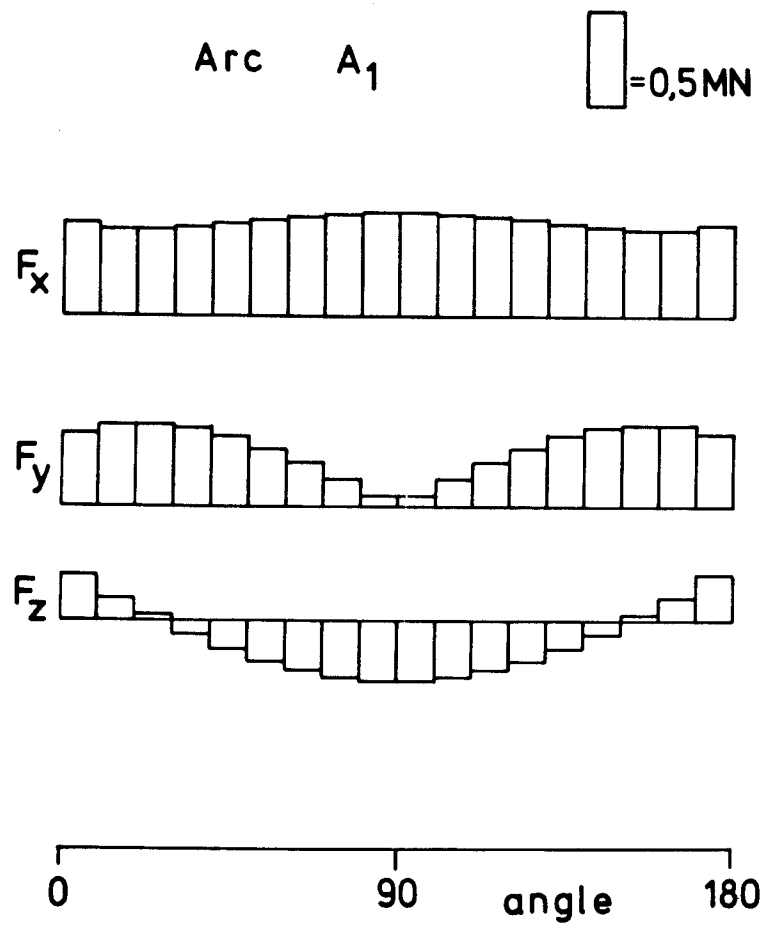


Fig. V.8-5 Forces on arc no.  $A_1$ .

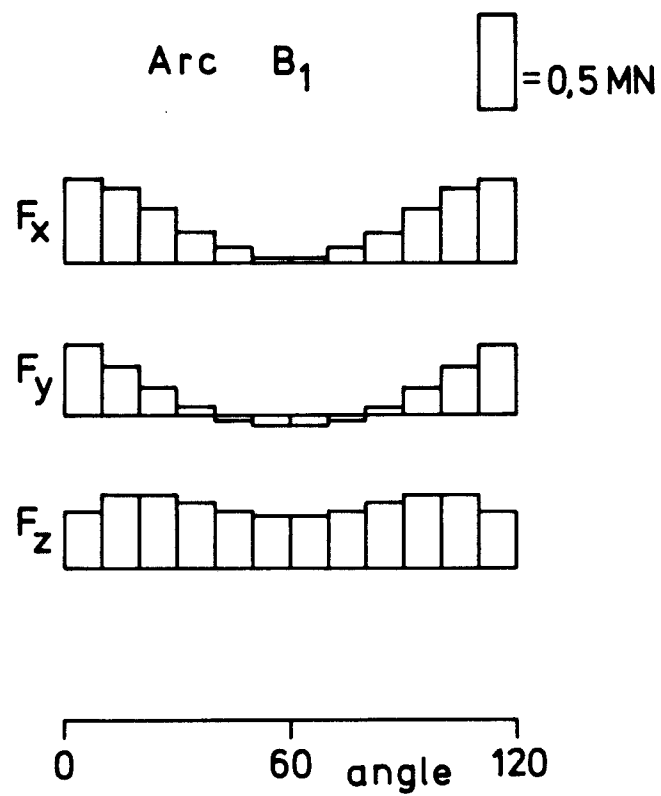


Fig. V.8-6 Forces on arc no.  $B_1$ .

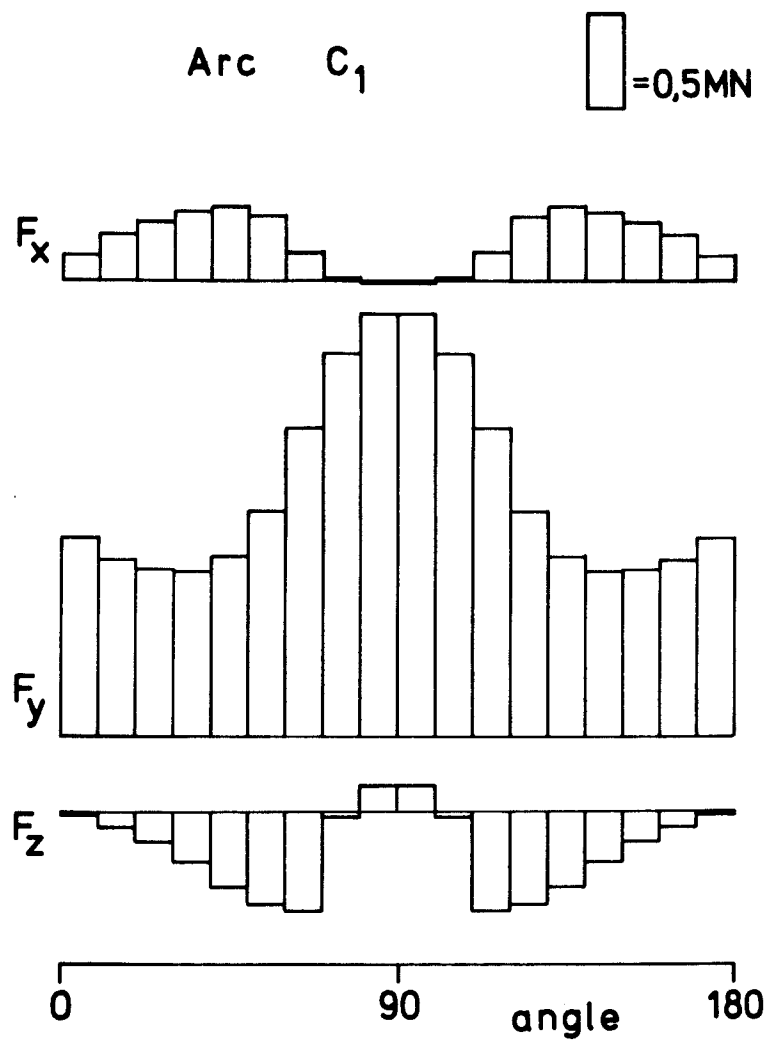


Fig. V.8-7 Forces on arc no.  $C_1$ .

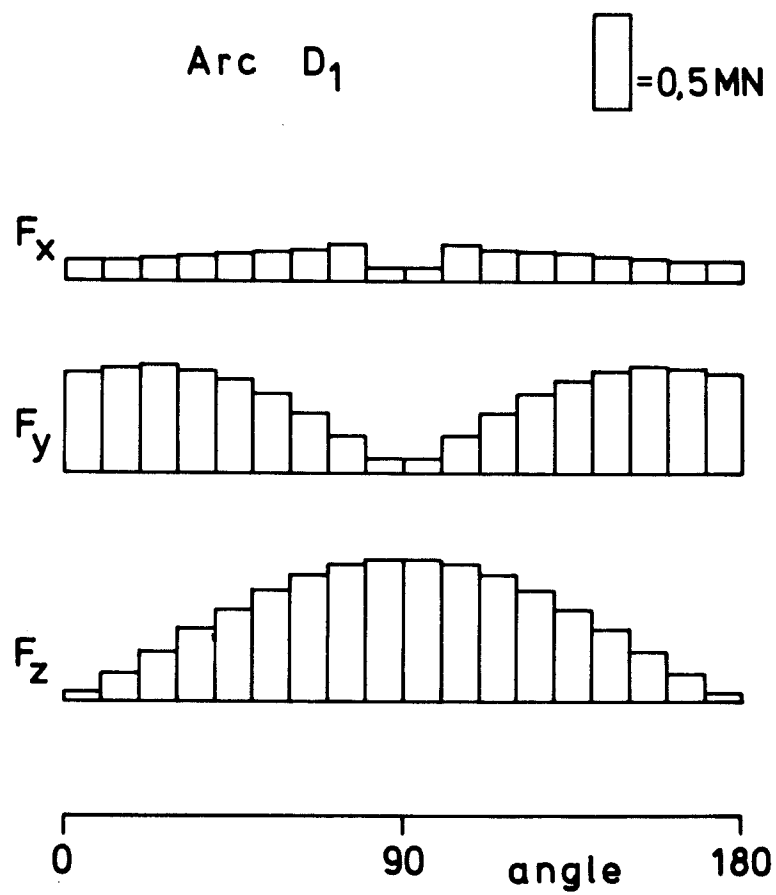


Fig. V.8-8 Forces on arc no.  $D_1$ .

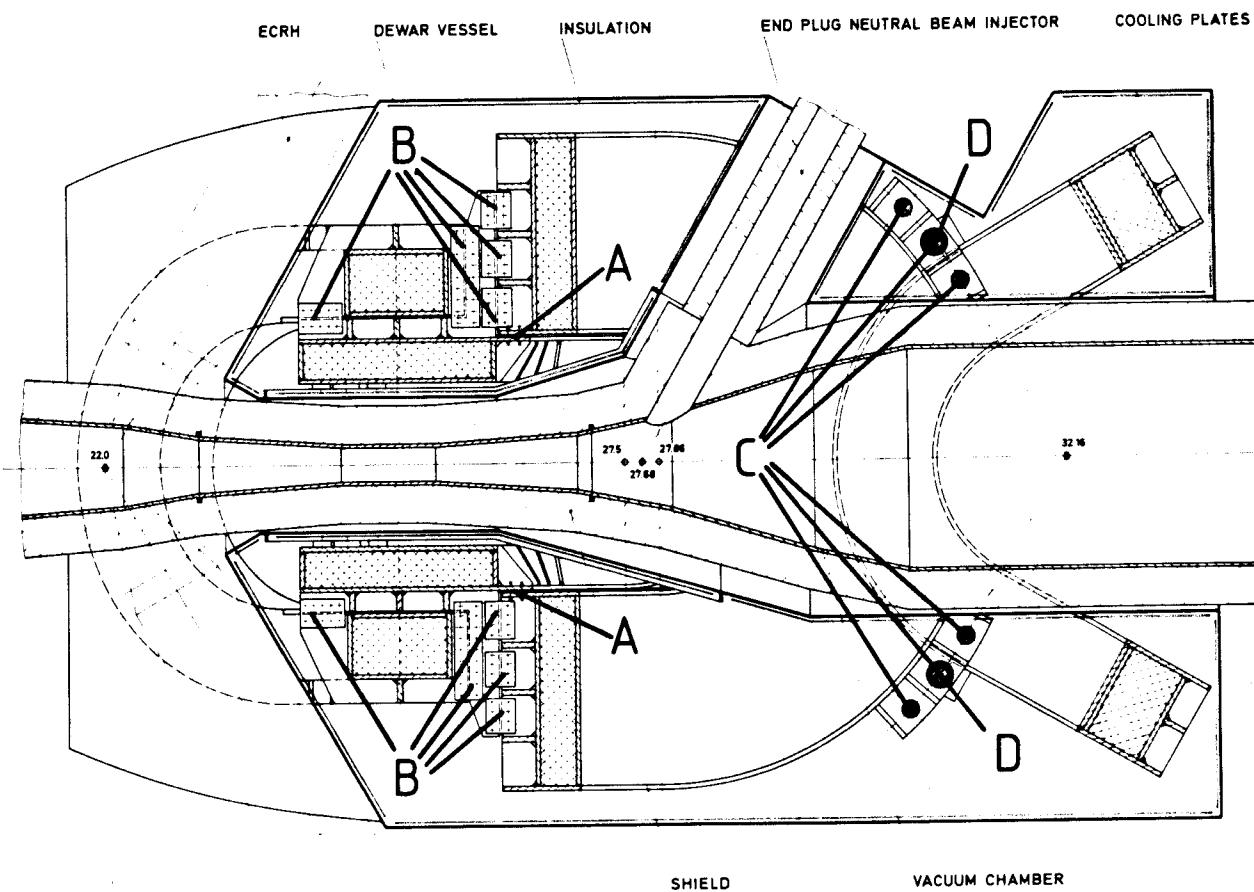
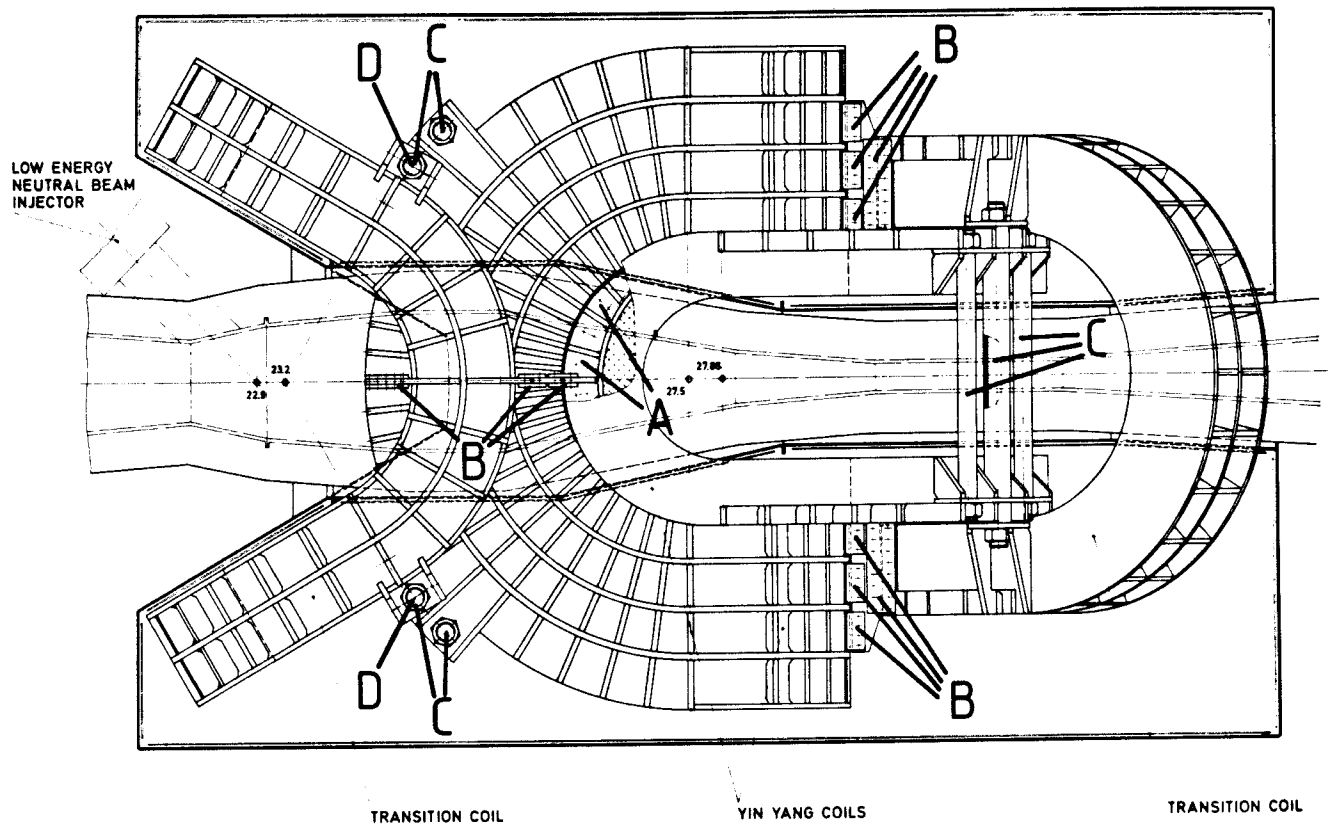


Fig. V.8-9: Support structure of the end plug coils



#### References for Section V.8

1. H. Preis, "Calculation of the Magnetic Field, Magnetic Forces and Behavior of Large Coil Systems for Fusion Experiments", IPP III/24, Max Planck Institut fur Plasmaphysik.
2. S.J. Sackett, "EFFI, A Code for Calculating the Electromagnetic Field, Force and Inductance in Coil Systems of Arbitrary Geometry", Lawrence Livermore Laboratory, Livermore, CA, UCRL-52402 (1978).

## V.9 The Impact of the Magnetic Properties of Ferritic Steels

### V.9.1 Introduction

The use of the martensitic HT-9 stainless steel in the central cell blanket modules has considerable advantages because of its high resistance to radiation damage (see materials section). However, because of the fact that HT-9 is a ferromagnetic material, its magnetic aspects should be considered carefully in the magnet design and in the blanket design of TASKA. In the following we will discuss the magnetostatic forces in the blanket and the magnetic field asymmetry which exists in the central cell as a result of the presence of the ferromagnetic blanket, and which may have an effect on the plasma confinement.

The 3-D computer program GFUN3D<sup>(1)</sup> has been used in the following study. First we will consider a necessary model used in GFUN to carry out this study.

### V.9.2 Homogenization Model

The computer code GFUN solves for the magnetization in the iron which must be divided into small elements. These elements can be prisms and/or tetrahedra. To achieve accuracy the aspect ratio of any element should not exceed 5:1. A maximum of 400 elements was allowed in the original version of the program. If  $n$  is the number of elements, then  $(3n)$  linear equations are solved by the program. The computer storage needed for such a problem is slightly larger than  $(3n)^2$  computer words.

If one tries to simulate the very fine details of the blanket, tube by tube, one would exceed, very quickly, the allowable computer storage. Consider, for example, a hypothetical case of a tube of  $l$  cm length,  $d$  cm inner diameter, and  $t$  cm thickness. This tube can be simulated by hexahedra fitting its wall. Each hexahedron will be divided into a minimum of five tetrahedra.

Abiding by the maximum aspect ratio allowed, the minimum number of hexahedra is estimated as  $\sim \frac{\pi d}{(5t)} \times \frac{\ell}{(5t)} = \frac{\pi}{25} \frac{d\ell}{t^2}$ , and the minimum number of tetrahedra (elements) as  $\frac{\pi}{5} \frac{d\ell}{t^2}$ . If  $d = 10$ ,  $\ell = 100$ , and  $t = .7$  cm, then the minimum number of elements is 1283.

Although GFUN has been modified to make use of the large computer storage of some modern computers, to accept a greater number of elements, the computer time needed to solve such a problem would neither be reasonable nor acceptable.

To avoid this problem a homogenization model has been used in which the iron is homogenized throughout the blanket. This means that the magnetization, the magnetic dipole density,  $\bar{M}$ , of the iron within a region of the blanket is distributed uniformly within that region, and this region is considered as a section of a ferromagnetic material with magnetization  $\bar{M}_b$  equal to  $M$  multiplied by the volume concentration of the iron in it. For example, if the B-H curve of the iron is as in Fig. V.9-1, then the B-H curve of a homogenized region containing 44% iron (volume ratio) will be as in Fig. V.9-2.

One would expect that the magnetic field calculated using this scheme to be precise everywhere except within the blanket itself, and especially when the iron is saturated, which will be the case in the central cell.

To check the validity of this model we present the magnetic field mapping for two cases. In both cases a 10 meter long solenoid with inner radius of 2.5 m, outer radius of 3 m, and current density of  $400 \text{ A/cm}^2$  is used to generate the magnetic field. In the first case, the iron region consists of three rows of circular rings. The width of each ring is 6 cm, and the axial distance between the rings (edge to edge) is also 6 cm.

MATERIAL CODE- 3

| B             | H       |
|---------------|---------|
| - 1) 0.0      | 0.0     |
| - 2) 10000.0  | 10.0    |
| - 3) 12000.0  | 20.0    |
| - 4) 14000.0  | 60.0    |
| - 5) 16000.0  | 160.0   |
| - 6) 17000.0  | 350.0   |
| - 7) 18000.0  | 600.0   |
| - 8) 19000.0  | 1200.0  |
| - 9) 20000.0  | 2300.0  |
| - 10) 40000.0 | 22300.0 |

# B/H CURVE OF IRON

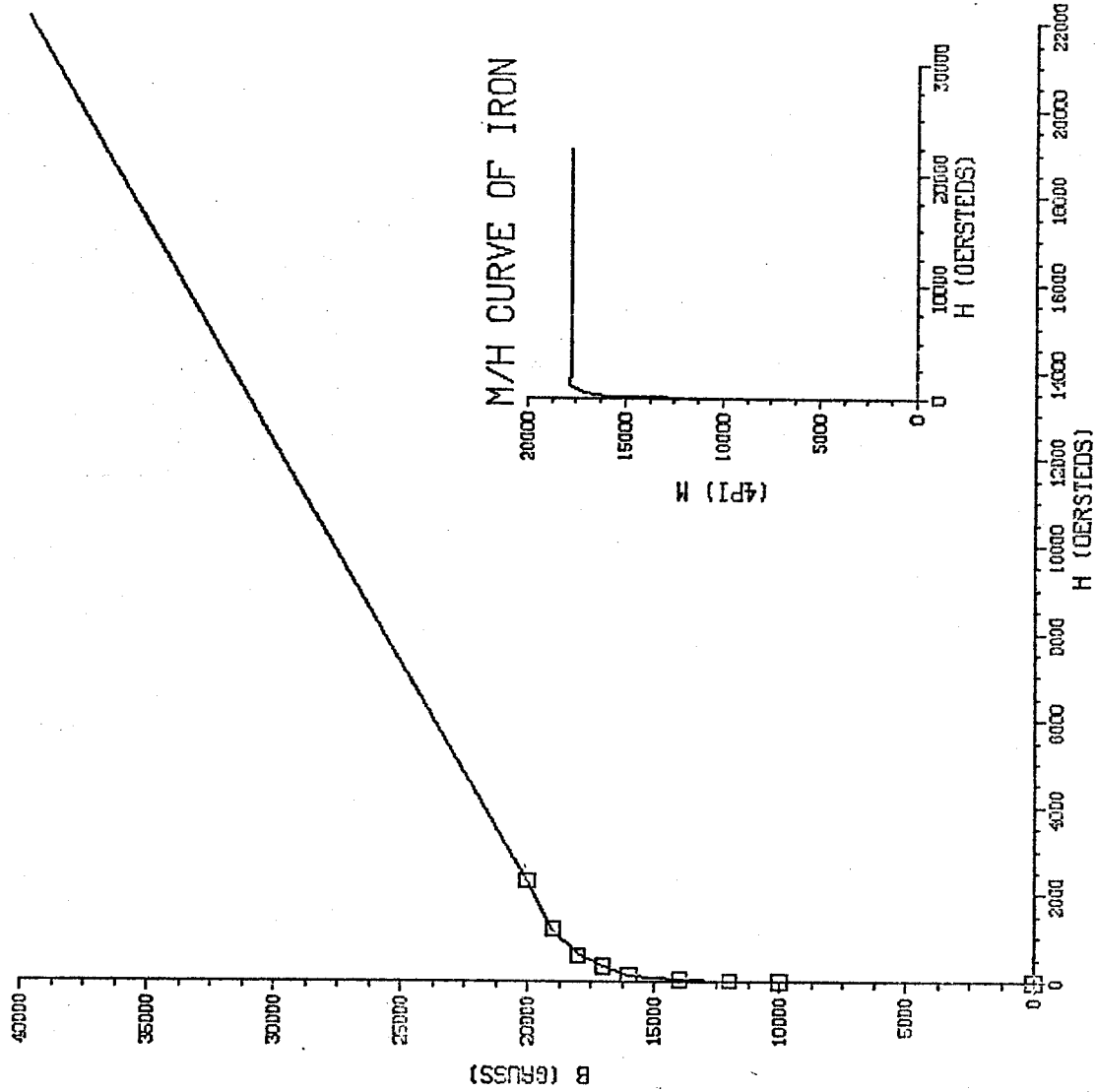


Fig. V.9-1 A typical B-H curve for iron.

FRAME 4, GFUN LDSN1 DATE 11/30/81 TIME 00:21:08  
 MATERIAL CODE- 3 VOL. (%) - 44.48

|       | B       | H       |
|-------|---------|---------|
| - 1)  | 0.0     | 0.0     |
| - 2)  | 4453.2  | 10.0    |
| - 3)  | 5348.3  | 20.0    |
| - 4)  | 6260.0  | 60.0    |
| - 5)  | 7205.1  | 160.0   |
| - 6)  | 7755.3  | 350.0   |
| - 7)  | 8338.9  | 600.0   |
| - 8)  | 9116.8  | 1200.0  |
| - 9)  | 10172.3 | 2300.0  |
| - 10) | 30172.3 | 22300.0 |

# B/H CURVE OF IRON

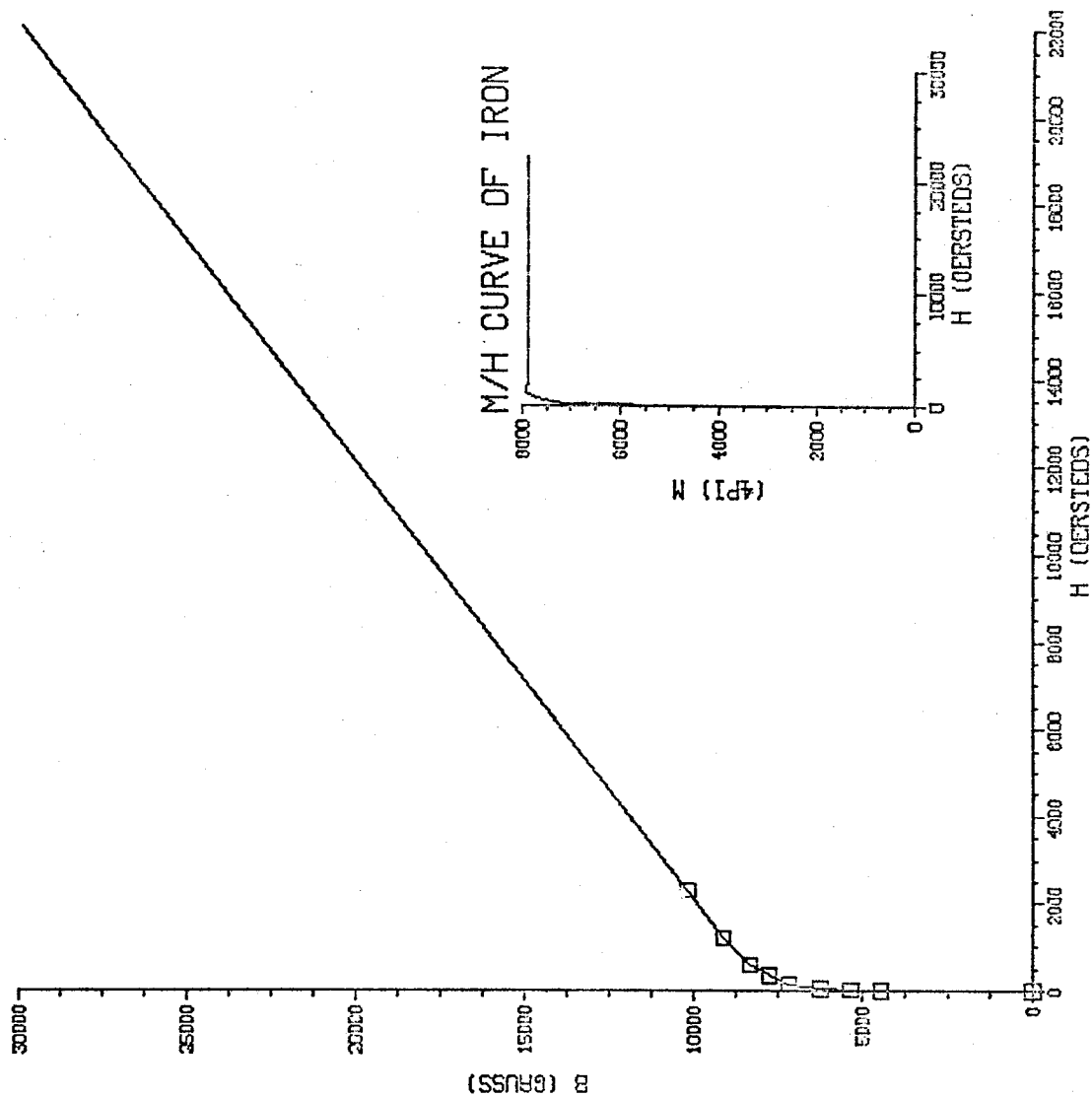


Fig. V.9-2 The B-H curve modified for homogenized material containing 44.5% iron.

The starting point of each row, the inner and outer diameters of its rings in cm are as follows: -123, 40, 45; -117, 46, 51; and -123, 52, 57, for the first, the second and the third row, respectively. In the second case, the homogenized case, the iron of the first case is assumed to uniformly fill a cylinder of 40 cm inner diameter and 57 cm outer diameter, and to extend axially from -123 to 123 cm. The volumetric ratio of the iron in this cylinder is .44, which was multiplied by the magnetization of the iron of the first case to produce new M-H and B-H curves for this homogenized case (Figs. V.9-1 and V.9-2).

The magnetic fields calculated in both cases are mapped and shown in Figs. V.9-3 and V.9-4 for the first and the second case, respectively. Note the fields are in kilogauss and are due only to the iron in both cases, i.e., the field of the solenoid is not included. One can see that the homogenized case has produced an almost identical field everywhere except near the iron region.

### V.9.3 The Magnetostatic Forces

#### V.9.3.1 One-Dimensional Estimate of the Magnetostatic Forces

The total magnetostatic force exerted on a magnetized body by sources of external origin has been expressed in various forms<sup>(2)</sup> which are all equivalent, although confused in some references. If a body has a volume  $V$  and magnetic moment  $\bar{M}$ , and is subjected to external magnetic field  $\bar{H}_0$ , the total force is given by<sup>(2)</sup>

$$\bar{F} = \int_V \bar{M} \cdot \nabla \bar{H}_0 d\tau . \quad (V.9-1)$$

There exists also a couple

FRAME 7, GFUN LOSN1 DATE 11/30/81 TIME 00:04:34  
 1 CONDUCTOR ELEMENTS 160 IRON ELEMENTS  
 Z-X PLANE  
 Y- 0.000

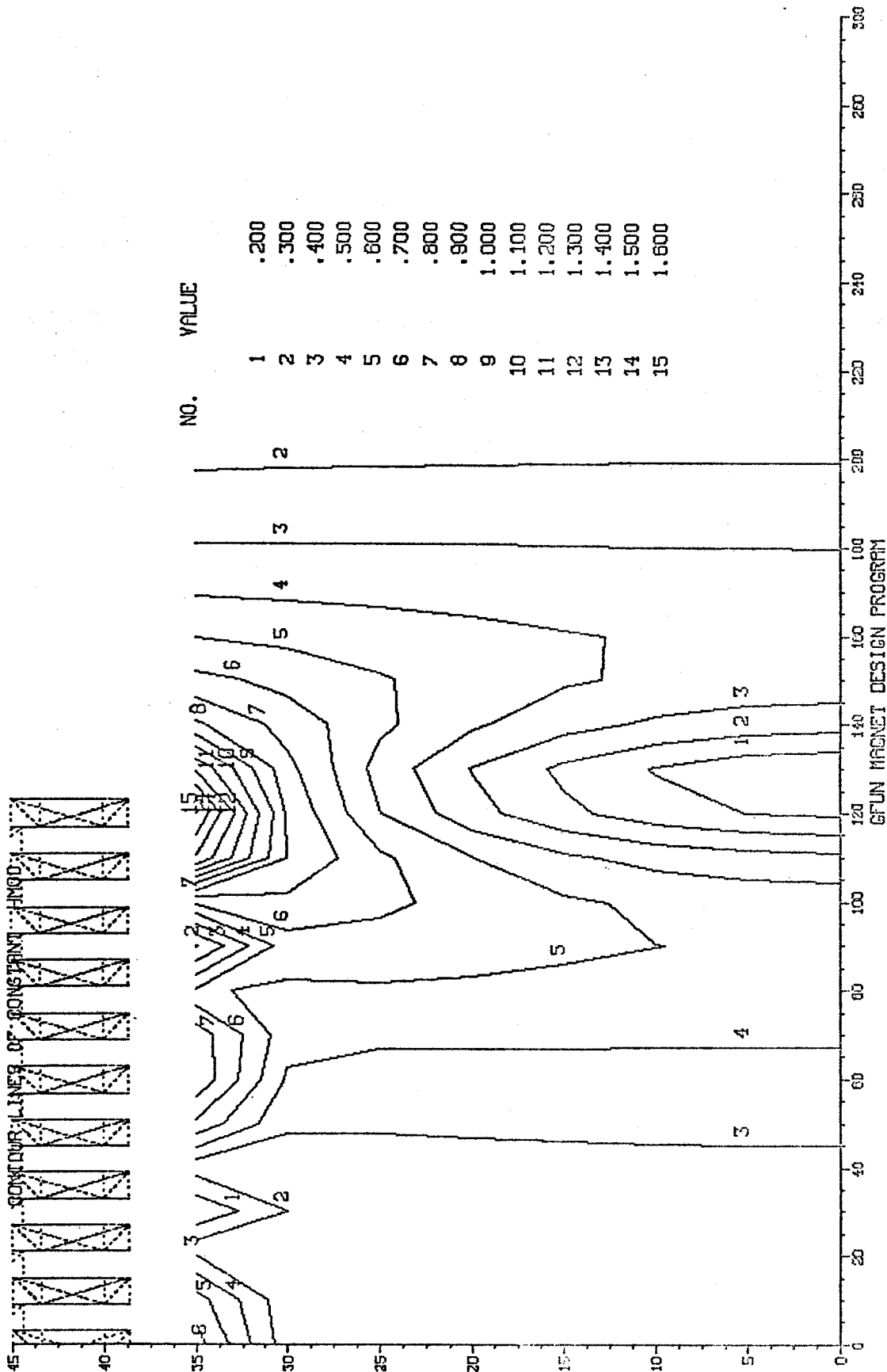


Fig. V.9-3 The magnetic field map for case 1.

FRAME 15, GFUN LGSN1    DATE 11/30/81 TIME 00:33:19  
 1 CONDUCTOR ELEMENTS    35 IRON ELEMENTS  
 Z-X PLANE  
 Y- 0.000

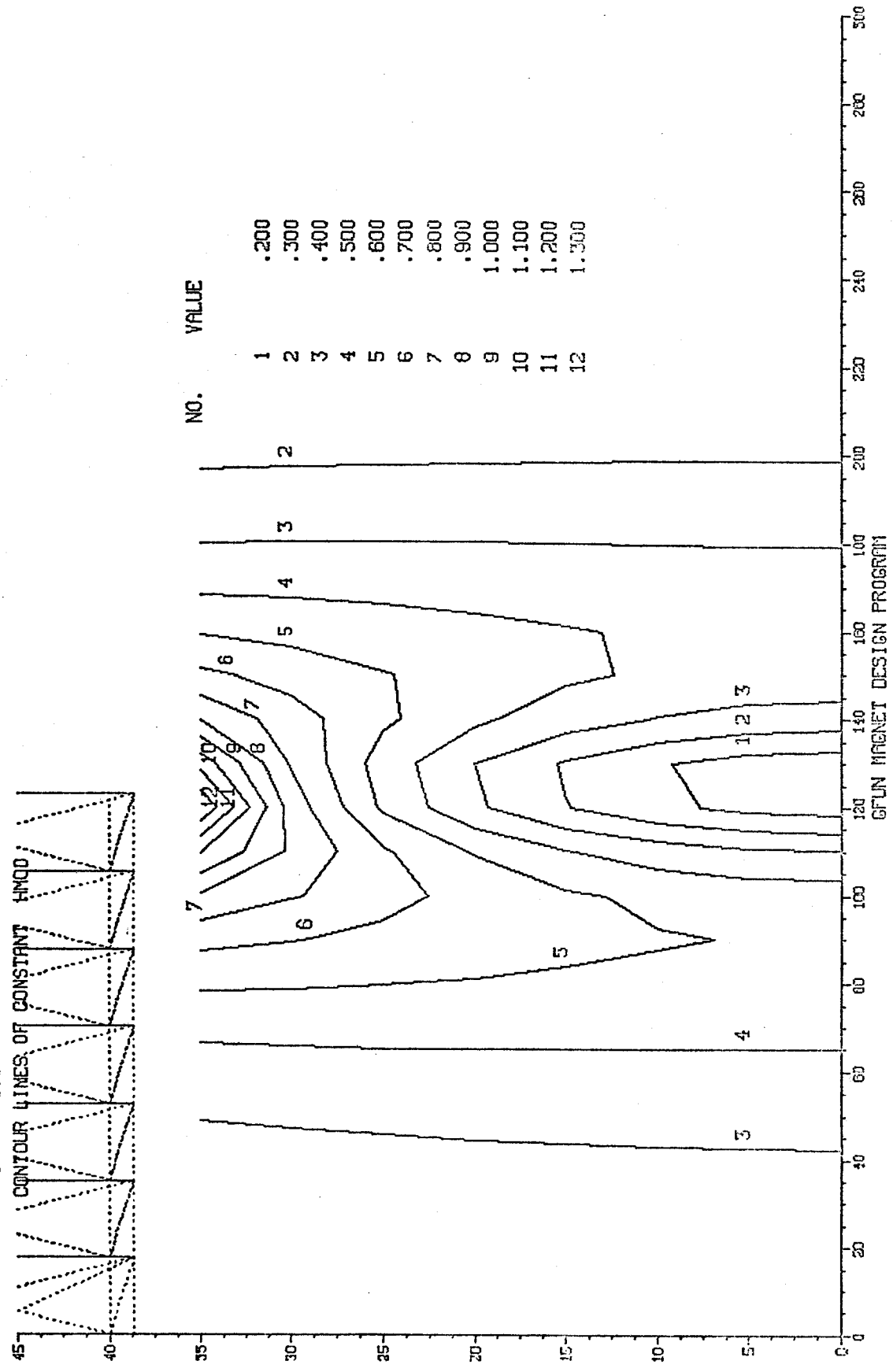


Fig. V.9-4 The magnetic field map for case 12, the homogenized case.



$$\vec{C} = \int_V \vec{M} \times \vec{H}_0 d\tau, \quad (V.9-2)$$

such that the torque about an arbitrary origin 0, with respect to which the position vector of the centroid of the volume element  $d\tau$  is  $\vec{r}$ , is given by

$$L = \int \vec{r} \times (\vec{M} \cdot \nabla \vec{H}_0) d\tau + \int \vec{M} \times \vec{H}_0 d\tau. \quad (V.9-3)$$

A simple one-dimensional estimate of the magnitude of  $\vec{F}$  can be made as follows. Assume the iron is saturated, and  $4\pi\vec{M}$  is on the order of 18000 gauss (see Fig. V.9-1). Also assume the gradient of the field  $\nabla H_0 = 10^3$  gauss/cm, then the maximum magnetostatic force  $\vec{F}$  the body is subjected to, is

$$\vec{F} = \frac{18000}{2\pi} \times 10^3 \times V \text{ dyne}$$

$$\vec{F}/V = 1433 \times 10^3 \text{ dyne/cm}^3$$

$$= 14.33 \text{ newton/cm}^3. \quad (V.9-4)$$

The weight per unit volume, assuming the density of the iron is  $7.81 \text{ g/cm}^3$  is,

$$\frac{\vec{W}}{V} = 7.81 \times 981 \times 10^{-5} \text{ newton/cm}^3$$

$$(V.9-5)$$

$$= .077 \text{ newton/cm}^3.$$

The magnetostatic force is about 187 times the weight of the ferromagnetic body.

#### V.9.3.2 GFUN3D Force Calculations

We have modeled the force calculations in GFUN3D, using only the field of the coils, i.e., neglecting any contribution of other existing ferromagnetic bodies which may contribute to the field on the body whose force is calculated. If included, this would have a small effect when the iron is saturated which is the case in the central cell. The gradient of the field  $\overline{\nabla H}_0$  is evaluated at the centroid of each iron element and assumed constant within it.

A section of a blanket module has been modeled as shown in Fig. V.9-5 in a cross section view of the blanket in the x-y plane, and in Fig. V.9-6 a cross section view in the z-y plane; note that the blanket starts at the center of the reactor and extends for only 20 cm.

The magnitude of the total magnetostatic force on this short section of the blanket was found to be 17076 newtons. The magnitude of each component of the total force and the total torque around the origin are shown in Figs. V.9-5 and V.9-6. The resulting force is about .2 the weight of the blanket. The field gradient in this part of the reactor is very small, and one would expect a larger force if the gradient is changed due to, for example, a loss of power in one of the central cell solenoids. Consequently, the design of the supports for the blanket should take that into account.

The obtained force distribution was used to calculate the stresses in the central pipe in the blanket using SAP4 code.<sup>(4)</sup> The maximum effective stress due to the magnetic force alone is about 0.1 MPa. This is negligible compared to the thermal stresses, which if included, would result in a maximum effective stress of 125.0 MPa.

FRAME 2, UN-GFUN LOSN1 DATE:02/02/82 TIME:11:13:52 NC:34, NI: 575  
 34 CONDUCTOR ELEMENTS 575 IRON ELEMENTS  
 X-Y PLANE  
 ELEMENT 1 TO 575 DISPLAYED MATE- 3

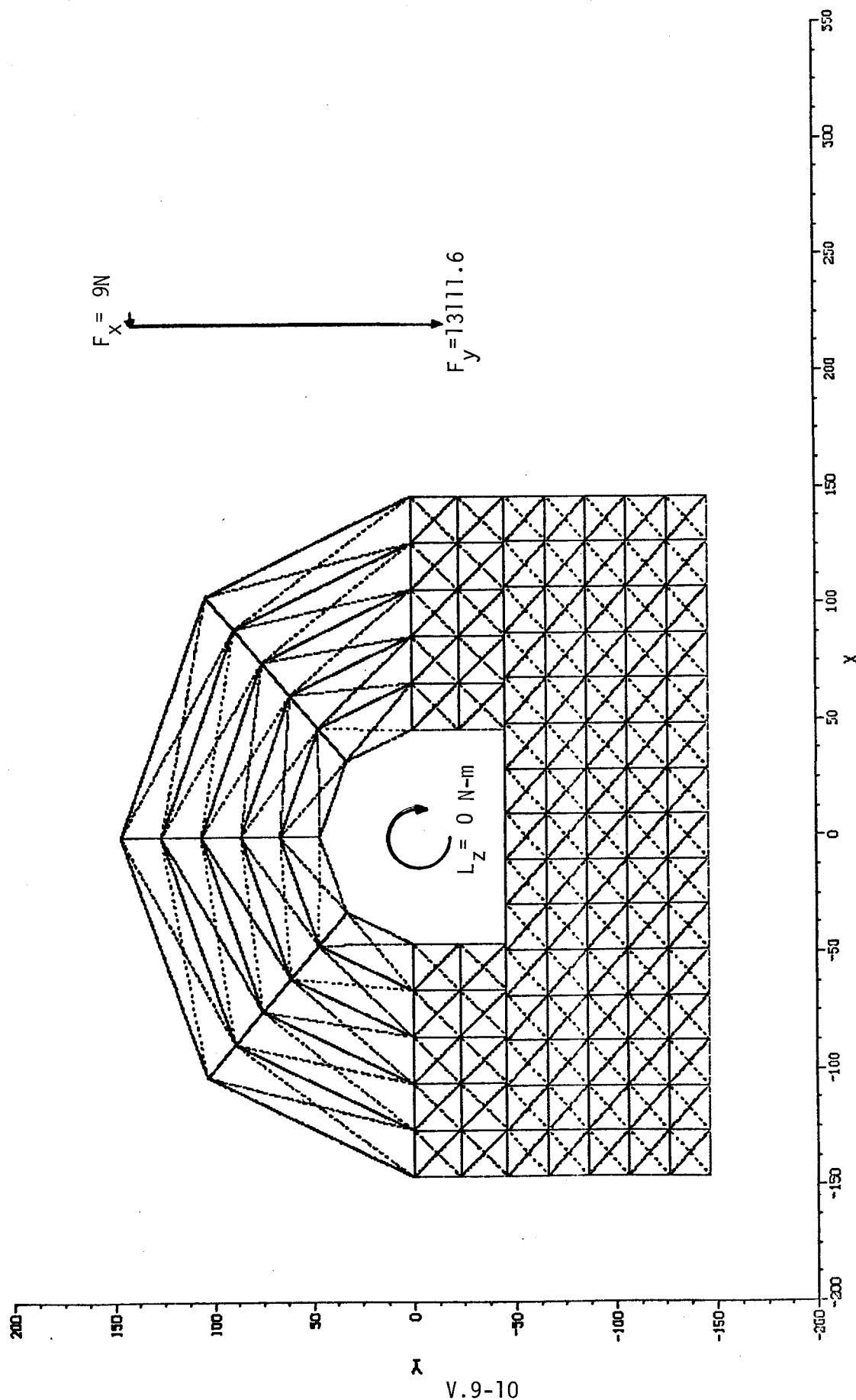


Fig. V.9-5 View of the blanket in x-y plane showing the components of the total magnetostatic force and torque.

FRAME 3, UN-GFUN L09N1    DATE:02/02/82 TIME:11:15:07    NC:34, NI: 575  
 34 CONDUCTOR ELEMENTS 575 IRON ELEMENTS  
 Z-Y PLANE  
 ELEMENT 1 TO 575 DISPLAYED MATE- 3

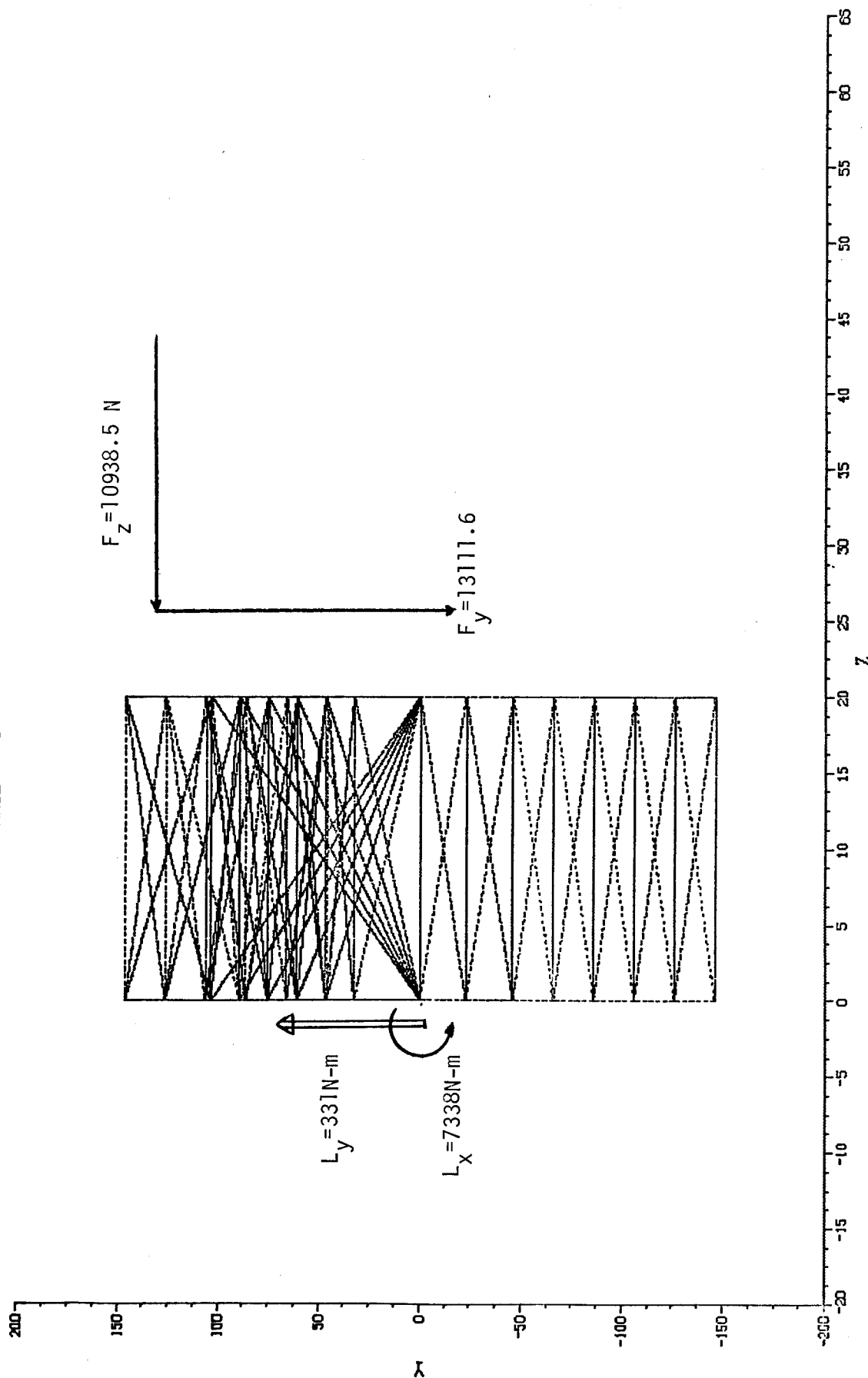


Fig. V.9-6 The view of the blanket in the z-y plane showing the components of the total magnetostatic force and torque.

#### V.9.4 Field Shielding and Field Asymmetry

The presence of the ferromagnetic HT-9 stainless steel in the blanket will partially shield the magnetic field in the central cell seen by the plasma. In addition, it will also reshape the central cell field configuration and possibly destroy its symmetry. However, both effects will vanish if the blanket is made long enough. In this case, the effects will appear only at the ends of the blanket.

Consider an infinite blanket which has any arbitrary shape. The magnetization  $\vec{M}$  can be replaced by two currents flowing on the inner and the outer surface of the blanket, and each is equal to  $\vec{n} \times \vec{M}$ , where  $\vec{n}$  is the outward normal of the surface. The currents are opposite in direction, and each will cancel the effect of the other inside the plasma region. This is a direct result of applying Ampere's law:  $\oint_C \vec{H} \cdot d\vec{\ell} = \Sigma I = 0$ . The field inside the plasma region will be exactly the same as if there is no iron.

For short blankets the end effects start to appear, and both field shielding and field shaping will occur. The shielding of the magnetic field depends on the permeability of the iron. For saturated iron, the shielding will be a minimum. For the blanket considered before, the field on the axis of the reactor with and without the blanket is shown in Figs. V.9-7 and V.9-8, respectively, and from  $z = -100$  cm to  $z = +100$  cm. The average reduction of the field over the 200 cm is about .01 tesla, as can be estimated from the field integrals shown in the figures, and the maximum reduction of the field is about .02 tesla and occurs at about the center of the blanket.

Figure V.9-9 shows the contours of  $\sqrt{B_x^2 + B_y^2}$  at  $z = 0$ . Figure V.9-10 shows the contours of the magnitude of the field at  $z = 10$ , i.e. at the center of the blanket section. Apparently the presence of the blanket has destroyed

FRAME 4, UN-GFUN LOSN1    DATE:02/07/82 TIME:21:08:25    NC:34, NI: 0  
 34 CONDUCTOR ELEMENTS    0 IRON ELEMENTS  
 Z-Y PLANE  
 FIELD INTEGRAL- 5825.1 (SIMP) H-AVE- 2.91

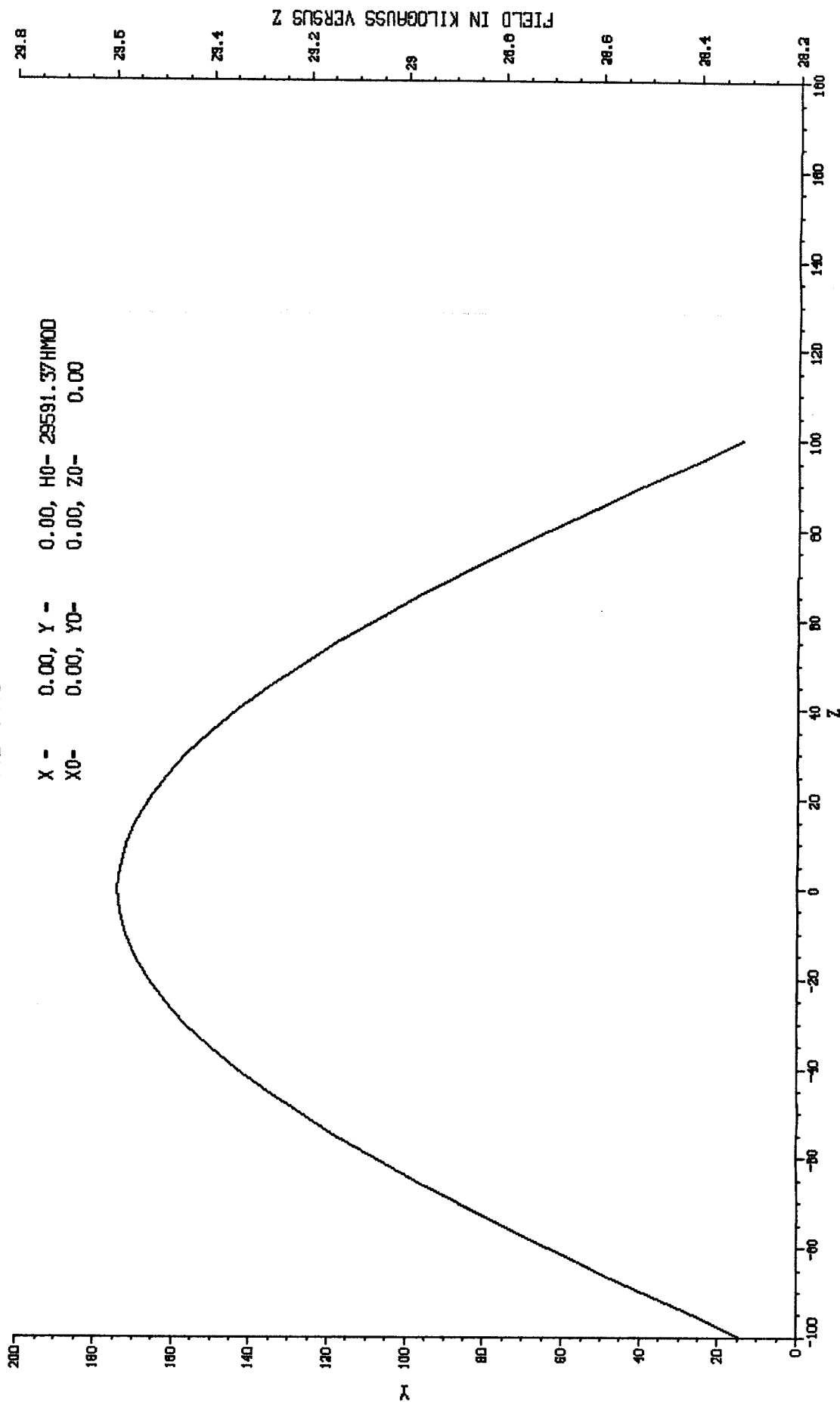


Fig. V.9-7 The magnetic field on the axis of the central cell without the ferromagnetic blanket.

FRAME 6, UN-GFUN LOSN1    DATE:02/07/82 TIME:21:13:01    NC:34, NI: 575  
 34 CONDUCTOR ELEMENTS 575 IRON ELEMENTS  
 Z-Y PLANE  
 FIELD INTEGRAL- 5812.9 (SIMP) H-AVE- 2.91

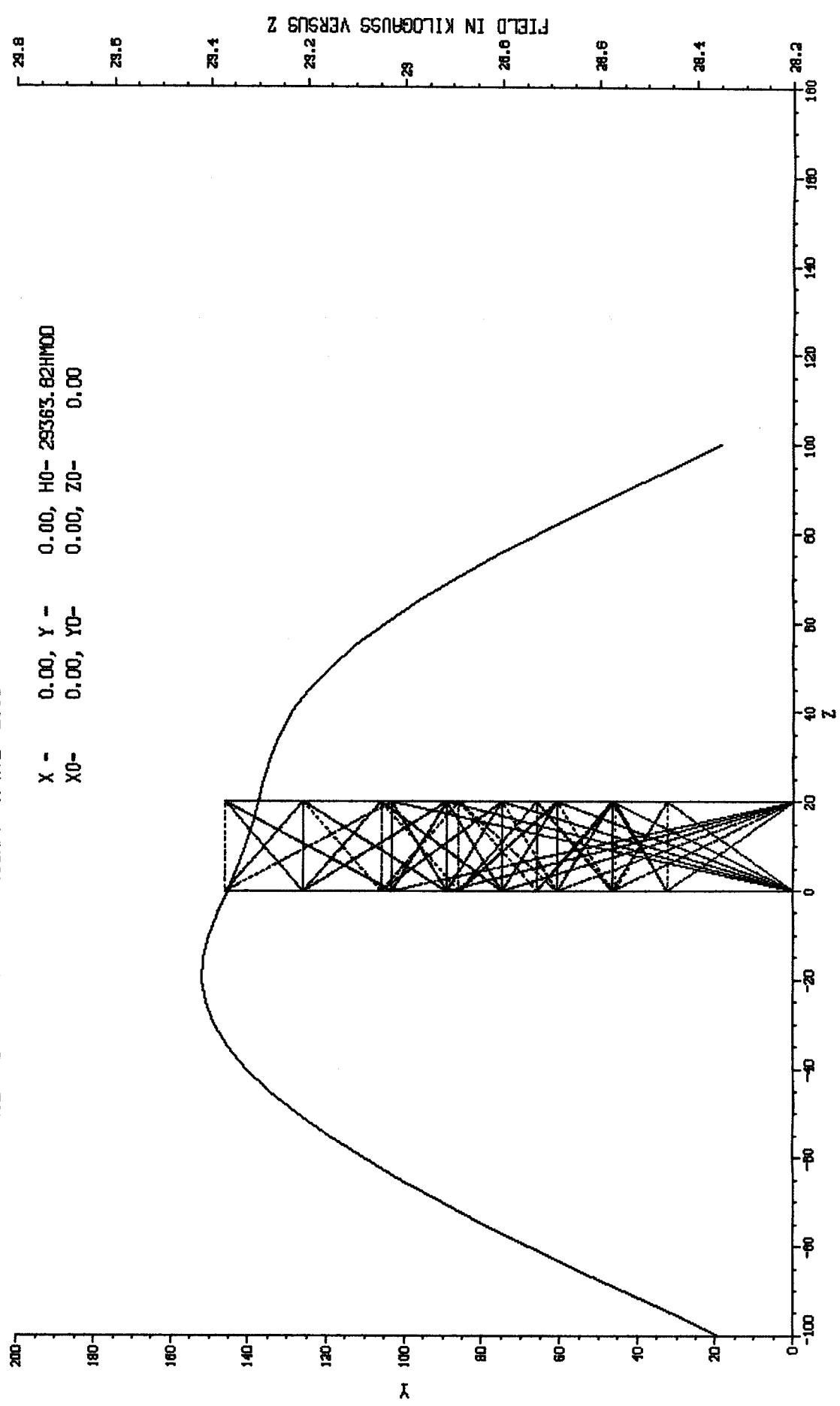


Fig. V.9-8 The magnetic field on the axis of the central cell, with the presence of the short section of the blanket.

34 CONDUCTOR ELEMENTS 575 IRON ELEMENTS

X-Y PLANE

ELEMENT 9 TO 9 DISPLAYED RATE- 0

Z- 0.000

CONTOUR LINES OF CONSTANT MODX 12

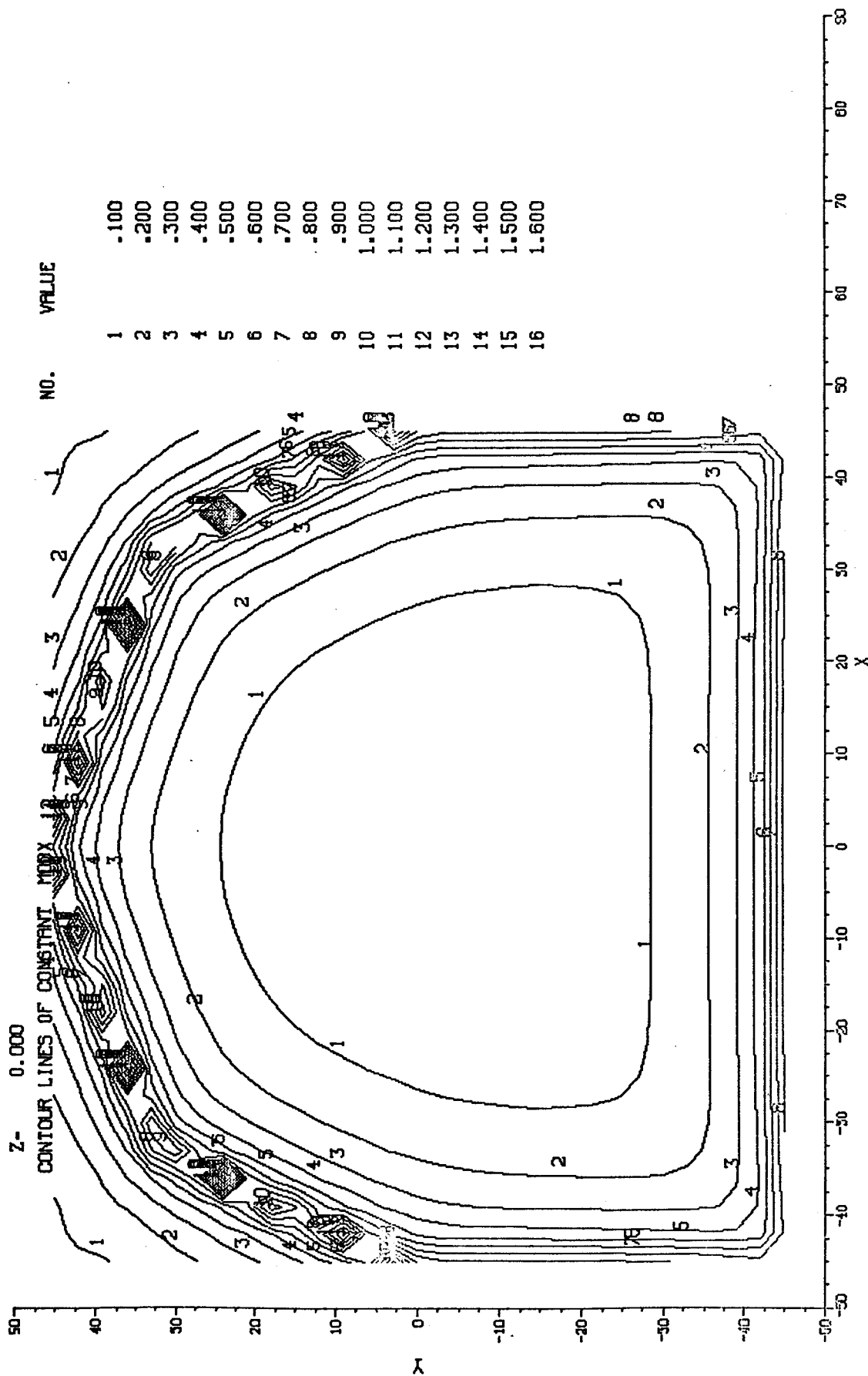


Fig. V.9-9 Map of the magnetic field;  $\sqrt{B_x^2 + B_y^2}$  at the left end of the blanket.



34 CONDUCTOR ELEMENTS 575 IRON ELEMENTS

X-Y PLANE

ELEMENT 9 TO 9 DISPLAYED MATE- 0

Z- 10.000

CONTOUR LINES OF CONSTANT HM00

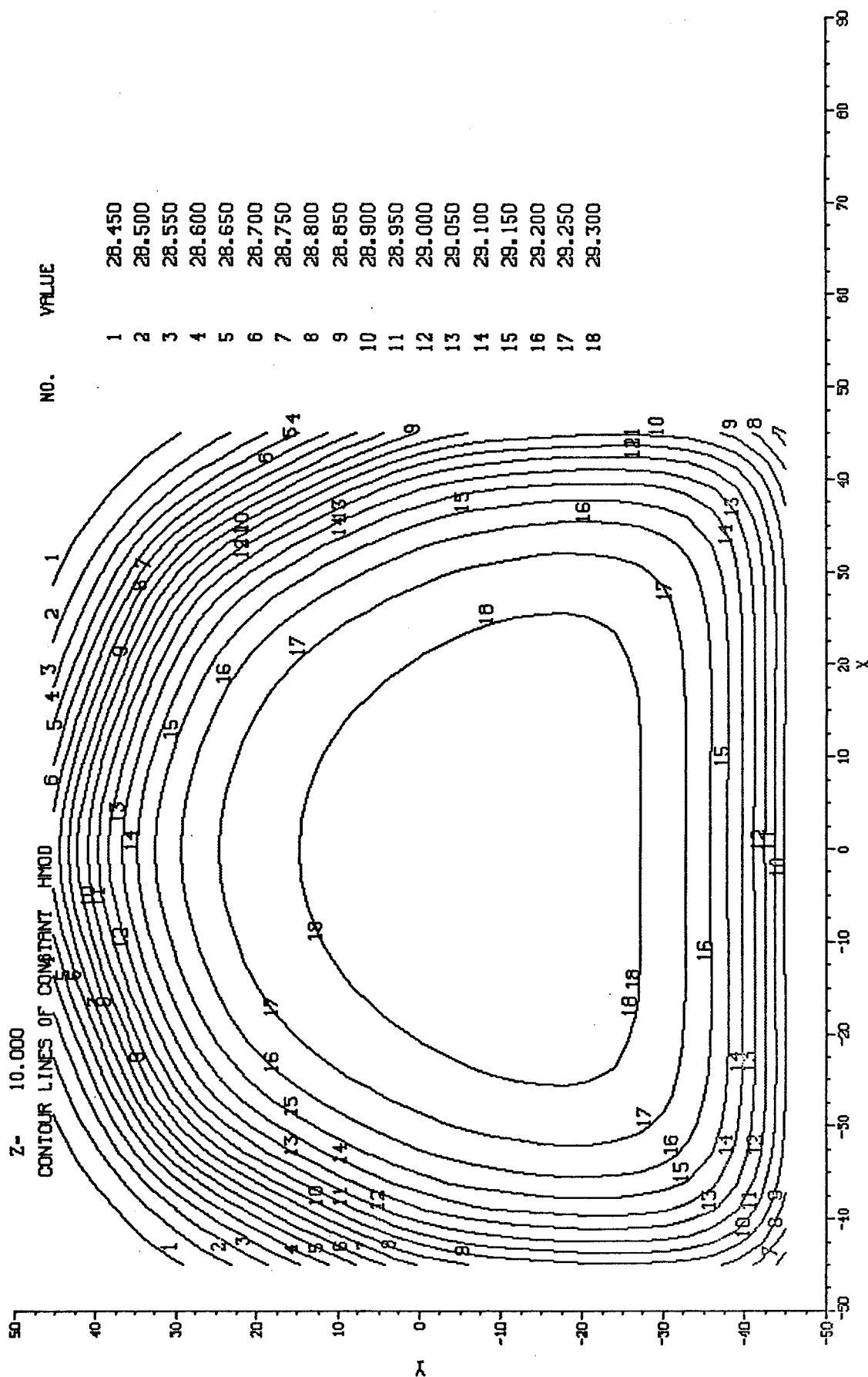


Fig. V.9-10 Map of the magnetic field;  $\text{mod}(B)$  at the center of the blanket section.

the symmetry of the field, and the field contours take almost the same shape as that of the blanket. The perturbation was initially thought to create radial diffusion of the plasma and hence destroys the confinement of the plasma.

To examine this problem further we have calculated the gradient of the field in cylindrical coordinates to evaluate the term  $\frac{\bar{B}_x \nabla |B|}{B^3}$  which can be readily used to estimate the drift in any direction.<sup>(3)</sup> The field, the gradient of the field, and the  $\frac{\bar{B}_x \nabla |B|}{B^3}$  components are shown in Figs. V.9-11a, V.9-11b, and V.9-11c, respectively, at  $z = 0$  and a radius of 30 cm, i.e., near the edge of the plasma. For further discussion and for the conclusions with respect to this problem refer to the plasma section.

#### V.9.5 Field Shielding and Perturbation in Case of Long Ferromagnetic Blankets

The magnetic field perturbation, and in particular the reduction in the magnetic field in the plasma, will also depend on the length of the blanket. In the previous section, the computations were done for a short section of the blanket because it was desired to model as accurately as possible the azimuthal distribution and configuration of the ferromagnetic blanket. The number of elements required is large so that the entire blanket could not be modeled in detail.

In order to study the effect of the length of the blanket on the magnetic field distribution, two different blanket configurations were examined in more detail.

One blanket model had the shape of a cylindrical annulus, and one the shape of a rectangular prism with a rectangular hole. The former model was treated analytically, whereas the latter was evaluated numerically with GFUN.

X-Y PLANE Z- 0.0, R- 30.0, PHI- 0.0

FIELD COMPONENTS IN K-GAUSS

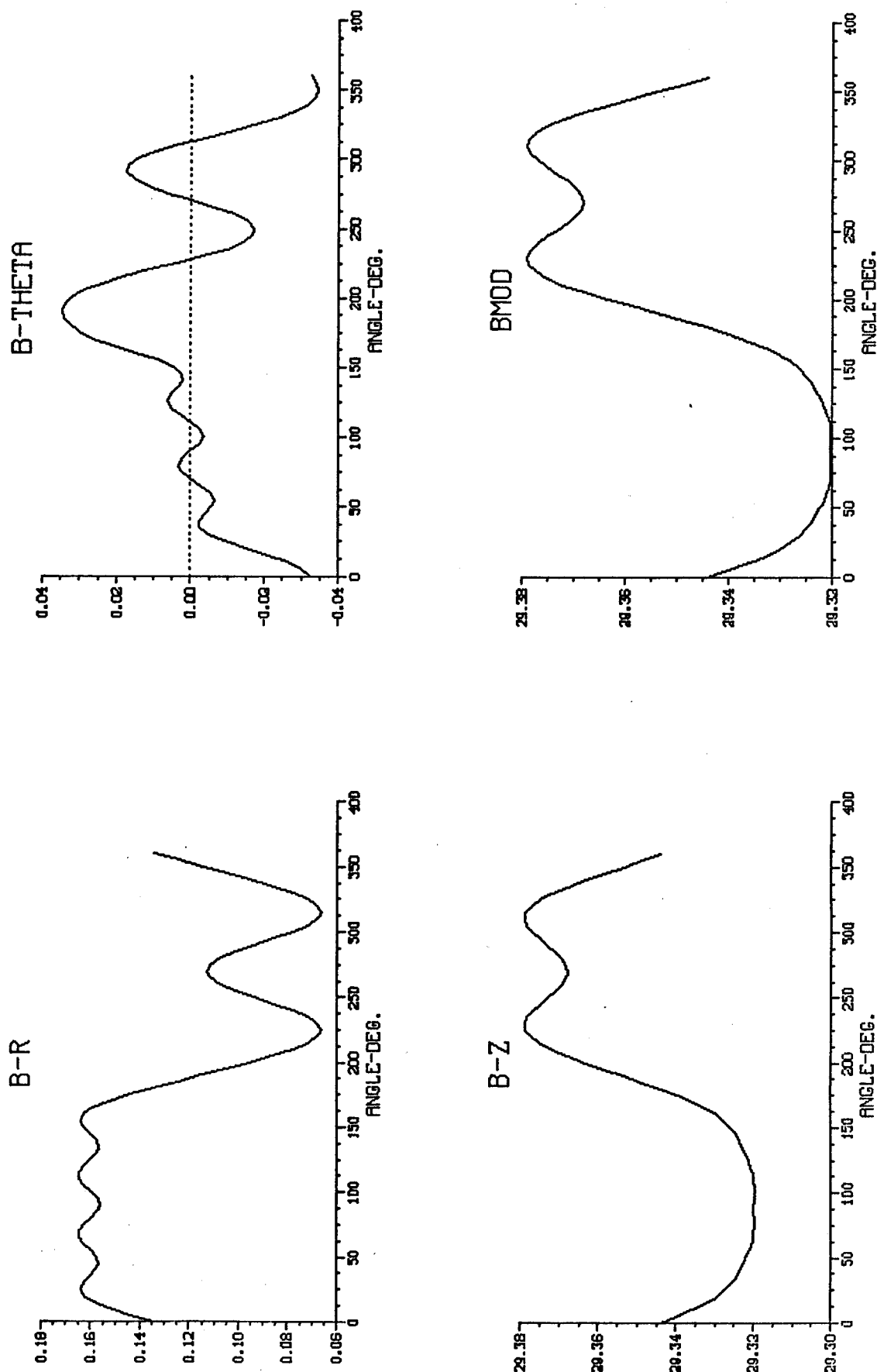


Fig. V.9-11a The azimuthal distribution of the magnetic field components at 30 cm radius and at the center of the central cell with the presence of the blanket included.

X-Y PLANE Z- 0.0, R- 30.0, PHI- 0.0

GRAD(B) COMPONENTS IN KG/CM

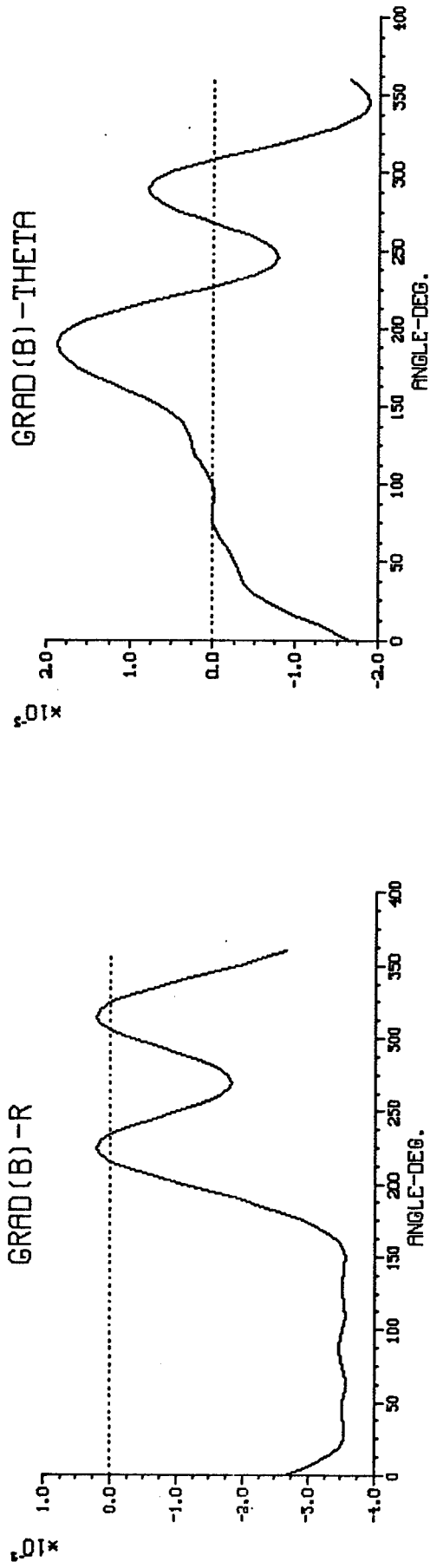


Fig. V.9-11b Azimuthal distributions of  $\nabla B$ .

NC:34, NI: 575

DATE:01/28/82 TIME:12:50:45

X-Y PLANE Z- 0.0, R- 30.0, PHI- 0.0

DRIFT COMPONENTS (B X GRAD(B)/B\*\*3) IN 1/(CM-KG)

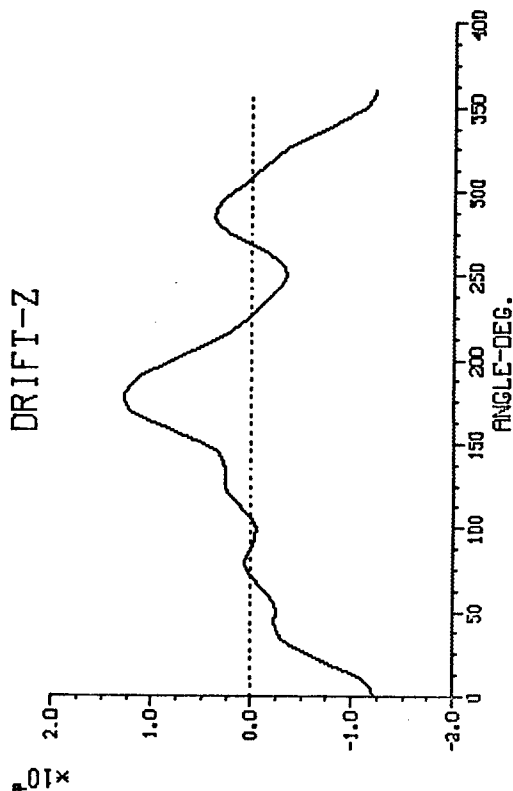
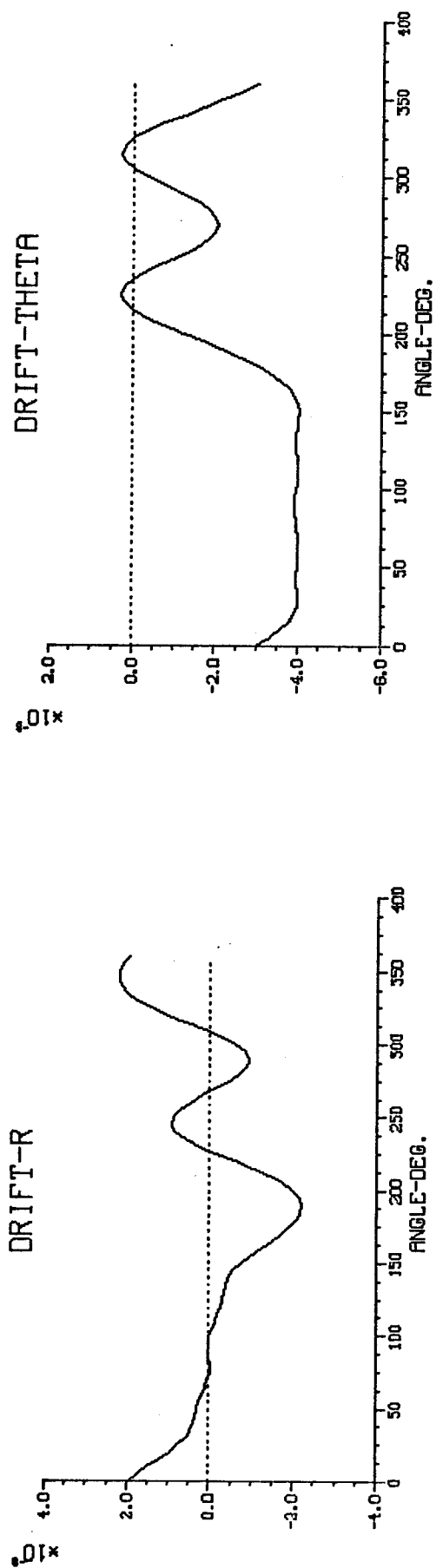


Fig. V.9-11c Azimuthal distributions of  $\frac{\bar{B}x\sqrt{B}}{B^3}$ , the drift term.

For the circular blanket, the inner and outer radii are designated  $a$  and  $b$ , respectively, whereas the total length is  $L = 2\ell$ . If it is assumed that the ferromagnetic blanket structure is magnetized to saturation, and if this saturation magnetization  $\mu_0 M$  is constant everywhere and parallel to the axial direction, then the field perturbation  $\Delta \vec{B}$  can be computed analytically. The total field is then the sum of the vacuum field  $\vec{B}_0$  and the field perturbation  $\Delta \vec{B}$ , i.e.  $\vec{B} = \vec{B}_0 + \Delta \vec{B}$ . The field perturbation can be computed from

$$\Delta \vec{B}(\vec{r}) = \frac{\mu_0}{4\pi} \oint \frac{(\vec{r} - \vec{r}') \times \vec{M} \times \vec{n}'}{|\vec{r} - \vec{r}'|^3} dS' - \frac{\mu_0}{4\pi} \int \frac{(\vec{r} - \vec{r}') \times \nabla' \times \vec{M}'}{|\vec{r} - \vec{r}'|^3} dV' . \quad (V.9-6)$$

For a constant magnetization the second term vanishes and the field  $\Delta \vec{B}$  is given by the surface integral. Since  $\vec{M}$  was assumed to be axial,  $\vec{M} \times \vec{n}$  vanishes on the end faces of the annulus, and contributions arise only from the cylindrical surfaces. In fact, the field  $\Delta \vec{B}$  is identical to the one produced by two concentric solenoids of equal length with equal but opposite currents. The axial field on axis is found to be given by

$$\begin{aligned} \frac{2\Delta B_z(z)}{\mu_0 M} &= \frac{(\ell + z)}{\sqrt{a^2 + (\ell + z)^2}} + \frac{(\ell - z)}{\sqrt{a^2 + (\ell - z)^2}} \\ &\quad - \frac{(\ell + z)}{\sqrt{b^2 + (\ell + z)^2}} - \frac{(\ell - z)}{\sqrt{b^2 + (\ell - z)^2}} \end{aligned} \quad (V.9-7)$$

where  $z$  is the distance from the midplane. The variation of the axial field on axis is shown in Fig. V.9-12 for differential lengths as a function of the reduced axial position  $z/\ell$ . It is seen that the field reduction on axis is due to end effects. For sufficiently long blankets, i.e. when  $L/2b \gg 1$ , then the field reduction on axis at the midplane becomes very small. The end

# FIELD REDUCTION AS A FUNCTION OF BLANKET LENGTH

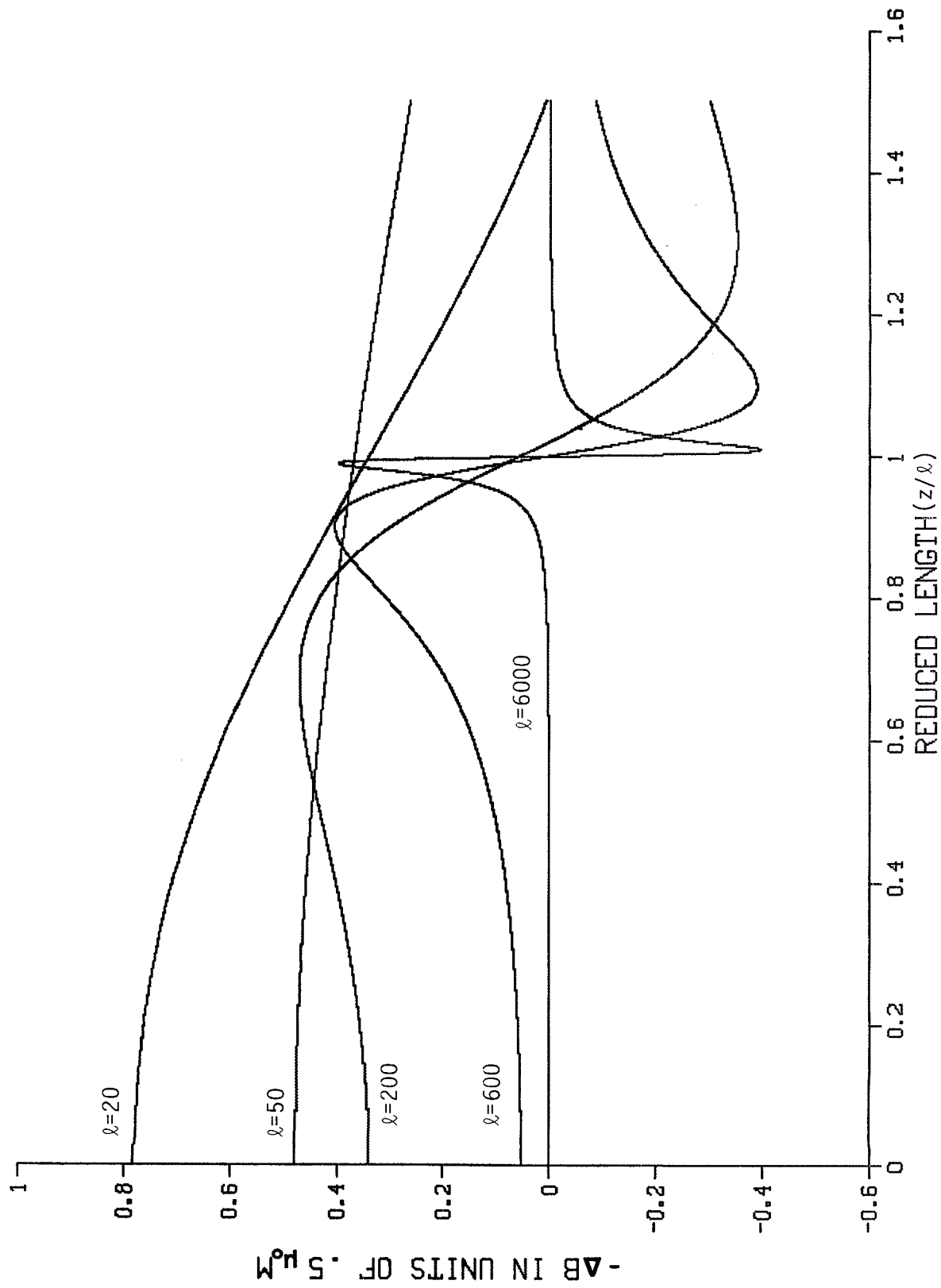


Fig. V.9-12. Axial field reduction on axis as a function of the reduced axial distance  $z/\lambda$ ; for different half-lengths  $\lambda$  of the blanket with 50 cm inner radius and 150 cm outer radius.

effects for large aspect ratios  $L/2b$  consists of an axial region inside the blanket where the field is reduced ( $-\Delta B$  is positive) and an outside region where the field is enhanced ( $-\Delta B$  is negative). Both regions have an axial extension of the order of the outer diameter of the blanket. As the blanket gets shorter, the two end effects overlap, reinforcing each other inside the blanket region.

To elucidate this point further, the field on axis at the midplane,

$$\frac{\Delta B_z(0)}{\mu_0 M} = \frac{1}{\sqrt{1 + a^2/\ell^2}} - \frac{1}{\sqrt{1 + b^2/\ell^2}} \quad (\text{V.9-8})$$

and the average field on axis

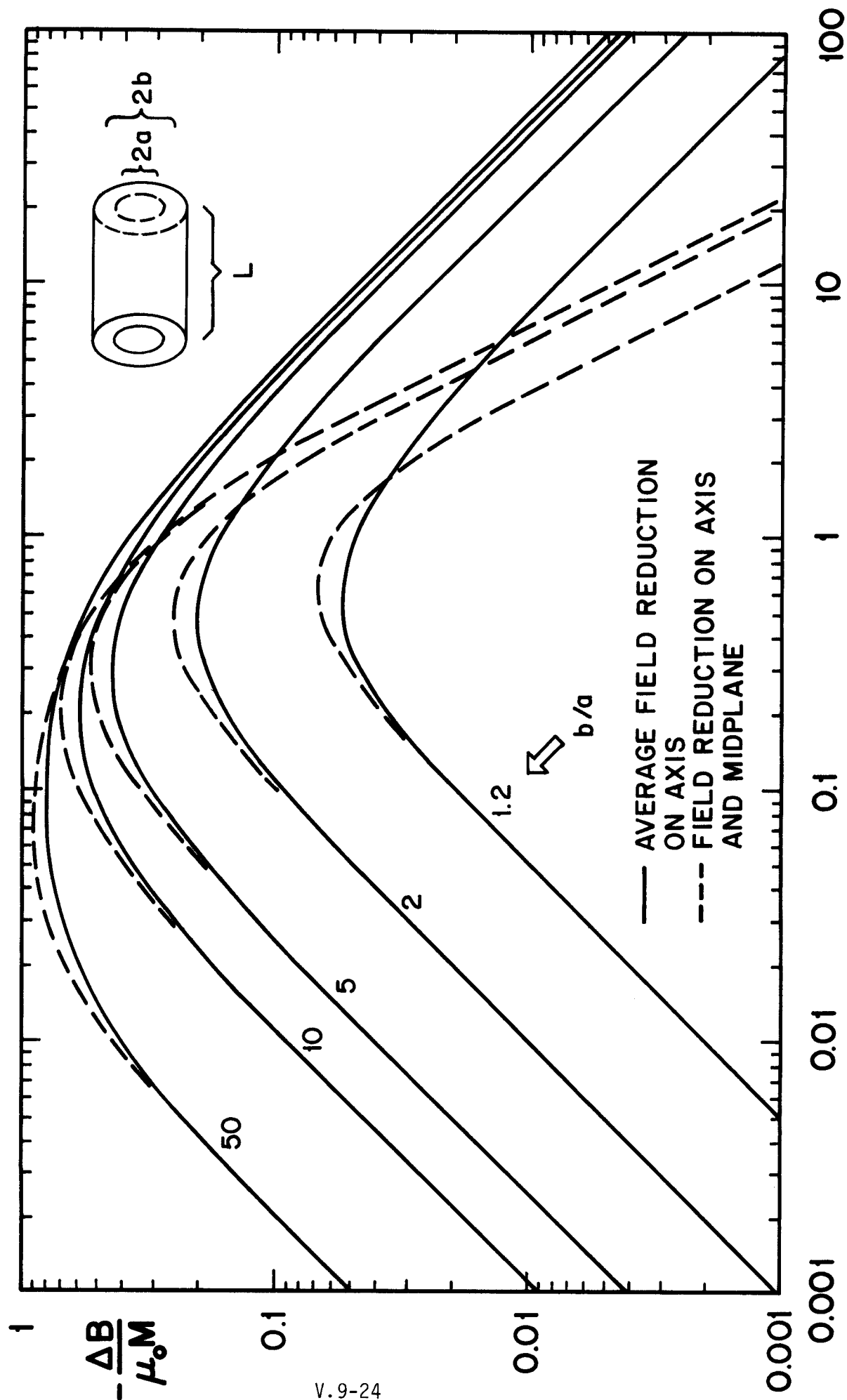
$$\frac{1}{\mu_0 ML} \int_{-\ell}^{+\ell} \Delta B_z(z) dz = \frac{\Delta \bar{B}_z}{\mu_0 M} = \sqrt{1 + \frac{a^2}{L^2}} - \frac{a}{L} - \sqrt{1 + \frac{b^2}{L^2}} + \frac{b}{L} \quad (\text{V.9-10})$$

were computed for various configurations. Figure V.9-13 shows that both the field reduction at the midplane and the average field reduction show a maximum when the aspect ratio  $L/2b$  is between 0.1 and 1. As this aspect ratio increases, the field reduction drops again at a rate proportional to  $2b/L$ .

So far, we modeled the blanket as a homogenized medium. Therefore, if  $M_0$  is the value of the magnetization in the ferromagnetic steel and  $f$  is the metal fraction in the blanket, then  $M = M_0 f$ .

In order to test the validity of this homogenization procedure, the field reduction was computed for a blanket model consisting of a number of concentric annuli of various thicknesses and various separation distances. For each individual annulus, the field reduction was computed according to the above equation, the contributions from all annuli added, and the result divided by





**LENGTH TO OUTER DIAMETER RATIO,  $L/2b$**

Fig. V.9-13. Effect of radial and axial dimensions on the field reduction on axis.

the average magnetization  $\mu_0 M_0 f$ . Some results are shown in Fig. V.9-14. It is found that the homogenization approach is indeed justified and gives very accurate results when the aspect ratio  $L/2b > 0.1$ . Furthermore, as the number of annuli is increased, the agreement with the homogenized result improves.

We also examined the effect of a radial variation of the magnetization. Since the saturation magnetization  $M_0$  is a function of the temperature of the ferromagnetic steel, temperature variations through the blanket are of concern.

For a constant magnetization gradient

$$\frac{dM}{dr} = \frac{[M(b) - M(a)]}{(b - a)}$$

the field reduction on axis and at the midplane is given by

$$\frac{\Delta B_z(0)}{\mu_0} = \frac{M(a)}{\sqrt{1 + a^2/\ell^2}} - \frac{M(b)}{\sqrt{1 + b^2/\ell^2}} + \frac{\ell[M(b) - M(a)]}{b - a} \ln \left[ \frac{b + \sqrt{b^2 + \ell^2}}{a + \sqrt{a^2 + \ell^2}} \right] . \quad (V.9-11)$$

Although a closed-form solution was found for  $\Delta B_z(z)$ , an analytical expression for  $\Delta \bar{B}_z$  could not be obtained. Therefore, we only show the results for  $\Delta B_z(0)/\mu_0 \bar{M}$  in Fig. V.9-15.  $\bar{M}$  is the volume-average of the magnetization. It is seen again that for large aspect ratios  $L/2b$ , the field reduction decreases, and that the detailed distribution of the magnetization has little effect. The effect of temperature on the magnetization can therefore be

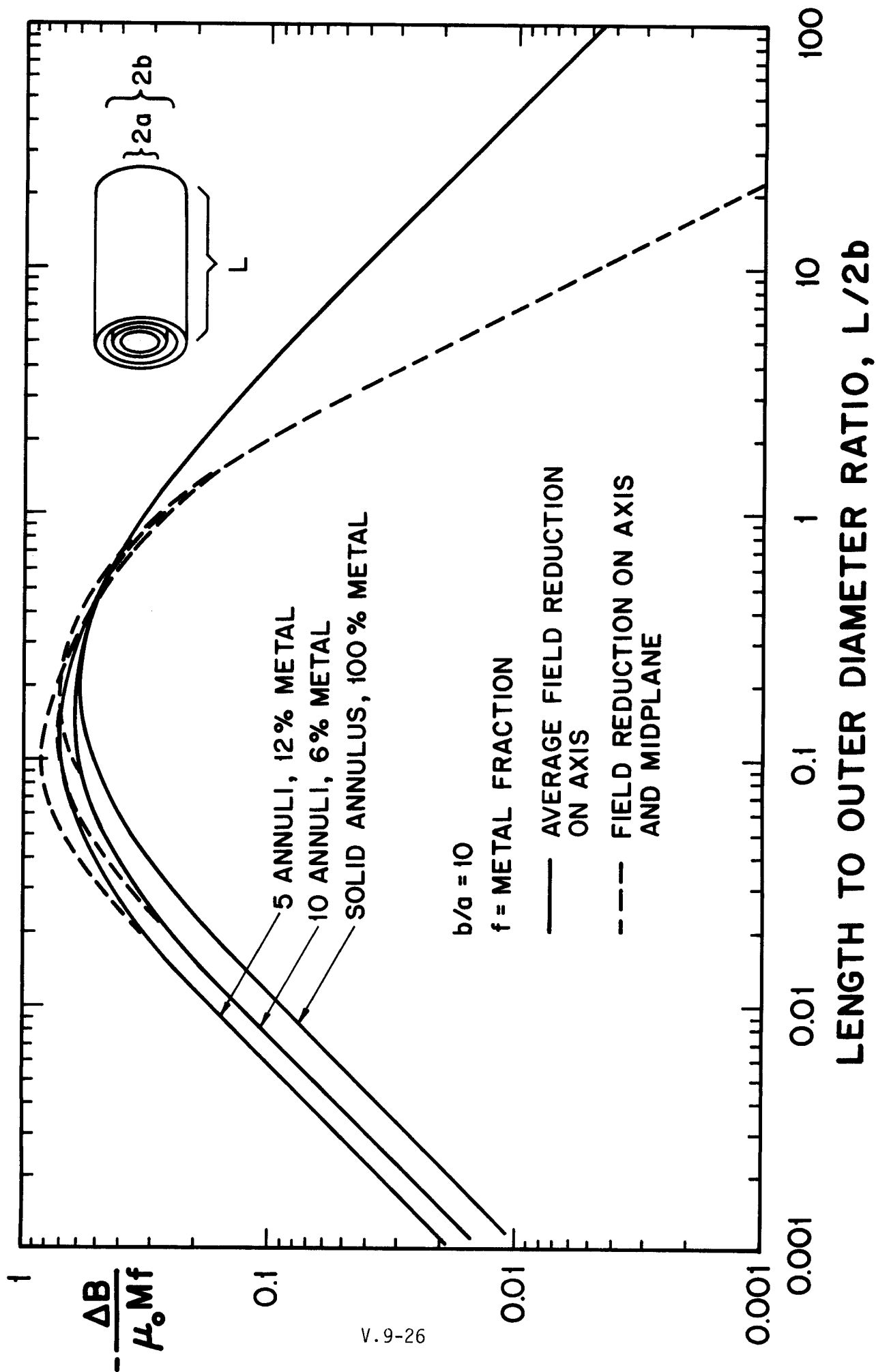


Fig. V.9-14. Effect of material distribution on the field reduction on axis.

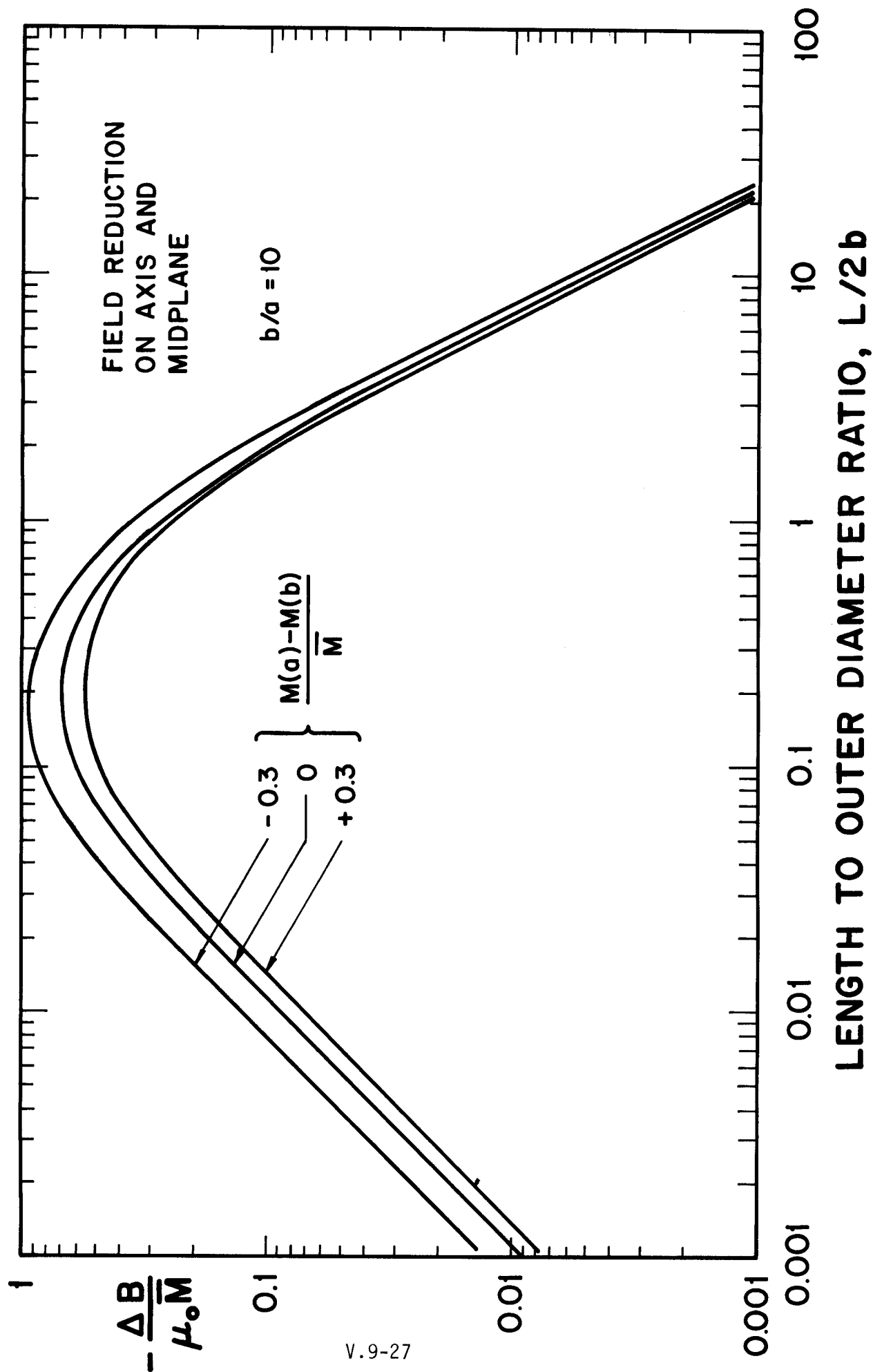


Fig. V.9-15. Effect of radial variations in the magnetization on the field reduction.

accounted for in an approximate manner if one employs the magnetization averaged over the entire blanket region.

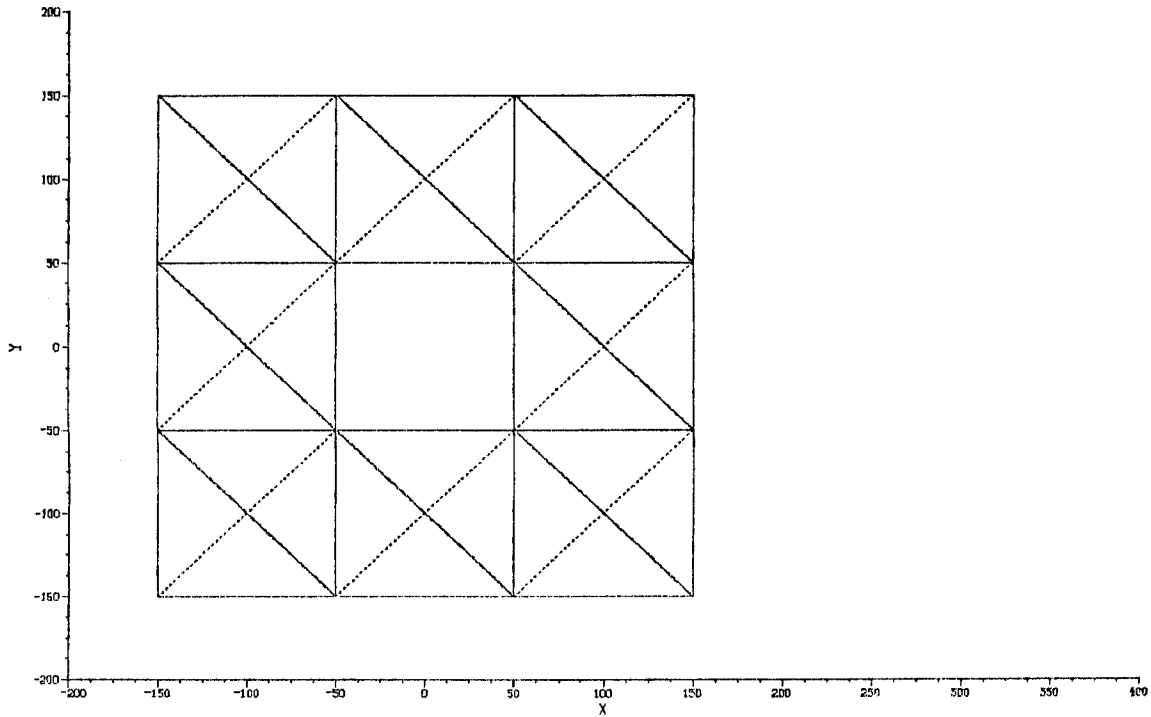
As mentioned in the beginning of this section we have studied the effect of long blankets with GFUN. Two blankets are considered which are shown in Figs. V.9-16 and V.9-17. The two blankets have the same length (12 m), but have different cross sections as shown in the figures. The magnetic field on the axis was calculated and is shown in Fig. V.9-18. The two blankets produced exactly the same field. The field with no blanket present in the central cell is also shown in the same figure with the dashed line. The maximum reduction in the field on the axis was about 500 gauss, and occurs at  $z = 550$  cm, near the end of the blanket. This is in agreement with the previous analytical results.

The shape of the blanket as discussed in the previous section should have an effect on the field especially near the blanket (i.e. at the plasma edge). Figures V.9-19-V.9-21 show the field components in cylindrical coordinates at radius = 20 cm and at the location  $z = 600$  cm with no blanket, with the square blanket, and with the rectangular blanket (note the fine scale on these figures), respectively. It is seen that the two different blanket cross sections affect the field in different ways which depends on the shape and symmetry properties of the blanket cross sections.

In summary, the detailed analysis of the magnetic field perturbation by various blanket configurations has shown that:

- a. the perturbation and reduction for a magnetically saturated structure is significant only near the ends, and over an axial region of the order of the outer blanket diameter or width;

FRAME 10, UN-9FUN LOSN1 DATE:03/01/82 TIME:09:10:25 NC:32, NI: 240  
 32 CONDUCTOR ELEMENTS 240 IRON ELEMENTS  
 X-Y PLANE  
 ELEMENT 1 TO 240 DISPLAYED MATE- 3



FRAME 11, UN-9FUN LOSN1 DATE:03/01/82 TIME:09:12:54 NC:32, NI: 240  
 32 CONDUCTOR ELEMENTS 240 IRON ELEMENTS  
 Z-Y PLANE

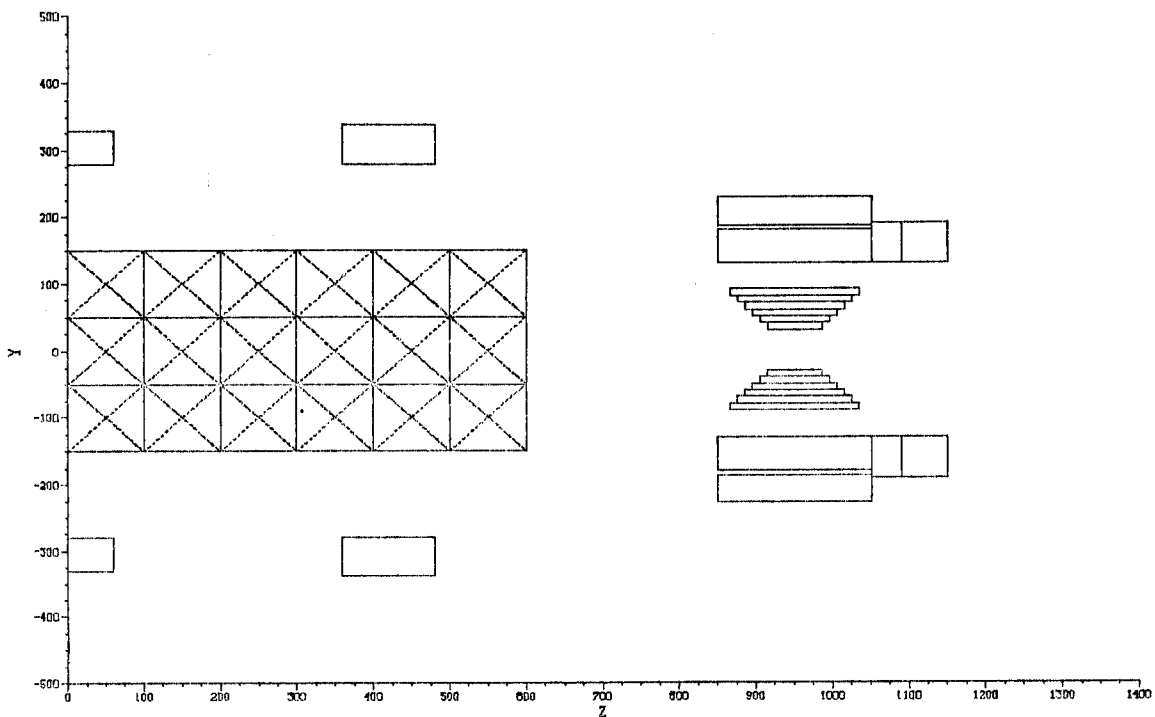
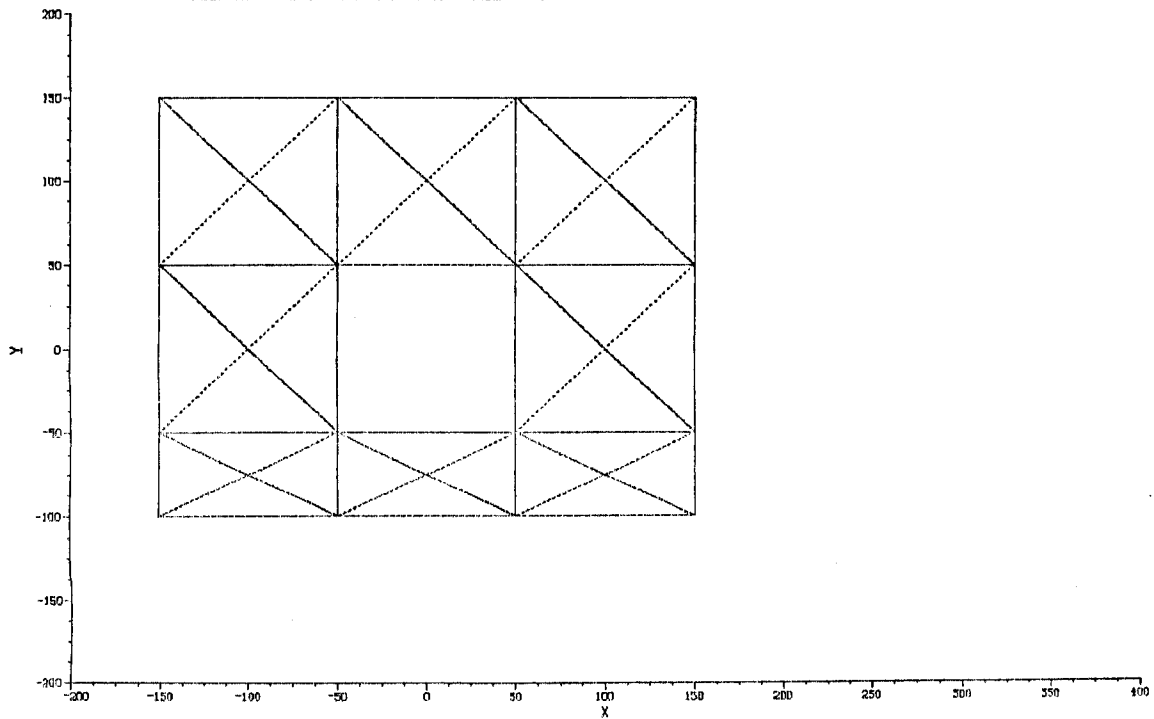


Fig. V.9-16. Two cross section views of the symmetric square cross section blanket.

FRAME 5, UW-OFUN LOSN1 DATE:03/03/02 TIME:13:41:08 NC:32, NI: 240  
 32 CONDUCTOR ELEMENTS 240 IRON ELEMENTS  
 X-Y PLANE  
 ELEMENT 1 TO 240 DISPLAYED MATE- 3



FRAME 6, UW-OFUN LOSN1 DATE:03/03/02 TIME:13:42:10 NC:32, NI: 240  
 32 CONDUCTOR ELEMENTS 240 IRON ELEMENTS  
 Z-Y PLANE

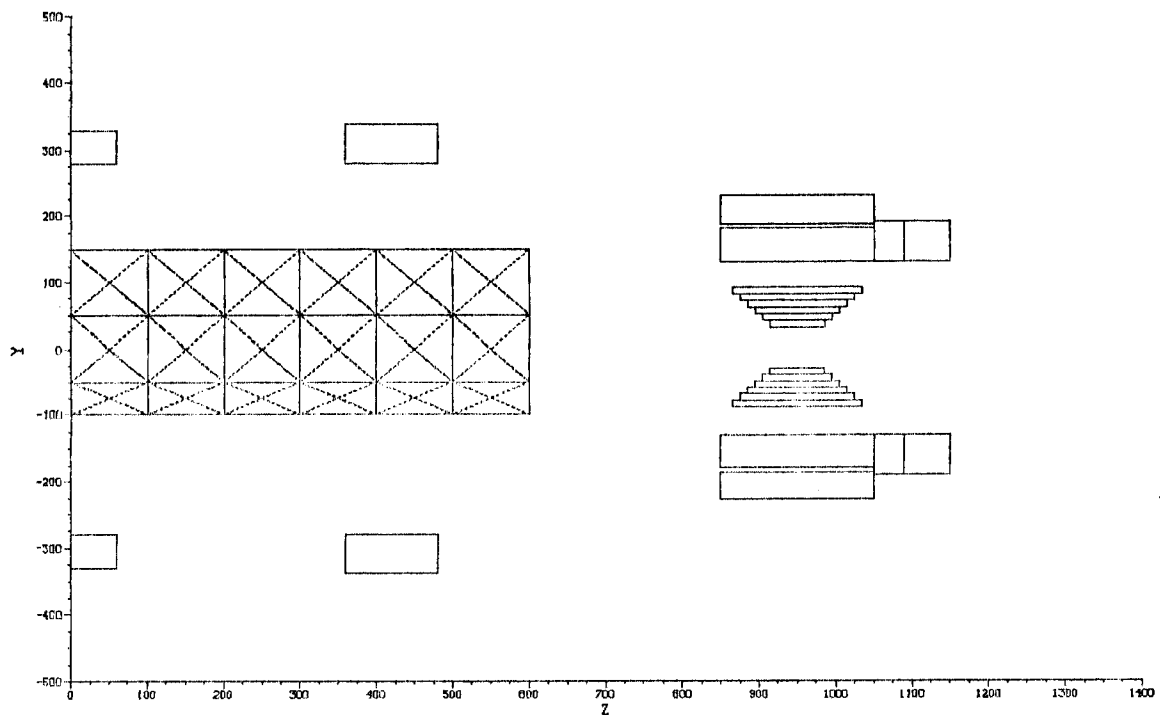


Fig. V.9-17. Two cross section views of the asymmetric rectangular blanket.

FRAME 6, UN-OFUN L03N1    DATE:03/03/82 TIME:08:36:38    NC:32, NI: 240  
 32 CONDUCTOR ELEMENTS 240 IRON ELEMENTS  
 Z-Y PLANE  
 FIELD INTEGRAL= 34160. (SIMP) H-AVE= 2.63

X = 0.00, Y = 0.00, H0= 24356.70HMOD  
 X0= 0.00, Y0= 0.00, Z0= 0.00

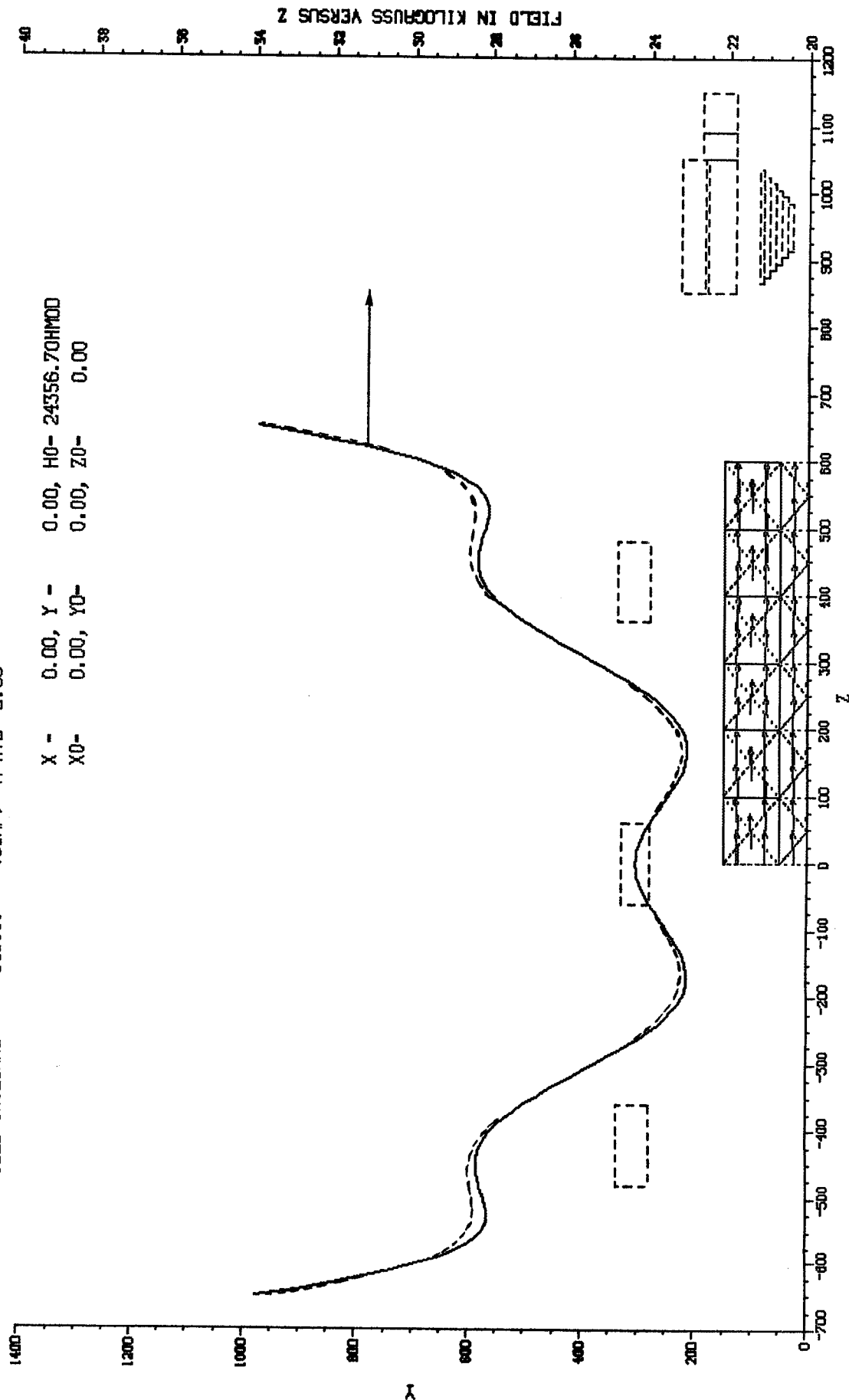


Fig. V.9-18. The magnetic field on the axis without the blanket (dashed line), and with the blanket (solid line).



FRAME 18, UN-GFVN L03N1 DATE:03/03/82 TIME:08:30:00 NC:32, NI: 0

X-Y PLANE Z=600.0, R= 20.0, PHI= 0.0

FIELD COMPONENTS IN K-GAUSS

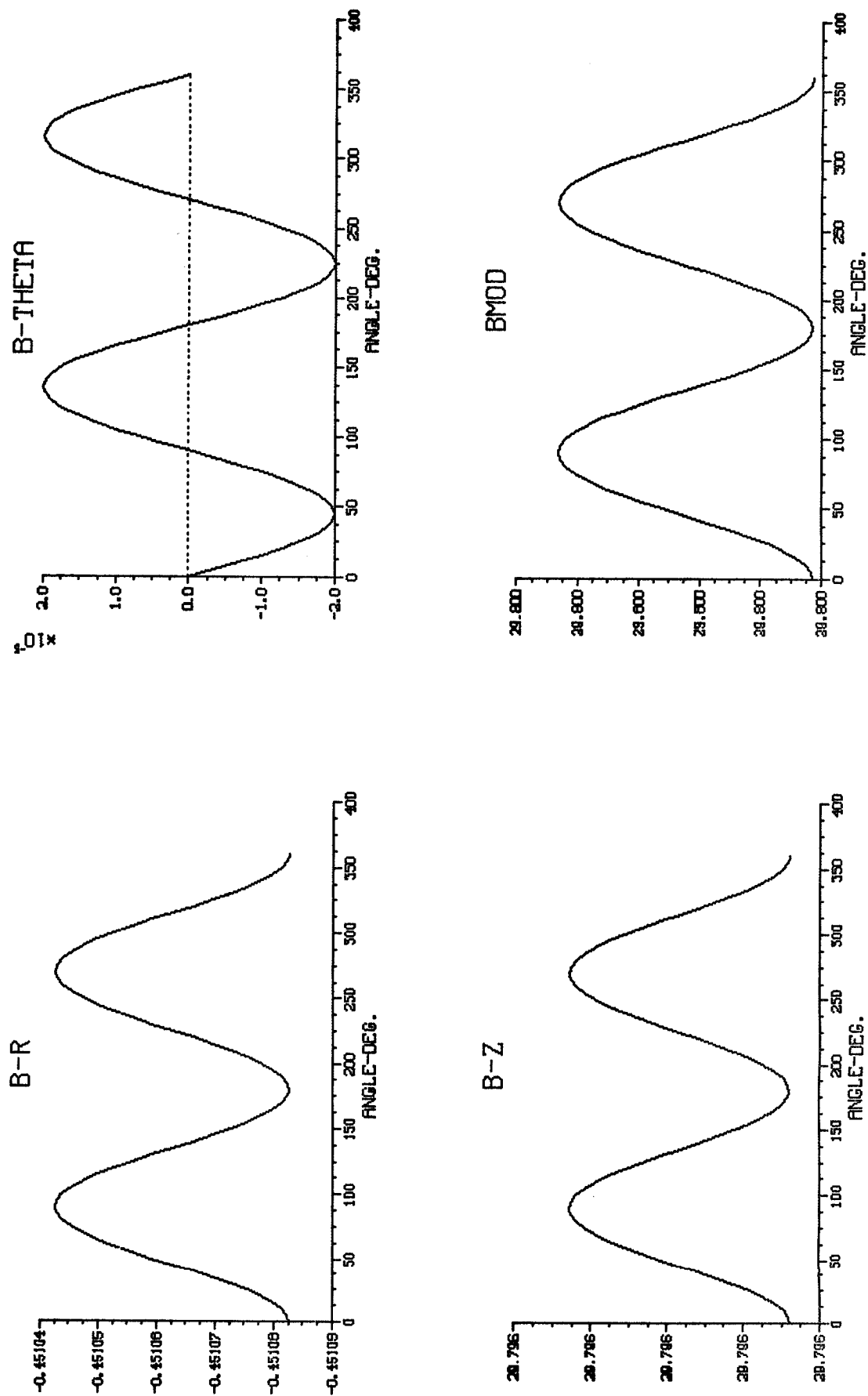


Fig. V.9-19. Field components as a function of the azimuthal angle at radius = 20 cm and at axial location = 600 cm, in the absence of the blanket.

FRAME 20, UN-GFUN L05N1    DATE:03/03/82 TIME:09:18:58    NC:32, NI: 240  
 X-Y PLANE Z-600.0, R- 20.0, PHI- 0.0  
 FIELD COMPONENTS IN K-GAUSS

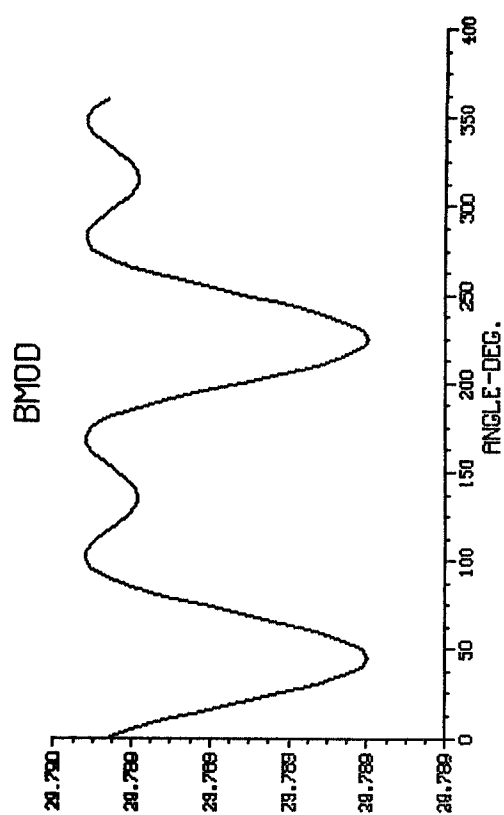
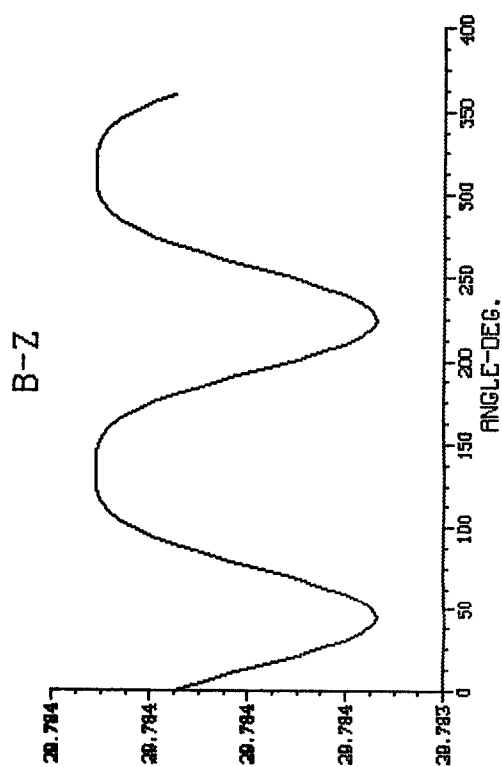
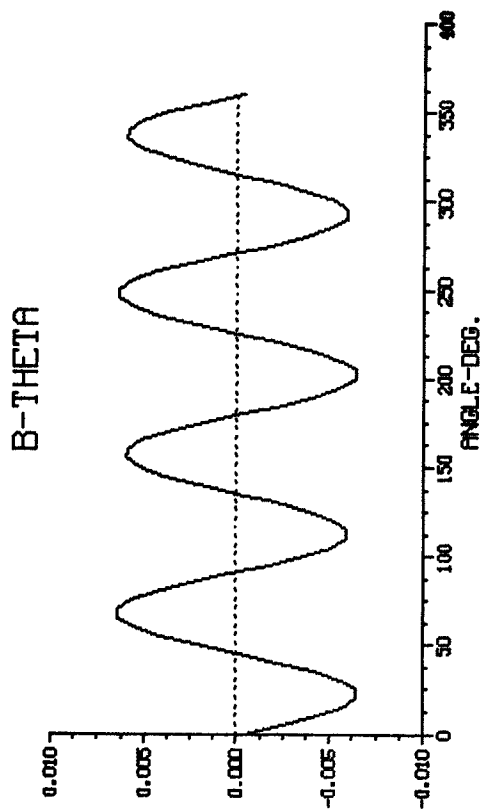
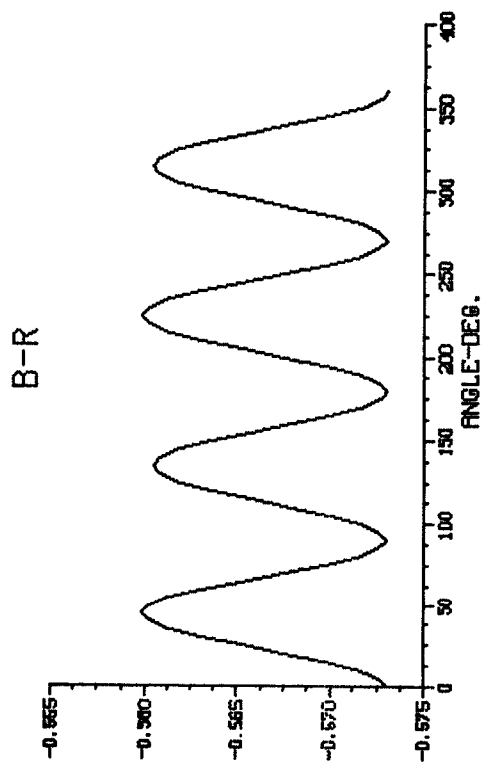


Fig. V.9-20. Field components as in Fig. V.9-19, but with the presence of the symmetrical square blanket (Fig. V.9-16).

X-Y PLANE Z=600.0, R= 20.0, PHI= 0.0

FIELD COMPONENTS IN K-GAUSS

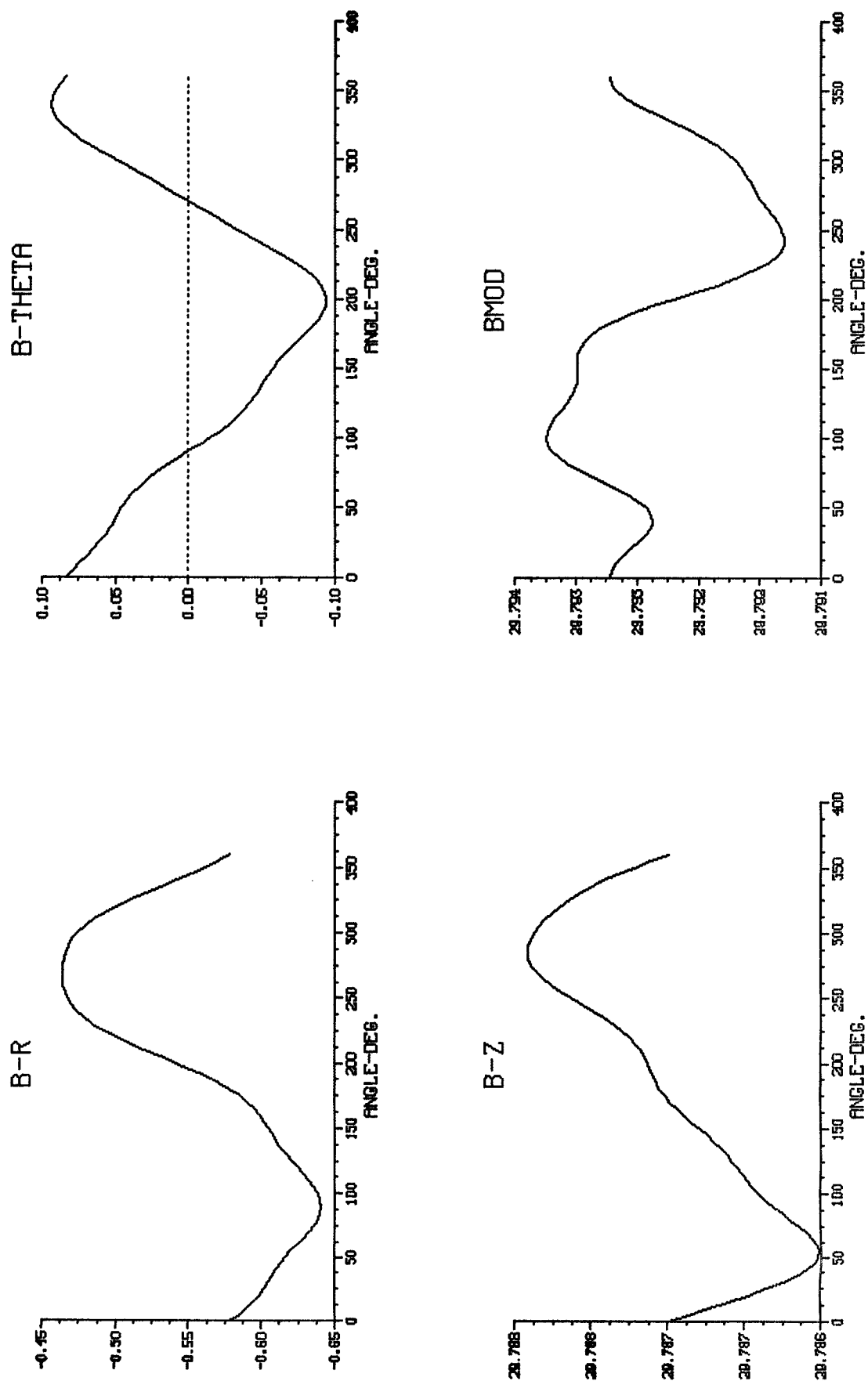


Fig. V.9-21. Field components as in Fig. V.9-19, but with the presence of the asymmetric rectangular blanket (Fig. V.9-17).

- b. if the length is of the order of the outer diameter or width, then the two end zones overlap and the perturbation extends over the entire length of the blanket;
- c. the use of a homogenized magnetization is indeed justified;
- d. the temperature dependence of the saturation magnetization can be accounted for by using an average magnetization;
- e. the field perturbation near the ends of the blanket depends however strongly on the configuration, the symmetry, and the material distribution of the blanket cross section.

#### References for Section V.9

- 1. H.G. Armstrong, et al., "GFUN3D User Guide", Rutherford Laboratory, United Kingdom, RL-76-029, 1979.
- 2. W.F. Brown, Jr., "Magnetoelastic Interactions", Springer-Verlag, NY, 1966.
- 3. W.M. Stacey, Jr., "Fusion Plasma Analysis", John Wiley & Sons, NY, 1981.
- 4. S.J. Sackett, "User Manual of SAP4," Lawrence Livermore Laboratory, UCID-18226, 1979.

## V.10 Conclusions

The following conclusions may be drawn for the TASKA magnet configurations:

- The DC nature of tandem mirror magnets is a big advantage compared with pulsed magnets or DC magnets with pulsed loads as in tokamaks. Fatigue is, therefore, not a limitation in this case.
- The technology of the central cell magnets and the end plug magnets is available today. This is proven by big bubble chamber magnets and the successful yin-yang test at Livermore.
- The technology of the barrier coils needs further development with respect to  $\text{Nb}_3\text{Sn}$  conductor technology in conjunction with hybrid coil inserts.
- Further investigations are required for radiation-resistant insulation materials as needed in the normal conducting insert of the barrier coil.

## VI Central Cell

### VI.1 General Design

#### VI.1.1 Introduction

The central cell in a tandem mirror can be regarded as the heart of the system, that is, where the useful energy is produced. Most of the fusion reactions take place in the central cell and thus it is here that fuel can be bred, thermal energy extracted and materials radiation damage tests performed. Although the end plugs are indispensable to the operation of the tandem mirror, they really play a supporting role by insuring the proper conditions within the central cell for fusion reactions to take place.

#### VI.1.2 Description of Central Cell

The central cell in TASKA consists of a long cylindrical structure which surrounds the plasma and is made up of a blanket, vacuum chamber, reflector and shield. This cylindrical structure fits inside three superconducting solenoids which provide the confining field within the central cell. On either end of the central cell there is an ICRF heating section, each consisting of four launching antennae. These antennae are recessed within the first wall and, therefore, do not intersect the path of charged particles streaming out toward the end plugs. The central cell coils are supported independently of the blanket/shield and are capable of being translated axially on tracks, to provide access for blanket modules located within the bore of the coils. Coolant inlet and outlet tubes, and cables are routed down into the reactor basement to free the space above for remote handling machines and the overhead crane.

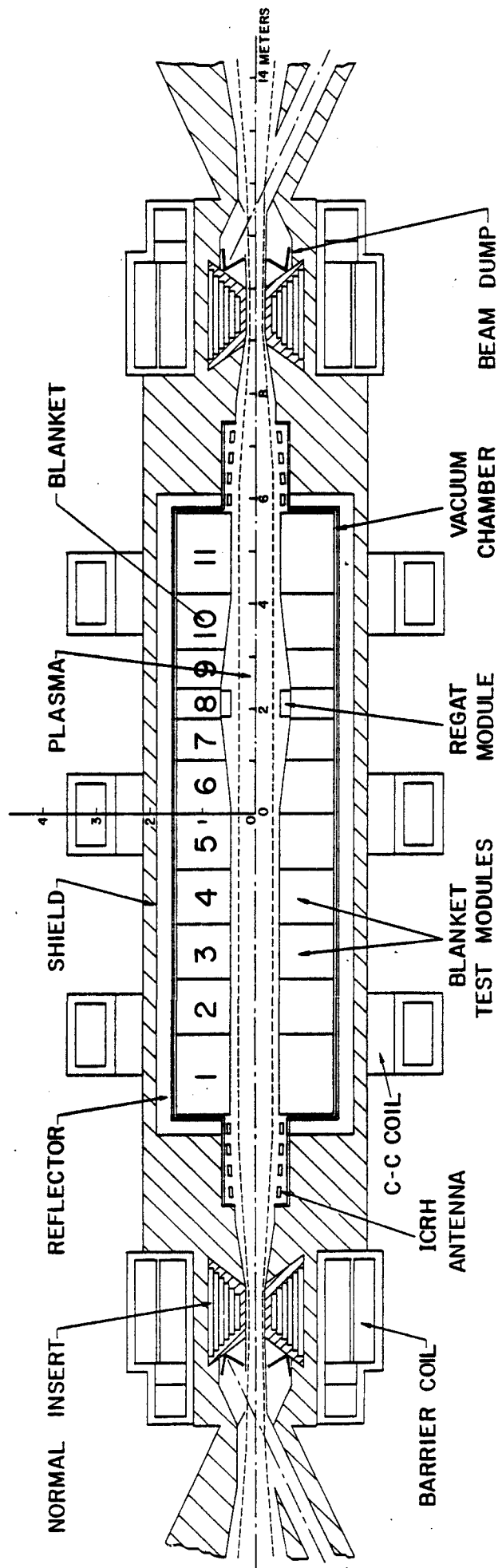
The primary mission of TASKA is to provide a test bed for materials and blanket system components. In order for it to be self sufficient with respect

to tritium, the blanket must have a breeding ratio greater than unity. Thus, the primary function of the central cell is to generate neutrons for the test bed. The blanket is provided for the purpose of breeding enough tritium as to make the reactor self sufficient. In a power reactor, the blanket energy is used to generate electricity. However, in TASKA, a small fraction of the blanket energy will be used to generate electricity for demonstration purposes only. The remaining energy will be disposed of.

Figure VI.1-1a shows a layout of the TASKA central cell with the blanket modules numbered 1-11 and Fig. VI.1-1b is another view of the central cell showing the intercoil structure. It can be seen that there are modules of several different shapes and lengths. The modules adjacent to the ICRF zones (1 and 11) are oversized whereas those adjacent to the REGAT module (7 and 9) are undersized. Blanket modules 2, 5, 6, and 10 have the same axial length, whereas modules 3 and 4, designated as blanket test modules, are somewhat shorter.

The inner surfaces of blanket modules 6, 7, 9, and 10 are tapered. This was done in order to enhance and more uniformly distribute the neutron flux to the REGAT module (see Chap. VII for details). Table VI.1-1 gives the room temperature dimensions and the locations of each module. Axial distance is measured from the center of the central cell and negative numbers indicate locations to the left of the centerline. The spaces between blanket modules are also indicated on the table. It will be noted that the spaces between blanket modules 1 and 2, and 10 and 11 are 1.5 cm whereas all the other spaces are 1.0 cm. This allowance is made to account for the greater expansion of the oversized modules.

# TASKA CENTRAL CELL LAYOUT



VI.1-3

Fig. VI.1-1a Numbered blanket modules to correspond with Table VI.1-1.



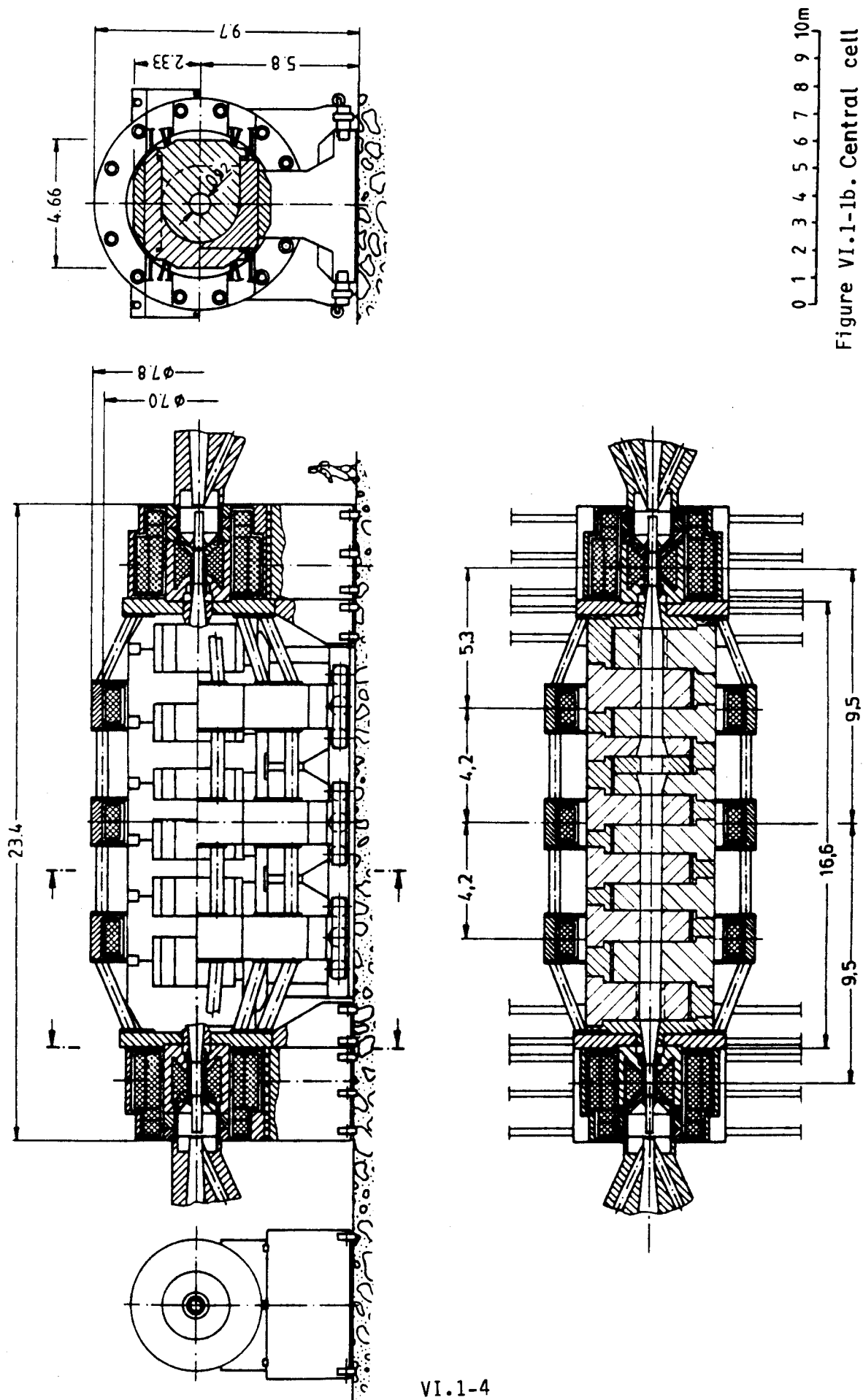


Figure VI.1-1b. Central cell

Table VI.1-1  
Blanket Module Dimensions and Locations  
in the Central Cell

| Blanket<br>Module<br>Number | Location of<br>Module Center<br>Line (cm) | Axial<br>Length<br>(cm) | Inner<br>Min.<br>(cm) | Radius<br>Max.<br>(cm) | Outer<br>Radius<br>(cm) |
|-----------------------------|-------------------------------------------|-------------------------|-----------------------|------------------------|-------------------------|
| 1                           | -499.25                                   | 156.5                   | 46                    |                        | 146                     |
| Space                       |                                           | 1.5                     |                       |                        |                         |
| 2                           | -367.50                                   | 104.0                   | 46                    |                        | 146                     |
| Space                       |                                           | 3.0                     |                       |                        |                         |
| 3                           | -262.00                                   | 101.0                   | 46                    |                        | 146                     |
| Space                       |                                           | 3.0                     |                       |                        |                         |
| 4                           | -158.00                                   | 101.0                   | 46                    |                        | 146                     |
| Space                       |                                           | 3.0                     |                       |                        |                         |
| 5                           | -52.50                                    | 104.0                   | 46                    |                        | 146                     |
| Space                       |                                           | 1.0                     |                       |                        |                         |
| 6                           | 52.50                                     | 104.0                   | 46                    | 64                     | 146                     |
| Space                       |                                           | 1.0                     |                       |                        |                         |
| 7                           | 142.25                                    | 73.5                    | 64                    | 73.5                   | 146                     |
| Space                       |                                           | 1.0                     |                       |                        |                         |
| 8                           | 210.00                                    | 60.0                    | 66                    |                        | 146                     |
| Space                       |                                           | 1.0                     |                       |                        |                         |
| 9                           | 277.75                                    | 73.5                    | 64                    | 73.5                   | 146                     |
| Space                       |                                           | 1.0                     |                       |                        |                         |
| 10                          | 367.50                                    | 104.0                   | 46                    | 64                     | 146                     |
| Space                       |                                           | 1.5                     |                       |                        |                         |
| 11                          | 499.25                                    | 156.5                   | 46                    |                        | 146                     |

Blanket No. 8 is the one which backs up the REGAT module. This is a non-breeding blanket and will contain the same material composition as the shield.

The blanket modules are surrounded by a vacuum chamber made of 4 cm thick 316 stainless steel. There is a 5 cm gap between the blanket and the vacuum chamber. Rectangular flanges attached to the vacuum chamber provide the closures for the blanket modules. Adjacent blanket modules have flanges on opposite sides of the vacuum chamber such that the flanges do not overlap each other. Thus, adjacent blanket modules are inserted and removed from the vacuum chamber from opposite sides. Since blanket module removal from the vacuum chamber is in the horizontal direction, the flange surface is always oriented vertically. The only exception is in the REGAT module (No. 8) which is designed for vertical removal. The flange surface for this module is oriented horizontally at the top of the vacuum chamber.

The vacuum chamber is followed by a 28 cm thick reflector zone consisting of water cooled 316 stainless steel. The function of the reflector is to reflect neutrons back into the blanket. Each blanket module has the vacuum chamber wall combined with the reflector as the module closure which seals against the flange. This portion of the reflector is attached to and is a permanent part of the blanket module as can be seen in Fig. VI.1-2.

The last element in the central cell is the shield which is needed to protect the central cell solenoids. The shield is 25 cm thick and is composed of ferritic steel structure,  $B_4C$ , Pb and water cooling. The shield part adjacent to the blanket module closure is also attached to the module as shown in Fig. VI.1-2. Table VI.1-2 gives the central cell engineering parameters.

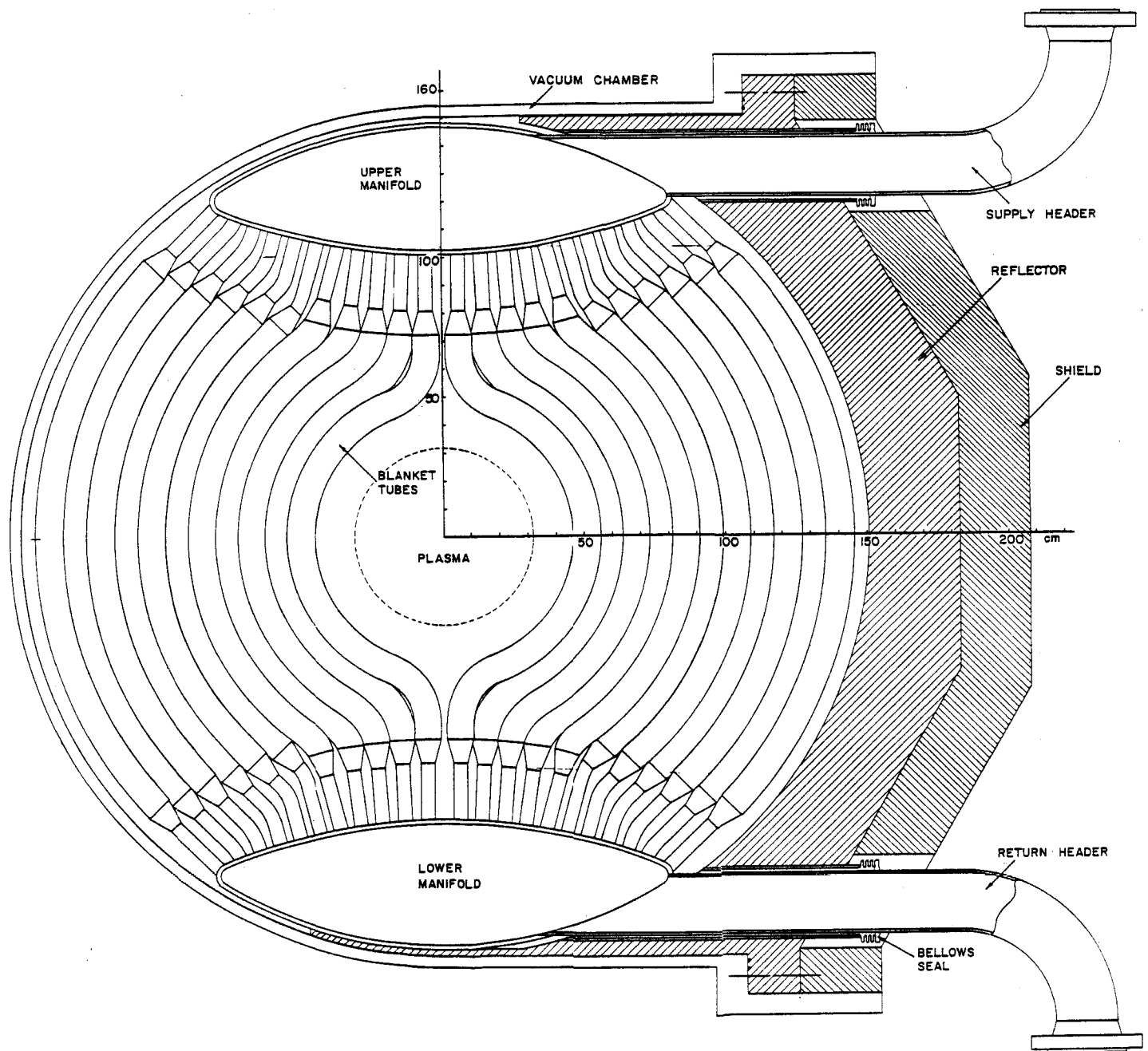


Fig. VI.1-2. Vertical section of TASKA blanket

Table VI.1-2. Central Cell Engineering Parameters

|                                                        |       |
|--------------------------------------------------------|-------|
| *Central cell length (Physics definition) (m)          | 19.2  |
| **Central cell length (Engineering definition) (m)     | 16.7  |
| Blanket inner diameter (m)                             | 0.92  |
| Permanent breeding zone length (m)                     | 8.76  |
| ICRH zone length (m)                                   | 3.07  |
| Blanket test module length (m)                         | 2.02  |
| Material test module length (m)                        | 0.60  |
| Breeding blanket first wall area (m <sup>2</sup> )     | 25.32 |
| ICRH launching area (m <sup>2</sup> )                  | 8.49  |
| Blanket test module first wall area (m <sup>2</sup> )  | 5.84  |
| Material test module first wall area (m <sup>2</sup> ) | 1.44  |
| Blanket thickness (m)                                  | 1.0   |
| Vacuum chamber inner diameter (m)                      | 3.02  |
| Vacuum chamber wall thickness (m)                      | 0.04  |
| Reflector thickness (m)                                | 0.28  |
| Shield thickness (m)                                   | 0.25  |

\*Distance between magnetic peaks.

\*\*Distance between inner edges of superconducting barrier coils.

### VI.1.3 Evolution of TASKA Blanket Design

The TASKA blanket design has undergone several iterations. Although there have been minor modifications along the way, there have been four major design changes since the first proposed blanket. These changes are shown graphically in Fig. VI.1-3 and are designated A-E. Heavy arrows indicate the direction of blanket module removal from the central cell. The reasons for the design changes are given in capsule form on Fig. VI.1-3 and will be further discussed in the following sections.

The blanket initially proposed for TASKA (designated "A" in Fig. VI.1-3) was similar to that used in WITAMIR,<sup>(1)</sup> a tandem mirror power reactor conceptual design completed at the University of Wisconsin in 1980. In this blanket, the breeding material came into a manifold at the top, then flowed down through tubes on both sides of the plasma, finally collecting in the bottom manifold where upon it was pumped to a steam generator. The removal scheme for these blanket modules was from the top. This was practical in WITAMIR because both shield enclosure and the reactor building were evacuated. There was no separate vacuum chamber. Blanket modules were removed in designated locations along the central cell and were translated axially to these locations within the shield.

Such a scheme did not seem practical for a materials test reactor, where frequent access to the plasma chamber will be needed. For this reason it was decided to provide TASKA with a separate vacuum chamber, and because it would be difficult to design overlapping closures for top removal of all the modules, it was proposed that alternating sideways removal be adopted instead. This led to the design designated "B" in Fig. VI.1-3.

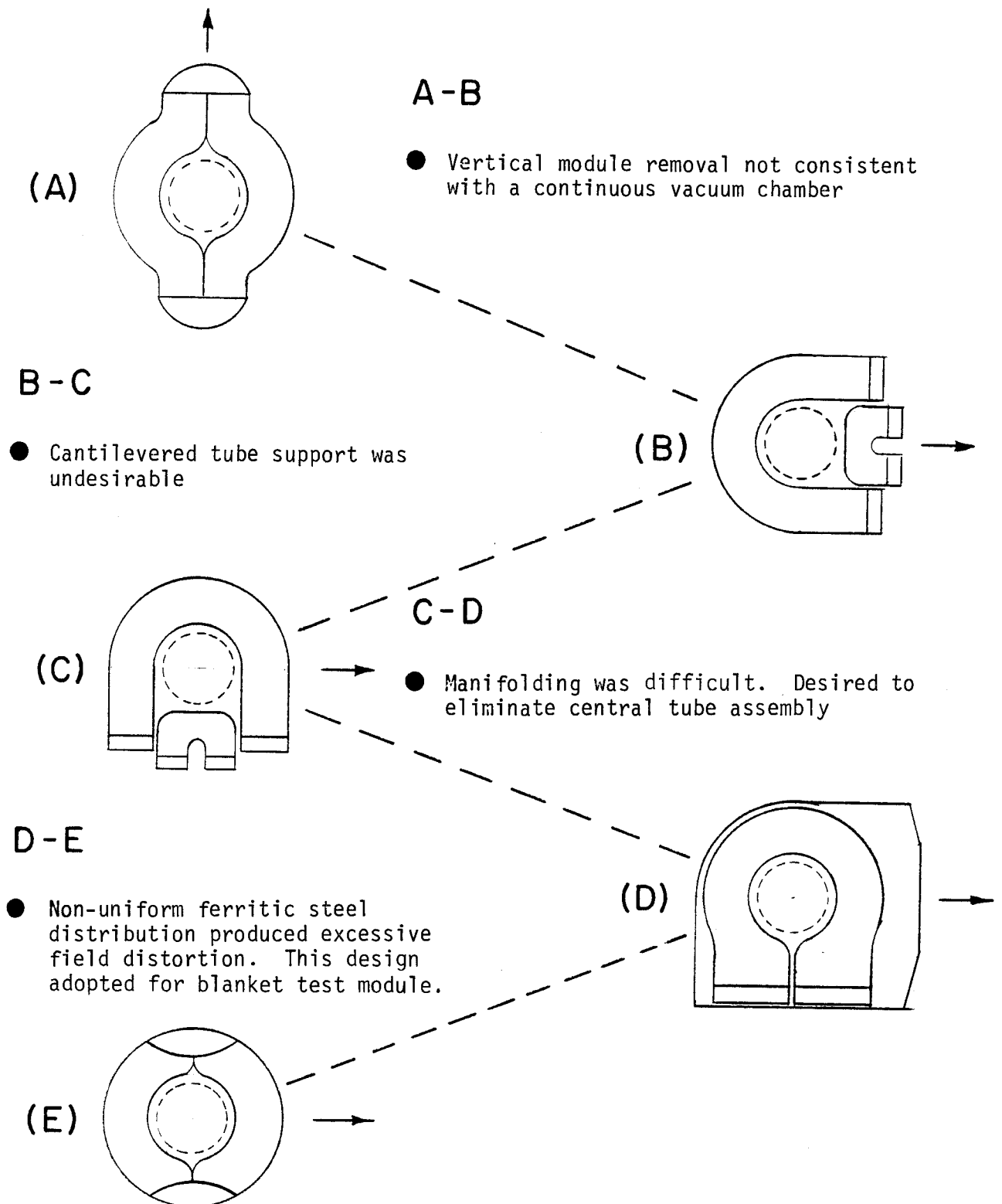


Fig. VI.1-3. Evolution of TASKA breeding blanket design

The new design retained the tubular structure but in a U-shaped configuration oriented sideways. The coolant thus flowed around the plasma on three sides and a separate segment of tubes was needed to fill the void in the U and cover the fourth side. The undesirable feature of this design was the unsupported tubes which were essentially cantilevered. It was also recognized that the MHD pressure drop would be somewhat higher in this design, although the effect was not that great. A solution to the unsupported tubes problem was proposed in which essentially all the features of the "B" design were retained but the blanket was rotated 90°, such that the U shape was oriented vertically with the open space on the bottom. This design is designated "C" in Fig. VI.1-3.

The "C" design was carried along in the course of the study for a long time and it will be noticed that many of the analyses in the TASKA report are predicated on it. The transition to "D" came in the final detailed blanket design version. In the detailed design it was concluded that it was difficult to manifold the additional segment of tubes needed to close the U. The new design "D" has tubes circumventing the plasma completely such that the additional segment of tubes was no longer needed. Furthermore, the new design had a casing which completely enclosed the blanket tubes on all sides. The casing was stepped on all sides in such a way as to prevent primary neutron streaming and was designed to match the steps in the adjacent blanket modules inserted from the opposite direction.

Design "D" appeared to be destined for final adoption except for one flaw. Since the material proposed for the blanket was the ferritic steel HT-9, the magnetic field code for ferritic steels, GFUN, (which became operational late in the study) predicted major field distortions in the plasma envelope due to



the grossly uneven distribution of steel in the "D" design. One other problem with the "D" design was the potential for neutron streaming through the space between manifolds, however, appropriately placed shield material can circumvent this problem. This design was adopted for the blanket test module (see Section VII.3).

Another design of the blanket subject to a near uniform distribution of ferritic steel and the constraints of a fixed blanket material composition resulted in the "E" design. This design comes as close as possible to a near uniform distribution of ferritic steel while still retaining the good features of the "C" and "D" designs. Although the casing has been eliminated, the tube and manifolds are stepped in such a way as to prevent streaming. A detailed design has been performed and will be discussed in the next section.

#### VI.1.4 Mechanical Description of Central Cell Components

##### VI.1.4.1. Blanket Design

The breeding blanket design for TASKA is shown in Fig. VI.1-2 and Fig. VI.1-4.

Figure VI.1-2 is a vertical cross section through the center of a module showing eleven rows of tubes on either side of the plasma. The end of each tube is attached to a smaller diameter tube by means of a conical transition section. The small diameter tube is then attached to the manifold.

The innermost tubes face the plasma on one side and are, thus, an effective first wall. The sharp bend radius toward the manifold is equal to four times the tube radius which is consistent with the minimum bend radius that can be produced without buckling the tube wall. All the other tubes have bend radii which are larger.

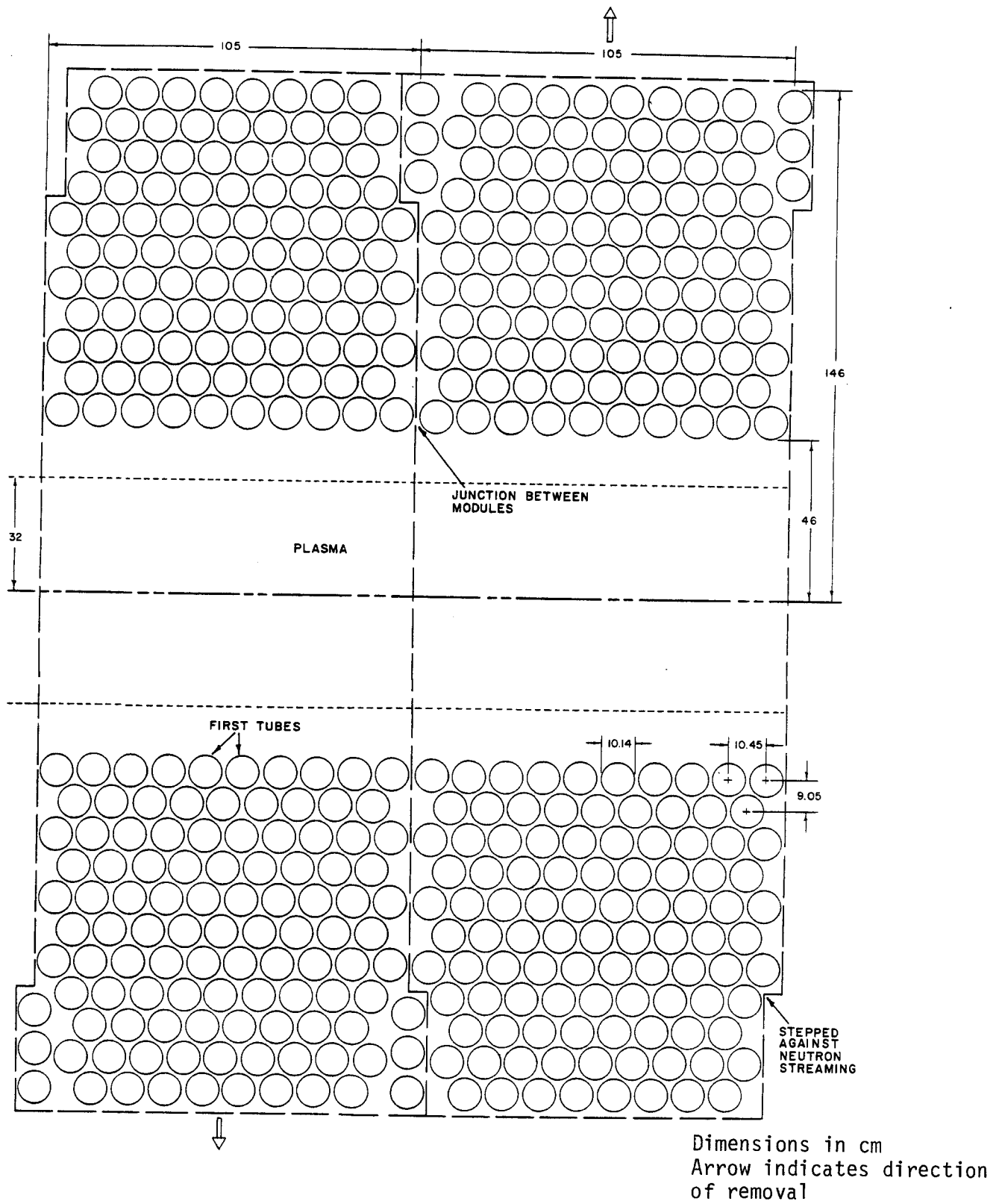


Fig. VI.1-4 Horizontal section through two blanket modules.

The outer diameter of the tubes is 10.14 cm and the wall thickness is 0.227 cm. The transition tubes are 5 cm OD and have a wall thickness of .175 cm. The axial centerline distance between tubes is 10.45 cm and the radial centerline distance is 9.05 cm.

Figure VI.1-4 is a horizontal cross section through two adjacent modules showing the stepping in the tubes where the modules interface. It also shows the staggering of tubes needed to allow the two modules to slide relative to each other at this interface. The missing tubes add to the void fraction within the blanket. The actual volumetric fractions in the blanket were based on a segment extending from one module centerline to the other, thus covering the interface between them. These fractions are 73% breeding material, 7% HT-9 structure and 20% void. The same proportions were also maintained in the blanket quadrant covering the manifold.

The modules shown in Fig. VI.1-4 have 104 cm axial length and have a blanket thickness of 1 m. The radius to the front surface of the first tube is 46 cm. There are ten tubes in the front row and a total of 208 tubes in each module. A separation of 1 cm per meter was allowed between modules at room temperature. At operating temperature the gaps close to  $\sim 0.4$  cm. Modules will be keyed in the center such that they will expand equally in both directions axially. From Fig. VI.1-1 it can be seen that there are three different axial lengths of blanket modules. For example, modules 1 and 11 are 156.5 cm long and have 15 tubes in the front row, whereas modules 7 and 9 are 73.5 cm long, and, if they were not conical, would have had seven front tubes.

This brings up the question of how a tapered effect can be produced with the proposed tubular construction. It will be recalled that the tapered blanket shape is needed to enhance and more uniformly distribute the neutron

flux to the REGAT module. Figure VI.1-5 shows how tubes can be eliminated in progressive rows in order to achieve this effect. A smooth tapered wall is not necessary to achieve the enhancement effect, rather a more extensive view of the plasma is needed, with reflection from the bulk of the blanket behind the tapered surfaces.

The manifolds in each module consist of two arcs joined at the points of intersection. The manifold walls are 1.7 cm thick. It is necessary to step the top half of the manifolds in order to prevent neutron streaming in the vertical direction. This is illustrated in Fig. VI.1-6. Table VI.1-3 gives all the pertinent mechanical parameters of the TASKA blanket.

A few words must be said about the question of whether or not casings are needed for the blanket modules. The only argument in favor of a casing for the blanket modules is the vacuum argument. This is a very important issue which cannot be ignored or deemphasized. Arguments against providing a casing are many, not the least of which is the design complication created by the need for cooling the casing. Thus, it appears that the need for a casing can only be justified on the basis of the vacuum requirement.

As far as vacuum requirements go, the casing performs two functions:

1. It reduces the exposed surface area which may be a source of impurities.
2. It isolates the long narrow channel between the tubes from the plasma chamber.

It is clear that in fabricating the blanket good high vacuum system practices have to be followed. This implies that weldments must be full penetration, leaving no reentrant cracks which can be sources of impurities. Furthermore, the blanket module must undergo a series of cleaning operations such as degreasing and pickling. After this, the blanket module must be baked

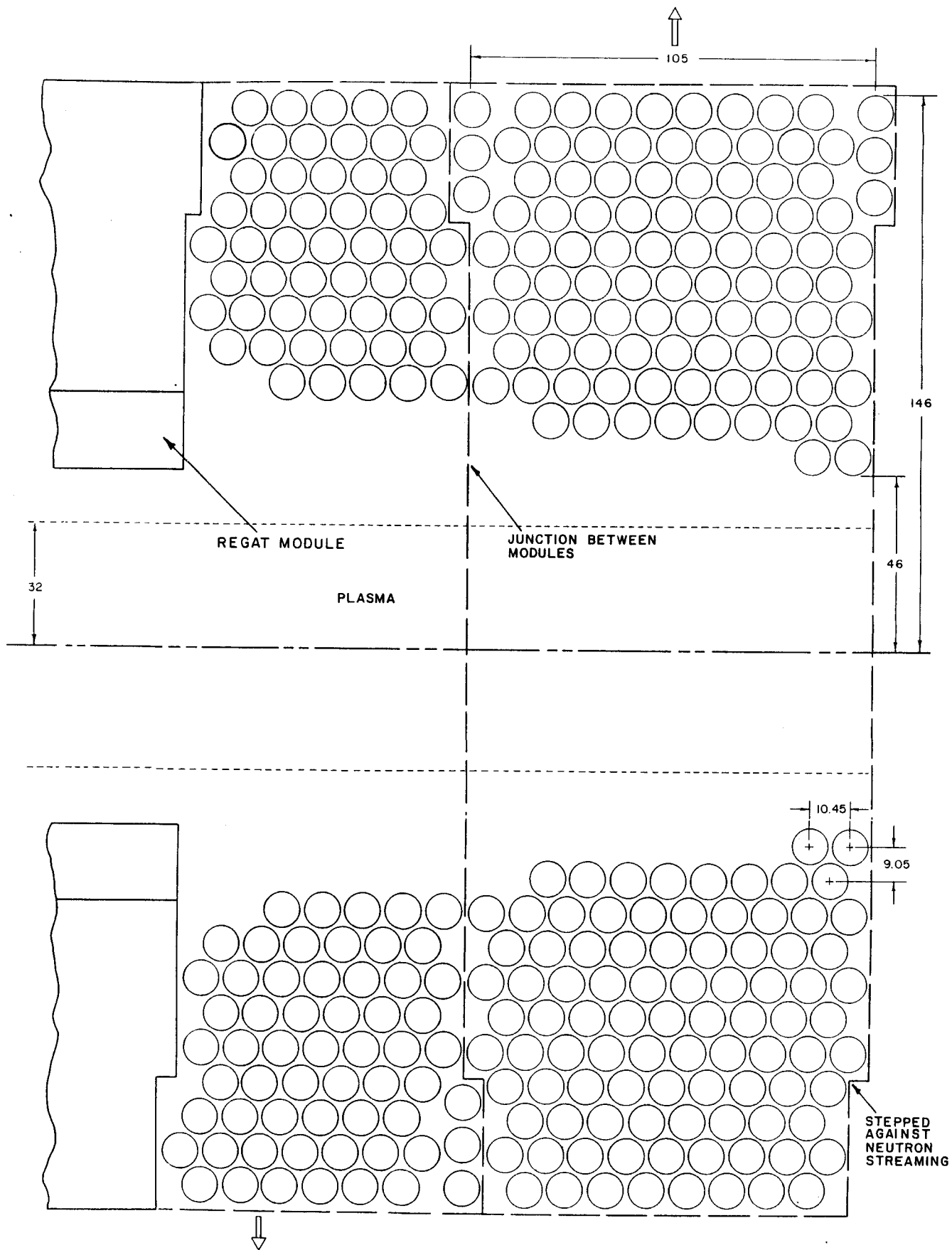


Fig. VI.1-5 Blanket shaping near REGAT modules.

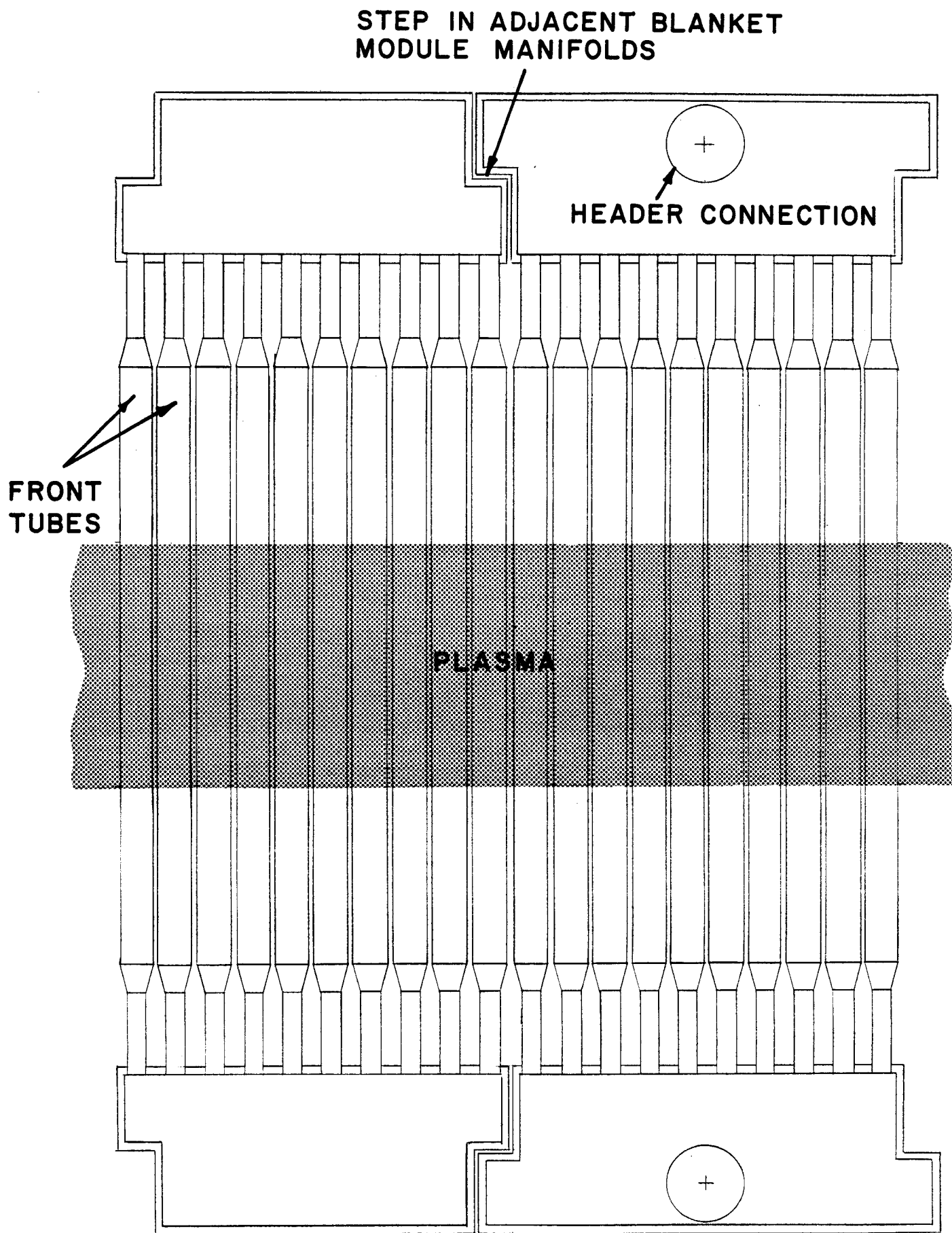


Fig. VI.1-6 Vertical section in the axial direction showing manifold stepping.

Table VI.1-3. Pertinent Mechanical Blanket Parameters

|                                                                             |       |
|-----------------------------------------------------------------------------|-------|
| Blanket material                                                            | HT-9  |
| Tube outer diameter (cm)                                                    | 10.14 |
| Tube wall thickness (cm)                                                    | 0.227 |
| Axial tube centerline distance (cm)                                         | 10.45 |
| Radial tube centerline distance (cm)                                        | 9.05  |
| Transition tube outer diameter (cm)                                         | 5.0   |
| Transition tube wall thickness (cm)                                         | 0.175 |
| Number of tubes per module                                                  | 108   |
| Manifold wall thickness (cm)                                                | 1.7   |
| Tube surface area/m of cc ( $\text{m}^2/\text{m}$ )                         | 140   |
| Volume of $\text{Li}_{17}\text{Pb}_{83}$ /m of cc ( $\text{m}^3/\text{m}$ ) | 4.45  |
| Mass of structure/m of cc ( $\text{kg}/\text{m}$ )                          | 3293  |

in a special oven at a temperature of  $\sim 600^{\circ}\text{C}$  for 10-20 hours. Transfer of the module to the reactor must be under conditions which will insure cleanliness. In situ backing of the blanket modules while the chamber is evacuated will also be necessary. With such preparation, the outgassing rate of mild steel can be reduced to  $\sim 10^{-9} \text{ Pa m}^3/\text{sec m}^2$ .<sup>(2)</sup>

The total tube surface area in TASKA is  $\sim 1600 \text{ m}^2$ . Using this outgassing rate the total gas throughput from the exposed surface is  $1.3 \times 10^{-5} \text{ torr l/s}$ . The total vacuum throughput of TASKA will be on the order of 10-100 torr l/s. It is evident that the level of impurities from the wall will be no more than  $1 \times 10^{-3}\%$  of the throughput. Furthermore, the bulk of the contaminants will be CO or  $\text{CO}_2$  which will dissociate at 100 eV and will travel within the plasma halo out of the reactor.

The second area of concern is the evacuation of the spaces between the tubes. In the case of the TASKA design, the space is equivalent to a 2 cm diameter tube, with the longest path being 1 m long. This  $\frac{L}{R}$  ratio of 100 gives a transmission probability of  $\sim .03$  if the sticking coefficient is assumed to be zero.<sup>(3)</sup> This effectively means that if the bulk of the chamber is evacuated to  $10^{-7} \text{ torr}$ , the innermost recesses of the spaces between the tubes will only reach a pressure of  $3 \times 10^{-6} \text{ torr}$  as long as there is a source of molecules present. Since at these pressures there will be several monolayers of molecules on the surface, a sticking coefficient of zero is reasonable to assume.

The initial baking and evacuation of the chamber after it had been exposed to atmospheric pressure will, therefore, require a long time, perhaps as long as 24-48 hours. However, once the reactor is in operation, the surfaces will be covered with hydrogenous species and helium. It would not be



detrimental to the operation of the reactor if the hydrogenous species pressure within the spaces between the blanket tubes was somewhat higher than in the plasma scrapeoff zone.

It is significant to note that experience on TMX has shown that only surfaces in close proximity to the plasma are sources of impurities whereas surfaces further removed do not seem to contribute. If this is true, then a casing will only be marginally helpful or not at all.

Although a casing can be incorporated into the present design, for the present we feel that it will not be needed. This brings up the question of how a leak can be determined in a blanket system without casings. Obviously, each module inlet and outlet coolant headers will require shut-off valves to provide the capability of isolating any one module from the rest of the system. Further, a capability must be provided for pressurizing each individual blanket with a suitable gas which can be detected with a mass spectrometer. After the breeding material is drained out of the modules, they can be individually pressurized and the leak isolated.

As in the previous designs, the blanket modules will be supported on the bottom, with the load path being through the vacuum chamber wall to the reflector and shield. Finally, the load is transferred to the structure attached to the floor of the reactor hall. Because of the substantial load which must be transferred through the bottom manifold, it will require some internal reinforcing. This reinforcing will be in the form of plates oriented perpendicular to the axis of the reactor. Care must be exercised to insure that the plates will not impede the flow of breeding material or add substantially to the overall system pressure drop.

Just looking at a normal blanket module of 104 cm axial length, it is interesting to note that the mass of the breeding material will be ~ 43.6 tonnes and the structure ~ 3.4 tonnes. These values do not include the masses of reflector and shield which will be part of the module. Nevertheless, once the breeding material is drained out of the module, there will be a substantial reduction in mass.

A few words must be said about the fabrication of such a blanket. We feel that the fabrication is relatively simple and well within present day technology.

The design calls for only 13 shapes of tubes. Fabrication of the tubes themselves will be fairly simple and can be made on a production line basis. The conical connectors are identical for all the tubes; however, the lengths and bends in each of the 13 shapes are different. A common method for bending tubes of this nature is to fill them with a soft material such as indium or woods metal, and then put them through dies which progressively bend them until the desired shape is achieved. The woods metal is then melted out and the tube cleaned in preparation for welding.

Joining sections of tubes together is performed with external welding. However, joining the tubes to the manifold plate will be done from the back side. Fig. VI.1-7 is a full scale drawing of how this can be accomplished. Notice that there are no reentrant gaps on the side where the tube is attached to the manifold. It is absolutely essential that there be no reentrant gaps at the points of attachment which can become sources of trapped gas.

It will also be noticed that the penetration into the manifold is always perpendicular to the tangent at that point. This will facilitate the assembly of the modules. The holes into the manifold plate are also made conical, with

Welding is done from this side

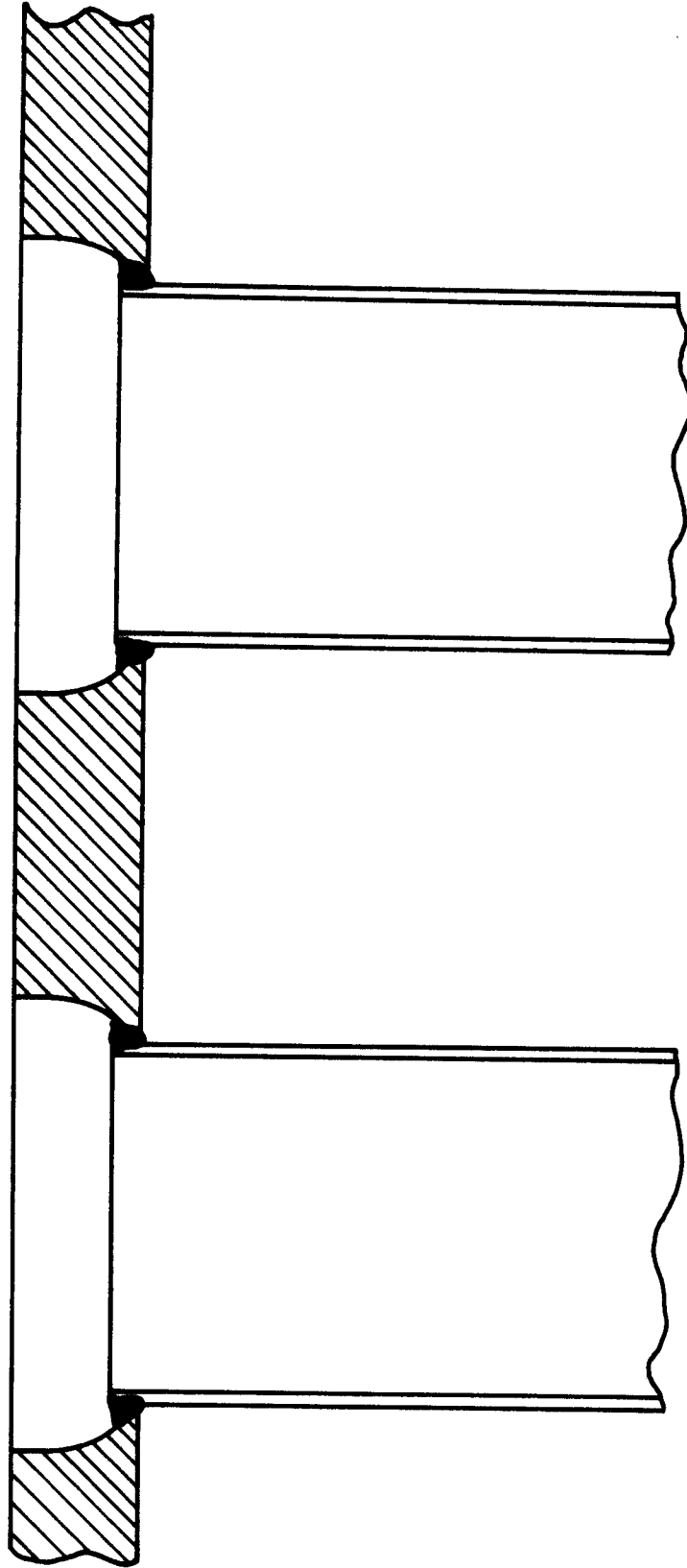


Fig. VI.1-7 Full scale drawing of how blanket tubes are welded to the bottom manifold plate.

the tube penetrating only part of the way. This will make it possible to insert the tubes individually rather than trying to assemble the whole module all at once.

When the tubes are all welded, the back side of the manifold is assembled and welded. The final step in the blanket assembly is the attachment of the header to the manifold. This is accomplished by welding from the outside. Finally, the assembly will be heat treated.

#### VI.1.4.2 Vacuum Chamber Design

The vacuum chamber in TASKA is a water cooled 4 cm thick walled cylinder, 151 cm internal radius. It is made of 316 stainless steel and surrounds the blanket. The chamber has rectangular flanged openings which alternate from one side to the other along the length of the central cell. These openings allow the blanket modules to be inserted and removed from the vacuum chamber. The part of the reflector which is attached to the blanket module makes a seal with the vacuum chamber flange as shown in Fig. VI.1-2.

In the vicinity of the ICRH antennae, the chamber necks down to a radius of 59 cm at a distance of 582.5 cm from the center of the reactor. This radius is maintained for 165 cm after which another necking down to a radius of 37 cm occurs. At this point the chamber is attached to the transition section leading to the barrier coil.

#### VI.1.4.3 Reflector Design

The reflector in TASKA is a 28 cm thick water cooled slab of 316 stainless steel. Actually there are three segments to the reflector as shown in Fig. VI.1-8:

1. The segment which constitutes the closure in each module.
2. The bottom segment which is part of the support path for the blanket.

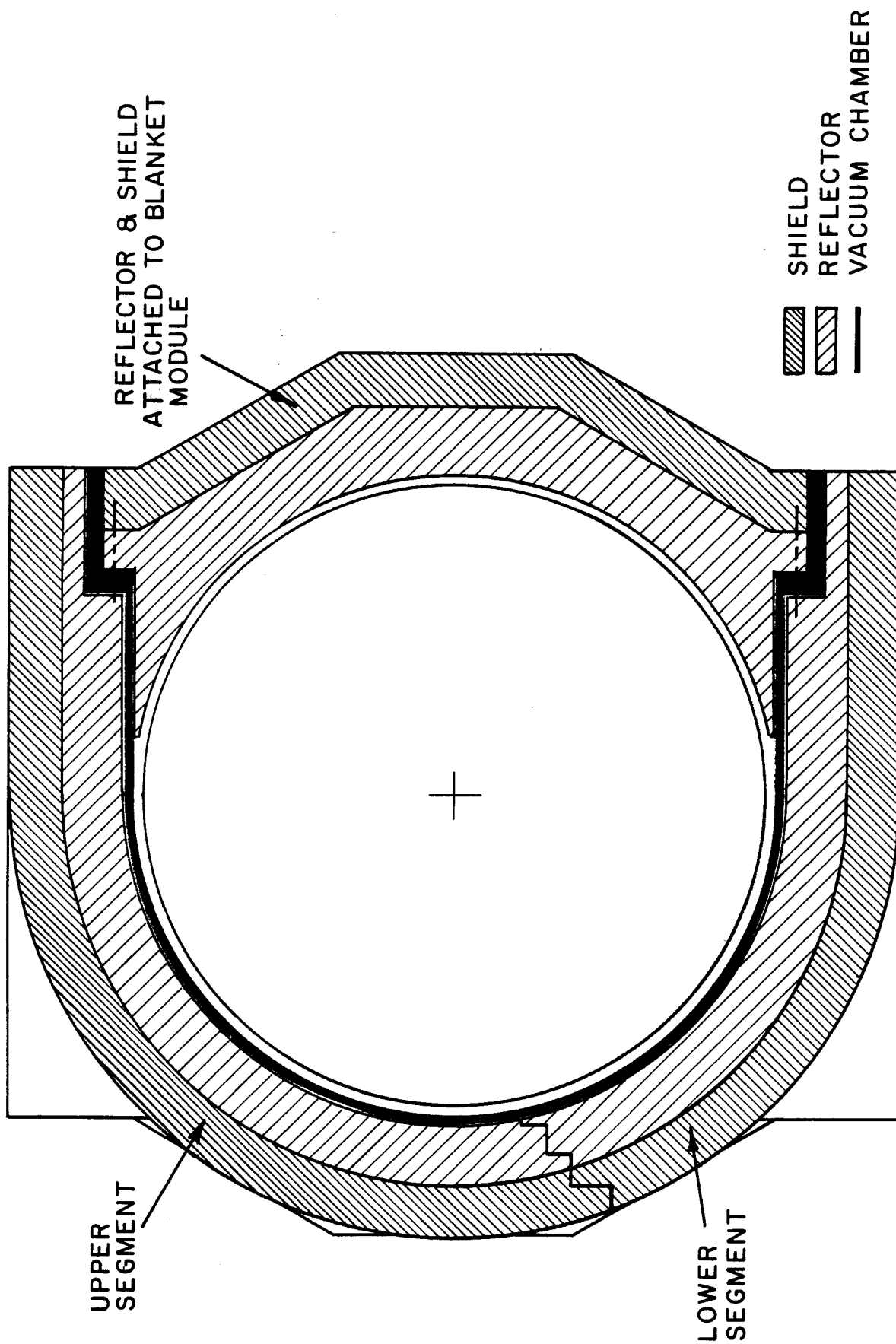


Fig. VI.1-8 Reflector and shield segmentation.

3. The top and side segment. This segment is designed to be removable to make it consistent with the KfK central cell maintenance scheme.

Figure VI.1-9 shows how the reflector can be fabricated. It essentially consists of 20 cm thick sections of stainless steel that have cooling passages machined in the sides. Holes drilled through tube sections at either extremities will allow the cooling water to traverse the assembly from one end to the other. The assembly is then bolted together and the junctions between the slabs welded shut. The bolt head and nut penetrations are also welded shut. The number of cooling water traverses depends on the size of the segment. Large pieces may have several supply and return connections.

One complication arises in the area where the blanket header has to penetrate the reflector. Here the coolant must traverse the slab above (or below) the penetration and a sleeve is used to seal off the penetration as shown in Fig. VI.1-10. A bellows is then used to seal the header to the sleeve in the reflector.

The shape of the reflector which is part of the blanket module is shown in Fig. VI.1-3. It will be noticed that it is thicker than the specified 25 cm thickness. The reason for that is the fact that the 4 cm thick vacuum wall is included in the reflector. This segment of the reflector also constitutes the closure for the blanket module, for it is the seal in the flanges between the reflector and the vacuum chamber which is the vacuum barrier. It is envisaged that this will be a double metal seal with an evacuated space in between.

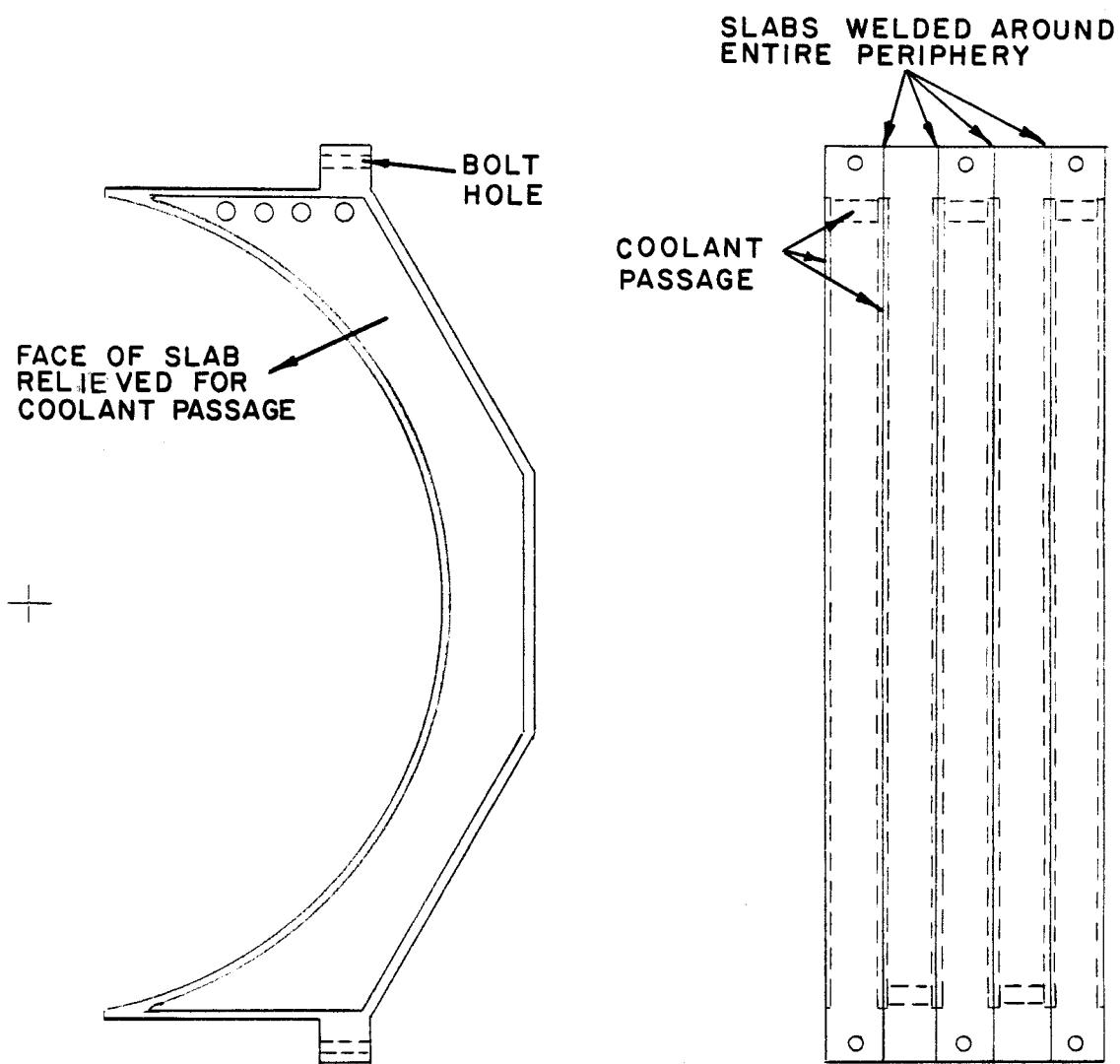


Fig. VI.1-9 Reflector fabrication

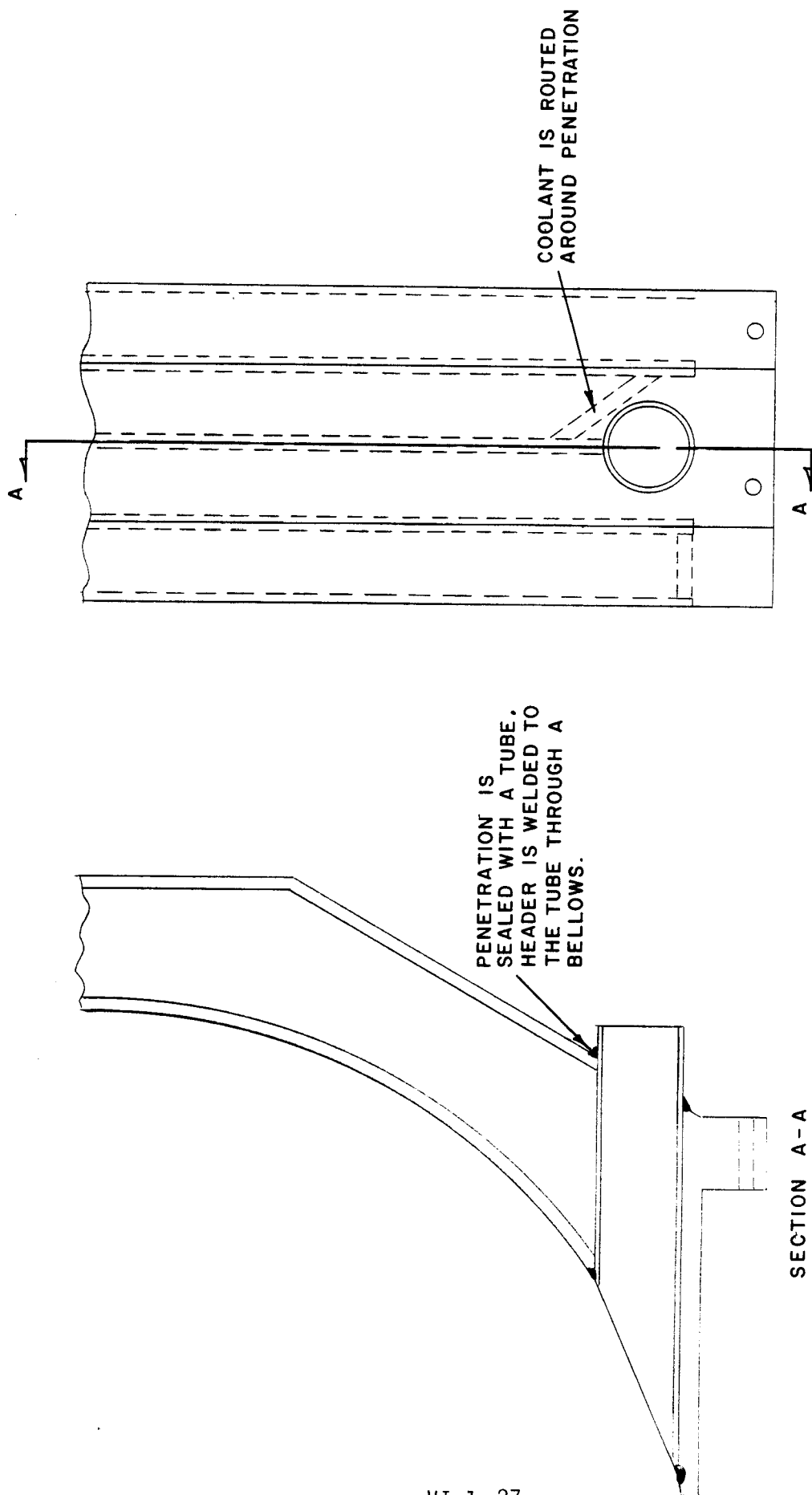


Fig. VI.1-10 Design of penetration through reflector closure for the blanket module.



#### VI.1.4.4 Shield

The shield in the central cell is 25 cm thick and is composed of 56% 1422 ferritic steel, 26%  $B_4C$ , 14% Pb and 4% cooling water. One possible shield geometry is shown in Fig. VI.1-11. The heavily ribbed structure can be oriented parallel to the chamber axis. The pockets will be filled with canned slabs of  $B_4C$  and Pb in the appropriate proportion with spaces in between for cooling water. The pockets are then capped off and welded shut.

Nuclear heating in the shield will be low enough to allow low velocity and low pressure water cooling. Thus, water traversing one shield pocket from one end to the other will be routed onto the next, snaking its way around the shield. In this way a minimum amount of coolant connections will be necessary.

The shield will also be divided into three segments, the same as the reflector (see Fig. VI.1-8). One segment will be permanently attached to the blanket module closure. The bottom segment constitutes some of the support structure for the central cell. The upper and the side opposite the blanket closure will be made into a continuous shield segment. An important aspect of the shield is to prevent neutron streaming, and to that end, the shield must have overlapping steps between adjacent shield segments.

The shield has to be self supporting, since the vacuum chamber is not strong enough to support it. Thus, the segments must be bolted together and must form a continuous self supporting structure.

Elsewhere in the reactor, the shield will vary in thickness from one place to another. With some variation, the same type of shield construction can be used in these areas as well.

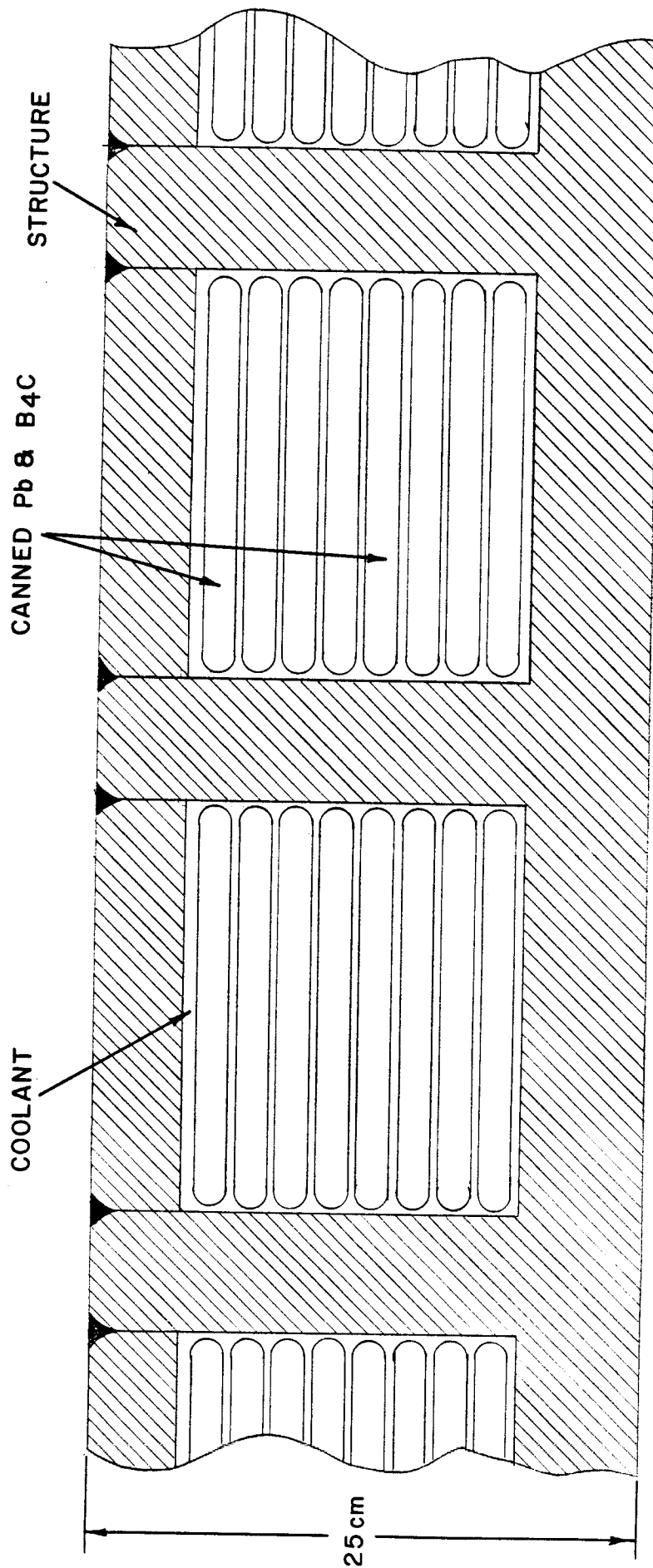


Fig. VI.1.11 Possible shield design.

#### References for Section VI.1

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2. L. Holland, W. Steckelmacher, and J. Yarwood, "Vacuum Manual", E&FN Spon, London.
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## VI.2 Neutronics

### VI.2.1 Design Goals

The neutronics design goals include adequate volumes and neutron fluences for the requirements of the materials and engineering test zones, tritium breeding, magnet and penetration shielding, and acceptable access to the building following shutdown. Energy production is given a low priority as achieving the other goals will ensure adequate heating for testing power conversion modules. The test zones are discussed in other sections of this report. The permanent blanket and shield are assessed in Section VI.2.2 with both one- and three-dimensional calculations. The special problems connected with shielding in the barrier and plug regions are taken up in Section VI.2.3. Radioactivity produced during operation is described in Section VI.2.4. A natural lithium test blanket design is discussed in Section VI.2.5.

### VI.2.2 Central Cell Permanent Blanket

It was felt that even though the power level of TASKA is modest and purchase of tritium from outside sources would be possible without an unmanageable economic impact, it would nevertheless be much more impressive if breeding was also achieved. This could easily be accomplished by a larger power rating where the test zones were a smaller perturbation on the central cell. As the reactor is made smaller while meeting the demands of the test zones, adequate breeding becomes difficult. The good breeding performance achieved with  $\text{Li}_{17}\text{Pb}_{83}$  in earlier studies made in connection with the WITAMIR<sup>(1)</sup> design led us to choose this material as the breeding material in TASKA. HT-9 was chosen as the structural material. As little HT-9 was used as possible while meeting structural demands in order to enhance breeding. A schematic of the central cell is shown in Fig. VI.1-1. The permanent blanket and its associ-

ated shield can be represented approximately by the one-dimensional schematic, Fig. VI.2-1. The general composition of the various regions are given in Table VI.2-1, including some that refer to a three-dimensional model of the entire central cell. The relatively large void fractions in the blanket are due to the tube configurations for the coolant-breeding material as shown in Fig. VI.1-4. The neutron and gamma fluxes for the one-dimensional model were calculated using the ONEDANT<sup>(2)</sup> discrete ordinates code on the MFE computer at LLNL. The data base is XSLIBA<sup>(3)</sup> (a 30 neutron - 12 gamma energy group cross section library<sup>(4)</sup>) and the problems were run in the  $S_4$ - $P_3$  approximation.

Key results from these calculations include the heating and radiation damage results for thermal hydraulic and materials design. The power density as a function of radius is shown on Fig. VI.2-2. The peak power density is  $11 \text{ W/cm}^3$  and falls rapidly into the blanket. The radiation damage as measured by the displacements per atom per full power year (dpa/FPY) as a function of the radius is shown in Fig. VI.2-3. It should be noted that the result is normalized to correspond to a wall loading of one megawatt per square meter. Thus if the time to reach a particular dpa is required, the result at a given radius using the dpa/FPY from the figure must then be divided by the actual wall loading in  $\text{MW/m}^2$  (1.52 in the present case) and multiplied by the capacity factor. The peak dpa rate in the first wall is 14.8 dpa per full power year. On the basis of this model, the tritium production as a function of position is given on Fig. VI.2-4.

Prior to this final calculation with the above model, several calculations were carried out to determine the shield thickness based on protection of the superconducting magnets. The magnet limits are set by several criteria: deterioration of insulation such as mylar and epoxy with dose, the

Fig. VI.2-1

## Central Cell

|                 | Radius<br>(cm) |
|-----------------|----------------|
|                 | 0              |
| Plasma          |                |
|                 | 32             |
| Vacuum          |                |
|                 | 46             |
| F.W.            |                |
|                 | 46.3           |
| Blanket         |                |
|                 | 146            |
| Vacuum          |                |
|                 | 151            |
| Vacuum Tank     |                |
|                 | 155            |
| Reflector       |                |
|                 | 183            |
| Shield-A        |                |
|                 | 196.95         |
| Shield-B        |                |
|                 | 204.5          |
| Shield-C        |                |
|                 | 208            |
| Void            |                |
|                 | 270            |
| Mylar           |                |
|                 | 280            |
| Magnet          |                |
|                 | 338            |
| Void            |                |
|                 | 1550           |
| Concrete Shield |                |
|                 | 1700           |

Table VI.2-1

| Region               | Thickness (cm) | Composition                                                                         |
|----------------------|----------------|-------------------------------------------------------------------------------------|
| First Wall           | 0.3            | 100 v/o HT-9                                                                        |
| Breeding Blanket     | 100            | 73 v/o $\text{Li}_{17}\text{Pb}_{83}$<br>7 v/o HT-9<br>20 v/o void                  |
| Blanket Test Modules | 100            | 73 v/o Li (nat.)<br>7 v/o HT-9<br>20 v/o void                                       |
| REGAT Module         | 20             | 20 v/o NaK<br>40 v/o 316 SS<br>15 v/o $\text{H}_2\text{O}$<br>25 v/o void           |
| Vacuum Tank          | 4              | 100 v/o 316 SS                                                                      |
| Reflector            | 28             | 90 v/o 316 SS<br>10 v/o $\text{H}_2\text{O}$                                        |
| Shield-A             | 13.95          | 95 v/o Fe 1422<br>5 v/o $\text{H}_2\text{O}$                                        |
| Shield-B             | 7.55           | 86 v/o $\text{B}_4\text{C}$<br>10 v/o Fe 1422<br>4 v/o $\text{H}_2\text{O}$         |
| Shield-C             | 3.5            | 100 v/o Pb                                                                          |
| Concrete Shield      | 200            | 87 v/o Ordinary Concrete<br>8 v/o Steel Reinforcement<br>5 v/o $\text{H}_2\text{O}$ |

Fig. VI.2-2

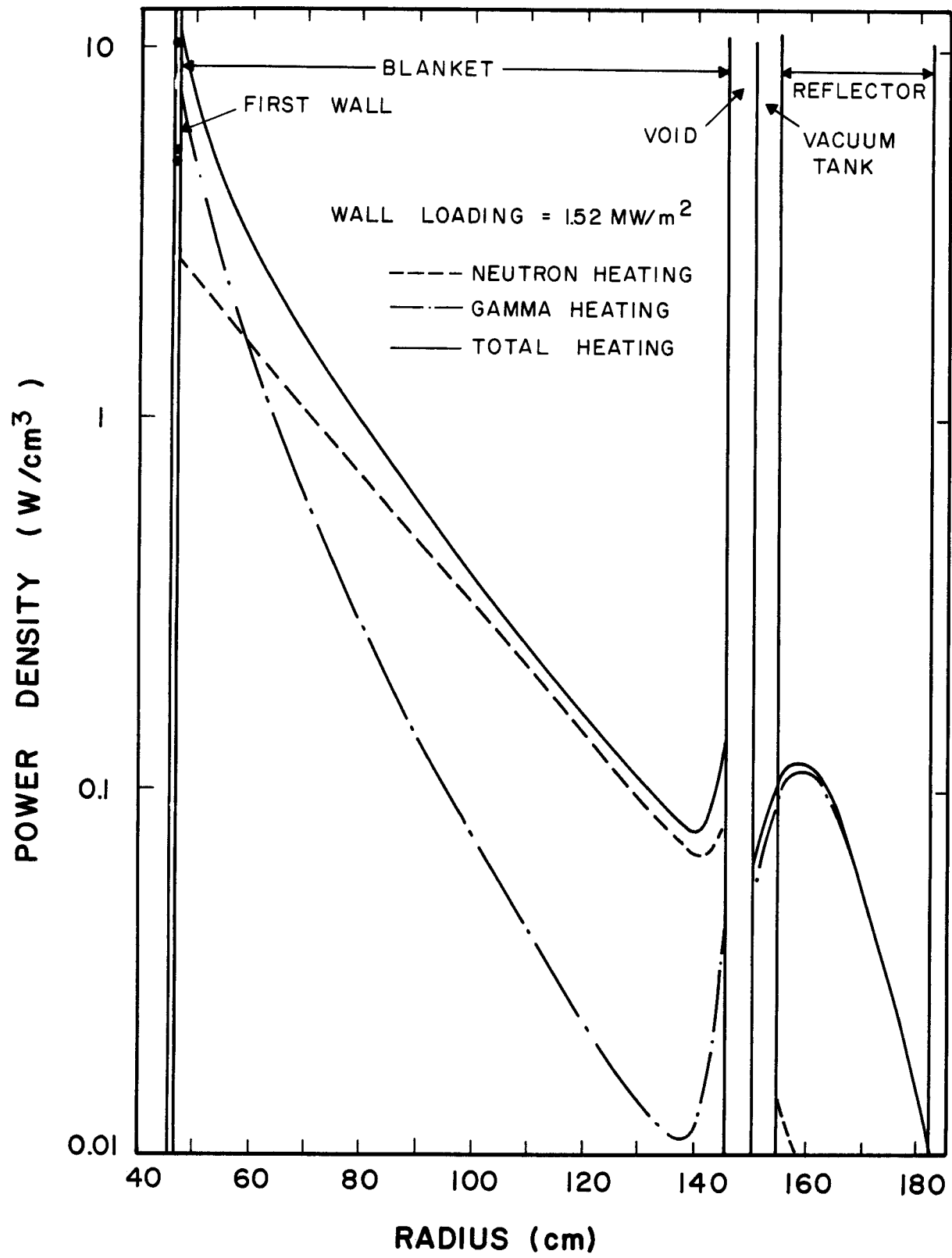




Fig. VI.2-3

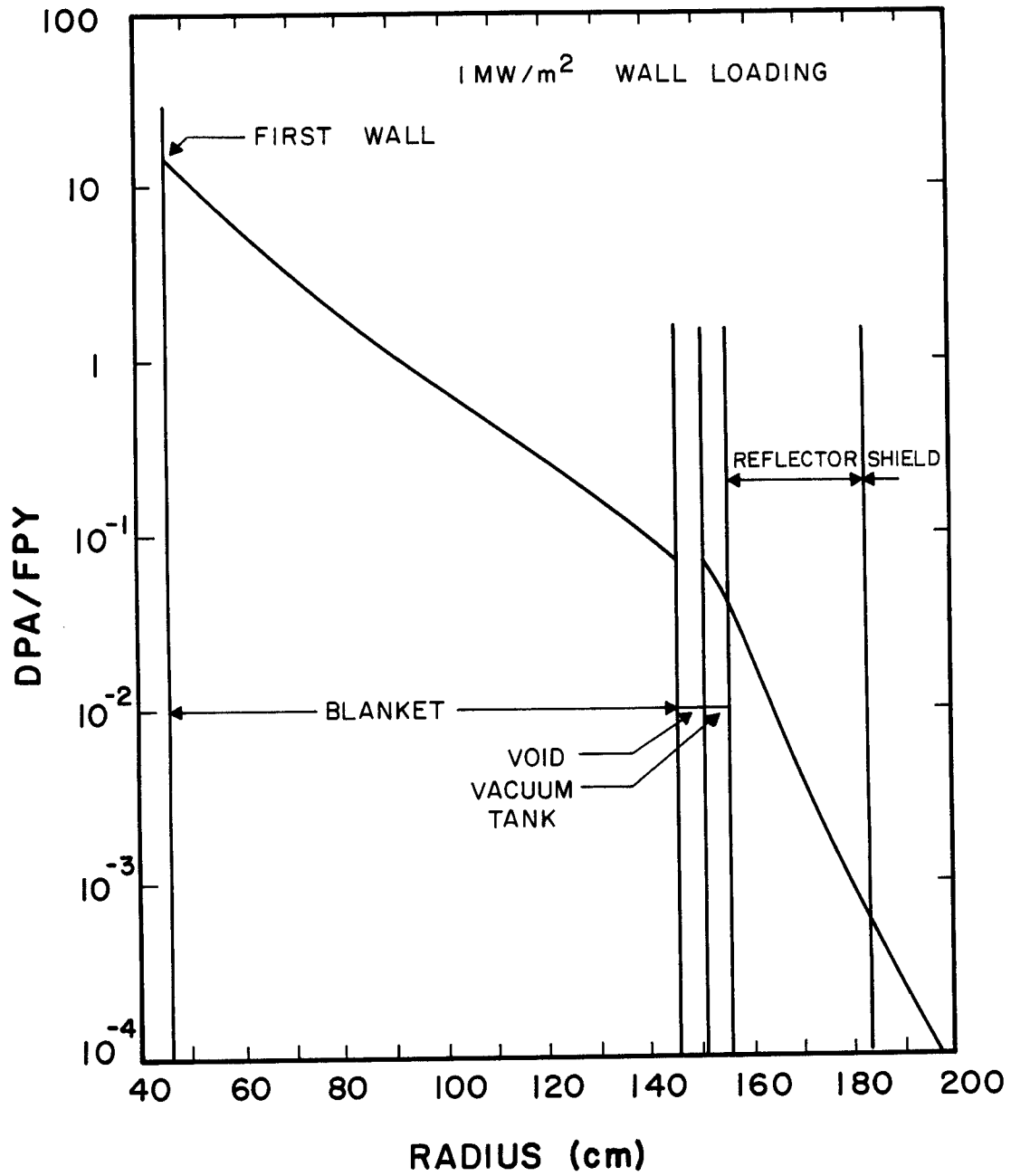
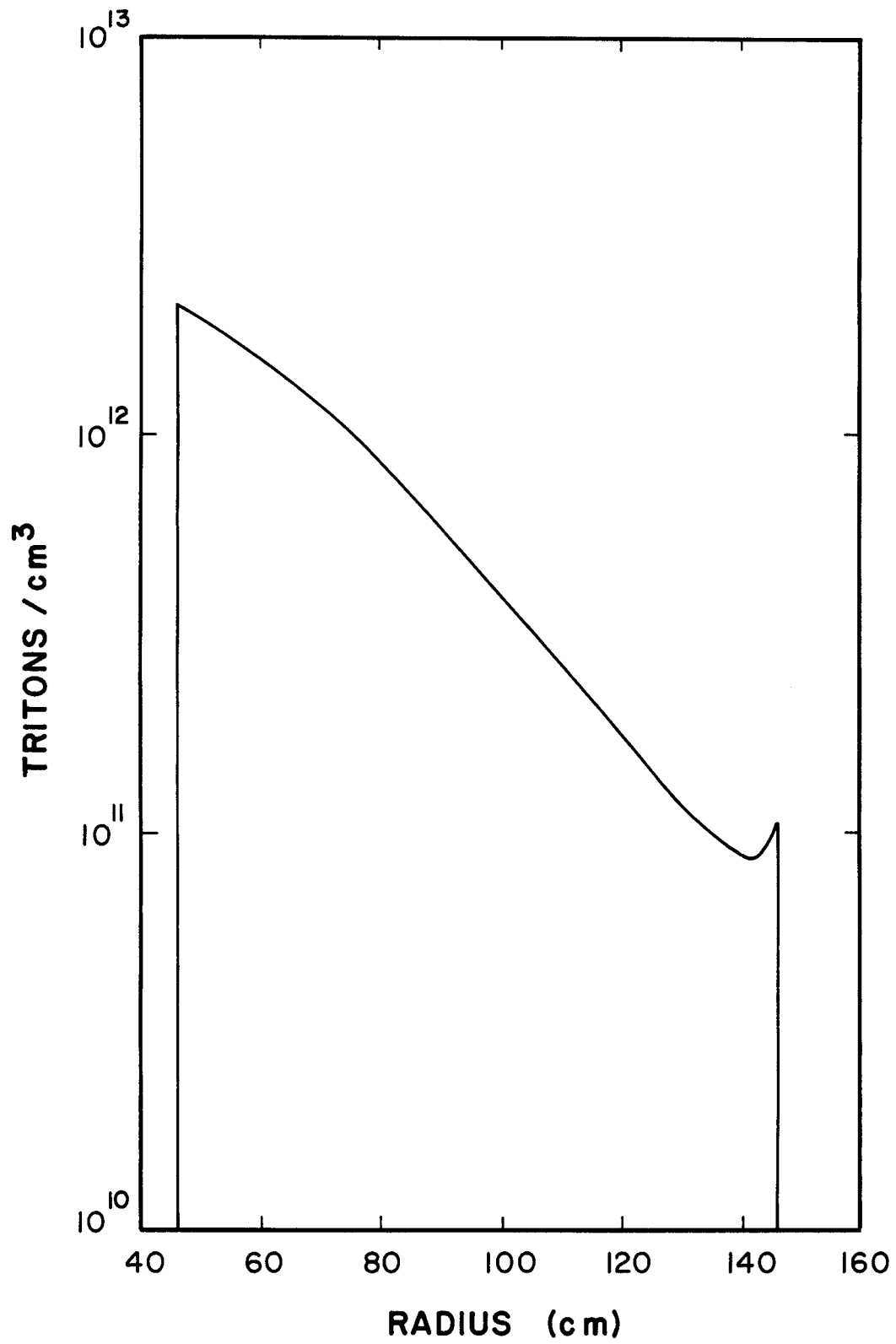


Fig. VI.2-4



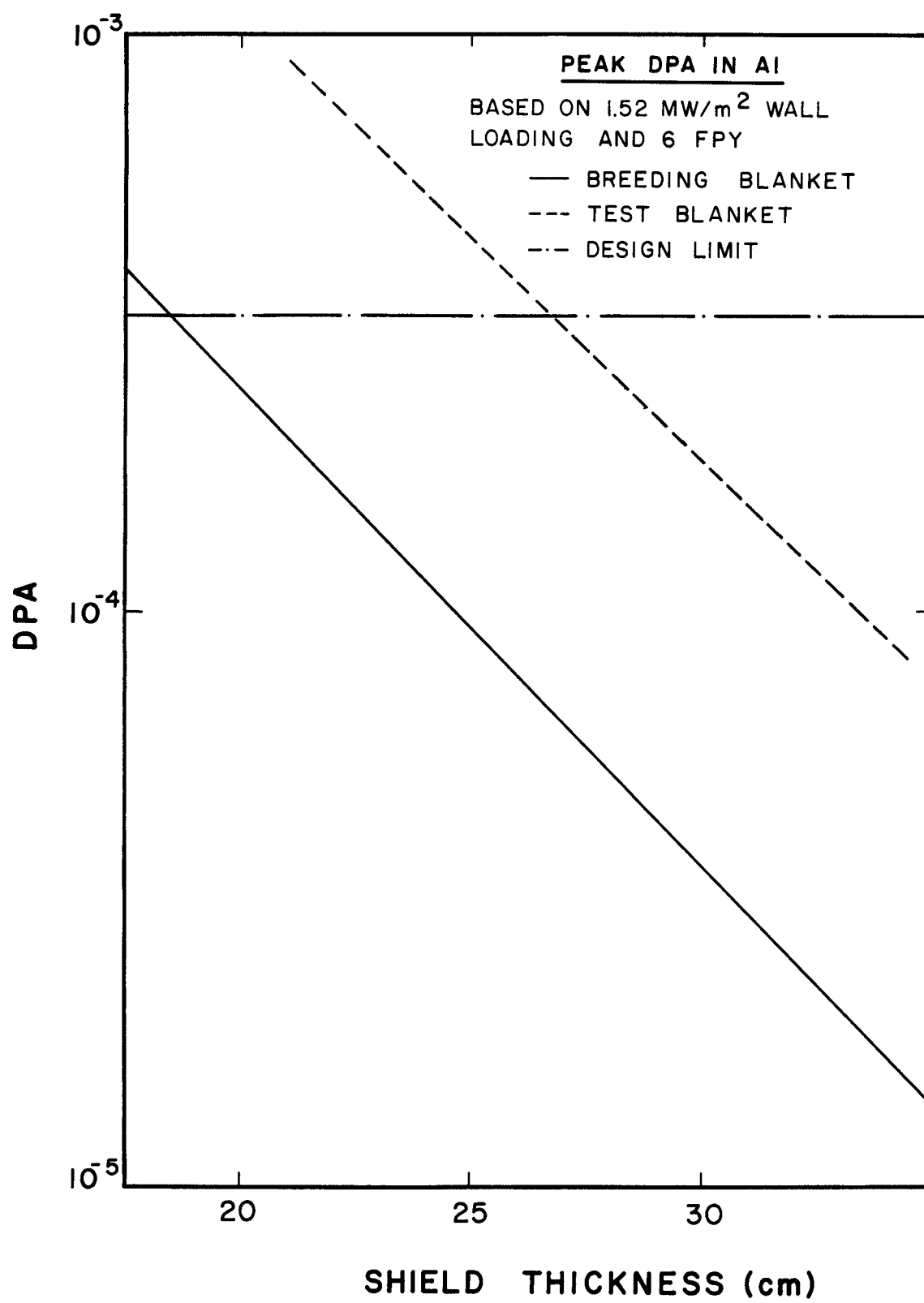
cryogenic heat load, and resistivity increases in the stabilizer for the superconductor. Dose limits for mylar and epoxy are taken as  $10^{10}$  and  $5 \times 10^9$  Rad, respectively. The coil design is based on the resistivity of the aluminum stabilizer not exceeding  $10^{-7} \Omega \text{ cm}$ . Since the resistivity of high purity aluminum is  $10^{-8} \Omega \text{ cm}$ , the induced resistivity ( $\rho_r$ ) due to irradiation should not exceed  $9 \times 10^{-8} \Omega \text{ cm}$ . For aluminum<sup>(5)</sup> the induced resistivity is given by

$$\rho_r = 8 \times 10^{-7} [1 - \exp (-366 d)] \Omega \text{ cm}$$

where  $d$  is the displacements per atom. With this set equal to  $9 \times 10^{-8} \Omega \text{ cm}$ , the value of  $d$  is  $3.26 \times 10^{-4} \text{ dpa}$ . Once this value of  $d$  is reached, the magnet must be annealed to ensure proper performance. Calculation for several shield thicknesses yields the dpa results shown on Fig. VI.2-5. On the basis of this, the shield thickness was set at 0.25 m which yields  $9.4 \times 10^{-5} \text{ dpa}$  in 6 FPY for the permanent blanket. Since TASKA is only expected to operate for 5.3 FPY this is more than adequate. The second curve representing the shield behind a test blanket corresponds to  $4.5 \times 10^{-4} \text{ dpa}$  on the same basis. This would imply annealing before starting the last year of operation. However, the latest designs do not call for magnets behind the test modules. The dose values in the mylar and epoxy are  $1.85 \times 10^7 \text{ Rad/FPY}$  and  $1.61 \times 10^7 \text{ Rad/FPY}$ , well below the design limits. Other data of interest are the peak helium and hydrogen production rates in the iron of the first wall. These are 107.1 and 347.6 appm/FPY, respectively. Finally based on this model, the peak biological dose outside the concrete shield is 1.12 mR/hr.

The tritium production and total heating in the central cell are based on three-dimensional continuous energy calculations. These were performed using

Fig. VI.2-5



the Monte Carlo code MCNP<sup>(6)</sup>, and the Recommended Monte Carlo Cross Section (RMCCS) library<sup>(7)</sup> on the MFE computer at LLNL. The coupled neutron and gamma ray model was used and results are based on 4,000 histories giving relative errors of less than 5%. As the plasma density varies in the central cell, barrier, and plug regions, so does the neutron source density. The radius of the plasma also varies. Assuming a flat profile, the source is uniform to the plasma boundary. Weighting the source density by the area of the cross section leads to the linear source density shown on Fig. VI.2-6. This can be approximated as constant in the central cell, but is fortunately much weaker in the barrier and plug zones. The Monte Carlo modeling of the central cell and nearby barrier regions has been carried out. A view emphasizing the central cell is shown on Fig. VI.2-7. This model and the related calculations give a tritium breeding ratio of 1.04. The region-wise details of the tritium production are shown in Table VI.2-2. With the same region-wise description the energy deposition in MeV/fusion neutron is given on Table VI.2-3. This results in the rather low blanket energy multiplication of 1.06. However, as pointed out earlier, this was not a design issue.

In summary, the central cell blanket and shield meet all the goals stated at the beginning of this section. The breeding margin is not great, but is probably adequate and relatively modest design changes could increase the breeding if necessary. Other performance goals have substantial margins for error.

### VI.2.3 Barrier Region

The barrier region contains the narrow throat area at the end of the central cell as shown in Fig. VI.2-8. The constriction is caused by the very high (20 T) magnetic field. This field is produced by a system of coils

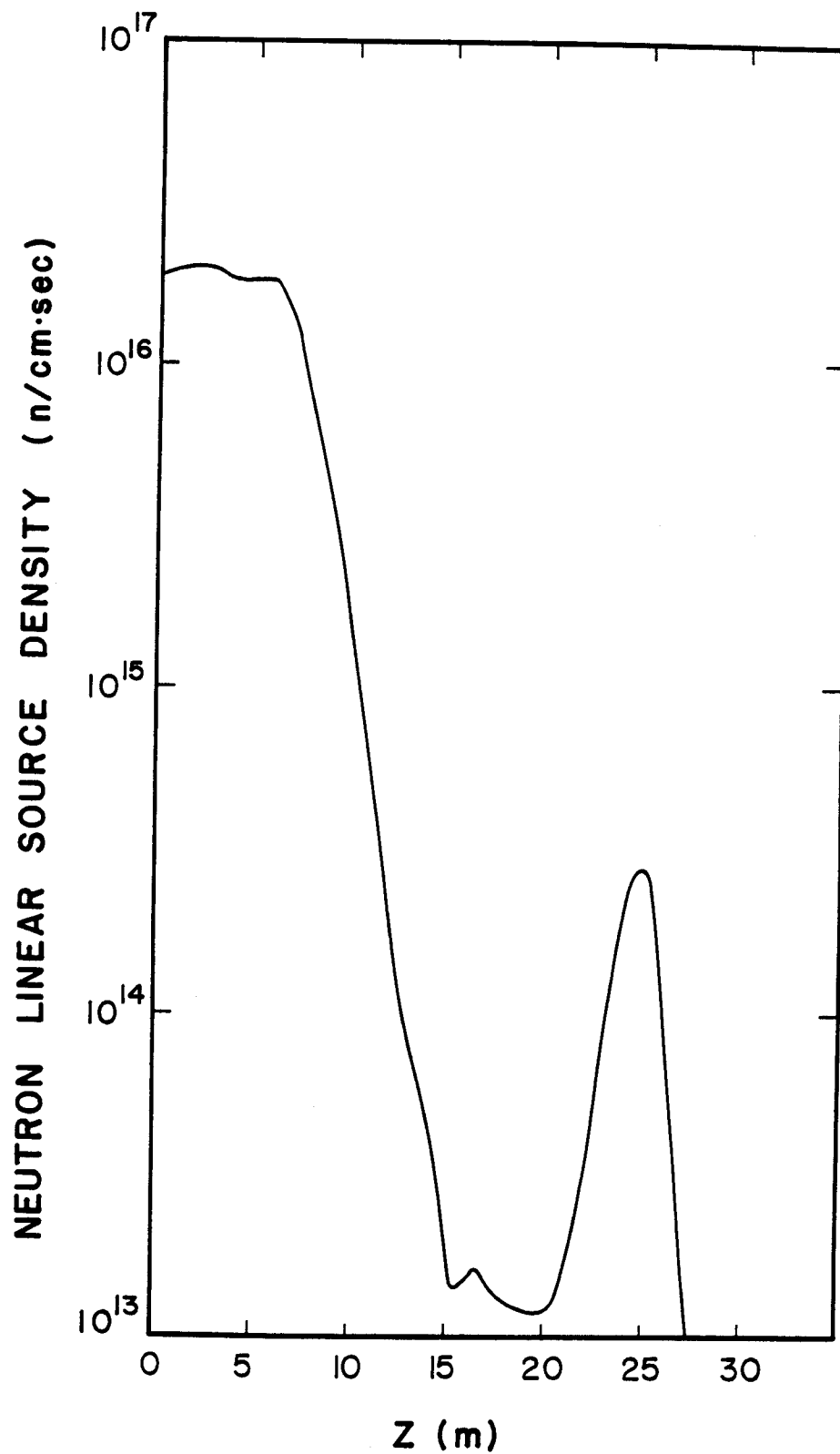


Figure VI.2-6 Variation of the neutron linear source density along the z axis.

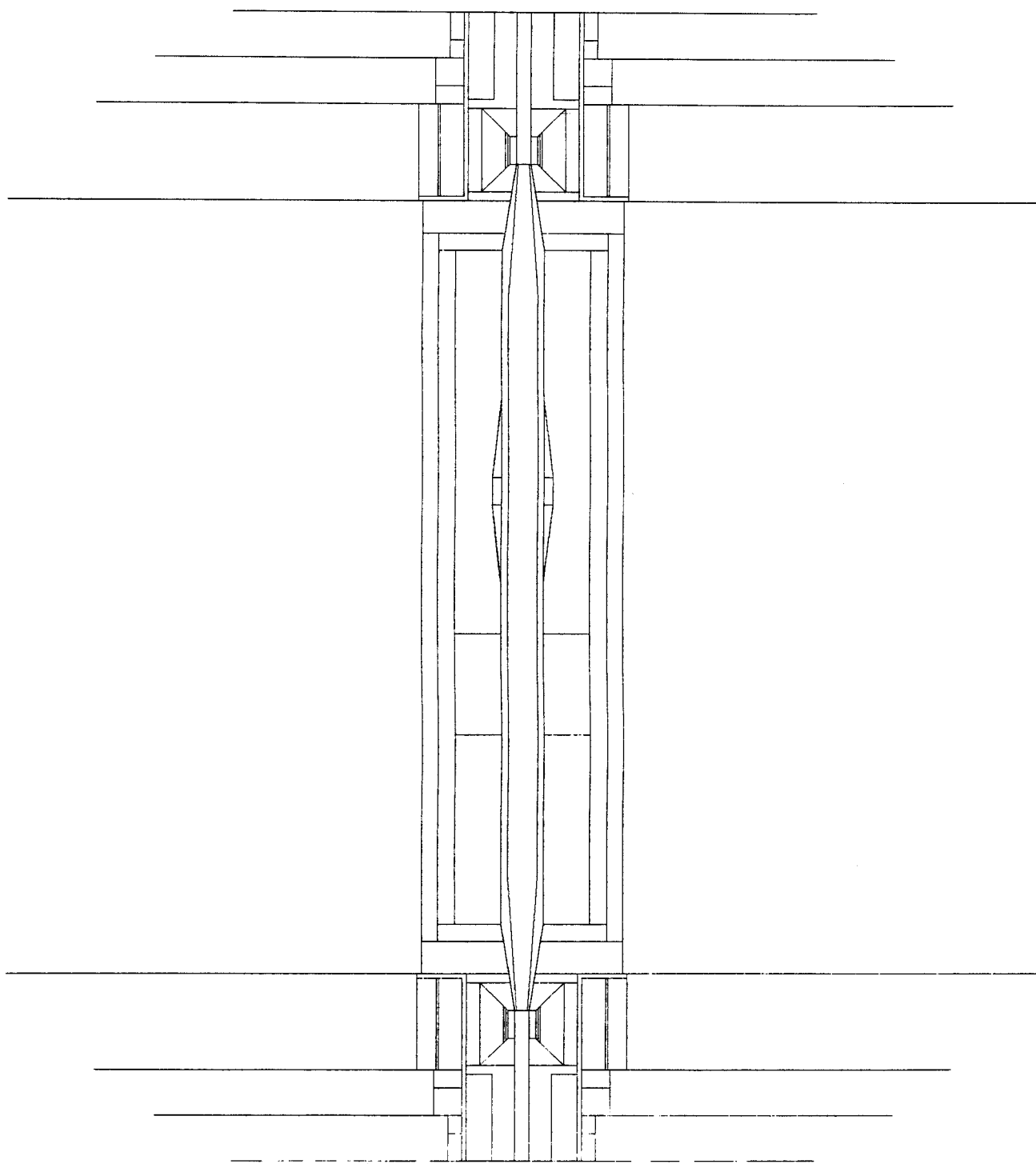


Fig. VI.2-7 Monte Carlo model of central cell and barrier region.

Table VI.2-2. Tritium Production

| Region               |       |              | T per Fusion    |                 |         |
|----------------------|-------|--------------|-----------------|-----------------|---------|
|                      |       |              | <sup>6</sup> Li | <sup>7</sup> Li | Total   |
| Breeding Blanket     |       | Width<br>(m) |                 |                 |         |
|                      | 1     | 1.565        | 0.15463         | 0.00027         | 0.15490 |
|                      | 2     | 1.80         | 0.16316         | 0.00027         | 0.16343 |
|                      | 3     | 1.80         | 0.16240         | 0.00027         | 0.16267 |
|                      | 4     | 1.05         | 0.10933         | 0.00021         | 0.10954 |
|                      | 7     | 1.05         | 0.09919         | 0.00018         | 0.09937 |
|                      | 8     | 1.565        | 0.13854         | 0.00023         | 0.13877 |
|                      | Total | 8.83         | 0.82725         | 0.00143         | 0.82868 |
| Blanket Test Modules | 5     | 1.05         | 0.07144         | 0.03144         | 0.10288 |
|                      | 6     | 1.05         | 0.07754         | 0.03186         | 0.10940 |
|                      | Total | 2.10         | 0.14898         | 0.06330         | 0.21228 |
| System Total         |       | 10.93        | 0.97623         | 0.06473         | 1.04096 |

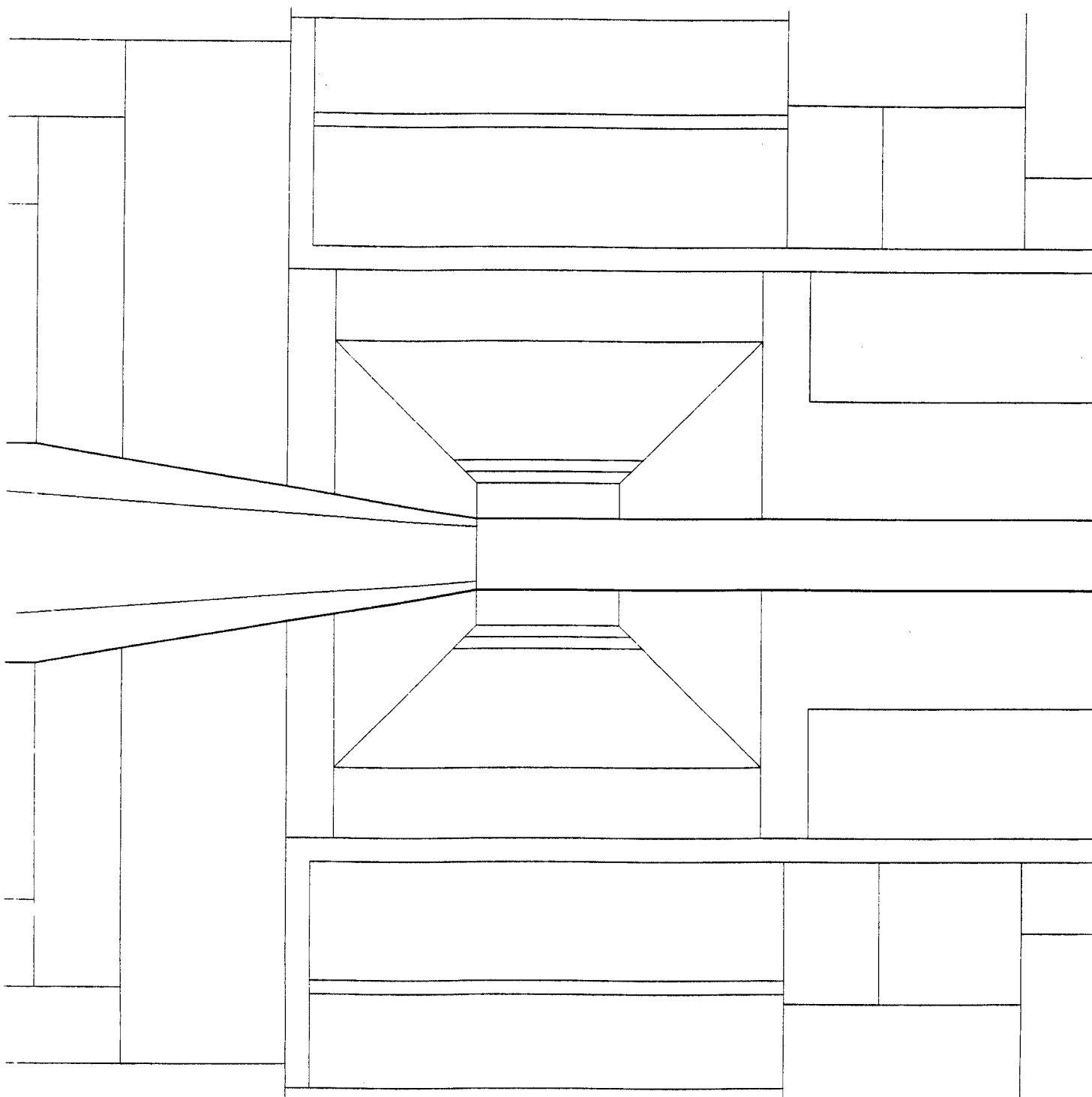


Table VI.2-3. Nuclear Heating in Central Cell (MeV/fusion)

| Region               | Neutron Heating | Gamma Heating | Total<br>Nuclear Testing |
|----------------------|-----------------|---------------|--------------------------|
| First Wall           | 0.0955          | 0.1197        | 0.2152                   |
| Breeding Blanket     | 4.9139          | 3.7665        | 8.6804                   |
| Blanket Test Modules | 1.6579          | 0.5287        | 2.1866                   |
| REGAT Modules        | 0.1854          | 0.3489        | 0.5343                   |
| Reflector            | 0.1344          | 0.9054        | 1.0398                   |
| Shield               | 1.2649          | 1.0985        | 2.3634                   |
| Total                | 8.2520          | 6.7677        | 15.0197                  |

Energy multiplication = 1.06

Fig. VI.2-8 Barrier and adjacent regions.



including Cu insert coils with only a thin shield in front. For this reason, a primarily tungsten shield is used to obtain maximum attenuation. A one-dimensional model described by the schematic shown on Fig. VI.2-9 is used as a basis for determining shield performance. The calculations are based on a linear source density of  $2.677 \times 10^{15}$  neutrons/cm·sec which with a wall radius of 17 cm gives a neutron wall loading of  $0.566 \text{ MW/m}^2$ . The resulting dose rate to the MgO insulation used in the normal coil is  $4.35 \times 10^{11}$  Rad/FPY. With a design limit of  $10^{12}$  Rad this implies a replacement of this coil every 2.3 FPY of operation. For TASKA's expected life and capacity factor only two replacements are required during its total operation.

There are other coils in the barrier region referred to as field shaping coils. These are also normal coils. Further one-dimensional calculations were used in choosing the required shielding. At 16.5 m from the midplane, the wall radius is 80 cm and the linear neutron source density is  $1.62 \times 10^{13}$ /cm sec leading to a  $7.28 \times 10^{-4} \text{ MW/m}^2$  maximum wall loading. With 20 and 30 cm shields, the peak dose rate in MgO is  $3.87 \times 10^8$  Rad/FPY which with a  $10^{12}$  Rad limit is more than adequate. During the operation the peak biological dose rate outside a surrounding concrete shield is determined to be 1.23 mR/hr easily meeting the upper bound of 10 mR/hr deemed acceptable.

A series of one-dimensional calculations were carried out to assess the shield needed to protect the yin-yang and the recircularizing coils which are both superconducting systems. The results are based on the linear source density of  $2.8 \times 10^{14}$ /cm sec corresponding to the peak at 25 m and using a wall radius of 18 cm leading to a  $0.056 \text{ MW/m}^2$  maximum neutron wall loading in this zone. Calculations for several shield thicknesses enable us to plot the peak dpa in the copper stabilizer versus shield thickness (Fig. VI.2-10), and

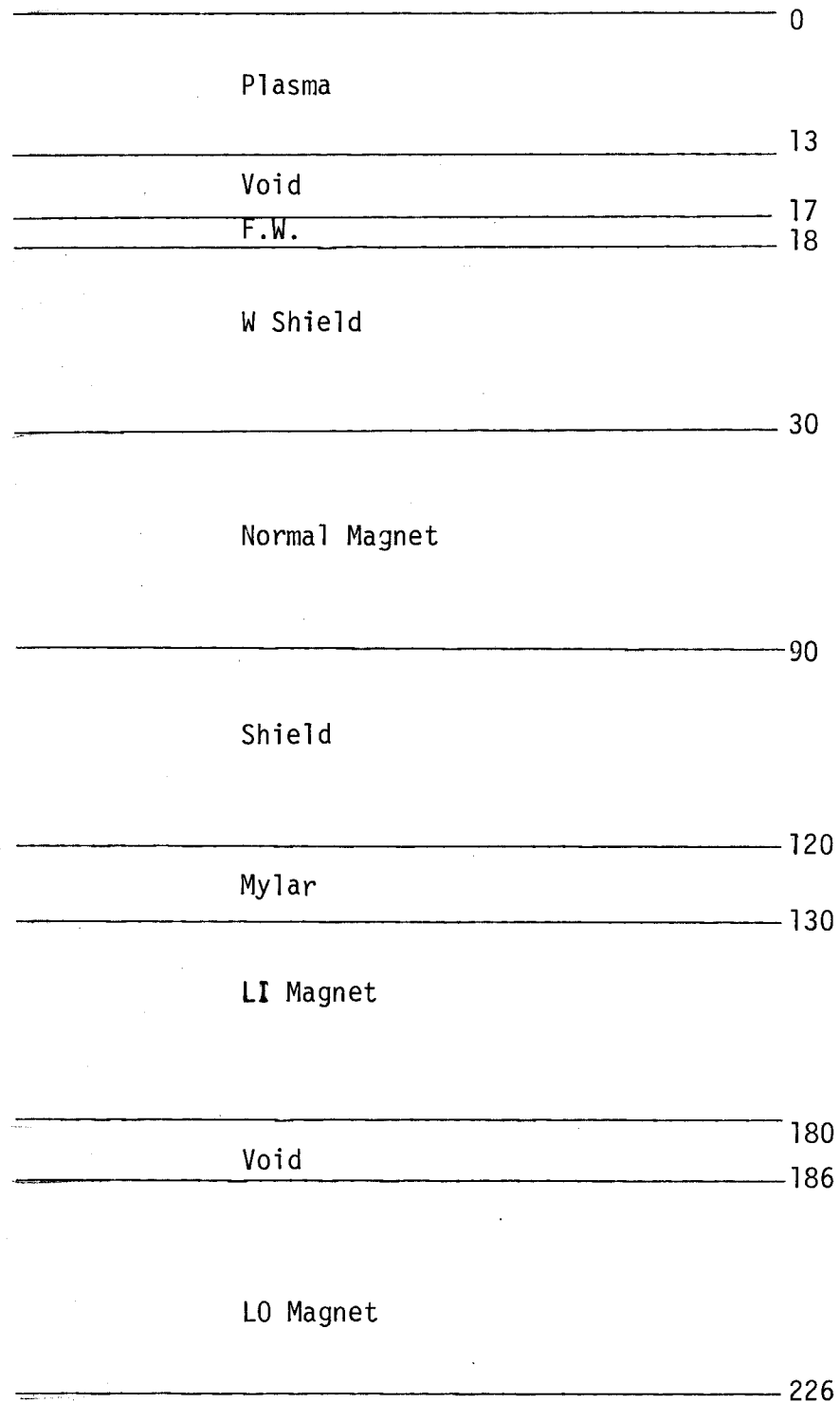
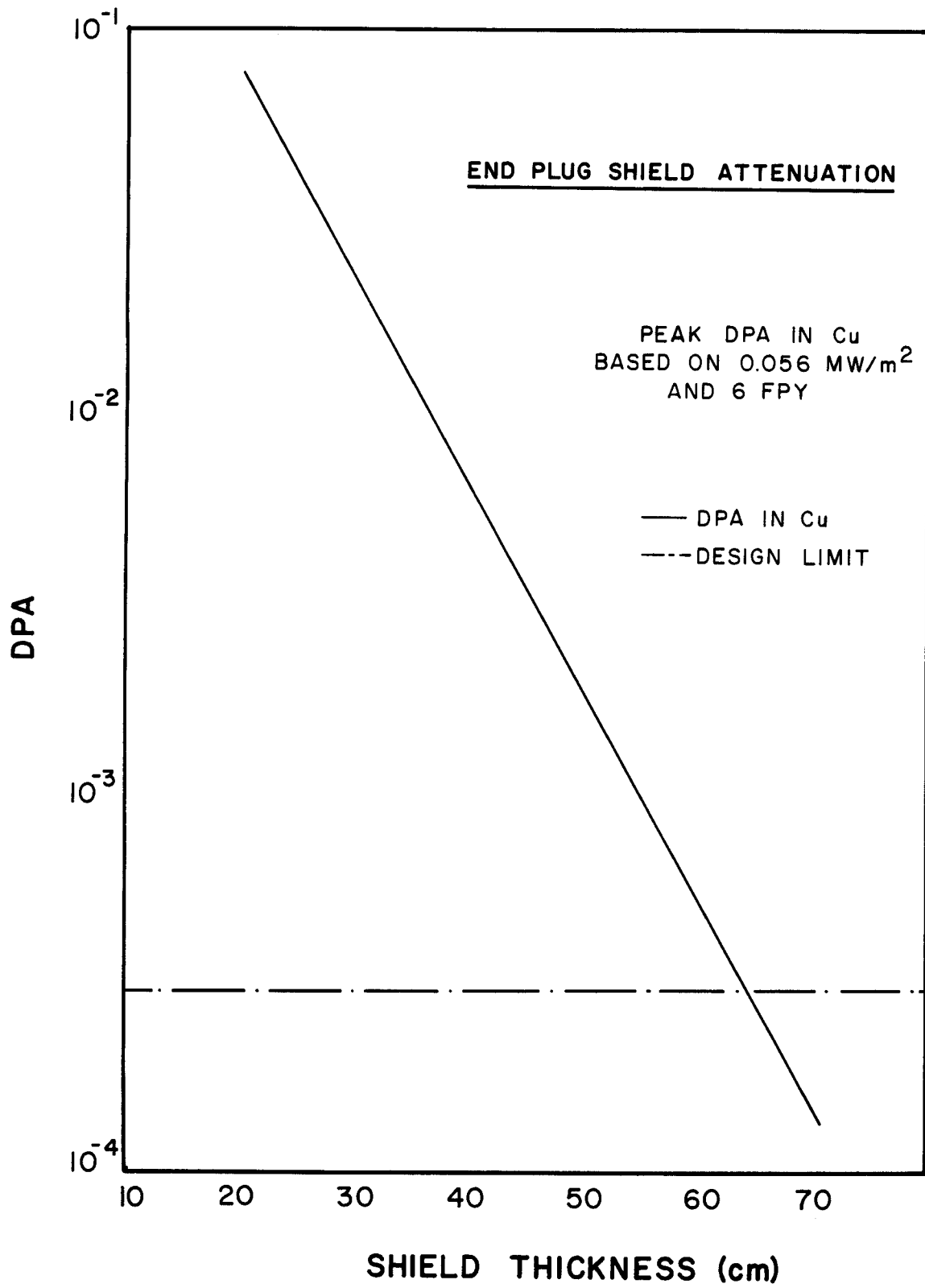


Fig. VI.2-9. Barrier midplane section.

Fig. VI.2-10



the peak dose in mylar and epoxy based on 6 FPY, Figs. VI.2-11 and VI.2-12. The copper resistivity should not exceed  $10^{-7} \Omega \text{ cm}$  to meet stabilization requirements, but in a magnetic field of 8 T its resistivity is  $5.3 \times 10^{-8} \Omega \text{ cm}$  without induced resistivity ( $\rho_r$ ). Thus, the induced resistivity should not exceed  $4.7 \times 10^{-8} \Omega \text{ cm}$ . For copper<sup>(5)</sup> the induced resistivity is

$$\rho_r = 3 \times 10^{-7} [1 - \exp(-563 d)] \Omega \text{ cm}$$

where d is the dpa, limiting d without anneal to  $3 \times 10^{-4}$  dpa.

The choice of a 60 cm shield then gives  $8.47 \times 10^{-5}$  dpa/FPY and a single anneal during TASKA's lifetime. The peak doses in mylar and epoxy under these conditions is  $2 \times 10^8$  and  $1.9 \times 10^8$  Rad/FPY respectively. With limits of  $10^{10}$  and  $5 \times 10^9$  Rad in the mylar and epoxy, this shield is adequate for our needs. The peak biological dose rate outside the surrounding concrete shield is found to be 0.62 mR/hr, again well below the 10 mR/hr limit.

#### VI.2.4 Radioactivity

The radioactivity for the TASKA study was calculated using the compositions and zones shown in Fig. VI.2-13. Cylindrical geometry was chosen and the flux was taken from the midplane of the reactor. The flux was calculated using the ANISN<sup>(8)</sup> code with a  $P_3S_8$  approximation. The transport cross sections were taken from a coupled 25 neutron - 21 gamma group data library.<sup>(9)</sup>

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Since these calculations were performed the thickness of the first wall of the blanket has been reduced to an equivalent of 0.3 cm. This will have the effect of reducing the activity in the first wall by about a factor of three and increasing the activity of the blanket near the first wall slightly. However, since the principal contribution to the dose after shutdown outside the shield comes from the regions near the edge of the shield, the change in the first wall thickness is expected to result in a negligible change in this quantity.

Fig. VI.2-11

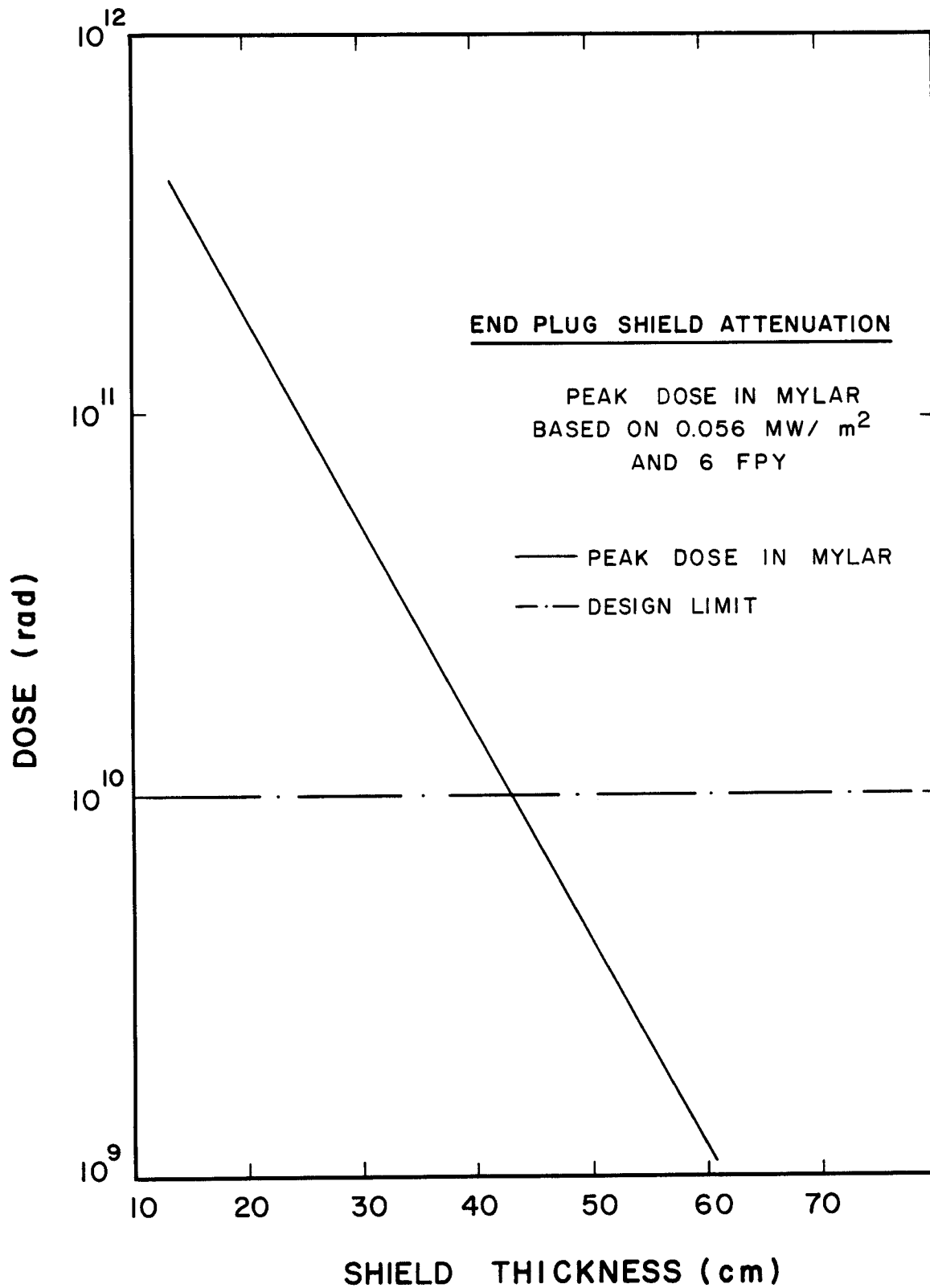


Fig. VI.2-12

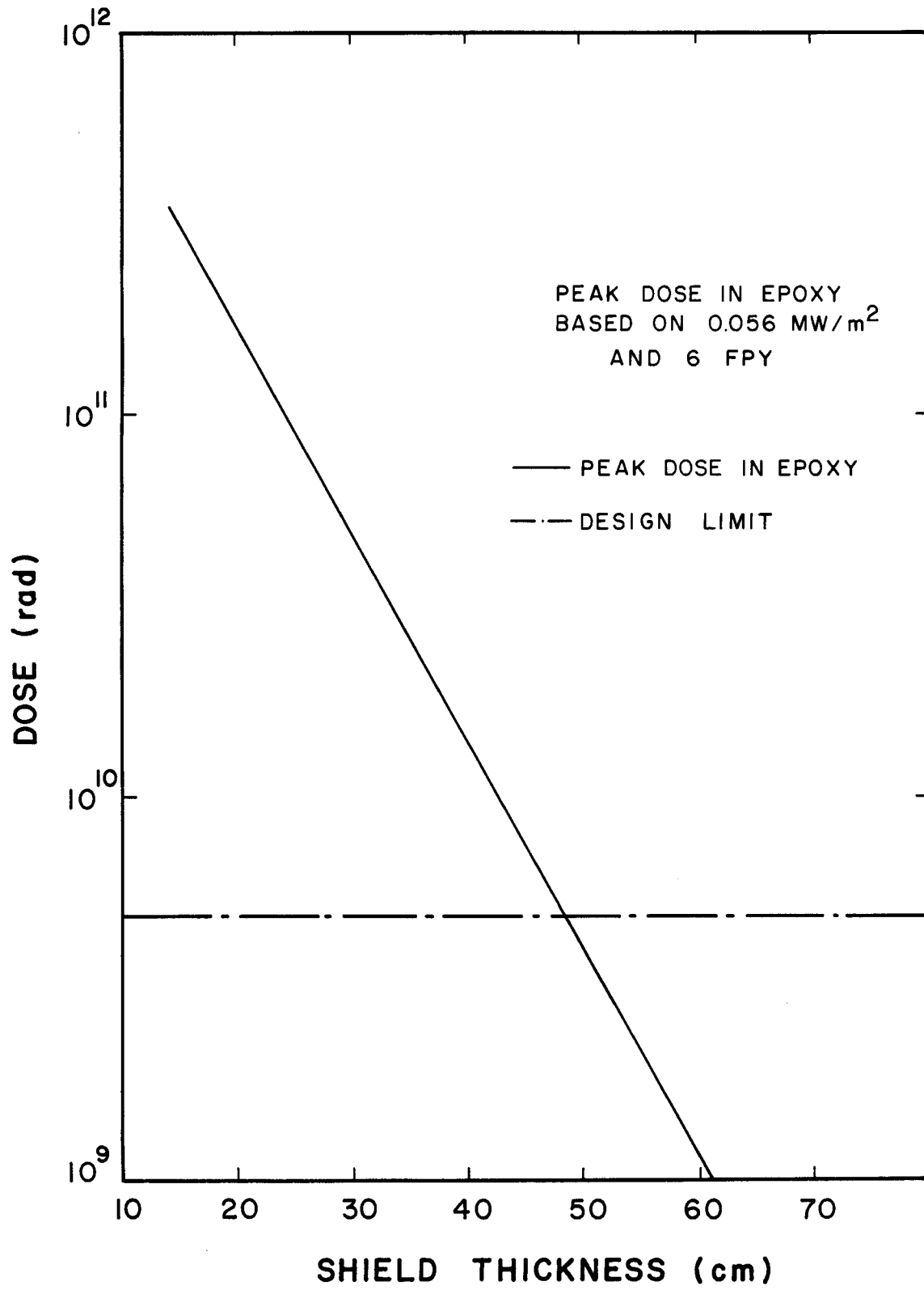
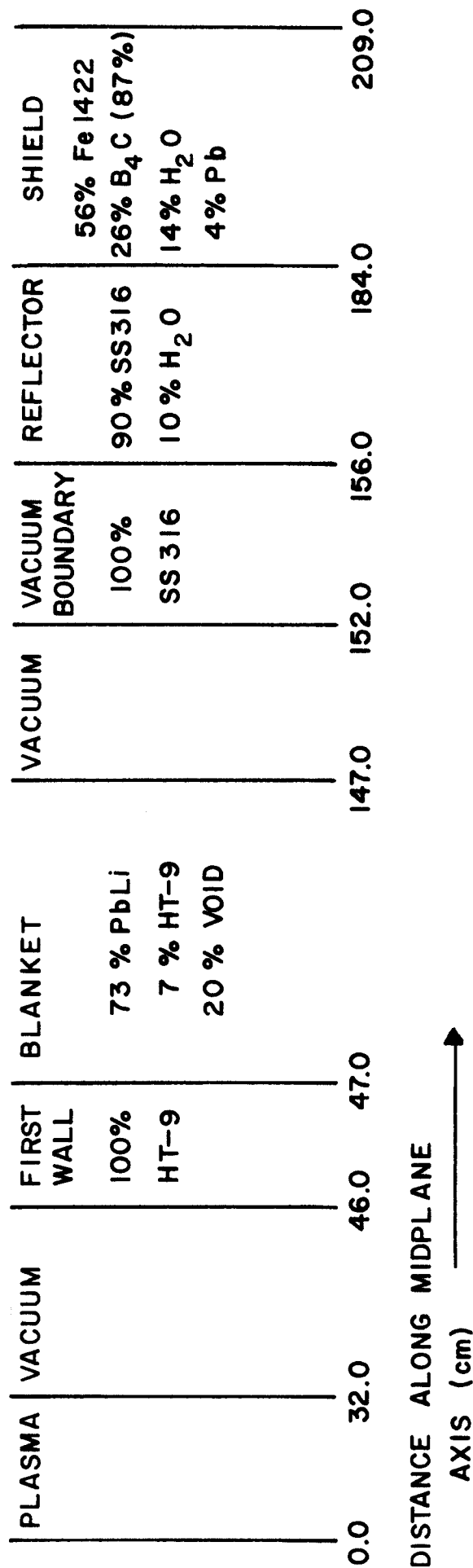




Fig. VI.2-13

DIAGRAM OF MODEL USED IN CALCULATING  
ACTIVATION PARAMETERS FOR TASKA



The flux was then used as the input to the DKR code.<sup>(10)</sup> The results of this calculation are shown in Figs. VI.2-14- VI.2-19. The calculations were done for an operating time of 2 years and the diagrams indicate the respective magnitudes of the quantities of interest for the combined reflector and shield and the combined blanket and first wall. The levels of the specific activity, biological hazard potential, and percent afterheat are all of the same order of magnitude as those of previous studies (see WITAMIR-I)<sup>(1)</sup>, i.e., the activity at shutdown is about 1 curie per watt, the biological hazard potential for air is about 100 km<sup>3</sup> per kilowatt and the afterheat is about .7% of the total operating power. The highest volumetric activity is found in the first wall but the inner part of the blanket is highly activated as well. In the reflector and shield, the induced activities are respectively  $3.41 \times 10^{-2}$  curies/cm<sup>3</sup> and  $6.69 \times 10^{-6}$  curies/cm<sup>3</sup> one day after shutdown.

The time dependence of the activity is basically the same as in most other steel reactors. After shutdown, Fe-55, Mn-56, Cr-51, and Pb-203 are the isotopes which contribute the most to the radioactivity. The relative contribution of Fe-55 is higher than in previous studies because of the relative thickness of the first wall (1 cm vs. 0.3 cm in earlier studies) and the choice of material (HT-9). The Mn-56 and Pb-203 both decay rapidly, diminishing the activity by a factor of two after one day. After this, the decay is influenced by the Fe-55 until about 50 years when the Ni-63 found in the 316 SS takes over. At this time the graphs show that the reflector and shield are the dominant sources of radioactivity. This is due to the compositions of the zones (see Fig. VI.2-13), i.e., the reflector has a very high concentration of nickel from the 316 SS. The long term decay is governed by Mo-93 and Nb-93m.

Fig. VI.2-14

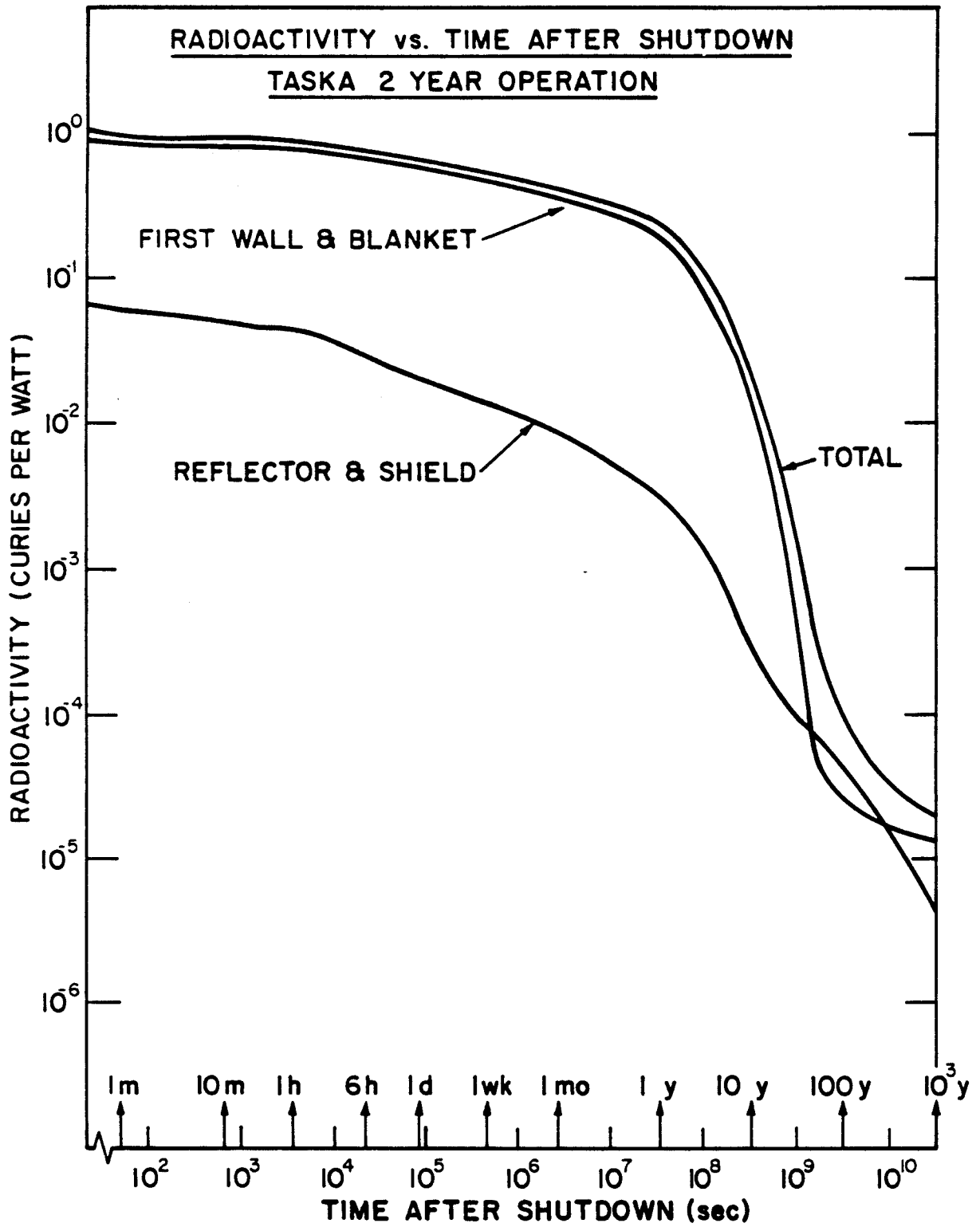


Fig. VI.2-15

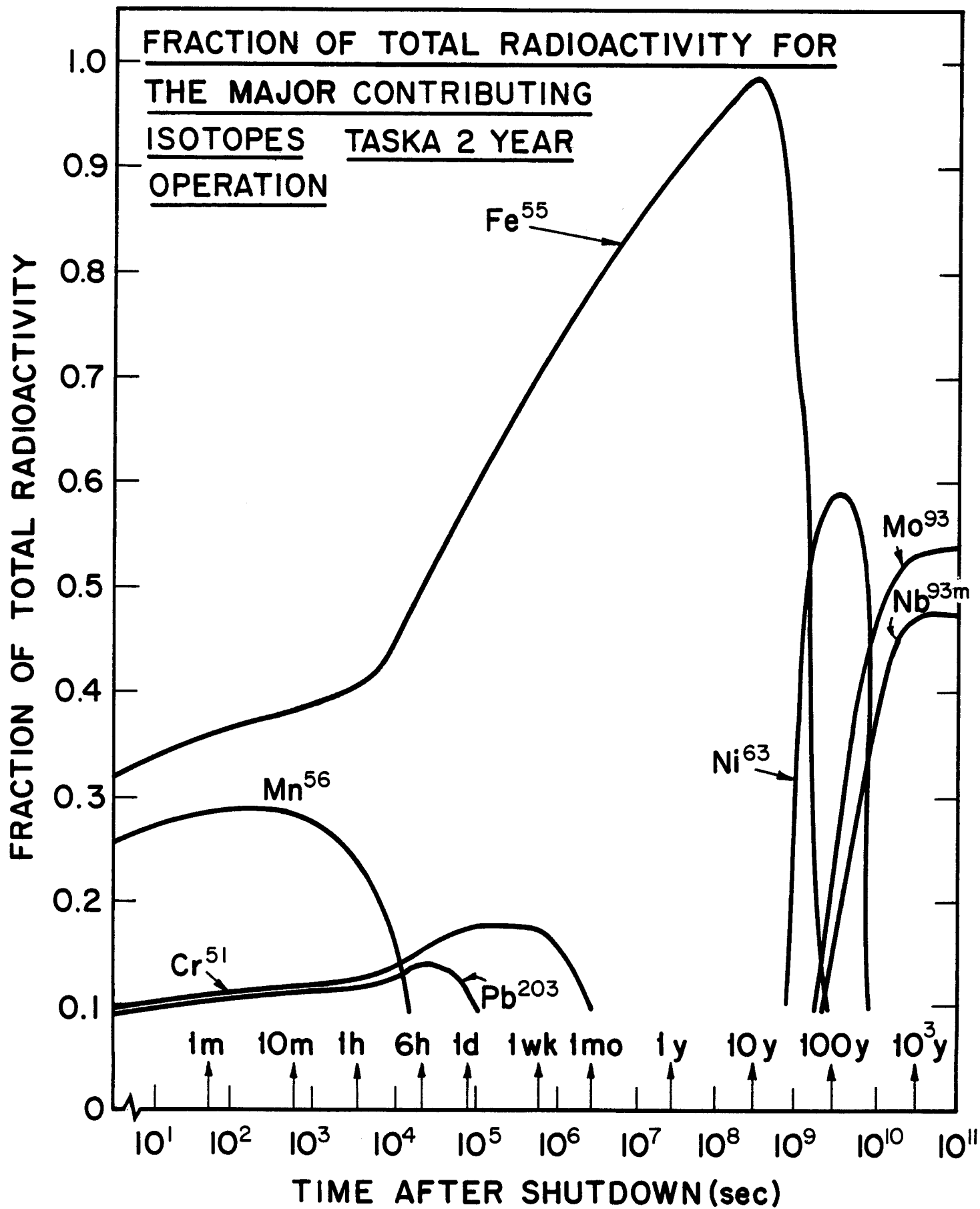


Fig. VI.2-16

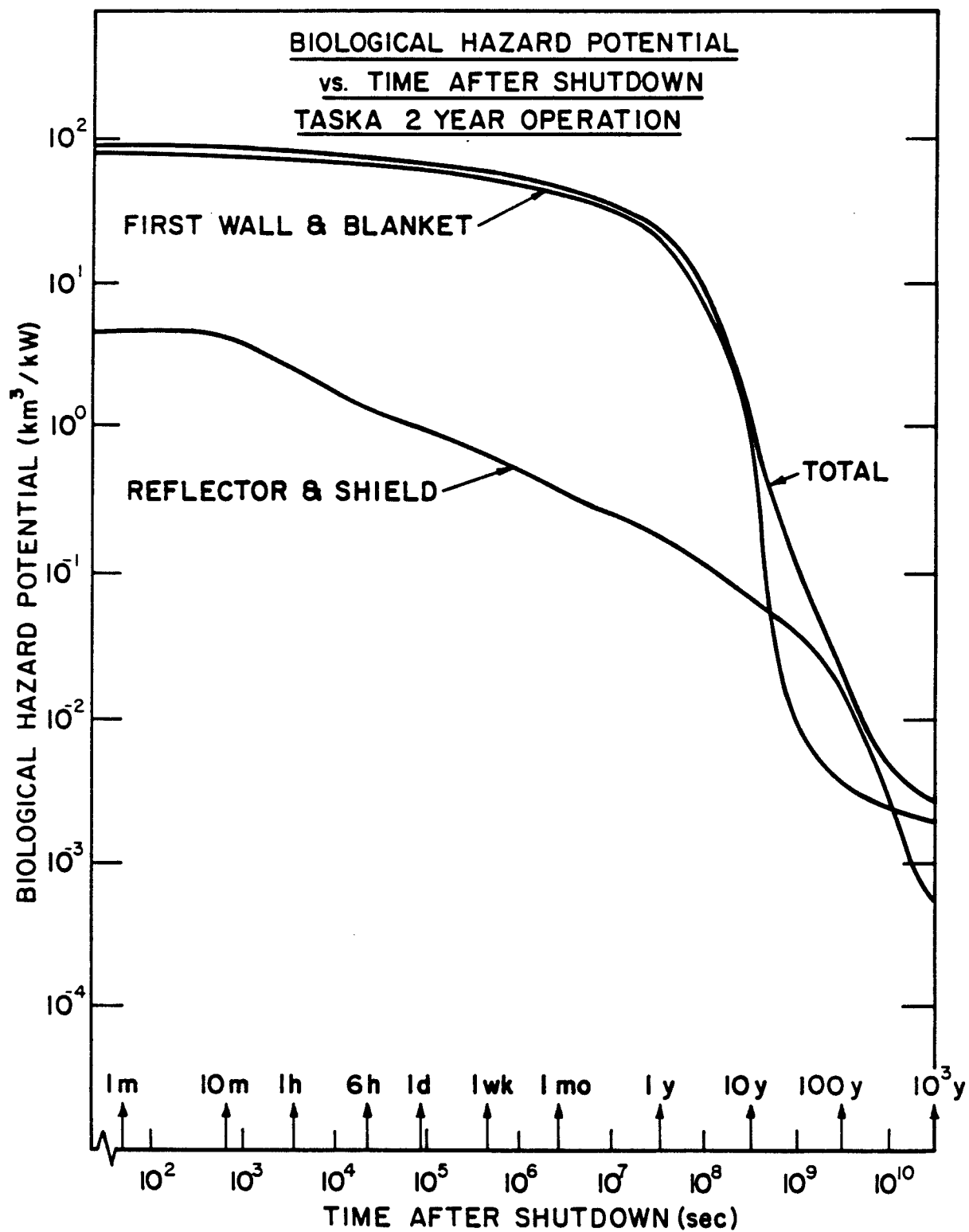


Fig. VI.2-17

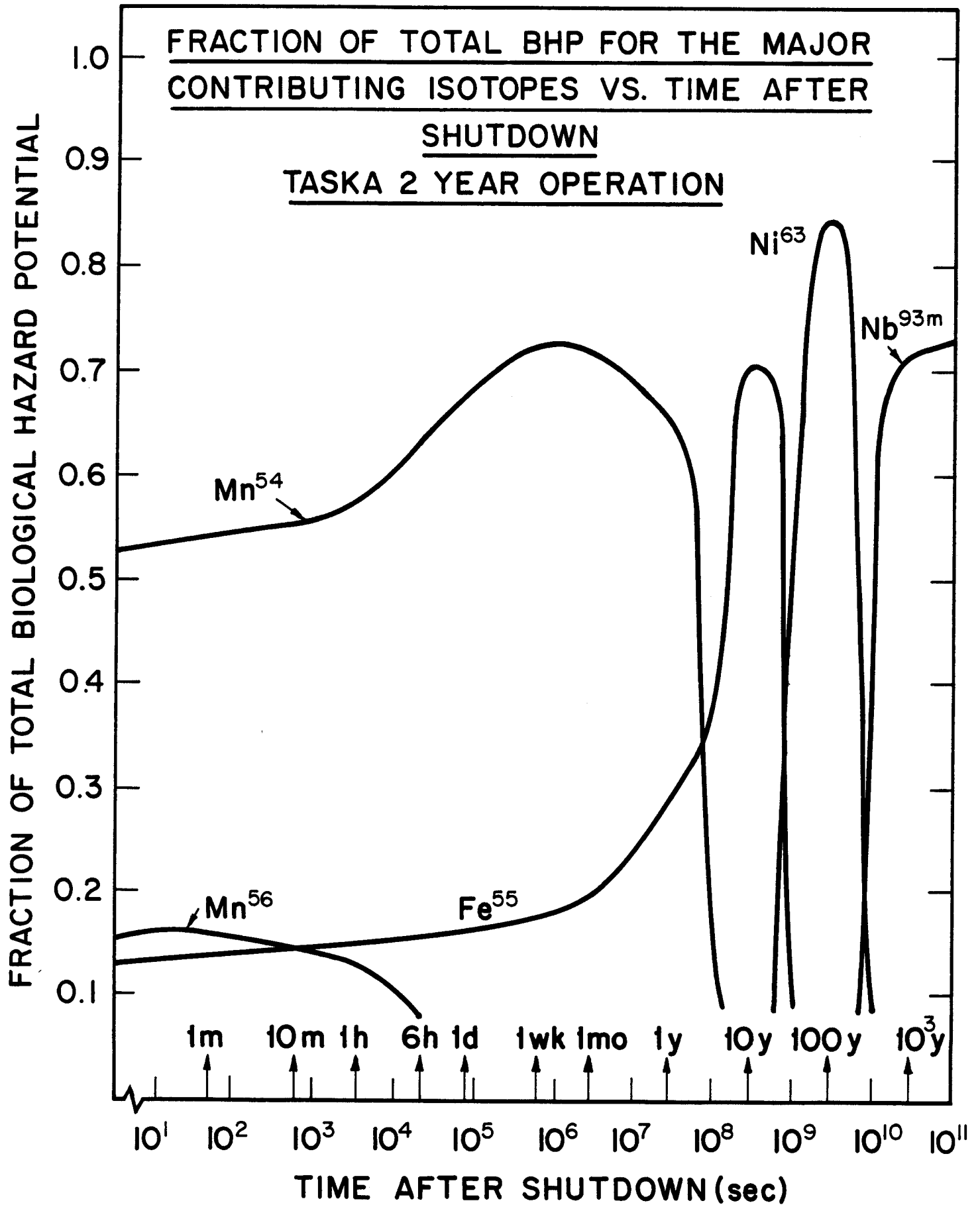


Fig. VI.2-18

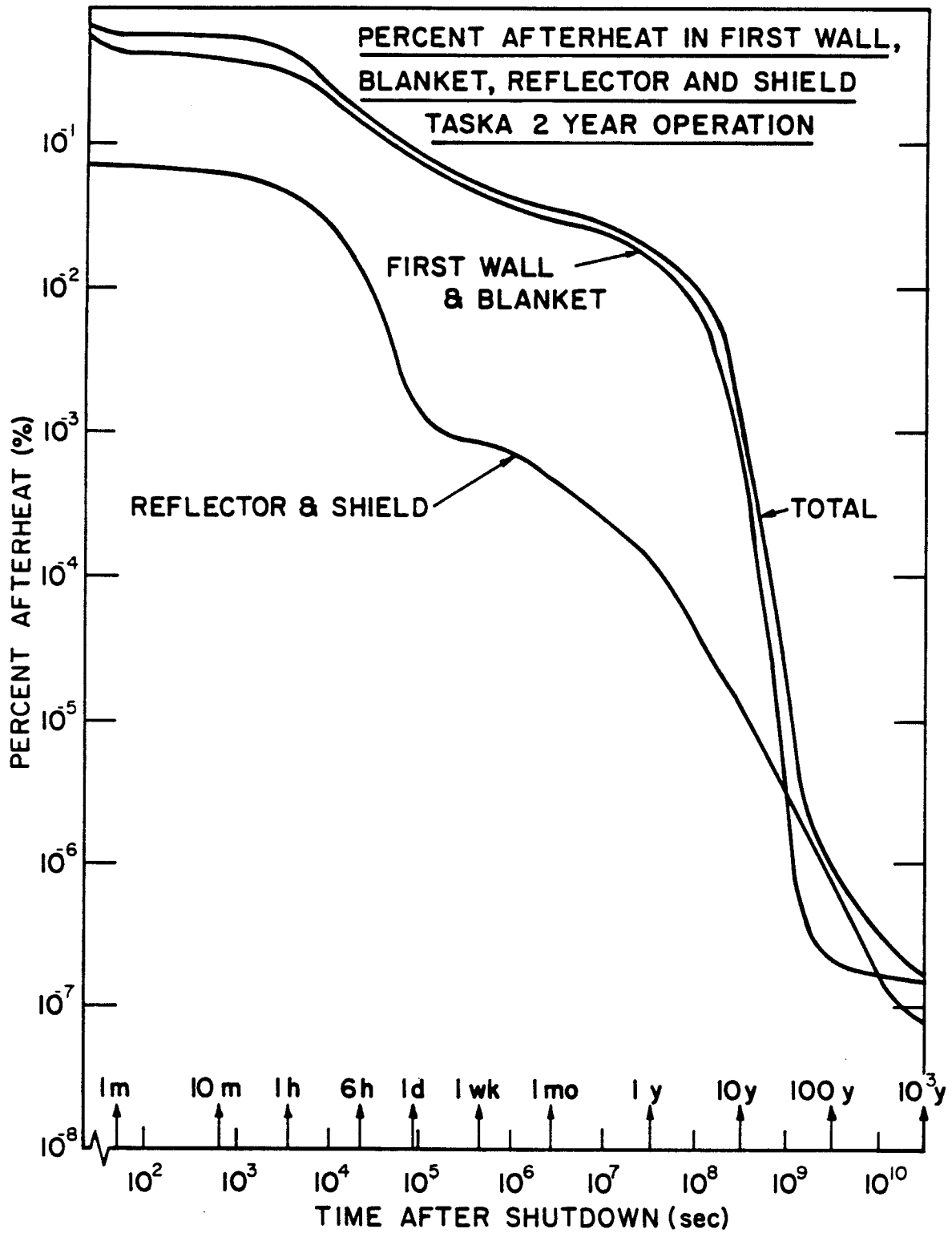
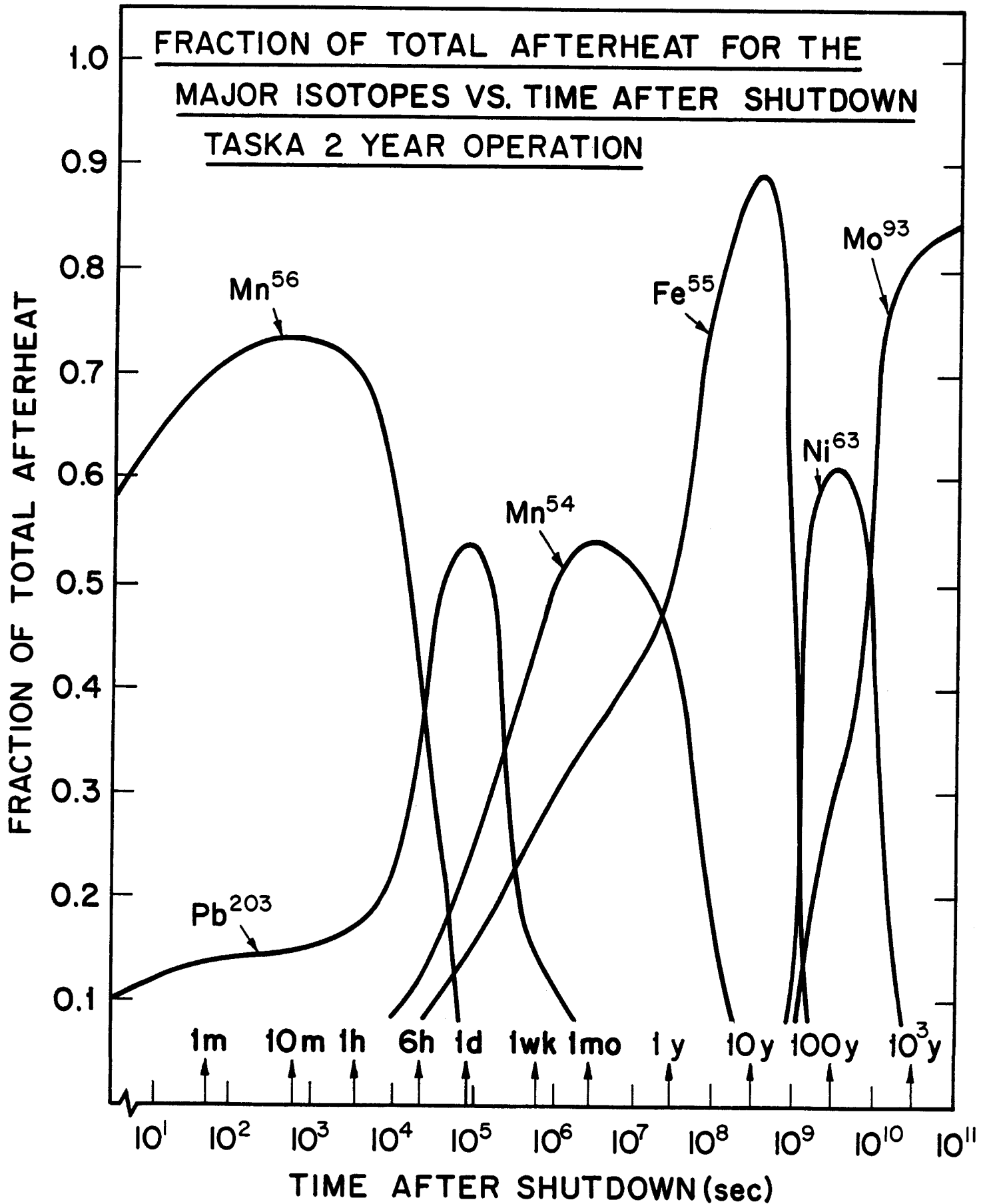


Fig. VI.2-19



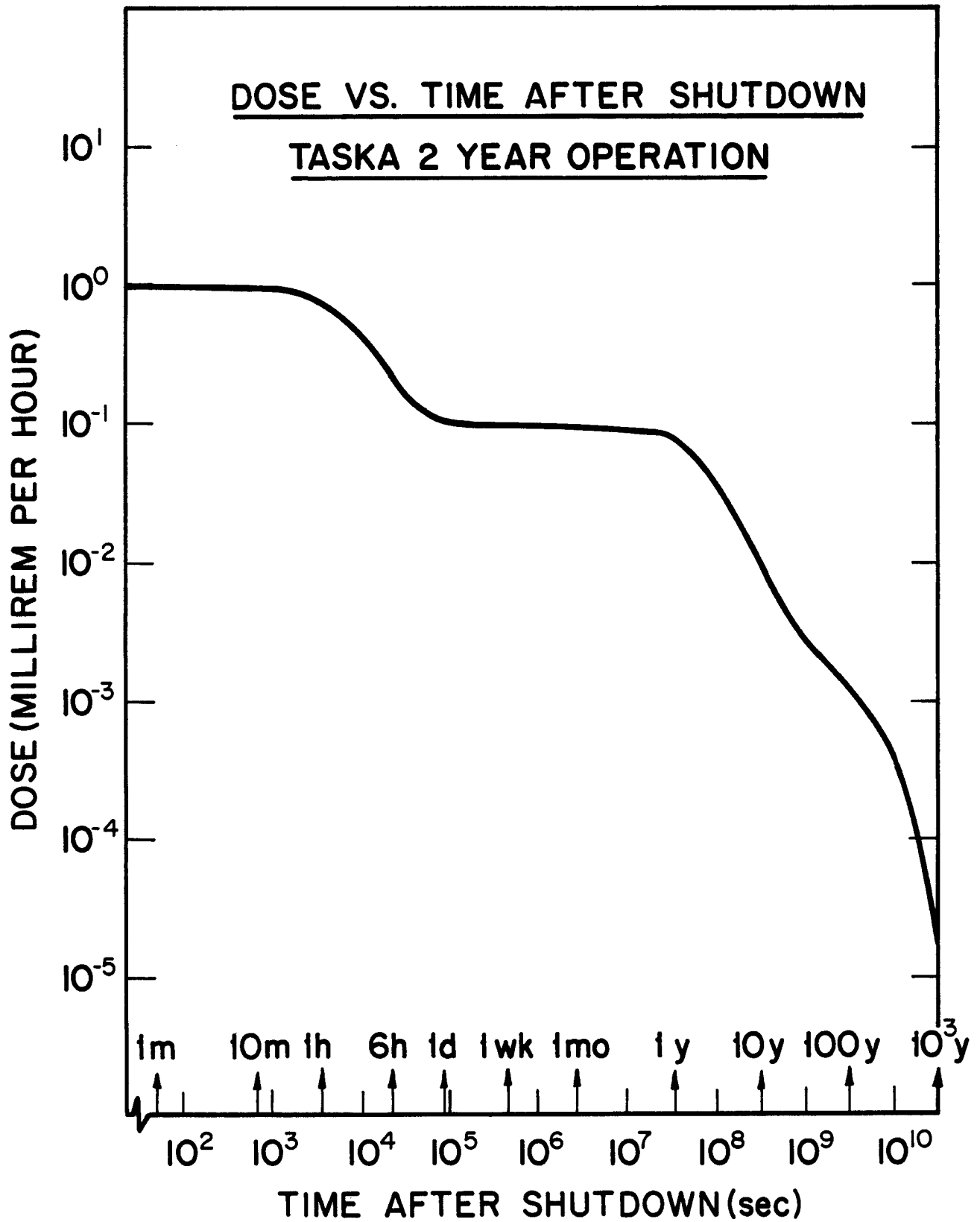


The time dependence of the afterheat differs from that of the activity because of the small decay heat of Fe-55. The isotopes Mn-56, Mn-54, and Pb-203 dominate the afterheat at shutdown. Because of the rapid decay of both of these isotopes, the afterheat drops by a factor of two in about two hours, and by a factor of ten in under a week. After these isotopes decay, Fe-55 dominates (from 1 y to 20 y) because of its abundance. The long term afterheat is governed by Ni-63 (20 y - > 300 y) and Mo-93 (300 y). The BHP follows essentially the same curve as the radioactivity. The dominant isotope here, however, is Mn-54 ( $t_{1/2} = .82$  years). Also giving significant contributions are Mn-56 and Fe-55. The Fe-55 eventually dominates until it decays and Ni-63 and Nb-93m take over. The BHP takes about one year to drop by a factor of two and 20 years to drop by a factor of 100.

As shown in the WITAMIR calculation, the activity, BHP, and afterheat in the magnets are very low and decay rapidly after shutdown (3-5 orders of magnitude in one day).

The dose rate after shutdown was calculated using two methods which yielded similar results. First, a forward calculation was done using the gamma source from the DKR code (this produces a spatially dependent case). Second, an adjoint calculation was performed using tissue kerma factors as a source in the last zone of the shield. The results of the dose calculation are shown in Fig. VI.2-20. At shutdown the dose rate outside the shield is about 1 mR/hr and is low enough to allow hands-on maintenance. It drops by one order of magnitude in about one day and by two orders of magnitude in under ten years. Since the blanket of TASKA has no major penetration and the flux falls off rapidly toward the ends of the machine, the dose calculated

Fig. VI.2-20



here (that of the midplane of the machine) should be an upper limit for the entire system and should allow immediate after shutdown hands-on maintenance.

#### VI.2.5 Test Blanket

For the TASKA central cell, a natural lithium blanket has been studied. This will allow testing of a blanket concept that is a prominent candidate for a power reactor and one which differs appreciably from the proposed permanent blanket of TASKA. Neutronic calculations were performed to determine:

- the tritium breeding.
- the conversion of the fusion neutron related energy into heat.
- the attenuation of the neutron and gamma fluxes in connection with magnet and equipment protection.
- the resistance to radiation damage.

##### VI.2.5.1 Codes and Data Libraries

The neutronic and photonic calculations were made with the one-dimensional code ONETRA.<sup>(11)</sup> A  $P_3$ - $S_8$  approximation in cylindrical geometry was used in the transport calculations. The fusion cross section library has been taken from the University of Wisconsin (UW) which is a combined  $P_3$ , VITAMIN-C,<sup>(12)</sup> MACKLIB-IV,<sup>(13)</sup> coupled 25 neutron-21 gamma group library. It contains data blocks for 48 isotopes with 4 data blocks for each isotope corresponding to a  $P_3$  Legendre expansion of the scattering cross section.

From these libraries one can generate the so-called response functions. Some important response functions for this study are:

- neutron and gamma kerma factors,
- displacement cross sections,
- $(n,2n)$ ,  $(n,3n)$  cross sections,
- hydrogen and helium production cross sections,

- ${}^6\text{Li}(n,\alpha)\text{t}$ ,  ${}^7\text{Li}$  inelastic continuous cross sections (for tritium breeding),
- absorption and total cross sections.

At KfK a special KfK-UW-library can be used immediately in ONETRA calculations as cross section blocks have the standard Los Alamos structure acceptable to the code.

The KfK cross section data set KfK INR<sup>(13)</sup> (26 neutron groups) was also used for some calculations.

#### VI.2.5.2 Calculations and Results

The calculational model of the blanket, reflector and shield together with the compositions in these zones is shown in Fig. VI.2-21. Note that TASKA, because it is basically a long cylinder, lends itself well to one-dimensional cylindrical modeling.

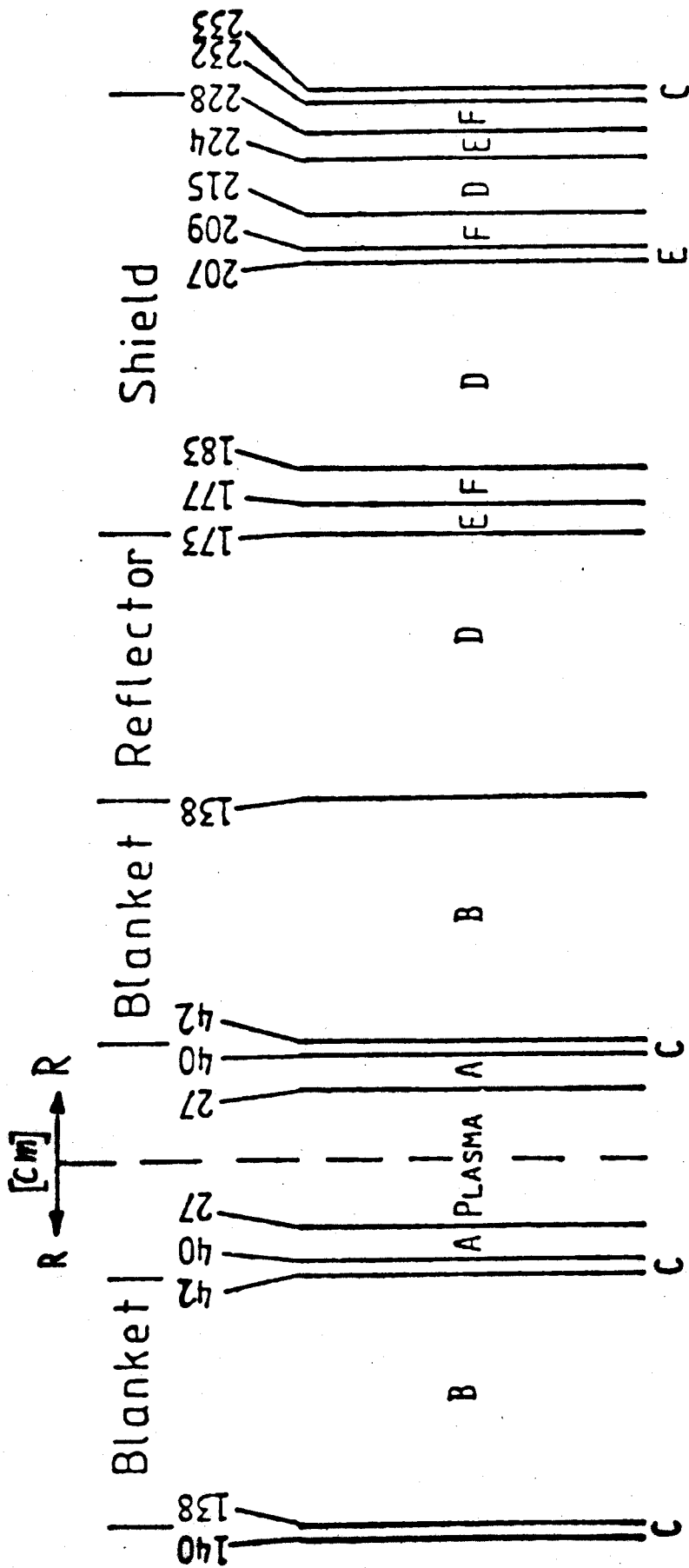
For simplifying the calculations the blanket region had to be considered as a homogeneous zone, consisting of breeding and structural materials.

Due to the  $P_3$ - $S_8$  approximation, the 46 energy groups, and the high number of fine mesh steps (about 160) in the calculations, the IBM-3033 computer at KfK needed about 6000 K-bytes of main storage.

Therefore, it was necessary to reduce the configuration to the outer radius of the reflector ( $r = 177$  cm). However, this fact does not have much influence on tritium breeding, heating rate distribution and some other parameters.

The results of the neutronic calculations are given in Table VI.2-4. These values are normalized to a wall load of  $1 \text{ MW/m}^2$ .

For a homogeneous blanket composition, consisting of 73 vol.%  $\text{Li}_{\text{nat}}$ , 7 vol.% SS 1.4970, 20 vol.% void, a total tritium breeding ratio of 1.32 T-atoms/source neutron was obtained.



- A = VACUUM  
 B = 73%  $\text{Li}_{\text{NAT}}$ ; 7% SS 1.4970; 20% VOID  
 C = 100% SS 1.4970  
 D = 95% SS 1.4970; 5%  $\text{H}_2\text{O}$   
 E = 95% Pb  
 F = 87%  $\text{B}_4\text{C}$

Fig. VI.2-21 TASKA blanket, reflector and shield composition.

Table VI.2-4. TASKA Neutronics Parameters  
(Based on 1 MW/m<sup>2</sup> Wall Load)

General

|                                           |            |
|-------------------------------------------|------------|
| 1. Breeding Ratio                         |            |
| <sup>6</sup> Li                           | 0.921      |
| <sup>7</sup> Li                           | 0.401      |
| Total                                     | 1.322      |
| 2. Energy Deposition per 14.1 MeV Neutron |            |
| Neutron                                   | 12.429 MeV |
| Gamma                                     | 5.572 MeV  |
| Total                                     | 18.001 MeV |
| 3. Energy Multiplication                  | 1.28       |

First Wall

|                                           |                           |
|-------------------------------------------|---------------------------|
| 1. Radius*                                | 40.0 cm                   |
| 2. Volume/Unit Length                     | 515.2 cm <sup>3</sup> /cm |
| 3. Thickness                              | 2.0 cm                    |
| 4. Power Density                          |                           |
| Neutron                                   | 2.98 W/cm <sup>3</sup>    |
| Gamma                                     | 5.04 W/cm <sup>3</sup>    |
| Total                                     | 8.02 W/cm <sup>3</sup>    |
| 5. DPA per yr/MW/m <sup>2</sup>           | 8.61                      |
| 6. H Production ppm/yr                    | 364.0                     |
| 7. He Production ppm/yr                   | 88.0                      |
| 8. Energy Deposition per 14.1 MeV Neutron |                           |
| Neutron                                   | 0.860 MeV                 |
| Gamma                                     | 1.457 MeV                 |
| Total                                     | 2.317 MeV                 |

Breeding Blanket

|                       |                           |
|-----------------------|---------------------------|
| 1. Radius             | 42.0 cm                   |
| 2. Volume/Unit Length | 54287 cm <sup>3</sup> /cm |
| 3. Thickness          | 96.0 cm                   |

---

\*The main design has changed to a new radius; however, the results are so insensitive that the calculations were not repeated.

Table VI.2-4. (continued)

|                                           |                           |
|-------------------------------------------|---------------------------|
| 4. Power Density                          |                           |
| Neutron                                   | 0.377 W/cm <sup>3</sup>   |
| Gamma                                     | 0.093 W/cm <sup>3</sup>   |
| Total                                     | 0.470 W/cm <sup>3</sup>   |
| 5. Energy Deposition per 14.1 MeV Neutron |                           |
| Neutron                                   | 11.472 MeV                |
| Gamma                                     | 2.840 MeV                 |
| Total                                     | 14.312 MeV                |
| <u>Reflector</u>                          |                           |
| 1. Radius                                 | 138.0 cm                  |
| 2. Volume/Unit Length                     | 38594 cm <sup>3</sup> /cm |
| 3. Thickness                              | 39.0 cm                   |
| 4. Power Density                          |                           |
| Neutron                                   | 0.0045 W/cm <sup>3</sup>  |
| Gamma                                     | 0.059 W/cm <sup>3</sup>   |
| Total                                     | 0.063 W/cm <sup>3</sup>   |
| 5. Energy Deposition per 14.1 MeV Neutron |                           |
| Neutron                                   | 0.097 MeV                 |
| Gamma                                     | 1.275 MeV                 |
| Total                                     | 1.372 MeV                 |

Although natural Li (92.5 at.%  $^7\text{Li}$ , 7.5 at.%  $^6\text{Li}$ ) was used as breeding material, the main contribution to the tritium breeding comes from  $^6\text{Li}$  (0.92 T-atoms/source neutron).

In Fig. VI.2-22 the radial breeding rate distribution is plotted for a 100 cm thick blanket.

The radial power density distribution is given in Fig. VI.2-23.

In the first wall and the reflector the gamma heating rate predominates whereas the contribution from neutrons to the heating rate is much higher in the blanket.

Hydrogen and helium production and the average dpa per year were considered for the first wall only (see Table VI.2-4).

Further neutronics parameter information is given in Section VII.3.2 (blanket modules) of this TASKA study.



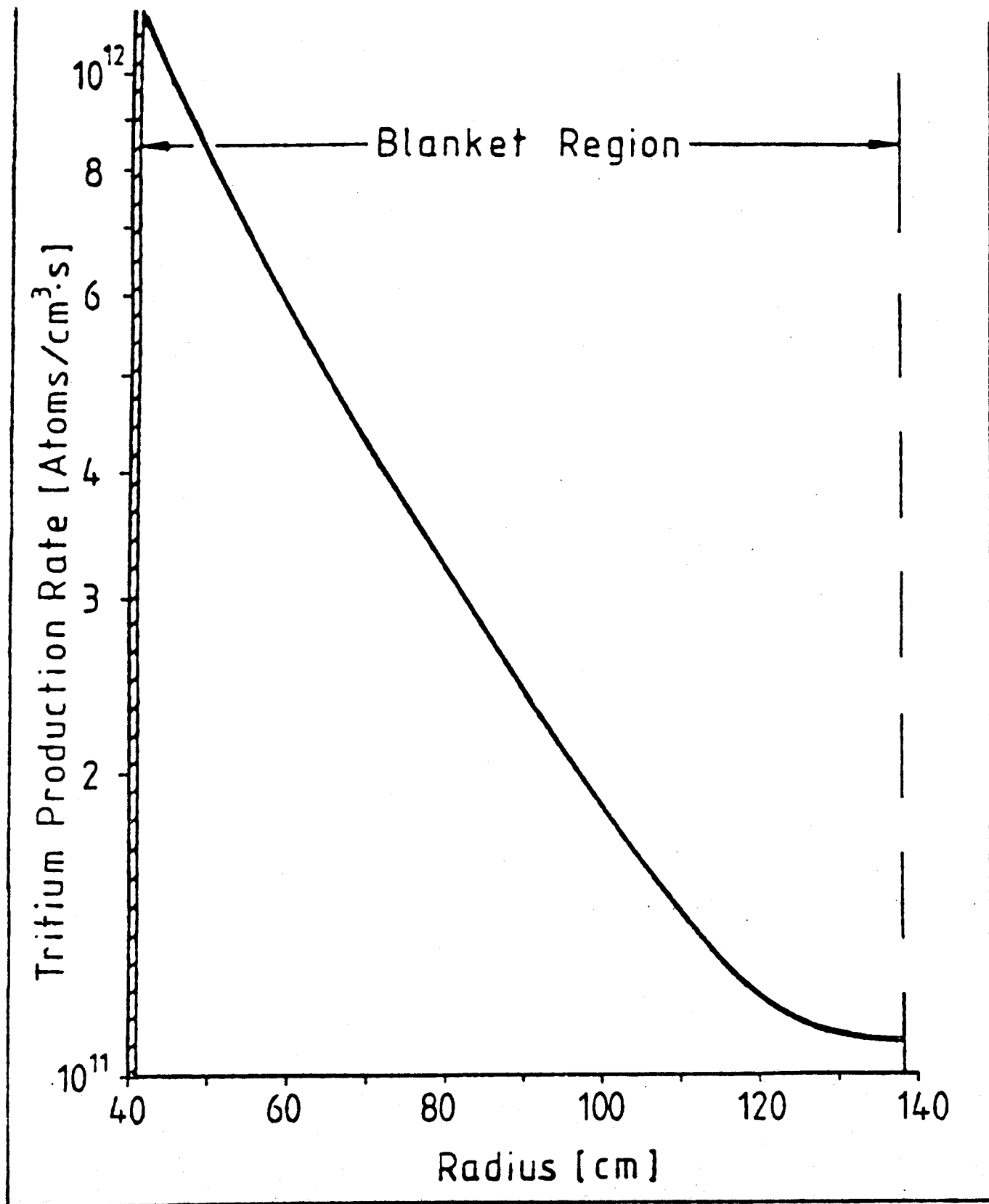


Fig. VI.2-22 Radial breeding rate distribution at 1 MW/m<sup>2</sup> neutron wall loading.

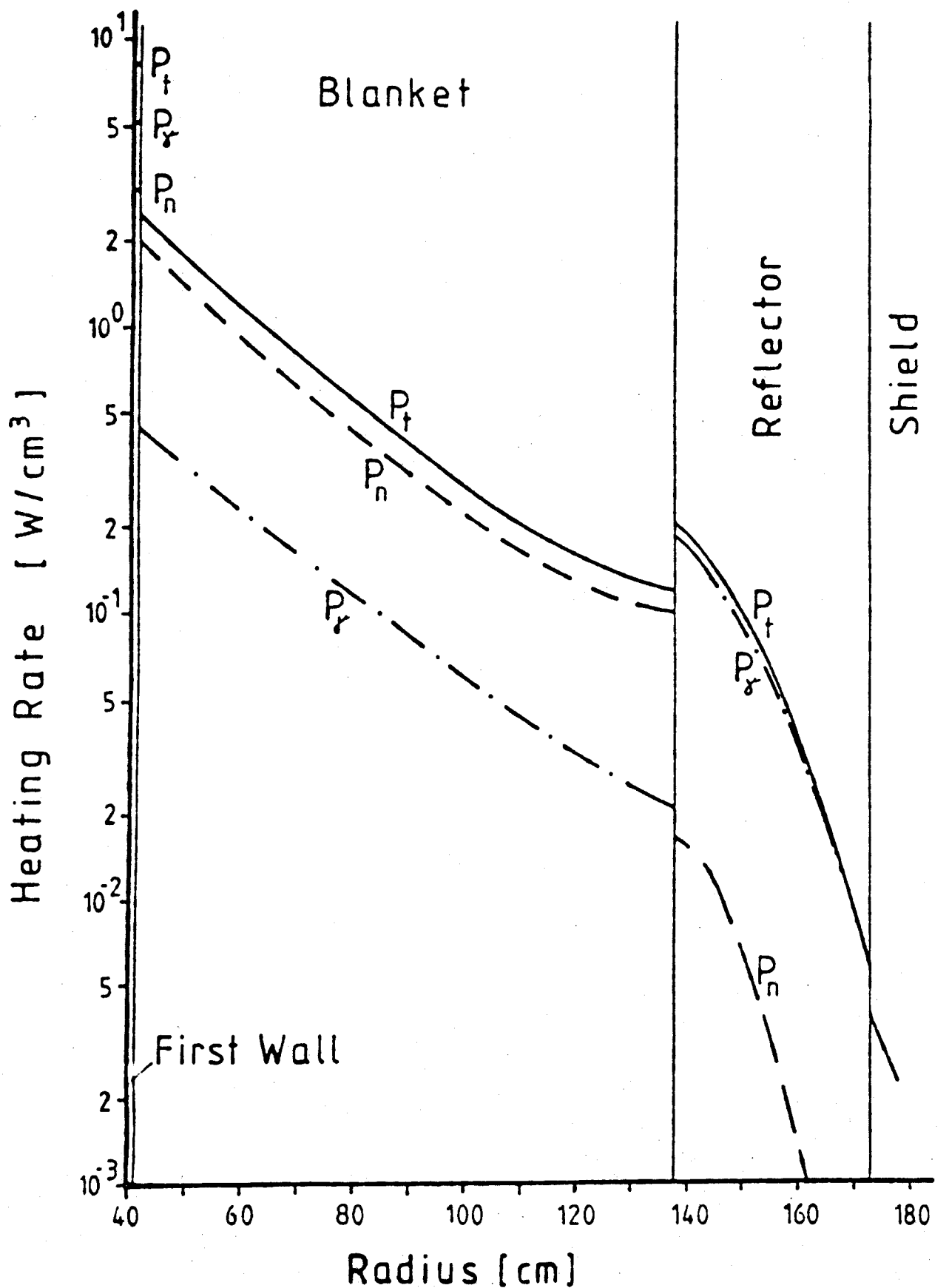


Fig. VI.2-23 Radial heating rate distribution at 1 MW/m<sup>2</sup> neutron wall loading.

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### VI.3 Thermal Hydraulics

#### VI.3.1 MHD Effects

The dominant force on a conducting fluid across the magnetic field lines is the MHD force. The effect of the MHD force is to increase the pressure drop and retard heat transfer by suppressing turbulence. In a D-T fusion reactor blanket, with the breeding material serving also as the coolant, the only severe heat transfer problem occurs at the first wall. For a TMR, the first wall surface heat load is only  $2 \text{ W/cm}^2$ . The effects of MHD on heat transfer are, therefore, not critical and the heat transfer calculations can be carried out by assuming no turbulence. The MHD pressure drop will increase the stress in the blanket and also increase pumping power. The lower magnetic field in TASKA compared to a tokamak will reduce the MHD pressure drop. Its magnitude and effects, however, still have to be assessed.

It is well known that whenever an electrically conducting fluid flows across field lines, eddy currents will appear wherever  $\text{curl } [\underline{v} \times \underline{B}]$  is non-zero. This produces a retarding force which causes extra pressure drops in the fluid. These pressure drops will be classified here into two categories<sup>(1)</sup>, the Hartmann and end-of-loop effects.

The Hartmann pressure gradient arises in fully-developed laminar flow with a uniform transverse magnetic field. For such a flow between parallel plates, in a uniform magnetic field normal to the plates, the pressure gradient is given by the following equation:<sup>(2)</sup>

$$-\frac{dP}{dx} = \frac{\mu v}{a^2} \left[ \frac{H_{\perp}^2 \tanh H_{\perp}}{H_{\perp} - \tanh H_{\perp}} + \frac{H_{\perp}^2 C}{1+C} \right] \quad (\text{VI.3-1})$$

where  $\mu$  = viscosity

$v$  = coolant velocity

$C$  = wall conductance ratio,  $\sigma_w t_w / \sigma a$

$a$  = channel half-width

$H$  = Hartmann number

$\sigma_w$  = electrical conductivity of the wall material

$\sigma$  = electrical conductivity of the coolant

$t_w$  = wall thickness.

When the fluid is in a channel with a conducting wall and under large magnetic fields ( $H \gg 1 + 1/C$ ), the last term of Equation VI.3-1 dominates, giving

$$-\frac{dP}{dx} = \frac{v B_{\perp}^2 \sigma_w t_w}{a(1 + C)} . \quad (\text{VI.3-2})$$

If the wall is strong enough to withstand the pressure and the corrosion of the lithium, the value of  $C$  is usually on the order of  $10^{-2}$  or larger; hence, Equation VI.3-2 is an excellent approximation in the present range of Hartmann numbers ( $H \approx 10^5$ ).

The end-of-loop effects are caused by gradients of  $[\underline{v} \times \underline{B}]$  in the flow direction. The resulting pressure drops are given approximately by the following equation<sup>(2)</sup>

$$\Delta P = K_{PIE} \frac{\sigma_b a v}{v_{av}} \Delta(v_{\perp}^2 B_{\perp}^2) . \quad (\text{VI.3-3})$$

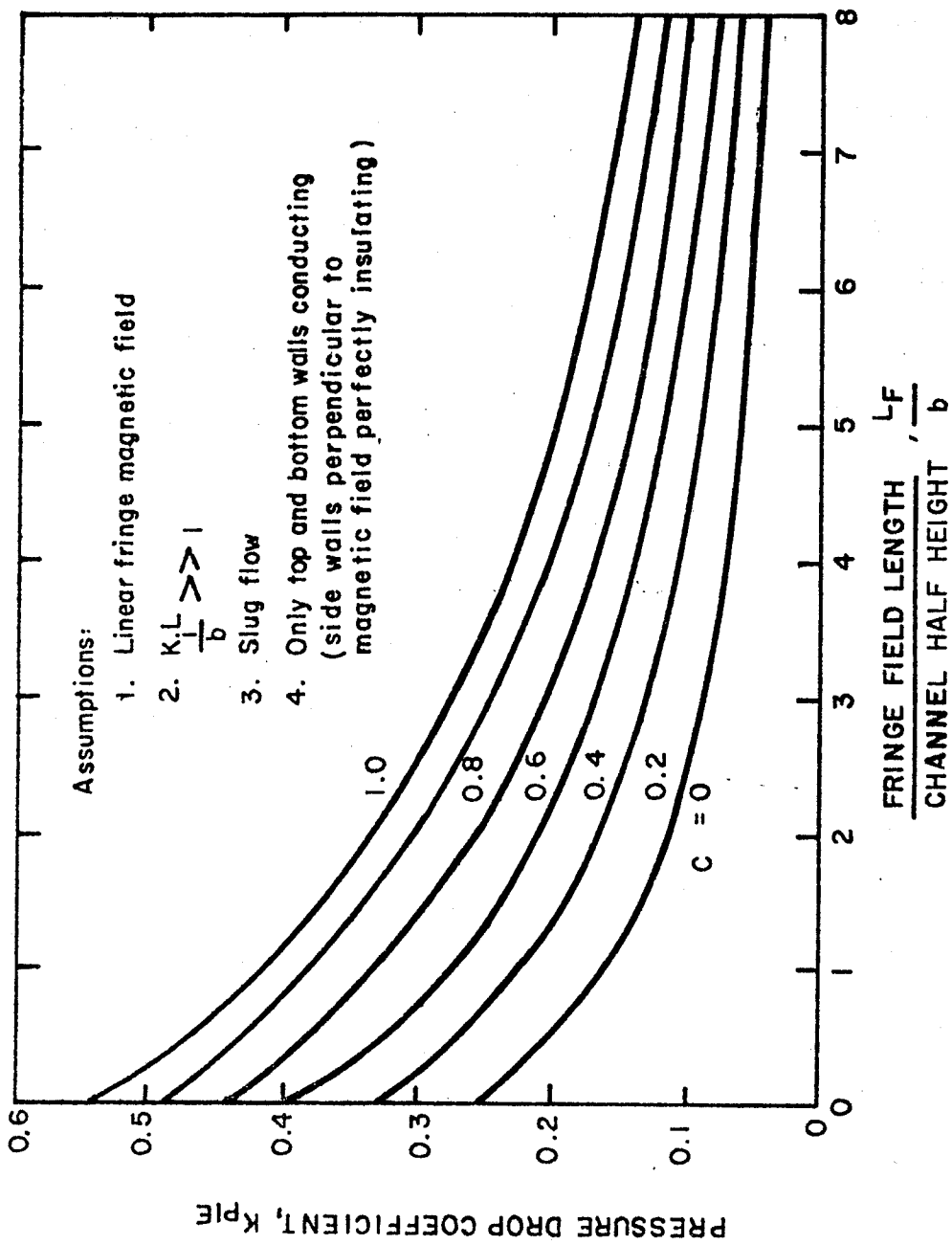


Fig. VI.3-1 - PRESSURE DROPS DUE TO ENTRANCE AND EXIT EDDY CURRENT FOR LINEAR FRINGE MAGNETIC FIELDS SEPARATED BY A LONG REGION OF UNIFORM MAGNETIC FIELD.

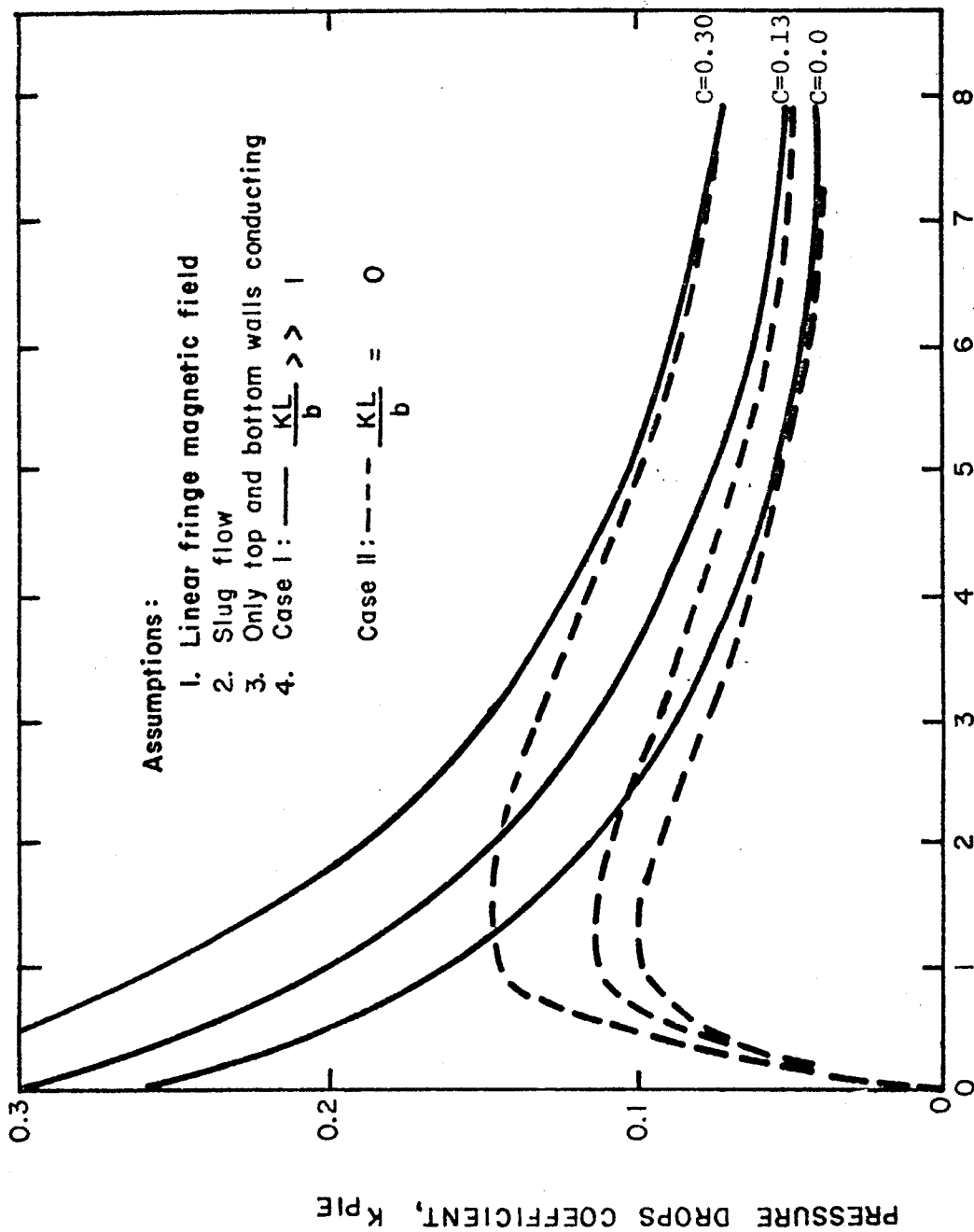


Fig. VI.3-2 PRESSURE DROP DUE TO ENTRANCE OR EXIT  
EDDY CURRENTS.

Here  $b$  = mean half-width of the local cross section in the direction normal to  $B_{\perp}$

$\bar{a}_v$  = arithmetic mean of values for initial and final cross sections.

The coefficient  $K_{PIE}$  has been presented by Hoffman and Carlson<sup>(1)</sup> for flows in tubes and ducts with varying  $B_{\perp}$ ; the results are summarized in Figures VI.3-1 and VI.3-2. We will use the same plots here to estimate  $K_{PIE}$  for changes of velocity and channel width; in this estimation  $L_F$  is taken as the downstream length of the region of varying  $vB_{\perp}$ .

The MHD pressure drop calculations for  $Li_{17}Pb_{83}$  coolant are calculated from Eq. VI.3-2 and VI.3-3. The results are summarized in Table VI.3-1. The calculations are based on local magnetic field = 2.7 tesla. The resulting pressures are shown in Fig. VI.3-3. The largest pressure drop occurs in the feed and discharge tubes due to the high local velocity. This pressure drop has been reduced by using a laminated tube structure, with a sleeve 1 mm thick, which is insulated from the main structural tube. The thin inner wall reduces the Hartmann pressure drop as can be seen from Eq. VI.3-2. The total MHD pressure drop in the reactor is 1.08 MPa, which corresponds to a pumping power requirement of 500 kW. The maximum blanket pressure is 1.02 MPa. The important blanket parameters are summarized in Table VI.3-2.

#### VI.3.2 Heat Transfer Calculations

The temperature profile in the coolant tube can be calculated by assuming only conduction. This can be justified due to the MHD effects on suppressing of turbulence. The most severe heat transfer problems occur in the first row of tubes. The energy deposited in the first row of tubes consists of a volumetric component due to nuclear heating and a surface heating load from the plasma. The effects of these two components can be decoupled. The

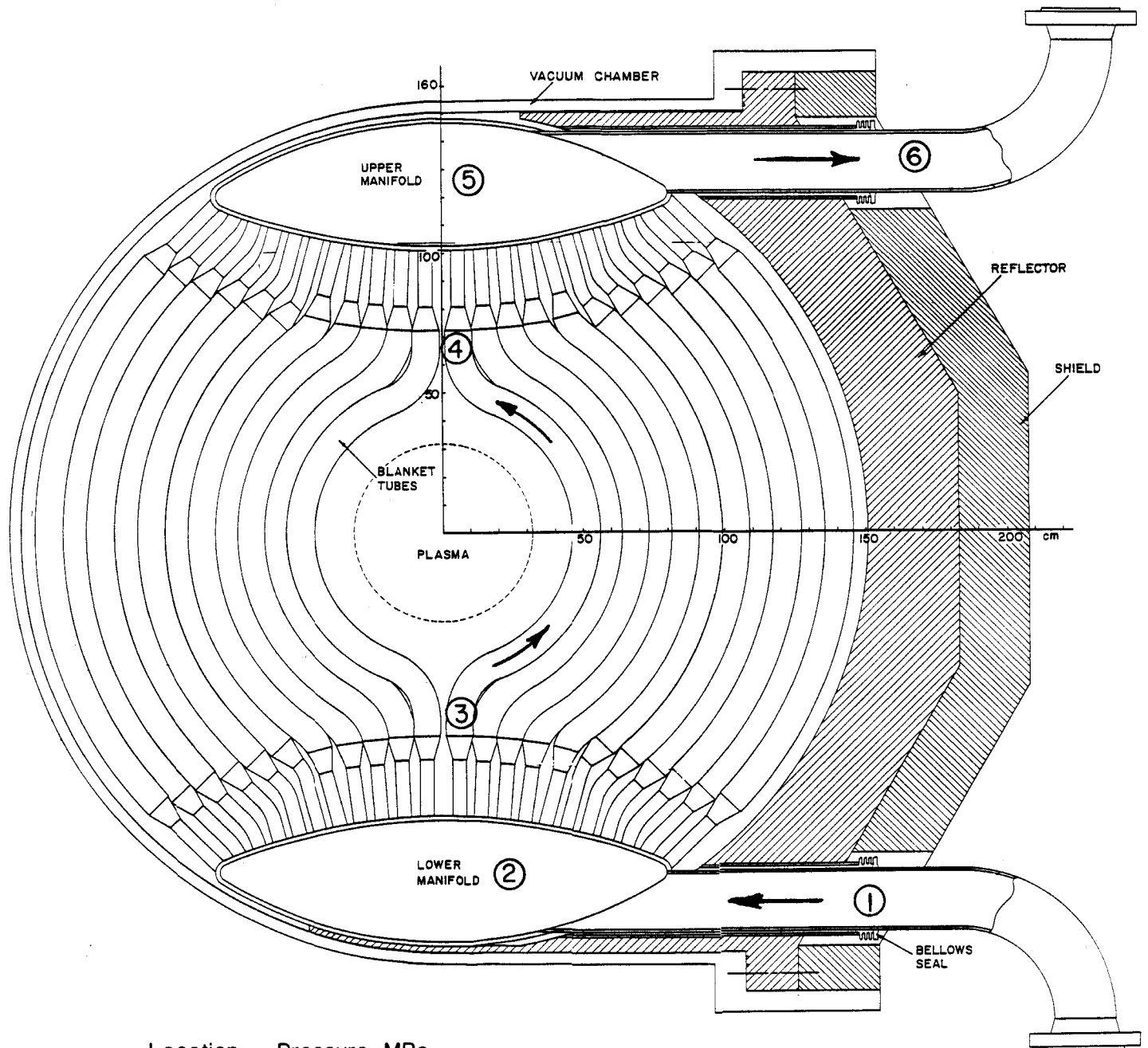


Table VI.3-1 Summary of MHD Pressure Drop Calculations

|                               | v,m/sec | r,cm  | L <sub>F</sub> /b | K <sub>PIE</sub> | t <sub>w</sub> ,mm | L,m  | ΔP <sub>E</sub> | ΔP <sub>M</sub> | ΣΔP      |
|-------------------------------|---------|-------|-------------------|------------------|--------------------|------|-----------------|-----------------|----------|
| Feed pipe<br>(discharge pipe) | 1.75    | 10    | 10                | .1               | 1.0*               | 3    | .08             | .36             | .44(x2)  |
| Connecting tube               | .42     | 2.325 | 2                 | .15              | 1.75               | .2   | .02             | .05             | .07(x2)  |
| Blanket tube                  | .097    | 4.843 |                   |                  | 2.27               | 1.45 |                 | .06             | .06      |
| Total                         |         |       |                   |                  |                    |      |                 |                 | 1.08 MPa |

\*Thickness of the inner sleeve of the pipe.

Fig. VI.3-3 Blanket Pressure Distribution



| Location | Pressure, MPa |
|----------|---------------|
| 1        | 1.46          |
| 2        | 1.02          |
| 3        | .95           |
| 4        | .61           |
| 5        | .54           |
| 6        | .10           |

Table VI.3-2 TASKA Blanket Parameters

|                                             |                          |
|---------------------------------------------|--------------------------|
| Total blanket power                         | 45 MW                    |
| Coolant inlet temperature                   | 300°C                    |
| Coolant outlet temperature                  | 400°C                    |
| Maximum blanket structural temperature      | 440°C                    |
| Minimum blanket structural temperature      | 300°C                    |
| Blanket coolant flow rate                   | $2.9 \times 10^3$ kg/sec |
| Maximum power per module                    | 8 MW                     |
| Maximum coolant flow rate per module        | $5.2 \times 10^2$ kg/sec |
| I.D. of feed (discharge) tube               | 20 cm                    |
| Coolant velocity in feed (discharge) tube   | 175 cm/sec               |
| Maximum coolant velocity in connecting tube | 42 cm/sec                |
| Maximum coolant velocity in blanket tube    | 9.7 cm/sec               |
| Maximum blanket pressure                    | 1.02 MPa                 |
| Total MHD pressure drop                     | 1.08 MPa                 |
| Pumping power required                      | 500 kW                   |

volumetric heating load increases the bulk temperature in the coolant, while the surface heating load generates a temperature gradient. The results of these two effects can then be superimposed to obtain a complete temperature profile.

The coolant temperature range is chosen between 300 and 400°C. The maximum volumetric heating is 11.24 W/cm<sup>3</sup>. The total coolant passage length is 145 cm. Therefore, the required coolant velocity is 9.7 cm/sec. The minimum coolant residence time in the blanket is 15 seconds.

The neutronic calculations are carried out for a homogenized cylindrical blanket. However, the curvature of the first wall is not a major factor in the model developed here and the first bank of tubes is assumed to be an infinite plate with a thickness of 10 cm. The mathematical problem then is conduction heat transfer to an infinite plate moving at a constant velocity with a uniform surface heating load, as shown in Fig. VI.3-4. If the heat conduction in the y direction is small in comparison to the x direction, the solution for the temperature can be obtained from the solution of T(t,x) from stationary systems with a constant heat input. The solution for an infinite plate with a constant heat input is:<sup>(3)</sup>

$$[T(t,x) - T_0] = \frac{2q}{k} (\alpha t)^{1/2} \sum_{n=1}^{\infty} \left[ \text{ierfc} \frac{(2n-1)R-x}{2(\alpha t)^{1/2}} + \text{ierfc} \frac{(2n-1)R+x}{2(\alpha t)^{1/2}} \right]$$

in which T(t,x)      temperature in a stationary system at time t and  
distance x from surface

T<sub>0</sub>                  initial temperature

q                    surface heat flux

k                    thermal conductivity

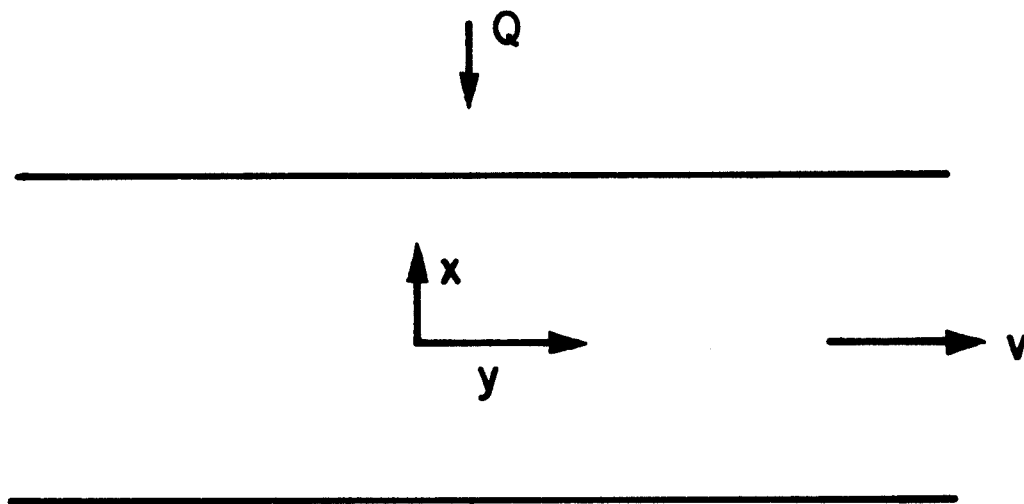


Fig. VI-3-4 GEOMETRY OF THE  
HEAT TRANSFER CALCULATIONS

$\alpha$             thermal diffusivity  
 $R$             thickness of the plate.

The required solution can then be obtained by

$$\theta_1(y,x) = T\left(\frac{y}{v}, x\right) - T_0 . \quad (\text{VI.3-5})$$

in which  $\theta(y,x)$     temperature rise  
 $v$             coolant velocity.

The bulk temperature rise due to nuclear heating is

$$\theta_2(y) = 100 \times \frac{y}{L} \quad (\text{VI.3-6})$$

in which  $L$  is the length of coolant passage. The superimposed temperature in the entire coolant passage is then

$$T(y,x) = \theta_1(y,x) + \theta_2(y) + T_0 . \quad (\text{VI.3-7})$$

The temperature distribution of the blanket is calculated by this technique. A finite difference calculation was also performed to confirm the results of this calculation and the difference is less than 5°C. Fig. VI.3-5 shows the maximum temperature of the first wall as a function of distance from the entrance. The temperature difference across the first wall is 4°C. Figure VI.3-6 shows the temperature across the coolant tube at the coolant exit. The maximum blanket temperature is 440°C. This is considerably higher

Fig. VI-3-5 FIRST WALL TEMPERATURE  
AS A FUNCTION OF DISTANCE FROM  
COOLANT INLET

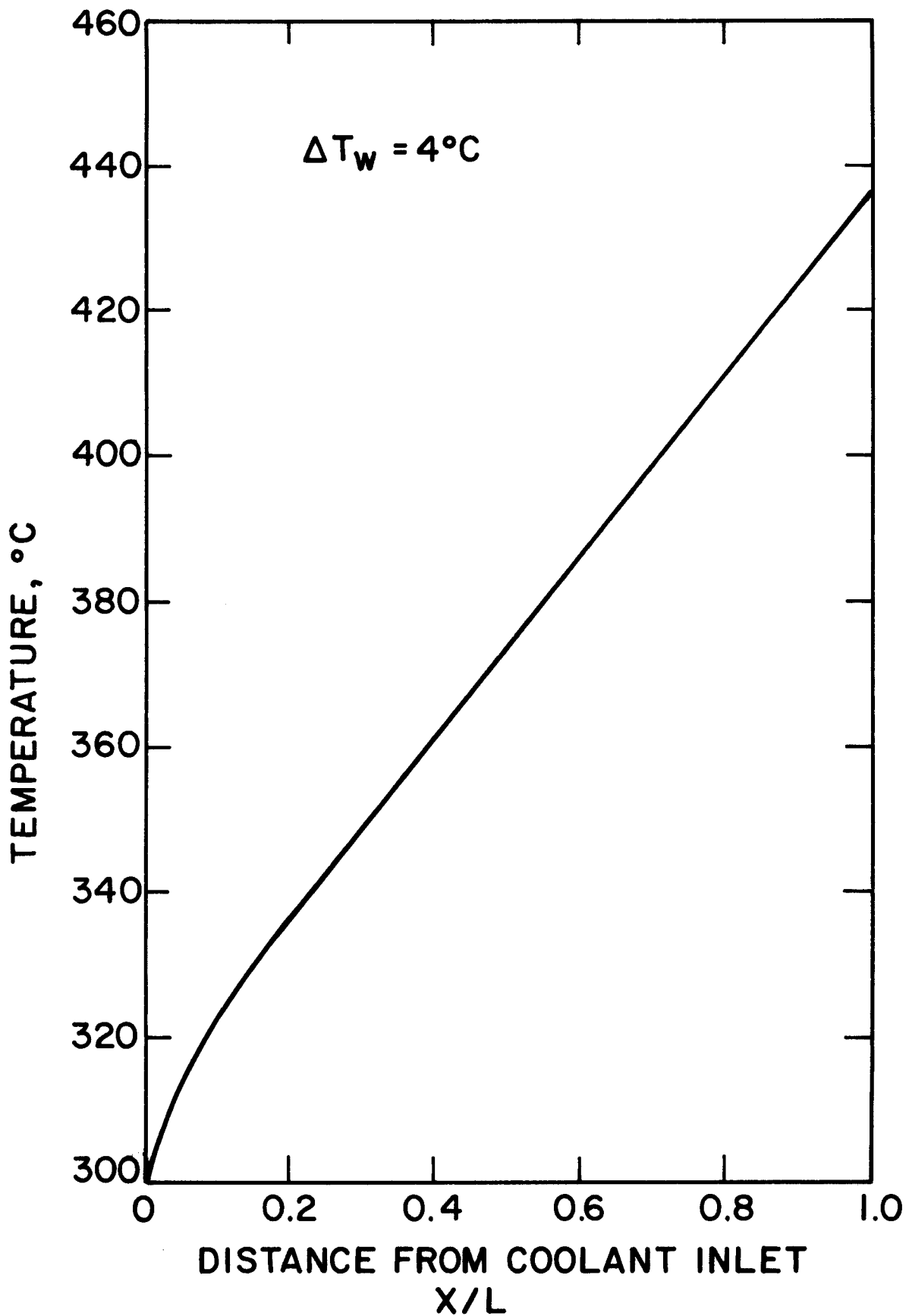
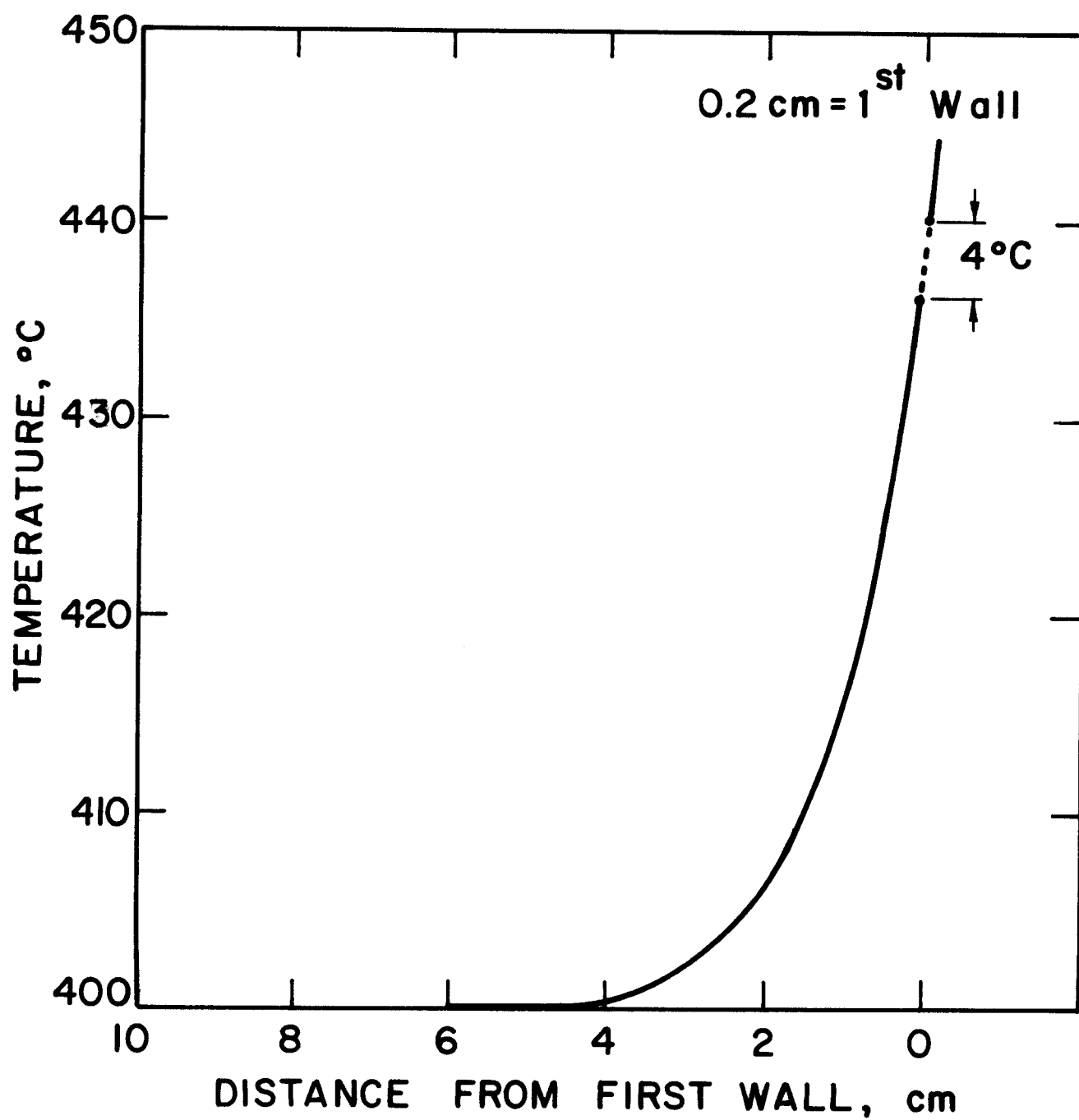


Fig. VI-3-6 COOLANT TEMPERATURE  
AT THE EXIT





from the plasma and nuclear heating from the first wall structure.

The large temperature difference from the coolant due to the bulk of the coolant is caused by MHD effects. The calculation assumes the absence of turbulence and heat transfer by conduction only. The total surface heating load on the coolant is  $4 \text{ watt/cm}^2$  which, with  $\Delta T = 36^\circ\text{C}$ , corresponds to a very small heat transfer coefficient of  $.11 \text{ watt cm}^2\text{-}^\circ\text{C}$ . This heat transfer coefficient is consistent with a pure conduction system with  $hD/k = 7$ . Therefore, a first wall design of this type is only applicable to a power system with a very small surface heat flux.

#### VI.3.3 Cooling of the Reflector and the Shield

The coolant tube arrangement in the reflector is shown in Fig. VI.3-7. The distance between two coolant channels is 20 cm. The maximum volumetric energy deposition is  $.1 \text{ W/cm}^3$ . Therefore, the maximum temperature difference between two coolant channels is less than  $30^\circ\text{C}$ . The maximum heat flux to the coolant wall is  $.1 \text{ W/cm}^2$  which can easily be removed by either helium or water. For this design, water will be used as the coolant. The inlet or outlet water temperatures are  $80$  and  $140^\circ\text{C}$ , respectively. The total power in the reflector is  $5.08 \text{ MW}$ , which corresponds to a coolant flow rate of  $20 \text{ kg/sec}$ . Since there is no need for good heat transfer, the maximum velocity needed can be less than  $1 \text{ m/sec}$ . The pressure drop is  $.01 \text{ MPa}$ . The pumping power required is less than  $1 \text{ kW}$ .

The shield is also cooled by forced convection water. For simplicity of the design, the water condition is chosen to be the same as the reflector. The shield design includes the two end modules of the central cell. The total energy deposited of  $11.3 \text{ MW}$  in the shield is, therefore, large compared to a conventional shield. The maximum energy flux to the coolant is  $\sim 10 \text{ W/cm}^2$

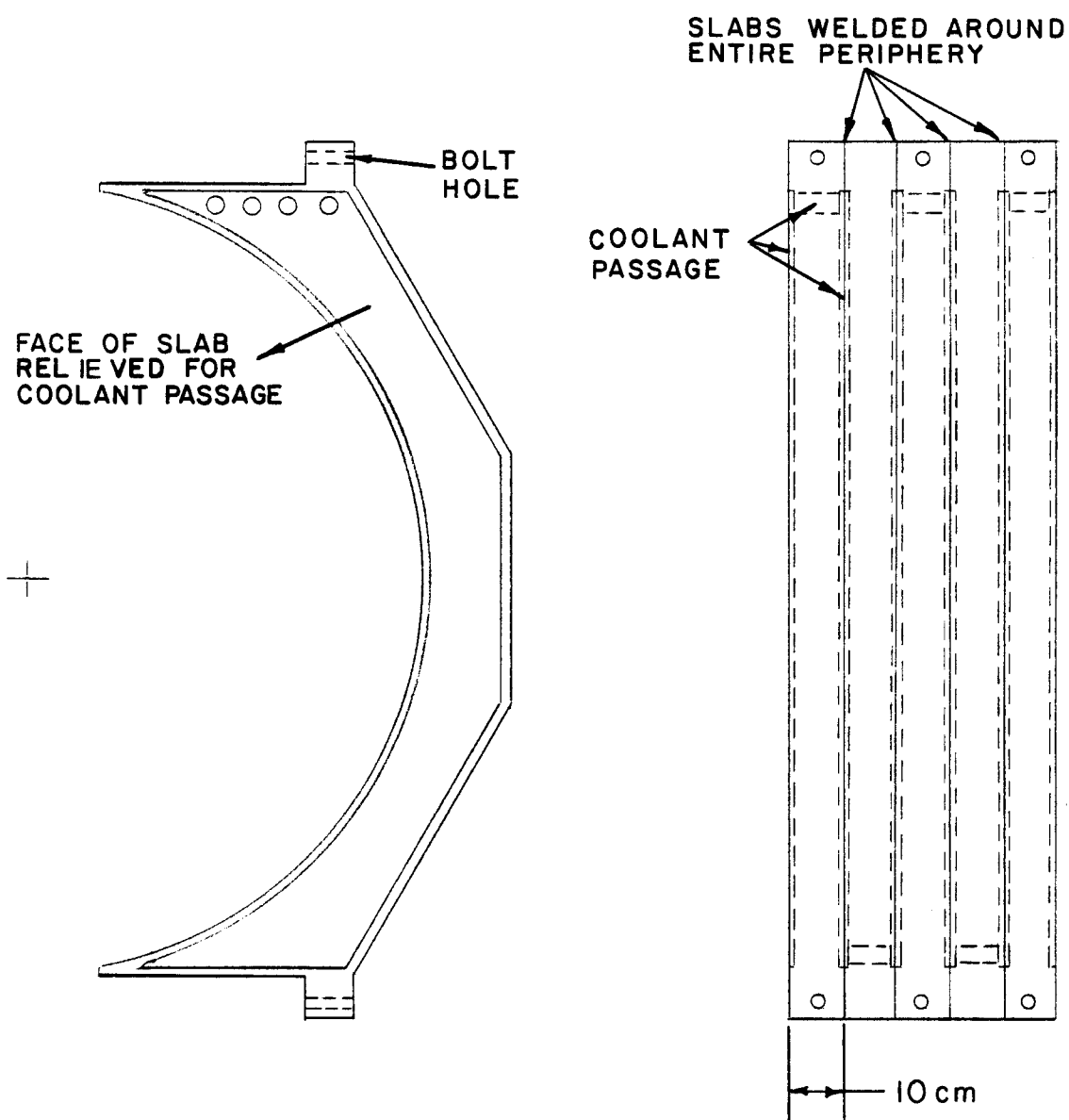


Fig. VI.3-7 Reflector Coolant Passage

conventional shield. The maximum energy flux to the coolant is  $\sim 10 \text{ W/cm}^2$  which can also be handled easily by forced convection water. With a water velocity of 1 m/sec in a 1 cm ID coolant tube, a heat transfer coefficient of  $1 \text{ W/cm}^2\text{-}^\circ\text{C}$  can be obtained, which corresponds to a film temperature drop of only  $10^\circ\text{C}$ . This is readily acceptable. The total coolant flow rate is 45 kg/sec. With a pressure drop of .01 MPa, the pumping power required is 2 kW.

The parameters of the reflector and the shield are summarized in Table VI.3-3.

Table VI.3-3 Parameters of the Shield and the Reflector

|                                          |                       |
|------------------------------------------|-----------------------|
| Coolant inlet temperature                | 80°C                  |
| Coolant outlet temperature               | 140°C                 |
| Total reflector power                    | 5.08 MW               |
| Maximum reflector energy deposition      | .1 W/cm <sup>3</sup>  |
| Maximum heat flux to reflector coolant   | .5 W/cm <sup>2</sup>  |
| Maximum reflector structural temperature | 170°C                 |
| Minimum reflector structural temperature | 80°C                  |
|                                          |                       |
| Total shield power                       | 11.3 MW               |
| Maximum shield energy deposition         | 7.5 W/cm <sup>3</sup> |
| Maximum heat flux to reflector coolant   | 10 W/cm <sup>2</sup>  |
| Maximum shield structural temperature    | 165°C                 |
| Minimum shield structural temperature    | 80°C                  |
|                                          |                       |
| Coolant velocity                         | 1 m/sec               |
| Total coolant flow                       | 65 kg/sec             |
| Total pumping power                      | 3 kW                  |

### References for Section VI.3

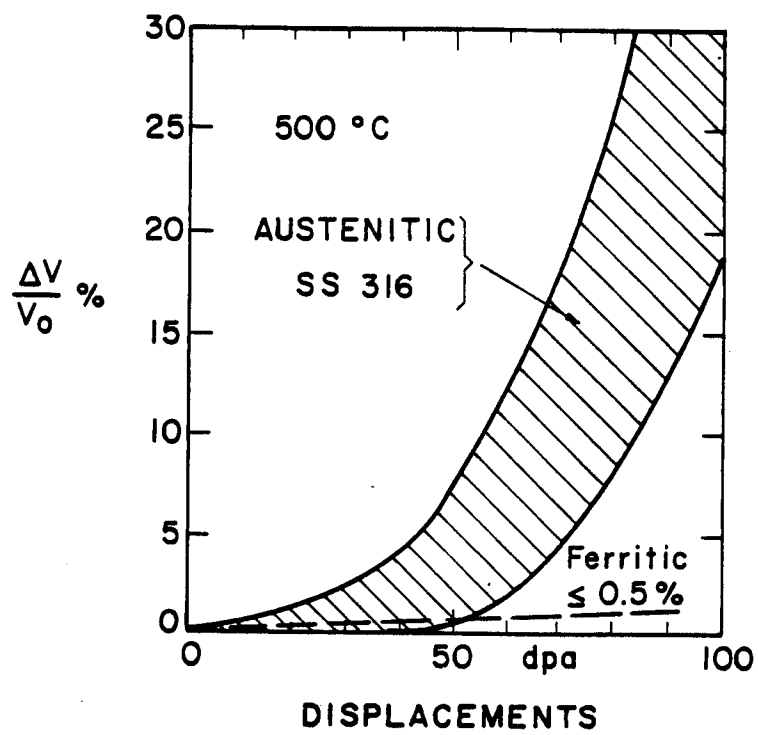
1. M.A. Hoffman and G.A. Carlson, "Calculation Techniques for Estimating the Pressure Losses for Conducting Flows in Magnetic Fields", USAEC Report UCRL-51010, 1971.
2. G.W. Sutton, and A. Sherman, "Engineering Magnetohydrodynamics", McGraw Hill, New York, 1956.
3. H.S. Carslaw and J.C. Jaeger, "Conduction of Heat in Solids", Oxford University Press, 1959.

## VI.4 Structural Materials for TASKA

### VI.4.1 General Considerations

Since TASKA is to be considered as one of the first fusion materials test facilities, structural materials to be used for its design must be those for which there exist a sufficiently broad data base in high radiation fields and in high temperature environments. Accordingly, the two alloy classes considered as potential candidates were the austenitic stainless steels and the ferritic (or martensitic) steels. Both materials classes were judged to be suitable structural materials for the blanket and the vacuum boundary. The major disadvantage of the austenitic stainless steels is their large propensity for swelling. This is clearly demonstrated by considering the swelling behavior of both austenitic type 316 stainless steel and ferritic steels as based on breeder reactor irradiation at about 500°C. As seen in Fig. VI.4-1, austenitic steels exhibit a rapid rate of swelling after an incubation dose varying between 30 and 60 dpa, depending on the particular heat treatment. In contrast, ferritic steels have not exhibited more than 0.5% swelling in similar irradiation experiments.<sup>(1)</sup> Based on the breeder reactor experience and the review given below, it appears that the blanket components exposed to the highest neutron fluence in TASKA would swell about 15% to 20% if made of 20% CW type 316 stainless steel. Although a linear expansion of 5% to 7% due to swelling can be accommodated with the blanket designs contemplated for TASKA, it is doubtful that a significantly larger expansion could be accommodated in a power producing tandem mirror fusion reactor. Therefore, although the austenitic stainless steels were judged suitable for TASKA they do not seem to be the answer for future fusion reactors.

Fig. VI.4-1. Swelling behavior of austenitic and ferritic steels irradiated in EBR-II.



In order to gain fusion reactor relevant experience with a structural material, it was decided to select ferritic steels as the prime candidates for TASKA. However, in the course of this design study, it was discovered that ferritic steels used as structural materials for the blankets may significantly perturb and lower the magnetic field and its distribution in the central cell region, particularly at either end of the central cell. The amount of field perturbation and reduction that can be tolerated in a mirror reactor is, at the present time, not known. It appears, however, that these issues can be resolved by appropriate designs of the blanket modules and by a modest increase of the magnetic field in the central cell region.

It will be evident from the discussions below that ferritic steels are to be preferred over austenitic steels for the following reasons:

- A) Much greater resistance to neutron induced void formation and metal swelling.
- B) A greater resistance to in-reactor creep below 600°C.
- C) Better ductility retained after high temperature fission neutron irradiation.
- D) Better thermal stress resistance, at least by a factor of two.
- E) Better compatibility with the  $\text{Pb}_{83}\text{Li}_{17}$  breeder/coolant material.
- F) Considerably more experience in heat exchanger design and aqueous environments.
- G) Lower material cost per unit weight.
- H) Less demand on Cr (12 vs. 18%) and Ni (0.5 vs. ~ 8%) resources which generally have to be imported.



The specific choice of HT-9 over other ferritic steels should not be taken too seriously. This alloy system has been identified as one particularly attractive to the fast breeder and, as such, gives us a convenient data base from which to extrapolate its performance in a fusion environment. It is quite possible that other ferritic steels might be developed to give even better properties in a fusion environment and future studies ought to be open to new ideas in this area.

#### VI.4.2 Physical and Thermal Properties of HT-9

The general class of alloys that we considered are the ferromagnetic steels which contain roughly 9 to 13% Cr with some small addition of various strengthening and stabilizing elements such as Mo. These steels are conventionally used in the normalized and tempered condition for high temperature applications. Although the heat treatment can result in either a tempered martensite or bainite structure, depending on the alloy and thermal treatment parameters, this class of materials is usually referred to as "martensitic" stainless steels for simplicity.

Because of the encouraging and rapidly expanding data base from the U.S. Fast Breeder Reactor Program and the significant operational history in Europe, the alloy Sandvik HT-9 (DIMX20CrMoWV121) was selected for TASKA. The normal composition of this alloy is given in Table VI.4-1 and Fig. VI.4-2<sup>(2)</sup> gives the binary section of the Fe-Cr-C phase diagram at 12%. For high temperatures and extended time service, the alloy is usually heat treated at 1050°C for 0.5 hours in the austenite zone (Fig. VI.4-3), air cooled and then tempered at 780°C.

The thermal properties of the 9 Cr-1 Mo system, which should be close to HT-9 are given in Table VI.4-2.<sup>(3)</sup>

Table VI.4-1. Nominal Composition of HT-9 Alloy

| <u>Element</u> | <u>Wt%</u> |
|----------------|------------|
| Fe             | Balance    |
| Cr             | 11.5       |
| Mo             | 1.0        |
| Ni             | 0.5        |
| W              | 0.5        |
| Mn             | 0.5        |
| V              | 0.3        |
| Si             | 0.25       |
| C              | 0.20       |

Fig. VI.4-2  
(Reference 2)

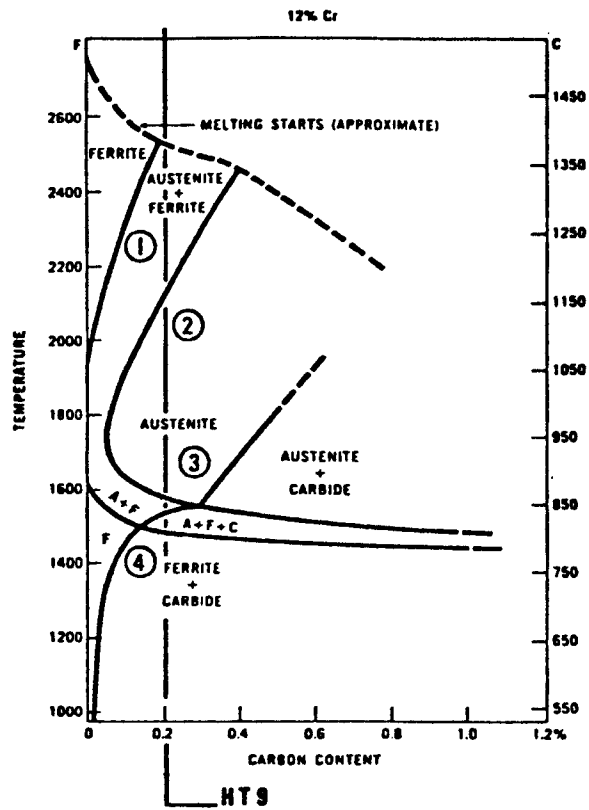


Fig. VI.4-3  
(Ref. 2)

## TYPICAL HEAT TREATMENT PROFILE FOR A MARTENSITIC STAINLESS STEEL

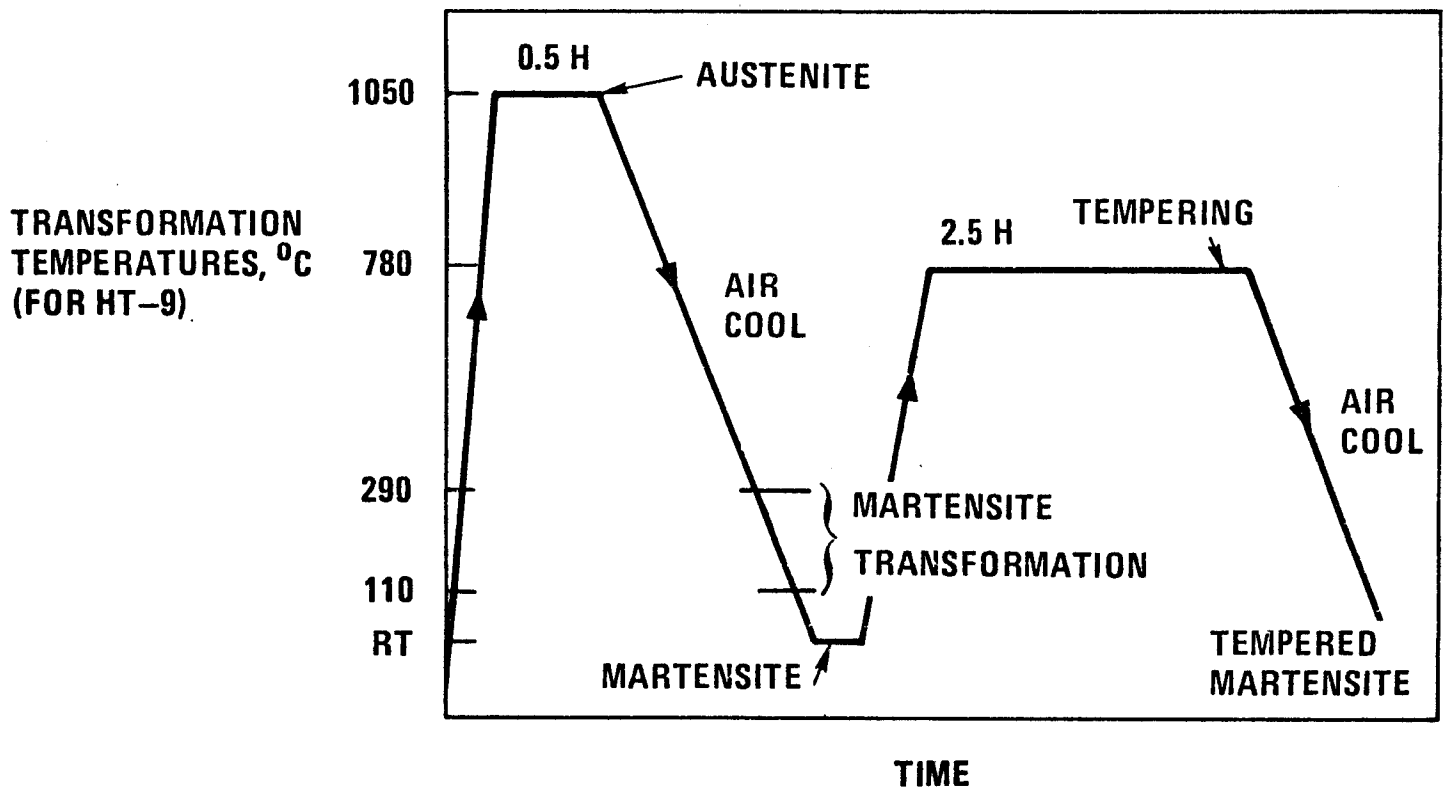


Table VI.4-2. Selected Thermal Properties of the 9 Cr-1 Mo Steel<sup>(20)</sup>

| Temperature |     | Thermal                          | Thermal                     | Specific                          | Instantaneous           | Mean Coefficient        |
|-------------|-----|----------------------------------|-----------------------------|-----------------------------------|-------------------------|-------------------------|
| °F          | °C  | Conductivity                     | Diffusivity                 | Heat                              | Coefficient of          | of Thermal              |
|             |     | $\text{W cm}^{-1} \text{C}^{-1}$ | $\text{cm}^2 \text{s}^{-1}$ | $\text{cal g}^{-1} \text{C}^{-1}$ | Thermal Linear          | Linear Expansion        |
|             |     |                                  |                             | $\times 10^3$                     | Expansion               | Expansion               |
|             |     |                                  |                             |                                   | $10^{-6} \text{C}^{-1}$ | $10^{-6} \text{C}^{-1}$ |
| 70          | 21  | .222                             | .066                        | .105                              | 3.23                    |                         |
| 100         | 38  | .227                             | .067                        | .106                              | 3.28                    | 3.25                    |
| 150         | 66  | .235                             | .067                        | .109                              | 3.35                    | 3.29                    |
| 200         | 93  | .242                             | .067                        | .112                              | 3.42                    | 3.34                    |
| 250         | 121 | .249                             | .068                        | .114                              | 3.49                    | 3.39                    |
| 300         | 149 | .254                             | .068                        | .117                              | 3.56                    | 3.42                    |
| 350         | 177 | .260                             | .067                        | .120                              | 3.63                    | 3.46                    |
| 400         | 204 | .263                             | .067                        | .123                              | 3.69                    | 3.49                    |
| 450         | 232 | .267                             | .066                        | .126                              | 3.75                    | 3.52                    |
| 500         | 260 | .270                             | .065                        | .129                              | 3.82                    | 3.56                    |
| 550         | 288 | .273                             | .064                        | .133                              | 3.87                    | 3.59                    |
| 600         | 316 | .275                             | .063                        | .136                              | 3.93                    | 3.62                    |
| 650         | 343 | .277                             | .062                        | .140                              | 3.98                    | 3.65                    |
| 700         | 371 | .277                             | .061                        | .144                              | 4.04                    | 3.68                    |
| 750         | 399 | .279                             | .059                        | .147                              | 4.09                    | 3.70                    |
| 800         | 427 | .279                             | .057                        | .152                              | 4.13                    | 3.73                    |
| 850         | 454 | .280                             | .055                        | .157                              | 4.18                    | 3.76                    |
| 900         | 482 | .279                             | .053                        | .163                              | 4.23                    | 3.79                    |
| 950         | 510 | .279                             | .051                        | .169                              | 4.27                    | 3.81                    |
| 1000        | 538 | .279                             | .050                        | .175                              | 4.31                    | 3.83                    |
| 1050        | 566 | .277                             | .047                        | .183                              | 4.35                    | 3.86                    |
| 1100        | 593 | .277                             | .045                        | .191                              | 4.39                    | 3.89                    |
| 1150        | 621 | .275                             | .043                        | .201                              | 4.43                    | 3.92                    |
| 1200        | 649 | .273                             | .040                        | .213                              | 4.46                    | 3.93                    |
| 1250        | 677 | .272                             | .037                        | .229                              | 4.49                    | 3.96                    |
| 1300        | 704 | .270                             | .034                        | .250                              | 4.52                    | 3.98                    |
| 1350        | 732 | .268                             | .030                        | .276                              | 4.55                    | 4.00                    |
| 1375        | 746 |                                  | .028                        | .295*                             |                         |                         |
| 1400        | 760 | .263                             | .038                        | .216                              | 4.57                    | 4.02                    |
| 1450        | 788 | .260                             | .043                        | .190                              | 4.60                    | 4.04                    |
| 1500        | 816 | .260                             | .046                        | .178                              | 4.62                    | 4.06                    |

\* Extrapolated value

The electrical conductivity of the ferritic steels is in the range of 0.002 to 0.005 mho depending on the composition and heat treatments. No specific value was found for HT-9.

#### VI.4.3 Use of Martensitic Steels

It is fortunate that there are so many applications for martensitic steels throughout the U.S. and the world. A recent summary by scientists at General Atomic revealed that there are major applications for high temperature martensitic stainless steels in the following areas:<sup>(4)</sup>

- Fossil Power Plants

- Superheaters

- Reheaters

- Evaporators

- Steam piping

- Nuclear Power Plants

- Superheaters, reheaters, U.K. Hinkley Point "B" AGR

- Steam generator: U.K. PFR

- He compressors, circulators: U.S. Fort St. Vrain HTGR

- Other Applications

- Gas and steam turbine components

- Aircraft and missile thin wall pressure vessels

- Petroleum industry pressure vessel linings

It was found that the 9-12 Cr steels were used in foreign fossil power plants under the following conditions:

| <u>Location</u> | <u>Usage</u> | <u>Alloy</u> | <u>Temp °C</u> | <u>Service Status</u> |
|-----------------|--------------|--------------|----------------|-----------------------|
| U. K.           | 13 plants    | 9 CrMo(V)    | 450-660        | < 110,000 hrs.        |
|                 | 12 plants    | 12 CrMo      | 430-850        | < 45,000 hrs.         |
| Austria         | 96 Tons      | 9 CrMo       | < 610          | < 110,000 hrs.        |
| Belgium         |              |              |                |                       |
| Czechoslovakia  |              |              |                |                       |
| Germany         |              |              |                |                       |
| Switzerland     |              |              |                |                       |
| Turkey          | > 250 Tons   | 12 CrMo(V,W) | < 610          | < 130,000 hrs.        |
| Yugoslavia      |              |              |                |                       |

In addition to past and current usage, several future areas of use are being considered.

| <u>System</u> | <u>Application</u>                               | <u>Alloys</u>      |
|---------------|--------------------------------------------------|--------------------|
| US-LMFBR      | Ducts (cladding)                                 | HT-9               |
|               | Steam generator                                  | MOD 9 Cr-1 Mo      |
| US-GCFR       | Ducts, cladding, steam generator, shielding      | HT-9/MOD 9 Cr-1 Mo |
| French Super  | Ducts                                            | 9-17 Cr            |
| Phenix        | Steam generator                                  | 9 Cr-1 Mo          |
| Japanese FBR  | Steam generator                                  | 9 Cr-2 Mo          |
| US-HTGR       | Steam generator & pumping, He-He heat exchangers | HT-9/MOD 9 Cr-1 Mo |
|               | He circulators                                   |                    |
| US-LWR        | Steam generator                                  | HT-9/MOD 9 Cr-1 Mo |

It is from this extensive experience that one can often obtain a great deal of mechanical property data which is summarized in the next section.

#### VI.4.4 Mechanical Properties of HT-9

The tensile strength of HT-9 is compared to that for several other candidate CTR materials in Fig. VI.4-4. This figure shows that HT-9 is superior to annealed 316 SS up to  $\sim 600^{\circ}\text{C}$  and it compares favorably to Ti-6Al-4V in the  $400\text{--}600^{\circ}\text{C}$  range. However, above  $500^{\circ}\text{C}$  the yield strength begins to drop, and it is best to limit its operating range to below  $500\text{--}550^{\circ}\text{C}$ .

The effect of long term exposure at  $600^{\circ}\text{C}$  on the tensile properties of HT-9 is shown in Fig. VI.4-5.<sup>(3)</sup> It can be seen that exposure to  $600^{\circ}\text{C}$  temperatures for 80,000 hours ( $\sim 10$  years) reveals that there is very little degradation of ultimate or 0.2% yield strength. The percentage elongation also stays between a comfortable 10 and 30% even after long term exposure. A recent study<sup>(6)</sup> also reports that HT-9 specimens subjected to 1 atm of hydrogen gas during testing at  $200\text{--}400^{\circ}\text{C}$  had no effect on ductility.

The ductility of HT-9 is compared to other CTR alloys in Fig. VI.4-6<sup>(5)</sup> and reveals that HT-9 retains a factor of 2 better ductility in the  $400^{\circ}\text{C}\text{--}600^{\circ}\text{C}$  range compared to 316 SS.

Long term stress rupture data for HT-9 and other alloys at  $500^{\circ}\text{C}$  is shown in Fig. VI.4-7. This information reveals that for up to 10 years of exposure, HT-9 can withstand stresses up to  $\sim 200$  MPa.

The effect of 80,000 hour- $600^{\circ}\text{C}$  service exposure on HT-9 stress-rupture properties is shown in Fig. VI.4-8. There is a slight degradation of the properties but it appears that stresses of 20 ksi ( $\sim 130$  MPa) or more are quite reasonable for 100,000 hour exposures.



Fig. VI.4-4a

Ultimate tensile strength versus temperature for candidate CTR alloys. (5)

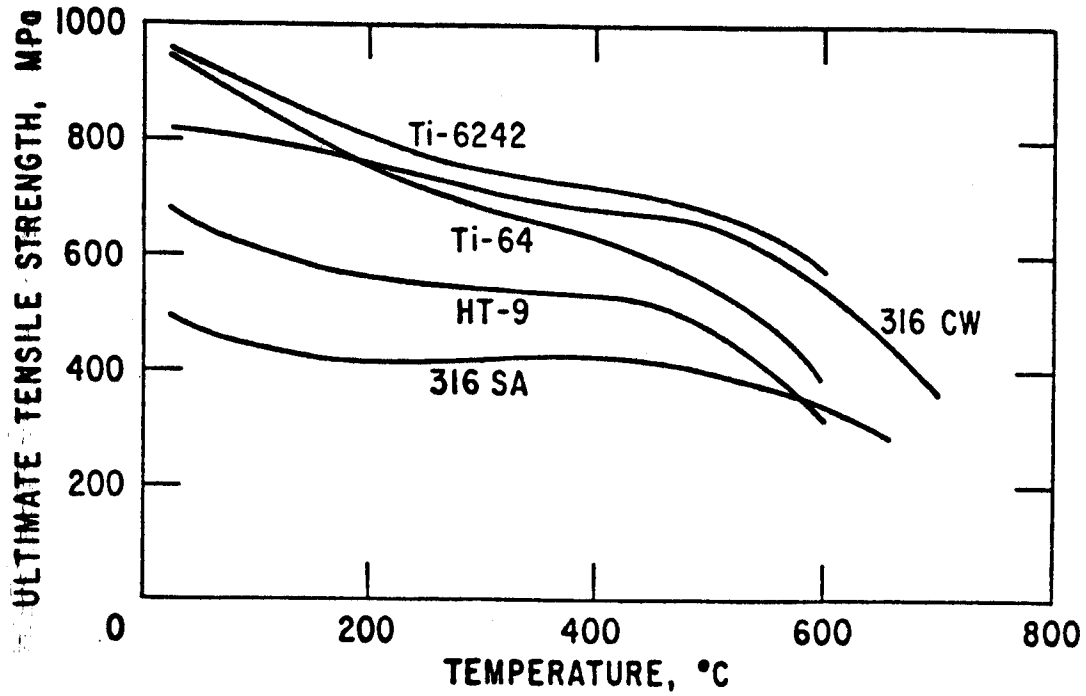


Fig. VI.4-4b

Tensile yield strength versus temperature for candidate CTR alloys. (5)

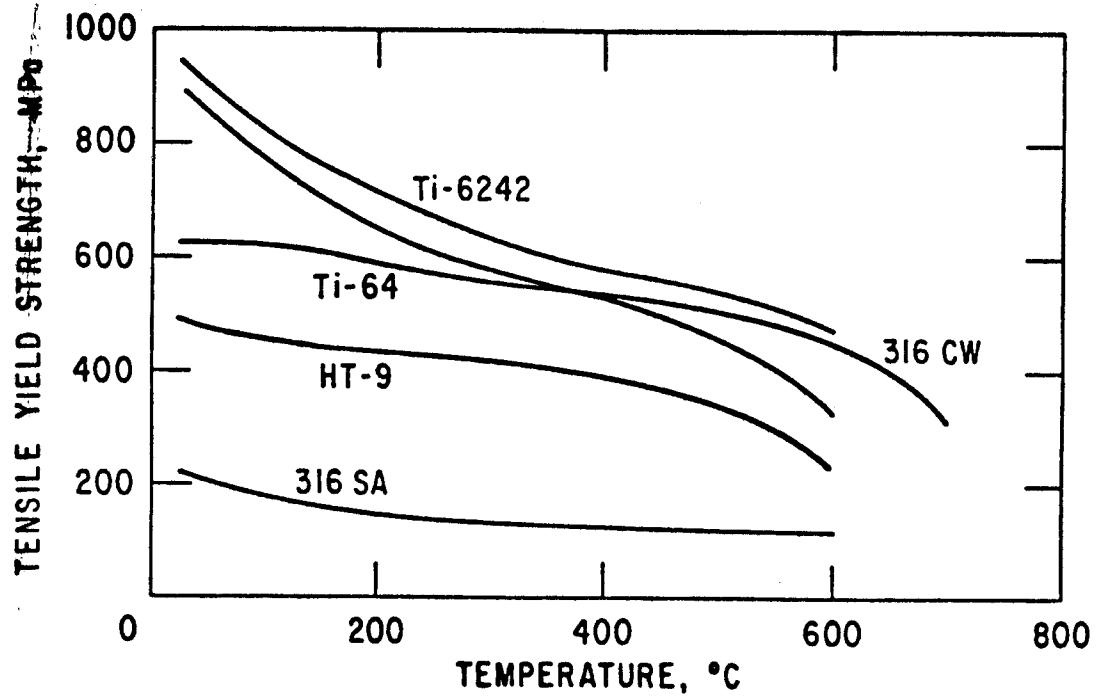


Fig. VI.4-5  
(Reference 3)

## EFFECT OF 80,000 HOUR, 600°C SERVICE EXPOSURE ON HT-9 TENSILE PROPERTIES

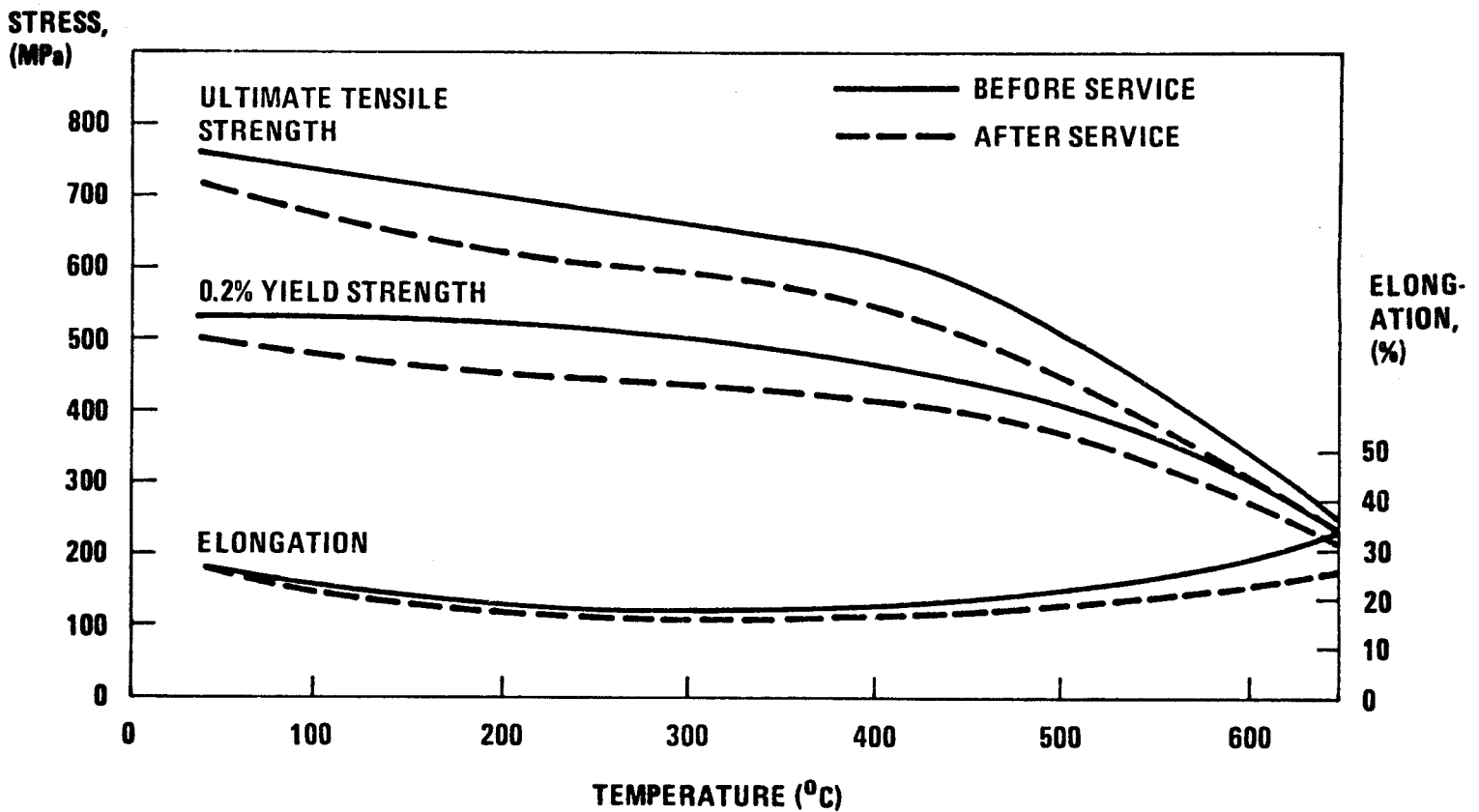
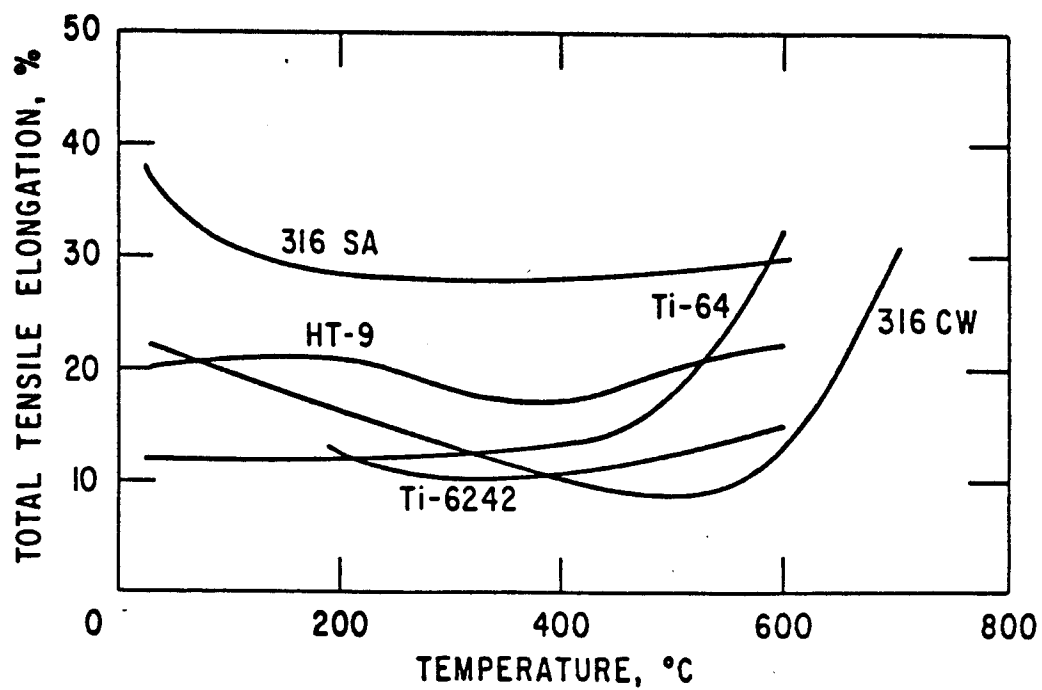
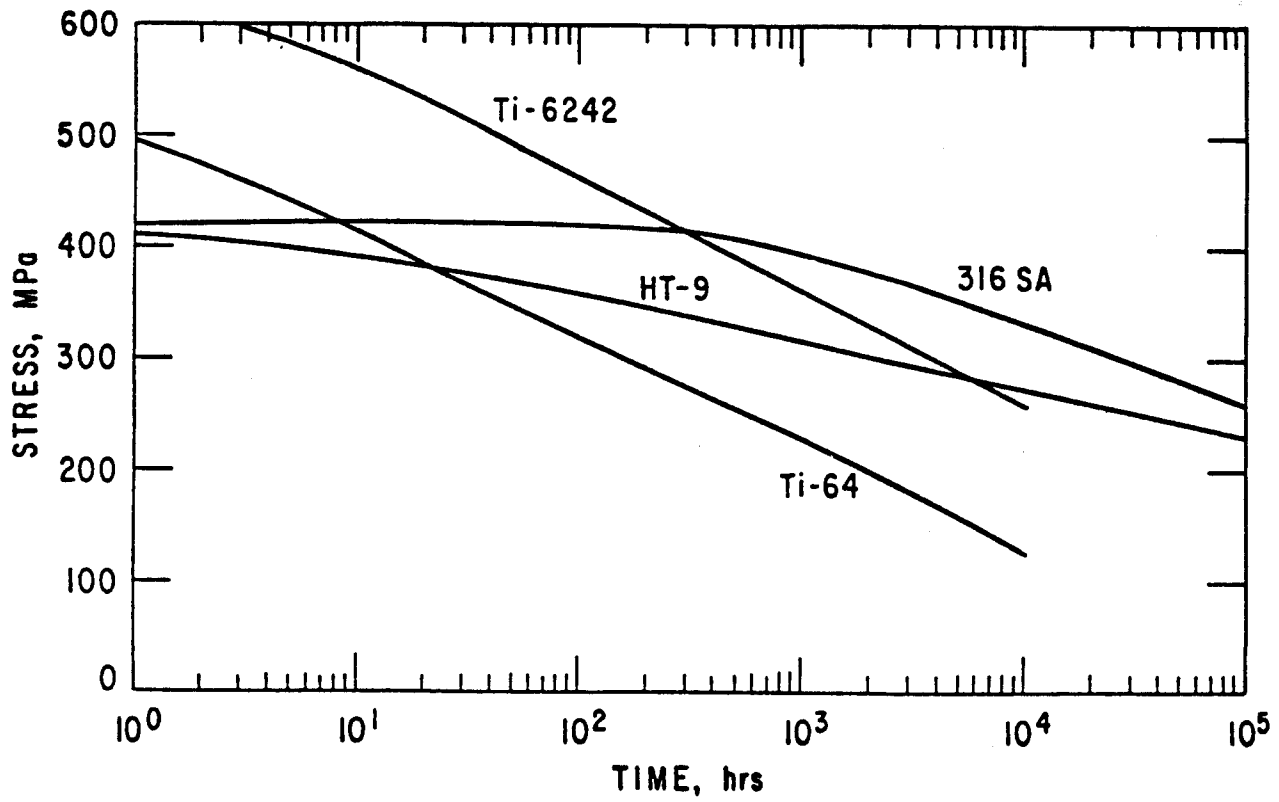


Fig. VI.4-6



Total tensile elongation versus test temperature  
for candidate CTR alloys. (22)

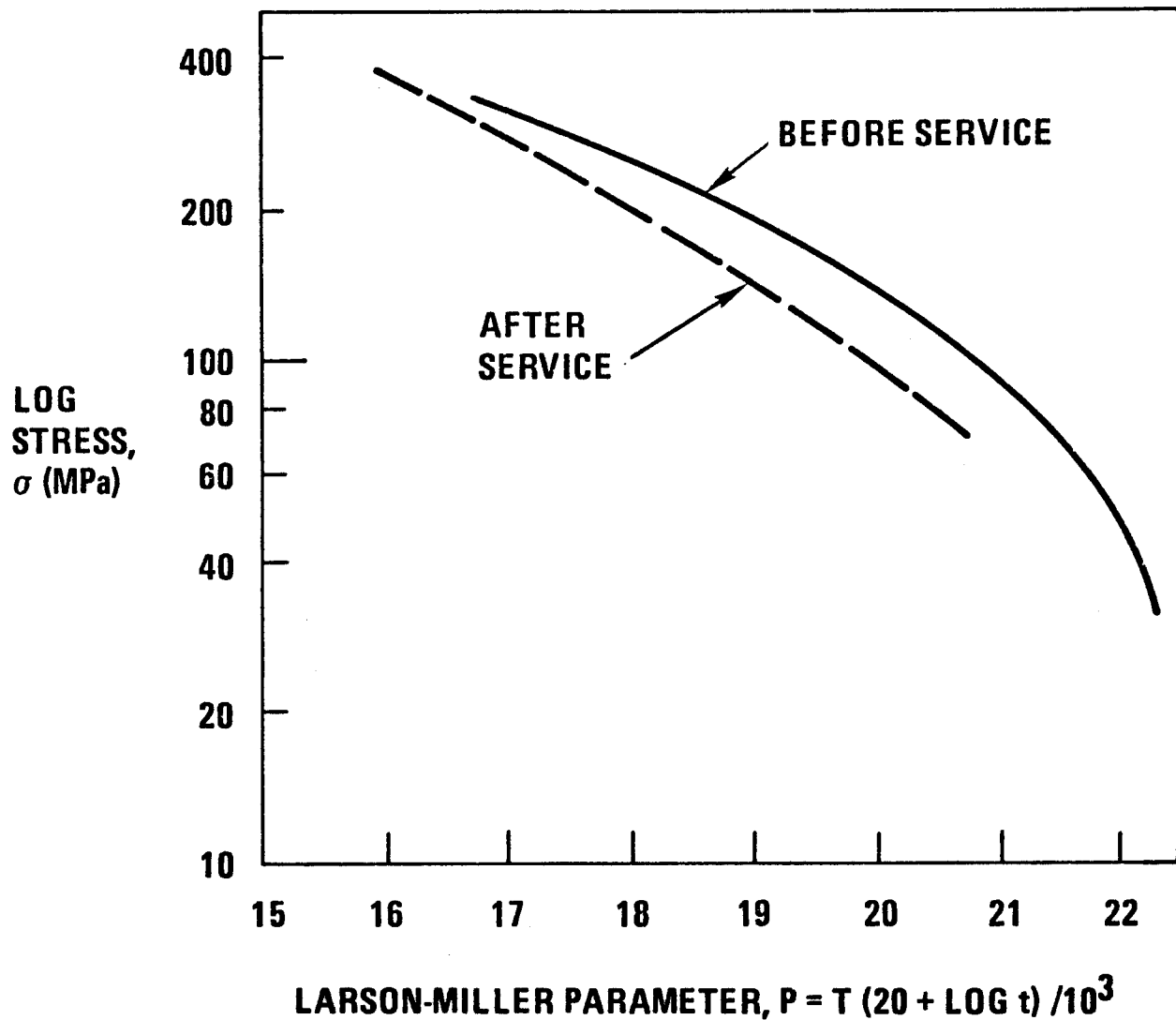
Fig. VI.4-7



Stress rupture data at 500°C for candidate CTR alloys.(22)

Fig. VI.4-8  
(Reference 4)

## EFFECT OF 80,000 HOUR, 600°C SERVICE EXPOSURE ON HT-9 STRESS-RUPTURE PROPERTIES



Even though TASKA is a steady state device, there will be numerous startups and shutdowns that will impose large stresses on the structure. Therefore, information on crack growth rates is important. Such information is given in Fig. VI.4-9 for helium environments and reveals that even under the most severe conditions the crack growth rates of unirradiated HT-9 are less than  $10^{-4}$  mm per cycle at 530°C operating temperatures. If the HT-9 is tested in air, a much more aggressive environment, the crack growth rates are increased, even at 425°C (Fig. VI.4-10). However, the growth rates are still roughly 4 times lower than found in 20% CW 316 SS or 9 Cr-1 Mo alloys.

#### VI.4.5 Thermal Stress Factor

When rigidly held plates are subjected to high heat fluxes, the resulting differential thermal expansion generates large stresses and in extreme circumstances can even cause failure. This can be a particularly difficult problem for tokamaks.

A typical figure of merit which has been used to indicate the resistance of metals to failure is the ratio of the yield stress to the thermally induced stress. If we assume equal thickness and nuclear heat generation rates, then the figure of merit, M (a high value is desired), is:

$$M = \frac{2\sigma_y K(1 - \nu)}{\alpha E}$$

where:  $\sigma_y$  is the yield strength;

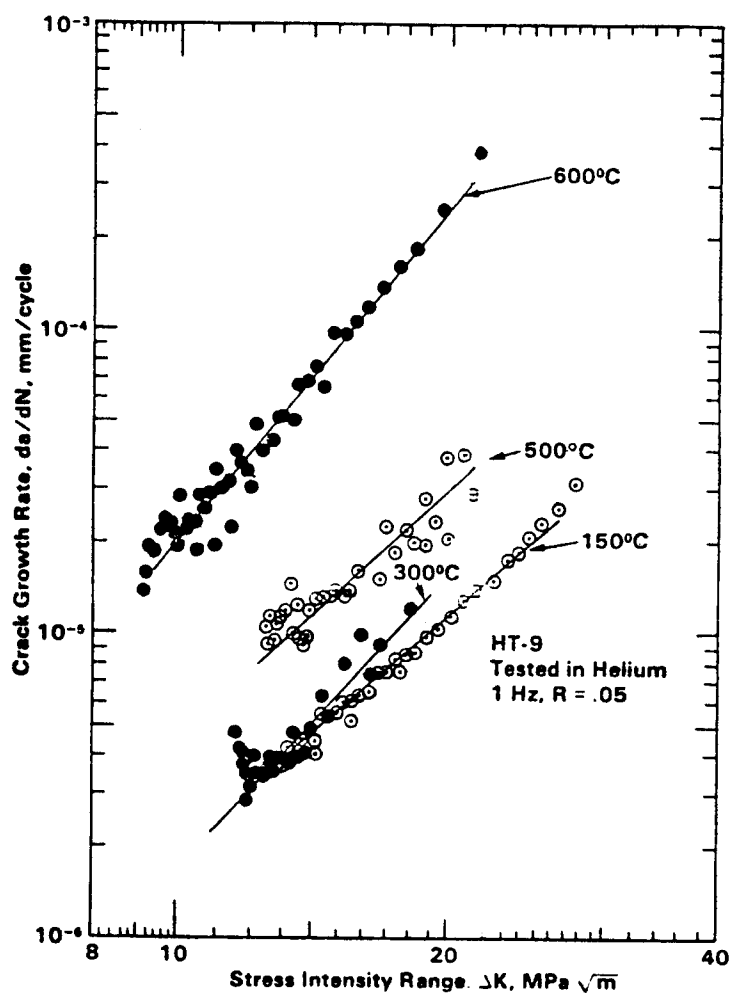
K is the thermal conductivity;

$\nu$  is Poisson's ratio;

$\alpha$  is the thermal expansion coefficient;

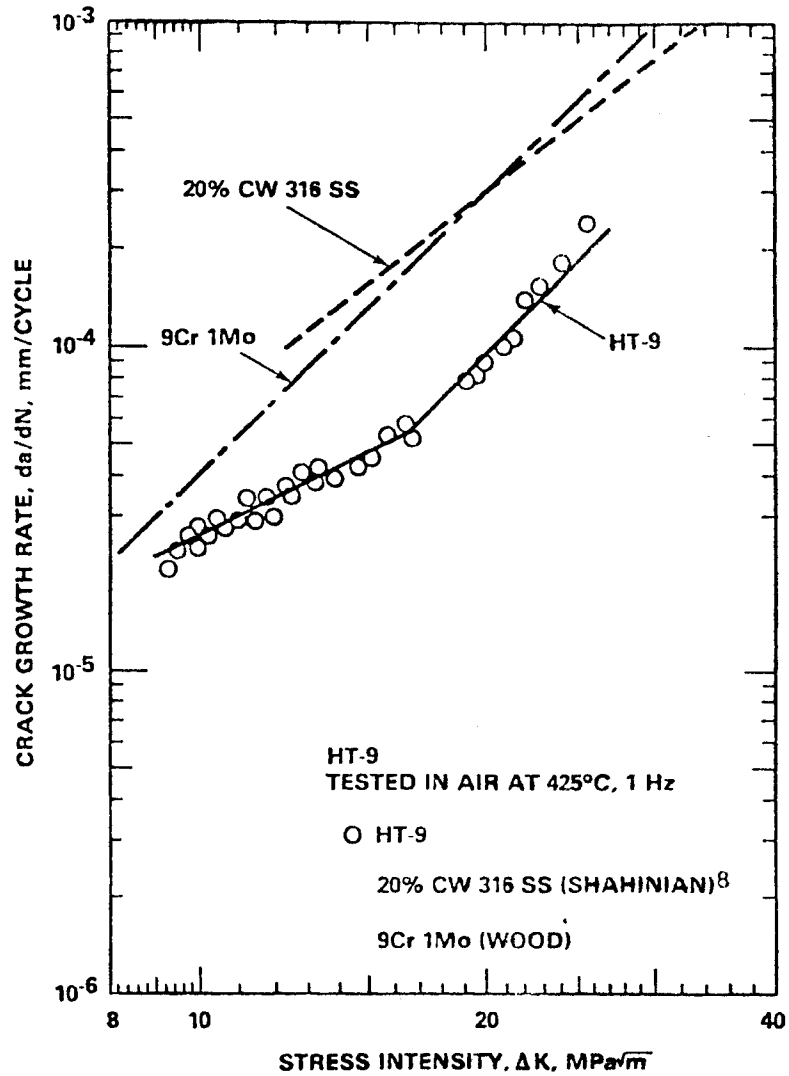
E is Young's modulus.

Fig. VI.4-9



Comparison of Crack Growth Between 150 and 600°C.

Fig. VI.4-10



HT-9 crack growth compared with 9 Cr-1 Mo and 20% CW 316 SS. Reference -- D. A. Mervyn et al., p. 132 in DOE/ER-0045/2, March 31, 1980.



Typical values of M have been calculated for 316 SS, HT-9, Ti alloys, Inconel 718, and V-20 Ti, at 500°C. These values are given in Table VI.4-3.

It can be seen that the HT-9 alloy has a figure of merit of at least twice that of cold worked 316 SS and slightly less than Ti and Mo alloys at 500°C. HT-9 is clearly inferior to refractory metals but, for lack of extensive irradiative experience we will not consider those systems in TASKA.

#### VI.4.6 Compatibility of HT-9 with Pb-Li Alloys

As might be expected, there is little data concerning the  $\text{Pb}_{83}\text{Li}_{17}$  eutectic alloy on HT-9 at temperatures of 350-550°C. There is a small amount of information on low alloy, high strength steels in pure lead,<sup>(7-9,11,14)</sup> and some data on 2-1/4 Cr-1 Mo alloys with a  $\text{Li}_{99}\text{Pb}_1$  system at 500°C.<sup>(10)</sup> The only information on liquid metals and HT-9 is a recent study at ANL in pure lithium and lithium contaminated with nitrogen at 482°C<sup>(12)</sup> and a recent ORNL investigation with  $\text{Pb}_{83}\text{Li}_{17}$ .<sup>(13)</sup>

Despite claims in earlier studies that low alloy steels were not embrittled by pure lead, recent results showed embrittlement could occur near the melting point of the lead (327°C).<sup>(11,14)</sup> The embrittlement of the high strength 4100 series steels (see composition in Table VI.4-4) started at ~ 200°C and reached a maximum (lowest ductility) at ~ 320°C. Above ~ 370°C there was degradation in the ductility (see Fig. VI.4-11). The addition of Sn increased the width of the embrittling zone.

At higher temperatures, 700°C, 2-1/4 Cr-1 Mo alloy suffered a severe attack after 300 hours of exposure to lead.<sup>(9)</sup> However, the addition of Ti to the Pb completely eliminated the corrosion under the same conditions. It has been known for a long time that the addition of Ti or Zr to Pb would inhibit corrosion by forming tightly adhering TiC and ZrC films on the steel surface.

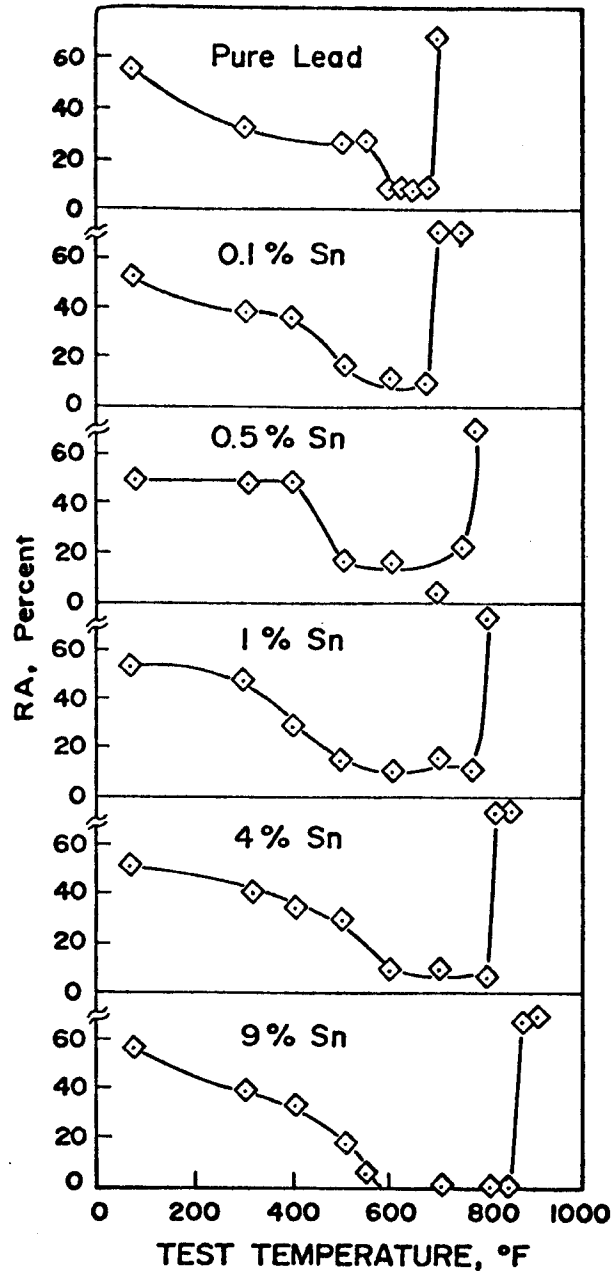
Table VI.4-3. Thermal Stress Resistance of  
Potential Fusion Materials at 500°C

| <u>Alloy</u>       | Thermal Stress Resistance      |
|--------------------|--------------------------------|
|                    | <u>(W/m) x 10<sup>-3</sup></u> |
| Annealed 316 SS    | 1.3                            |
| 20% CW 316 SS      | 4.9                            |
| HT-9               | 8.5                            |
| Ti-6Al-4V          | 10.2                           |
| Ti-6Al-2Sn-4Zr-2Mo | 10.4                           |
| Inconel-718        | 10.7                           |
| V-20 Ti            | 12.2                           |

Table VI.4-4. Comparison of Some Alloy Steels Which Have Been Tested  
in Pb, Li or Pb-Li Alloys

| <u>Alloy</u> | Element - wt% (Bal. Fe) |           |           |          |           |
|--------------|-------------------------|-----------|-----------|----------|-----------|
|              | <u>Cr</u>               | <u>Mn</u> | <u>Mo</u> | <u>C</u> | <u>Si</u> |
| 4140         | 0.96                    | 0.87      | 0.18      | 0.41     | 0.23      |
| 4145         | 1.00                    | 0.88      | ---       | 0.47     | 0.33      |
| Croloy       | 2.25                    | 0.5       | 1.0       | 0.12     | ---       |
| HT-9         | 11.5                    | 0.5       | 1.0       | 0.2      | 0.25      |

Fig. VI.4-11



Reduction of area, RA, for 200-ksi (1379-MPa) 4145 steel, surface wetted with various Pb-Sn alloys as a function of test temperature. (30)

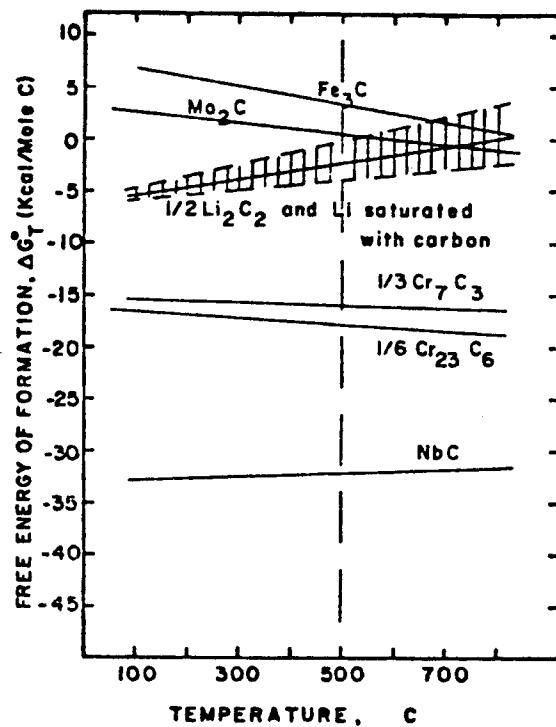
On the other end of the spectrum, a  $\text{Li}_{99}\text{Pb}_1$  alloy was shown to attack  $\text{Fe}_3\text{C}$  and  $\text{Mo}_2\text{C}$  in the welded zones of a 2-1/4 Cr-1 Mo alloy at 500°C after exposure up to 1600 hours. Figure VI.4-12, from the paper by Anderson et al.<sup>(10)</sup>, shows that the cementite and  $\text{Mo}_2\text{C}$  are unstable with respect to Li below 500°C while the chromium carbides are stable over a wide range.

Work at Harwell<sup>(15)</sup> has shown that low alloy steels can be significantly decarburized by exposure to Li even at ~ 200°C. However, higher Cr containing steels such as 2-1/2 Cr-1 Mo, or the 9 Cr-1 Mo alloys showed much greater resistance.<sup>(16)</sup> Presumably, the higher Cr content of HT-9 would help to protect that alloy from decarburization and hence embrittlement.

A recent study by Chopra and Smith<sup>(12)</sup> has shown that nitrogen can have a large effect on the fatigue life of HT-9 in flowing Li at 482°C. They found that in short term fatigue tests, where the nitrogen level was between 80 and 130 ppm, the fatigue life in Li was the same as that obtained in Na. However, when the nitrogen level reached > 1000 ppm of nitrogen, there was a dramatic drop in the fatigue life (Fig. VI.4-13) by as much as a factor of 5. This strongly suggests that close control of the nitrogen level in the TASKA coolant will be necessary.

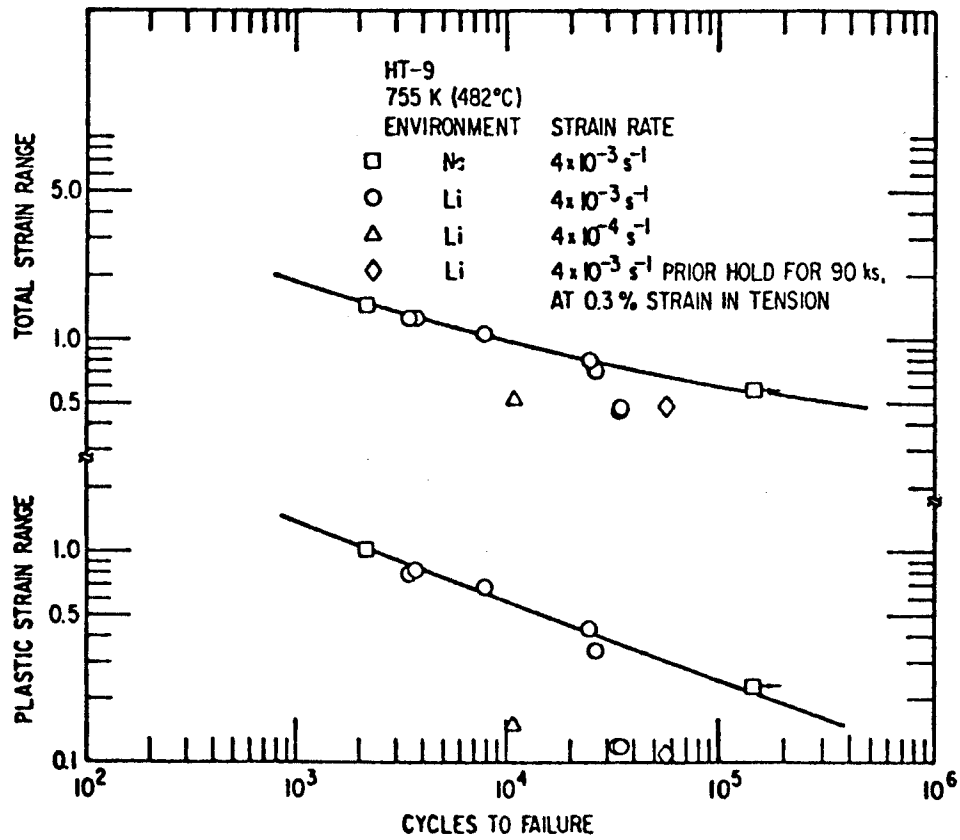
Tortorelli and DeVan have most recently studied the compatibility of type 316 stainless steel and HT-9 with static Pb-17 at.% Li in the temperature range from 300 to 500°C. The effects of corrosion were characterized by changes in specimen weight and tensile properties after exposure. Figure VI.4-14 shows the weight change as a function of exposure at a temperature of 500°C. Less weight change was obtained at lower temperatures, as seen from the results in Table VI.4-5.

Fig. VI.4-12



Relative thermodynamic stability of several carbides in carbon saturated lithium.

Fig. VI.4-13



Total and plastic strain range vs. cycles to failure for HT-9 alloy tested in lithium at 755 K.

Fig. VI.4-14  
Weight loss in liquid Li and  $\text{Li}_{17}\text{Pb}_{83}$ .

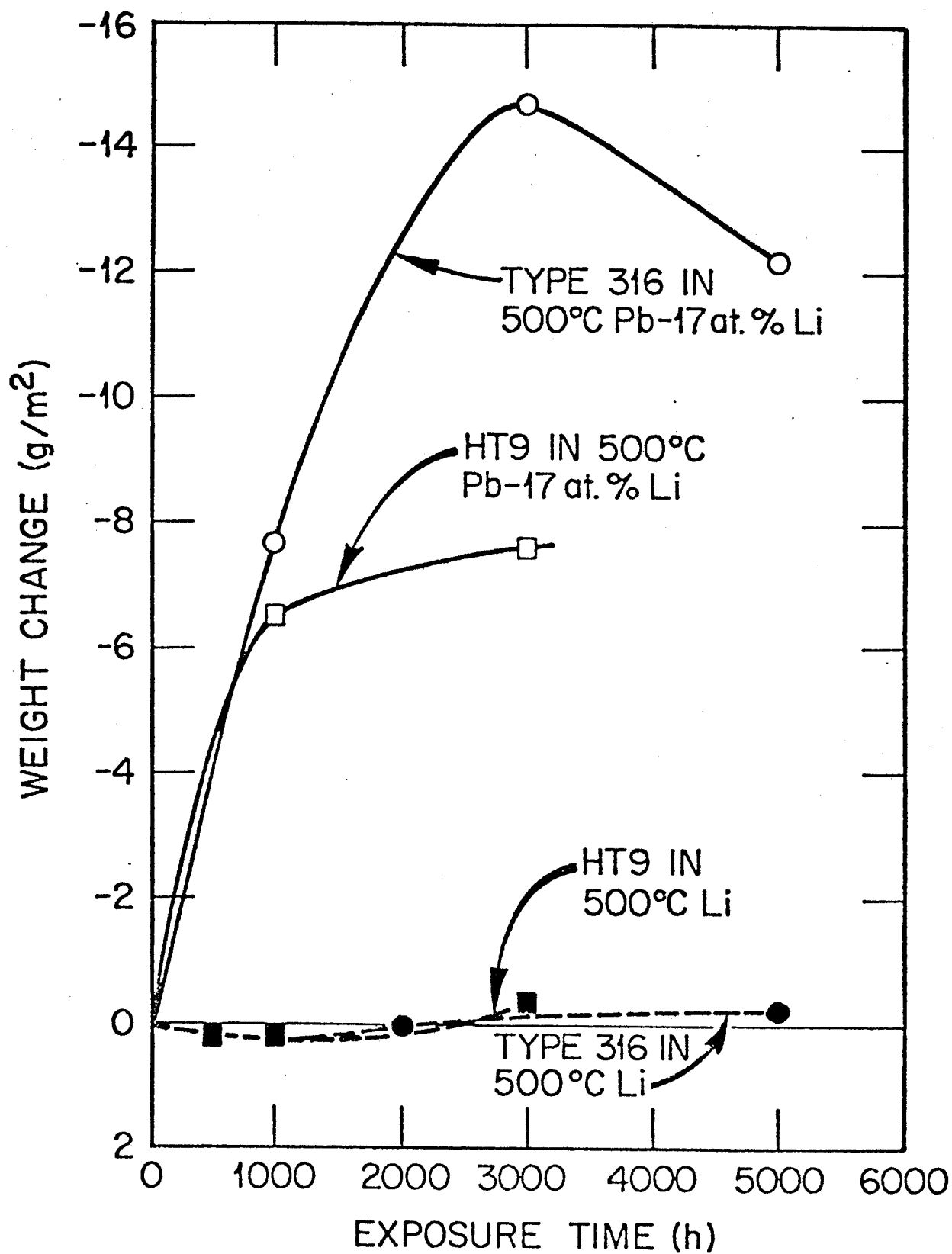


Table VI.4-5. Weight Changes of Type 316 Stainless Steel and Sandvik HT-9<sup>(a)</sup>  
Exposed to Static Argon and Pb-17 at.% Li

| Exposure Temperature<br>(°C) | Exposure Time<br>(h) | Weight Change <sup>(b)</sup> (g/m <sup>2</sup> ) |                   |                  |               |
|------------------------------|----------------------|--------------------------------------------------|-------------------|------------------|---------------|
|                              |                      | Type 316<br>in Pb-Li                             | Type 316<br>in Ar | HT-9 in<br>Pb-Li | HT-9 in<br>Ar |
| 300                          | 1000                 | -0.4                                             | -0.1              | -2.3             |               |
|                              | 3000                 | -0.4                                             | 0.0               | -6.6             |               |
|                              | 5000                 | -1.0                                             | 0.0               | -6.6             |               |
| 400                          | 1000                 | -0.9                                             | -0.1              | -3.2             |               |
|                              | 3000                 | -1.2                                             | 0.0               | -5.6             |               |
|                              | 5000                 | -3.2                                             | 0.0               | -5.6             |               |
| 500                          | 1000                 | -7.6                                             | +0.1              | -6.5             | +0.2          |
|                              | 3000                 | -14.7                                            | 0.0               | -7.6             | 0.0           |
|                              | 5000                 | -12.2                                            | +0.1              |                  |               |
| 900                          | 20                   | -11.9 <sup>(c)</sup>                             | +0.1              |                  |               |

(a) Normalized and tempered.

(b) Average of measurements from two specimens unless otherwise noted.

(c) Average of measurements from six specimens.

Although both type 316 stainless steel and HT-9 show significant weight losses when exposed to static Pb-17 at.% Li at 500°C, no significant changes in room temperature tensile properties could be detected, as demonstrated by the results in Table VI.4-6.

In summary, it appears that Pb<sub>83</sub>Li<sub>17</sub> can have a corrosive effect on HT-9 at temperatures of 500°C and presumably higher temperatures. The application of molten Pb<sub>83</sub>Li<sub>17</sub> as a breeding and coolant material together with ferrous alloys may therefore be limited to temperatures below 500°C.



Table VI.4-6. Tensile Results<sup>(a)</sup> for Type 316 Stainless Steel  
Exposed to Static Argon and Pb-17 at.% Li

| Exposure<br>Temperature<br>(°C) | Exposure<br>Time<br>(h) | Strength (MPa)       |                    |                  |                    | Elongation <sup>(c)</sup><br>(%) |    |
|---------------------------------|-------------------------|----------------------|--------------------|------------------|--------------------|----------------------------------|----|
|                                 |                         | Yield <sup>(b)</sup> |                    | Ultimate Tensile |                    |                                  |    |
|                                 |                         | Pb-Li                | Ar                 | Pb-Li            | Ar                 | Pb-Li                            | Ar |
| 300                             | 1000                    | 242                  | 229                | 577              | 574                | 84                               | 86 |
|                                 | 3000                    |                      | 248                |                  | 583                | 85                               | 88 |
|                                 | 5000                    | 238                  | 246                | 582              | 584                | 88                               | 86 |
| 400                             | 1000                    | 235                  | 237                | 574              | 568                | 88                               | 88 |
|                                 | 3000                    | 242                  | 243                | 584              | 582                | 85                               | 88 |
|                                 | 5000                    | 245                  | 245                | 589              | 580                | 85                               | 84 |
| 500                             | 1000                    | 240                  | 240 <sup>(d)</sup> | 570              | 576 <sup>(d)</sup> | 88                               | 87 |
|                                 | 3000                    | 250                  | 251                | 575              | 592                | 84                               | 89 |
|                                 | 5000                    | 246                  | 253                | 570              | 594                | 89                               | 87 |
| 900                             | 20                      | 248 <sup>(e)</sup>   | 215                | 593              | 518 <sup>(e)</sup> | 86                               | 89 |

(a) Average of measurements from two specimens unless otherwise noted (scatter within  $\pm 10\%$ ).

(b) 0.2% offset.

(c) In 12.7 mm.

(d) One specimen measurement.

(e) Average of measurements from six specimens.

#### VI.4.7 Magnetic Properties of HT-9

Figure VI.4-15 shows the hysteresis loops for 9 and 12% Cr steels. HT-9 exhibits a similar hysteresis loop as shown in Fig. VI.4-16.<sup>(4)</sup> It is found that the area of the hysteresis loop is reduced at the operating temperature goes up and the value of  $B_{sat}$  is reduced from  $\sim 1.4$  T at 25° to 1.1 T at

Fig. VI.4-15  
(Reference 4)

## COMPARISON OF MAGNETIC PROPERTIES OF TWO CANDIDATE MAGNETIC MATERIALS

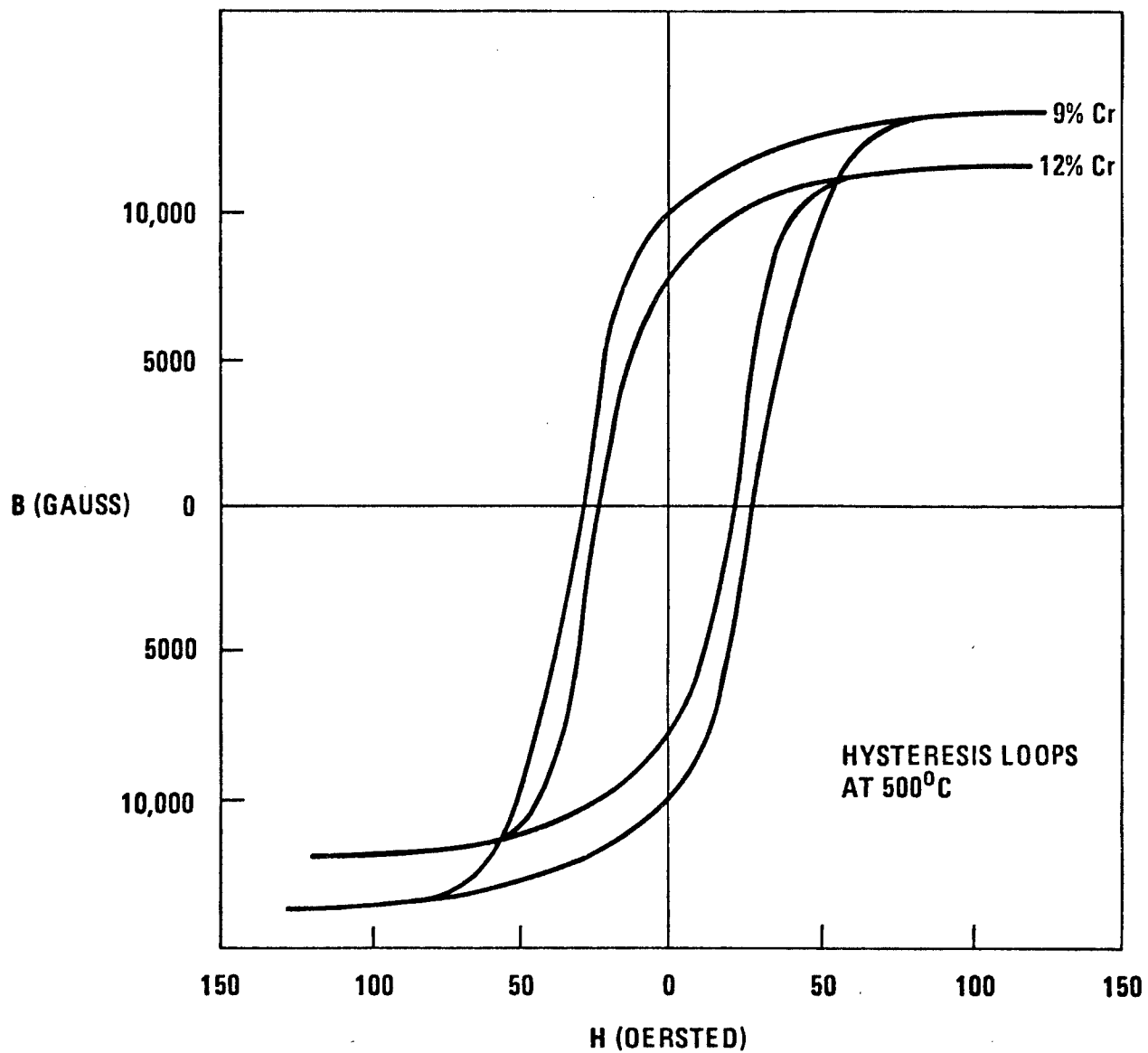
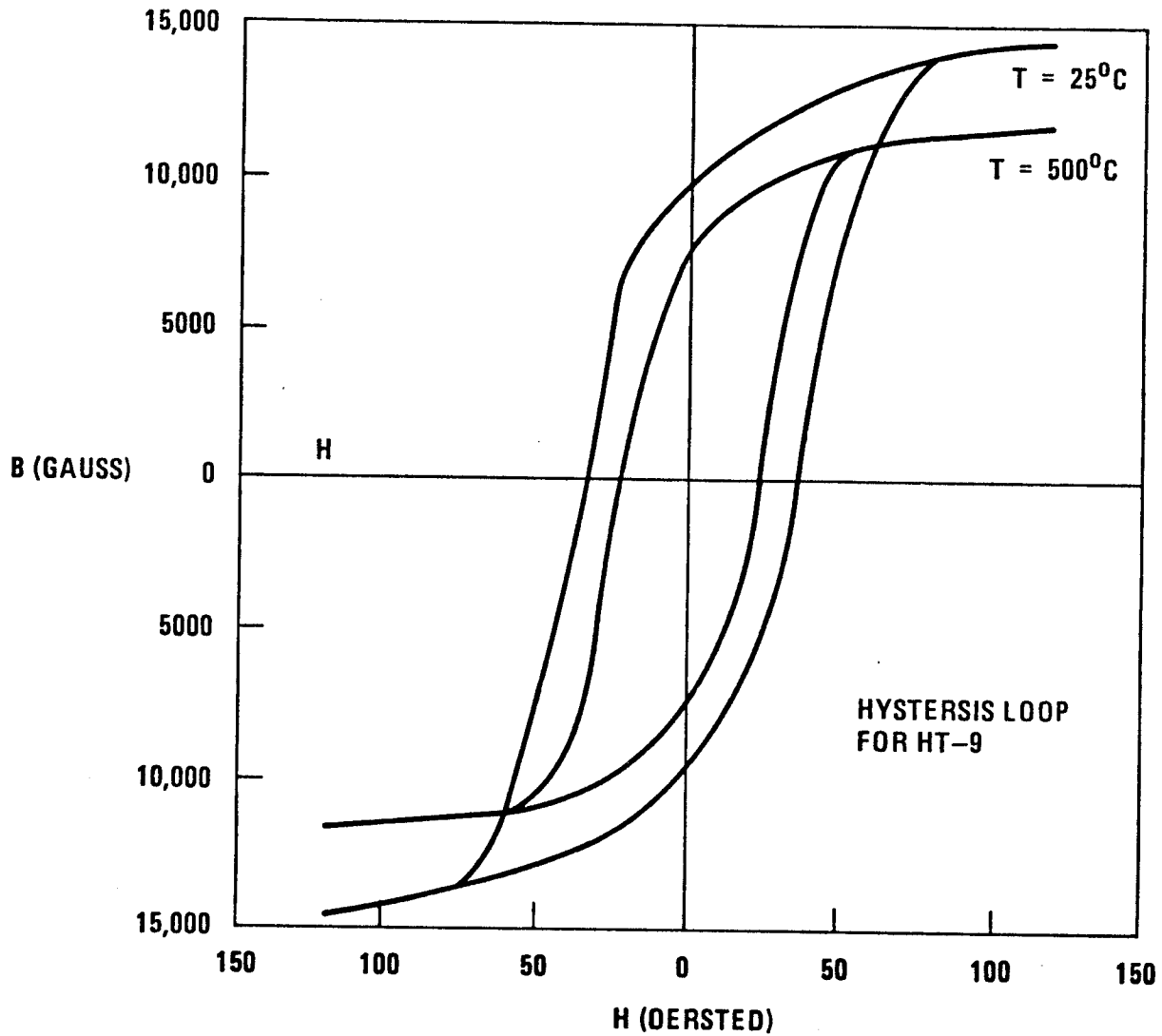


Fig. VI.4-16

## MAGNETIC MATERIALS IN MAGNETIC FUSION DEVICES ARE IN SATURATION



500°C. Since the magnetic field at the blanket from the central cell coils is roughly 3 tesla, it is clear that the blanket material will be saturated during the operation of TASKA. Another way of stating the above is that the permeability as defined,

$$B = \mu H$$

is very close to one under TASKA conditions.

Since the saturation magnetization is a substantial fraction of the magnetic field produced by the central coil, in contrast to tokamaks where it is a small fraction, it is expected that the ferromagnetic blanket structure will perturb and reduce the central cell magnetic field. This is discussed further in Section V.9.

#### VI.4.8 Effects of Irradiation on HT-9 Type Alloys

##### VI.4.8.1 Introduction

It was previously pointed out there is a considerable data base for low temperature, relatively low fluence, irradiation of ferritic steels. More recently, because of interest in the applications of ferritic steels in the core structures of LMFBR's, some information about the high temperature behavior of ferritic steels irradiated to moderately high fission neutron fluences has begun to appear.<sup>(17)</sup> Unfortunately, the data is not as extensive as that for the austenitic system but a great deal of information should be available in the next year or two. It must also be pointed out that there is no data on HT-9 (or any other ferritic steel) irradiated with D-T neutrons. Therefore, all of the results quoted here from fission neutron, heavy ion, and electron irradiation will have to be extrapolated to typical TASKA conditions.

#### VI.4.8.2. Survey of Irradiated Ferritic Steel Information Pertinent to TASKA

There are four sources of data which may be used for our study.

- 1) Fast neutron data from EBR-II.
- 2) Thermal neutron data from test reactors and commercial power plants.
- 3) Heavy ion data.
- 4) High voltage electron microscopy data.

It is not the purpose of this section to present a complete literature survey, but only to highlight those papers which might have a direct bearing on the performance of HT-9 in TASKA. A summary of the work of interest to this study is given in Table VI.4-7.

There are four areas in which measurements have been made on irradiated ferritic steel:

- A) swelling due to voids;
- B) tensile properties;
- C) in-reactor creep;
- D) shift in ductile to brittle transition temperature (DBTT).

#### VI.4.8.3 Void Swelling in Ferritic Steel

The void swelling problem, which has plagued the austenitic steels in the LMFBR program and has presented some serious problems for the design of first walls in fusion reactors, seems to be greatly reduced in the ferritic steel system. Smidt et al.<sup>(26)</sup> showed in 1976 that the ferritic steels (including HT-9) were very resistant to the production of voids during heavy ion bombardment. In fact, he found that the maximum swelling rate was  $\sim 0.01$ - $0.02\%$  per dpa compared to values at least 10 times higher for the 316 stainless steel systems. One MeV electron irradiation experiments also revealed that ferritic

Table VI.4-7. Summary of Pertinent Irradiation Data for HT-9 in WITAMIR-I

| <u>Type of<br/>Irradiation<br/>(Reference)</u> | <u>Material</u>                                              | <u>T<sub>irr</sub><br/>°C</u> | <u>Fluence<br/>x 10<sup>22</sup> n/cm<sup>2</sup></u> | <u>Type of Test</u>             | <u>Comments</u>             |
|------------------------------------------------|--------------------------------------------------------------|-------------------------------|-------------------------------------------------------|---------------------------------|-----------------------------|
| fast n <sup>(17)</sup>                         | HT-9                                                         | 419                           | 1.1                                                   | DBTT, $\sigma_y$                | 100°C shift                 |
| fast n <sup>(18)</sup>                         | 405 SS                                                       | 400-405                       | 2.3                                                   | $\Delta V$ , $\sigma_y$ , duct. | ---                         |
| fast n <sup>(19)</sup>                         | HT-9                                                         | 540-565                       | 4.0                                                   | $\epsilon_{irr}$                | ---                         |
| fast n <sup>(20)</sup>                         | FV-448                                                       | 380, 420,<br>460, 615         | (30 dpa)                                              | $\Delta V$                      | ---                         |
| fast n <sup>(21)</sup>                         | HT-9                                                         | 605<br>625-635                | 2<br>4                                                | $\epsilon_{irr}$                | ---                         |
| fast n <sup>(22)</sup>                         | 2-1/4 Cr<br>1 Mo                                             | 371-427                       | 0.29                                                  | FCP, DBTT                       | No effect of<br>irradiation |
| fast n <sup>(23)</sup>                         | 15% Cr-Fe<br>FV-448                                          | 380-615                       | 30                                                    | $\Delta V$                      | No $\Delta V$ change        |
| fast n <sup>(24)</sup>                         | HT-9                                                         | 440-600                       | 1.1                                                   |                                 | ---                         |
| fast n <sup>(25)</sup>                         | HT-9                                                         | 550                           | 7.5                                                   | duct., $\Delta V$               | ---                         |
| Heavy Ion <sup>(26)</sup>                      | HT-9, EM-12                                                  | 450-650                       | (250 dpa)                                             | $\Delta V$                      | No He                       |
| 1 MeV Elec. <sup>(27)</sup>                    | S4                                                           | 395-550                       | (20 dpa)                                              | $\Delta V$                      | 10 ppm He                   |
| LWR n <sup>(25)</sup>                          | HT-9, 9 Cr 1 Mo,<br>2-1/4 Cr                                 | 50                            | 1.3 (9.3 dpa)                                         | $\sigma_y$ , duct.              | 10-80 ppm He                |
| fast n <sup>(29)</sup>                         | 430 F, 416,<br>EM-12, H11                                    | 400-650                       | 17.6                                                  | $\Delta V$                      | $\Delta V < 0.57\%$         |
| fast n <sup>(30)</sup>                         | 2-1/4 Cr-1 Mo<br>430 F, 416,<br>EM-12, H11,<br>2-1/4 Cr-1 Mo | 425                           | 5.05                                                  | TEM                             |                             |

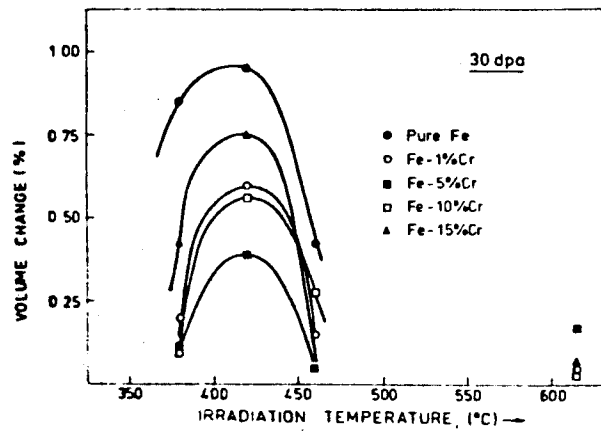
steels were quite resistant to the formation of voids, even after predoping the specimens with 10 appm of He.<sup>(32)</sup>

Neutron irradiation of Fe-10 Cr and FV-448 (11% Cr, 1 Mn, 0.6 Mo, 0.65 Ni) alloys showed very little swelling up to 30 dpa in the temperature range of 375°C-460°C.<sup>(23)</sup> Figures VI.4-17 and VI.4-18 show that the addition of up to 5% Cr reduces the swelling compared to pure iron. Further additions of Cr have less effect but still reduce the swelling to ~ 0.02% per dpa level. Other fission neutron irradiations of FV-448 to ~ 30 dpa in the 380°C-615°C temperature range show that < 0.1% swelling was observed. Similarly irradiation of 405 SS (15 Cr, 0.3 Ni, 0.6 C, 0.15 Al) to 400°C with  $2.3 \times 10^{22}$  n/cm<sup>2</sup> (E > 0.1 MeV) revealed no swelling at all.<sup>(18)</sup> Recent results from EBR-II irradiations<sup>(30)</sup> on various ferritic steels whose composition and structure is given in Table VI.4-8 has further confirmed the expectation that the swelling resistance of the ferritic class of steels is generic rather than dependent on one particular composition or heat treatment. As shown in Fig. VI.4-19, swelling is less than 1% for displacement doses between 70 and 90 dpa.

Table VI.4-8. Comparison of the Nominal Major Element Compositions  
and Microstructures of the Ferritic Alloys Examined

| <u>Alloy</u>  | <u>Cr</u> | <u>Mo</u> | <u>C</u> | <u>Structure</u>            |
|---------------|-----------|-----------|----------|-----------------------------|
| AISI 430F     | 18        | 0.04      | 0.05     | Duplex - Ferrite/Martensite |
| AISI 416      | 13        | 0.2       | 0.1      | Ferritic                    |
| EM-12         | 10        | 2.0       | 0.06     | Duplex - Ferrite/Martensite |
| H-11          | 5         | 1.4       | 0.4      | Martensitic                 |
| 2-1/4 Cr-1 Mo | 2.2       | 1.0       | 0.1      | Martensitic                 |

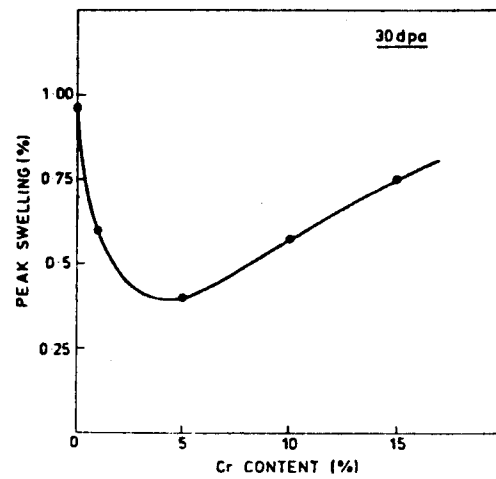
Fig. VI.4-17



Swelling as a function of temperature for binary iron-chromium alloys compared with pure iron.



Fig. VI.4-18

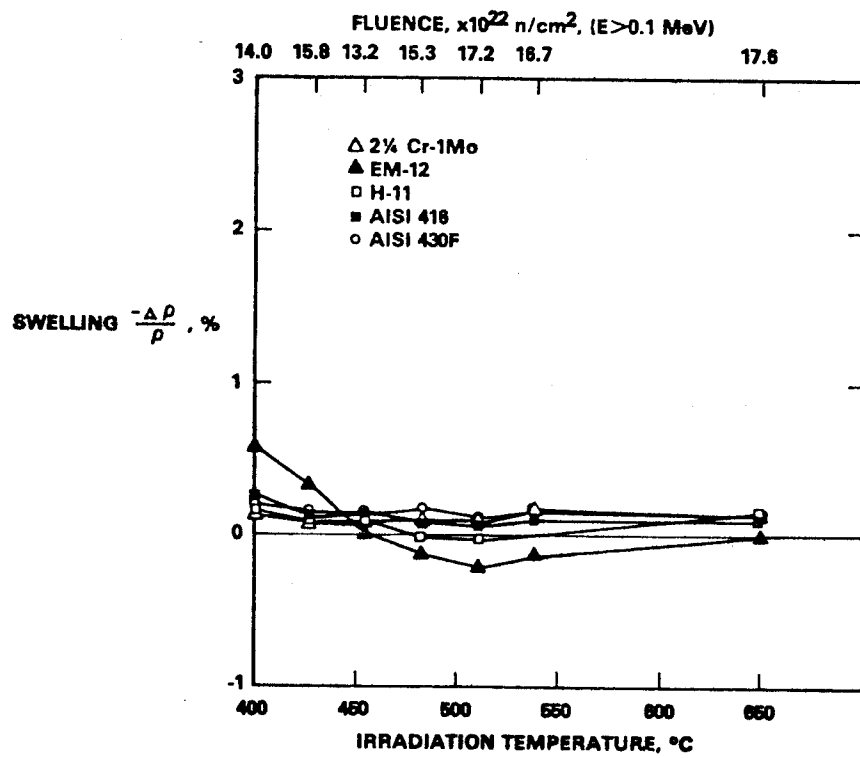


Magnitude of peak swelling as a function of chromium content for binary iron-chromium alloys.

Fig. VI.4-19

Swelling in the ferritic alloys.

**SWELLING IN COMMERCIAL ALLOYS:  
FERRITIC ALLOYS (PATH E)**



Irradiation of HT-9 to  $1.1 \times 10^{23}$  n/cm<sup>2</sup> (48 dpa) over the range of temperature from 400°C-600°C reveals that the swelling rate is less than 0.01% per dpa compared to a value of 0.35% per dpa for 316 SS (Fig. VI.4-20).<sup>(25)</sup> Such studies do not simulate the helium production in HT-9 correctly, but HT-9 doped with 1% Ni is now being inserted into thermal fission reactors to obtain the necessary higher helium concentrations. However, only tensile properties on doped HT-9 have so far been measured.

#### VI.4.8.4 Effects of Neutron Irradiation on the Tensile, Creep, and Stress-Rupture Properties of Ferritic Steel

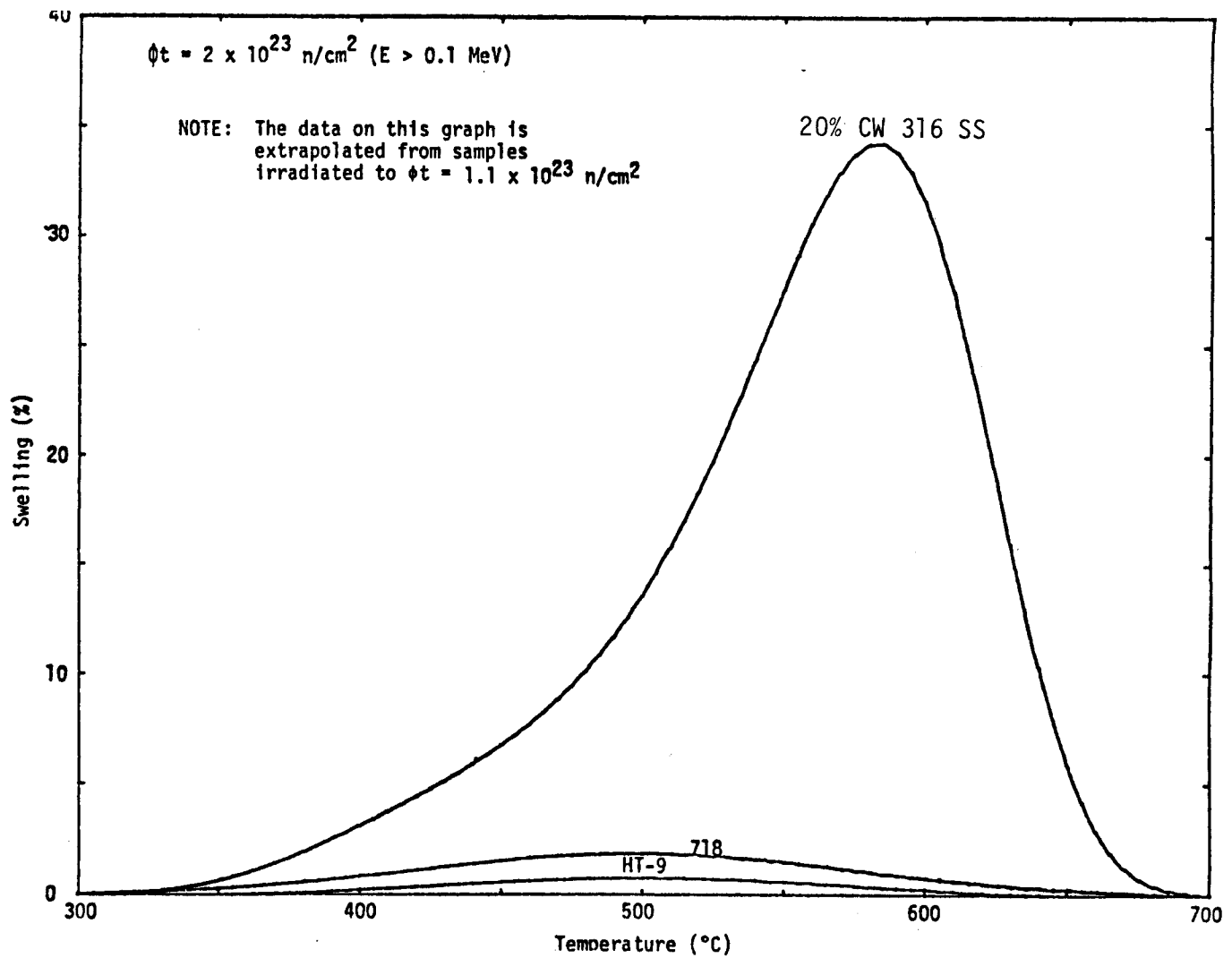
There are three results reported in the literature that are pertinent to the use of HT-9 in TASKA. An early study by Garr et al.<sup>(18)</sup> showed that 405 SS, which was given a 1 hour anneal at 980°C before irradiation, suffered relatively little loss in ductility after being irradiated to  $2.3 \times 10^{22}$  n/cm<sup>2</sup> (10 dpa) at 400°C-425°C (see Table VI.4-9). The total elongation remained at 11.2% at 500°C while the uniform elongation dropped from 9.4% to 4.0%.

In another study<sup>(25)</sup> where HT-9 was irradiated to  $1.1 \times 10^{23}$  n/cm<sup>2</sup> (48 dpa) at 550°C the total elongation remained above 10% when tested at 550°C and 670°C.

Smidt et al.<sup>(17)</sup> found that irradiation to ~ 5 dpa at 419°C resulted in very little reduction of ductility during post-irradiation room temperature tests. They found that after such an irradiation, there was still 6.3% uniform elongation and 13.9% total elongation remaining.

More recently, Klueh and Vitek<sup>(30)</sup> irradiated tensile specimens of various ferritic steels with the compositions as given in Table VI.4-10. The HFIR irradiations were carried out at 50°C to doses between 6.5 and 9.5 dpa and to helium concentrations ranging between 10 to 80 appm He. The low helium

Fig. VI.4-20



Swelling of candidate CTR materials during neutron irradiation<sup>(25)</sup>.

Table VI.4-9. Pertinent Data of Irradiation Effects on  
Tensile Properties of Ferritic Steel

| <u>Material</u> | <u>T<sub>irr</sub>/Test</u> | <u>dpa<sup>(a)</sup></u> | <u>Appm<br/>He<sup>(b)</sup></u> | <u>Yield<br/>Stress-MPa</u> | <u>Tensile<br/>Stress-MPa</u> | <u>Uniform<br/>Strain-%</u> | <u>Total<br/>Strain-%</u> |
|-----------------|-----------------------------|--------------------------|----------------------------------|-----------------------------|-------------------------------|-----------------------------|---------------------------|
| 405 SS          | ---/500                     | Control                  | ---                              | 346                         | 712                           | 9.4                         | 11.2                      |
| (18)            | 425/500                     | 10                       | 3.9                              | 461                         | 516                           | 4.0                         | 11.2                      |
| HT-9            | 550/550                     | 48                       | 19                               | N.R.                        | N.R.                          | N.R.                        | 14.                       |
| (25)            | 550/670                     | 48                       | 19                               | N.R.                        | N.R.                          | N.R.                        | > 10                      |
| HT-9            | ---/253                     | Control                  | ---                              | 653                         | 823                           | 7.5                         | 16.4                      |
| (17)            | 419/25                      | 5                        | 2                                | 971                         | 1037                          | 6.3                         | 13.9                      |

N.R. = Not Reported

(a) use 4.4 dpa per  $10^{22}$  n/cm<sup>2</sup> (E > 0.1 Mev)

(b) use 0.39 appm He/dpa

Table VI.4-10. Composition of Ferritic Steels Irradiated in HFIR at 50°C.

| Element | Alloy Content, <sup>a</sup> wt % |                |                        |                        |                                 |
|---------|----------------------------------|----------------|------------------------|------------------------|---------------------------------|
|         | HT9 Breeder (91354)              | HT9 (XAA-3587) | HT9 + 1% Ni (XAA-3588) | HT9 + 2% Ni (XAA-3589) | HT9 + 2% Ni Adjusted (XAA-3592) |
| C       | 0.20                             | 0.21           | 0.20                   | 0.20                   | 0.15                            |
| Mn      | 0.47                             | 0.50           | 0.47                   | 0.49                   | 0.32                            |
| P       | 0.004                            | 0.011          | 0.010                  | 0.011                  | 0.008                           |
| S       |                                  | 0.004          | 0.004                  | 0.004                  | 0.004                           |
| Si      | 0.31                             | 0.18           | 0.13                   | 0.14                   | 0.07                            |
| Ni      | 0.54                             | 0.43           | 1.14                   | 2.27                   | 2.30                            |
| Cr      | 11.2                             | 11.99          | 11.97                  | 11.71                  | 13.60                           |
| Mo      | 0.96                             | 0.93           | 1.04                   | 1.02                   | 1.59                            |
| V       | 0.31                             | 0.27           | 0.31                   | 0.31                   | 0.30                            |
| Nb      |                                  | 0.018          | 0.015                  | 0.015                  | 0.016                           |
| Ti      |                                  | 0.003          | 0.003                  | 0.003                  | 0.002                           |
| Co      | 0.02                             | 0.017          | 0.015                  | 0.021                  | 0.018                           |
| Cu      | 0.04                             | 0.05           | 0.05                   | 0.05                   | 0.05                            |
| Al      | 0.03                             | 0.030          | 0.017                  | 0.028                  | 0.018                           |
| B       | 0.0007                           | <0.001         | <0.001                 | <0.001                 | <0.001                          |
| W       | 0.5                              | 0.54           | 0.53                   | 0.54                   | 0.64                            |
| As      | 0.01                             | <0.001         | 0.002                  | <0.002                 | <0.002                          |
| Sn      | 0.01                             | 0.002          | 0.001                  | 0.002                  | 0.002                           |
| Zr      |                                  | <0.001         | <0.001                 | <0.001                 | <0.001                          |
| N       |                                  | 0.020          | 0.016                  | 0.017                  | 0.014                           |
| O       |                                  | 0.005          | 0.007                  | 0.007                  | 0.007                           |

<sup>a</sup>Balance iron.

| Element | Concentration, <sup>a</sup> wt % |                             |                                      |
|---------|----------------------------------|-----------------------------|--------------------------------------|
|         | 9 Cr-1 Mo (XA-3590)              | 9 Cr-1 Mo + 2% Ni (XA-3591) | 9 Cr-1 Mo + 2% Ni Adjusted (XA-3593) |
| C       | 0.09                             | 0.064                       | 0.067                                |
| Mn      | 0.36                             | 0.36                        | 0.36                                 |
| P       | 0.008                            | 0.008                       | 0.008                                |
| S       | 0.004                            | 0.004                       | 0.004                                |
| Si      | 0.08                             | 0.08                        | 0.10                                 |
| Ni      | 0.11                             | 2.17                        | 2.24                                 |
| Cr      | 8.62                             | 8.57                        | 12.30                                |
| Mo      | 0.98                             | 0.98                        | 1.70                                 |
| V       | 0.209                            | 0.222                       | 0.29                                 |
| Nb      | 0.063                            | 0.066                       | 0.074                                |
| Ti      | 0.002                            | 0.002                       | 0.002                                |
| Co      | 0.013                            | 0.015                       | 0.017                                |
| Cu      | 0.03                             | 0.04                        | 0.03                                 |
| Al      | 0.013                            | 0.015                       | 0.016                                |
| B       | <0.001                           | <0.001                      | <0.001                               |
| W       | 0.01                             | 0.01                        | 0.01                                 |
| As      | <0.001                           | <0.001                      | <0.001                               |
| Sn      | 0.003                            | 0.003                       | 0.003                                |
| Zr      | <0.001                           | <0.001                      | <0.001                               |
| N       | 0.050                            | 0.053                       | 0.059                                |
| O       | 0.007                            | 0.006                       | 0.007                                |

<sup>a</sup>Balance iron.

concentration range corresponds to alloy compositions with the lowest nickel concentration, whereas the high helium concentrations were achieved with the alloys containing about 2% nickel. Post-irradiation tensile tests were performed at room temperature (25°C) and at elevated temperature (300°C), and the results are listed in Tables VI.4-11 and VI.4-12.

As can be seen from Figs. VI.4-21 and VI.4-22, both the yield and the ultimate strength increased after irradiation. However, the uniform and total elongation were reduced by irradiation as shown in Figs. VI.4-23(a) and VI.4-23(b).

In particular, the uniform elongation dropped to values less than 0.5% for the room temperature post-irradiation tests, whereas the total elongation remained above about 2%. This behavior is indicative of localized plastic flow, a phenomenon also observed in highly irradiated austenitic steels.

By comparing the ratios of irradiated to unirradiated tensile properties among the alloys with different nickel concentration, Klueh and Vitek concluded that helium (at least in the range from 10 to 85 appm He) has no effect on either the increase in yield and ultimate strength, or on the reduction in uniform and total elongation. Hence, the observed changes in tensile properties are entirely due to the displacement damage. It should be noted, however, that this conclusion may not be applicable for irradiations at elevated temperature and to higher doses. Nevertheless, all the results so far on the tensile properties of irradiated ferritic steels look very promising. Recent investigations by Wassilev<sup>(28)</sup> on post-irradiation rupture life of the martensitic steel 1.4914 (whose composition is similar to HT-9 but contains additionally 0.25% Nb as a carbon-stabilizing element) showed modest reduction in the rupture life after an irradiation dose of about 5 dpa and a helium

Table VI.4-11

Tensile properties of unirradiated and irradiated normalized-and-tempered<sup>a</sup>  
12 Cr-1 MoVW Steels

| Fluence,<br>>0.1 MeV,<br>(neutrons/m <sup>2</sup> ) | Displace-<br>ment Level<br>(dpa) | Helium<br>Concentration <sup>b</sup><br>(at. ppm) | Test<br>Temperature<br>(°C) | Strength (MPa) |          | Strength<br>Ratios <sup>c</sup> |                | Elongation (%) |       |
|-----------------------------------------------------|----------------------------------|---------------------------------------------------|-----------------------------|----------------|----------|---------------------------------|----------------|----------------|-------|
|                                                     |                                  |                                                   |                             | Yield          | Ultimate | R <sub>y</sub>                  | R <sub>u</sub> | Uniform        | Total |
| × 10 <sup>25</sup>                                  |                                  |                                                   |                             |                |          |                                 |                |                |       |
| 12 Cr-1 MoVW (91354)                                |                                  |                                                   |                             |                |          |                                 |                |                |       |
| 0                                                   |                                  |                                                   | 25                          | 549            | 716      |                                 |                | 6.64           | 9.91  |
| 1.3                                                 | 9.3                              | 26                                                | 25                          | 983            | 987      | 1.79                            | 1.38           | 0.31           | 2.12  |
| 0                                                   |                                  |                                                   | 300                         | 490            | 647      |                                 |                | 5.53           | 8.63  |
| 1.2                                                 | 9.1                              | 20                                                | 300                         | 784            | 800      | 1.60                            | 1.24           | 1.36           | 4.49  |
| 12 Cr-1 MoVW (XXA-3587)                             |                                  |                                                   |                             |                |          |                                 |                |                |       |
| 0                                                   |                                  |                                                   | 25                          | 553            | 759      |                                 |                | 8.06           | 11.15 |
| 1.3                                                 | 9.3                              | 22                                                | 25                          | 980            | 992      | 1.77                            | 1.31           | 0.39           | 2.89  |
| 0                                                   |                                  |                                                   | 300                         | 483            | 652      |                                 |                | 5.10           | 7.98  |
| 1.2                                                 | 9.1                              | 18                                                | 300                         | 783            | 815      | 1.62                            | 1.25           | 1.80           | 5.08  |
| 12 Cr-1 MoVW-1 Ni (XXA-3588)                        |                                  |                                                   |                             |                |          |                                 |                |                |       |
| 0                                                   |                                  |                                                   | 25                          | 576            | 800      |                                 |                | 7.09           | 10.64 |
| 1.3                                                 | 9.3                              | 45                                                | 25                          | 1115           | 1134     | 1.94                            | 1.42           | 0.56           | 2.96  |
| 0                                                   |                                  |                                                   | 300                         | 519            | 761      |                                 |                | 5.94           | 9.04  |
| 1.2                                                 | 9.1                              | 41                                                | 300                         | 899            | 931      | 1.73                            | 1.22           | 1.11           | 3.78  |
| 12 Cr-1 MoVW-2 Ni (XXA 3589)                        |                                  |                                                   |                             |                |          |                                 |                |                |       |
| 0                                                   |                                  |                                                   | 25                          | 719            | 899      |                                 |                | 4.63           | 7.81  |
| 1.2                                                 | 9.1                              | 74                                                | 25                          | 1227           | 1249     | 1.71                            | 1.39           | 0.51           | 2.19  |
| 0                                                   |                                  |                                                   | 300                         | 646            | 827      |                                 |                | 3.49           | 6.18  |
| 0.89                                                | 8.5                              | 50                                                | 300                         | 1019           | 1128     | 1.58                            | 1.36           | 0.83           | 3.02  |
| 12 Cr-1 MoVW-2 Ni (adjusted) (XAA-3592)             |                                  |                                                   |                             |                |          |                                 |                |                |       |
| 0                                                   |                                  |                                                   | 25                          | 769            | 938      |                                 |                | 4.76           | 7.65  |
| 1.3                                                 | 9.3                              | 83                                                | 25                          | 1310           | 1338     | 1.70                            | 1.43           | 0.57           | 2.78  |
| 0                                                   |                                  |                                                   | 300                         | 696            | 894      |                                 |                | 3.21           | 6.29  |
| 1.2                                                 | 9.1                              | 63                                                | 300                         | 1056           | 1094     | 1.52                            | 1.22           | 0.65           | 3.07  |

<sup>a</sup>All heats were normalized by heating 0.5 h at 1050°C, then rapidly cooled in flowing helium. Heats 91354, XXA-3587, and XXA-3588 were tempered 2.5 h at 780°C; heats XXA-3589 and XXA-3592 were tempered at 700°C for 5 and 8 h, respectively.

<sup>b</sup>Calculated level of helium from <sup>58</sup>Ni and <sup>10</sup>B. [Each alloy was assumed to contain 0.007 wt % B (total)].

<sup>c</sup> $R_y$  = ratio of irradiated-to-unirradiated yield strengths;  $R_u$  = ratio of irradiated to unirradiated ultimate tensile strengths.



Table VI.4-12

Tensile properties of unirradiated and irradiated<sup>a</sup> normalized-and-tempered<sup>b</sup>  
9 Cr-1 MoVNb Steels

| Fluence,<br>>0.1 Mev<br>(neutrons/m <sup>2</sup> ) | Displace-<br>ment Level<br>(dpa) | Helium<br>Concentration <sup>c</sup><br>(at. ppm) | Test<br>Temperature<br>(°C) | Strength (MPa) |          | Strength<br>Ratio <sup>d</sup> |                | Elongation (%) |       |
|----------------------------------------------------|----------------------------------|---------------------------------------------------|-----------------------------|----------------|----------|--------------------------------|----------------|----------------|-------|
|                                                    |                                  |                                                   |                             | Yield          | Ultimate | R <sub>Y</sub>                 | R <sub>U</sub> | Uniform        | Total |
|                                                    |                                  |                                                   |                             |                |          |                                |                |                |       |
| 9 Cr-1 MoVNb (XA-3590)                             |                                  |                                                   |                             |                |          |                                |                |                |       |
| 0                                                  |                                  |                                                   | 25                          | 541            | 656      |                                |                | 5.11           | 9.61  |
| 1.3 × 10 <sup>26</sup>                             | 9.3                              | 11                                                | 25                          | 878            | 878      | 1.62                           | 1.34           | 0.23           | 3.23  |
| 0                                                  |                                  |                                                   | 300                         | 483            | 581      |                                |                | 3.59           | 7.13  |
| 1.1                                                | 7.6                              | 10                                                | 300                         | 716            | 716      | 1.48                           | 1.23           | 0.22           | 3.70  |
| 9 Cr-1 MoVNb-2 Ni (XA-3591)                        |                                  |                                                   |                             |                |          |                                |                |                |       |
| 0                                                  |                                  |                                                   | 25                          | 734            | 851      |                                |                | 3.74           | 7.50  |
| 1.3                                                | 9.3                              | 80                                                | 25                          | 1289           | 1297     | 1.76                           | 1.52           | 0.36           | 1.60  |
| 0                                                  |                                  |                                                   | 300                         | 650            | 757      |                                |                | 2.29           | 5.29  |
| 1.2                                                | 8.6                              | 72                                                | 300                         | 1088           | 1107     | 1.67                           | 1.46           | 0.41           | 2.72  |
| 9 Cr-1 MoVNb-2 Ni (adjusted) (XA-3593)             |                                  |                                                   |                             |                |          |                                |                |                |       |
| 0                                                  |                                  |                                                   | 25                          | 817            | 927      |                                |                | 3.44           | 7.46  |
| 1.3                                                | 9.3                              | 82                                                | 25                          | 1286           | 1298     | 1.57                           | 1.40           | 0.48           | 2.63  |
| 0                                                  |                                  |                                                   | 300                         | 766            | 878      |                                |                | 3.57           | 6.59  |
| 1.1                                                | 7.7                              | 63                                                | 300                         | 1093           | 1114     | 1.43                           | 1.27           | 0.51           | 2.31  |

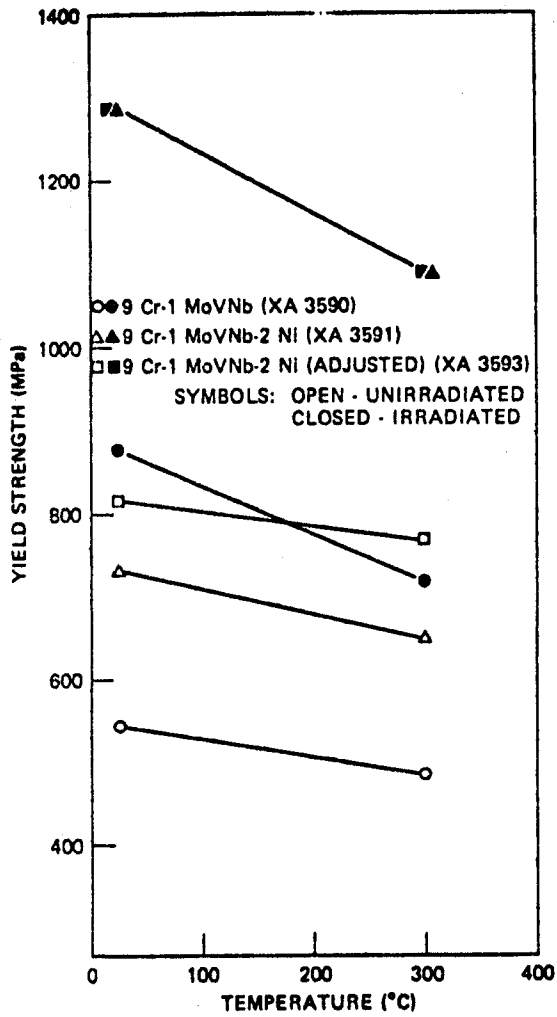
<sup>a</sup>Irradiation was in HFIR at 50°C.

<sup>b</sup>All heats were normalized by heating 0.5 h at 1040°C, then rapidly cooled in flowing helium. Heat XA-3590 was tempered 1 h at 760°C; heats XA-3591 and XA-3593 were tempered at 700°C for 5 and 8 h, respectively.

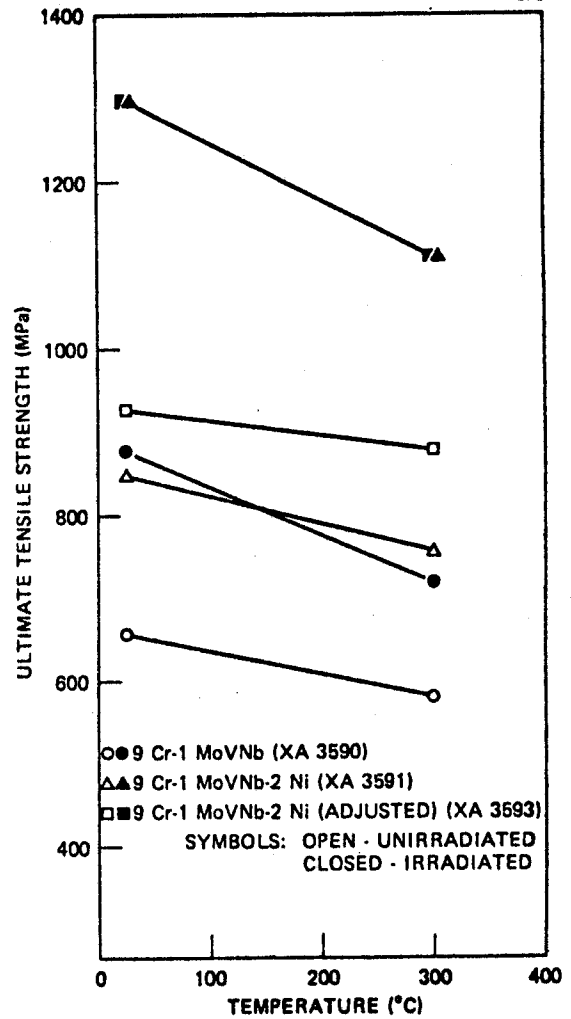
<sup>c</sup>Calculated level of helium from <sup>58</sup>Ni and <sup>10</sup>B. [Each alloy was assumed to contain 0.0007 wt % B (total)].

<sup>d</sup>R<sub>y</sub> = ratio of irradiated-to-unirradiated yield strengths; R<sub>U</sub> = ratio of irradiated-to-unirradiated ultimate tensile strengths.

Fig. VI.4-21

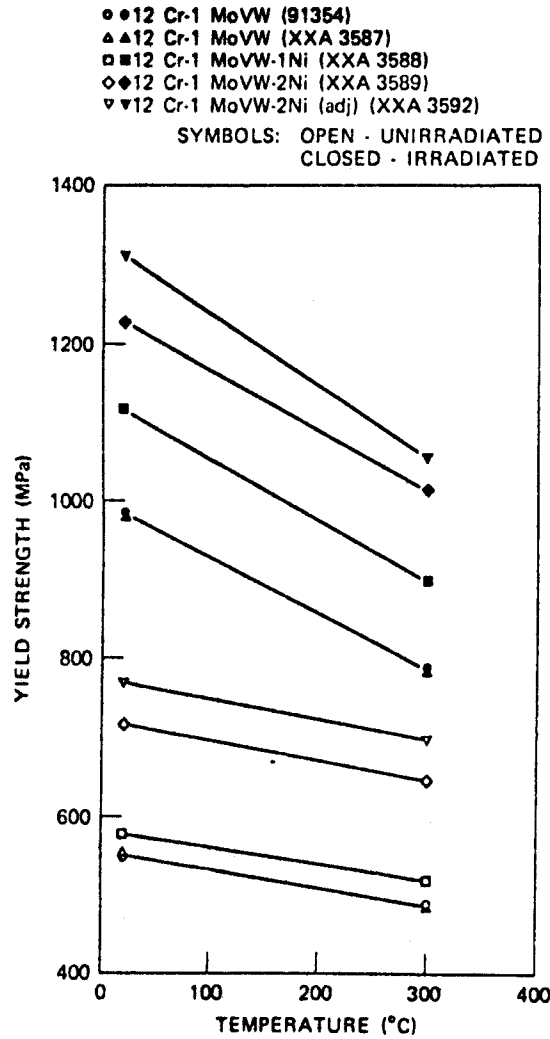


a) The 0.2% yield strengths of 9 Cr-1 MoV Nb steels unirradiated and after HFIR irradiation to produce up to 9.3 dpa and 10 to 80 at. ppm He.

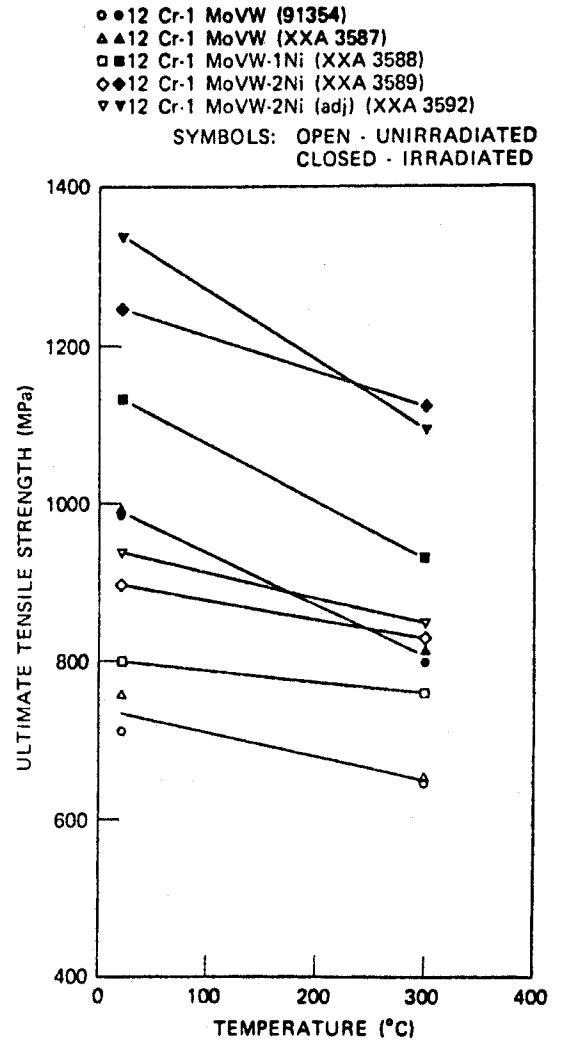


b) The ultimate tensile strengths of 9 Cr-1 MoV Nb steels.

Fig. VI.4-22

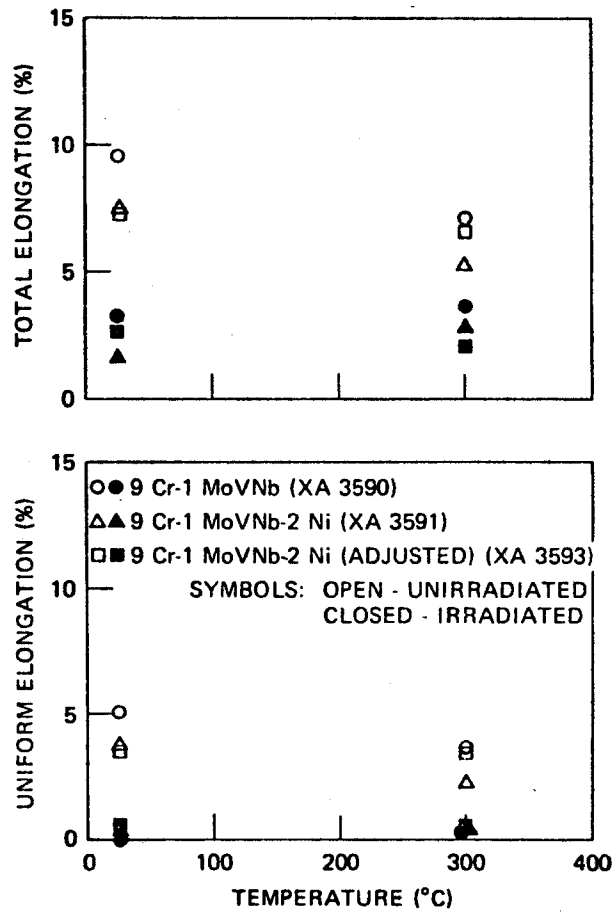


a) The 0.2% yield strengths of 12 Cr-1 MoVW steels unirradiated and after irradiation in HFIR to produce up to 9.3 dpa and 20 to 80 at. ppm He.



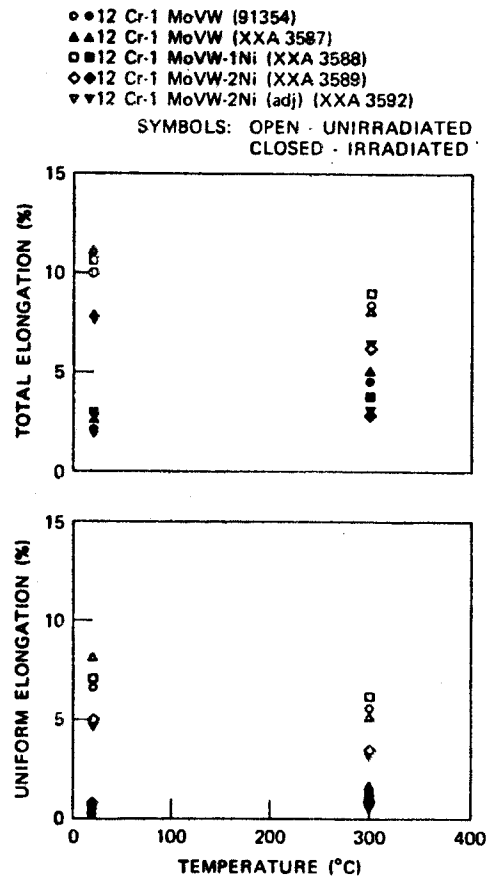
b) The ultimate tensile strength of 12 Cr-1 MoVW steels unirradiated and after irradiation in HFIR.

Fig. VI.4-23



(a) The uniform and total elongation of 9 Cr-1 MoVNb steel unirradiated and after irradiation in HFIR to produce up to 9.3 dpa and 10 to 80 at. ppm He.

Fig. VI.4- 23



(b) The uniform and total elongation of 12 Cr-1 MoVW steels unirradiated and after irradiation in HFIR to produce up to 9.3 dpa and 10 to 80 at. ppm He.

accumulation of 90 appm. The results are shown in Fig. VI.4-24(a). The reduction in rupture life can be explained by two simultaneous effects of radiation damage. First, there is an increase in post-irradiation thermal creep as shown in Fig. VI.4-24(b). Second, the ductility for the post-irradiation test is reduced as illustrated by the results of Fig. VI.4-24(c). It should be noted that the post-irradiation ductility loss of this martensitic steel is much less severe than for austenitic alloys. Figure VI.4-24(d) shows the strain to rupture versus rupture life for both the irradiated martensitic steel 1.4914 and the irradiated, and Ti-stabilized austenitic steel 1.4970. It is seen that the rupture strain decreases rapidly with increasing rupture life for the austenitic steel, but it remains nearly independent of the rupture life and at a more than adequate level of  $\gtrsim 5\%$  for the martensitic steel.

Microstructural evaluation showed that the dramatic loss in ductility is due to the well-known helium embrittlement of grain boundaries in the austenitic steel, leading to intergranular failure. In contrast, the fracture mode in the martensitic steel is characterized by ductile transgranular failure in spite of the fact that helium is found to promote also the grain boundary bubble formation. It appears that the martensitic steels have a much reduced susceptibility for helium embrittlement as compared to austenitic steels.

#### VI.4.8.5 In-Reactor Creep of HT-9

There are basically two sources of information in this area, both from Paxton et al.<sup>(19,24)</sup>. They found that the in-reactor creep of HT-9 is superior to CW 316 SS, especially in the 550°C to 650°C range (much higher than we propose to use HT-9). Figure VI.4-25 shows that at stress levels of 70 MPa ( $\sim 10$  ksi) the total creep at 560°C after  $1.5 \times 10^{23}$  n/cm<sup>2</sup> ( $\sim 75$  dpa) is

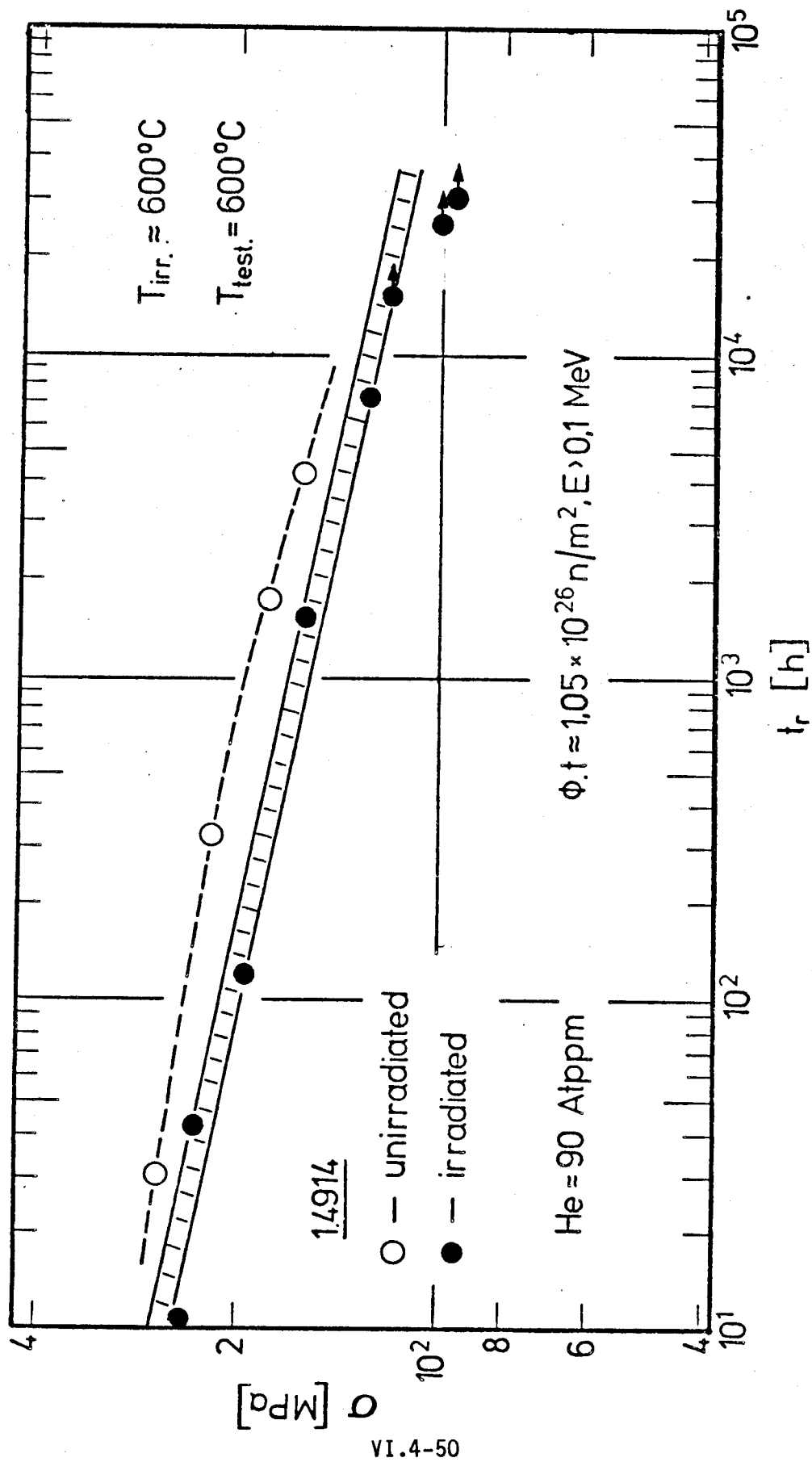
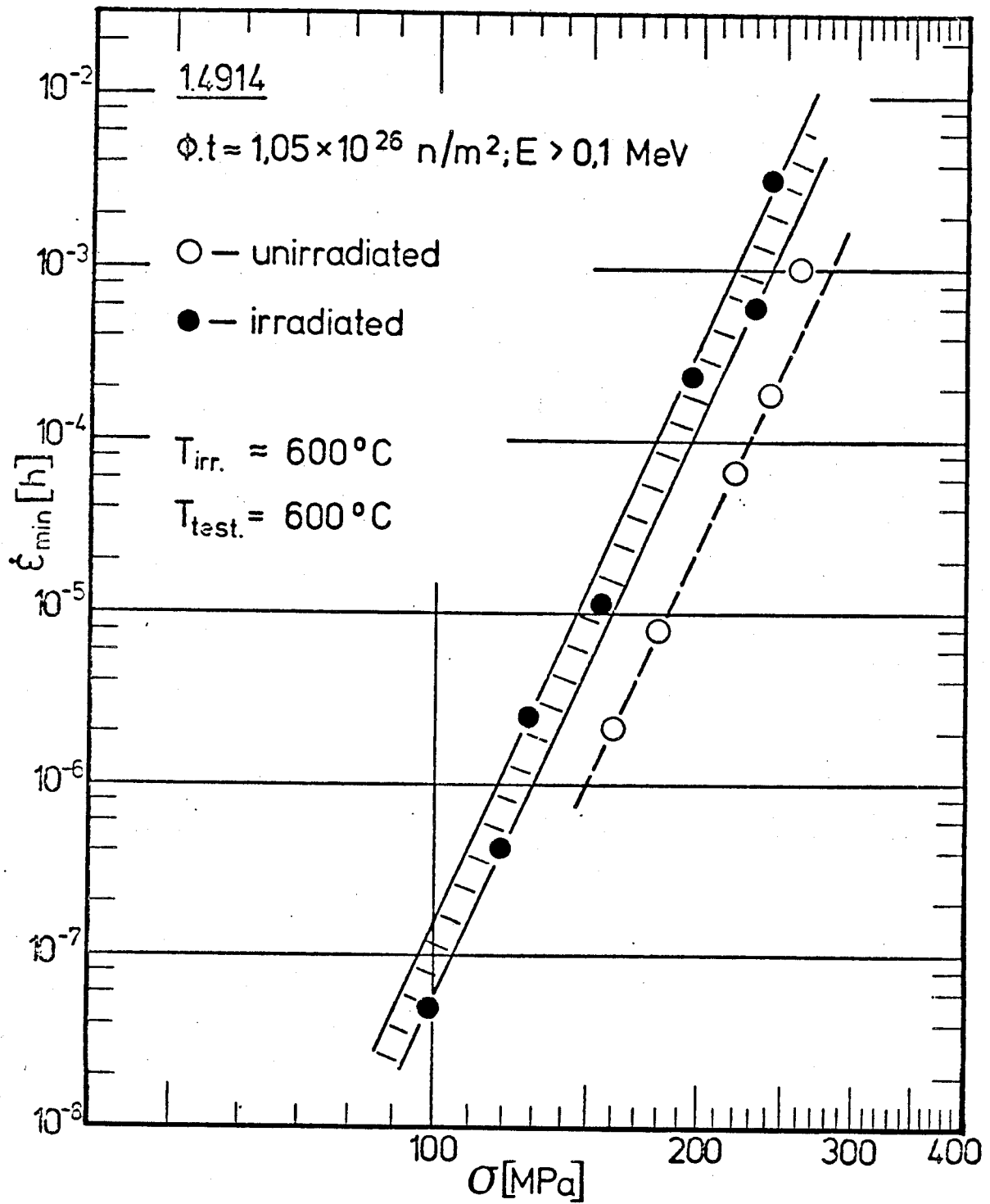


Fig. VI.4-24a Rupture life of the unirradiated and irradiated martensitic steel 1.4914.

Fig. VI.4-24(b)

Minimum creep rate of the unirradiated and irradiated martensitic steel 1.4914.





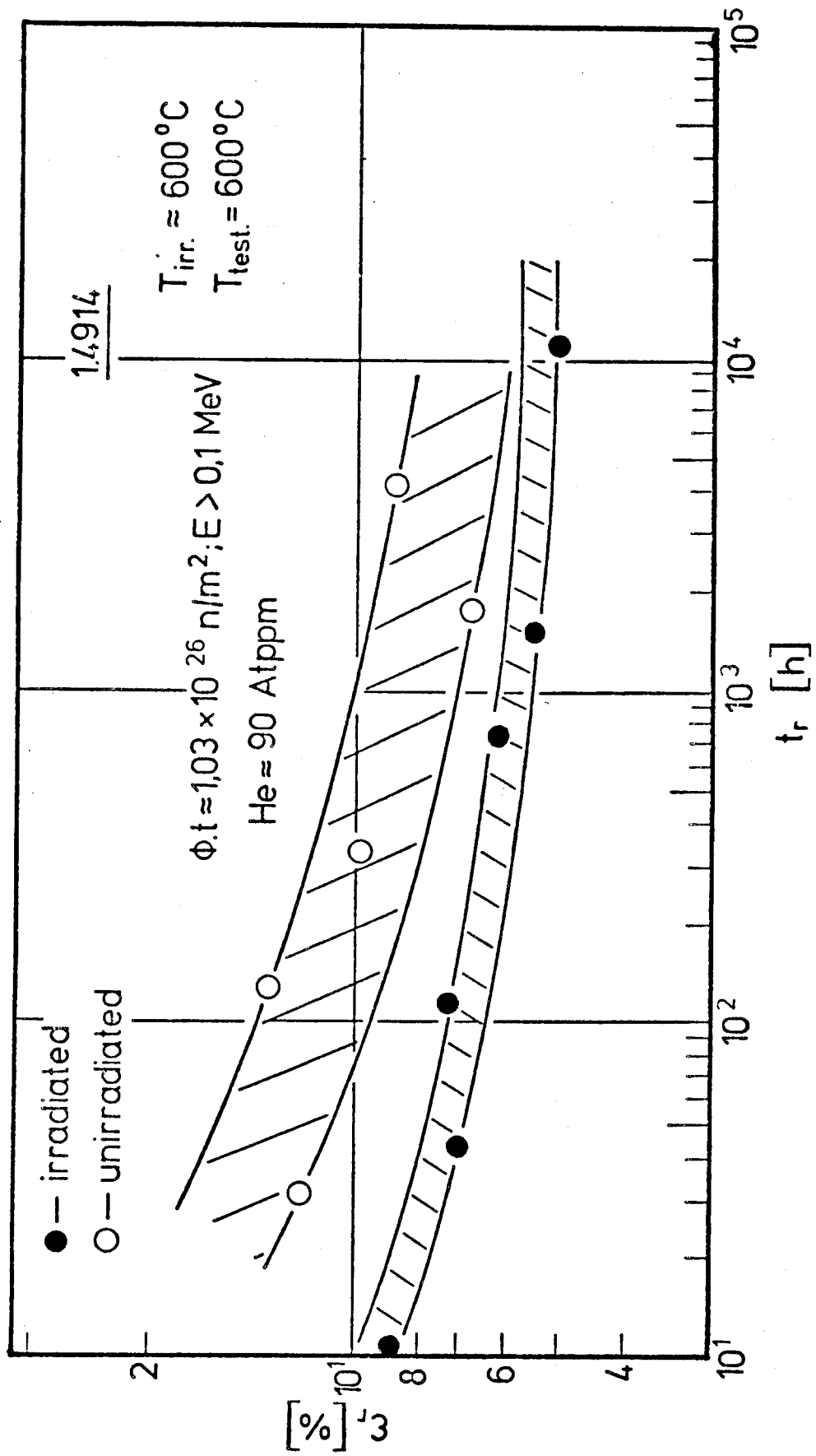


Fig. VI.4-24(c) Creep ductility of the unirradiated and irradiated martensitic steel 1.4914.

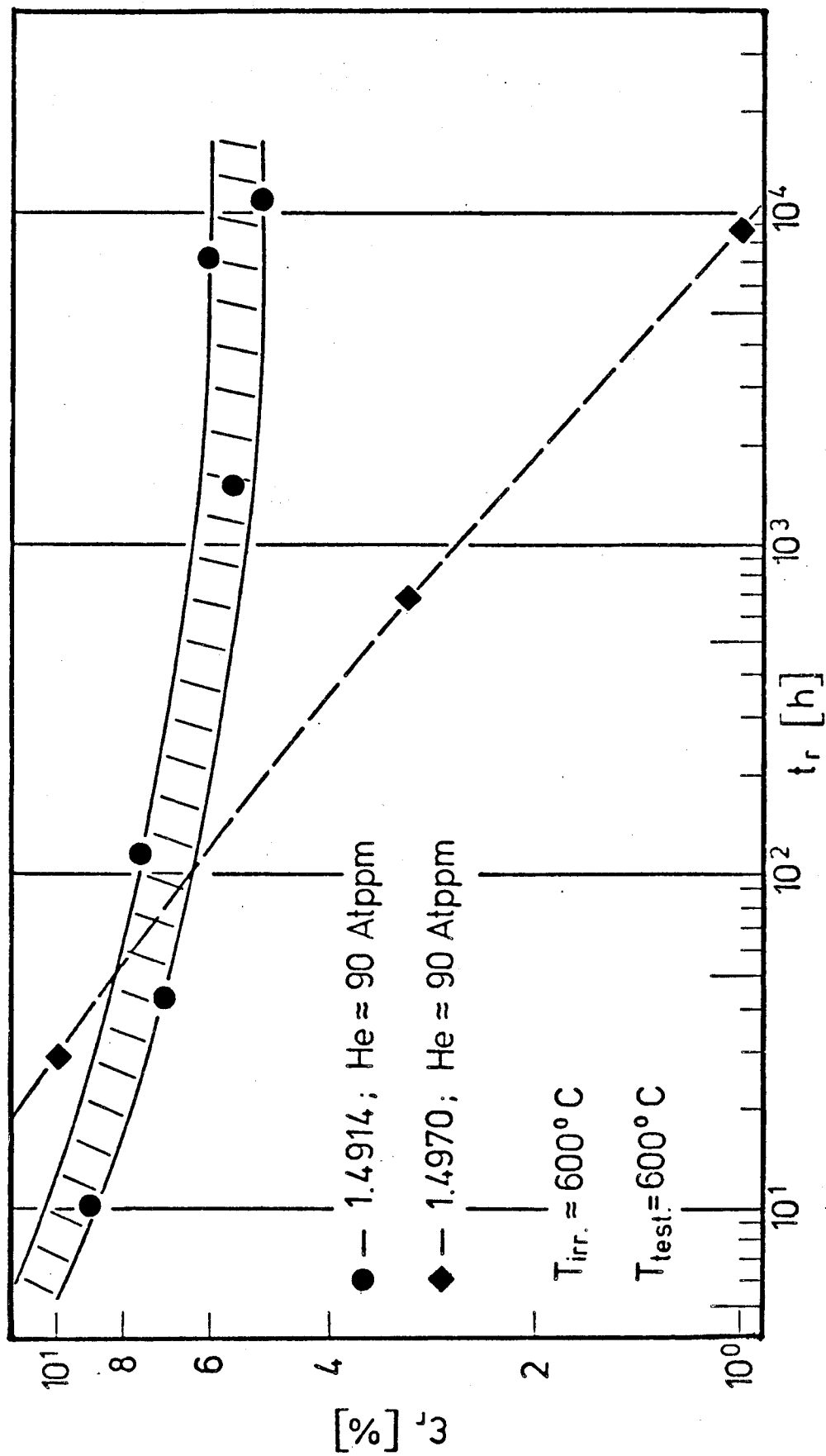
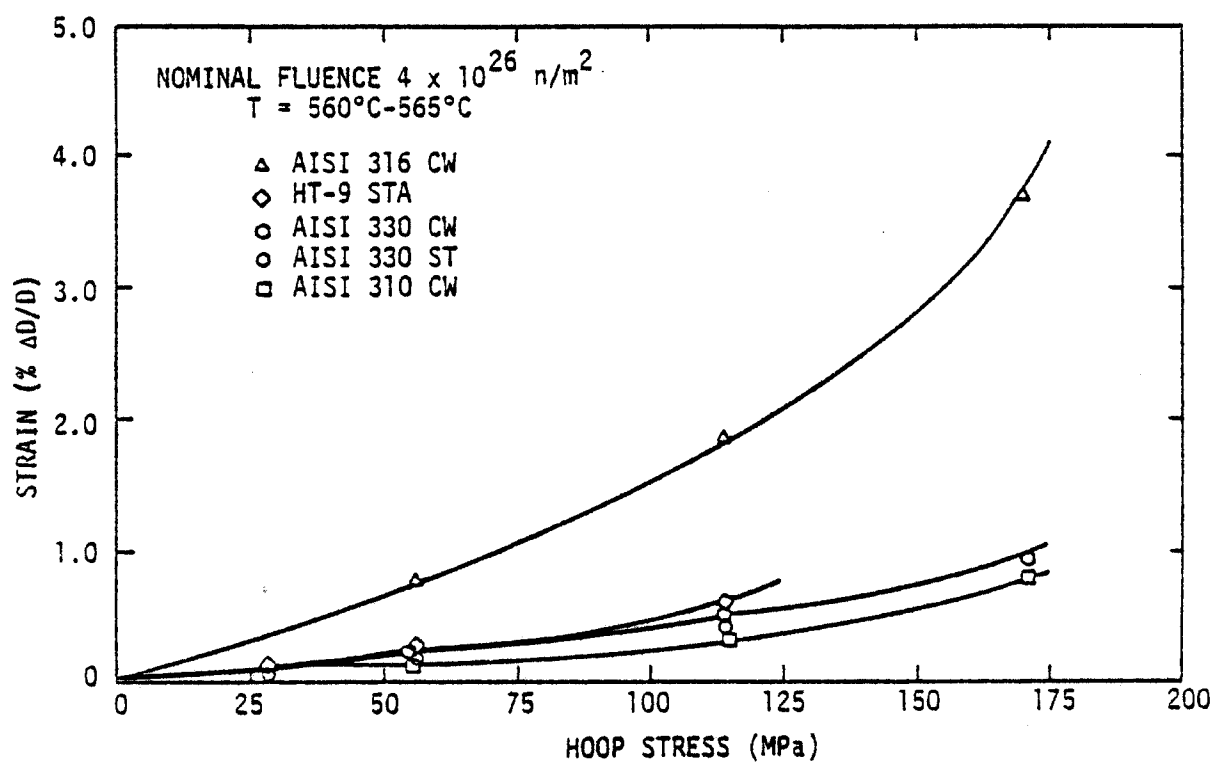


Fig. VI.4-24(d) Comparison of the ductility between the irradiated steels 1.4914 and 1.4970.

Fig. VI.4-25



In-reactor creep of HT-9 (12Cr-1Mo-0.3V) compared to 20% CW 316SS (ref. 19,24).

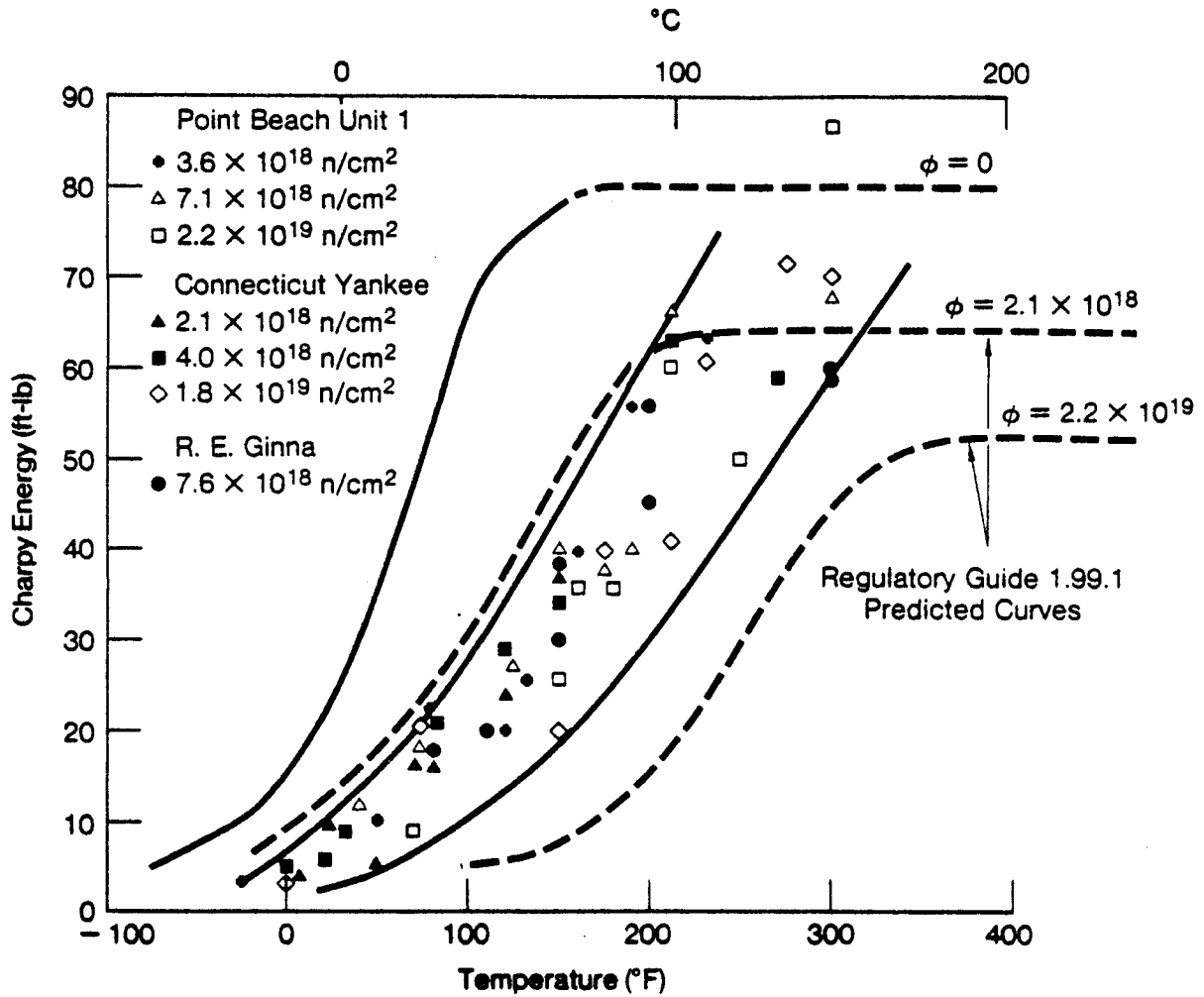
~ 1.3%. This is reduced to ~ 0.7% at 450°C. Paxton et al.<sup>(19,24)</sup> show that at hoop stresses of 70 MPa and  $2 \times 10^{22}$  n/cm<sup>2</sup>, diametral strains of only 0.4% are observed at 605°C. However, recent irradiation creep measurements by Herschback et al.<sup>(29)</sup> on 2-1/2 Cr-1 Mo steel did not show any radiation-enhanced creep at a temperature of 450°C and a tensile stress of 100 MPa. At higher temperatures there appears to be an enhancement of thermal creep. Similar trends have also been observed in a 12 Cr martensitic steel according to private communications with Herschback and Ehrlich.

#### VI.4.8.6 DBTT Shift During Neutron Irradiation

Practically all the information available on DBTT shifts for ferritic steels has been obtained through low temperature (< 300°C) and low fluence (<  $10^{21}$  ncm<sup>-2</sup> or ~ 0.5 dpa) data. Past experience has shown that if ferritic steels are irradiated under these conditions, significant DBTT shifts can occur. Figure VI.4-26 gives data from seven different fluence levels on A 302B ferritic steel.<sup>(33)</sup> It can be seen that the DBTT is ~ 100°C at irradiation temperatures of 288°C up to  $2.2 \times 10^{19}$  n/cm<sup>2</sup> (0.01 dpa). This DBTT shift can be partially recovered and the 50% recovery time is 17 hr at 400°C or ~ 2 hr at 440°C.<sup>(28)</sup> Anderko<sup>(21)</sup> has recently reviewed the known DBTT shift and given the tentative correlation with irradiation temperature as shown in Fig. VI.4-27.

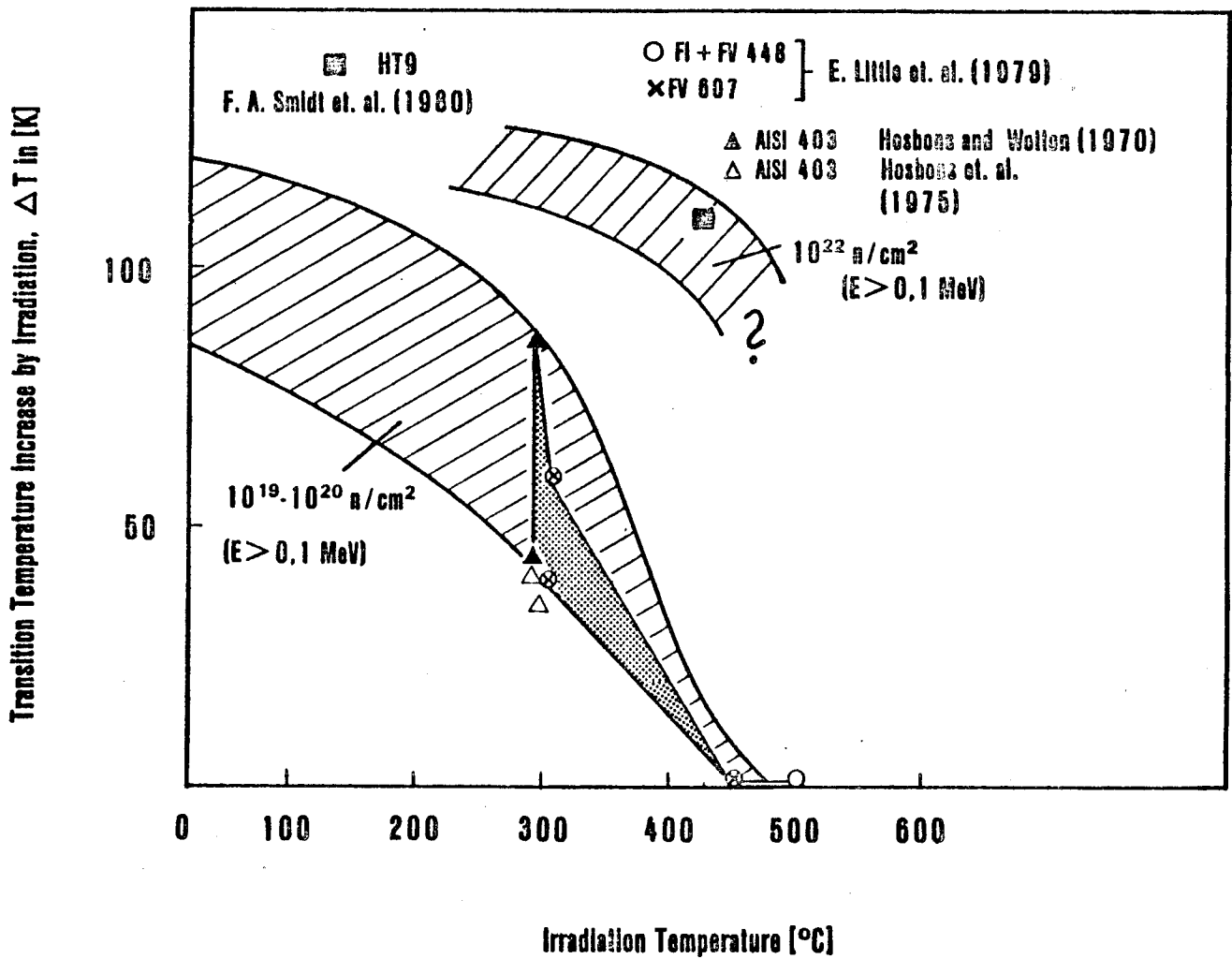
There is only one set of data on the HT-9 alloy at high temperatures and fluences. Smidt et al.<sup>(17)</sup> found that the transition temperature shifted from ~ 0°C (as received) to ~ 100°C after a 5 dpa ( $1 \times 10^{22}$  n/cm<sup>2</sup>) irradiation at 427°C (Fig. VI.4-28). Presumably this shift will drop rapidly at higher temperatures so operation up to 530°C may not be a problem. Indeed, British scientists have found there is no DBTT shift following neutron irradiation at

Fig. VI.4-26



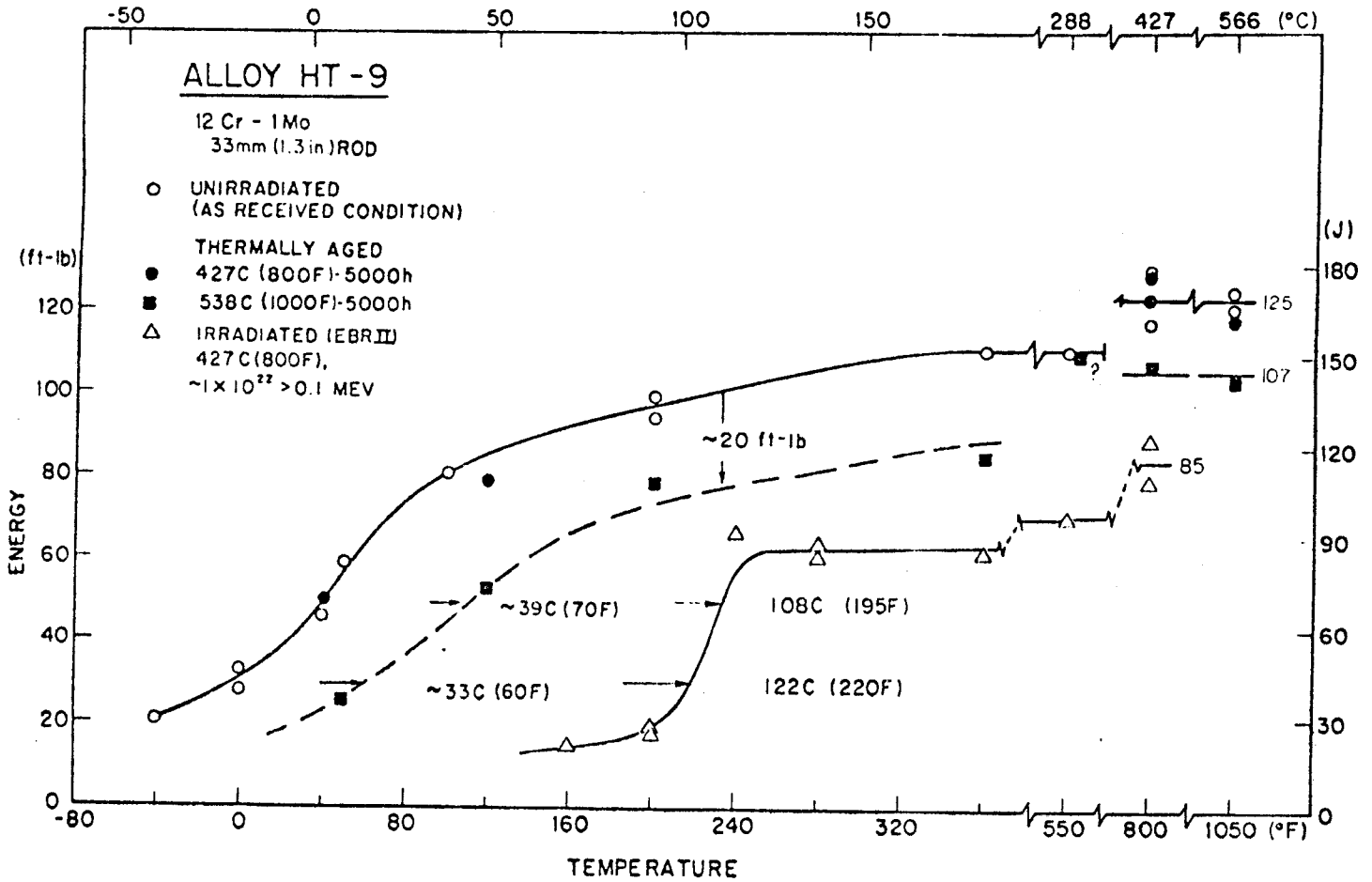
Low flux irradiation results for ASTM Reference A 302 B (0.20 weight percent Cu, 0.011 weight percent P) from three reactors. Charpy data for seven fluence levels with predicted Charpy curves (Regulatory Guide 1.99.1) for the lowest and highest levels.(28)

Fig. VI.4-27



Increase of impact transition temperature as a function of irradiation conditions (Charpy notch specimens).

Fig. VI.4-28



450°C.<sup>(33)</sup> However, it is not known how much the DBTT shift will be after irradiation in the low temperature region (~ 350°C), but it is expected to be a maximum of ~ 40°C per dpa. Damage accumulated in that temperature region may have to be removed by periodic annealing.

#### VI.4.9 Fabrication of TASKA Blanket Module

Several considerations should be discussed pertaining to the fabrication of an HT-9 TASKA blanket module. They consist of the fabrication and procurement of the proper product forms, manufacture of pieces into the component's configuration through welding, the post weld heat treatment (PWHT) of the weldments, and non-destructive examination to ensure a high reliability of the completed structure. These issues are discussed below.

##### VI.4.9.1 Manufacture of HT-9 TASKA Blanket Module Tubes

Table VI.4-13 lists the tube and pipe material specifications for use of 12 Cr-1 Mo steels including the alloy grade and designation and known producers. This is an indication of the common use of HT-9 in pipe and tubular products and their acceptance by industry and standards. Table VI.4-14 lists some specific applications in CEGB boilers for HT-9 (or the 12 Cr-1 Mo equivalent alloy) which use welded tubes on the same order of size as those of the TASKA plant blanket design. Also included are notations of failures and the reasons.

As is seen in Table VI.4-14, tubes of nearly the same dimensions as those for TASKA are used frequently and successfully. These tubes are placed into service in the fully tempered condition. The manufacturers' recommend an austenitization at 1050°C for 0.5-1.0 hrs with air-cool, followed by a tempering treatment at 760-780°C for 1.5-2.5 hrs, also followed by an air-cool. This is done to "normalize" the microstructure, which relieves stresses and



Table VI.4-13. Tube and Pipe Material Standards that Include 12 Cr-1 Mo Ferritic/Martensitic Steel Alloys(a)

| Country of Material Standard | Standard Number           | Standard Titles                                                                                                      | Alloy Grade and/or Designation           | Alloy Composition (wt%) |     |      | Additions                            | Trade Names or Alloys |
|------------------------------|---------------------------|----------------------------------------------------------------------------------------------------------------------|------------------------------------------|-------------------------|-----|------|--------------------------------------|-----------------------|
|                              |                           |                                                                                                                      |                                          | Cr                      | Mo  | C    |                                      |                       |
| U.K.                         | BS 3604-78                | Steel pipes and tubes for pressure purposes; ferritic alloy steel with specified elevated temperature properties     | CFS 762<br>HFS 762                       | 12                      | 1   | 0.2  | 0.3V-0.5W-0.5Ni                      | (Equivalent to HT-9)  |
|                              | None                      | This alloy has not been used in tube form; applications have been in bar and forgings                                | ---                                      | 12                      | 1.7 | 0.1  | 2.5Ni-0.35V                          | Jethete M152          |
| FRG                          | DIN17175-79               | Seamless steel tubes for elevated temperatures                                                                       | Werkstoff No.                            | 12                      | 1   | 0.2  | 0.5Ni-0.3V                           | HT-91                 |
|                              | Votuev Werkstoffblatt 110 |                                                                                                                      | Werkstoff No. 1.4935; alloy X20CrMoVW121 | 12                      | 1   | 0.2  | 0.5Ni-0.3V<br>0.5W-0.1Nb             |                       |
| Russia                       | None                      | Product forms not known                                                                                              | 20KH12WNMF                               | 12                      | 0.6 | 0.2  | 0.7Ni-0.3V-0.1Ti<br>0.9W-0.1Nb-0.2Cu | Unknown               |
|                              |                           |                                                                                                                      | 15KH12VNMF                               | 12                      | 0.6 | 0.15 | 0.6Ni-0.2V<br>0.9W                   |                       |
|                              | 5-2006-73                 | Steel heat exchanger and communication tubes (closest applicable standard which only includes the lower Cr-Mo alloy) | 18KH12VMBFP                              | 12                      | 0.5 | 0.18 | 0.6Ni-0.2V<br>0.5W-0.3Nb-0.002B      |                       |

(a) Manufacturing process: seamless tube and pipe, normalized and highly treated.

Table VI.4-14. Superheater and Reheater Applications of Martensitic/Ferritic Steels in CEGB Boilers

| Material  | Station               | Position                                               | Operating Temperature (°C)      | Wall Dimensions |                |                  | Life (h) | Failures (h)     | Notes                                        |
|-----------|-----------------------|--------------------------------------------------------|---------------------------------|-----------------|----------------|------------------|----------|------------------|----------------------------------------------|
|           |                       |                                                        |                                 | Dia. (OD) (mm)  | Thickness (mm) | Total Length (m) |          |                  |                                              |
| 12Cr-Mo-V | Aberthaw (2)          | Radiant                                                | 430-460                         |                 |                |                  | 16,900   | 6/17,000         | Failures due to overheating                  |
|           | "A" 1<br>4            | superheater<br>superheater                             | 430-460                         |                 |                |                  | 18,600   | 8 tubes          |                                              |
|           | Aberthaw (2)<br>"B" 7 | Division<br>wall pendant<br>superheater<br>Superheater | 600-650 but<br>can reach<br>800 |                 |                |                  | 3,700    |                  |                                              |
|           | 8                     |                                                        |                                 |                 |                |                  | 16,500   | 2,000<br>2 tubes | Overheating<br>Most of creep<br>life used up |
|           | 9                     | Superheater                                            |                                 |                 |                |                  | 11,000   |                  |                                              |
|           | High Marnham<br>(1) 5 | Final stage<br>reheater                                | 540-650                         | 54              | 4.5            | 18               | 35,000   | None             |                                              |
|           | Staythorpe            | Horizontal                                             | 550                             | 58.5            | 4.5            |                  | 4,000    | None<br>reported |                                              |
|           | Northfleet (6)        | Radiant<br>superheater                                 | 540-850                         |                 | 727            | 24,000           | 20,000   | Numerous         | Overheating                                  |
|           | Bankside (6)          | Secondary<br>superheater                               | 590                             |                 |                | 9                | 22,000   | None             | Loss of tube wall<br>thickness               |
|           | Kingsnorth (6)        | Downcomer<br>tubes                                     | Up to 622                       | 49              | 6              | 10               | 8,400    | None             |                                              |

promotes recovery and recrystallization of any material work-hardened by the forming or tube-drawing process.

The TASKA blanket tubes call for a tapered end for the reasons discussed in Section VI.1.3 which fit in the manifold for welding. A taper is put on a tube by a process called "spinning" whereby the tube-end is circumferentially compressed in a mandril while the tube is spun. This sort of process is usually done at warm-forming temperatures,  $\sim 700^{\circ}\text{C}$ , but below the transformation temperature to austenite. The tapered end will have a gradually thickening wall up to the narrow end. If this is perceived as an issue when welding the conical section to the manifold, the tapered end could be machined to a uniform wall thickness; however, there is no apparent reason for concern.

The bends of the order required for design can be done either at warm temperatures ( $\sim 700^{\circ}\text{C}$ ), or at low temperatures by using a low melting point alloy, solidified inside the tube. This inhibits the buckling of the tube wall by internal hydrostatic pressure. Tubes of HT-9 have been successfully produced and cold and warm bent for heat exchanger tubing in the FRG where the technology is well developed, while 9 Cr-1 Mo tubes similar in properties to HT-9, have been processed in the U.S.

#### VI.4.9.2 Welding and Heat Treatment of the TASKA Module

The use of the martensitic steel has been thought (at least in the U.S.) to be a feasibility issue in the large, high restrained structures. This is primarily due to the lack of U.S. experience in pressure vessel and piping applications, in contrast to the situation in the European community. Table VI.4-15 gives a synopsis of the current applications of HT-9 in steam piping. Although weldability is perceived as an issue, a study released by General

Table VI.4-15. Typical Service Data for X20CrMoV(W)121 (Alloy HT-9 Type) Steam Piping

| Name               | Plant Data | Location | Size<br>(MW) | Main Steam      |              |                       | Reheat Steam    |              |                       | Pipe<br>o.d.<br>(in.) | Pipe<br>Wall<br>(in.) | Start-up<br>Year |
|--------------------|------------|----------|--------------|-----------------|--------------|-----------------------|-----------------|--------------|-----------------------|-----------------------|-----------------------|------------------|
|                    |            |          |              | Press.<br>(atm) | Temp<br>(°C) | Pipe<br>o.d.<br>(in.) | Press.<br>(atm) | Temp<br>(°C) | Pipe<br>o.d.<br>(in.) |                       |                       |                  |
| Arzberg            |            | FRG      | 300          | 220             | 540          | 11                    | 55              | 540          | 17                    | 0.5                   |                       | 1974             |
| Neurath A, B, C    |            | FRG      | 300          | 200             | 535          | 14                    | --              | --           | --                    | --                    |                       | 1972             |
| Neurath D, E       |            | FRG      | 600          | 195             | 535          | 20                    | --              | --           | --                    | --                    |                       | 1975             |
| Weisweiler G, H    |            | FRG      | 600          | 195             | 535          | 20                    | --              | --           | --                    | --                    |                       | 1974             |
| Frimmersdorf Q     |            | FRG      | 300          | 200             | 535          | 14                    | --              | --           | --                    | --                    |                       | 1970             |
| Robert Frank IV    |            | FRG      | 450          | 286             | 540          | 8.5                   | --              | --           | --                    | --                    |                       | 1973             |
| Lausward           |            | FRG      | 300          | 215             | 540          | 12                    | 55              | 540          | 29                    | 1.4                   |                       | 1976             |
| Asnaesvaerket      |            | Denmark  | 255          | 210             | 545          | 9                     | --              | --           | --                    | --                    |                       | 1968             |
| Stignsnaevaerket   |            | Denmark  | 250          | 210             | 560          | 9                     | --              | --           | --                    | --                    |                       | 1969             |
| Skaerbackvaerket   |            | Denmark  | 250          | 190             | 545          | 11                    | --              | --           | --                    | --                    |                       | 1969             |
| IS Vestkraft       |            | Denmark  | 250          | 193             | 545          | 12                    | --              | --           | --                    | --                    |                       | 1969             |
| Studstrupvaerket   |            | Denmark  | 250          | 174             | 545          | 13                    | --              | --           | --                    | --                    |                       | 1972             |
| Fynsvaerket        |            | Denmark  | 256          | 185             | 535          | 8                     | --              | --           | --                    | --                    |                       | 1974             |
| NEFO               |            | Denmark  | 300          | 199             | 535          | 13                    | --              | --           | --                    | --                    |                       | 1976             |
| Kraftwerk          |            |          |              |                 |              |                       |                 |              |                       |                       |                       |                  |
| Hanasaari          |            | Finland  | 150          | 160             | 535          | 9                     | --              | --           | --                    | --                    |                       | 1973             |
| KW Amer-Centrale   |            | Holland  | 175          | 195             | 575          | 6                     | --              | --           | --                    | --                    |                       | 1960             |
| Centrale-Velsen    |            | Holland  | 460          | 181             | 535          | 18                    | --              | --           | --                    | --                    |                       | 1974             |
| KW Maasvlakte      |            | Holland  | 550          | 199             | 540          | 17                    | --              | --           | --                    | --                    |                       | 1974             |
| KW Rovinari        |            | Rumania  | 330          | 196             | 540          | 10                    | --              | --           | --                    | --                    |                       | 1974             |
| KW Wolfersheim     |            | FRG      | 65           | 210             | 550          | 7.5                   | --              | --           | --                    | --                    |                       | 1962             |
| KW Standinger      |            | FRG      | 250          | 260             | 545          | 9                     | --              | --           | --                    | --                    |                       | 1964             |
| Robert Frank III   |            | FRG      | 300          | 260             | 545          | 10                    | --              | --           | --                    | --                    |                       | 1967             |
| KW Westfalen       |            | FRG      | 320          | 226             | 550          | 12                    | --              | --           | --                    | --                    |                       | 1968             |
| CKW Weser Veltheim |            | FRG      | 300          | 198             | 535          | 13                    | --              | --           | --                    | --                    |                       | 1969             |
| KW Offleben C      |            | FRG      | 325          | 181             | 525          | 10                    | --              | --           | --                    | --                    |                       | 1972             |
| KW Certsheimwerk   |            | FRG      | ?            | 181             | 530          | 13                    | --              | --           | --                    | --                    |                       | 1972             |
| KW Franken I       |            | FRG      | 300          | 225             | 540          | 14                    | --              | --           | --                    | --                    |                       | 1972             |
| KW Neideraubein    |            | FRG      | 600          | 175             | 530          | 18                    | --              | --           | --                    | --                    |                       | 1973             |
| KW Lunen           |            | FRG      | ?            | 136             | 525          | 10                    | --              | --           | --                    | --                    |                       | 1971             |
| KW Moorberg        |            | FRG      | 515          | 215             | 540          | 16                    | --              | --           | --                    | --                    |                       | 1973             |
| KW Scholven        |            | FRG      | 714          | 210             | 535          | 19                    | --              | --           | --                    | --                    |                       | 1974             |
| KW Emsland         |            | FRG      | 365          | 181             | 530          | 13                    | --              | --           | --                    | --                    |                       | 1974             |
| KW Frauen          |            | FRG      | 300          | 225             | 540          | 14                    | --              | --           | --                    | --                    |                       | 1975             |
| KW Veltheim        |            | FRG      | 450          | 235             | 540          | 12                    | --              | --           | --                    | --                    |                       | 1975             |
| KW Hückingen       |            | FRG      | 300          | 176             | 530          | 13                    | --              | --           | --                    | --                    |                       | 1975             |
| HKW Niehler Hafen  |            | FRG      | 300          | 178             | 540          | 13                    | --              | --           | --                    | --                    |                       | 1975             |

Atomic has noted that only two failures related to welding are known to have occurred.

Commercial experience, therefore, has demonstrated that this alloy is readily weldable if proper precautions are taken regarding preheat and post weld heat treatment (PWHT). However, the as-welded joint is untempered martensite, which is prone to fracture if the weld is handled improperly and to hydrogen cracking if the weld is allowed to sit idle prior to the PWHT. The function of the preheat is to input sufficient heat to slow the cooling rate after welding, to drive off any moisture or hydrocarbons absorbed in the plate surface prior to welding, and to reduce the thermal shock to the weld region. In the case of multipass welds, interpass temperatures are sometimes specified to prevent the transformation to martensite, which occurs around 250°C. The PWHT is generally recommended immediately after welding to temper the joint, which lowers the strength considerably, but increases its toughness.

The conical section of each tube of the TASKA blanket will be welded to the manifold as shown in Fig. VI.1-7. As stated, the welds must be full penetration to insure that outgassing will not be a significant problem for the vacuum system. In order to insure this, non-destructive examinations by radiography would be needed on each row of tube-to-manifold weldment as they are completed. This could be done by applying film to the manifold side of the welds with a radiation source on the other. With that, and visual examination, it is felt the full penetration requirement could be met without re-welding on the underside which would be difficult and costly. In order to insure that each tube is completely sealed and no through-cracks are missed

during NDE, each tube could be lightly pressurized and leaks detected prior to welding the next row of tubes.

After the last row of tubes is welded to the manifold and the manifold top is welded to the bottom section, the structure may be subject to hydrogen attack. This is not uncommon in high strength, stressed materials exposed to moisture. A welded structure such as the TASKA blanket will also be subject to triaxial stresses in the as-welded joints. The first row of tubes might be in the non-stress relieved condition during welding and inspection of all others. A pretreatment to reduce the strength and increase the toughness could sufficiently slow the crack growth due to hydrogen. This pretreatment may be a low-temperature heat treatment ( $< 400^{\circ}\text{C}$ ) for enough time to increase the ductility of the material significantly. The coolant tubes are placed fairly close to each other in the manifold ( $\sim 4$  cm) and therefore, the welding of the next row of tubes may provide sufficient heat input to lightly temper the previous row of weldments. This will also reduce the potential cracking before the final heat treatment.

After welding, care must be exercised in handling or transporting the blanket modules. A heat treatment is required to temper the welded joints which is traditionally a post-weld heat treatment (PWHT) of  $760^{\circ}\text{C}$  for 2.5 hrs. This can be performed by using insulated heating tapes wrapped around the weldments. However, this may be only one option. It will "fully" temper the weldment giving it superior thermal properties to the parent metal (as indicated by so few weldment failures in boiler service). Another choice is to post-weld heat treat the entire structure. This could be done using furnaces available in many steel processing plants, although heating and cooling rates would need to be carefully controlled.

There is some concern about the effect of neutron damage on weldments and heat affected zone (HAZ) microstructures which might still exist after the PWHT. The concern is primarily due to the lack of irradiated mechanical property data on weldments. Irradiated HAZ properties may not be different from the parent metal, but until that is shown by data, it is prudent to consider eliminating the weld zone by re-austenitizing the entire structure. This will have two primary benefits. The first is to eliminate the metallurgically different zones of a weld joint, while the second is to eliminate any potential problems caused by exacerbated irradiation damage in heat affected zones.

A weld joint is one which can be characterized as "as-cast", that is, the material in the fusion zone was cooled from a liquidous state such that chemical segregation will occur at dendrite structures. The "as-cast" material is not re-austenitized at  $760^{\circ}\text{C}$ , and so is retained even after the PWHT. Furthermore, delta ferrite tends to form at the dendritic boundaries (and also in the material not melted during welding immediately adjacent to the fusion line in the parent metal). The ferrite in a dual-phased structure such as this, will swell disproportionately from the martensite. This difference in swelling may promote cracking near the fusion line interface.

Other options which might be considered to avoid deleterious microstructures include re-austenitization followed by a full tempering treatment. This could be done using higher temperature heating tapes to attain the required  $1050^{\circ}\text{C}$ , but such a solution would require some development. Another approach would be to heat treat the entire structure in a large furnace typical of those in steel mills for high-temperature soaking of ingots. Large forgings have been heated to temperatures on the order of  $1000^{\circ}\text{C}$  with care-

fully controlled heating and cooling rates to minimize thermal stresses. In air, high temperatures would cause scale formation (which is tightly adherent on 12 Cr-1 Mo steel) so a protective atmosphere of argon or helium could be provided. After austenitization at 1050°C, the structure would be cooled sufficiently fast to prevent the formation of bainitic or perlitic decomposition products, yet slowly enough to minimize stresses due to the transformation to martensite.

Finally, the temperatures needed to austenitize the structure may lead to some distortion. The exact degree of this problem is not known, but manufacturers of heat treating furnaces suggest that development work on a prototype could answer the feasibility question fairly rapidly. The design of a true engineering structure will require strict tolerances to be built into the components. When these are known and when the thermal distortion is quantified, the engineering feasibility of this process will be established.

#### VI.4.10 Expected Irradiation Performance of HT-9 in TASKA

In this section we will attempt to extrapolate the data discussed in previous sections to TASKA conditions. The reader is certainly aware by now that the uncertainties in this extrapolation process are very large and certainly more experimental and theoretical data is required before one could confidently predict component lifetimes. Nevertheless, we will proceed with our projections based on what we now know.

It appears that the swelling rate in the HT-9 alloy will be in the range of  $0.01\% \text{ dpa}^{-1}$  or smaller over the 350°C-530°C operating range. At the maximum damage rate of 14.8 dpa per operating year, this amounts to a maximum swelling rate of  $\sim 0.15\%$  per year. The effect of much higher helium



concentrations on this swelling is unknown and close attention should be paid to experiments currently in U.S. mixed-spectrum neutron fission reactors.

As the high temperature ( $\sim 550^{\circ}\text{C}$ ) irradiations indicated; adequate total elongation ( $> 10\%$ ) is retained for low He levels ( $\sim 19$  appm) and modest doses ( $\sim 50$  dpa). Higher levels of He are likely to reduce ductility at temperatures above  $500^{\circ}\text{C}$ , but not at lower temperatures. On the other hand, the low temperature irradiations ( $\sim 50^{\circ}\text{C}$ ) in HFIR resulted in a more severe radiation-induced embrittlement with total elongations of the order of 5% or less, and uniform elongations around 0.5%. Fortunately, helium was not found to influence this reduction in ductility.

There is still no information on the radiation-induced embrittlement in the temperature range from  $350$  to  $450^{\circ}\text{C}$ .

The creep rates appear to be in the range of 0.2% per year and are not greatly affected by irradiation in the TASKA operating range. Since the creep rate depends on the damage rate, which is about the same in TASKA ( $0.8 \times 10^{-6} \text{ s}^{-1}$ ) as in EBR-II ( $\sim 1 \times 10^{-6} \text{ s}^{-1}$ ) where the data was taken, we feel somewhat more confident in predicting the 0.2% per year value.

The DBTT shift of ferritic steels is of concern for irradiation temperatures between  $300$  to  $400^{\circ}\text{C}$ , the temperature range for TASKA in the blanket region. The shift at the lowest irradiation temperature occurs rapidly over a few days, but then saturates at a temperature around  $200^{\circ}\text{C}$ . Saturation appears to be reached after about 0.05 to 0.1 dpa. The shift in DBTT decreases with increasing irradiation temperature. From post-irradiation annealing experiments on nuclear pressure vessel steels, irradiated at about  $300^{\circ}\text{C}$  and subsequently annealed at  $400^{\circ}\text{C}$  for 168 hours, it is found that the shift can

be reversed and the DBTT be restored to 55 to 75% of its initial value.

Figure VI.4-29 summarizes the annealing data.<sup>(34)</sup>

The implications of the DBTT shift for TASKA are the following. While the reactor is on power or the blanket coolant  $\text{Pb}_{83}\text{Li}_{17}$  above its melting point ( $\sim 280^\circ\text{C}$ ), the blanket structure remains everywhere above the DBTT. Above this temperature, the material is ductile and the fracture toughness is more than adequate.

When a blanket module is scheduled for removal, the coolant flow should be reversed for several days while the reactor remains on power. As a result of this operation, the parts of the blanket structure previously exposed to a low irradiation temperature will recover, while the parts previously exposed to the high temperature will experience now some shift in the DBTT. However, the overall effect on the entire blanket structure should be beneficial. Although the DBTT in thick section specimens will most likely remain above room temperature in spite of the annealing procedure, it is expected that the DBTT for the thin walled tubes used in the TASKA blankets will indeed be below room temperature.

Figure VI.4-30 shows the effect of the section thickness of fracture specimens on the DBTT for types 409, 437, and other proprietary ferritic steels.<sup>(35)</sup> It is seen that the DBTT is lower by about  $100^\circ\text{C}$  for a thickness of 2.5 mm as compared to the conventionally thicker specimens of 1 cm or larger.

Since the wall thickness of the blanket coolant tubes is 2.77 mm, the DBTT for the blanket structure is expected to be at least  $50^\circ\text{C}$  lower, if not more, than the present data indicates. Therefore, DBTT shift may not present

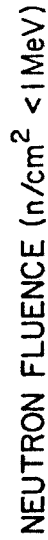
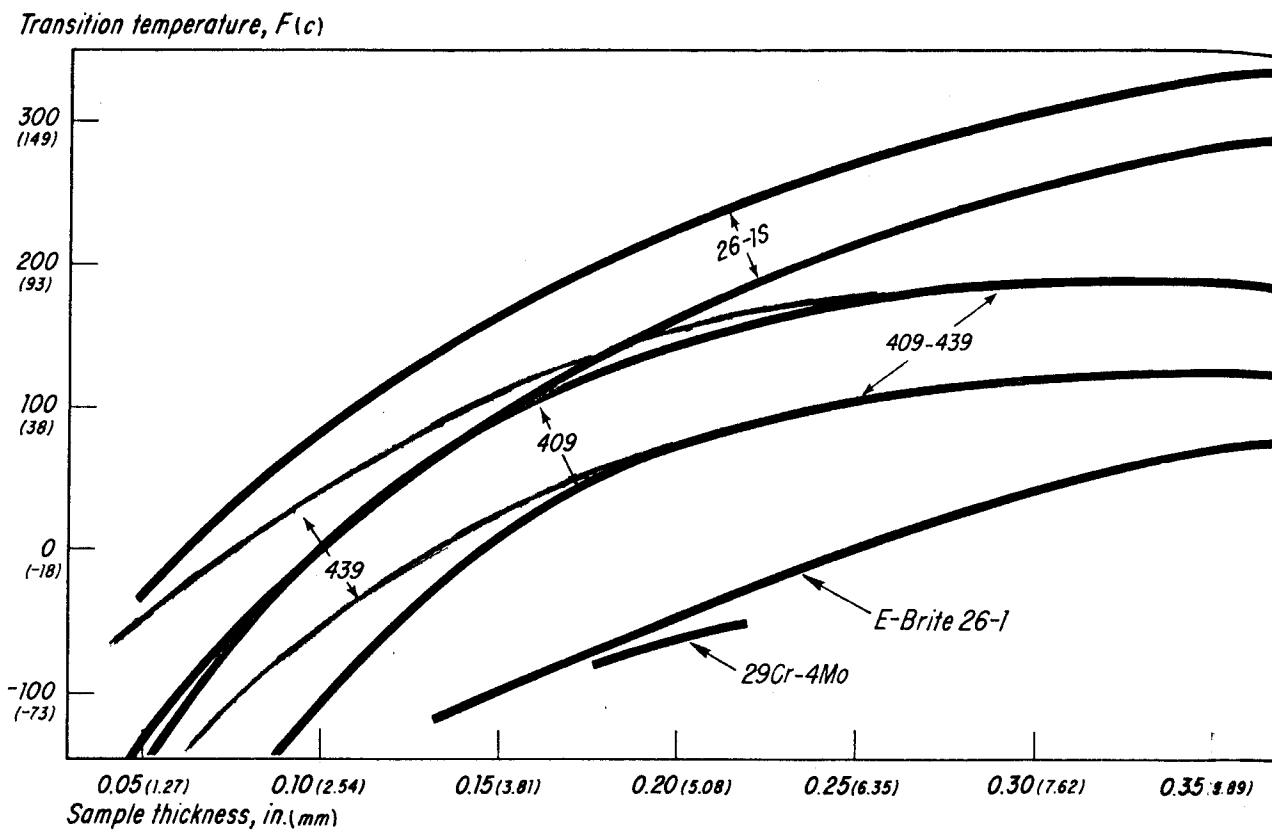


Fig. VI.4-29 Summary of transition-temperature changes (41-J index) observed for several steels with 288 C (550 F) irradiation (open symbols) and with 399 C (750 F) postirradiation heat treatment (filled symbols). Limited data for 371 C (700 F) and 427 C (800 F) annealing are also shown.

Fig. VI.4-30



Ductile-to-brittle transition temperatures (DBTT) for ferritic stainless steels rise with section thickness. Bands for 26-1S, 409, and 439 indicate data scatter.

any restriction on the use of ferritic steels in fusion reactors, and TASKA in particular.

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## VI.5 Tritium Extraction and Purification

This chapter is divided into two parts. Section VI.5.1 outlines the tritium parameters in the  $\text{Li}_{17}\text{Pb}_{83}$  eutectic, the blanket, the heat exchange cycle which includes an organic intermediate loop and discusses the tritium extraction scheme. Section VI.5.2 describes the TASKA fuel purification systems.

### VI.5.1 Blanket and Heat Exchange Systems

In TASKA the main reactor breeding blanket consists of HT-9 tubes through which the liquid lithium-lead breeding material is circulated. The lithium-lead is a eutectic of composition 17 atom % Li and 83 atom % Pb which melts at  $235^{\circ}\text{C}$ . The  $\text{Li}_{17}\text{Pb}_{83}$  is operated at temperatures between 300 and  $400^{\circ}\text{C}$  and serves as the heat transfer agent to an organic intermediate loop. The blanket and coolant characteristics in the primary and intermediate loop are summarized in Tables VI.5-1 and VI.5-2.

The low solubility of tritium in  $\text{Li}_{17}\text{Pb}_{83}$  results in low inventories and presents the possibility of extracting the tritium by allowing it to permeate through the breeding blanket tubes. This in-situ extraction method is feasible if an appropriate tritium barrier is present on the organic coolant side of the heat exchange loop.

#### VI.5.1.1 Blanket Inventory

In order to determine the tritium inventory in the breeding material the solubility of tritium in  $\text{Li}_{17}\text{Pb}_{83}$  at the reactor temperatures ( $300\text{--}400^{\circ}\text{C}$ ) must be known. The solubility of hydrogen in dilute solutions is expressed by the Sievert's constant ( $K_s$ ) where:



Table VI.5-1. Characteristics of the Primary Heat Exchange Cycle

Blanket Parameters:

|                |                     |
|----------------|---------------------|
| Structure      | HT-9                |
| Tube Thickness | .227 cm             |
| Temperature    |                     |
| High           | 440°C               |
| Low            | 300°C               |
| Average        | 370°C               |
| Surface Area   | 1234 m <sup>2</sup> |

Primary Coolant:

|                                |                                   |
|--------------------------------|-----------------------------------|
| Composition                    | Li <sub>17</sub> Pb <sub>83</sub> |
| Temperature                    |                                   |
| High                           | 400°C                             |
| Low                            | 300°C                             |
| Average                        | 350°C                             |
| Li Vapor Pressure (350°C)      | $1.2 \times 10^{-10}$ torr        |
| Pb Vapor Pressure (350°C)      | $1.3 \times 10^{-8}$ torr         |
| Density (350°C)                | 9.46 g/cm <sup>3</sup>            |
| Volume in blanket              | 39.3 m <sup>3</sup>               |
| Mass (Blanket and Ext. Piping) | $5.7 \times 10^5$ kg              |
| Flow Rate                      | $2.9 \times 10^6$ g/sec           |

Table VI.5-2. Characteristics of the Intermediate Loop

IM Loop:

|                |                    |
|----------------|--------------------|
| Composition    | HT-9               |
| Tube Thickness | .227 cm            |
| Temperature    |                    |
| High           | 385°C              |
| Low            | 285°C              |
| Average        | 335°C              |
| Surface Area   | 895 m <sup>2</sup> |

Intermediate Coolant:

|             |                          |
|-------------|--------------------------|
| Composition | HB-40                    |
| Temperature |                          |
| High        | 371°C                    |
| Low         | 271°C                    |
| Average     | 321°C                    |
| Mass        | 1.4 x 10 <sup>4</sup> kg |
| Flow Rate   | 169.1 kg/s               |

$$K_s = \frac{N}{p^{1/2}}$$

where  $N$  = the concentration of hydrogen in the alloy (appm), and

$P$  = the hydrogen pressure above the alloy (torr).

The Sievert's constant for deuterium in Li-Pb alloys has been determined by Ihle et al.<sup>(1)</sup> at 677 and 767°C, much higher temperatures than the reactor conditions. The accuracy of this data is questionable and direct extrapolation to lower temperatures results in unreasonable solubility values. The solubility has also been predicted from models.<sup>(11)</sup> E. Veleckis<sup>(2)</sup> at Argonne National Laboratory is conducting Sievert's constant experiments on lithium-lead alloys in the temperature range of 400 to 600°C. The value obtained for hydrogen in  $\text{Li}_{17}\text{Pb}_{83}$  is  $(7.1 \pm 1.5) \times 10^3 \text{ atm}^{1/2} (\text{at.fr.H})^{-1}$  which corresponds to 5.1 appm H/torr<sup>1/2</sup>. Preliminary results also indicate that in this temperature range the Sievert's constant is independent of temperature within experimental error.

The low value and temperature independence of the solubility seem to be consistent with the thermodynamic and physical properties of the lithium-lead system. A comparison of the hydrogen solubility of pure lithium<sup>(3)</sup> and lead<sup>(4)</sup> to that of other metals<sup>(5,6,7)</sup> (Fig. VI.5-1) shows that lithium has one of the highest hydrogen solubilities of any known metal while lead has one of the lowest hydrogen solubilities. Therefore, when hydrogen dissolves in the alloy it interacts exclusively with the lithium, because lead-hydrogen interactions are weak. The degree to which the lithium is available for bonding is related to the activity coefficient for lithium in lithium-lead. Activity measurements on the alloy<sup>(1,8,9,10)</sup> indicate very low activities of

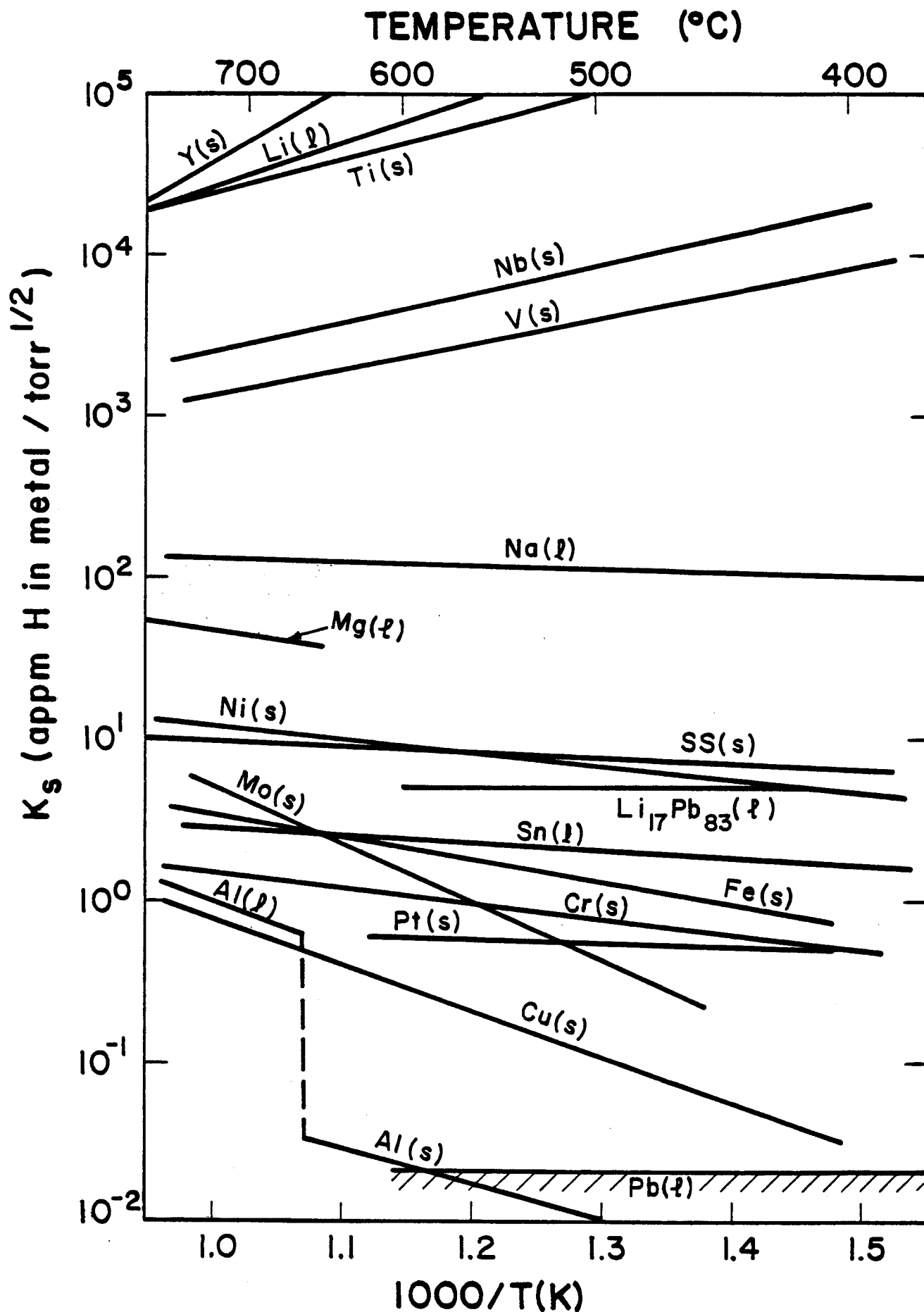


Figure VI.5-1. THE SOLUBILITY OF HYDROGEN IN SELECTED METALS AND ALLOYS AS A FUNCTION OF TEMPERATURE.

lithium in the  $\text{Li}_{17}\text{Pb}_{83}$  eutectic ( $1.3 \times 10^{-4}$  at  $500^\circ\text{C}$ ) which predicts low solubilities.

The temperature independence also appears reasonable on examination of Fig. VI.5-1. Lithium dissolves hydrogen exothermically while metals with low solubilities (such as lead) generally dissolve hydrogen endothermically. This probably leaves Li-Pb alloys in an intermediate category showing little temperature dependence.<sup>(11)</sup>

For TASKA, the Sievert's constant for hydrogen ( $5.1 \text{ appmH/torr}^{1/2}$ ) is converted to tritium units by assuming the  $\sqrt{3}$  dependence as observed in the pure lithium hydrogen isotope solubility studies. The resulting value is

$$0.051 \frac{\text{wppm T in Li}_{17}\text{Pb}_{83}}{\text{torr}^{1/2}} .$$

To determine the blanket inventory the tritium pressure above the coolant must be specified. The pressure required for the tritium to be extracted at the breeding rate can be calculated from the permeability of tritium through HT-9<sup>(12)</sup> given in Eq. (VI.5-1):

$$P = \frac{65.3}{\sqrt{3}} \exp \left( \frac{-11100}{RT} \right) \frac{\text{mol T}_2 \cdot \text{mm}}{\text{d} \cdot \text{m}^2 \cdot \text{torr}^{1/2}} . \quad (\text{VI.5-1})$$

For the TASKA parameters:

$$P = 1.3 \times 10^5 \text{ Ci/d (breeding rate)}$$

$$T = 350^\circ\text{C}$$

$$\text{Area} = 1234 \text{ m}^2$$

$$\text{Thickness} = 0.227 \text{ cm,}$$

the resulting pressure is 0.39 torr. At a pressure of 0.39 torr the solubility is 0.032 wppmT in  $\text{Li}_{17}\text{Pb}_{83}$ . For a total Li-Pb mass of  $5.7 \times 10^5$  kg, the tritium inventory in the blanket is 18.2 g T.

At this high tritium pressure, we also need to determine the tritium inventory in the structural material. The hydrogen solubility in HT-9<sup>(12)</sup> is given by Eq. (VI.5-2):

$$K_s = \frac{2.3 \times 10^{-4} \exp \left( \frac{-7500}{RT} \right)}{\sqrt{3}} \frac{\text{mol}_2\text{H}}{\text{cm}^3 \text{ metal} \cdot \text{atm}^{1/2}} \quad (\text{VI.5-2})$$

At 370°C and 0.39 torr for 3.84 m<sup>3</sup> of metal the inventory is 0.12 g T. The tritium blanket parameters are summarized in Table VI.5-3.

#### VI.5.1.2 Tritium Extraction

From the previous section it is apparent that with the low Sievert's constant the tritium pressure can be allowed to build up to a fairly high pressure (0.39 torr) and still maintain a relatively modest blanket inventory (18.32 g T). A way to take advantage of this phenomenon is to allow the tritium to permeate through the blanket tubing into the reactor vessel<sup>(13)</sup> where it will be pumped and recovered along with the exhaust gases. This scheme necessitates an effective containment scheme to prevent losses to the heat exchanger. For TASKA, the containment is provided by the use of an organic coolant in the intermediate loop (IM loop) of the heat exchange cycle. Tritium permeating into the organic will exchange slowly with hydrogen and be effectively contained. The flux of tritium into the organic must be minimized to prevent gross decomposition of the coolant.

The tritium permeation into the organic IM loop can be calculated from Eq. (VI.5-1) assuming the same wall thickness, temperature, and tritium

Table VI.5-3. Blanket Tritium Parameters

|                                                         |                               |
|---------------------------------------------------------|-------------------------------|
| Breeding Ratio                                          | 1.0                           |
| Breeding Rate                                           | $1.3 \times 10^5$ Ci/d        |
| Tritium Flow Rate                                       | $9.28 \times 10^{-2}$ g T/sec |
| Tritium Partial Pressure (@ 350°C)                      | 0.39 torr                     |
| Tritium Concentration in $\text{Li}_{17}\text{Pb}_{83}$ | 0.032 wppm                    |

Blanket Tritium Inventory:

|                                        |               |
|----------------------------------------|---------------|
| $\text{Li}_{17}\text{Pb}_{83}$ Coolant | 18.2 g        |
| HT-9 Structure                         | <u>0.12 g</u> |
| Total:                                 | 18.32 g       |

pressure assuming no tritium barrier is present. A barrier that will provide a factor of ~ 150 or more reduction in tritium permeation is necessary to reduce the permeation to less than 500 Ci/d. The tritium permeation should be minimized to prevent high decomposition rates in the organic.

#### VI.5.1.3 Tritium Barriers

Tritium barriers are commonly invoked to reduce permeation to the steam cycle. For example in STARFIRE<sup>(14)</sup> a barrier factor of 150 is used for both the 316 SS blanket (500°C) and the Inconel (300°C) in the steam generator. The most common barrier is the oxide coating that is present on the steam-side of the heat exchange tubing. Table VI.5-4 illustrates the magnitude of permeation barriers for steels at 660°C. A factor of 150 for a barrier seems reasonable.

There are many factors that affect the magnitude of the oxide barrier. The oxidation temperature is important. Stainless steel membranes oxidized at 750°C for 200 hrs had 20 times lower permeation rates than those oxidized at 550°C for an equivalent time period.<sup>(17)</sup> Surface analyses revealed that Mn and Cr oxides formed on the surface of the high temperature membranes and Fe oxides have been formed at lower temperatures. After prolonged operation, 6000 to 24000 hrs at 500 to 600°C, Mn/Cr oxide coatings were observed at HEDL. These data imply that for steam generators running at low temperatures (<500°C), oxide coatings may have to be preformed at higher temperatures to obtain the more resistant barrier. Studies also show that the permeability does not necessarily decrease linearly with oxide thickness.<sup>(18)</sup> Thick oxides (>1  $\mu\text{m}$ ) often showed permeabilities with activation energies close to that of the substrate. This indicates that cracks or defects form in the oxide which permit hydrogen access to metal surfaces.<sup>(18,19)</sup>



Table VI.5-4. Permeation Barrier Factors of Ferritic and Austenitic  
Steels Produced by Steam Oxidation of the Downstream Side

|                      | Ferritic<br>Fe-2-1/4 Cr-1 Mo | Ferritic<br>SS 406 | Austenitic<br>SS 316 |
|----------------------|------------------------------|--------------------|----------------------|
| Reference            | 16                           | 15                 | 15                   |
| Temperature (°C)     | 472                          | 660                | 660                  |
| Steam Pressure (atm) | 2                            | 0.94               | 0.94                 |

Factors by Which Permeabilities are Reduced

Time

|        |     |     |     |
|--------|-----|-----|-----|
| 0.33 d | 10  | 13  | --- |
| 1 d    | 25  | 72  | 1.6 |
| 6 d    | 100 | 144 | 2.8 |
| 24 d   | 170 | 291 | 3.8 |
| 40 d   |     | 434 | 4.3 |

Experiments with palladium over oxides indicate that a Pd coating on the oxide has an effect on the permeability on the upstream face, but that very little difference is observed if the downstream face is coated.<sup>(18)</sup> This suggests that surface effects on the upstream side are rate controlling. The difference between the two measurements, however, was less than a factor of two.

The liquid-metal side of the steam generator tubing is not capable of maintaining an oxide coating. A nickel-aluminide layer has been developed which shows a reduction in diffusion rate of two orders of magnitude for 304 SS at 550°C and 10 torr H<sub>2</sub>.<sup>(17)</sup> The pressure dependence of the diffusion rate appears to be linear. This indicates that for the low tritium pressure expected in fusion heat exchangers, the Ni-Al barrier will lower the permeation by several orders of magnitude. This barrier can be formed in-situ by addition of Al to the coolant and it is resistant to attack by hot sodium and lithium. In TASKA, the blanket and heat exchange material is HT-9, which is very low in nickel and not expected to form the Ni-Al layer. To form this barrier a different structural material with a higher nickel concentration would have to be chosen for the heat exchanger. The resistant coating would then form on the heat exchange tubing while a low Ni alloy would be used in the blanket to allow permeation. Experimentation is needed to verify the barrier effectiveness at low pressures.

Multilayered metal laminates containing at least one layer with a relatively low hydrogen permeability have been studied<sup>(20)</sup> and the use of doubled walled tubes with an oxidizing purge gas in the gap<sup>(21)</sup> have been proposed. The cost and feasibility of constructing these types of special tubing may be prohibitive. The use of double walled tubes is an especially effective

tritium barrier, as tritium in the gap is converted to the oxide, reducing the HT partial pressure to  $\sim 10^{-16}$  torr.

The use of  $\text{Li}_{17}\text{Pb}_{83}$  as a coolant in any design will result in a high tritium partial pressure and thus a containment problem. For TASKA, the organic IM loop acts as a tritium getter and, therefore, tritium permeating to the organic will not be released to the environment. However, the permeation into the organic must be minimized to prevent decomposition. Oxide coatings have been shown to be effective barriers, reducing permeation by 2 to 3 orders of magnitude. A conservative barrier assumption of 150 will result in 370 Ci/d lost to the IM loop.

#### VI.5.1.4 Organic Intermediate Loop

The intermediate loop contains a mixture of partially hydrogenated terphenyls with the proprietary name of the Monsanto Company HB-40. The coolant is a pale yellow liquid at room temperature and has the characteristics of a light lubricating oil. After exposure to heat and radiation the coolant is opaque black and contains  $\sim 30\%$  high boiling (HB) decomposition products as well as  $\sim 1\%$  hydrogen and gaseous hydrocarbons and  $\sim 5\%$  volatile ( $\text{C}_6\text{-C}_{12}$ ) hydrocarbons. The AECL (Atomic Energy of Canada Limited) has done considerable work in the development and use of organic terphenyls as heat transfer media for nuclear reactors. In particular, the AECL has over 8 years of operating experience with a heavy water moderated, organic cooled reactor, WR-1, at the Whiteshell Nuclear Research Establishment. There is a vast amount of literature available on the Canadian experience (see Summary Report, Ref. 22). In this section a few of the characteristics of the organic that impact the tritium processes are outlined.

It is difficult to ascertain the effects of tritium leakage into the organic. The organic decomposes in both radiation and thermal environments and the coolant is periodically vented and partially replaced. The organic that is removed from the IM loop and the vented gases can be oxidized to  $\text{CO}_2$  and  $\text{H}_2\text{O}$  and the isolated tritiated water is then sent to a water recovery unit. The thermal decomposition rate of the organic is 1.4 kg/hr. If the tritium intake for the primary coolant is 370 Ci/d and the coolant inventory is  $1.4 \times 10^4$  kg HB-40, then the tritium concentration in the coolant is 11 Ci/kg and the total inventory in the IM loop is 15g T.

The terphenyls are also known to form fouling films on heat transfer surfaces. The effect of these films on the tritium permeation and the oxide barrier is unknown. However, experience with organic coolants and zirconium alloys has shown that if 50 mg water/g is added to the coolant the fouling is reduced and the integrity of the oxide layer is maintained. The organic must also be sent through purification cycles to remove any chlorine species originally present in the organic. Chlorine can be absorbed into the oxide film and can destroy its protective characteristics. Therefore, by careful chemical control of chlorine species and addition of water, it is felt that the oxide barrier can be maintained in the presence of fouling films.

An initial analysis indicates that the organic in the intermediate loop may serve as a possible tritium getter in the steam cycle. The radiolytic damage due to tritium decay must be quantified and the safety aspects involved in using a flammable organic should be addressed.

#### VI.5.2 Purification Systems

The design of the purification systems for TASKA is patterned after the Tritium Systems Test Assembly (TSTA).<sup>(23-25)</sup> The philosophy of the TASKA

purification system is to provide gases for the neutral beam lines free of impurities (O,N,C) in order to maintain the predicted efficiencies and component lifetimes of the injection system and getter panels, to provide D-T fuel of the appropriate mixture, to limit tritium losses and to attempt to produce a simplified yet efficient design.

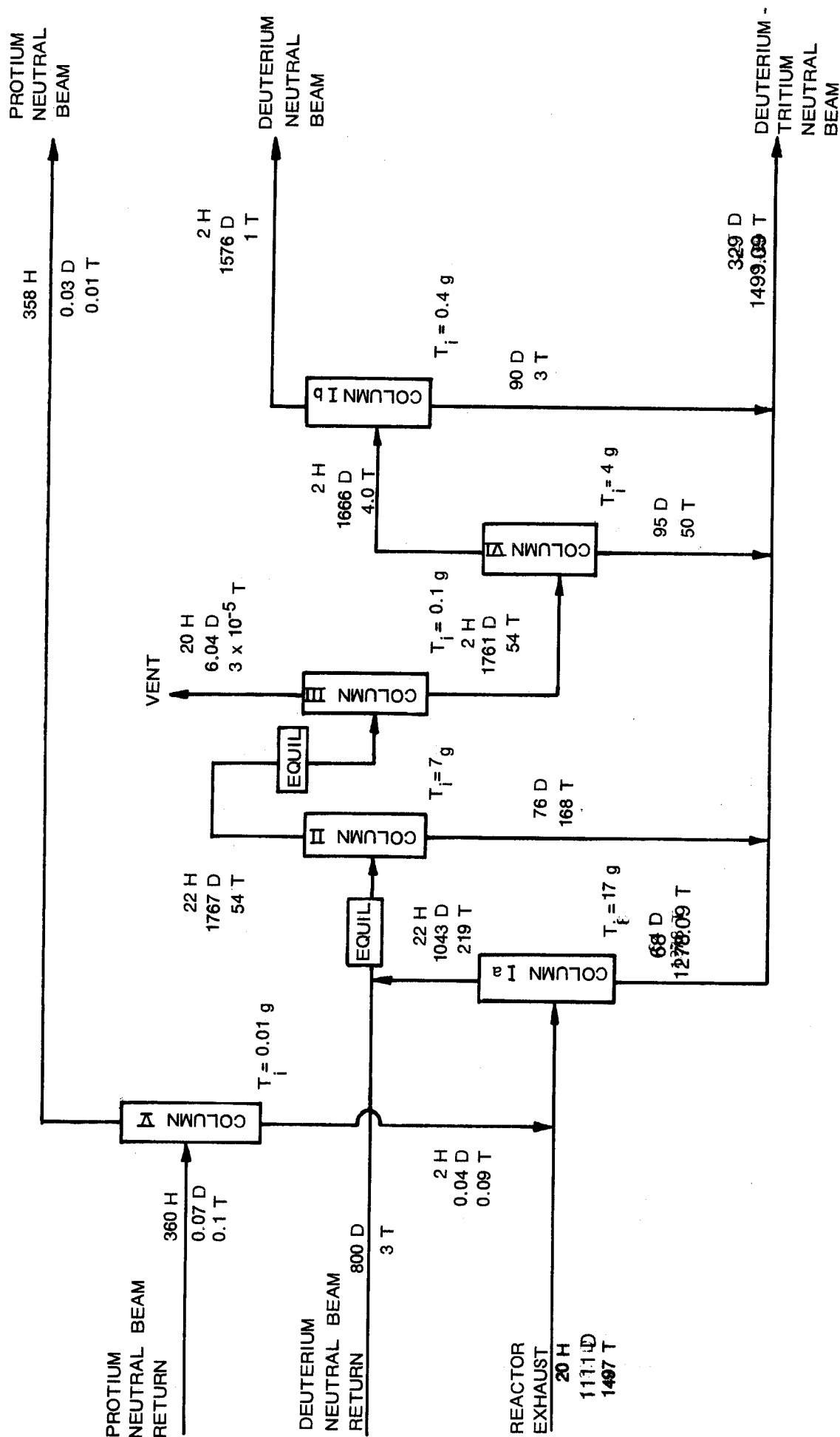
#### VI.5.2.1 Purification of NB Recycle Gases Deuterium and Proton NBI

The gases that are regenerated from the Zr-Al getter panels in the deuterium and proton NB lines are expected to contain small amounts of tritium impurities. It is possible to recycle only a fraction of these beam lines and to let the tritium level build up to a certain maximum (~5%). Although this would have a very small impact on the tritium inventory, it would increase losses to the dump coolants. In order to minimize these losses, the entire gas load is continuously sent to the reactor isotopic separation unit (ISU). The Zr-Al panels are effective getters for oxygen, nitrogen and CO, and somewhat less effective getters for water and hydrocarbons. These impurities are chemically trapped in the getter. No re-emission of N<sub>2</sub>, CO, CO<sub>2</sub>, or O<sub>2</sub> occurs at regenerations below 1100°C.<sup>(26)</sup> Therefore, the recycled isotope gases can be sent directly to the cryogenic systems where helium is removed and then to the reactor ISU. Alternatively, the gases can pass through liquid nitrogen cooled charcoal beds before being sent to the cryogenic system if further purification proves necessary. The deuterium recycle from the LE-NB (low energy-neutral beam) and the HE-NB (high energy-neutral beam) are combined and the proton stream is kept separate. These two streams are injected into the isotopic separation columns as shown in Fig. VI.5-2. Purified gases (~99.9%) are returned to the NBI systems.

Fig. VI.5-2

TASKA ISOTOPIC SEPARATION UNIT

(Flow rates in g/d)



The mixed DT beam will contain helium as its main impurity after being released from the getter panels. Hydrogen impurities from backstreaming are expected to be very small. A small side stream of the DT gas can be sent to the ISU periodically if there is any hydrogen buildup. It is unnecessary, in general, to send the recycle gases from this beam to the ISU.

The DT recycle gases can be purified simply by sending them over uranium beds to absorb the deuterium and tritium and separate helium. The purified DT recycle stream receives purified gases from the ISU to achieve the appropriate fuel mixture. The inventory in the DT fuel cleanup unit is approximated as 33 g T.

#### VI.5.2.2 Purification of Reactor Exhaust

The reactor exhaust is composed of unburned and bred tritium (1500 g/d), unburned deuterium (1111 g/d), hydrogen from the plug neutral beam (20 g/d), helium (35 g/d), and impurities produced in the harsh reactor environment.

The hydrogen isotopes and helium are pumped from the surface of the thermal dump with compound cryopumps. It has been demonstrated that a compound pump can simultaneously pump hydrogen isotopes and helium. The separation of the hydrogen on the condensation panel from helium on the cryosorption panel is extremely sharp. Thus, separation of the hydrogen isotopes from the helium ash is accomplished at this step. The helium panel is first regenerated, after which the cryocondensation panel is warmed to remove the hydrogen isotopes from the pump.<sup>(24)</sup> The hydrogen isotopes are then sent to the fuel cleanup unit.

The fuel cleanup unit serves to purify the hydrogen isotopes from any tritiated water, methane, ammonia, argon or any other species which would freeze out at the liquid hydrogen temperatures employed in the isotope

separation system.<sup>(23-25)</sup> Impurity levels must be maintained below 1 ppm to ensure safe, long-term operation of the isotope separation system.<sup>(25)</sup>

The tritiated water and ammonia are condensed out of the hydrogen isotope stream, after which the low melting impurities are separated from the hydrogen molecules by adsorption on molecular sieve beds at 75 K. The recovered impurities are then catalytically oxidized to form tritiated water and tritium-free compounds. The water is finally condensed out and electrolyzed to recover the tritium.<sup>(25)</sup> For a 500 mole T/day flow rate TSTA has estimated an inventory of 14 g.<sup>(23)</sup> In TASKA the exhaust cycles 1500 g T/d (500 moles T/d) and should, therefore, have the same inventory.

From the fuel cleanup unit the gases are sent to cryogenic distillation (Fig. VI.5-2). Cryogenic distillation columns are used to separate the hydrogen isotopes into 4 streams: HD waste which is vented to the atmosphere, high purity  $D_2$  for the deuterium neutral beam injector, D-T stream for the use in the DT-NBI and high purity  $H_2$  for use in the end plug neutral beam injectors. The cryogenic distillation columns are modeled on the separation scheme used in WITAMIR<sup>(12)</sup> with minor modifications. Since the breeding ratio in TASKA is 1.0, minor redistribution of tritium into the pure deuterium gas flow from the columns has two consequences: (1) a small amount of tritium must be supplied to the DT beams or (2) assume that the tritium in the deuterium beams will fulfill the fueling requirements. A sixth column is used to reduce the amount of tritium sent to the deuterium beam lines. A breeding ratio of  $\sim 1.1$  would compensate for this small redistribution effect. The tritium inventory in the columns is estimated as 28 g.



### VI.5.3 Summary of the Total Tritium Inventory and Comments

The overall hydrogen isotope pathways and the total tritium inventory in the fuel cycle (IV.5.2) and the blanket and purification systems are given in Fig. VI.5-3 and Table VI.5-5. The total inventory is 5347 g T ( $5.3 \times 10^7$  Ci).

Further investigation for TASKA should include details of the water recovery unit and the secondary and tertiary containment systems.

Fig. VI.5-3

# HYDROGEN ISOTOPE PATHWAYS IN TASKA

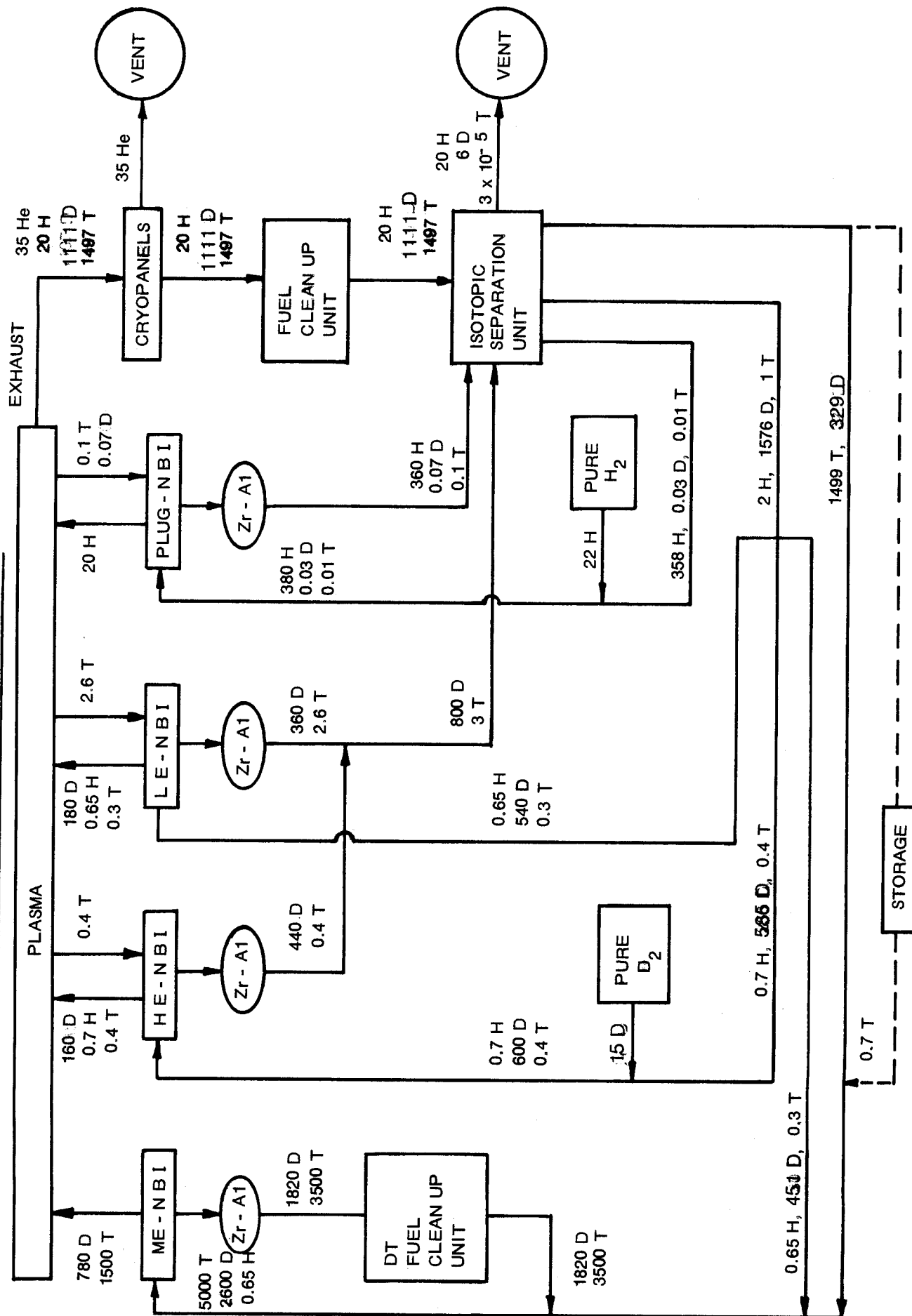


Table VI.5-5. Total TASKA Tritium Inventory

|                                   |      |             |
|-----------------------------------|------|-------------|
| Fuel and Exhaust Cycle (g)        |      | 385         |
| NBI Getter Panels                 | 259  |             |
| Cryopanels                        | 125  |             |
| Dumps                             | 0.52 |             |
| Blankets (g)                      |      | 18.32       |
| Li <sub>17</sub> Pb <sub>83</sub> | 18.2 |             |
| HT-9                              | 0.12 |             |
| Organic IM loop (g)               | 15.0 | 15.0        |
| Purification (g)                  |      |             |
| Fuel Cleanup                      |      | 75.0        |
| DT-NBI Recycle                    | 33.0 |             |
| Exhaust Recycle                   | 14.0 |             |
| Cryogenic Distillation            | 28.0 |             |
| Subtotal                          |      | 493         |
| Storage (g)                       | 5000 | <u>5000</u> |
| Total                             |      | 5493        |

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## VI.6 Heat Removal Cycle

### VI.6.1 Introduction

The heat removal cycle for TASKA has the primary objective of demonstrating practical power production while operating with the highest possible reliability together with the desired flexibility required for an experimental facility.

While meeting the above requirements has top priority, it appears (whenever possible and practical) that the system should have features which will provide valuable experience needed to produce better, more efficient power cycle designs for later generations of TASKA (emphasizing power production to a greater degree). It is also desirable to identify power cycle refinements applicable to these later generation plants so that component design and configuration options available in the TASKA cycle can be used intelligently.

The purpose of this subsection is to describe the proposed 45 MW<sub>t</sub> power module cycle and its major components with the above objectives in mind and then to identify unique power cycle refinements suitable for later generations of this concept.

### VI.6.2 TASKA Power Cycle

The power cycle temperatures are set to comply with the steam generating system temperature diagram shown in Fig. VI.7-2. Though not fully optimized, these temperatures appear to provide the following:

- . A 29°C (52°F) log mean temperature difference in the intermediate heat exchanger. This appears to be a practical value from the standpoint of limiting IHX heating surface (a tritium diffusion path) and

retaining sufficient  $\Delta T$  for a power cycle similar to those used for pressurized water reactors.

- . A 138°C (280°F) final feedwater temperature. This lower than normal temperature will allow steam generation at 6.55 MPa (950 psia) using a pinch point  $\Delta T$  of 13.9°C (25°F). This steam pressure is typical of pressurized water reactor power cycles.

This is desirable since it makes much PWR steam plant technology available to the project in fields such as water chemistry, steam turbine designs, steam generator designs, etc. In later designs, the lower feed temperature used to reach the desired steam pressure can be avoided by methods described later in this subsection.

- . A steam temperature of 343°C (650°F) making full use of the HB-40 temperature (max.) of 371°C (700°F) and capable of providing improved PWR power cycle performance. A simplified turbine schematic is shown in Fig. VI.6-1.
- . A moisture separator reheater shown here is optional. Its use will bring the cycle closer to preferred PWR cycle operating conditions and experience.

The steam generating equipment (see Section VI.7) will consist of separate heat exchanger units for the pre-boiler (economizer), evaporator and superheater. This allows use of more reliable heat exchanger configurations which, in addition, are required for the more advanced cycle features to be described below in this subsection.

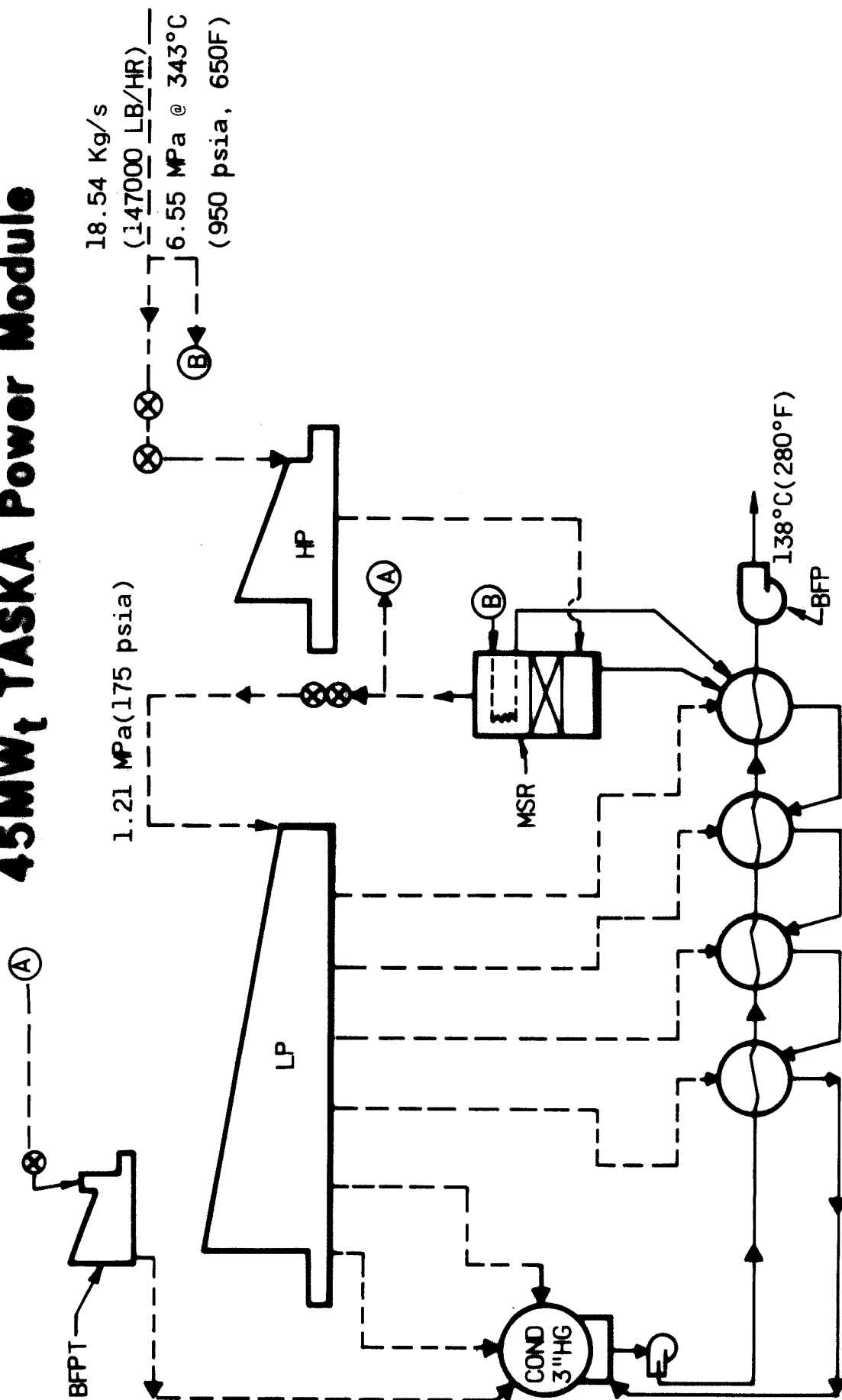
### VI.6.3 Future Steam Cycle Improvements

It is believed desirable to identify methods of improving steam cycle efficiency applicable to later versions of TASKA to help confirm the organic



# Turbine Power Cycle Schematic for

## 45MW<sub>t</sub> TASKA Power Module



cooled intermediate loop as an attractive approach to fusion power reactor design.

Heat made available from a fusion or fission source at relatively low temperatures poses various problems in effectively using a suitable power cycle (Rankine) to achieve adequate thermal efficiency.

In the study to increase efficiency, it was assumed that PbLi and HB-40 maximum loop fluid temperatures could each be increased about 50°C above the values assumed for TASKA. These hot and cold PbLi temperatures would be 450° (842°F) and 350°C (662°F), respectively. Similarly, HB-40 hot and cold temperatures become 421°C (790°F) and 321°C (610°F). These improved system temperatures will permit use of the power cycles listed in Table VI.6-1.

The cycle shown in the first column would use heat exchange components arranged in a manner typical of present day PWR plants. There is no HB-40 heated pre-boiler or superheater. Thus, the HB-40 would be cooled from 421° C to 321°C in the evaporator. All heat would then be transferred at saturated steam temperature. This would put the evaporator pinch point  $\Delta T$  at the cold end of the temperature diagram so that the permissible steam pressure would be about 8.28 MPa (1,200 psia). Such a power cycle would yield a thermal efficiency of about 35 per cent at the generator terminals in a large plant (1,000 MWe).

In PWR practice, a "once-through" steam generator capable of providing pre-boiler and superheat surface (as well as evaporator surface) will produce a temperature diagram in which the pinch point (at the low temperature end of the evaporator surface) will move to the right. This effect can be realized more fully with separate heat exchangers for the pre-boiler, evaporator, and

# Applicable STM Cycles for Advanced TASKA

| CYCLE PARAMETERS                  | SINGLE STM HTD RH |                | SINGLE STM HTD RH |              | DOUBLE HB-40 HTD RH |                |
|-----------------------------------|-------------------|----------------|-------------------|--------------|---------------------|----------------|
|                                   | NO ECONOMIZER     | NO SUPERHEATER | HB-40 HTD ECON.   | HB-40 HTD SH | HB-40 HTD SH        | HB-40 HTD ECON |
| ● STEAM PRESS @ T.T. ~ Mfa (psia) | 8.27, (1200)      |                | 12.41, (1800)     |              | 16.55 (2400)        |                |
| ● FINAL FEED TEMP. ~ °C (°F)      | 227, (440)        |                | 227, (440)        |              | 246, (475)          |                |
| ● STEAM TEMP. @ T.T. °C(°F)       | 297, (567)        |                | 399, (750)        |              | 385, (725)          |                |
| ● RH NO. 1 STM. TEMP. °C(°F)      | 282, (540)        |                | 313 (595)         |              | 343, (650)          |                |
| ● RH NO. 2 STM. TEMP. ~°C(°F)     | -                 |                | -                 |              | 343 (650)           |                |
| ● EST. CYCLE EFF (NET @ GEN)      | -35               |                | -37               |              | -39                 |                |

Table VI.6-1

superheater functions. Consequently, a scheme such as this was selected for the TASKA facility and is shown in Fig. VI.6-1.

When applied to the HB-40 temperature levels used in Table VI.6-1, it is possible to increase steam pressure to 12.41 MPa (1,800 psia) as shown in the second column of this table. These improved steam conditions produce a rise in plant thermal efficiency of two percentage points to 37 per cent as shown.

Past work on steam cycles at B&W has shown that steam pressure can be raised further if the pre-boiler temperature curve slope can be decreased so that the pinch point is moved further to the right of the temperature diagram. It was found that reheat steam conditions could be slightly adjusted so that reheat begins at a temperature within the pre-boiler water temperature range (final feed to saturation). If some HB-40 is used to perform this reheat function, there will be less HB-40 available for the pre-boiler. This reduced HB-40 flow in the pre-boiler will reduce the slope of the corresponding water temperature curve and push the pinch point to the right on the temperature diagram, thereby allowing an increase in steam pressure.

To maximize this effect in a steam cycle suitable for future power producing versions of TASKA, it was decided to use a double reheat power cycle as shown in Fig. VI.6-2 with characteristics as listed in Column 3 on Table VI.6-1. This should permit an unprecedented thermal efficiency rise to about 39 per cent and a steam pressure of 16.55 MPa (2,400 psia).

Figure VI.6-3 shows the required arrangement of pre-boiler and reheater stages and stage sections. This more complicated arrangement should be justified by the resulting increase in plant MWe output, especially in view of the high balance of plant capital costs that are apparently characteristic of fusion power systems.

# Proposed Future TASKA Power Cycle and Temperature Diagram

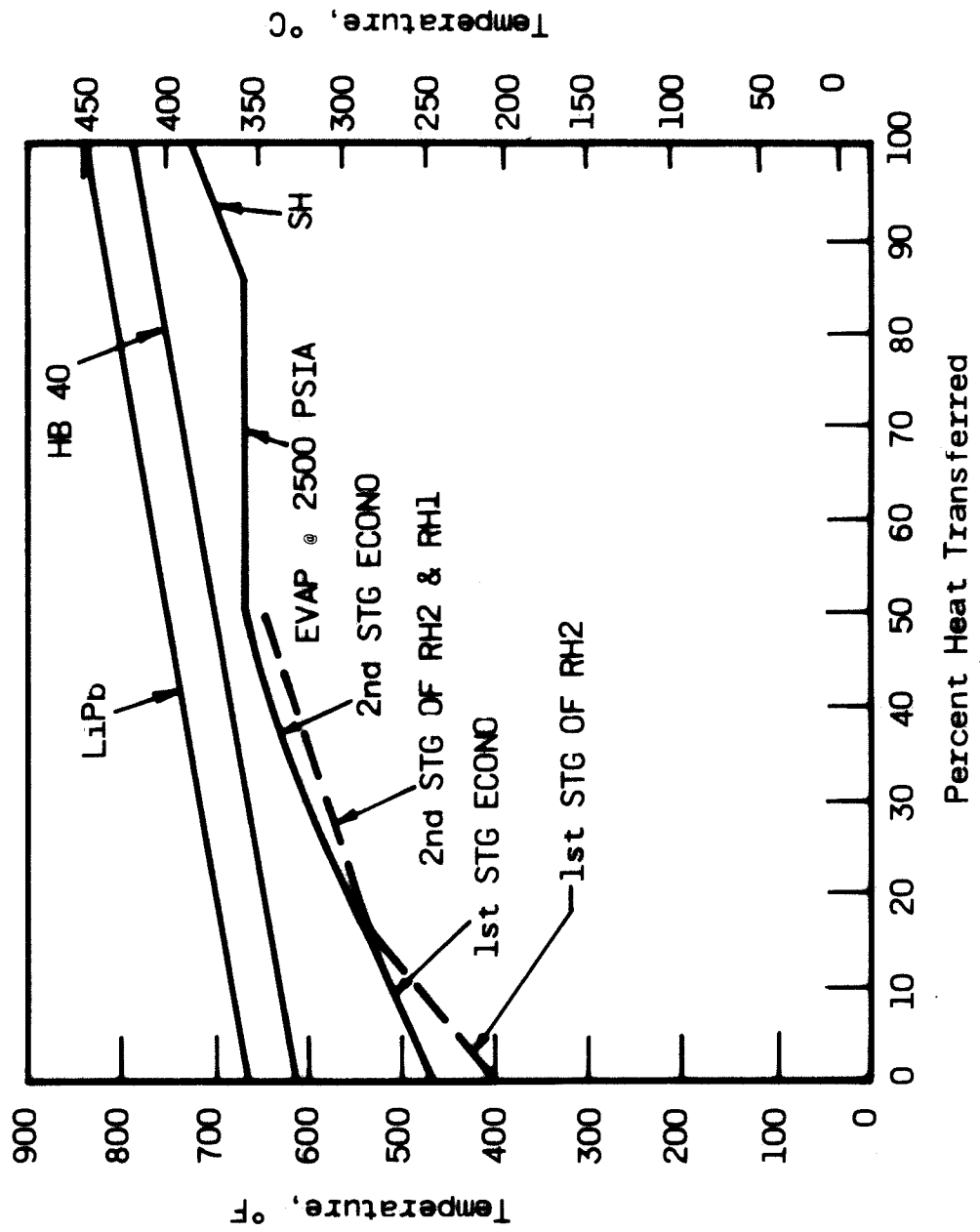
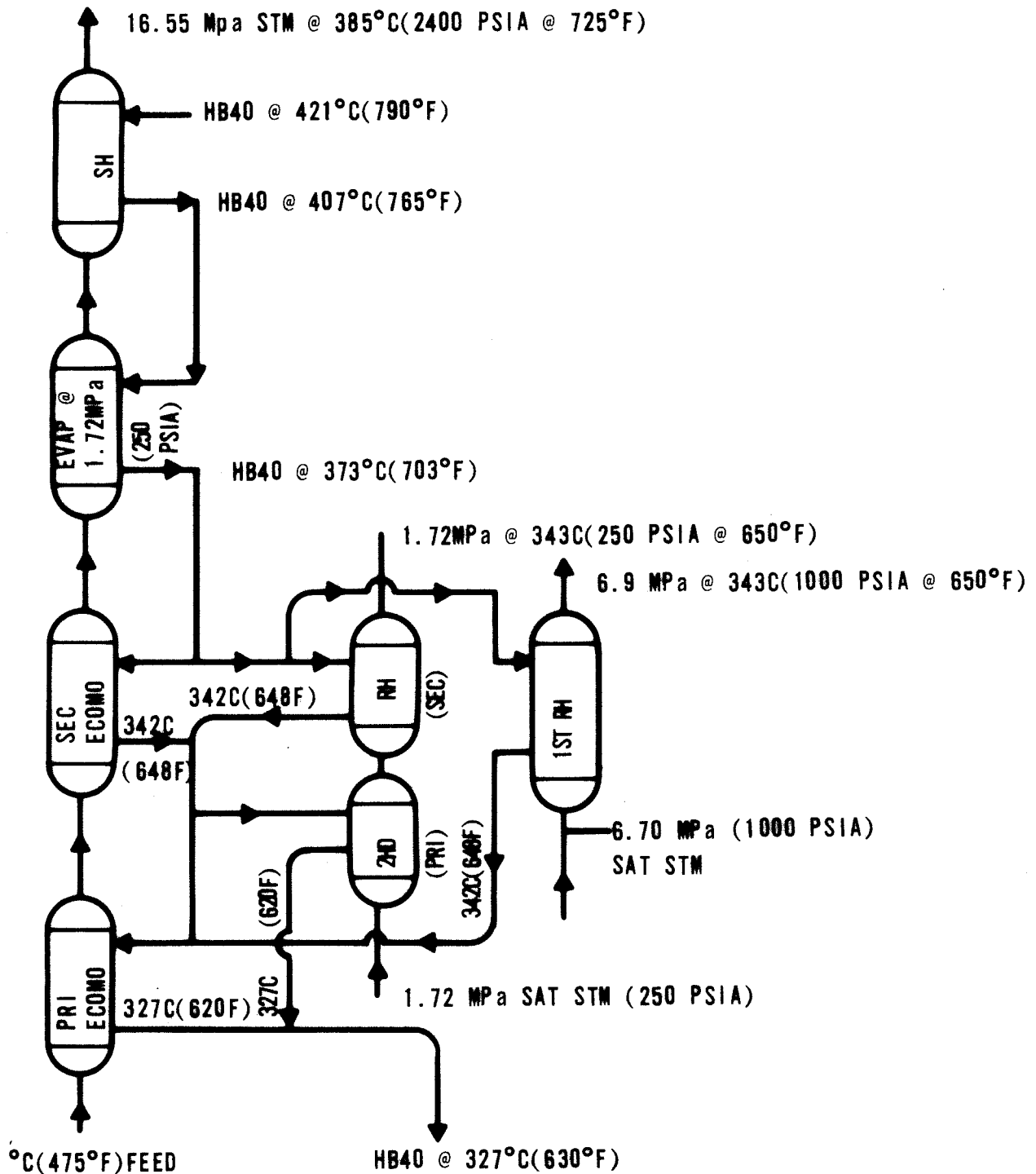


Fig. VI.6-2

Fig. VI.6-3

# **Steam Generator Flow Diagram for Advanced TASKA Tandem Mirror Power Cycle**



This potential future advantage helps justify the use of separate heat exchangers in the TASKA system for the pre-boiler, evaporator and superheater.

## VI.7 Electricity Production

### VI.7.1 Introduction

The objective of this study is to design a heat transport system to remove heat from the reactor blanket and provide steam to a conventional turbo-generator system. Many of the major features of heat transport systems and components, conceived for fusion reactors, are dictated by the need to provide a barrier to tritium diffusion. For the WITAMIR-I project, the reactor coolant was transported directly to the steam generator which utilized double wall tubes in its design as a tritium barrier. For the purposes of the present study, an optional position was taken to utilize more conventional steam generator design requiring some other means for preventing tritium diffusion. The method chosen was the use of an organic coolant loop to absorb tritium which diffuses from the PbLi loop. A basic requirement of the system is, therefore, to provide an intermediate loop of organic fluid between the PbLi reactor coolant and the steam generating system.

### VI.7.2 General System Requirements

Schematics of the heat transport system and its temperature diagram are shown on Figs. VI.7-1 and VI.7-2, respectively. The requirements dictated by the primary system are that the heat removal rate from the reactor blanket be  $45 \text{ MW}_t$  with inlet and outlet operating temperatures of  $300^\circ\text{C}$  ( $572^\circ\text{F}$ ) and  $400^\circ\text{C}$  ( $752^\circ\text{F}$ ), respectively. Heat is transferred to the organic coolant via an intermediate heat exchanger (IHX) utilizing a horizontal U-tube, U-shell arrangement. The organic fluid, HB-40, has been successfully used as an experimental reactor coolant in Canada. The organic fluid flows in series



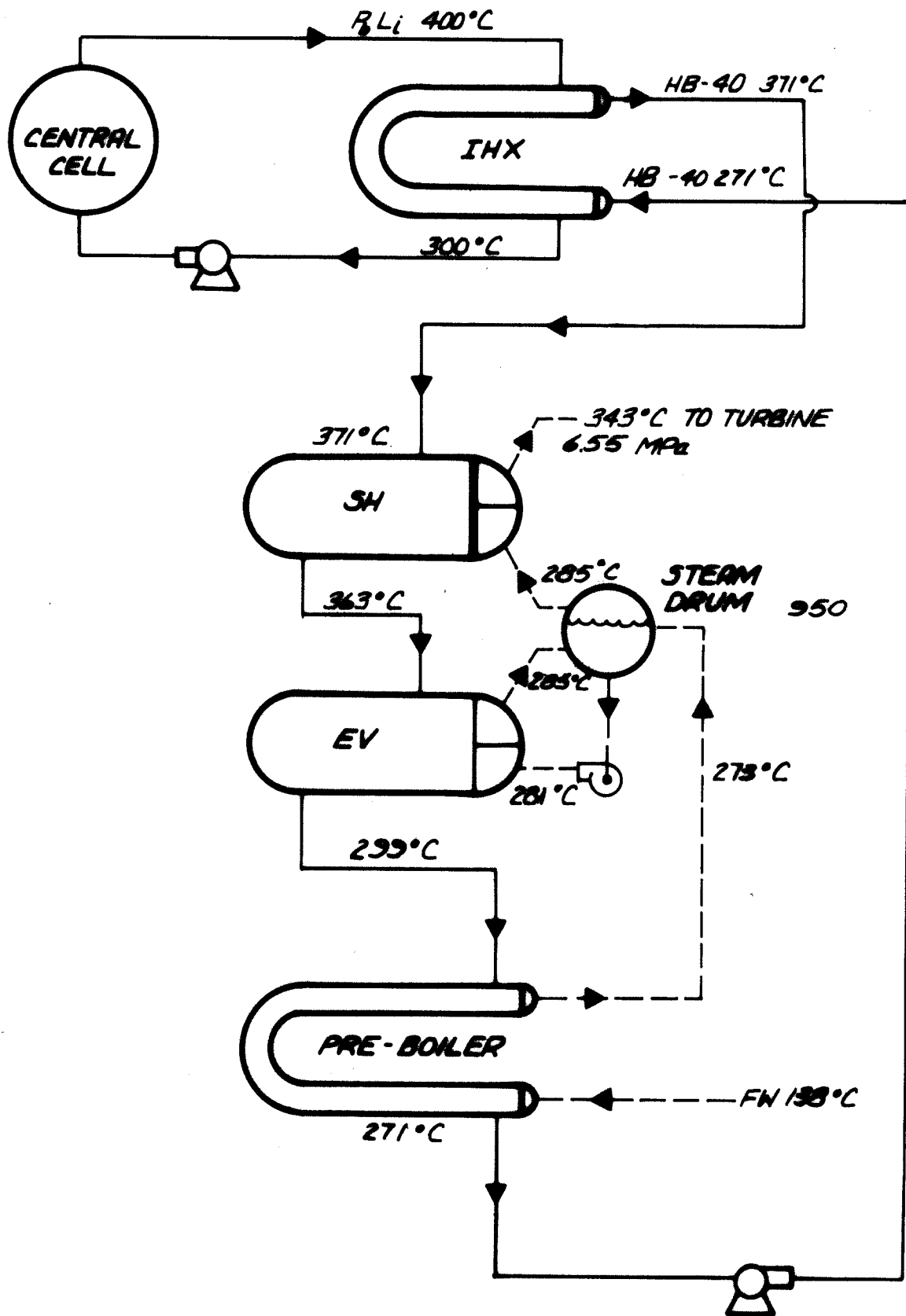


FIGURE VI.7-1 SYSTEM LAYOUT SCHEMATIC

# Power Module Temp. Diagram

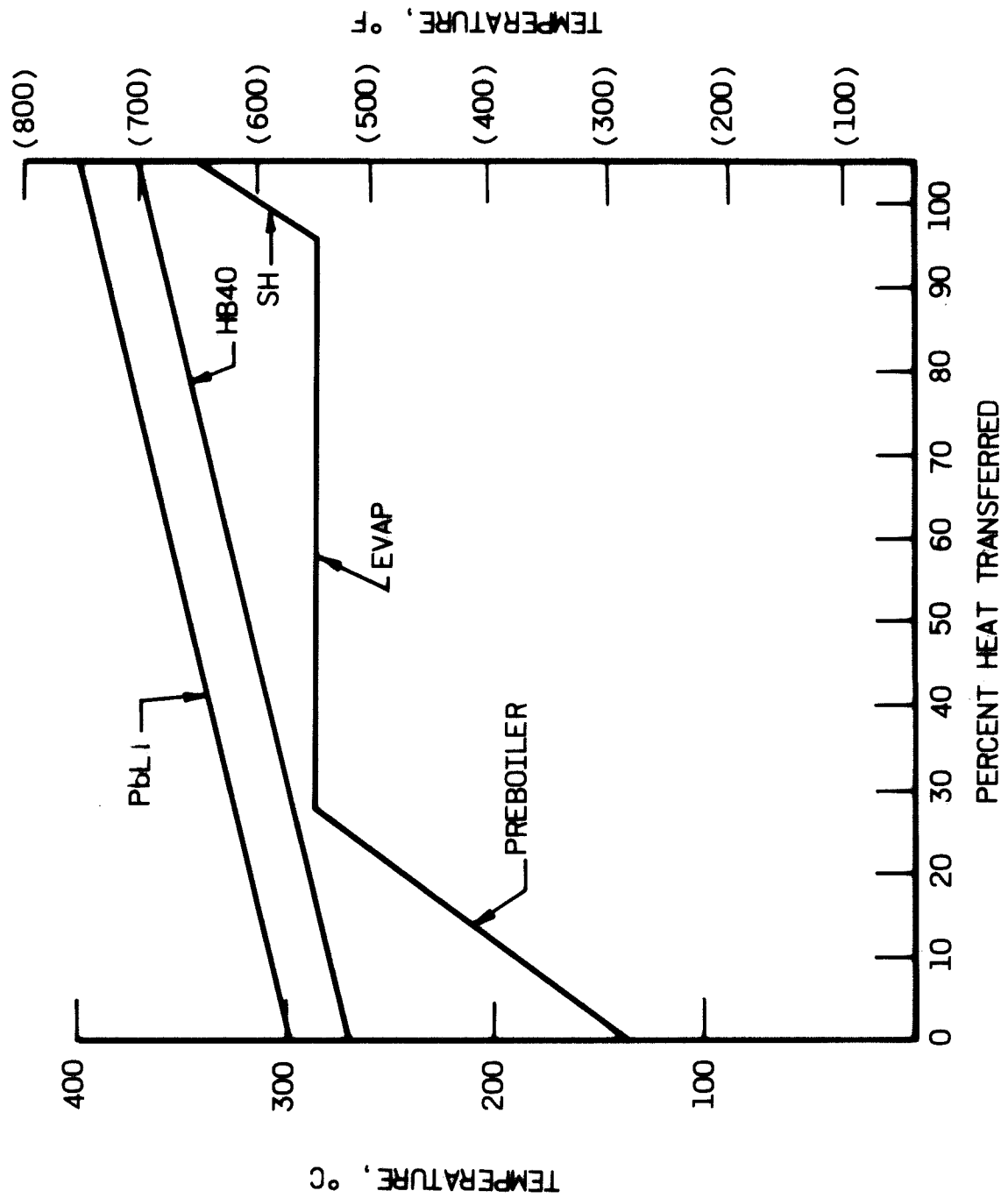


Fig. VI.7-2

through the superheater, evaporator and pre-boiler components of the steam generator system before being pumped back to the IHX. Steam conditions of 6.55 MPa (950 psia), 343°C (650°F) were chosen which would meet the requirements of small turbines designed for conditions close to those of PWR systems and still permit practical LMTD's in the respective heat exchange components.

Before sizing the components, the organic fluid flow rate and operating temperatures were selected. The temperature limits were determined after a study was made to trade-off the IHX surface against the steam generator surface.

#### VI.7.3 Steam Generation System

The major components of the steam generation system are the superheater, evaporator, pre-boiler, steam drum, and recirculating pump. A pumped circulating system for the evaporator was chosen over natural circulation primarily because of its flexibility of operation which makes it especially attractive for use in small pilot plants. Pumped circulation systems are often used in the power industry where frequent startups or major load swings need to be accommodated. A circulation ratio of 3.3 was chosen to avoid the transition from nucleate boiling to film boiling within the evaporator. By avoiding the transition, the probability of tube wall corrosion (tubeside) due to deposits is greatly diminished.

Most of the components can be mounted horizontally at one elevation. The only exception is the steam drum which would require some additional elevation above the evaporator to insure positive head on the pump. The horizontal orientation, as opposed to vertical orientation, provides an opportunity to save costs, both in the support of the components and in building height.

A separate pre-boiler has been designed to minimize the total heat transfer surface of the pre-boiler and evaporator. While the pre-boiler and evaporator may be combined in one unit, the total surface will increase. For the purpose of this study, a separate pre-boiler was assumed to provide the overall lowest cost system.

#### VI.7.4 Components

Heat exchanger components need to be designed to accommodate the temperature differences between the heating and cooling fluid and to withstand thermal transients imposed during operation. U-tubes are employed in all components to provide the necessary flexibility and accommodate the differential expansion between the tubes and the shell. For the evaporator and superheater, a straight shell arrangement is used to house the tube bundle. For the pre-boiler and IHX, the higher temperature differences across the unit would impose excessive thermal stresses on a single tubesheet arrangement and, therefore, a U-shell is used to house the U-tube bundle (see Table VI.7-1, Figs. VI.7-3 to VI.7-6).

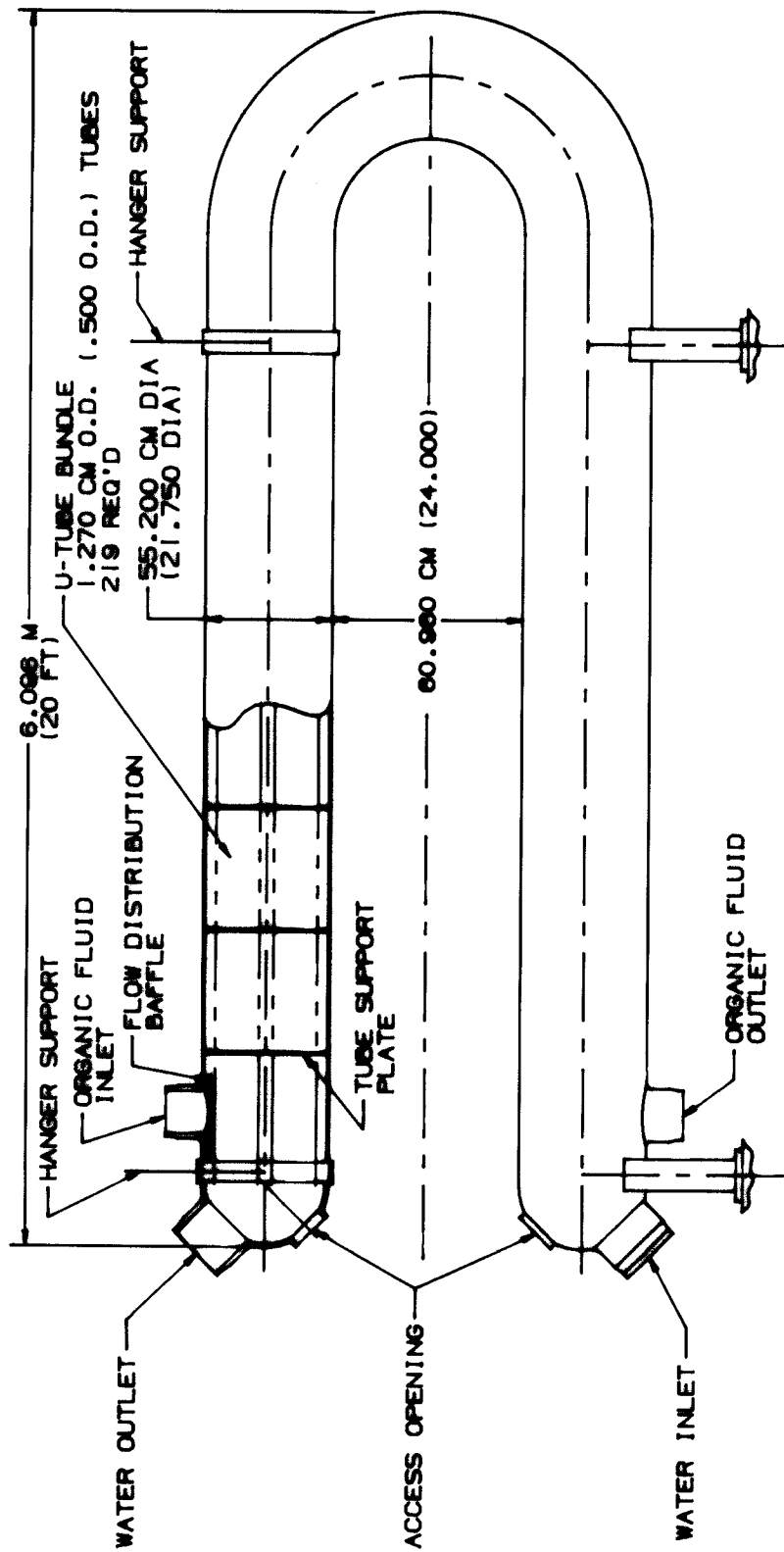
Both the U-tube, U-shell and the U-tube, straight shell arrangements are conventional designs which have been used extensively for nuclear applications.

In the design of the IHX, PbLi was chosen as the shell side fluid and the HB-40 as the tubeside fluid. Putting HB-40 on the tube side was considered appropriate in this component where the HB-40 is in contact with the heating surface at temperatures approaching the limit permitted by thermal decomposition considerations. Removal of surface fouling is more easily accomplished on the tube side.

Table VI.7-1 IHX/Steam Generator Sizing (45 MWe)

| Heat Exchanger Type                    | Units                                              | IHX                                    | Pre-Boiler                          | Evaporator                                 | Superheater                                   |
|----------------------------------------|----------------------------------------------------|----------------------------------------|-------------------------------------|--------------------------------------------|-----------------------------------------------|
|                                        |                                                    | U-Tube<br>U-Shell, Pure<br>Counterflow | U-Tube, U-Shell<br>Pure Counterflow | U-Tube, Straight Shell<br>Segmented Baffle | U-Tube, Straight<br>Shell Segmented<br>Baffle |
| Material                               |                                                    | 9 Cr-1 Mo                              | SA-210 Carbon<br>Steel              | SA-210 Carbon<br>Steel                     | 2 1/4 Cr-1 Mo                                 |
| Number of Tubes<br>(All 0.5 inch O.D.) |                                                    | 1,235                                  | 219                                 | 772                                        | 221                                           |
| Shell Side<br>Fluid Type               |                                                    | PbLi                                   | Organic                             | Organic                                    | Organic                                       |
| Temperatures                           |                                                    |                                        |                                     |                                            |                                               |
| Inlet                                  | °C/°F                                              | 400/752                                | 299/571                             | 361/683                                    | 371/700                                       |
| Outlet                                 | °C/°F                                              | 300/572                                | 271/520                             | 299/571                                    | 361/683                                       |
| Norm. Pressure                         | MPa/psia                                           | .55/80(inlet)                          | .21/30(inlet)                       | .28/40(inlet)                              | .34/50(inlet)                                 |
| Flow Rate                              | kg/s/lb/hr                                         | 2881/22.9x10 <sup>6</sup>              | 169.1/1.34x10 <sup>6</sup>          | 169.1/1.34x10 <sup>6</sup>                 | 169.1/1.34x10 <sup>6</sup>                    |
| Tube Side<br>Fluid Type                |                                                    | Organic                                | Water                               | Water/Steam                                | Steam                                         |
| Temperature                            |                                                    |                                        |                                     |                                            |                                               |
| Inlet                                  | °C/°F                                              | 271/520                                | 138/280                             | 273/535                                    | 285/545                                       |
| Outlet                                 | °C/°F                                              | 371/700                                | 273/535                             | 285/546                                    | 343/650                                       |
| Norm. Pressure                         | MPa/psia                                           | .41/60(inlet)                          | 3.24/1,050(outlet)                  | 6.90/1,000(outlet)                         | 6.6/950(outlet)                               |
| Flow Rate                              | kg/s/lb/Hr                                         | 169.1/1.34x10 <sup>6</sup>             | 7.24/.147x10 <sup>6</sup>           | 7.24/.147x10 <sup>6</sup>                  | 7.24/.147x10 <sup>6</sup>                     |
| Heat Transfer Surface                  | m <sup>2</sup> ft <sup>2</sup>                     | 639/9,633                              | 94.5/1,016                          | 388/4,169                                  | 54.4/585                                      |
| Overall Coefficient                    | (W/m <sup>2</sup> °C/<br>BTU/hrft <sup>2</sup> °F) | 1,741/307                              | 2,133/376                           | 1,947/343 (boiling<br>range)               | 1,704/300                                     |
| Estimated Weight                       | kg/lb <sub>m</sub>                                 | 17,689/39,000                          | 4,644/10,232                        | 4,900/22,810                               | 5,521/12,179                                  |

146080B



NOTE:  
 APPROX. DRY WEIGHT - 4,644 KG (10,230 LBS)  
 MATERIAL - SA-210 CARBON STEEL

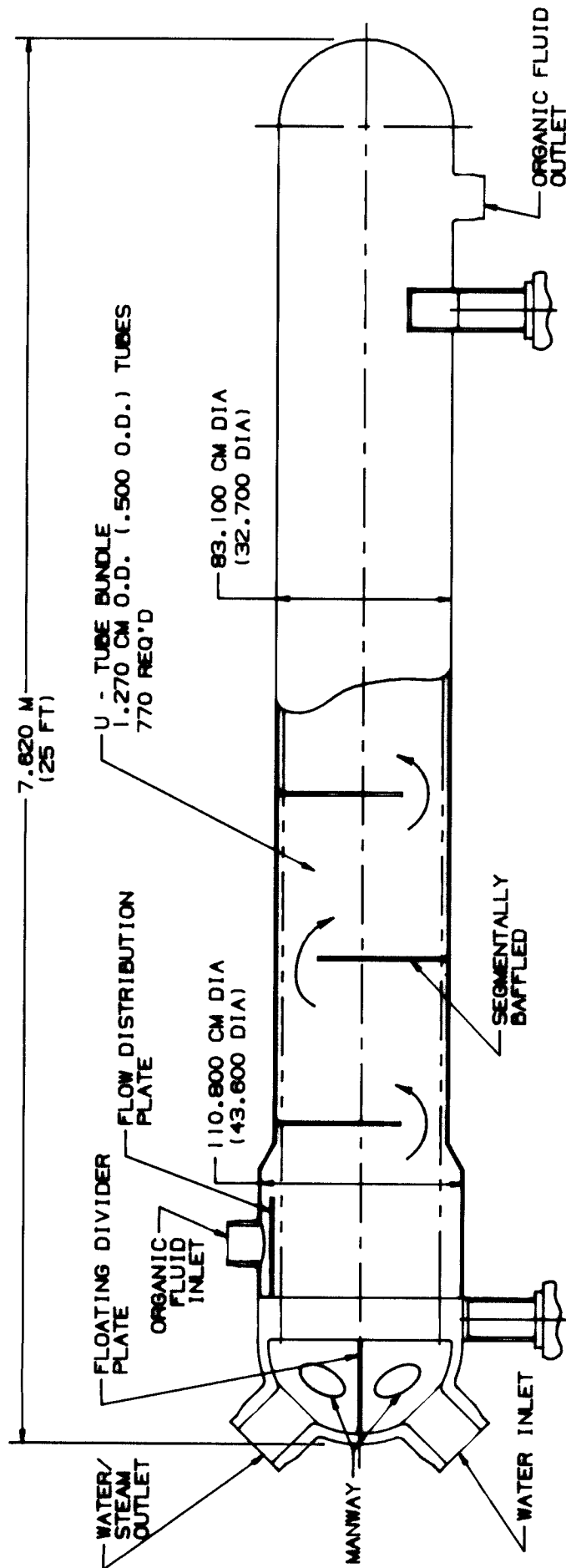
THE ENGINEERING COMPANY

REV. 1, - REVISED VESSEL DIA., TUBE QUANTITY AND WEIGHT.

PREBOILER

146080B-1

148081B



NOTE:  
APPROX. WEIGHT - 10,791 KG (23,770 LBS)  
MATERIAL - SA-210 CARBON STEEL

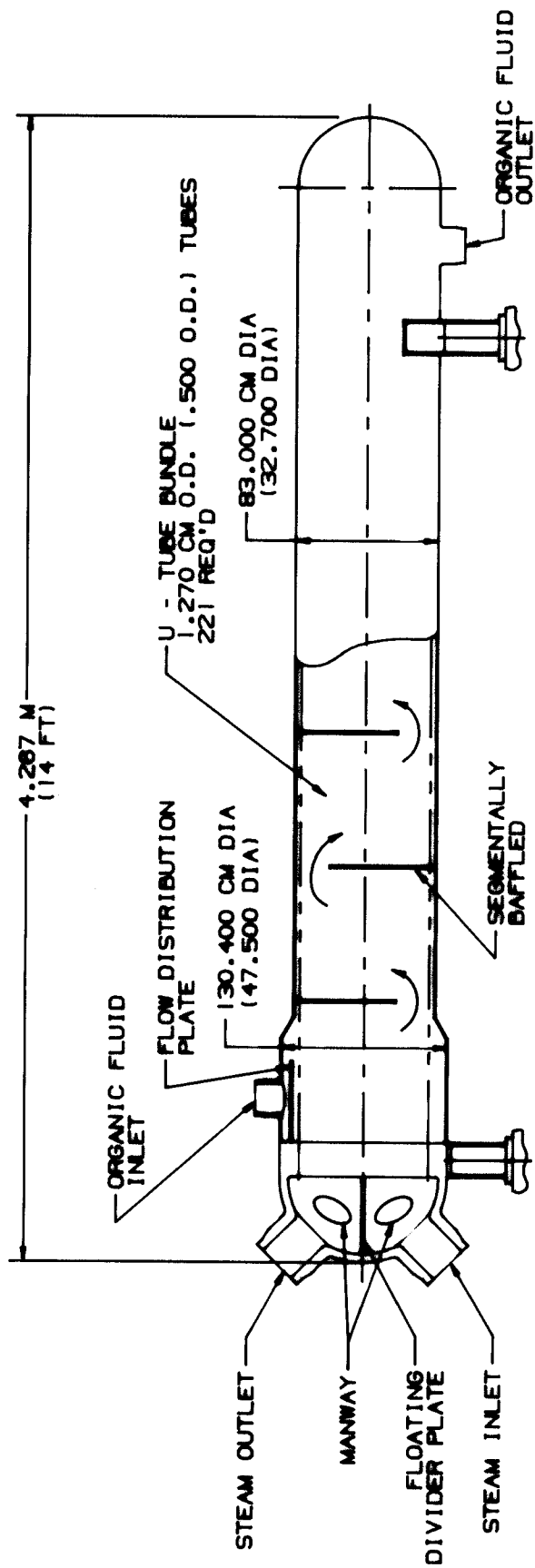
THE ENGINEERING COMPANY

REV. 1 - REVISED VESSEL DIAMETERS, TUBE QUANTITY & WEIGHT.

EVAPORATOR

148081 B-1

46082B



NOTE:  
APPROX. DRY WEIGHT - 5,528 KG (12,180 LBS)  
MATERIAL - 2 1/4 CR-1MO

THE ENGINE & TOOL COMPANY

SUPERHEATER

REV. 1 - REVISED VESSEL DIAMETERS, TUBE QUANTITY & WEIGHT.

DATE: 11/1/82

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Fig. VI.7-5



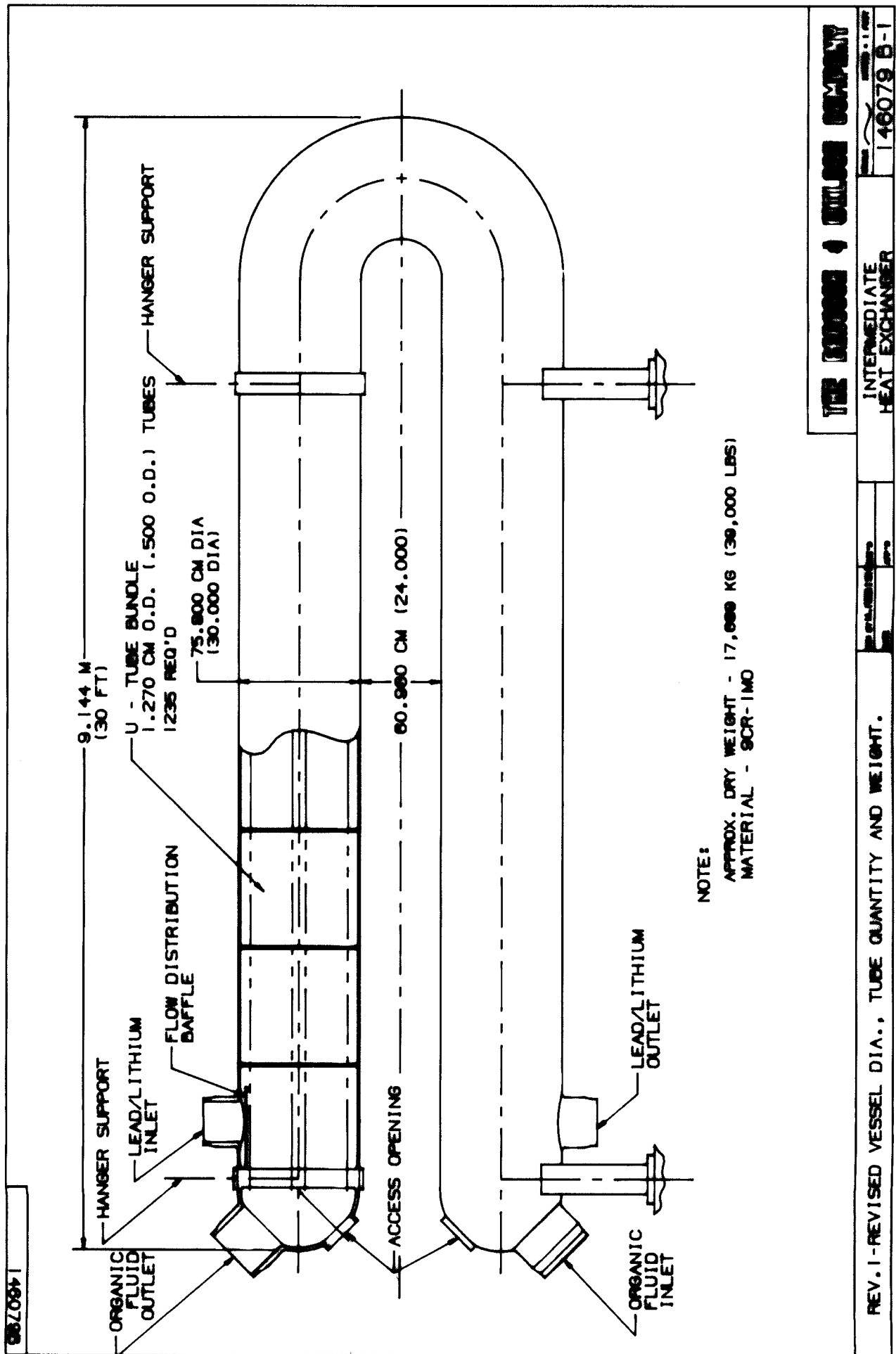


Fig. VI.7-6

Tube side fouling tends to be at a minimum over a velocity range of 1.8 to 3.1 meters/sec (6 to 10.0 ft/sec). A velocity of 3.1 meters/sec was chosen to improve heat transfer and thereby minimize IHX heating surface. Reduced heating surface is of course desirable to reduce tritium diffusion as much as possible.

In all of the components, shell side velocities are typical of values normally used to avoid damaging flow induced vibration. In addition, tube support plates are spaced to provide a tube natural frequency above the flow eddy force frequencies produced by turbulence at velocities. The support plates are broached to allow longitudinal passage of the shell side fluid.

Materials have been chosen to lower unit costs consistent with allowable stresses at design temperatures and with environmental compatibility. The 9 Cr-1 Mo steel was mandated for the IHX to ensure compatibility with the PbLi. For the pre-boiler and evaporator, carbon steel was chosen. For the superheater, 2 1/4 Cr-1 Mo was chosen to provide higher strength than is available with carbon steel at the required design temperatures.

#### VI.7.5 Component Costs

Estimated component costs are listed below:

| <u>Component</u>            | <u>Cost</u>           |
|-----------------------------|-----------------------|
|                             | (fob Barberton, Ohio) |
| Pre-Boiler                  | \$ 672,000            |
| Evaporator                  | 1,200,000             |
| Superheater                 | 620,000               |
| Intermediate Heat Exchanger | 3,264,000             |
| Boiler Steam Drum           | <u>124,000</u>        |
| Total                       | \$5,880,000           |

Cost estimates for the circulating pump, valves, piping, instrumentation, controls, foundations, etc., can be made at a later phase of the project when reactor parameters are more nearly finalized.

Low pressure in the PbLi and HB-40 loops plus the respective compatibility of these fluids with 9 Cr-1 Mo and carbon or low alloy steel (Table VI.7-1) will assure a relatively low component and system cost when compared with concepts using other heat transport fluids involving less favorable materials compatibility, higher operating temperatures or higher pressures.

## VII Testing Program

### VII.1 Overview

The main function of TASKA is to provide a reactor relevant testing environment for both materials and blanket modules. To that end, the central cell region of TASKA has been designed to maximize the access to the test modules while at the same time provide sufficient tritium breeding to make TASKA self-sufficient. We have also separated the materials testing modules from those testing integrated blanket concepts because, in general, we wanted to maximize the damage rate in the materials modules.

It is also important to realize the time frame for which TASKA is designed. If it were to operate in the early 1990's, then TASKA could be providing useful information by the mid 1990's. Such information would be of great value to confirm the choice of materials and blanket concepts chosen for Demonstration reactors which themselves might be operating at the turn of the century. It is most probable that the DEMOs would be under construction in the mid 1990's. This distinction between confirmation and determination is quite important here. If we wanted to use TASKA to determine what materials and blankets should be used in the DEMO, then we would have to wait until high fluence data from TASKA is thoroughly analyzed and integrated in the DEMO design. This could not happen before the late 1990's which means that the DEMO would probably not be operating until the year 2010 or so. On the other hand, if we make the choices for the DEMO based on other test devices, such as light water and fast fission reactors, RTNS-II, or FMIT, then TASKA could be used to confirm that those choices were indeed reasonable. This would allow a DEMO to be designed in the early 1990's, built in the late 1990's and operating near the turn of the century. If the materials or blankets chosen for the

DEMO do not withstand the full fusion environment of TASKA, then the construction of the DEMO could be delayed until a suitable solution could be found.

Another aspect of the testing capabilities of TASKA which needs to be discussed is the ability of such a device to provide information for many other fusion concepts such as tokamaks, stellarators, etc. The real question is, can the pulsed environment (on the time scale of one burn cycle, ~100 seconds) of a tokamak or the plasma wall interaction of any toroidal device be simulated by a steady state, open-ended machine? Certainly one could surround samples with electrical heaters which could be periodically varied to produce temperature cycles. It is also possible to mechanically or pneumatically impose loads on structural members to simulate periodic stresses. However, such simulation procedures will require more complex testing modules and may seriously reduce the "high flux" test volume available for samples. Nevertheless, it could be done and in future designs such test capsules will be designed.

There are two areas where TASKA cannot reasonably simulate the environment of a tokamak, i.e., disruptions and first wall erosion. Disruptions are so violent that both the electromagnetic and thermal loads far exceed what could be programmed into a tandem mirror machine. The rapidly changing magnetic fields will impose loads on blankets and first walls which may cause them to fail prematurely. If we are to learn about such phenomena, they probably will have to be tested in a tokamak.

The high flux of charge exchange particles characteristic of divertor slots, limiters, or neutral beam ports can cause erosion by sputtering and/or blistering. While such effects may be present in the end plug region of TASKA, the lack of such an environment in the central cell where the neutrons

are produced means that the first wall environment in a TMR will be far less severe than for tokamaks. If a blanket concept or particular first wall material is susceptible to such particle fluxes, it will be very difficult to properly simulate it in TASKA.

In summary, TASKA is designed to test both materials and blanket concepts in a high temperature, high neutron fluence environment. The main task of TASKA is to confirm DEMO, material and blanket choices already made by previous tests at low fluence, or on the basis of a few specimens tested to high fluence in neutron spectra similar to, but not exactly the same as in a fusion device. After confirmation is completed, TASKA can be used to develop the materials and blankets for the first generation of commercial reactors. The useful life of TASKA is expected to be ~15 years but with some modifications, 20 years may be possible. This would allow materials and blankets to be tested well into the 21st century.

## VII.2 Materials Testing

### VII.2.1 Introduction

The testing of materials for a Demonstration Reactor at the turn of the century has placed some severe requirements on TASKA. For example, we need to obtain high integrated fluences (in the neighborhood of  $5 \text{ MW-years/m}^2$ ) under reactor relevant conditions in the shortest possible time. These requirements prompted the REGAT (Reduced Damage Gradient) concept to be examined and a large amount of this chapter is devoted to its analysis. At this time the neutronic characteristics of the design have been investigated in the most detail, but some thought has also been devoted to the question of temperature control and ease of replacement.

The whole question as to how many, and what kind of materials samples need to be tested has been addressed in a manner similar to that in the INTOR, ETF, and to a limited degree, the FED projects. We have tried to benefit from these previous analyses and also tried to tailor the materials testing program to the tandem mirror program. Certainly much more work needs to be done in the future as feedback from the materials community is obtained.

The organization of this subchapter is as follows. First a generalized description of the neutronics of fusion test blankets is given and the particular geometrical advantages of the REGAT module are highlighted. Next, the specific test matrix considered for the magnetic fusion program is given along with the design of the TASKA-REGAT module. Finally, the neutronics analysis of the TASKA-REGAT module is given.

## VII.2.2 Neutronics Survey Analysis

### VII.2.2.1 Introduction

One problem with past materials test modules<sup>(1-4)</sup> has been the large spatial variation in displacement damage (dpa), gaseous transmutation rates, and nuclear heating from the first wall to the back of the module. In an effort to minimize those gradients, we have designed a Reduced Gradient Test Module (REGAT) for TASKA. In the REGAT design, the blanket region is cut back from the first wall to leave the test module protruding into the vacuum chamber (see Fig. VII.2-1). This results in direct exposure of the module sides to source neutrons. Therefore, the damage gradients in the module are reduced as compared to the traditional design which is surrounded on all sides by the normal blanket and shield. A survey neutronics analysis of the REGAT module is presented here to assess its effectiveness in reducing damage gradients and enhancing the materials testing capability. The results for the REGAT module used in TASKA are given and compared to the results for the current INTOR<sup>(2)</sup> materials testing module in Section VII.2.4.

### VII.2.2.2 Calculational Model

The materials testing module is placed at the center of the cylindrical central cell section of the tandem mirror reactor. Fig. VII.2-1-a shows a schematic of the traditional test module. The module is immediately behind the first wall and surrounded on all sides by the normal blanket and shield. In Fig. VII.2-1-b, the original REGAT design is shown with the test module protruding into the vacuum chamber and directly exposed to neutrons on the sides. An improved REGAT design is shown in Fig. VII.2-1-c. In this design, the blanket region surrounding the module is tapered to increase the contribution to the damage in the module from neutrons reflected from the blanket.



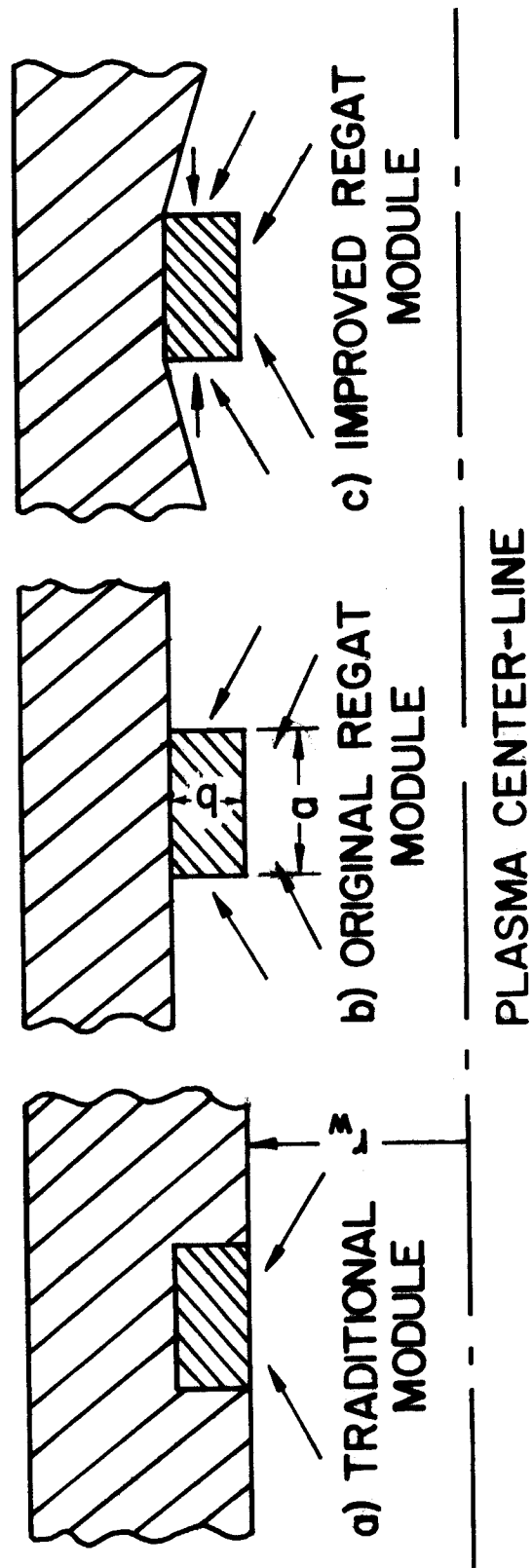


Fig. VII.2-1 Schematic of REGAT and traditional test modules.

Further enhancement of damage can be obtained when a neutron multiplier is used in the region of the blanket surrounding the module. In TASKA, neutron multiplication results from using  $\text{Pb}_{83}\text{Li}_{17}$  liquid metal eutectic as a coolant/breeder. The REGAT module, with width  $a$  and thickness  $b$ , has an inner radius  $r_w$ . A blanket consisting of 81 v/o  $\text{Pb}_{83}\text{Li}_{17}$  and 10 v/o structure made of the ferritic steel alloy HT-9 was considered in this work. Although the TASKA design was used in the calculations, the results are more general and apply to any linear fusion device.

Detailed neutronics calculations have been performed, using the three-dimensional Monte Carlo code MORSE,<sup>(5)</sup> to determine the spatial variation in the damage of different module designs. A coupled 25 neutron-21 gamma cross section library, consisting of the RSIC DLC-41B/VITAMIN-C data library<sup>(6)</sup> and the DLC-60/MACKLIB-IV response data library,<sup>(7)</sup> was used.

The combinatorial geometry capability of the MORSE code was used to model the geometry of the problem. The test module was divided into 60 zones with volume detectors used to estimate the quantities of interest in the different zones. An isotropic 14.1 MeV neutron source was sampled uniformly from a cylindrical plasma zone which is 20 m long and 32 cm in radius. Because of symmetry, only half the central cell was modelled with a reflecting albedo boundary used at the center of the module.

The geometry for the computational model used is given in Fig. VII.2-2.  $L$  is the central cell length and  $L_0$  is the length over which the blanket is tapered.  $L$  and  $L_0$  were taken to be 20 m and 4 m, respectively. The zones 1 to 30 were used to get the spatial variation of damage in the REGAT module. A 30 cm thick blanket was used in the calculations and is represented by zone 31.

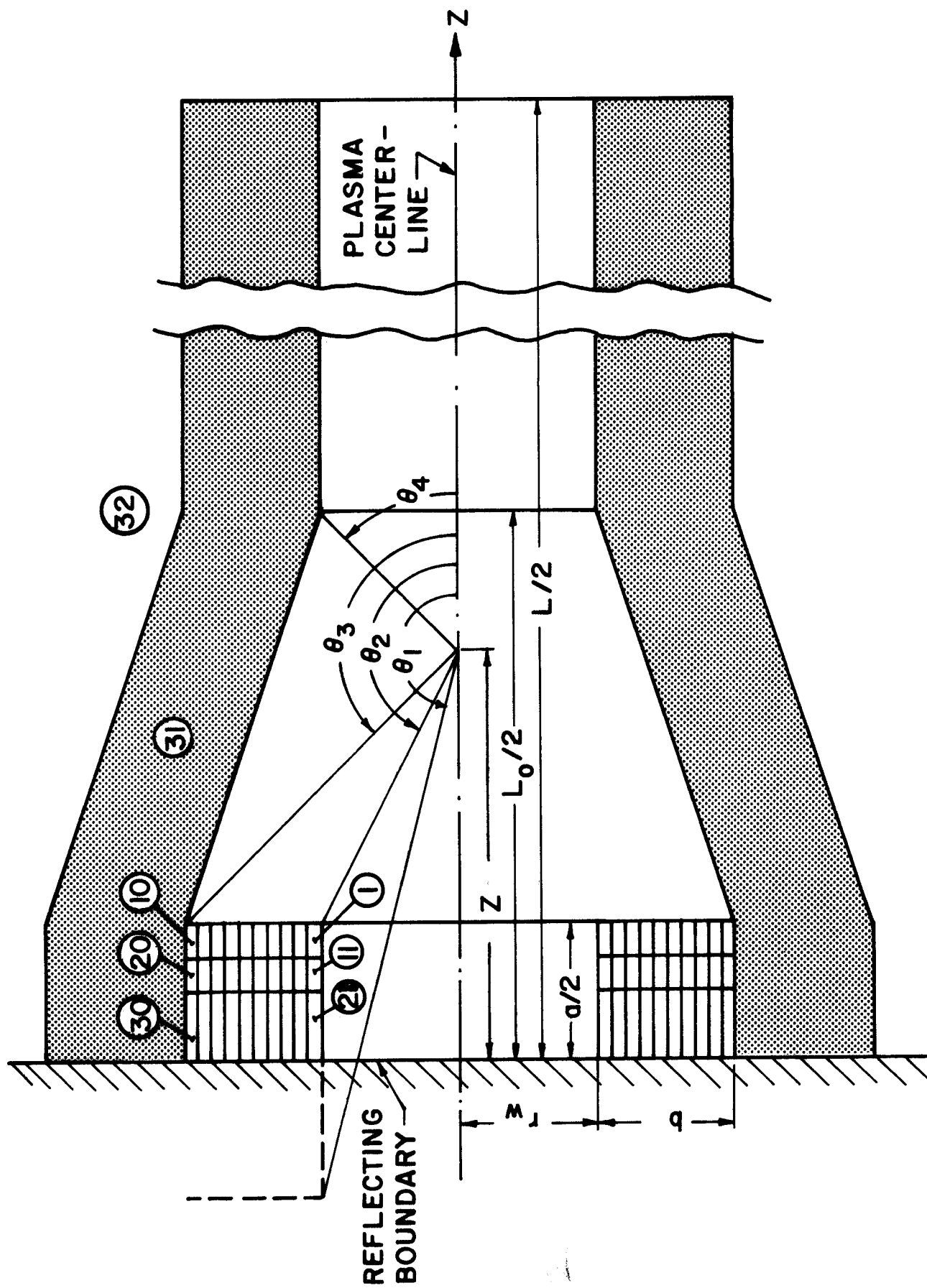


Fig. VII.2-2 Geometrical model used in Monte Carlo calculations.

Since the width of the test module is much smaller than the central cell length, a position dependent angular source biasing technique was used to get statistically adequate estimates for the neutron flux in the test module. This technique is similar to that used previously for the analysis of the end plug of a tandem mirror fusion reactor.<sup>(8)</sup>

First an axial position  $z$  is sampled uniformly along half the central cell from the distribution function

$$P(z)dz = 2dz/L . \quad (\text{VII.2.1})$$

The polar angle  $\theta$  is then picked from a biased distribution with 95% of the neutrons forced to have polar angles between  $\theta_{\min}$  and  $\theta_{\max}$ . This biased distribution is given by

$$P(\cos\theta)d\cos\theta = \begin{cases} \frac{0.95 d\cos\theta}{\cos\theta_{\min} - \cos\theta_{\max}} , & \theta_{\min} < \theta < \theta_{\max} \\ \frac{0.05d\cos\theta}{2 + \cos\theta_{\max} - \cos\theta_{\min}} , & \theta_{\min} > \theta > \theta_{\max} . \end{cases} \quad (\text{VII.2-2})$$

The value of  $\theta_{\min}$  is determined depending on the axial position of the source. The smallest of  $\theta_2$  and  $\theta_3$  is compared to  $\theta_4$  and the larger angle is considered as  $\theta_{\min}$ . For all axial positions,  $\theta_{\max}$  takes the value of  $\theta_1$ . This leads to particles reaching the REGAT module without collision 95% of the time and particles suffering at least one collision before reaching the module 5% of the time. For the final estimates to be unbiased the statistical weight of the source is modified by the ratio of the unbiased to the biased distribution functions at the polar angle picked. An azimuthal angle  $\phi$  is then sampled

uniformly from the distribution function

$$P(\phi)d\phi = d\phi/2\pi . \quad (\text{VII.2-3})$$

Finally, a radius is sampled for the source particle from the distribution function

$$P(r)dr = 2rdr/r_p^2 , \quad (\text{VII.2-4})$$

where  $r_p$  is the plasma radius.

50,000 histories were used in the Monte Carlo calculations yielding less than 10% statistical uncertainties in the estimates for the damage rates in the different module zones. The results in this work are normalized to a wall loading of  $1 \text{ MW/m}^2$  at the front surface of the module.

#### VII.2.2.3 Analysis of REGAT Module Performance

A series of survey calculations was performed for the REGAT module with different values for the width and thickness to investigate the effect of dimensions on the testing capability of the module. Calculations have also been performed for the traditional module design to assess the effectiveness of the REGAT module in reducing the damage gradients. In these survey calculations, a first wall radius of 40 cm was used. The test module was considered to consist of 50 v/o 316 SS and 50 v/o void. All the results presented here are normalized to a  $1 \text{ MW-yr/m}^2$  fluence.

Figure VII.2-3 shows the spatial variation of atomic displacement and helium production rates in a 5 cm wide and 100 cm long REGAT module as compared to the equivalent traditional module. The peak damage rate occurring

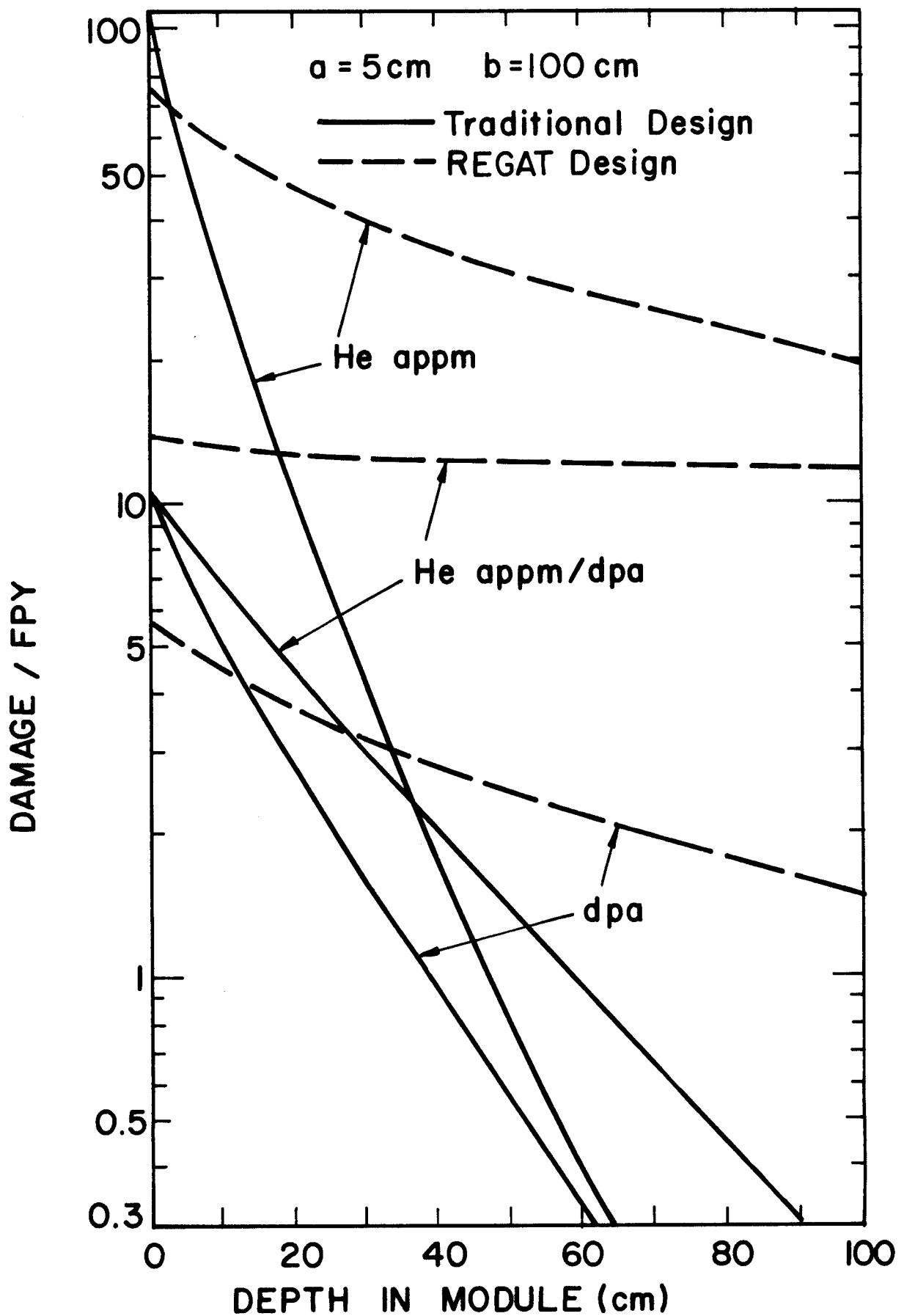


Fig. VII.2-3 Spatial variation of damage in traditional and REGAT modules.

at the front of the module is decreased in the REGAT module. The reason is that source neutrons missing the module will impinge on the adjacent first wall and after being slowed down can contribute to the damage of the front of the traditional module. This effect is more pronounced on the dpa which can be produced by neutrons with energies as low as  $\sim 1$  keV. However, the gradients for both dpa and helium production rates are reduced considerably when the REGAT module is used. The net effect is that the average damage rates in the REGAT module are greater than those in a traditional module with the same dimensions. The spatial variation of helium to dpa ratio, which is an important parameter affecting microstructure evolution,<sup>(9)</sup> is also shown. A nearly uniform He/dpa ratio is obtained in the REGAT module. This results from the fact that less neutron spectrum softening occurs in the REGAT module when compared to the traditional module.

Table VII.2-1 gives the average and peak/average ratios for dpa and helium production rates for different REGAT module dimensions. Fixing the module width and decreasing the thickness results in increasing the average damage rates and reducing the gradients. This results from the reduced attenuation in the radial direction. Decreasing the width for a fixed length results also in increasing the average damage rates and reducing the gradients because of reduced attenuation in the axial direction.

It is clear from the results of Table VII.2-1 that decreasing the thickness  $b$  has more pronounced effect on increasing the average damage rates than does decreasing the width  $a$ . This results primarily from decreasing the peak damage rate occurring at the center of the front surface of the module as the width,  $a$ , decreases. Hence, for a fixed test module volume, it is expected that the average damage rate increases if one uses a wider module. In Fig.

Table VII.2-1. Effect of Module Dimensions  
on Damage Rates in the REGAT Module

| a(cm) | b(cm) | dpa/FPY |           | He appm/FPY |           |
|-------|-------|---------|-----------|-------------|-----------|
|       |       | Avg.    | Peak/Avg. | Avg.        | Peak/Avg. |
| 5     | 100   | 2.37    | 2.271     | 30.62       | 2.535     |
| 5     | 50    | 3.25    | 2.131     | 41.63       | 2.254     |
| 7.5   | 50    | 3.19    | 2.302     | 39.20       | 2.564     |

VII.2-4, the volume integrated damage rate, used as a measure of the testing capability of the module, is plotted versus module length for a fixed module volume of  $0.153 \text{ m}^3$ . For the purpose of comparison, the results for the traditional module are also included. Notice that the two modules give the same results in the limiting case when the testing zone covers the whole central cell length. The REGAT module has a higher performance factor than that of the traditional module with the improvement being more pronounced when a thick module is used. Hence, one gains significantly by using the REGAT module when the area devoted for materials testing is limited by other considerations such as tritium breeding.

Figure VII.2-5 shows the variation of dpa rate with depth in a 7.5 cm wide and 50 cm thick module for the traditional design and three REGAT module cases. In the first case, no neutron reflection from the surrounding blanket is considered. The second case represents the original REGAT design with a



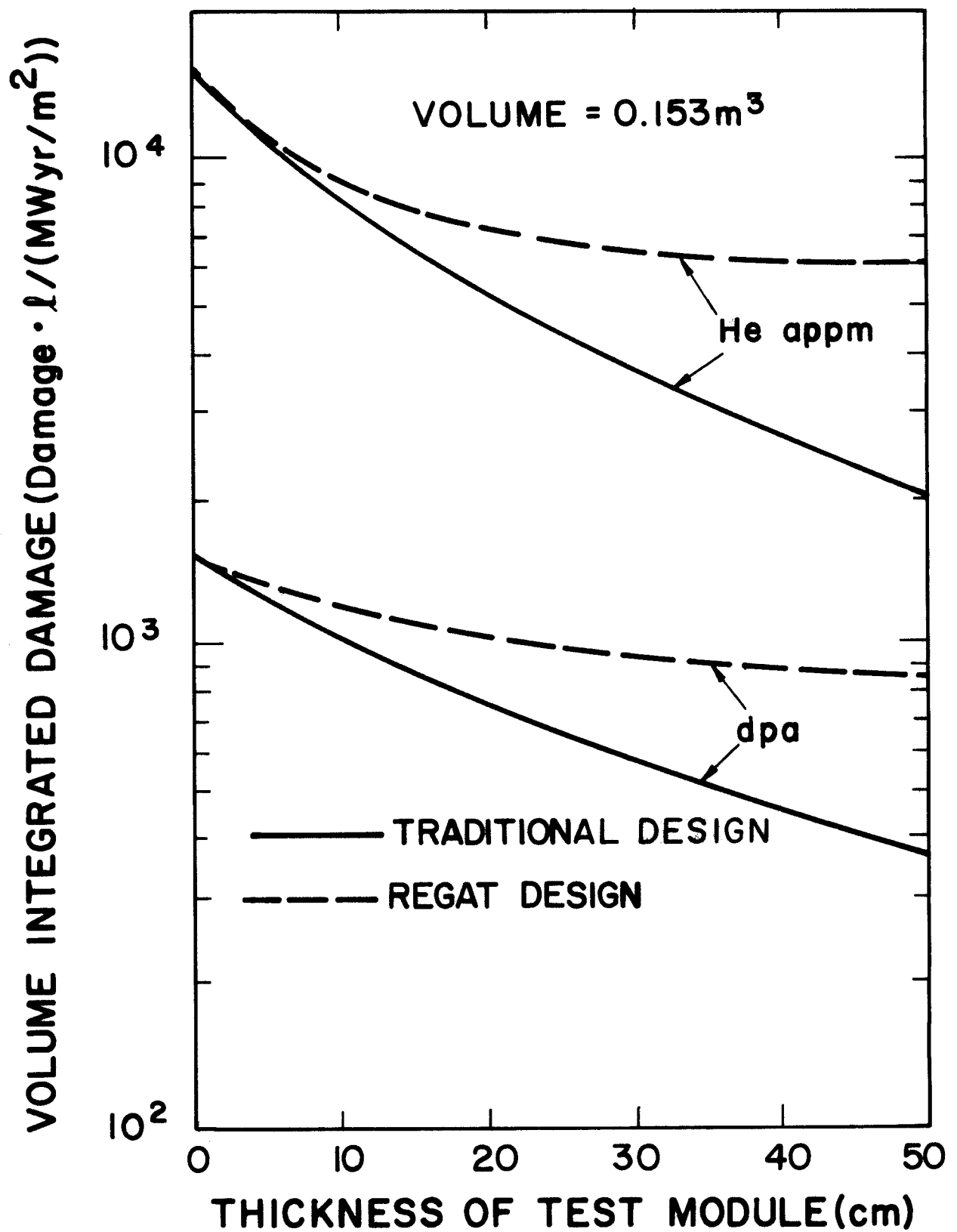


Fig. VII.2-4 Volume integrated damage as a function of module thickness for a fixed volume.

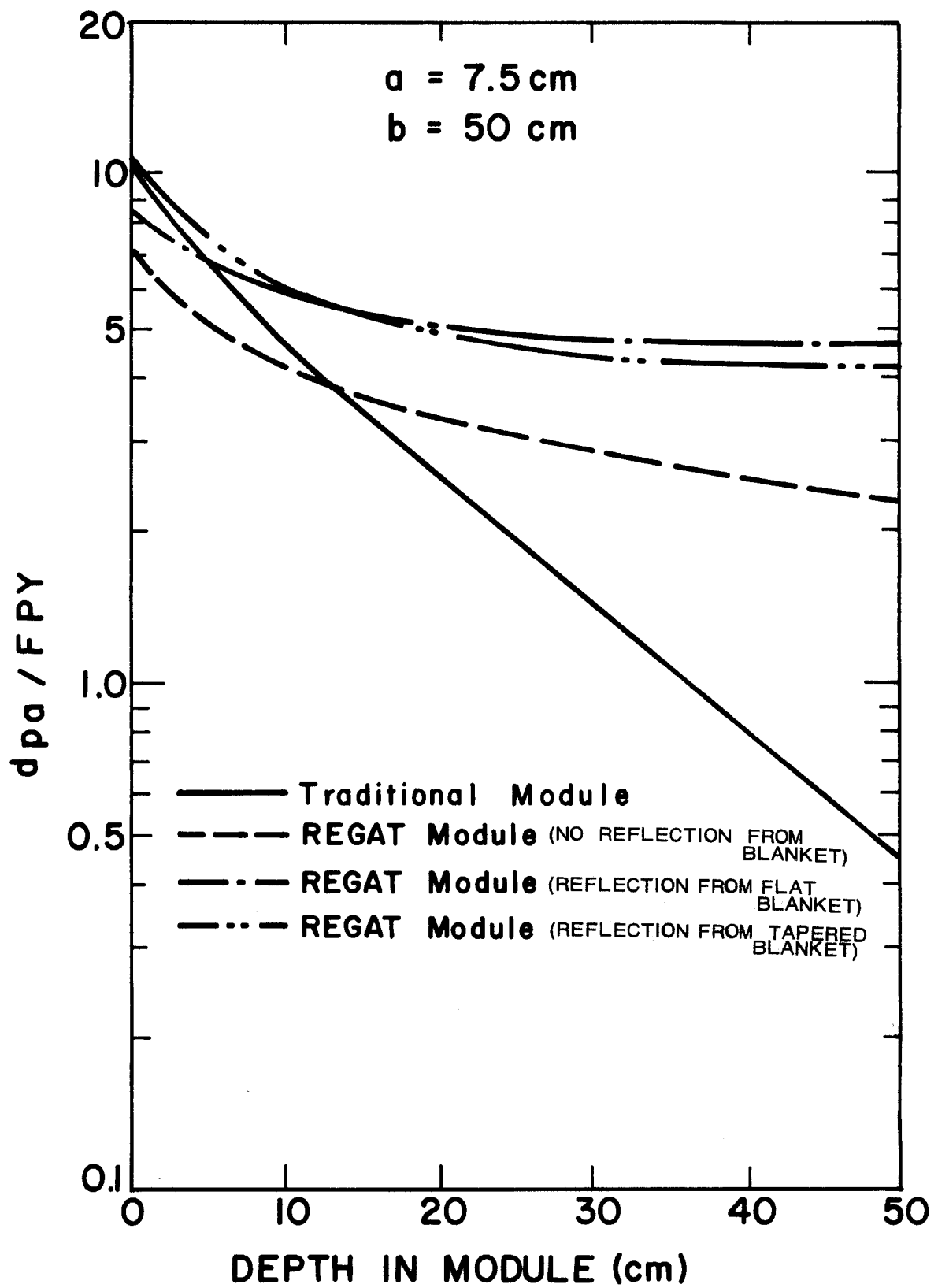


Fig. VII.2-5 Spatial variation of dpa rate for different module designs.

flat blanket adjacent to the module and the third case represents the improved REGAT design with the adjacent blanket tapered as shown in Fig. VII.2-1.

Table VII.2-2 shows the average and peak/average ratios for dpa and helium production in these cases. It is clear that, even in the case when no reflection from the blanket is considered, the REGAT module has higher average damage rates and less gradients than the traditional module. However, the peak damage in this case is much lower than that in the traditional module because of the lost contribution from source neutrons missing the module.

When neutron reflection and multiplication in the flat adjacent blanket are considered, the damage rate increases at all points of the module with the effect being more pronounced at the back of the module (which is closest to the blanket where the reflected neutrons originate). This results in higher average damage rates and less damage gradients. The effect on helium production is not as pronounced as that on dpa because most of the neutrons produced in  $(n,2n)$  reactions with lead in the blanket do not contribute to helium production.

The peak damage rate is increased further by tapering the inner surface of the blanket surrounding the module. On the other hand, the damage rate at the back of the module decreases because less source neutrons impinge directly on the back of the module side. This results in a slightly higher damage gradient. The effect is more pronounced for helium production, most of which is caused by direct source neutrons. The net result is that the average dpa rate increases while the average helium production rate decreases with tapering. Even though the damage gradients are slightly increased, using the improved REGAT design is recommended because it results in a higher average dpa rate and allows some test specimens to be exposed to high damage levels.

Table VII.2-2. Effect of Neutron Reflection From the Blanket  
on Damage Rates in a Test Module 7.5 cm Wide and 50 cm Thick

|                                                   | dpa/FPY     |                  | He appm/FPY |                  |
|---------------------------------------------------|-------------|------------------|-------------|------------------|
|                                                   | <u>Avg.</u> | <u>Peak/Avg.</u> | <u>Avg.</u> | <u>Peak/Avg.</u> |
| Traditional module                                | 2.342       | 4.27             | 13.045      | 7.69             |
| REGAT module (no reflection)                      | 3.188       | 2.30             | 39.202      | 2.56             |
| REGAT module (reflection from<br>flat blanket)    | 5.140       | 1.59             | 43.535      | 2.30             |
| REGAT module (reflection from<br>tapered blanket) | 5.347       | 1.95             | 39.591      | 2.53             |

#### VII.2.2.4 Conclusions

An improved materials testing module for engineering test facilities is presented. Detailed neutronics analysis shows that the REGAT design reduces the damage gradients and increases the average damage levels in the module significantly. The results of the survey calculations show that the improvement over the traditional design is more pronounced when a thick module is used. Hence, one gains significantly by using the REGAT module when the area devoted for materials testing is limited by other considerations such as tritium breeding. Tapering the inner surface of the blanket surrounding the REGAT module increases the contribution to the damage from reflected neutrons and hence increases the average dpa rate and allows more test specimens to be exposed to high damage levels.

### VII.2.3 Test Matrix and Test Module Design Considerations

The test matrix for materials irradiations in TASKA is very similar to the matrix developed by the INTOR Test Program group<sup>(2)</sup> during its 1980 Phase-1 study. The TASKA test matrix is a list of materials test specimens and their required environmental and irradiation conditions. Test matrices in general provide a starting point for both experiment and facility designers by defining, in exact terms, what is to be irradiated and under what conditions. Given this information, along with the physical and neutronics features of the reactor, the experiment designer can then, in principle, design a specific test module or modules to physically contain and environmentally control the necessary number of materials test specimens. The test matrix specifications combined with the resultant test module designs are then used as a basis for many of TASKA's mechanical and operational requirements. Hence, the test matrix provides a basis for experiment design, which in turn provides a design basis for that portion of the TASKA facility that interfaces with and is required to support the testing program.

#### VII.2.3.1 Test Matrix

The TASKA test matrix specifies the type of materials with detail on specimen numbers, irradiation temperatures, fluences, stress, changeout frequencies and postirradiation testing conditions. The test matrix provides a basis for estimating the volume needed for materials testing in TASKA plus other information such as range of test temperatures, etc. Also, the frequency of specimen changeout must be accommodated by the test module design with minimal impact on reactor availability.

The test matrix is shown in Table VII.2-3. The presumed scope of the testing specified in the test matrix is that one or two candidate materials

TABLE VII.2-3

## Proposed Materials Test Matrix -TASKA

| Materials Surveillance       | Mat. Var. <sup>a</sup> | Dup. <sup>b</sup> | Temp. <sup>c</sup> | Fluenced <sup>d</sup> | Conditions         | Total Specimens     |                      | Total Capsules <sup>g</sup> |                      |
|------------------------------|------------------------|-------------------|--------------------|-----------------------|--------------------|---------------------|----------------------|-----------------------------|----------------------|
|                              |                        |                   |                    |                       |                    | Mirror <sup>e</sup> | Tokamak <sup>f</sup> | Mirror <sup>e</sup>         | Tokamak <sup>f</sup> |
| Tensile                      | 3                      | 2                 | 2                  | 2                     | 2 rate/temp.       | 48                  | 48                   | 3                           | 3                    |
| Fatigue                      | 0(3)                   | 0(2)              | 0(2)               | 0(2)                  | 0(4) stress        | 0                   | 96                   | 0                           | 4                    |
| Crack growth                 | 1(2)                   | 2                 | 2                  | 3                     | 3 stress levels    | 36                  | 72                   | 1                           | 2                    |
| Fracture                     | 2                      | 2                 | 2                  | 2                     | 2 temp.            | 32                  | 32                   | 4                           | 4                    |
|                              |                        |                   |                    |                       | Subtotal           | 116                 | 248                  | 8                           | 13                   |
| Structural Materials         |                        |                   |                    |                       |                    |                     |                      |                             |                      |
| Tensile                      | 5                      | 3                 | 6                  | 5                     | 6 rate/temp.       | 2700                | 2700                 | 10                          | 10                   |
| Fatigue (High Cycle)         | 0(8)                   | 0(3)              | 0(2)               | 0(4)                  | 0(4) stress        | 0                   | 768                  | 0                           | 29                   |
| Crack growth                 | 3(6)                   | 2                 | 4                  | 4                     | 3 stress levels    | 288                 | 576                  | 3                           | 6                    |
| Fracture                     | 4                      | 3                 | 4                  | 2                     | 2 temp.            | 192                 | 192                  | 5                           | 5                    |
| Swelling                     | 10                     | 5                 | 10                 | 5                     | 4 post irr. tests  | 10,000              | 10,000               | 5                           | 5                    |
| Stress relax.                | 6                      | 4                 | 6                  | -                     | 5 stresses         | 720                 | 720                  | 3                           | 3                    |
| Creep-rupture                | 6                      | 2                 | 6                  | -                     | 5 stresses         | 360                 | 360                  | 2                           | 2                    |
| In situ-cyclic               | 0(4)                   | 0(2)              | 0(2)               | -                     | 0(4) stress levels | 0                   | 64                   | 0                           | 64                   |
|                              |                        |                   |                    |                       | Subtotal           | 14,260              | 15,380               | 28                          | 124                  |
| Other Materials <sup>h</sup> |                        |                   |                    |                       |                    |                     |                      |                             |                      |
| Fatigue                      | 5                      | 3                 | 4                  | 4                     | 2 strain range     | 480                 | 960                  | 11                          | 22                   |
| Tensile                      | 15                     | 3                 | 6                  | 4                     | 4 rate/temp.       | 4,320               | 4,320                | 15                          | 15                   |
| Dimensional stab.            | 15                     | 5                 | 6                  | 4                     | 4 post tests       | 7,200               | 7,200                | 4                           | 4                    |
| Creep                        | 10                     | 2                 | 6                  | -                     | 4 stresses         | 480                 | 480                  | 2                           | 2                    |
| In situ-cyclic               | 0(6)                   | 0(2)              | 0(2)               | -                     | 0(4) stresses      | 0                   | 96                   | 0                           | 96                   |
| Fracture                     | 3                      | 2                 | 2                  | 2                     | 6 temp.            | 144                 | 144                  | 7                           | 7                    |
| Elect. prop.                 | 6                      | 3                 | 2                  | 4                     | 3 tests            | 432                 | 432                  | 18                          | 18                   |
| Therm. Cond.                 | 6                      | 3                 | 4                  | 4                     | ---                | 288                 | 288                  | 12                          | 12                   |
|                              |                        |                   |                    |                       | Subtotal           | 13,344              | 13,920               | 69                          | 176                  |
|                              |                        |                   |                    |                       | Total              | 27,720              | 29,548               | 105                         | 313                  |

<sup>a</sup>Materials x variations<sup>b</sup>Duplication<sup>c</sup>Number of irradiation temperatures<sup>d</sup>Fluence levels<sup>e</sup>Specifically for the Mirror Program<sup>f</sup>Required for both Mirror and Tokamak Programs<sup>g</sup>Capsule volume-390 cm<sup>3</sup><sup>h</sup>Ceramics, electrical, and heat dump materials

( ) Addition for tokamak program

have been identified (through extensive fission reactor and accelerator source screening programs) for the various applications (first wall, structure, insulators, etc.) and the extent of testing is determined by the number of specimens needed to determine quantitatively the critical properties of the irradiated materials.

The first portion of the test matrix (surveillance and structural materials) was developed for candidate first wall materials and other structural components receiving high exposure and the second portion of the test matrix was developed for other high exposure applications such as high heat flux components, armor, and RF. The test matrices reveal why large numbers of specimens are needed to provide design data.

In the test matrix, the types of tests are listed in the left hand column, and important test parameters are summarized as column headings across the top. The first parameter is "materials x variations" and refers to the number of base (candidate) materials and the number of variations in heat treatment or different conditions such as weldments to be tested. The second column refers to duplication of specimens, primarily to verify the reproducibility of data. In some cases such as specimens for swelling and phase stability, the duplication also allows for attrition in the postirradiation preparation of samples.

The next two columns refer to the two basic parameters of irradiation temperature and fluence. The numbers of separate irradiation temperatures and goal fluences are given in these columns.

Typically, parallel tests are performed at several irradiation temperatures in order to map expected temperature-dependent behavior over the range of interest. Usually, linear dependence of materials properties on tempera-

ture is not expected, especially near ceiling temperatures (above which useful service is not expected).

The microstructure of materials is often studied at more temperature conditions than are mechanical properties because swelling and phase stability are quite sensitive to temperature, and the more complete characterization of microstructures with temperature (and fluence) provides a qualitative guide for interpolating a coarser matrix of observations on materials properties. For example, microstructural observations were performed at intervals of about 50°C in irradiation tests supporting alloy development for the U.S. breeder program. In the test matrices for TASKA, the number of irradiation temperatures varies from four temperatures for several mechanical properties tests to ten temperatures for observations of swelling and phase stability.

Irradiation experiments typically include tests to a goal fluence, representing end-of-life conditions if possible, and several interim fluences. Irradiation experiments are expensive and take a long time. Consequently, preliminary predictions of material behavior are used for redirecting experimental objectives and for preliminary design data, pending corroboration with data at high fluences. However, the extrapolation of low fluence data to predict properties at high fluences can be grossly misleading. A rule of thumb is extrapolation by less than a factor of two in fluence (for moderate fluence levels, i.e. exceeding incubation doses for property changes) is acceptable. Data at several values of moderate to high fluences is desirable to define trends, especially where the properties after irradiation are only marginally acceptable. Five goal fluences are appropriate for some tests in TASKA on structural materials (components using "other" materials were assumed to be more easily replaceable than the first wall and have lower lifetime fluences).



The column heading "Conditions" refers to test conditions other than temperatures and fluence. Most of the conditions, for example the strain ranges in fatigue tests, are conditions in postirradiation tests performed after the specimens are removed from TASKA. Another example of postirradiation test conditions is for tensile tests where both the strain rate and the postirradiation test temperature are variables. Typically, postirradiation tensile tests are performed at the irradiation temperatures, at a temperature slightly above the irradiation temperature, and at the temperature expected in the material during refueling or during repair. Designers use these data to predict behavior during operation and for fault conditions such as overheating and seismic response.

In the case of creep tests, the other "condition" is stress. The current techniques for in-reactor creep tests utilize sealed tubes pressurized to different stress levels. The diameters of the tubes are measured at several intervals of fluence after which the specimens are again placed in the reactor for further exposure.

The last two columns give the total number of specimens and the number of irradiation capsules (see Section VII.2.3.2.2 for capsule description) for two cases: (1) mirror relevant testing, and (2) mirror and tokamak relevant testing. The key difference between the two cases is that in (2), fatigue and crack growth specimens are not included. The total specimen numbers are the product of the previous columns. The number of capsules is simply the number of capsules required to hold the specified number of specimens.

The mirror, and mirror and tokamak cases require similar numbers of test specimens: 27,720 vs. 29,548, respectively. However, the irradiation volume required for the mirror and tokamak testing is nearly three times that of the

mirror-only test program. This is due to the relatively large number of specimens required to carry out the stress cycling tests. The configuration and size of material test specimens selected for the TASKA matrix are shown in Appendix VII.2-A.

Test Matrix Related Operations. The fluence requirements specified in the TASKA test matrix necessitate test module removal from the reactor at periodic intervals. The number and time between removals is a function of (1) the test matrix goal fluences, and (2) the rate at which damage is accumulated over the necessary specimen volumes.

The TASKA goal fluences are assumed to be 2, 5, 15, 25, and 50% of some goal fluence - assumed to be 200 dpa for structural materials. Hence, specimens will be discharged from the reactor at 4, 10, 30, 50, and 100 dpa levels.

Figure VII.2-6 is used to determine the time required to achieve the goal dpa levels. The figure illustrates the dpa level throughout a specimen capsule. The average dpa value is used to determine the discharge frequency.

Based on the above fluence level requirements and the dpa rate available in TASKA, the resultant capsule discharge sequence will be as illustrated in Fig. VII.2-7. The capsule discharge sequence is based on an initial test module loading of 252 capsules. At each interim examination, approximately 63 capsules are removed from the test module for specimen analysis. Sixty-three fresh capsules are then loaded into the test module for further irradiation along with the ~ 189 previously irradiated capsules which remained in the test module. This operational sequence minimizes the total time required to achieve the specified goal fluences. Table VII.2-4 summarizes some salient test matrix results and operational frequencies.

FIGURE VII.2-6

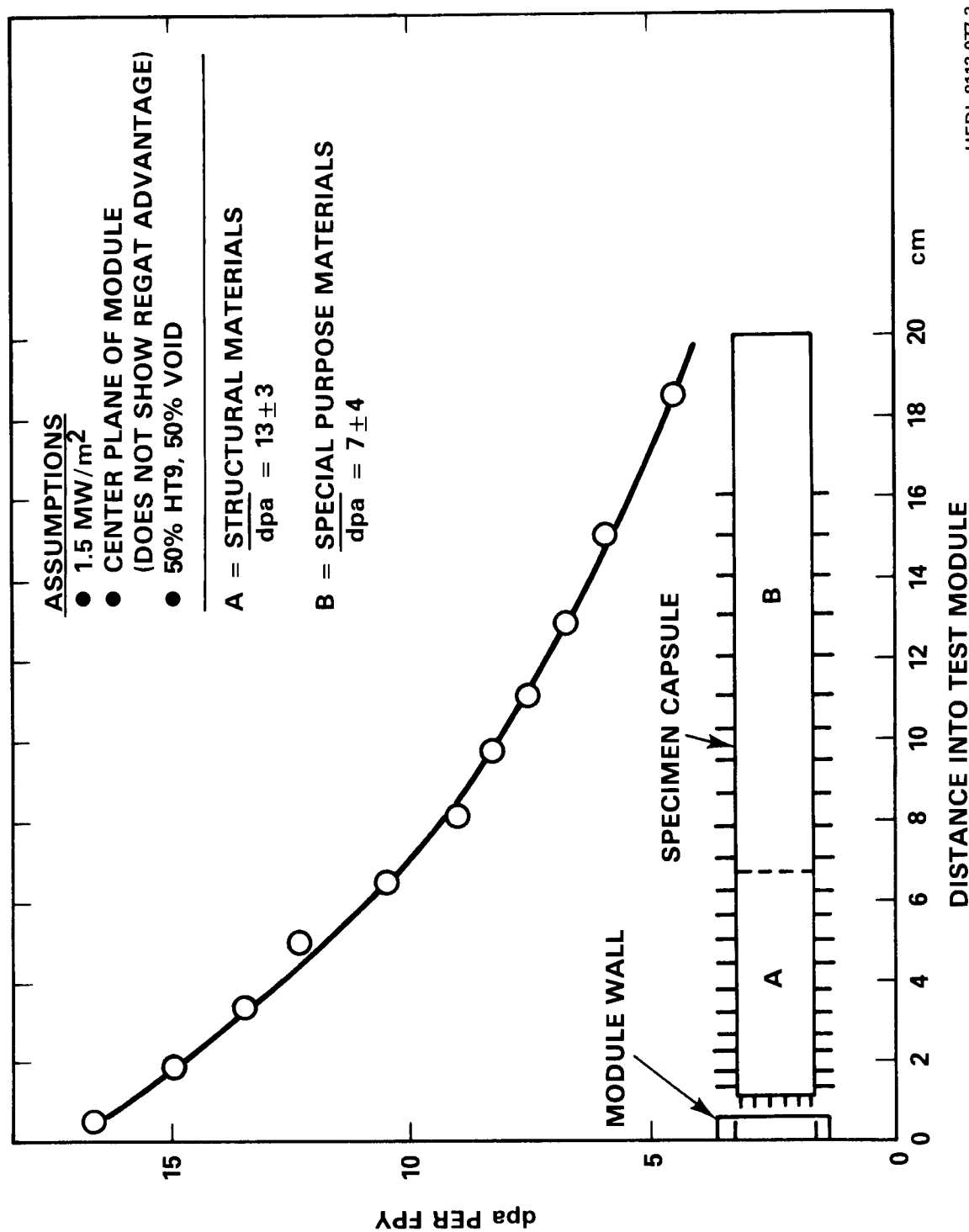


FIGURE VII.2-7

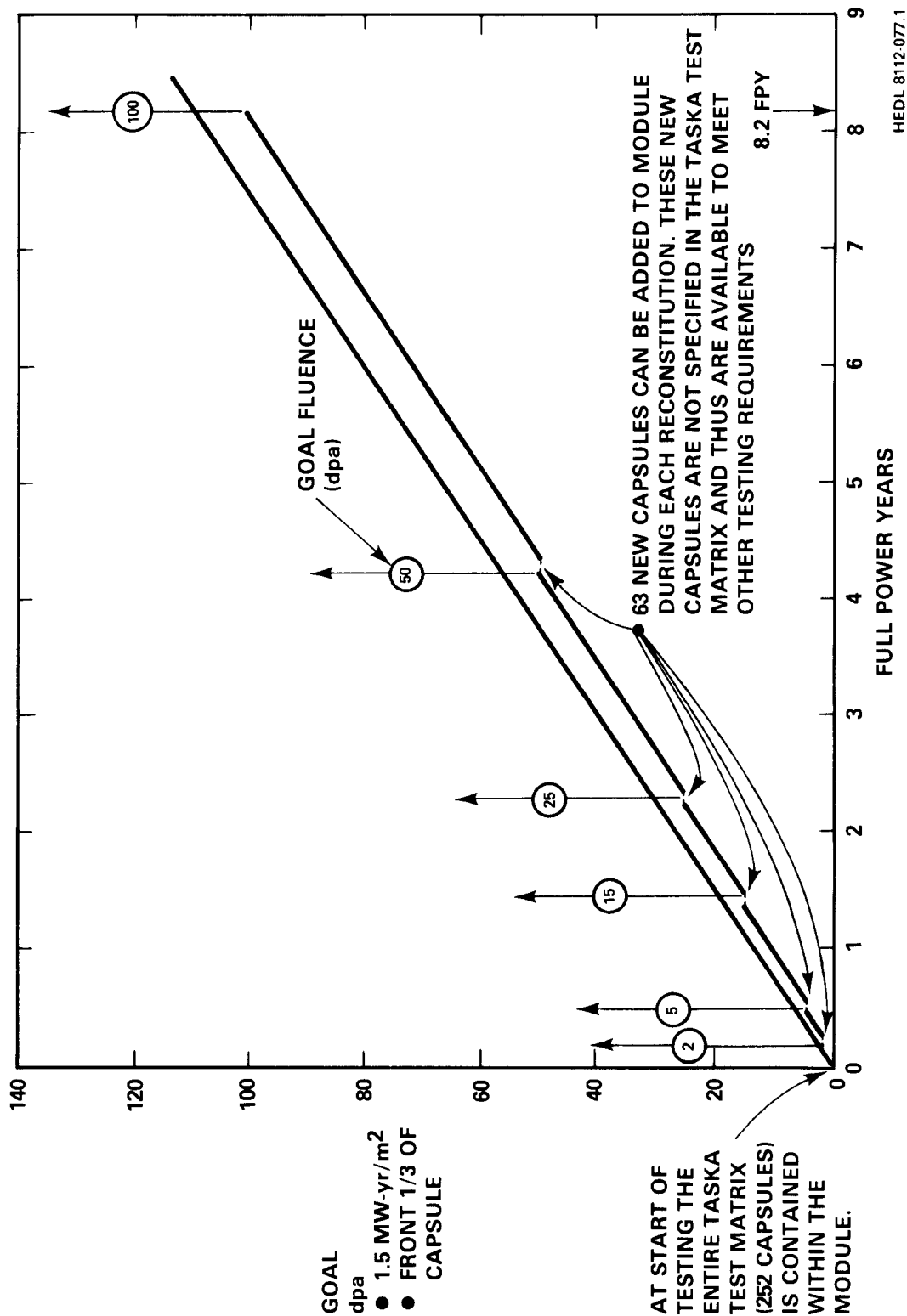


Table VII.2-4. Parameters Derived from TASKA Test Matrix

| <u>Item</u>                                                       | <u>Value</u>     |
|-------------------------------------------------------------------|------------------|
| 1. Number test modules (TM) in TASKA                              | 1                |
| 2. Number specimen capsules in TM                                 | 252 <sup>*</sup> |
| 3. Time to achieve maximum fluence (FPY)                          | 8.2              |
| 4. Ave. time between TM changeouts from reactor (da) <sup>+</sup> | 497              |
| 5. Minimum time between TM changeouts (da) <sup>+</sup>           | 86               |
| 6. Maximum time between TM changeouts (da) <sup>+</sup>           | 1432             |

<sup>\*</sup>Mirror and tokamak specimens considered.

<sup>+</sup>Full power days.

### VII.2.3.2 Test Module Design Considerations

The test module is the hardware that contains the materials test specimens and provides environmental containment as specimens are irradiated in TASKA. Figure VII.2-8 illustrates the key features of the test module concept design for TASKA. The key components of the test module, that will be referred to often in this chapter, are the (1) specimen capsule which contains the groups of individual material specimens, and (2) the annular test module which acts as the heat sink and holding fixture for specimen capsules.

The test module concept must be consistent with the requirements of the test matrix (Section VII.2.3.1) and with the mechanical and neutronics characteristics of the TASKA reactor. The reactor configuration therefore strongly affects the test module design and conversely the test module will have an impact on the availability, floor space, services, access, remote handling, and operations required by TASKA.

#### VII.2.3.2.1 Test Module Design Criteria and Concepts

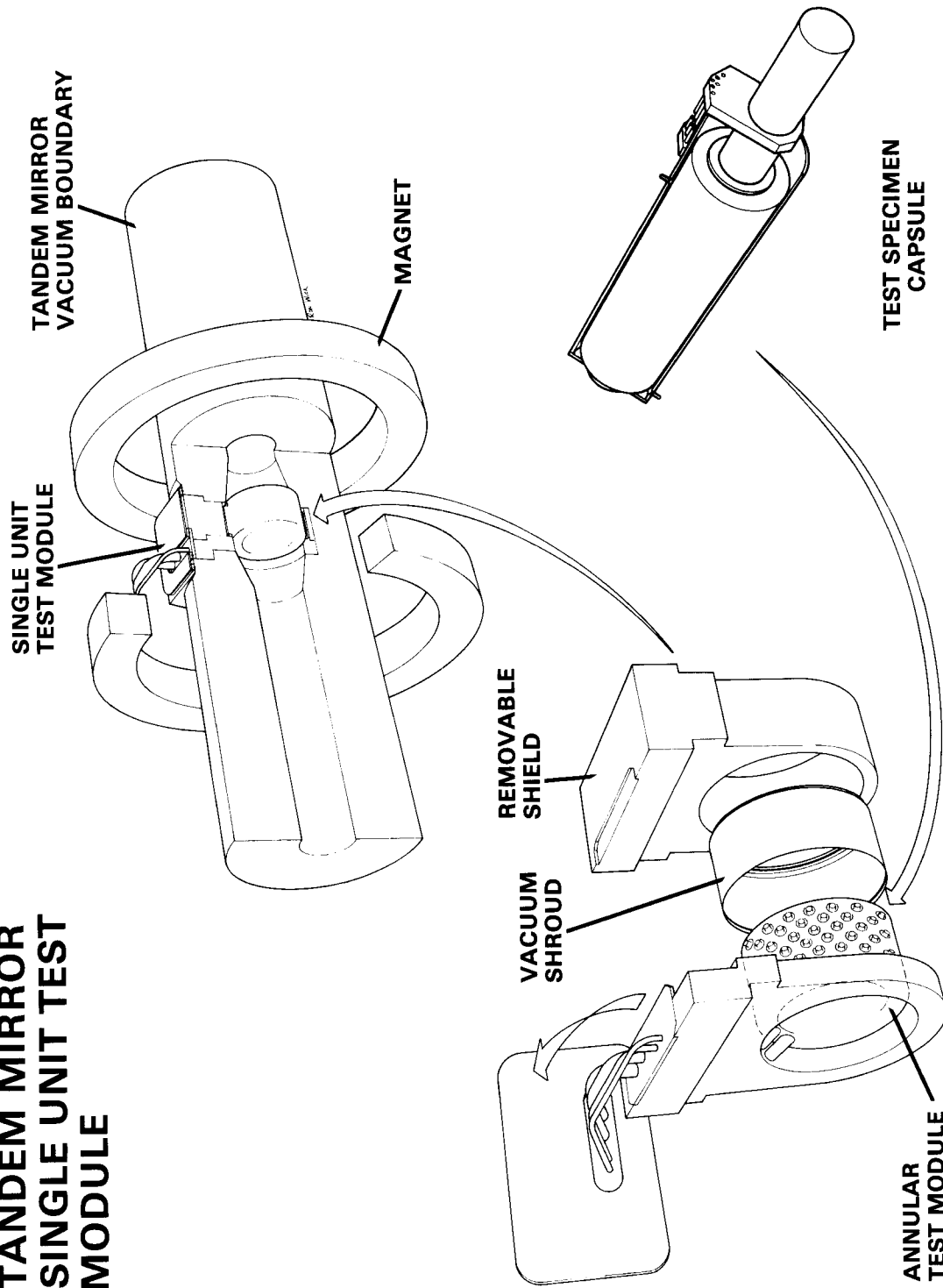
The test module is based on the following general design requirements:

1. Reliability.
2. Minimum specimen temperature variation.
3. Quick turn-around (reconstitution).

Based on the above general test module requirements and features of the conceptual TASKA reactor design, the following more detailed criteria (Table VII.2-5) were established to guide the TM design. These requirements and criteria are consistent with the TASKA test matrix and were established as a result of materials testing experience gained through involvement in (1) the fast breeder reactor program, and (2) design studies supporting the Fusion Materials Irradiation Test Facility and the INTOR Phase-1 Study.

# **TANDEM MIRROR SINGLE UNIT TEST MODULE**

FIGURE VII.2-8



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Table VII.2-5. Design Criteria

1. Test module shall permit simultaneous irradiation of an array of specimen capsules within the first 20 cm of blanket region. Maximum axial reactor opening is 95 cm.
2. Test module components shall have a design lifetime of at least four full power years.
3. Temperature range available in designing specimen capsule shall be 300-650°C with a calculated variation in specimen temperature of less than 10°C.
4. Each specimen capsule shall have independent temperature control and monitoring capability.
5. Test modules shall be capable of accepting previously irradiated specimens and undergo reconstitution.
6. A portion of the test module shall interface directly with the plasma chamber. There will be no first wall between test module and plasma.
7. The test module shall interface with the adjoining blanket and shield consistent with the REGAT concept, and consistent with biological shielding requirements.
8. Test module design shall maximize utilization of the available flux-volume.

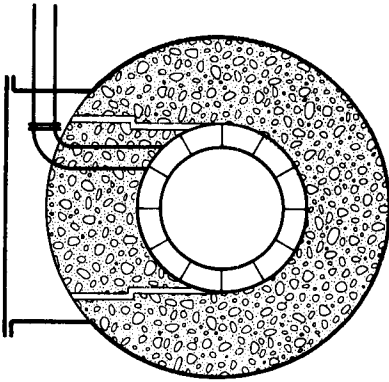
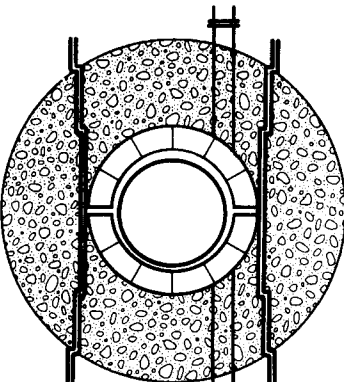


Test Module Design Concepts. Since the above criteria do not uniquely define a single test module, a number of concepts were considered. Table VII.2-6 graphically illustrates four TM concepts that were considered. The concepts were evaluated according to the considerations listed horizontally along the top of the table. Questions posed by designers in evaluating the considerations are as follows. Reliability: Will the design last and is the design solid and simple? Versatility: Can the design accommodate a range of specimen types and sizes and can the overall test module configuration be changed as programmatic considerations mature and evolve? Fabricability: Can standard welding, machining and forming techniques be employed? Remote Handling: Can the test module be inserted and removed from the reactor utilizing standard remote systems? Can service leads be connected and disconnected; is there sufficient space and access for remote operations? Specimen Access: Can specimen capsules be easily removed from the test module; can the specimens be removed from the capsules routinely? Volume Utilization: Is the volume occupied by the TM being used to irradiate an optimum number of material specimens? Sealing: Can the TM systems and components be hermetically sealed?

Based on a qualitative assessment of the above considerations the "single unit test module" (Table VII.2-6) was selected as the reference design concept. The reference test module is the simplest design structurally and is inserted into the reactor vertically requiring only an overhead crane. Access for service connection and atmospheric sealing is from the top, thus avoiding the congested areas typically found along the sides of a reactor. The TM is removed from TASKA as a single unit - and it is recommended that the TM be directly transported into a hot cell for disassembly.

# TEST MODULE CONCEPTS

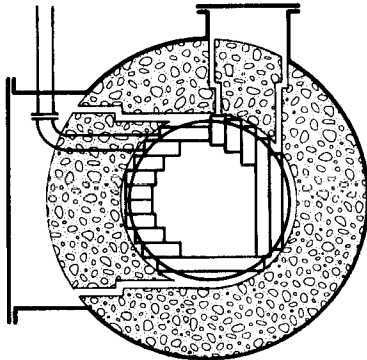
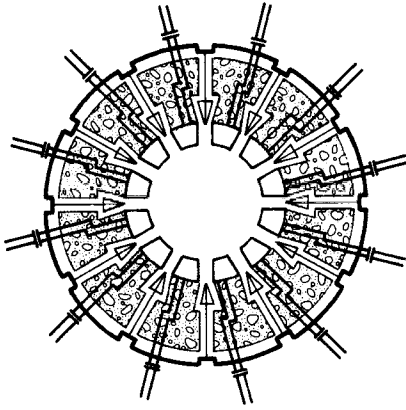
TABLE VII.2-6

| CONCEPT                                                                                                      | RELIABILITY | VERSATILITY | FABRICABILITY | REMOTE HANDLING | SPECIMEN ACCESS | VOLUME UTILIZATION | SEALING | COMMENTS                        |                                                |
|--------------------------------------------------------------------------------------------------------------|-------------|-------------|---------------|-----------------|-----------------|--------------------|---------|---------------------------------|------------------------------------------------|
|                                                                                                              |             |             |               |                 |                 |                    |         | DUAL HORIZONTAL ACCESS REQUIRED | TO REMOVE SPECIMENS ENTIRE MODULE IS WITHDRAWN |
| <br>1. SINGLE UNIT MODULE | +           | +           | +             | +               | -               | +                  | +       | +                               |                                                |
|                                                                                                              |             |             |               |                 |                 |                    |         |                                 |                                                |
| <br>2. CLAMSHELL MODULE  | +           | +           | O             | O               | -               | +                  | +       |                                 |                                                |
|                                                                                                              |             |             |               |                 |                 |                    |         |                                 |                                                |

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# TEST MODULE CONCEPTS

TABLE VII.2-6 (cont.)

| CONCEPT                                                                                                               | RELIABILITY | VERSATILITY | FABRICABILITY | REMOTE HANDLING | SPECIMEN ACCESS | VOLUME UTILIZATION | SEALING | COMMENTS                                                |
|-----------------------------------------------------------------------------------------------------------------------|-------------|-------------|---------------|-----------------|-----------------|--------------------|---------|---------------------------------------------------------|
|                                                                                                                       |             |             |               |                 |                 |                    |         |                                                         |
|  <p>3. MULTIUNIT MODULE</p>        | ○           | +           | -             | ○               | ○               | ○                  | -       | SPECIMENS CAN BE REMOVED WITHOUT REMOVING ENTIRE MODULE |
|                                                                                                                       |             |             |               |                 |                 |                    |         |                                                         |
|  <p>4. DIRECT ACCESS MODULES.</p> | ○           | ○           | ○             | -               | +               | ○                  | ○       | REQUIRES 360° ACCESS                                    |
|                                                                                                                       |             |             |               |                 |                 |                    |         |                                                         |

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The three other TM concepts are considered to be viable; however, they are progressively more complex and have more "parts", require more access space and in general are considered less reliable and harder to handle remotely. A more detailed description of the TM design will now be given.

#### VII.2.3.2.2 Test Module Design Details

The major components of the TASKA materials test module, shown in Fig. VII.2-8, are the helium cooled heat sink (annular test module), the biological shield and the specimen capsules. The temperature within each of the 351 specimen capsules is pre-determined by engineering the thermal barrier between the specimen capsule and the heat sink. Individual capsules can be independently accessed and are designed to operate at a predetermined temperature within the range of 300 to 650°C. The total irradiation volume within a single TASKA test module is 138 liters. Table VII.2-7 lists the detailed test module design values determined during the initial concept design.

Test Module. TASKA allocates one reactor opening for materials testing - located between two of the central cell magnets. The test module is inserted into TASKA vertically through a rectangular 140 x 200 cm opening in the vacuum vessel. Appendix VII.2-B illustrates test module design details and location within the reactor. The annular test module consists of a titanium, helium filled pressure vessel structure which is penetrated by 351 capsule holes. The capsule holes are arranged with their long axes along radial lines with the open end of the capsule hole facing away from the plasma. The capsule holes are arranged in staggered rows on a triangular pitch. Helium flows circum-ventially inside the module perpendicular to, and outside of, the capsule holes. Instrumented specimen capsules shown in Fig. VII.2-9 can be inserted in any of the 351 capsule holes. The specimen capsule outside diameter is

Table VII.2-7. Design Values

Reactor Opening

- Maximum axial opening of 95 cm

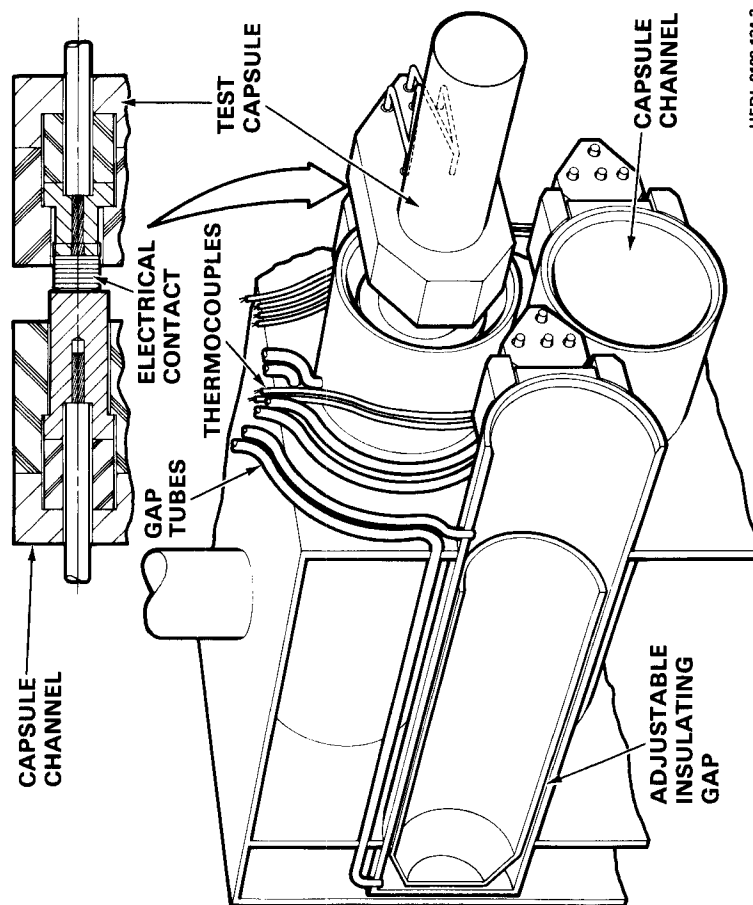
Test Module

- Material of construction: 0.635 cm thick titanium alloy (Ti-6Al-4V)
- Size: Inside radius - 50 cm  
Outside radius - 70 cm  
Length - 69.2 cm (end to end) 44.2 cm straight section
- No. of capsule tubes: 351
- Total heat generation: 1145 kW
- Coolant: helium at 3448 kPa and 125°C
- Flow rate:  $72.9 \times 10^3$  l/min
- Coolant pipe size: 20 cm diameter
- Instrumentation per capsule tube: 2 pair thermocouple leads - chromel-alumel  
2 gas leads - 0.158 cm diameter for gas gap control

Specimen Capsule

- Material of construction: 316 stainless steel
- Size: 5 cm diameter by 20 cm long
- Specimens: various shapes packed to 50% of capsule volume in NaK (22/78)
- Instrumentation: (2) 0.158 cm diameter stainless steel sheathed MgO insulated thermocouples
- Sealing: all welded construction with bellows for NaK expansion
- Gas gap: variable annulus 0.018-0.076 cm filled with helium or neon or mixtures of the two gases

FIGURE VII.2-9



HEDL 8108-134.2

spaced from the inside wall of the capsule hole by a gas gap. This gap becomes "sealed" when the specimen capsule is inserted into the capsule hole.

Neutron and gamma heat deposited in the test specimens is transferred by conduction via the NaK liquid metal to the capsule wall, across the insulating gap, through the outer wall, which is cooled on the outside by the circulating helium system. Two 1.6 mm O.D. tubes are provided to access the insulating gap near the top and bottom of each of the 351 capsule holes. These tubes can provide inert gas to the insulating gap space. Two (2) lines are provided to the gap to permit either a static or dynamic supply of gas. In addition, they allow for on-line change-out of gas.

Two pair of thermocouple leads are connected to each of the 351 capsule holes and are routed between capsules along the top front surface of the module. The electrical leads terminate in an electrical connector which is positioned near the entry of each capsule hole. This connector becomes mated with the specimen capsule thermocouple leads when the capsule is inserted in the capsule hole. The electrical leads (702 pair of thermocouples) and tubes (702) are routed from the module, along the helium coolant lines through stepped openings in the shield plug. From there they are run to instrument cabinets located in the experiment support area.

Coolant System. The coolant system is designed to remove heat from the test module structure and the specimen capsules contained within it. The various coolant media considered are listed in Table VII.2-8; they are: water, sodium and helium. From a heat transfer and remote handling standpoint, water is the most effective coolant. However, as discussed in Section VII.2.1, the presence of water in the test module moderates the high energy neutrons to such a degree that their damage energy is reduced by a factor of

Table VII.2-8. Comparison of Coolants for TASKA Test Module

|                                         | <u>Water</u>              | <u>Sodium*</u>            | <u>Helium</u>              |
|-----------------------------------------|---------------------------|---------------------------|----------------------------|
| DPA reduction to test specimen          | Large                     | Medium                    | None                       |
| MHD effects                             | Medium                    | Large                     | None                       |
| Compatibility with structural materials | Good                      | Medium                    | Best                       |
| Pressure required, psig                 | 100                       | 30                        | 500                        |
| (kPa)                                   | (690)                     | (207)                     | (3448)                     |
| Pipe size to module, inches             | 1-1/2                     | 2                         | 8                          |
| (cm)                                    | (3.8)                     | (5.0)                     | (20)                       |
| Coolant circulation component           | Centrifugal pump          | E.M. Pump                 | Compressor blower          |
| Activation of coolant                   | Some                      | Severe Na <sup>22</sup>   | Slight to none             |
| Structural materials                    | Thin wall stainless steel | Thin wall stainless steel | Thick wall stainless steel |
| Stress levels                           | Low                       | Very low                  | High                       |
| Heat transfer coefficient (H)           | Medium                    | High                      | Low                        |

\*Sodium requires trace heating and cannot be operated below 125°C.



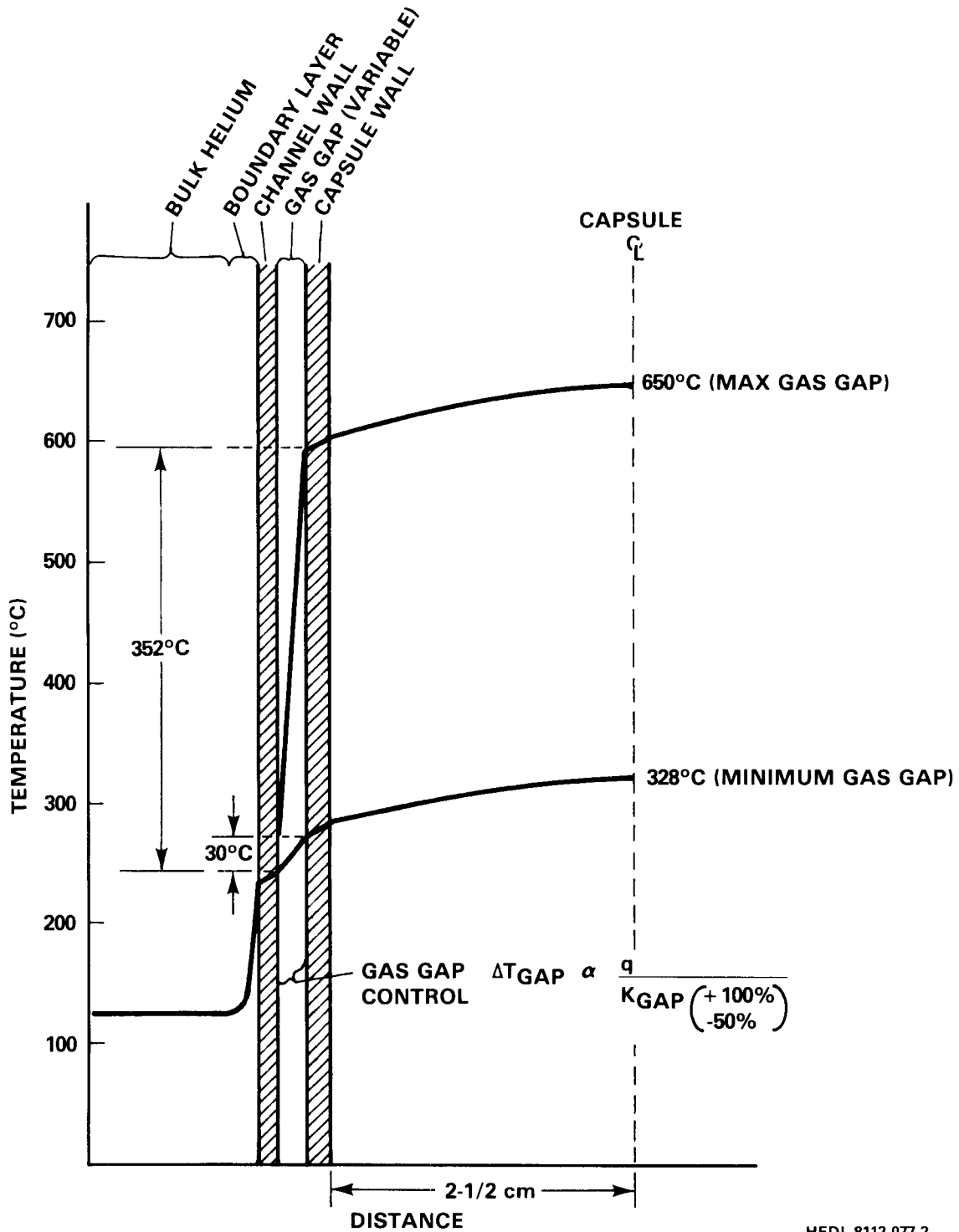
~ 2. Sodium is difficult to handle, becomes highly activated, requires a trace heat system and has an adverse effect on neutron damage energy (though not as great as water). Helium, while not as good as water or sodium from a heat removal standpoint, is inert, relatively easy to handle and has minimal impact on the dpa-volume as discussed in Section VII.2.1. Therefore, helium was selected as the reference coolant for the initial TM conceptual design.

To evaluate the performance of the helium cooled TM the temperature profile from the bulk helium coolant to the center of the specimen capsule was calculated. Figure VII.2-10 illustrates the results of one such calculation for the case in which the total heat removed from the specimen capsule was uniformly averaged over the capsule wall area (i.e., heating gradients were not considered). The average neutron induced heating rate was assumed to be approximately  $5 \text{ W/cm}^3$  over the capsule volume. For the materials within the capsule,  $5 \text{ W/cm}^3$  corresponds to  $1.26 \text{ W/g}$ . Table VII.2-9 summarizes the temperature profile data for the average case (Fig. VII.2-10) and for the region of the specimen capsule receiving the maximum power input (region closest to plasma).

As heat flows from the inside of the specimen capsule to the helium coolant it encounters its greatest resistance at (1) the engineered gas gap, and (2) at the boundary layer between the helium coolant and the outside of the capsule hole. The thermal conductivity of the engineering gas gap can be controlled by varying the composition of gas (helium and neon) within the gap. By varying the mixture of the two gases, it is possible to vary the thermal conductivity ( $K$ ), and hence the temperature drop across the gap ( $\Delta T$ ), by approximately +100%, -50%. This allows the temperature of the capsule to be controlled to a certain degree. The range of control is, however, strongly

FIGURE VII.2-10

TEMPERATURE PROFILE FROM HELIUM COOLANT INTO  
SPECIMEN CAPSULE ( $q = 1.26 \text{ W/gm}$ )



HEDL 8112-077.2

Table VII.2-9. Test Module Temperature Profile

Basis: Avg. Heating Rate + 1.26 W/g (Max. = 2.57 W/g)

Bulk Helium  $\Delta T = 50^{\circ}\text{C}$

| Helium Convective Heat Transfer             | Temp. $^{\circ}\text{C}$ | Temp. $^{\circ}\text{C}$ |
|---------------------------------------------|--------------------------|--------------------------|
| Coefficient = $794 \text{ W/m}^2\text{C}^*$ | <u>Average</u>           | <u>Maximum</u>           |
| Helium Inlet                                | 100 <sup>**</sup>        | 100 <sup>**</sup>        |
| Bulk Helium $\Delta T$                      | 50                       | 50                       |
| Helium Outlet                               | 150                      | 150                      |
| Helium Film $\Delta T$                      | 86                       | 174                      |
| Capsule Tube Outside                        | 246                      | 234                      |
| Capsule Tube $\Delta T$                     | 10                       | 20                       |
| Capsule Tube Inside                         | 256                      | 344                      |
| Gas Gap $\Delta T$ (Variable)               | 30-352                   | 30-222                   |
| Specimen Capsule $\Delta T$                 | 42                       | 84                       |
| Capsule Center                              | 328-650                  | 458-650                  |

\*140 BTU/hr·ft<sup>2</sup>°F for capsule tube with extended surface.

\*\*Provides  $\Delta T$  of  $75^{\circ}\text{C}$  to cool with water at  $25^{\circ}\text{C}$ .

<sup>+</sup>Based on  $1.8 \text{ MW/m}^2$ .

dependent upon the magnitude of the  $\Delta T$  across the gap. As illustrated in Fig. VII.2-10 for the case in which the capsule centerline temperature is 650°C, the  $\Delta T_{\text{gap}}$  (352°C) represents a significant fraction of the total temperature drop between the capsule and bulk helium coolant. For the low temperature capsule, the  $\Delta T_{\text{gap}}$  (30°C) is only a small fraction of the total  $\Delta T$ , hence, temperature control is virtually lost.

The difficulty in achieving temperature control (as well as the minimum required temperature of 300°C) is due to the large helium boundary layer temperature drop. The boundary layer dominates the  $\Delta T$  between the bulk helium coolant and specimen capsule centerline temperature (at temperatures approaching  $T_{\text{helium}}$ ). The obvious solutions to this problem are to (1) increase the velocity of the helium, which reduces the boundary layer  $\Delta T$ , and/or (2) to reduce the inlet helium temperature which allows the  $\Delta T_{\text{gap}}$  to become a larger fraction of the total  $\Delta T$ . Both solutions were considered to be impractical due to the already significant helium mass flow rates ( $73 \times 10^3$  l/min) and power (1145 kW) contained in the helium. Other means of attaining a lower and more controllable minimum temperature include reduction of specimen capsule diameter and optimization of overall test module mass. Further design is required to access these options.

Specimen Capsule (Fig. VII.2-9). Each specimen capsule is a circular stainless steel cylinder 5 cm diameter by 20 cm long. Material test specimens are positioned inside each capsule. A NaK liquid metal heat transfer bond is provided between test specimens and capsule wall. Each capsule is provided with two or more chromel-alumel thermocouples for measuring test specimen temperature. The capsule wall thickness is adjusted during fabrication to provide a gap between it and the capsule hole. This gap is sized so that when

it is filled with a particular insulating gas it will provide the desired test specimen temperature. This gap can be of uniform width or it can be stepped along the capsule axis to compensate for different heating rates between the front and the back end of the capsule. The top of each capsule includes a cylindrical canopy which seals the gap when the capsule is inserted in the tube. A metal bellows with a compressible space is provided for thermal expansion of the NaK bond.

To cover the entire temperature range (300-650°C) requires relatively high and low thermal conductivity media in the gap. Helium and neon were selected - helium because of its good thermal conductivity and neon because of its lower thermal conductivity and reduced activation (when compared to argon). The practical range of gas gaps is approximately 0.0175 to 0.10 cm. A 0.0175 cm gap is believed to be the minimum controllable gap width from a fabrication standpoint and for gap widths above approximately 0.10 cm convection starts adding to conduction which makes temperature prediction less reliable.

In summary, for gas gap widths ranging from 0.0175 to < 0.10 cm filled with an atmosphere of helium or neon, the range of average specimen capsule temperatures is 328 to 650°C. For that portion of the specimen capsule receiving the maximum neutron heating rate ( $\sim 10 \text{ W/cm}^3$ ) the attainable specimen capsule temperatures are 458-650°C. (This assumes no axial heat transfer - only radial.) Hence, the initial design concept does not meet the specified low temperature criteria of 300°C. Design optimization and further thermal analysis will be required to meet the specified temperature criteria.

### VII.2.3.3 Test Module Operations and Handling

The frequency of test module removal is a function of the test matrix fluence requirements discussed in Section VII.2.3.1 and the reliability of the test module. If one looks only at the fluence requirements the materials test module will be removed from the reactor and transported to a disassembly hot cell every ~ 500 full power days (Table VII.2-4). There will be additional removals from the reactor as random failures occur within the test module and test module control and data acquisition equipment.

When problems and failures occur within the test module the entire test module will have to be removed from the reactor for repair. The design does not accommodate maintenance internal to the vacuum flange while in the reactor. Therefore, in order to maximize TASKA availability a second test module should always be available for insertion when the primary module requires repair. The transfer of the module from reactor to the disassembly hot cell should be a simple, efficient operation with minimal impact on other reactor systems. Simplicity in module removal is the principal reason vertical access was chosen for the materials test module. It is believed that vertical access has minimal impact on surrounding reactor support systems, requires minimal floor space and involves straight line, vertical lifts using only an overhead crane.

The operations required to remove a test module from TASKA are given in Table VII.2-10.

### VII.2.3.4 Summary and Conclusions

A materials testing scenario for TASKA was presented in the form of a test matrix. The test matrix lists the type of materials test, the irradiation temperatures, goal fluences and details relating to stresses and

Table VII.2-10. Operations Required for Test Module Removal from Reactor

| <u>Operation</u>                                                                                  | <u>Comment</u>                                                                        | <u>Approximate Time<br/>to Complete<br/>Operations (hrs)</u> |
|---------------------------------------------------------------------------------------------------|---------------------------------------------------------------------------------------|--------------------------------------------------------------|
| 1. Following reactor shutdown and cooldown period, disconnect helium piping and instrument leads. | Hands-on operation.                                                                   | 16                                                           |
| 2. Unbolt flange from vacuum boundary of machine.                                                 | Hands-on.                                                                             | 4                                                            |
| 3. Unbolt large module cover-plate from pipe penetration flange.                                  | Hands-on.                                                                             | 4                                                            |
| 4. Swing cover-plate over helium piping and remove. Protect sealing surfaces.                     | Hands-on.                                                                             | 8                                                            |
| 5. Secure OH crane to test module. Lift entire test module from reactor.                          | As soon as lift begins operations become remote.                                      | 4                                                            |
| 6. Transfer test module to hot cell for disassembly (or load TM into a large cask for transfer).  | Totally remote viewing and handling.                                                  | 3                                                            |
| 7. Place test module in rotating, indexing figure in hot cell.                                    | Hot cell operation utilizing crane, electro-mechanical and master slave manipulators. | 8                                                            |

Table VII.2-10 (Cont'd.)

|                                                                                                                 |                                                                              |              |
|-----------------------------------------------------------------------------------------------------------------|------------------------------------------------------------------------------|--------------|
| 8. Remove shield plug and shear vacuum cover seal which covers specimen capsule tubes.                          | Remote operation.                                                            | 16-32        |
| 9. Withdraw specimen capsules from capsule holes and replace with new capsules.                                 | Utilize rotating removal machine which indexes on each row of capsule holes. | 24-48        |
| 10. Verify integrity of test module and all specimen capsules. Leak test system, service lead continuity check. | All systems and components must be verified operable.                        | 48-120       |
| 11. Replace vacuum cover and weld, reassemble shield plug to module. Prepare for transfer back to reactor.      | ---                                                                          | 48-72        |
| Total Time:                                                                                                     |                                                                              | 183-319 hrs. |

The times estimated in Table VII.2-10 are actual working hours (assumes 60 minutes are worked each hour) assuming success-oriented operations.



postirradiation test conditions. It was shown that the entire TASKA test matrix consisting of nearly 30,000 test specimens could be completed in approximately eight full power years utilizing on the average less than half of the volume in a single test module. The maximum fluence achieved by any group of specimens is over 100 dpa.

The test module was designed to contain and thermally control the test specimens within the vacuum boundary of the reactor. Helium was chosen as the bulk heat removal fluid because it has the least moderating effect upon the neutrons and therefore results in the greatest dpa-volume for a given test module space envelope. Helium is acceptable from a heat removal standpoint; however, further design optimization will be required to verify the limits of test module operation.

The operational impacts of materials testing upon TASKA should be minimal as long as reliable test modules can be designed and fabricated. The test module concept presented in this section requires minimal handling in the vicinity of the reactor - the majority of test module remote operations occur during test module disassembly which occurs in hot cell facilities removed from the reactor operating gallery.

## VII.2.4 Neutronics Analysis for the REGAT Module of TASKA

### VII.2.4.1 Introduction

One major requirement of TASKA is to provide a capability to test materials for the mirror program and preferably for the tokamak program as well. Testing materials for the tokamak program requires including fatigue and crack growth specimens. While the number of such specimens is not great, the volume per specimen is large. The TASKA materials testing module is designed to accommodate specimens to test materials for both programs. Results of the neutronics analysis for the REGAT module of TASKA are given here and compared to the results for the current INTOR materials test module.

### VII.2.4.2 Neutronics Results

The improved REGAT design shown in Fig. VII.2-1-c is used in TASKA. The module has an inner radius of 46 cm and a thickness of 20 cm. The neutronics results presented here are given for two designs. In the first design, the REGAT module is cooled by helium and has a width of 70 cm, yielding a total module volume of 493 liters. In the second design, water is used as a coolant and a width of 50 cm is used, yielding a total module volume of 352 liters. Water was considered as a coolant because of the difficulty of maintaining low irradiation temperatures with helium cooling. A smaller module width is used in the water cooled case in order to compensate for the increased damage gradients resulting from higher neutron attenuation in water. In this case, the module is still large enough to accommodate specimens to test materials for both the mirror and tokamak programs. The current INTOR materials testing module has a volume of 135 liters.

In TASKA, the test capsules represent 40 v/o of the module. The capsules consist of 50 v/o NaK, which is used as a thermal contact material, and 50 v/o

316 SS, which represents the specimens and the capsule structural material. The module structure represents 20% of the module volume and is made of Ti-6Al-4V and 316 SS in the gas and water cooled cases, respectively. Water occupies 15 v/o of the water cooled volume. Because of severe neutron attenuation in water, an effort was made to reduce the water content in the REGAT module of TASKA. The INTOR materials test module has a 36 v/o of water. Table VII.2-11 gives the test module composition for INTOR and the two designs used in TASKA.

The neutronics results presented in this section are based on the initial assumption that the test capsules represent 40% of the module volume. However, later in the study when the modules were designed in detail, it was found that the cylindrical capsules represent only 28% of the module volume because of the annular geometry of the module. The later number corresponds to a 30% reduction in the capsule volume. However, the lower steel content will result in  $\sim 10\%$  increase in the average dpa, or a 20% decrease in the dpa $\cdot x$  value. The 40% capsule volume fraction can be easily maintained by designing conically shaped test capsules with diameters of 5 and 5.37 cm at the front and back of the module, respectively. Therefore, we will address the 40% volume fraction case in the rest of this chapter.

For the purpose of comparison, one-dimensional neutronics calculations have been performed for the current INTOR materials test module which is 15 cm thick and 135 liters in volume. Figure VII.2-11 shows the variation of dpa rate with depth in the test module as obtained from the one-dimensional cylindrical calculations. The results are given for the test module of INTOR and the two designs used in TASKA. Note that the results for TASKA give the conditions at the center of the module with effects of using the REGAT design

Table VII.2-11. Test Module Composition

|                                   | INTOR                      | H <sub>2</sub> O Cooled<br>TASKA | He Cooled<br>TASKA  |
|-----------------------------------|----------------------------|----------------------------------|---------------------|
| Module structure                  | 316 SS<br>24 v/o           | 316 SS<br>20 v/o                 | Ti-6Al-4V<br>20 v/o |
| Specimen and<br>capsule structure | 316 SS<br>17 v/o           | 316 SS<br>20 v/o                 | 316 SS<br>20 v/o    |
| Coolant                           | H <sub>2</sub> O<br>36 v/o | H <sub>2</sub> O<br>15 v/o       | He<br>40 v/o        |
| Thermal contact<br>material       | NaK<br>17 v/o              | NaK<br>20 v/o                    | NaK<br>20 v/o       |
| Void                              | 6 v/o                      | 25 v/o                           | 0 v/o               |

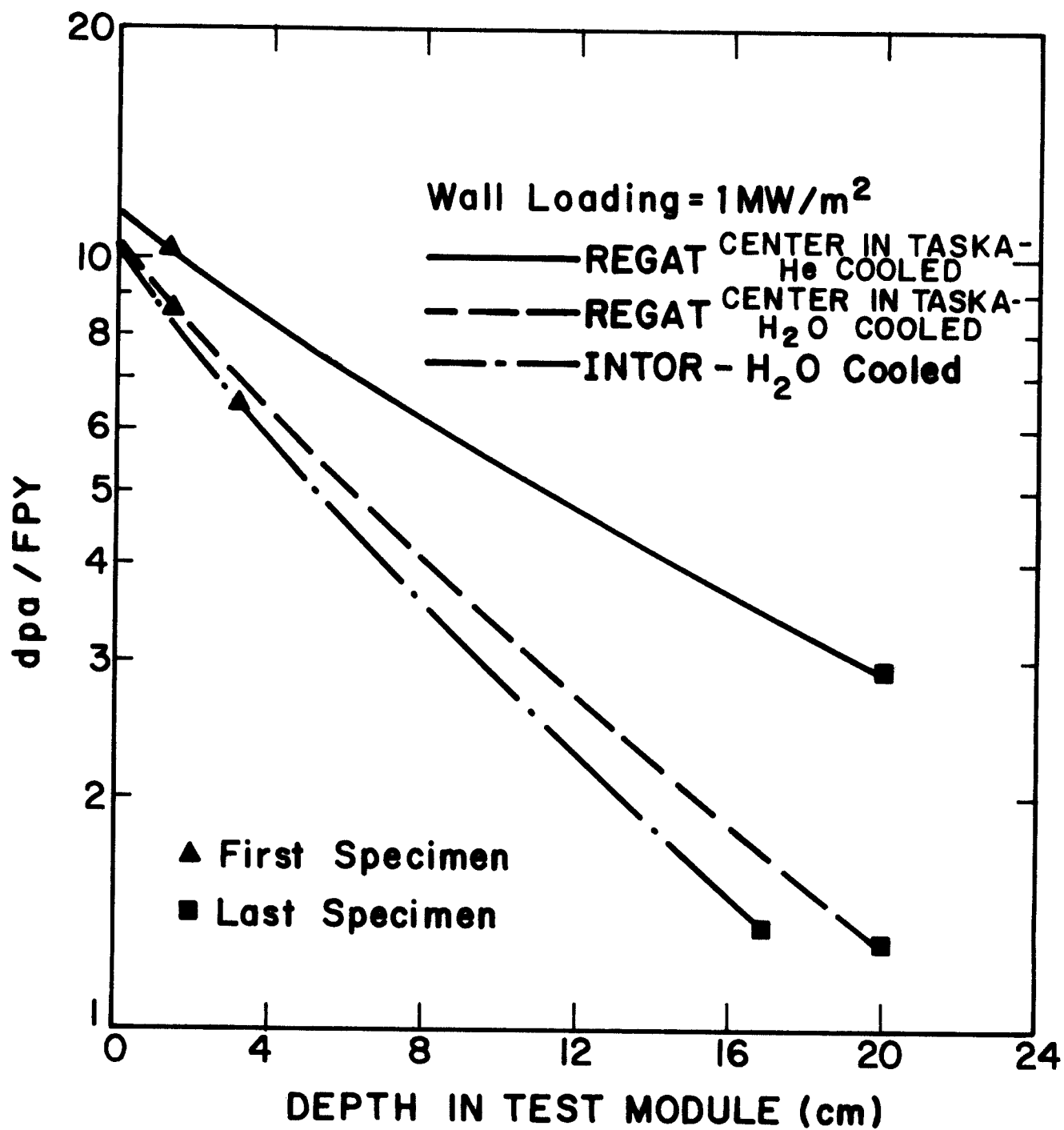


Fig. VII.2-11 One-dimensional results for dpa in TASKA and INTOR modules.

not shown. The results are normalized to unit wall loading. Because of the high heat flux in INTOR, a 19.6 mm thick separate water cooled first wall is required. The module and capsule walls have a total thickness of 13 mm. This implies that the specimen exposed to the highest irradiation level is placed at a depth of 32.6 mm from the first surface in INTOR and 13 mm in TASKA. This results in a smaller peak damage rate in INTOR.

Detailed three-dimensional neutronics calculations were performed to calculate the damage profiles in the REGAT materials testing module designs of TASKA. Figures VII.2-12 and VII.2-13 show the equal dpa contours for the gas and water cooled cases, respectively. The results are given for a fluence of  $1 \text{ MW} \cdot \text{yr}/\text{m}^2$ . Note that if the traditional design is used, these contours will reduce to a set of parallel horizontal lines. The effect of using the REGAT design on reducing the damage gradient is clear. These results suggest also that small test specimens might be irradiated at the center of the module while larger specimens, such as those required for the tokamak program, can be irradiated at the sides of the module. It is clear that the water cooled design results in higher damage gradients because of the increased neutron attenuation in water.

Figures VII.2-14 and VII.2-15 give the equal helium appm contours for the gas and water cooled cases, respectively. Again, it is clear that the gas production gradient is reduced as a result of using the REGAT design. Since the threshold energy for the helium production reaction is much higher than that for atomic displacement, the helium production decreases sharply as one moves from the side of the module towards the center. Hence, using the REGAT design results in a more pronounced effect on reducing dpa gradients than on reduced helium production gradients. Higher helium production gradients are

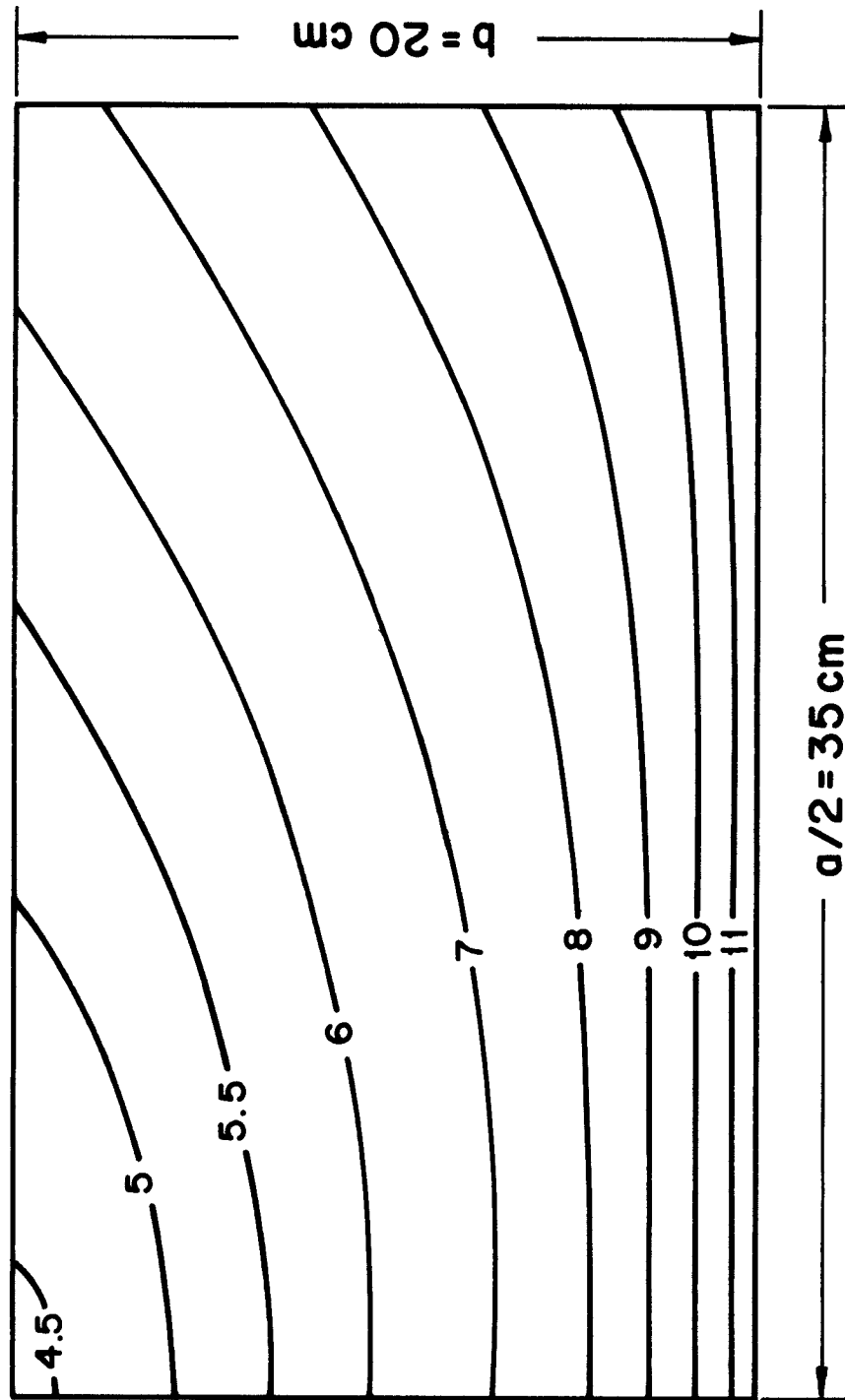


Fig. VII.2-12 Equal dpa contours in the gas cooled REGAT module.

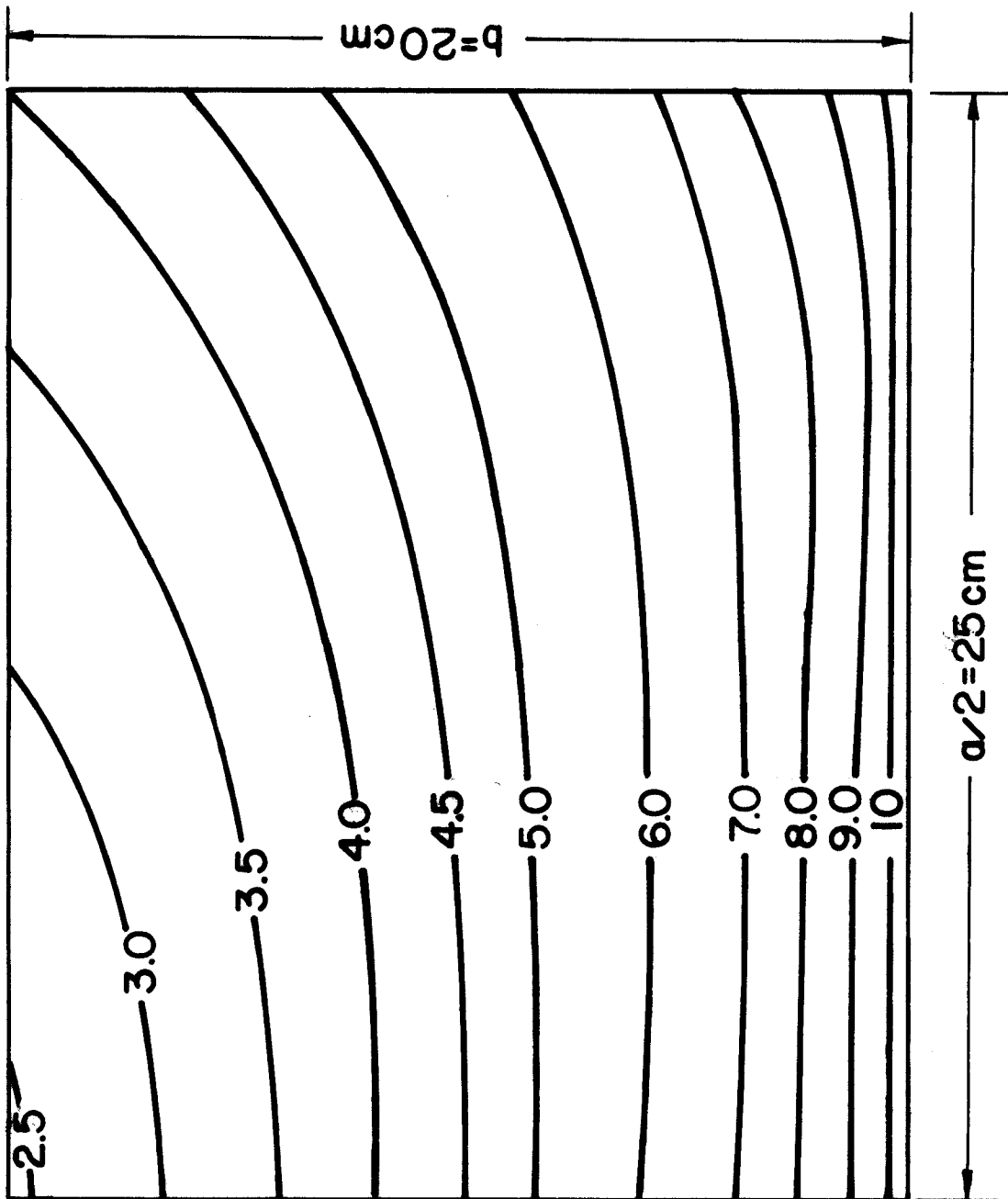


Fig. VII.2-13 Equal dpa contours in the water cooled REGAT module.



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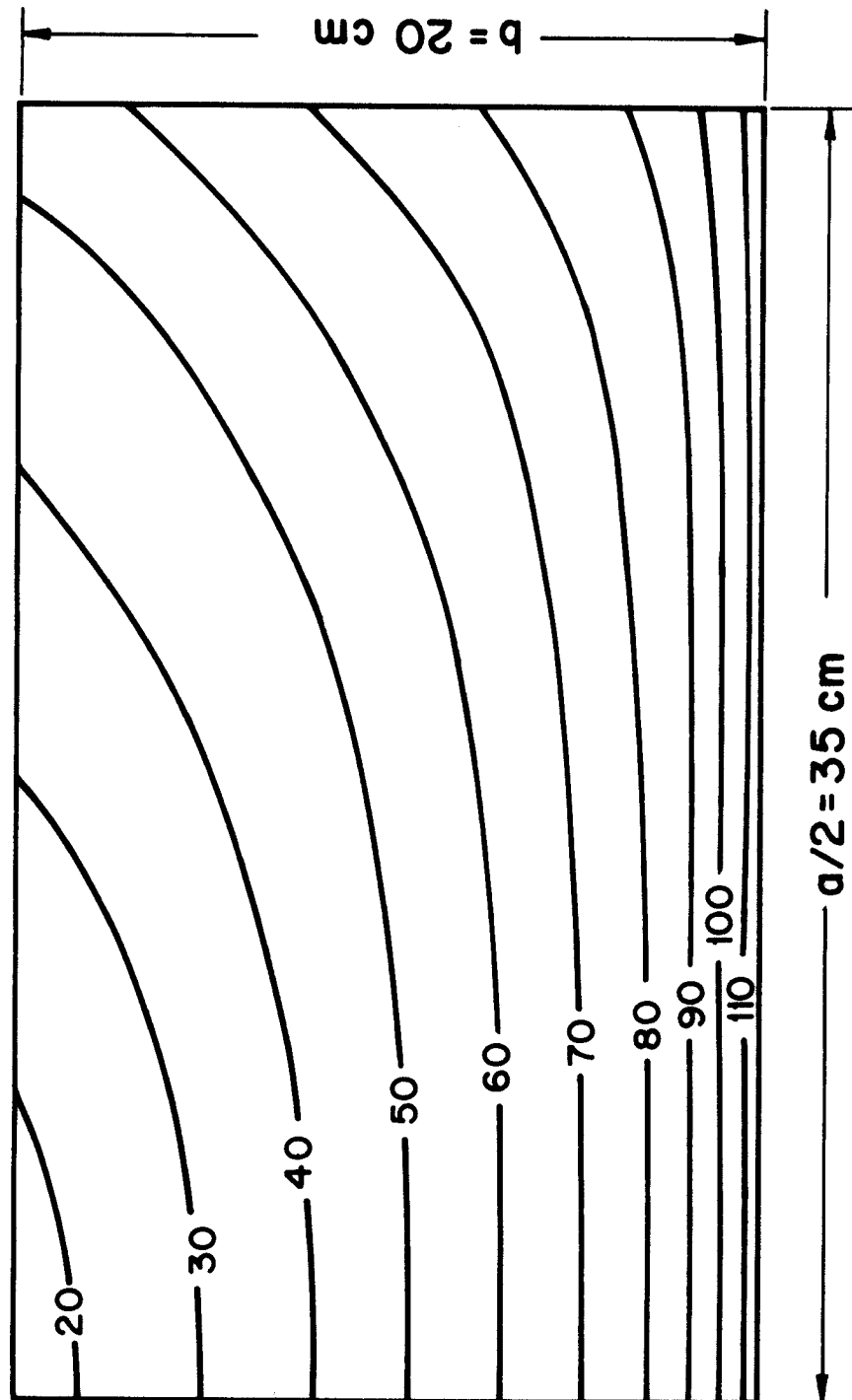


Fig. VII.2-14 Equal helium appm contours in the gas cooled REGAT module.

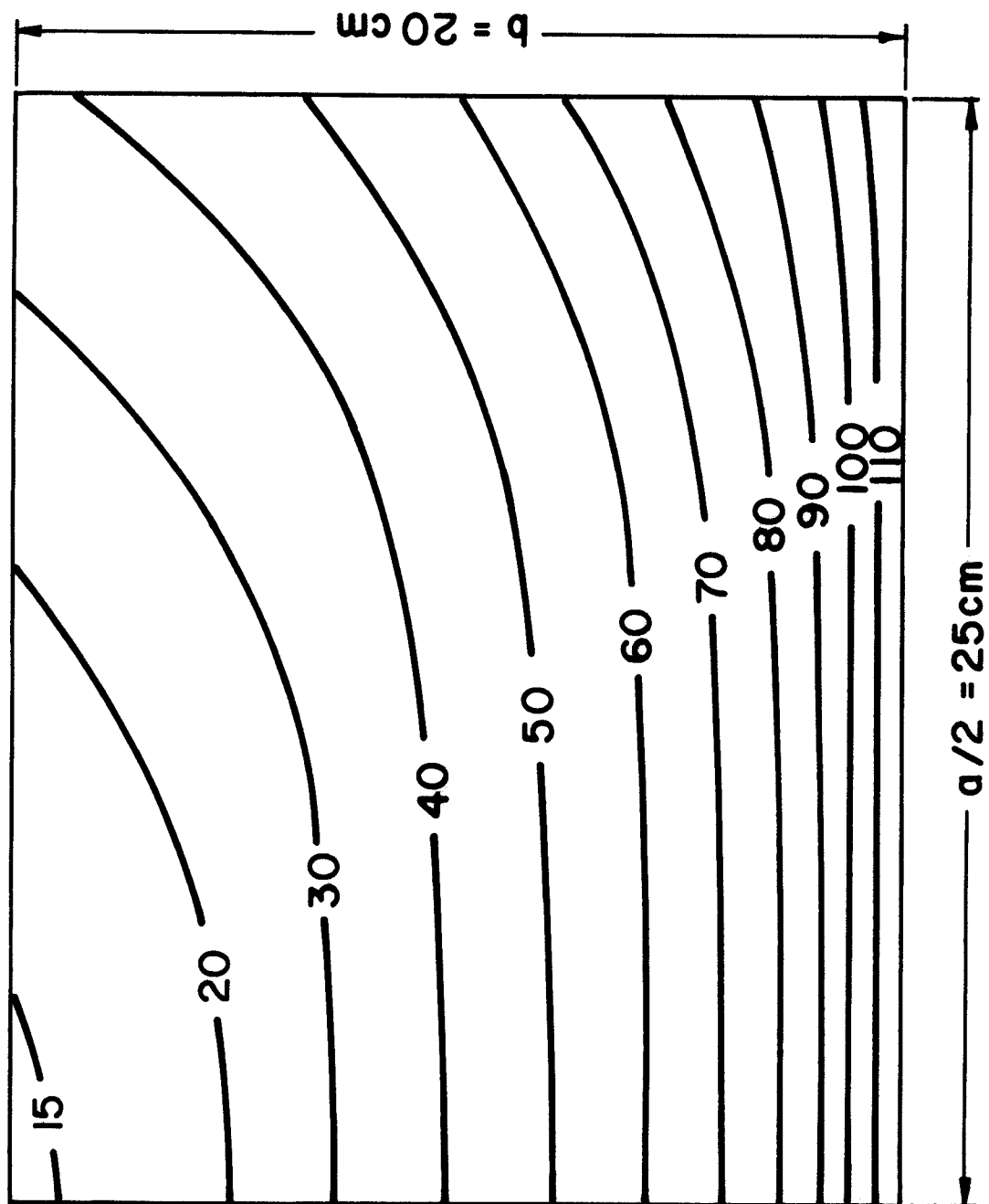


Fig. VII.2-15 Equal helium appm contours in the water cooled REGAT module.

obtained when water is used as compared to the gas cooled case. However, the effect on helium production, which is produced by neutrons with energies  $\gtrsim 1$  MeV, is not as pronounced as that on dpa which can be produced by neutrons with energies as low as  $\sim 1$  keV.

Figures VII.2-16 and VII.2-17 show the dpa levels that can be obtained in different volumes of the gas and water cooled modules, respectively. Figures VII.2-18 and VII.2-19 give the results for helium production. For the purpose of comparison, the results for the traditional design are also included. The results are normalized to a wall loading of  $1 \text{ MW/m}^2$ . It is clear that in using the REGAT design, one can achieve higher damage levels using the same test volumes. The results show that using the REGAT design has a more pronounced effect on increasing dpa levels than on increasing gas production levels. It is clear also that the improvement over the traditional design is more pronounced in the water cooled case than in the gas cooled case.

Table VII.2-12 gives the average and peak/average dpa and helium production rates in the gas and water cooled REGAT modules of TASKA and the INTOR materials testing module. The results are normalized to a unit wall loading. Coupled neutronics and photonics calculations were performed to calculate the power density profiles in the different modules. The results are summarized in Table VII.2-13. The results are normalized to a unit wall loading.

Another way of expressing the damage in the test specimen is to sum the product of the damage level times the volume of the test zone that can produce that damage level. This parameter accounts for the number of specimens that can be irradiated to high levels. It can be calculated by multiplying the average damage level by the total test capsule volume in the module. For a unit fluence of neutron irradiation, the dpa times capsule volume values are

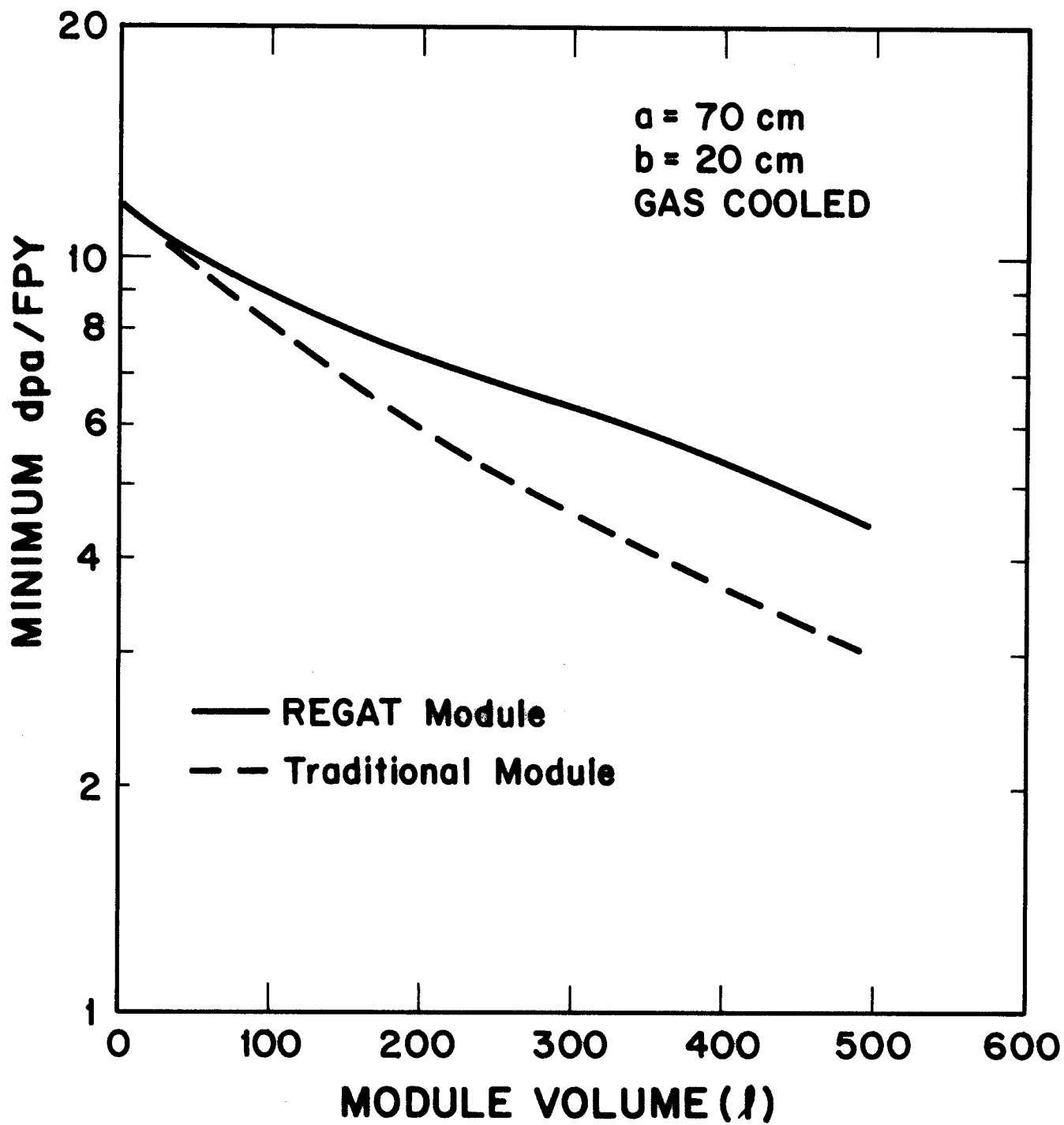


Fig. VII.2-16 Dpa rate vs. volume in the gas cooled REGAT module.

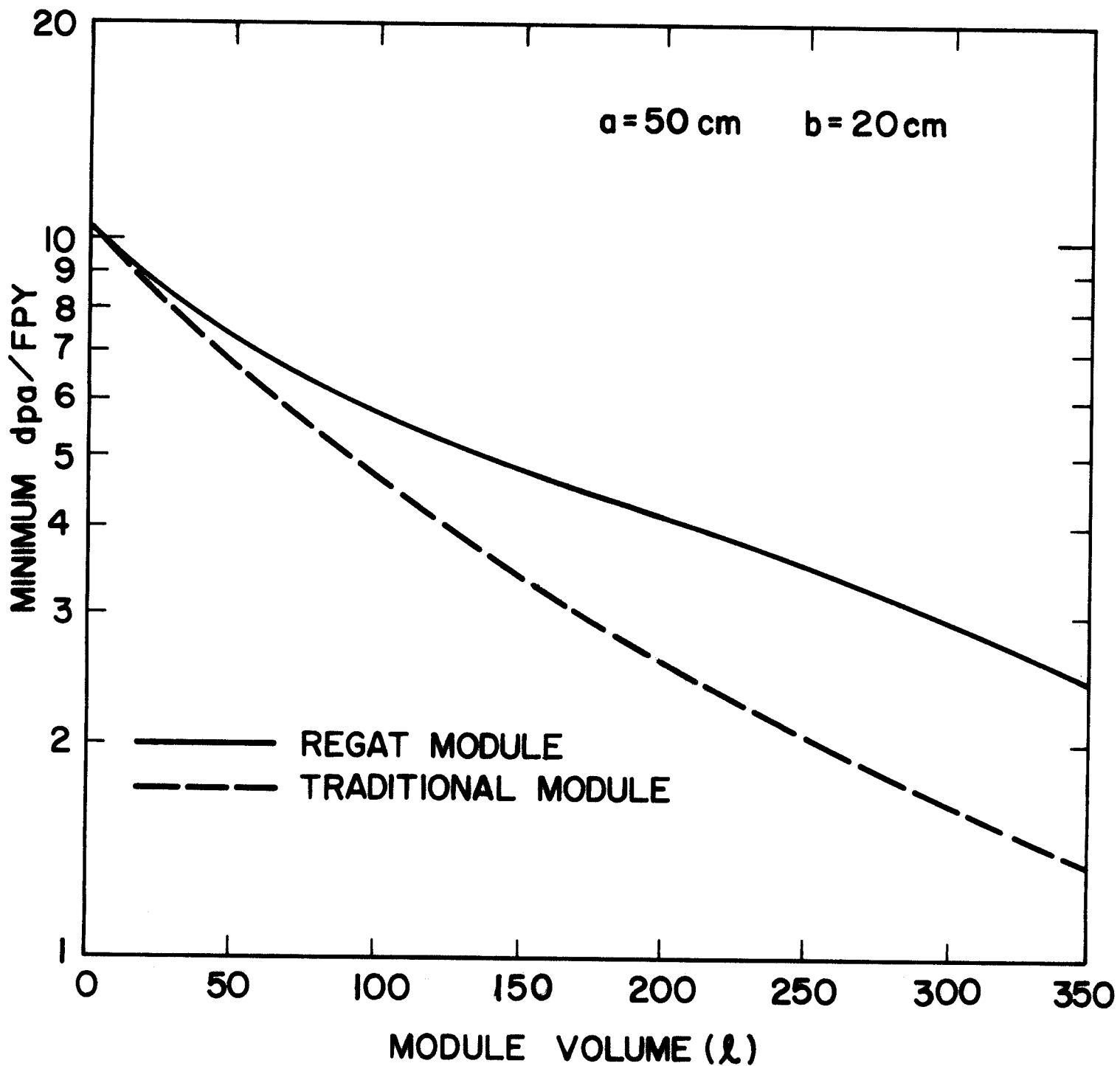


Fig. VII.2-17 Dpa rate vs. volume in the water cooled REGAT module.

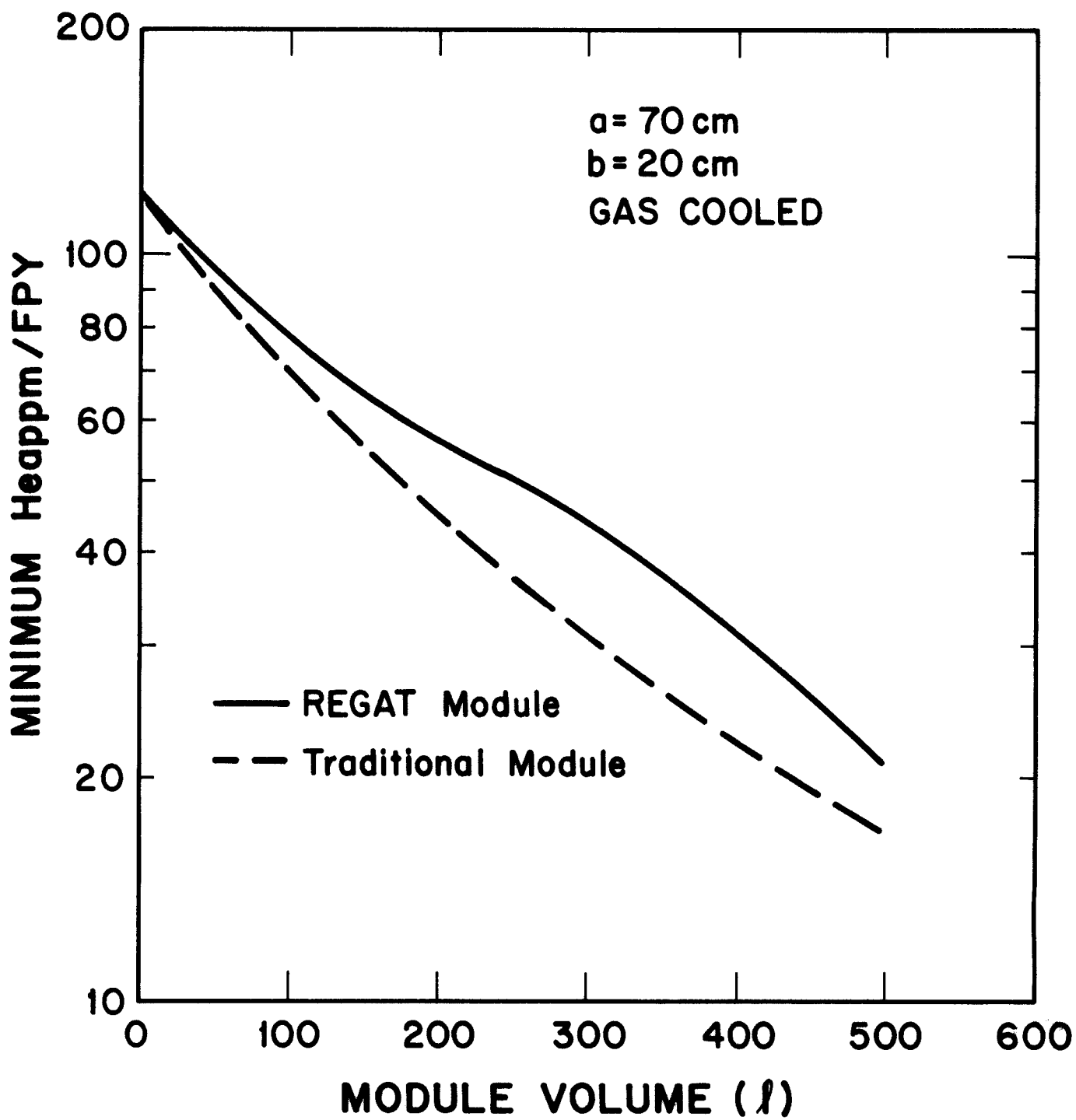


Fig. VII.2-18 Helium appm rate vs. volume in the gas cooled REGAT module.

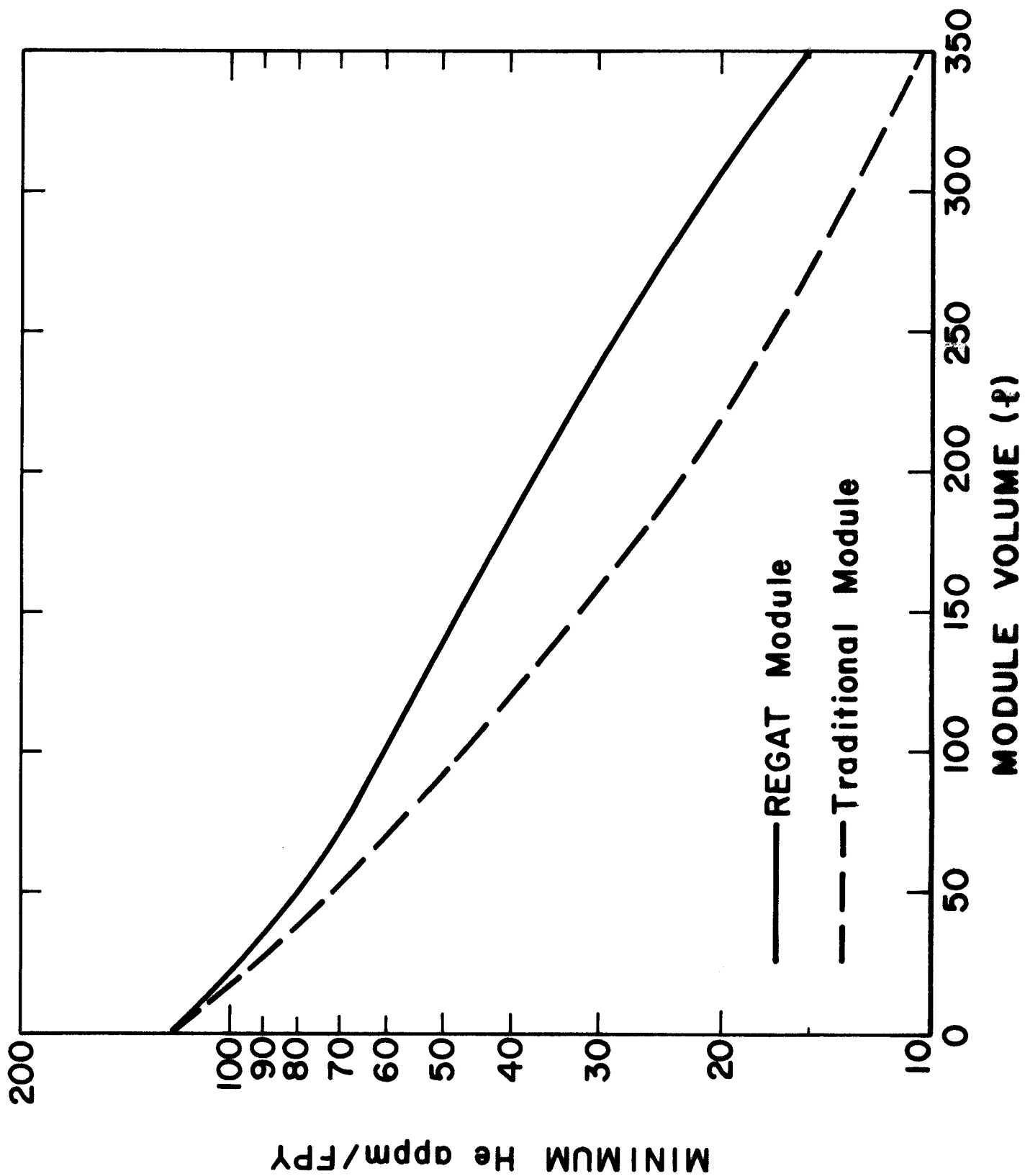


Fig. VII.2-19 Helium appm rate vs. volume in the water cooled REGAT module.

Table VII.2-12. DPA and Helium Production Rates in the  
Different Test Modules for Unit Wall Loading

|                    | dpa/FPY |               | He appm/FPY |               |
|--------------------|---------|---------------|-------------|---------------|
|                    | Ave.    | Peak/<br>Ave. | Ave.        | Peak/<br>Ave. |
| Gas cooled REGAT   | 7.30    | 1.56          | 49.03       | 2.12          |
| Water cooled REGAT | 5.14    | 1.96          | 44.00       | 2.31          |
| Water cooled INTOR | 3.90    | 2.24          | 40.00       | 2.59          |

Table VII.2-13. Power Densities in the Different Test  
Modules for Unit Wall Loading

|                    | Power Density (W/cm <sup>3</sup> ) |         |         |
|--------------------|------------------------------------|---------|---------|
|                    | Peak                               | Average | Minimum |
| Gas cooled REGAT   | 3.87                               | 2.15    | 1.78    |
| Water cooled REGAT | 7.74                               | 4.84    | 2.46    |
| Water cooled INTOR | 10.69                              | 6.28    | 3.37    |



1438, 725, and 176 dpa  $\lambda$  for gas cooled REGAT, water cooled REGAT, and INTOR, respectively. The values for helium appm times capsule volume are 9659, 6204, and 1800 He appm  $\lambda$ , respectively.

We have assumed that TASKA will operate on the following schedule which is identical to that of INTOR.<sup>(2)</sup>

| <u>Year</u> | <u>% Availability</u>    |
|-------------|--------------------------|
| 1           | 10% (hydrogen operation) |
| 2-3         | 15%                      |
| 4-7         | 25%                      |
| 8-15        | 50%                      |

Therefore, the device is expected to operate for 5.3 full power years (FPY). Combining this with the neutron wall loading of  $1.5 \text{ MW/m}^2$  in TASKA yields a total fluence of  $7.95 \text{ MW-yr/m}^2$ . Hence, the accumulated dpa times volume in the helium cooled REGAT module is 11,416 dpa  $\lambda$  and 5764 dpa  $\lambda$  in the water cooled module of TASKA. If the traditional water cooled design is used, the corresponding value is 4149 dpa  $\lambda$ . This implies that a gain of  $\sim 40\%$  in the materials testing capability of TASKA can be achieved with the REGAT design.

Combining the availability values with a startup date of 1990 and a wall loading of  $1.5 \text{ MW/m}^2$  yields the average dpa and cumulative dpa  $\lambda$  values given in Figs. VII.2-20 and VII.2-21, respectively. The values for the cumulative He appm  $\lambda$  are given in Fig. VII.2-22. For the purpose of comparison, the results of the materials testing modules of FMIT, RTNS-II, and INTOR are included. The results are summarized in Table VII.2-14.

The first observation one can make from Fig. VII.2-20 is that current devices like RTNS-II cannot provide either the damage level or the volume in a

# CUMULATIVE DAMAGE IN FUSION MATERIALS TEST FACILITIES

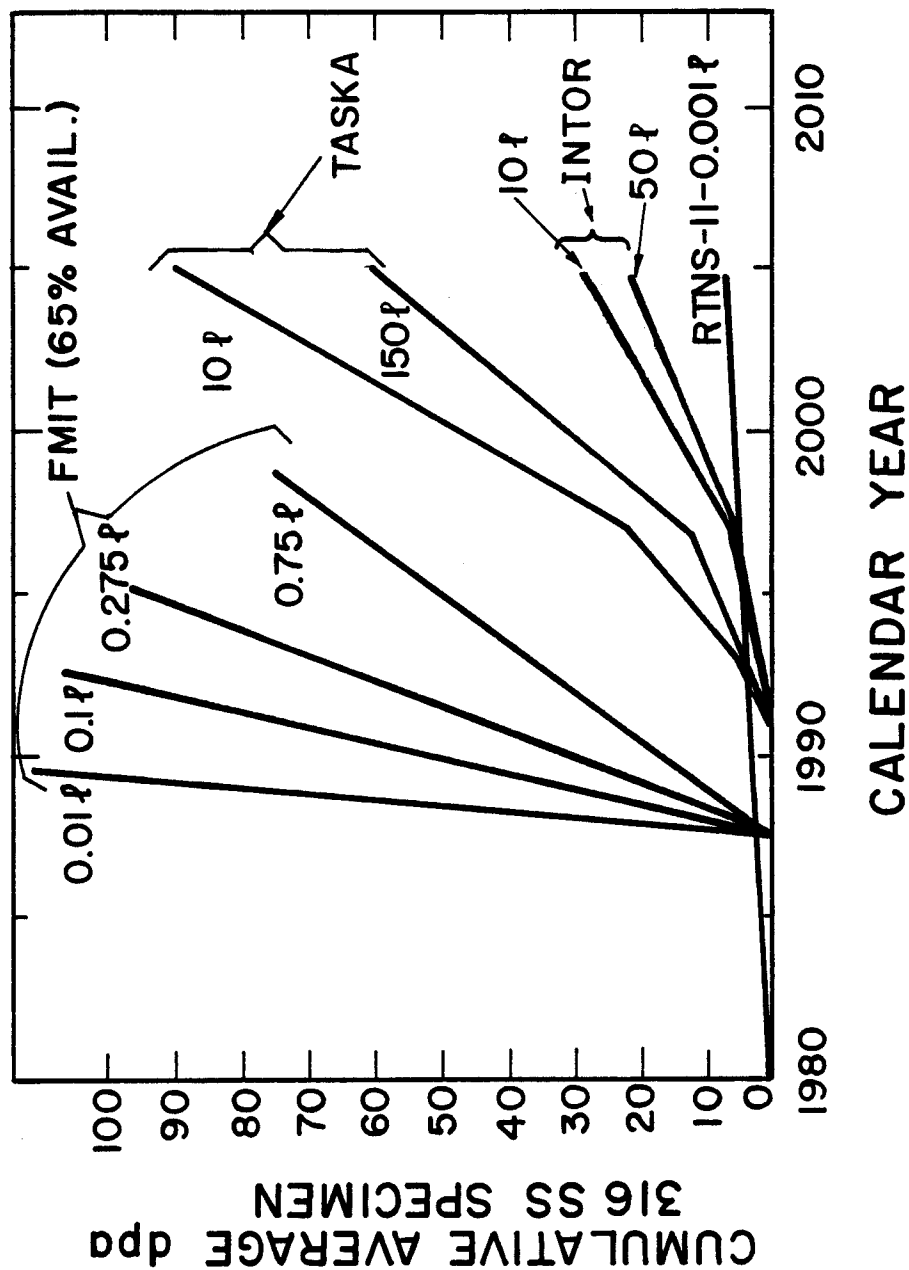


Fig. VII.2-20

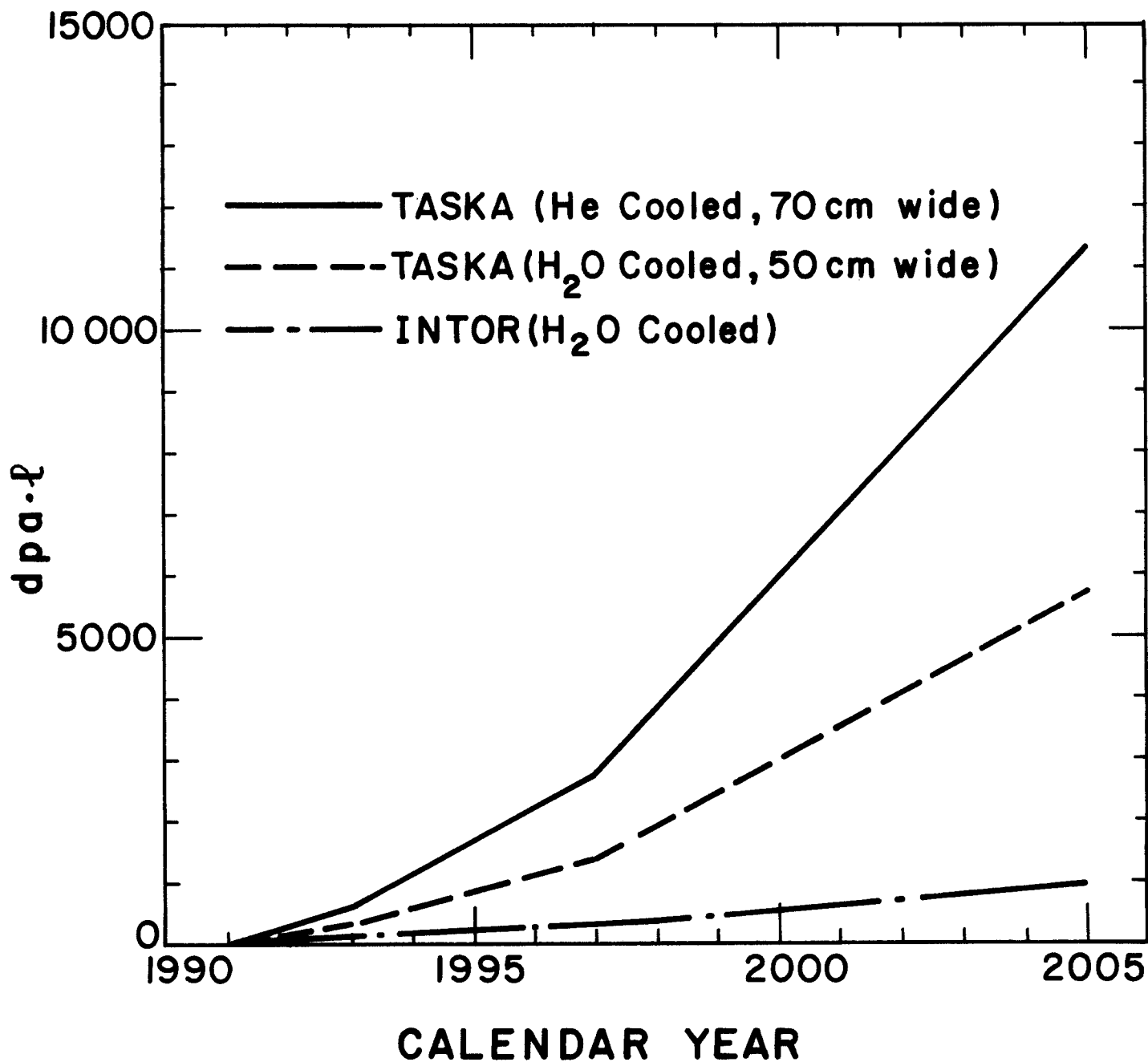


Fig. VII.2-21 Cumulative dpa x test module volume for TASKA and INTOR.

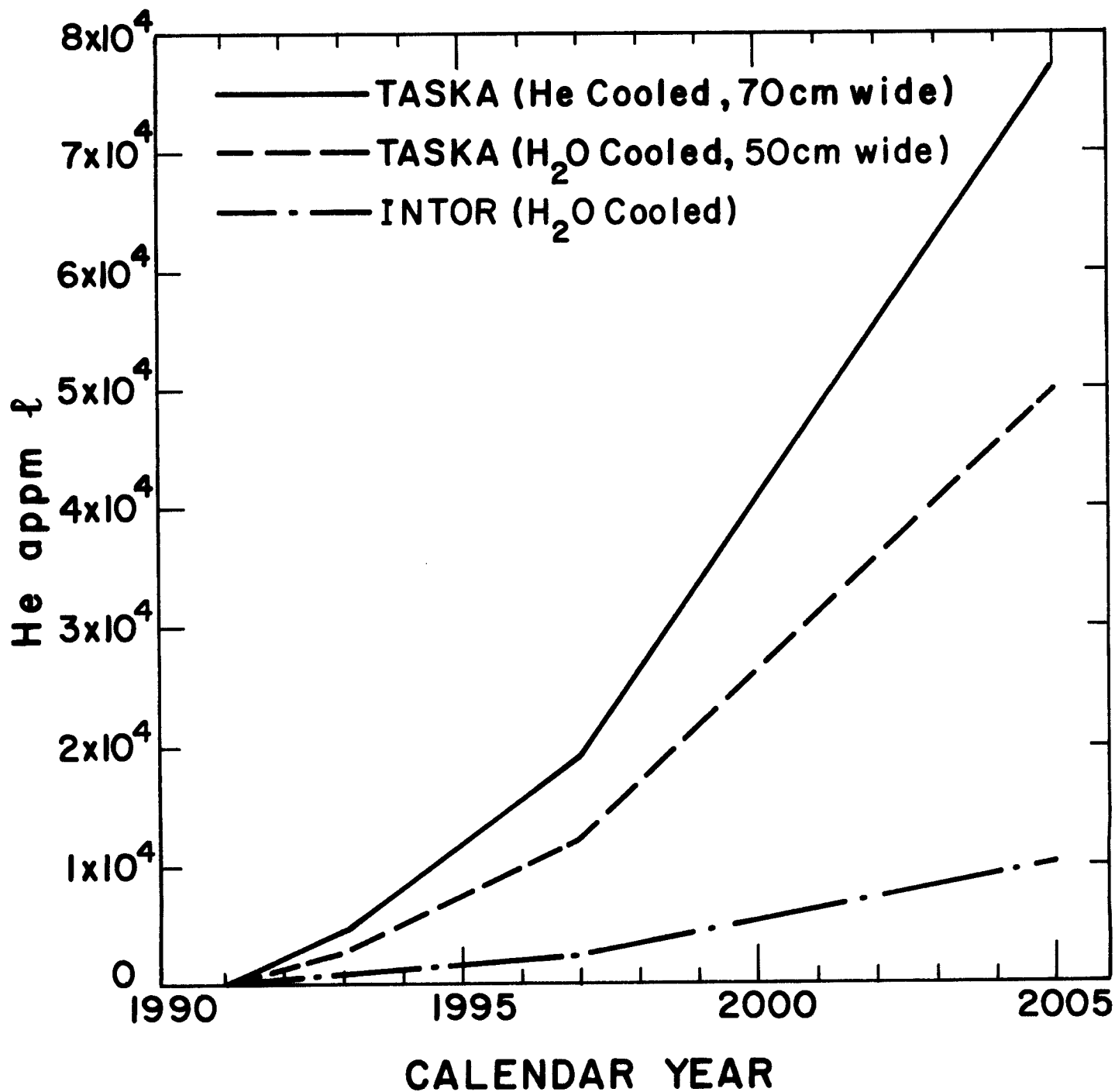


Fig. VII.2-22 Cumulative helium appm x test module volume for TASKA and INTOR.

Table VII.2-14. Comparison Between Different Test Module Designs

|                                                   | He Cooled<br>TASKA | H <sub>2</sub> O Cooled<br>TASKA | H <sub>2</sub> O Cooled<br>INTOR |
|---------------------------------------------------|--------------------|----------------------------------|----------------------------------|
| Area (m <sup>2</sup> )                            | 2.02               | 1.45                             | 0.90                             |
| Ave. dpa per MW-yr/m <sup>2</sup> (10 $\lambda$ ) | 11                 | 7                                | 5                                |
| Module volume ( $\lambda$ )                       | 493                | 352                              | 135                              |
| Capsule volume ( $\lambda$ )                      | 197                | 141                              | 45                               |
| Ave. dpa per MW-yr/m <sup>2</sup> (capsule)       | 7.3                | 5.14                             | 3.9                              |
| Cumulative dpa $\cdot\lambda$                     | 11416              | 5764                             | 965                              |
| Relative figure of merit                          | 12                 | 6                                | 1                                |
| Average He appm per MW-yr/m <sup>2</sup>          | 49                 | 44                               | 40                               |
| Cumulative He appm $\cdot\lambda$                 | 76861              | 49322                            | 9922                             |
| Relative figure of merit                          | 8                  | 5                                | 1                                |

reasonable time. If FMIT is operating in 1986, very high values of damage can be achieved by 1990. It is seen that dpa levels of more than 100 could be obtained in a testing volume of  $\sim 10 \text{ cm}^3$ . Lower damage rates are obtained in larger testing volumes of FMIT and values of 100 dpa could be achieved by the mid-1990's in  $\sim 500 \text{ cm}^3$  of test volume. As valuable as this information is, the qualification of an engineering alloy requires 10's of liters of test volume.

The large test volume requirements are most likely to be filled by devices like INTOR or TASKA. Figure VII.2-20 shows that one could achieve  $\sim 90$

dpa in specimens by 2005 in 10 liters of TASKA test volume or  $\sim 30$  dpa by the same time period in INTOR. Obviously, either of these devices could be used for a major alloy qualification program but TASKA seems to have a higher performance factor.

The above conclusions can be restated by comparing the dpa $\cdot$ l values for TASKA and INTOR. This is done in Fig. VII.2-21. The water cooled REGAT module of TASKA has a dpa times volume figure of merit that is a factor of  $\sim 6$  higher than that for the current INTOR materials testing module. A relative figure of merit of  $\sim 12$  is obtained when the REGAT module is cooled by helium. There are several reasons for the low values obtained in INTOR as compared to TASKA. The wall loading in TASKA is 1.5 vs. 1.3 MW/m<sup>2</sup> in INTOR. While TASKA is a steady state machine, INTOR has a duty cycle of  $\sim 0.8$ . A thick, separate water cooled first wall is required in INTOR because of the high heat flux. This results in further attenuation of the neutron flux to the samples. While an effort was made to decrease the water volume in the test module for TASKA, the INTOR module has a 36 v/o of water. Finally, using the REGAT design in TASKA allows using large test volumes without the penalty of reduced damage level.

#### VII.2.4.3 Conclusions

We conclude from this analysis that a gain of  $\sim 40\%$  in the materials testing capability of TASKA can be achieved with the REGAT design. The damage profiles in the REGAT module suggest that large test specimens, such as those required for the tokamak program, can be irradiated at the sides of the module where the lowest damage gradients occur. The water cooled REGAT module of TASKA has a dpa times volume figure of merit that is a factor of  $\sim 6$  higher

than that for the current INTOR materials testing module. A relative figure of merit of  $\sim 12$  is obtained when the gas cooled REGAT module is used in TASKA.

#### References for Section VII.2

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2. W.M. Stacey, Jr., et al., "USA Conceptual Design Contribution to the INTOR Phase-I Workshop", INTOR/81-1 (June 1981).
3. D. Steiner, W.R. Becraft, and P.H. Sager, J. Fusion Energy 1, (1981) 5.
4. C.A. Flanagan, D. Steiner, and G.E. Smith, ORNL/TM-7777 (June 1981).
5. RSIC Code Package CCC-203, "MORSE-CG", Radiation Shielding Information Center, ORNL.
6. RSIC Data Library Collection, "VITAMIN-C, 171 Neutron, 36 Gamma-Ray Group Cross Sections Library in AMPX Interface Format for Fusion Neutronics Studies", DLC-41, ORNL.
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8. M. Ragheb and C. Maynard, "Neutronics Shielding Analysis for the End Plug of a Tandem Mirror Fusion Reactor", University of Wisconsin Fusion Engineering Program Report UWFDM-398 (1981).
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## Appendix VII.2-A

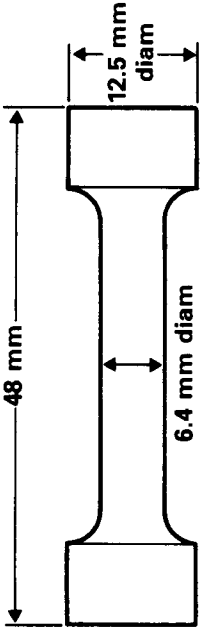
### Materials Test Specimens Specified in the TASKA Test Matrix



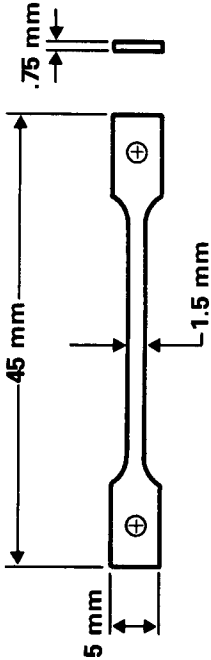
Figure VII.2-A-1

BULK MATERIALS TESTING  
**MATERIALS TEST SPECIMENS**

TENSILE SPECIMEN  
SURVEILLANCE TENSILE (ST)



TENSILE SPECIMEN  
FLAT TENSILE (FT)



TENSILE SPECIMEN  
ROUND TENSILE (RT)

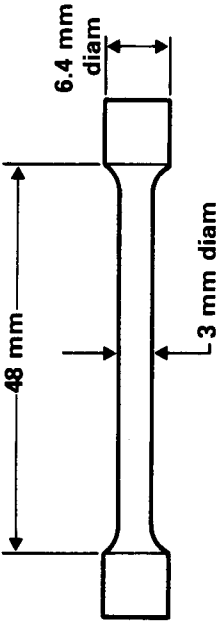


Fig. VII.2-A-2

# BULK MATERIALS TESTING **MATERIALS TEST SPECIMENS**

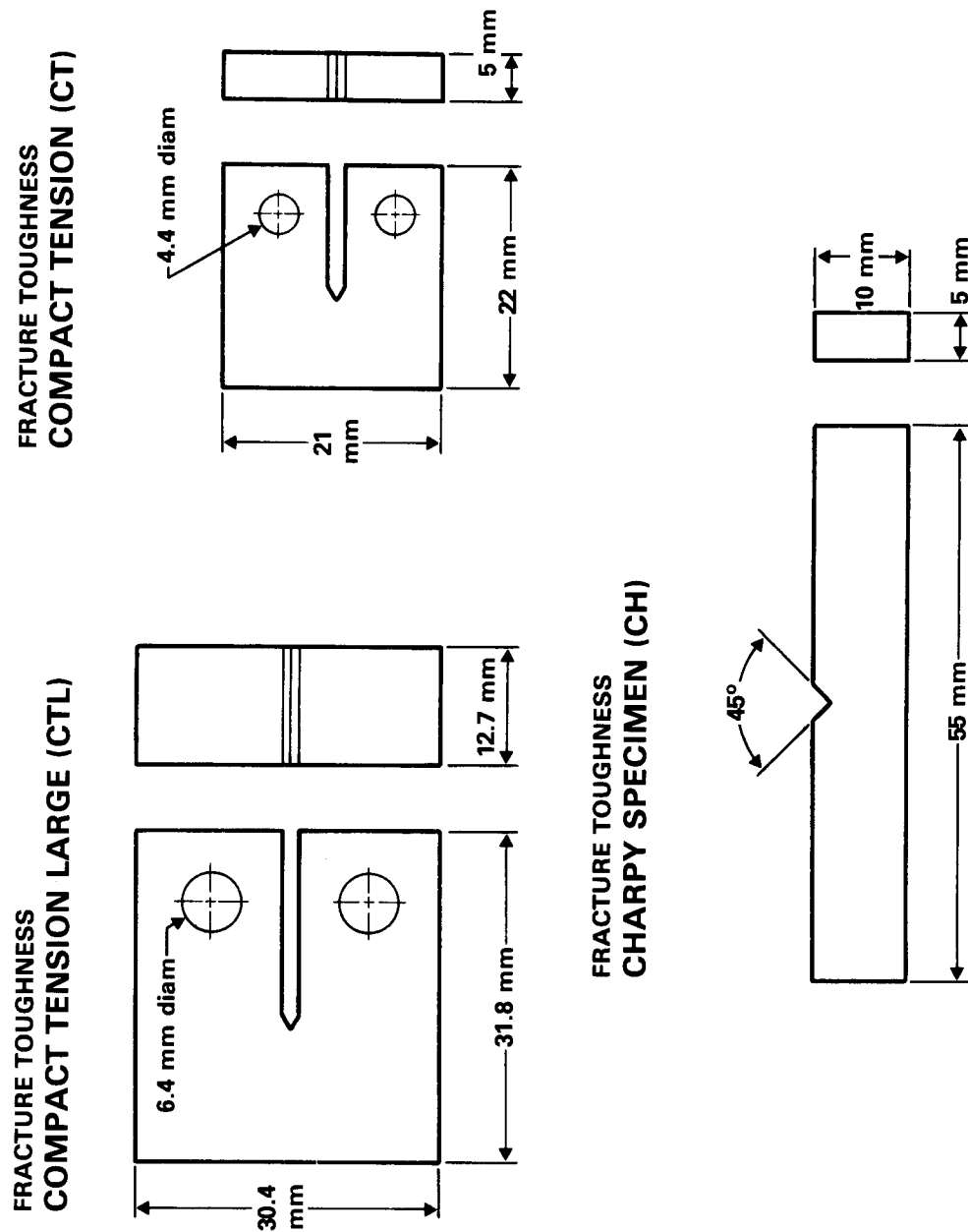
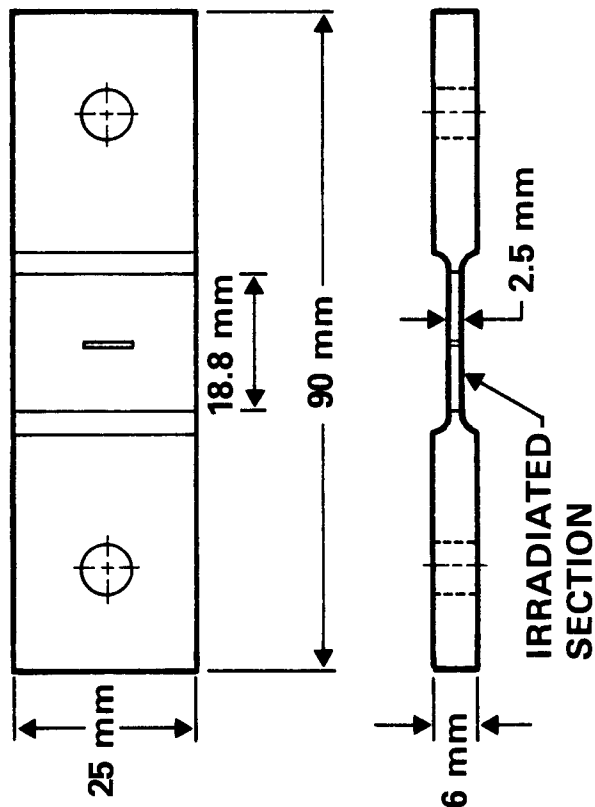


Fig. VII.2-A-3

BULK MATERIALS TESTING

# MATERIALS TEST SPECIMENS

## FATIGUE-CRACK GROWTH RATE SPECIMEN



FULL SCALE

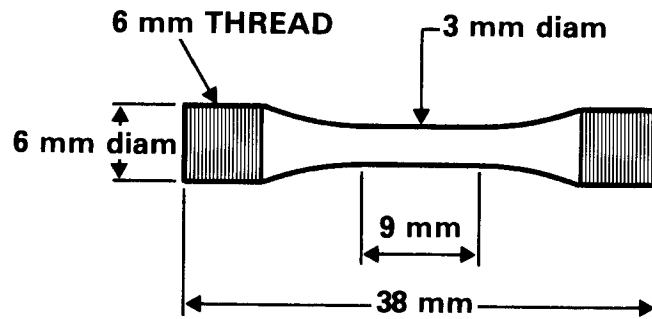
HEDL 8106-158.4

Fig. VII.2-A-4

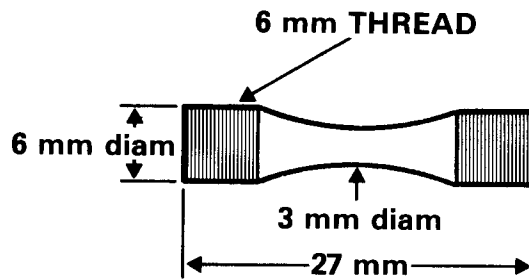
BULK MATERIALS TESTING

# MATERIALS TEST SPECIMENS

## CONSTANT-AMPLITUDE LOAD-CONTROLLED HIGH-CYCLE FATIGUE SPECIMEN (HC)



## CONSTANT-AMPLITUDE STRAIN-CONTROLLED LOW-CYCLE FATIGUE SPECIMEN (LC)

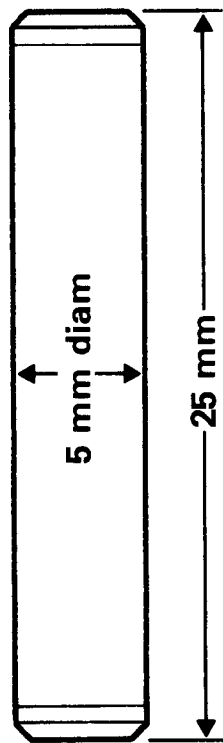


HEDL 8106-158.7

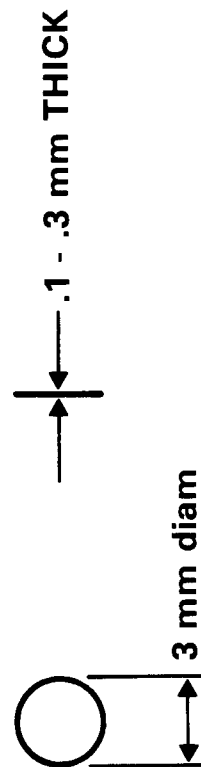
Fig. VII.2-A-5

BULK MATERIALS TESTING  
**MATERIALS TEST SPECIMENS**

**PRESSURIZED TUBE-CREEP SPECIMEN (C)**



**SWELLING AND MICROSTRUCTURAL SPECIMEN (TEM)**

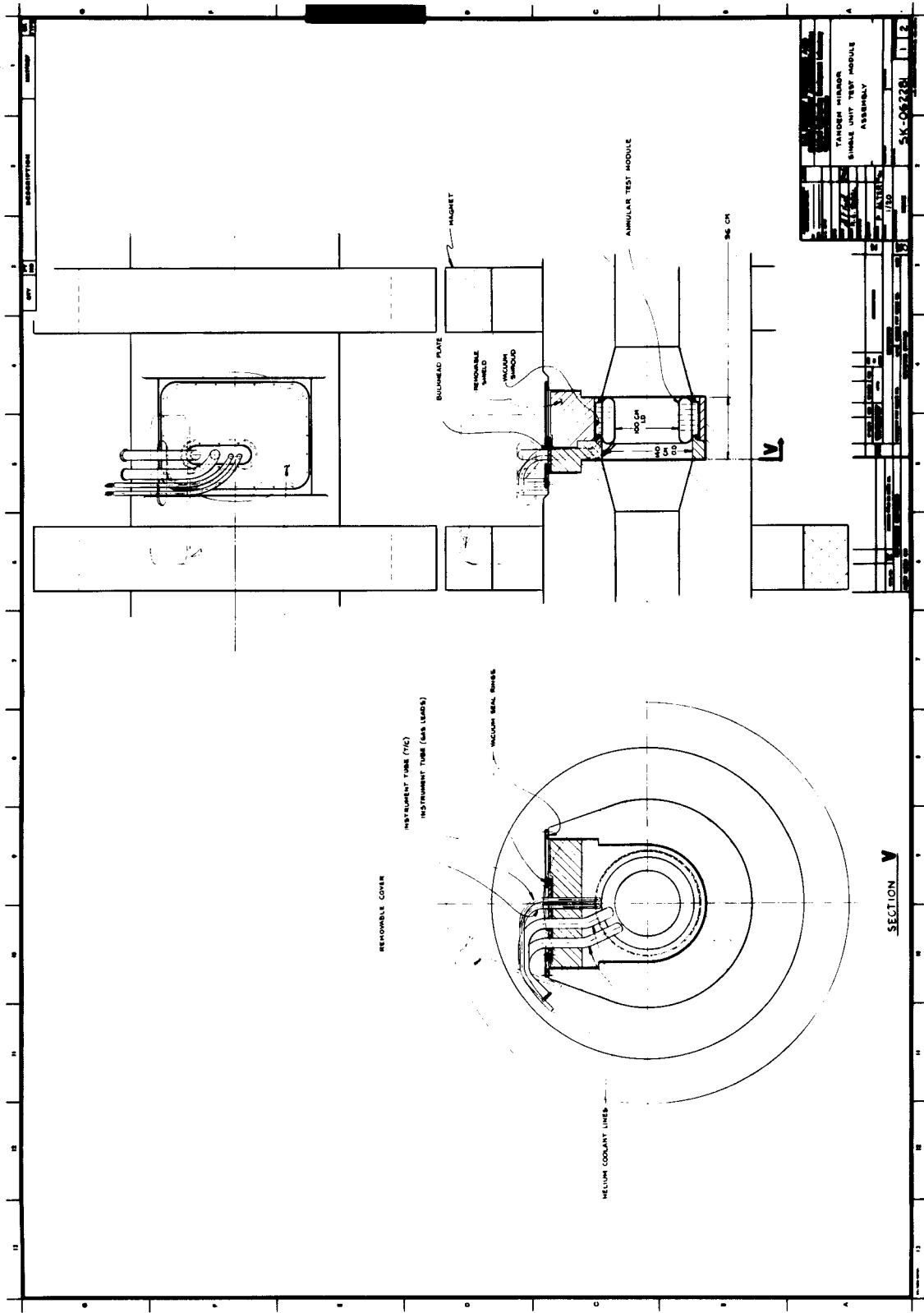


## Appendix VII.2-B

### Test Module and Specimen Capsule Design Details

[illegible]

Fig. VII.2-B-2





### VII.3 Concept of a Test Module With Liquid Lithium Blanket

#### NOTICE

The following section (VII.3) is a design study of an alternative test blanket module for TASKA utilizing liquid lithium. The study was performed by INTERATOM Internationale Atomreaktorbau GmbH, and was conducted independently of the main University of Wisconsin design study of the permanent Pb-Li breeding blanket for TASKA. Accordingly, some dimensional inconsistencies may be encountered between this design and the design for the permanent blanket system (for example, a first wall radius of 40 cm was used for the Li test blanket compared with 46 cm for the Li-Pb permanent blanket).

### VII.3.1 General Considerations

In addition to the permanent Pb-Li breeding blanket for TASKA (see Chapter VI), two test blanket concepts were considered:

- a test blanket with liquid material, where the breeding material lithium is pumped as a liquid through the blanket.
- a test blanket with solid material, where the breeding material is present as a solid and cooled by a gas stream.

In both cases, the tritium must be separated from the breeding material in an extraction process.

It was decided to pursue the blanket concept with liquid material for this TASKA study. The next section gives a detailed description of a blanket module with liquid lithium as breeding material and of the extraction process.



### VII.3.2 Concept of a Test Module with Liquid Lithium Blanket

#### VII.3.2.1 Philosophy of the Concept

The test modules consist of three components:

- Blanket
- Reflector and shield
- Casing

The blanket is a hollow cylinder arranged around the plasma. It has an inner radius of 40 cm (first wall radius) and an outer radius of 140 cm.

Natural lithium is planned as breeding material and coolant for the test module with liquid metal blanket. The target coolant temperature in the blanket is in the range of 300 °C to 500 °C.

As lithium is liquid at temperatures higher than 180 °C, a blanket configuration is selected in which the lithium is pumped through tubes bent around the first wall cylinder. This tube bundle forms the structure of the blanket. The tubes are arranged inside the blanket volume in such a way that a large breeding material inventory is achieved.

The lithium flowing through the blanket permits a low tritium inventory in the blanket, continuous tritium extraction and simultaneous heat removal from the blanket. As the power is mainly deposited in the coolant itself, problems concerning heat transfer play only a secondary role. Due to the large surface of the blanket, caused by the tube construction, the blanket is enclosed in a casing for two reasons:

- a) The pumping time required to evacuate the vacuum vessel is kept low, as the volume to be evacuated and the exposed surface is smaller,
- b) The tritium permeation through the blanket tubes can be controlled by the build-up of a suitable casing atmosphere.

Furthermore, this casing serves to shield the wall of the vacuum vessel from direct neutron irradiation. This concept has, however, a disadvantage in that the cramped space conditions at the side walls of the casing and the first wall make the heat removal from these structures difficult linked with problems of the integration of the cooling facilities.

The part of the reflector and the shield which is integrated in the test module is also enclosed by the casing. Reflector and shield are designed as a plug to simplify handling and permit re-use. They are screwed on to the casing.

The modules must be exchanged from the central cell by means of remote control due to the high radiation level. For this reason, the module is not to be welded to the central cell, but rather is to be equipped with a flanged coupling joint. This means that stringent requirements are imposed on the flange sealings which must be able to maintain the vacuum in the central cell.

#### VII.3.2.2 Design

##### VII.3.2.2.1 Casing

The casing surrounds the blanket and the coolant piping and holds the reflector and shield plug. It is made of austenitic steel 1.4970. The side walls are graduated horizontally and vertically to prevent the vacuum wall being directly irradiated by 14 MeV-neutrons. The graduations (shielding edges) are arranged taking into account the thermal expansion of the casing and a linear material expansion rate of more than 4 % under neutron irradiation. Sufficient distance is thereby guaranteed between the neighbouring modules (Fig. VII.3-1). The width of the shielding edges amounts to 50 mm, when installed, the distance between the shielding edges of neighbouring modules is 50 mm and that between the side walls 40 mm. Fig. VII.3-2 contains a slide-in presentation of the modules in the central cell, in relation to the position of the base plates to each other.

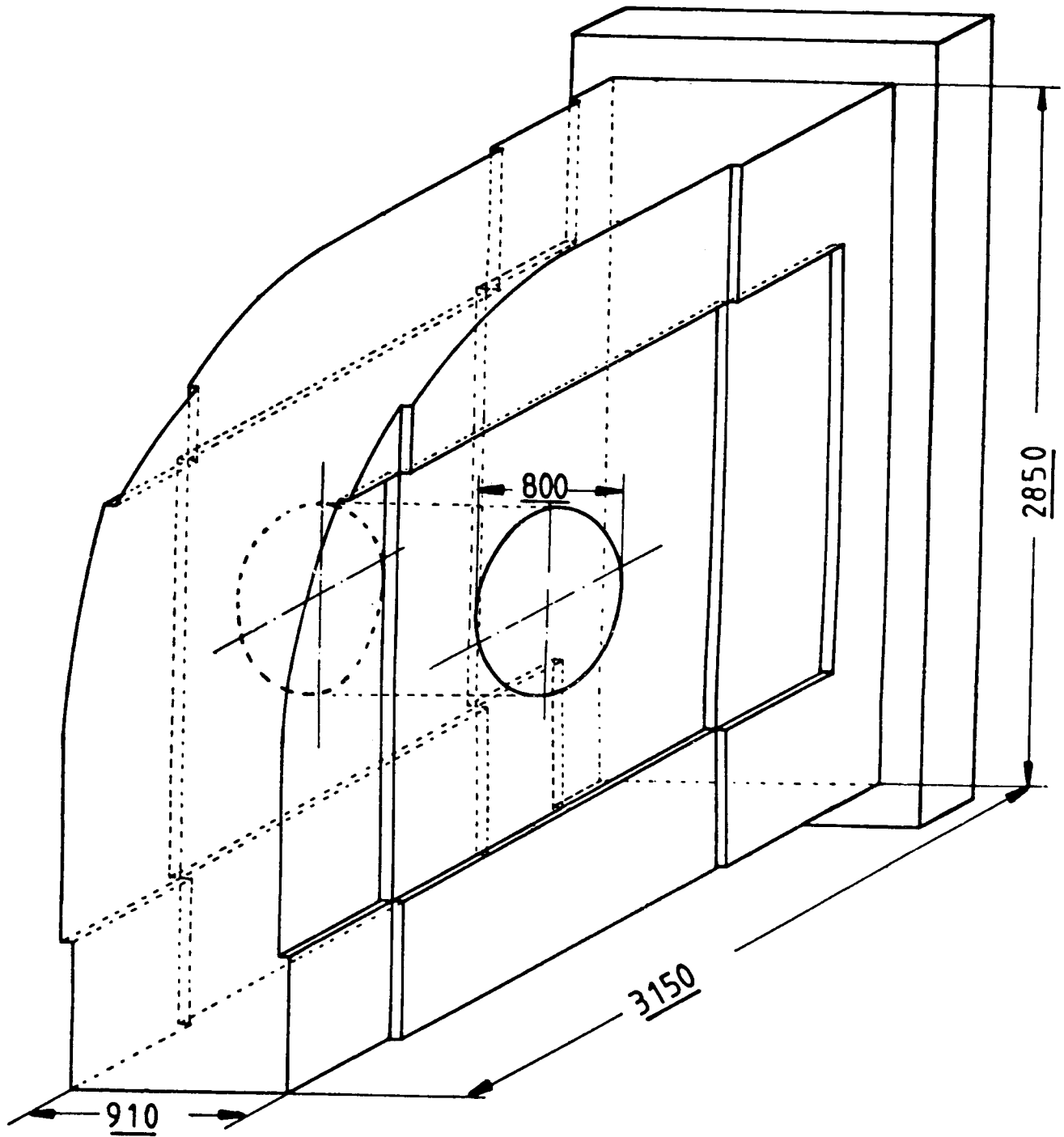
With the exception of the base plate, which is 40 mm thick, the wall thickness of the casing amounts to 20 mm. The first wall is 10 mm thick and has an inner area of 2.54 m<sup>2</sup>.

Table I of the Appendix contains a survey of the geometric data of the casing.

The wall thicknesses are dimensioned assuming low pressure differences and the integration of suitable support structures to retain the rigidity of the casing even after dismantling of the blanket and shield. Similarly, the installation of shielding structures takes into account the requirements for the shielding of the vacuum wall along the shielding edges.

Due to its cylindrical form, the first wall is sufficiently dimensioned against internal and external overpressure (max. 0.1 MPa) with a wall thickness of 10 mm. This pressure difference only occurs in the case of accident (leaky vacuum seals) or during short-term maintenance work in the blanket area without removal of the module.

During reactor operation, vacuum conditions ( $\approx 10^{-3}$  Pa) are assumed both inside and outside the casing.



**Fig. VII.3-1 TASKA test module casing**

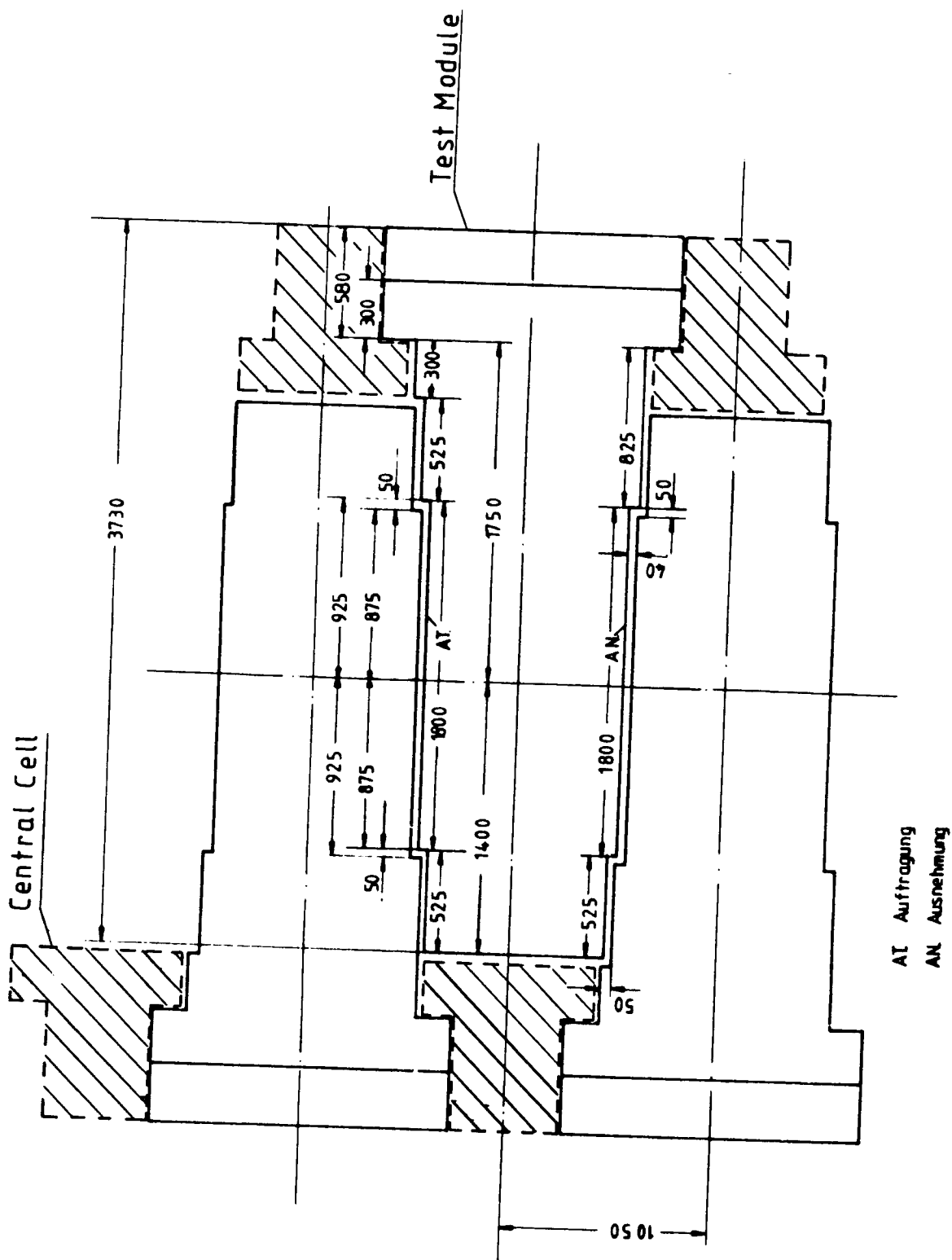


FIG. VII.3-2 Arrangement of the modules inside central cell

For strength reasons, the maximum permissible temperature of first wall and the side walls of the casing has been fixed to 550 °C. Due to the heat production in these structures, they must be provided with cooling facilities. The cooling system of the first wall is dealt with in the following Chapter VII.3.2.2.2 as lithium is planned as coolant and the system is considered as an integral part of the blanket.

At this stage in the concept, we are trying to find a basic method of solving the cooling problem of the side walls.

The concept of a surface cooler, as illustrated in Fig. VII.3-3 is conceivable as a cooling system for the side walls. Thereby it is assumed that the heat production drops so drastically in the radial direction that only the area from  $r = 40$  cm (first wall cylinder) to the first shielding edge needs to be cooled. This is realistic as the outer wall parts located behind are largely shielded by the blanket of the adjacent modules.

The inner side of the area to be cooled (approximately 3 m<sup>2</sup> per side) is covered by a plate in which coolant guides are imprinted. The plate is welded to the wall of the casing at the first wall and the shielding edges and is leak-proof.

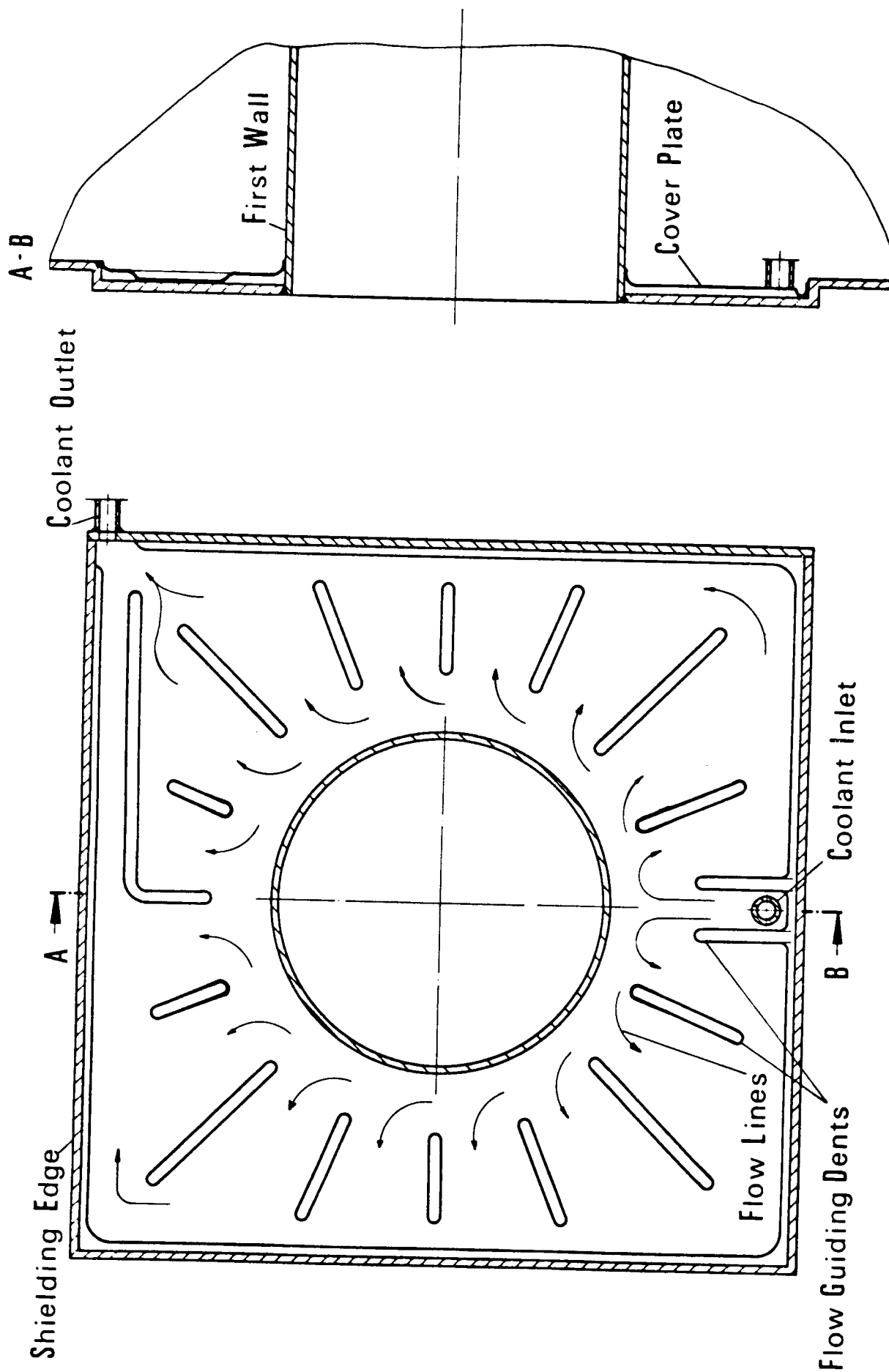
With this configuration it is therefore possible to achieve a directed coolant guidance, which takes into account the gradient of heat production and should limit the average temperature in the wall structure to approximately 350 °C.

The coolant pressure has to be kept very low to avoid high additional wall stresses.

For handling the test module is guided along on support rails inside the central cell. The support rails are equipped with roller rails to improve the sliding ability of the module casing on the rails.

In order to simplify the remotely controlled exchange of the test modules, they are bolted to the central cell. This imposes high requirements on the flange sealing which must guarantee the vacuum conditions inside the central cell. The sealing is therefore a double O-ring seal with interspace evacuation. Viton or Inconel which is vapour deposited with silver can be used as sealing materials.

The maximum leakage rate of the flange should not exceed  $10^{-3} \frac{\text{Pa.l}}{\text{s}}$ .



**Fig. VII.3-3 Proposal for a cooling method for the casing wall**



#### VII.3.2.2.2 First Wall/Blanket

##### VII.3.2.2.2.1 First Wall

From a construction point of view, blanket and first wall will be regarded as one unit here. The first wall must be cooled due to the power production in the structure of the first wall and the heat flow from the plasma. Liquid lithium which is planned as breeding material and coolant for the blanket, is to be used as coolant.

The basic arrangement of the first wall cooling system is presented in Fig. VII.3-4.

It consists of half-tubes welded on to steel bands which are arranged around the first wall cylinder. The half-tubes have an outer radius of 22.5 mm and should have a center distance of approximately 70 mm. The steel bands are pressed against the first wall cylinder by means of a holding fixture device, see Fig. VII.3-5, in order to attain the best possible heat transfer.

At the same time, this approach serves to prevent non-permissible high stresses in the cooling tubes, which can occur as a result of the thermal expansion of the first wall.

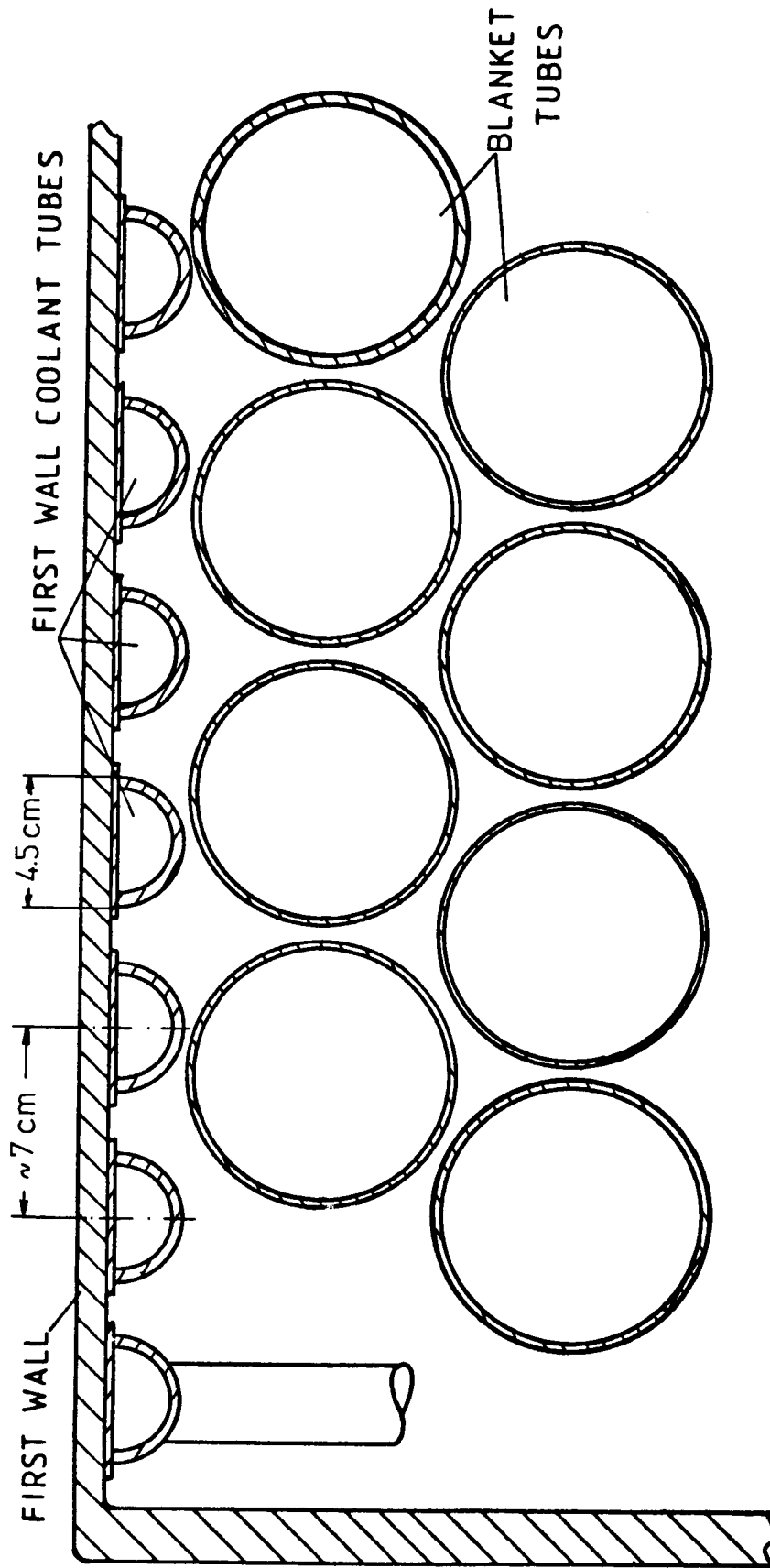
The temperature pattern in the first wall for a center distance of 70 mm between the cooling tubes which is illustrated in Fig. VII.3-6 shows that the maximum temperature difference between the inner side of the first wall and a cooling tube is approximately 10 °C per W/cm<sup>2</sup>. This value can just be tolerated, but nevertheless calls for an optimization of the thickness of the first wall, taking into consideration the power production and the stability requirements for the specified pressure differences. The temperature difference between the first wall and the cooling tube plate amounts to approximately 35 °C for a heat transmission coefficient of 0.57 W/°C cm<sup>2</sup>.

The cooling system of the first wall is connected to the coolant supply of the blanket.

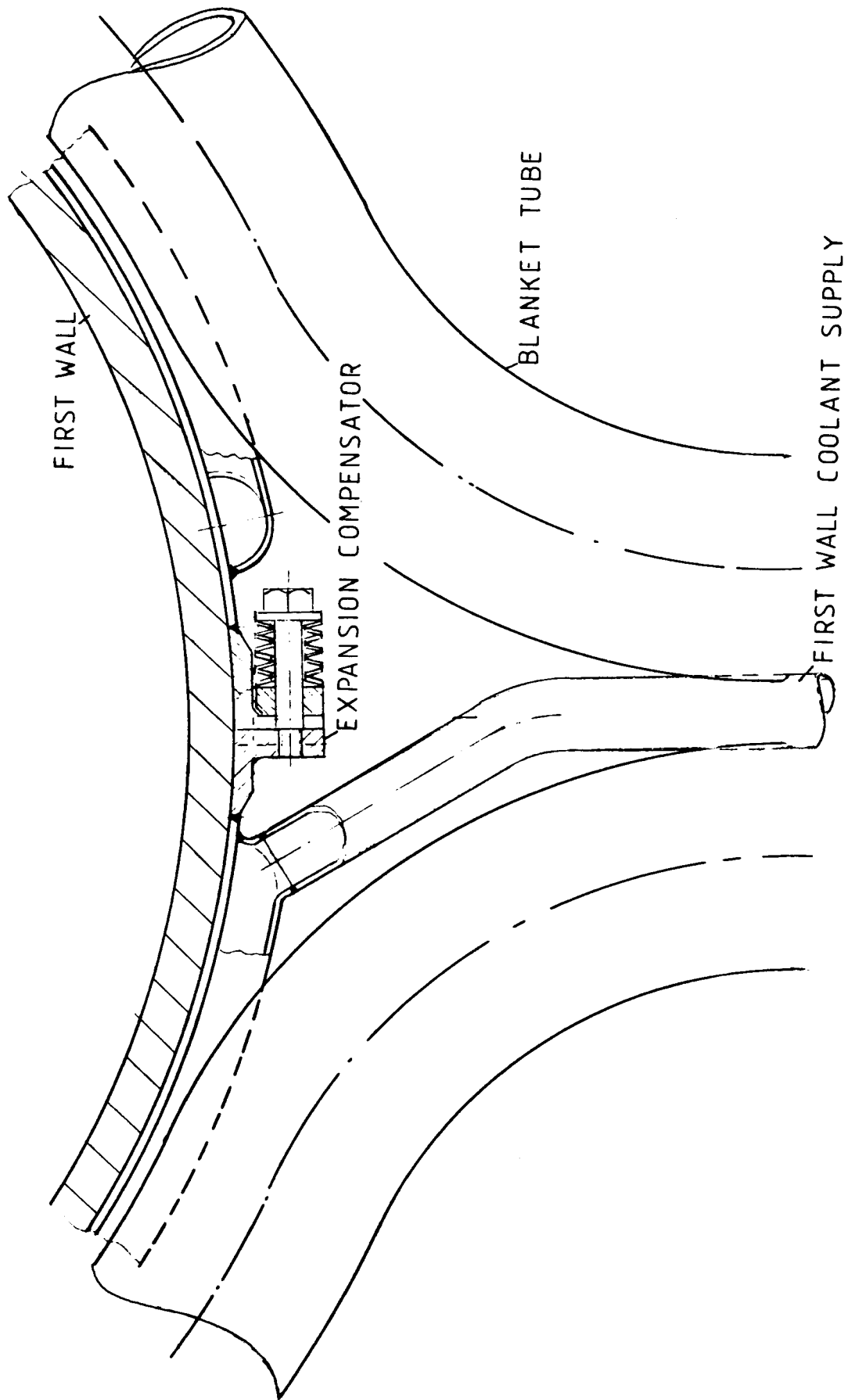
##### VII.3.2.2.2.2 Blanket

The blanket is designed in such a way that an acceptable tritium breeding rate and power production are attained with the given breeding material.

It consists of tubes bent around the first wall cylinder which are made of austenitic steel 1.4970. They have a diameter of approximately 10 cm and a wall thickness of 2.6 mm (Fig. VII.3-7). The distance between the tubes amounts to approximately 5 mm. Spacers prevent the tubes from touching each other.



**Fig. VII.3-4 Arrangement of the first wall coolant tubes**



**Fig. VII.3-5 Holding fixture of the first wall coolant tubes**

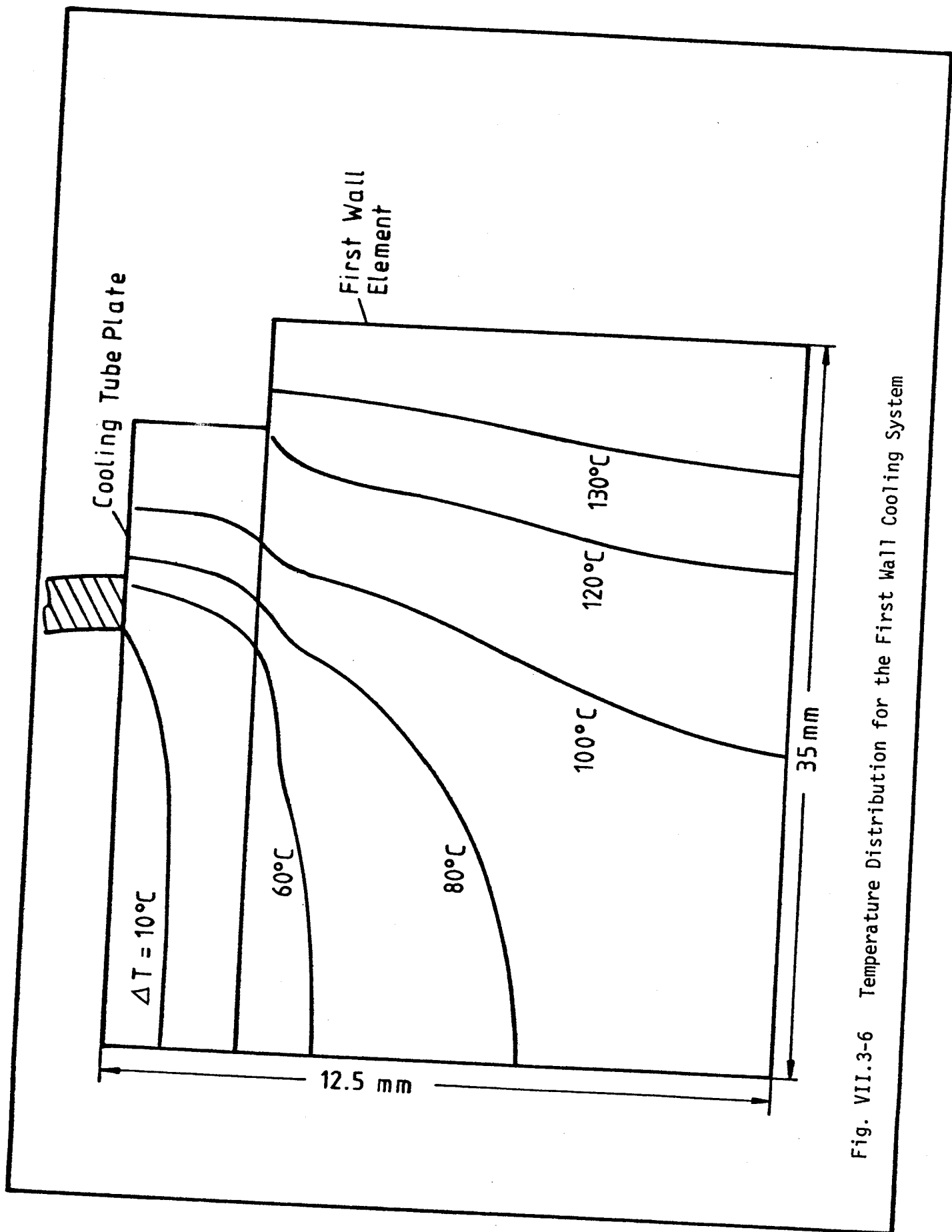


Fig. VII.3-6 Temperature Distribution for the First Wall Cooling System

In the radial direction, ten layers of tubes are positioned in staggered form behind each other. The greatest possible utilization of the blanket volume with breeding material is attained as a result of this design. The fraction of breeding material in the blanket amounts to 73 % within the tube region.

The axial number of layers amounts to 4 rows of 8 tube bends and 6 rows of 7 tube bends from the inside to the outside caused by the geometry of the module casing. The average radius of the inner tube bend is 483 mm and that of the outer tube bend 1317 mm.

The liquid lithium is supplied via the coolant inlet pipe to the distribution header. This is connected to the blanket tubes by nozzles (Fig. VII.3-8). After it has flowed through the blanket area, the lithium is removed from the module via collecting header and outlet pipe. A different coolant mass flow is necessary in the tubes in accordance with the power distribution in the blanket. This is achieved by means of different tube lengths and the throttles in the nozzles which connect the blanket tubes to the headers.

The area of the inner side of the blanket which is exposed to fusion neutrons amounts to  $2.3 \text{ m}^2$  or 91 % of the first wall area of the test module.

The important parameters of the TASKA blanket with liquid lithium as breeding material and coolant may be taken from Table VII.3-1.

Blanket tubes, nozzles, headers and coolant pipes were designed for a maximum coolant pressure of 0.7 MPa at  $T = 550 \text{ }^\circ\text{C}$ . Due to the very low coolant velocities in the bundle tubes (max. 7 cm/s), at this temperature the eroding corrosion which is to be expected during the maximum operating period of 2.5 years at load factor of 80 % has already been taken into account for the wall thickness. Austenitic steel 1.4970 which was used for the blanket tubes, is also used for nozzles, headers and coolant pipes. According to Larsen and Miller, its maximum permissible loading was estimated as being  $110 \text{ N/mm}^2$  at  $550 \text{ }^\circ\text{C}$  under neutron irradiation. After a first estimate, the stresses to be expected for blanket tubes, nozzles, headers and main cooling pipes are below this limit.

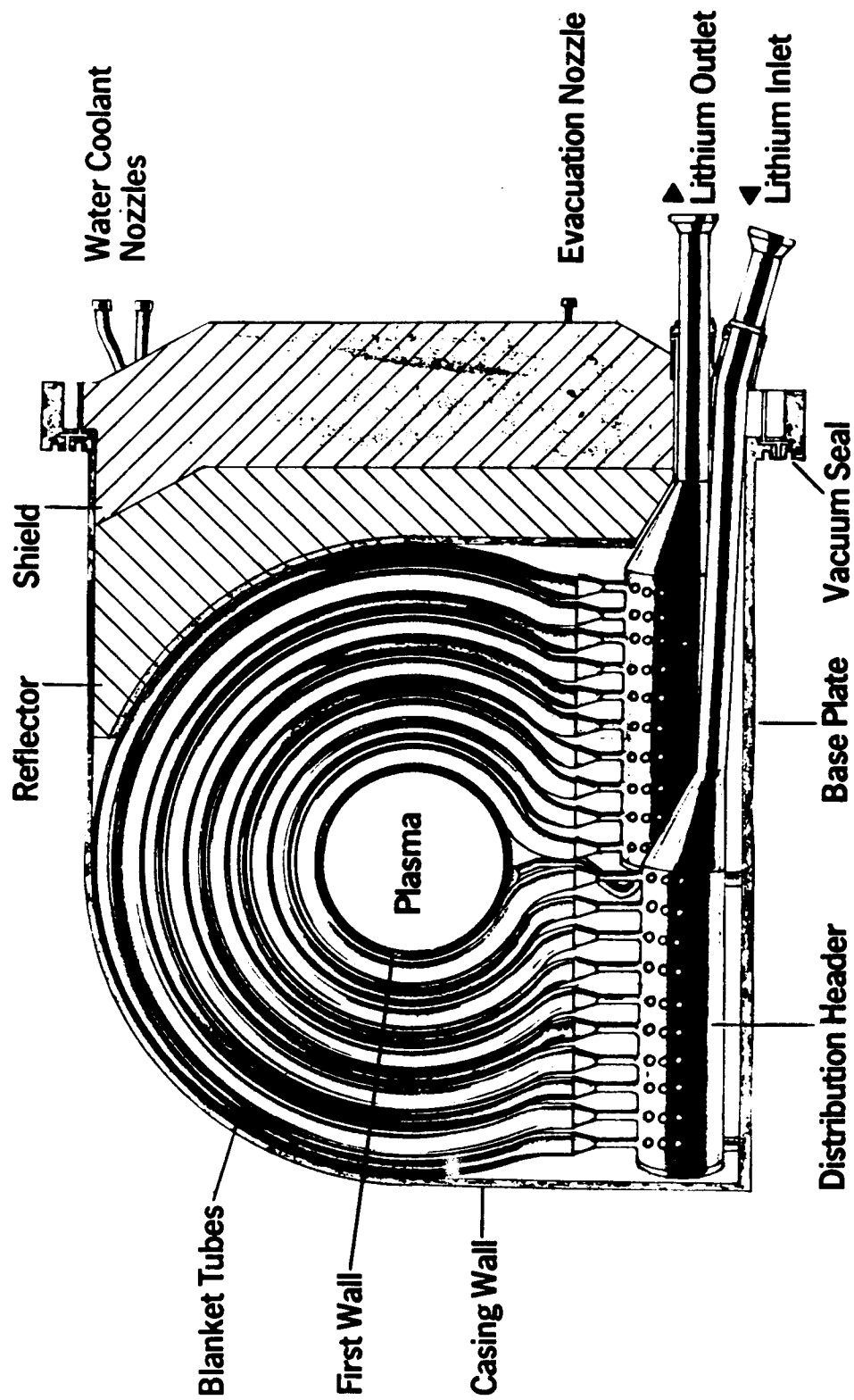
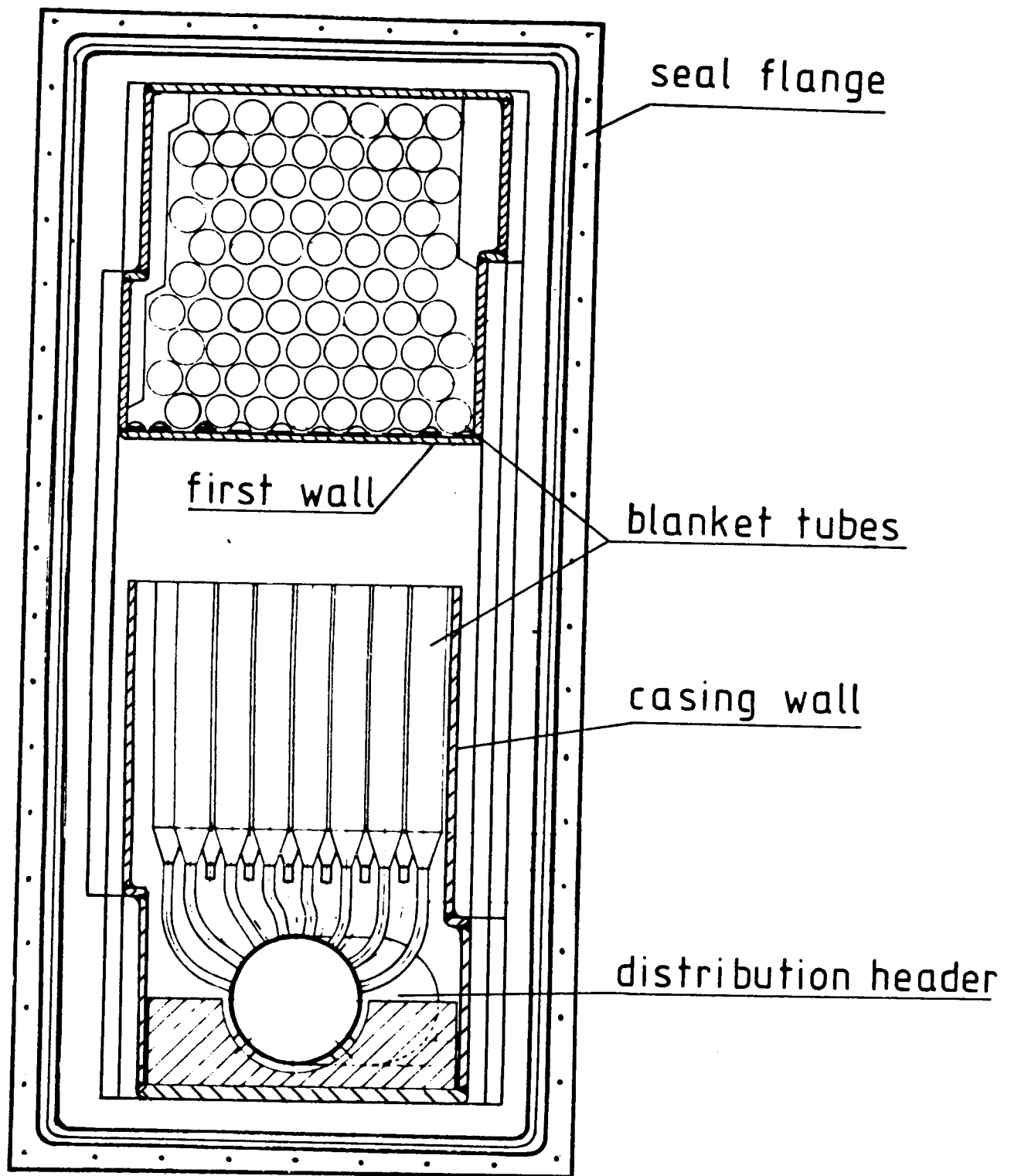


Fig. VII.3-7 TASKA test module blanket view



**Fig. VII.3-8 TASKA test module vertical cut**

Table VII.3-1: Data of the TASKA Liquid Lithium Blanket

|                               |                                            |
|-------------------------------|--------------------------------------------|
| Breeding material and coolant | Natural Lithium<br>(92.5 % Li7, 7.5 % Li6) |
| Radius of the First Wall      | 40 cm                                      |
| Thickness of the blanket      | 96 cm                                      |
| Length of the blanket         |                                            |
| $\bar{r}$ = 42 cm             | 92 cm                                      |
| $r$ = 138 cm                  | 82 cm                                      |
| Composition (tube region)     | 73 % Lithium<br>7 % Structure<br>20 % Void |
| Structural material           | Austenitic Steel 1.4970                    |
| Diameter of the blanket tubes | 10 cm                                      |
| Number of blanket tubes       | 74                                         |



### VII.3.2.2.3 Reflector and Shield

#### VII.3.2.2.3.1 Design

Reflector and shield are designed as a plug which is inserted into the module casing behind the blanket as one component. Fig. VII.3-9 shows the schematic composition of reflector and shield.

The reflector is made of austenitic steel 1.4970 and is provided with cooling holes, through which water with an average temperature of approximately 80 °C and a pressure of approximately 0.5 MPa flows. The average power density in the reflector amounts to approximately 0.1 W/cm<sup>3</sup>.

The purpose of the shield is to guarantee sufficient protection against the neutron radiation and the secondary gamma radiation during reactor operation and shield also against the  $\gamma$ -radiation of activated structures after reactor shutdown.

In contrast to this, there is a demand for the thinnest possible shield, as this keeps the quantity of shield material required low.

Furthermore, the thickness of the shield influence the inner diameter of the superconducting magnet coils, the size of which also greatly affects the costs of the plant.

The shield construction, which comprises 3 layers of Pb/B<sub>4</sub>C/ steel structure, was specified.

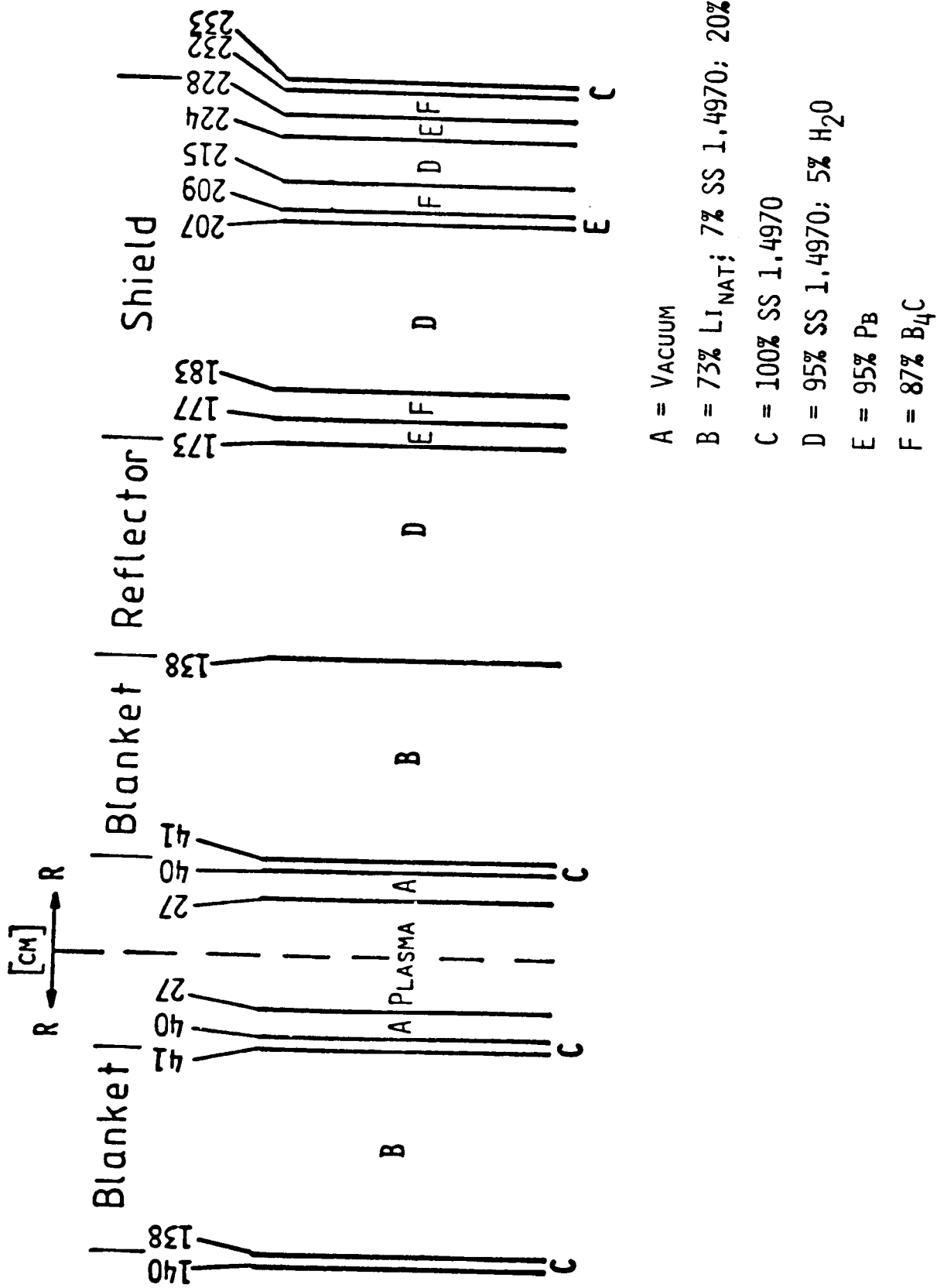
The shield is cooled by water to remove the heat produced in the structure and in the boron layers. The maximum structure temperature is limited to 200 °C.

The reflector and shield plug is hermetically bolted to the casing (leakage rate  $< 10^{-3} \frac{\text{Pa.l}}{\text{s}}$ ). A double O-ring seal with inter space evacuation, the suction channel of which is attached to the seal of the casing flange, is envisaged as the seal.

#### VII.3.2.2.3.2 Shield Calculations

The calculations dealt with the shielding effect with reference to the neutron and gamma dose rates during reactor operation and the  $\gamma$ -dose rates from the activated structures after plant shutdown in the case of an installed test module.

The two dimensional calculation of the neutron flux distribution (26-groups-ABBN-structure) was performed using the INTERATOM computer code DLS in accordance with the diffusion method, whereby the Removal Sources were taken into account. The INTERATOM computer code DLS-GAMMA is used to calculate the gamma dose rate during operation and after reactor shutdown. This program works according to the Point-Kernel method and takes the build-up factor into consideration.



**Fig. VII.3-9 TASKA blanket, reflector and shield composition**

The 26-groups-flux at the outer blanket boundary of the KfK-INR-neutron transport calculations / 1 / was used as boundary flux for DLS.

The calculations resulted in a neutron dose rate of 4,000 rem/h and a gamma dose rate of 1 rem/h at a distance of 1 m from the outer edge of the module during reactor operation. The neutron flux distribution is shown in Fig. VII.3-10.

After reactor shutdown, the dose rate due to structure activation of the reflector/shield plug amounts to 12 mrem/h at the outer edge of the module and to about 6mrem/h at a distance of 1 m. Only the long-lived isotopes Co60 ( $T_{1/2} = 5.3$  a), Ta 182 ( $T_{1/2} = 115$  d) and Fe59 ( $T_{1/2} = 44$  d) were considered in these calculations.

The concentration of the precursors in steel was assumed to be:

500 ppm Co 59

500 ppm TA 181

3100 ppm Fe 58

In the case of a cyclic mode of operation over a period of 30 months with a load factor of 80 % / 2 / the fraction of the saturation activity achieved at the end of the operating period is as follows:

23 % for Co 60

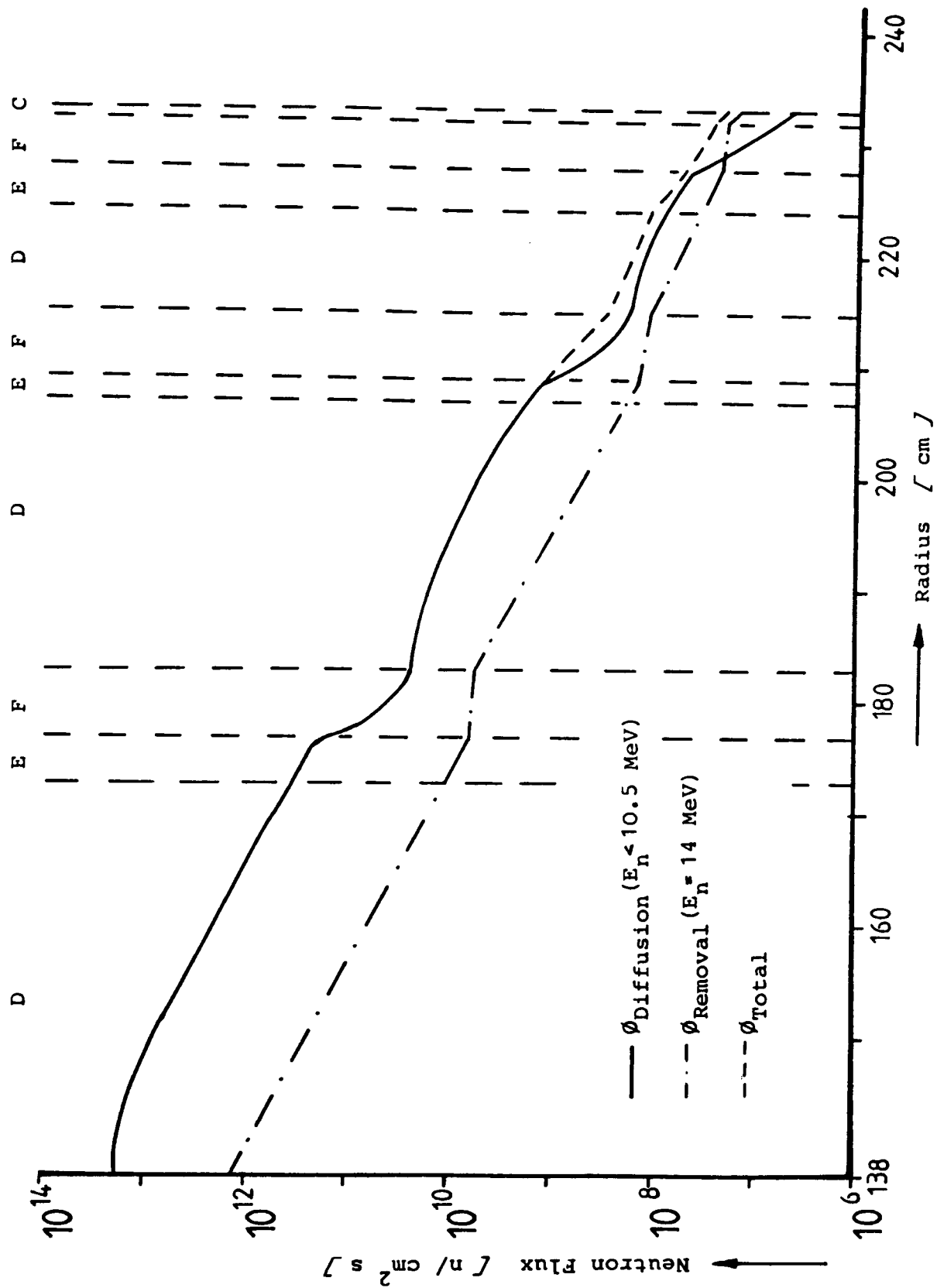
89 % for Ta 182

98 % for Fe 59

Table VII.3-2 contains a summary of the results calculated.

With reference to the sandwich construction of the shield, it is recommended that this be reduced to 3 layers. This could lead to a simpler construction with lower manufacturing and material costs.

Furthermore, the thicknesses of the absorber layers and the reflector should be optimized.



**Fig. VII.3-10 Neutron flux distribution inside reflector and shield plug at  $1\text{MW/m}^2$  neutron wall loading**

Table VII.3-2: Radiation Exposure at the TASKA Test Module with  
Liquid Metal Blanket

Dose rate during operation:  
at a distance of 1 m

due to neutrons  $\dot{D}_n = 4000 \text{ rem/h}$   
due to gammas  $D_\gamma = 1 \text{ rem/h}$

Dose rate after reactor shutdown  
( $T > 1d$ ):

$T_{\text{operation}} = 2.5 \text{ years, LF} = 80 \%$

Nuclide concentrations

500 ppm Co 59  
500 ppm Ta 181  
3100 ppm Fe 58

| Nuclide | $\dot{D}_{\text{radial}} / \text{mrem/h} /$ |              |
|---------|---------------------------------------------|--------------|
|         | Module outer edge                           | 1 m Distance |
| Co 60   | 1.1                                         | 0.6          |
| Ta 182  | 10.0                                        | 5.0          |
| Fe 59   | 0.5                                         | 0.3          |
| Total   | 11.6                                        | 5.9          |

### VII.3.2.3 Cooling Loop

#### VII.3.2.3.1 Power and Coolant Balance

The power generated in the test module is made up of the power generated in the first wall and in the cooling system of the first wall and the blanket power.

Fig. VII.3-11 illustrates the radial power distribution in the first wall and in the blanket for a neutron wall loading of  $1 \text{ MW/m}^2$ . A homogeneous blanket composition of 73 % natural lithium, 7 % structure and 20 % void was assumed for the calculation.

The power generated per test module amounts to 3.5 MW for an average power density of approximately  $0.9 \text{ W/cm}^3$  in the blanket. The neutron wall loading for TASKA is  $1.53 \text{ MW/m}^2 / 3 /$ .

The blanket energy multiplication factor is 0.9. This results in a blanket power of 3.2 MW.

The blanket energy multiplication factor has been estimated as follows.

From the one dimensional (1-D) neutronics calculations, a blanket multiplication factor ( $M_{1D}$ ) of 1.02 was obtained. However, the true energy multiplication factor ( $M$ ) of the blanket must take into account the fact that the blanket composition, structure and void fractions are different in different radial directions across the blanket (see Fig. VII.3-7). A first approximation to  $M$  may be obtained by weighting the 1-D computation of the multiplication factor by the ratio of the blanket volume fraction to the blanket volume fraction plus the headers volume fraction. Therefore,

$$M \approx M_{1D} \frac{V_{\text{Blanket}}}{V_{\text{Blanket}} + V_{\text{Headers}}}$$

where  $M_{1D}$  is the multiplication factor from the 1-D blanket neutronics calculation

$V_{\text{Blanket}}$  is the volume fraction of the blanket ( $\approx 3.6 \text{ m}^3$ )

$V_{\text{Headers}}$  is the volume fraction of the headers region ( $\approx 0.5 \text{ m}^3$ )

Table VII.3-3 contains the power to be removed per tube row of the blanket. The inlet temperature of the coolant amounts to  $300^\circ\text{C}$  and the outlet temperature should be varied between  $400^\circ\text{C}$  and  $500^\circ\text{C}$ . An average coolant outlet temperature of  $450^\circ\text{C}$  was taken as the basis for the coolant balance. The corrosion at this temperature leads to a material deduction of approximately  $10^{-5} \text{ m/a} / 4 /$ , and is an acceptable value for the operating period maximum of 2.5 years as planned for the TASKA test modules.

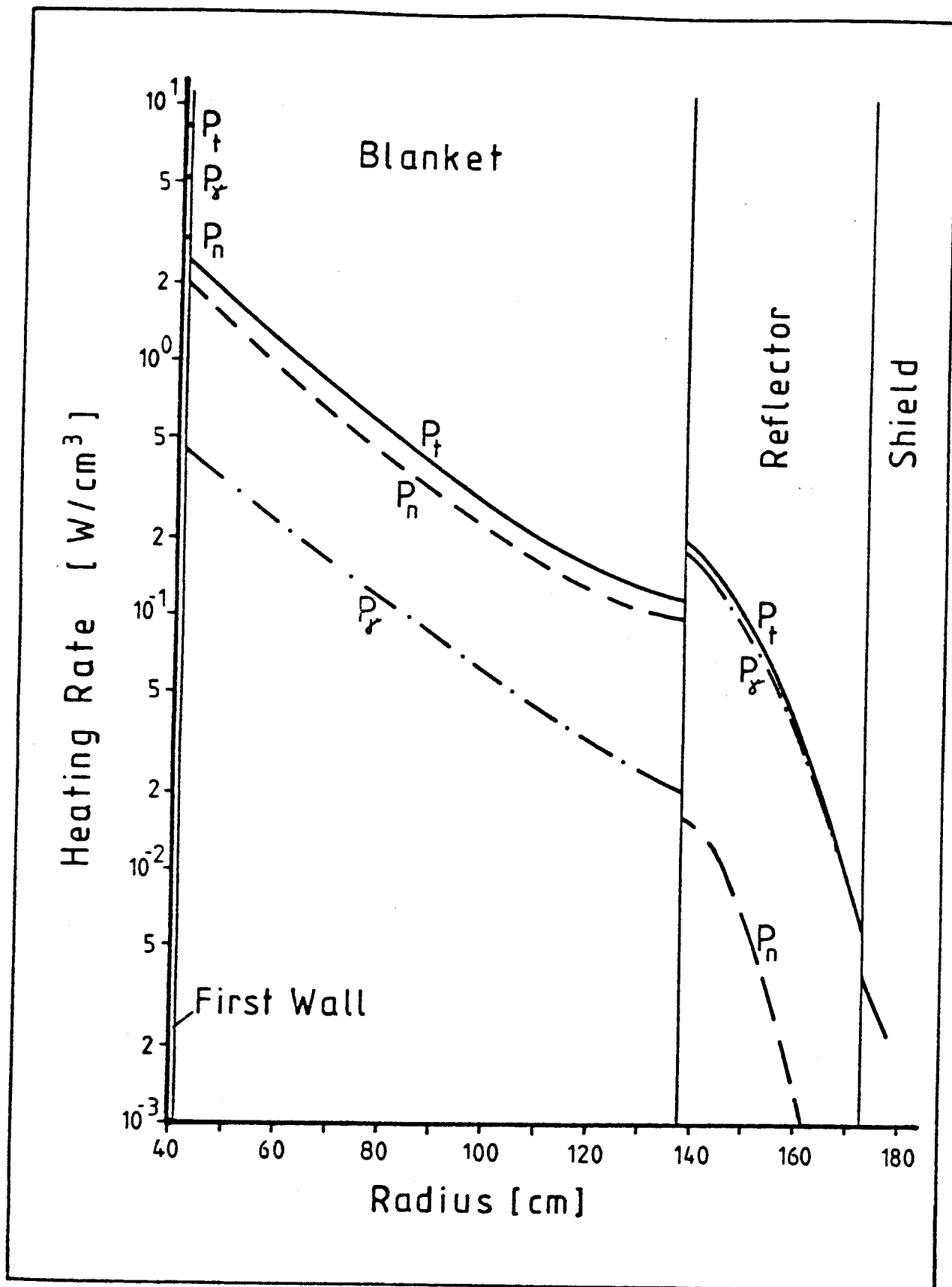


Fig. VII.3-11 Radial Heating Rate Distribution at  
1 MW/m<sup>2</sup> Neutron Wall Loading

A linear temperature increase in the coolant was assumed from the tube beginning to the end of the tube.

The main data concerning the balance of power and coolant for a test module may be taken from Table VII.3-4.

The calculations were based on the material data for lithium, which are listed in Table II of the Appendix.

Table VII.3-3: Tube Row Power in the TASKA Liquid Lithium Blanket

| Tube<br>Row | $\bar{r}$<br>/ cm / | $\bar{p}$<br>/ W/cm <sup>3</sup> / | Power per Row<br>/ MW / | Number of<br>Blanket Tubes |
|-------------|---------------------|------------------------------------|-------------------------|----------------------------|
| 1           | 48                  | 2.9                                | 0.86                    | 8                          |
| 2           | 57                  | 2.0                                | 0.61                    | 8                          |
| 3           | 67                  | 1.4                                | 0.46                    | 8                          |
| 4           | 76                  | 0.95                               | 0.36                    | 8                          |
| 5           | 85                  | 0.7                                | 0.27                    | 7                          |
| 6           | 95                  | 0.5                                | 0.19                    | 7                          |
| 7           | 104                 | 0.38                               | 0.15                    | 7                          |
| 8           | 113                 | 0.3                                | 0.11                    | 7                          |
| 9           | 122                 | 0.24                               | 0.10                    | 7                          |
| 10          | 132                 | 0.2                                | 0.09                    | 7                          |



Table VII.3-4: Power and Coolant Balance of a TASKA Test Module with  
Liquid Lithium Blanket

|                                                            |                        |
|------------------------------------------------------------|------------------------|
| Neutron wall loading                                       | 1.53 MW/m <sup>2</sup> |
| Module power                                               | 3.5 MW                 |
| Average power density in the blanket                       | 0.9 W/cm <sup>3</sup>  |
| Blanket energy multiplication factor                       | 0.9                    |
| Power peaking factor $\frac{P_{\max}}{P_{\text{average}}}$ | 4                      |
| Coolant                                                    | Liquid Lithium         |
| Coolant temperatures                                       |                        |
| Inlet nozzle                                               | 300 °C                 |
| Outlet nozzle                                              | 450 °C                 |
| Coolant pressure                                           | 0.5 MPa                |
| Coolant mass flow                                          | 6 kg/s                 |
| Coolant inventory in cooling loop                          | approx. 3.5 t          |
| Max. coolant velocity at inlet nozzle                      | 2 m/s                  |
| Max. coolant velocity in the blanket tubes                 | 7 cm/s                 |

#### VII.3.2.3.2 General Design

This presents a survey of the general design of the primary coolant loop for a test module. In principle this survey may only be transferred to a plant loop, as no statement is made concerning the extrapolation to larger power and units to be supplied.

The primary coolant loop for a test module with liquid metal blanket essentially consists of the following components, when viewed in the direction of the coolant flow (Fig. VII.3-12):

- module
- main coolant pipes
- degasification system
- purification system
- intermediate heat exchanger
- balancing tank
- primary pump
- drain system

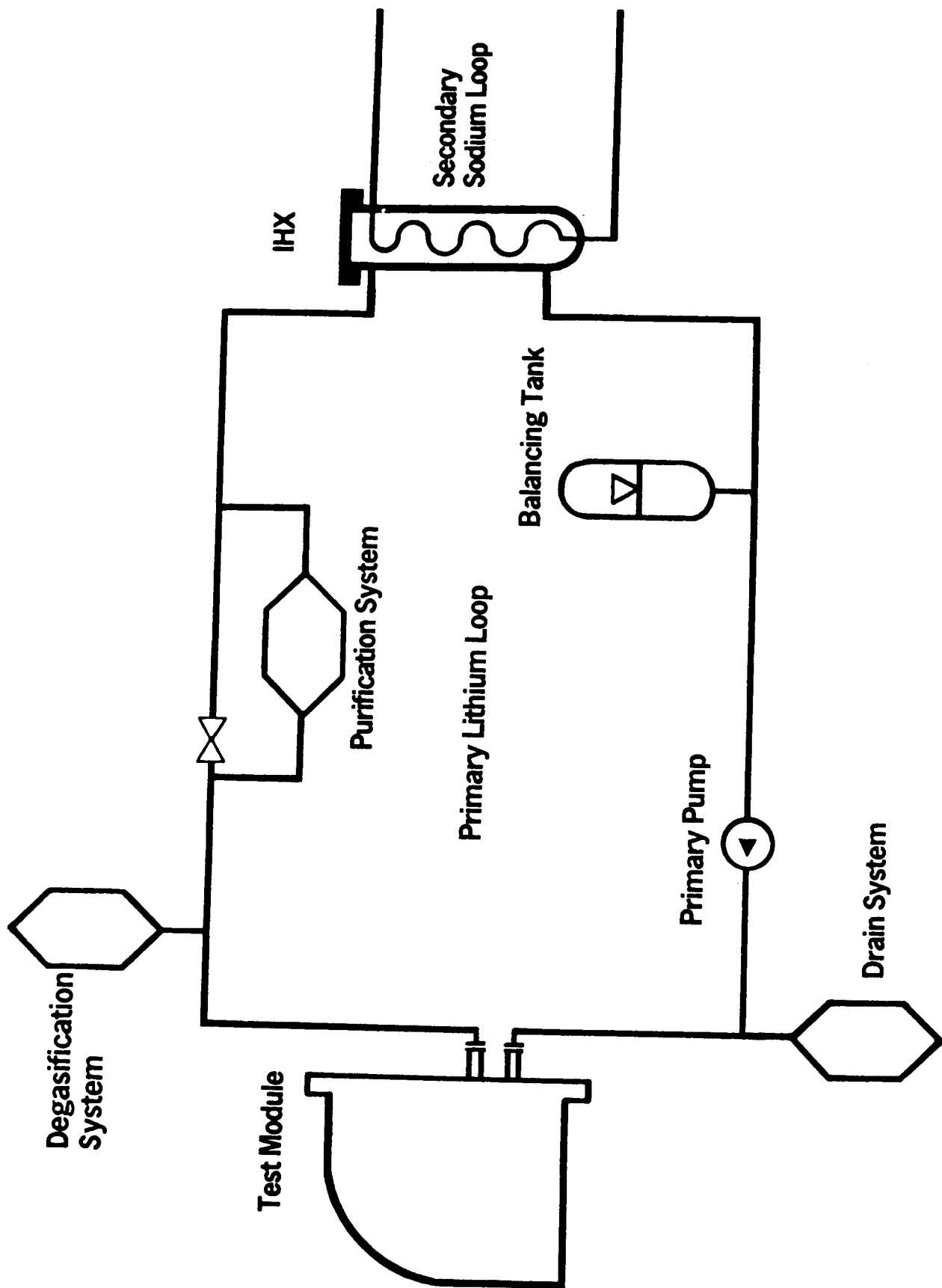
The coolant leaves the test module with a maximum temperature of 500 °C, and a pressure of approximately 0.4 MPa. Thereby, a pressure loss of about 0.1 MPa is assumed in the module.

The gas bubbles present in the coolant are removed in the degasification system which must be located at the highest point in the geometry.

Approximately 10 % of the coolant flow in the primary loop is led through the purification system, which is situated in the hot leg of the loop. The purification system also contains the cold trap system required for the tritium extraction.

The intermediate heat exchanger must be designed for a power of approximately 3 MW per module. In order to keep the coolant pressure and the pumping capacity as low as possible, the pressure loss in the heat exchanger should be less than 0.1 MPa. Approximately 100 m<sup>2</sup> are assumed as heat exchanger area.

The purpose of the balancing tank is to balance any lithium losses which might occur due to leakage and purification as well as any alterations in the volume of the coolant caused by temperature.



**Fig. VII.3-12 TASKA primary coolant loop**

The primary coolant pump transports a maximum of 12 kg/s lithium at a pressure difference of approximately 0.2 MPa per test module.

The temperature at the inlet nozzle of the test module amounts to approximately 300 °C at 0.5 MPa pressure.

The drain system is located between the primary coolant pump and the test module. Its capacity must be at least the inventory of one loop (3.5 t). The drain system is required for the exchange of the module and in the case of repairs to the cooling loop.

The components of the cooling system, in particular all the tanks, must be equipped with heating systems to ensure that the lithium temperature in the loop is higher than the melting temperature of 180 °C.

#### VII.3.2.4 Tritium

##### VII.3.2.4.1 Tritium Production

The calculation of the tritium production rate is based on the neutronic calculation performed by the KfK / 1 /. A tritium breeding factor of 1.32 was determined for the blanket assembly given in Chapter VII.3.2.2 using natural lithium as breeding material. Li6-enrichment is not necessary.

An estimation for a 90 % Li6-enrichment resulted in a 20 % lower breeding factor.

Fig. VII.3-13 illustrates the radial distribution of the tritium production rate in the blanket. It is evident that approximately 75 % of the tritium is produced in the blanket zone up to  $r = 90$  cm.

Taking a tritium breeding factor of 1.32, one obtains a tritium production rate of 280 g/a at  $1.53 \text{ MW/m}^2$  neutron wall loading.

The inventory of breeding material in the test module amounts to approximately 1.4 t. This is approximately 40 % of the lithium inventory stipulated for the cooling loop of one module.

The tritium inventory in the entire cooling loop is limited to 16.6 g as a result of continuous extraction in the purification system.

Table VII.3-5 presents the essential parameters for the production of tritium.

Table VII.3-5: Main Parameters for the Production of Tritium

|                                                   |                                       |
|---------------------------------------------------|---------------------------------------|
| Neutron wall loading                              | 1.53 MW/m <sup>2</sup>                |
| Tritium breeding factor                           | 1.32                                  |
| Tritium production rate                           | 280 g/a<br>= 3 . 10 <sup>6</sup> Ci/a |
| Inventory of breeding material<br>per test module | 1.4 t                                 |
| Tritium inventory in<br>the blanket               | 6.6 g                                 |
| Tritium concentration in<br>the breeding material | 4.75 ppm                              |

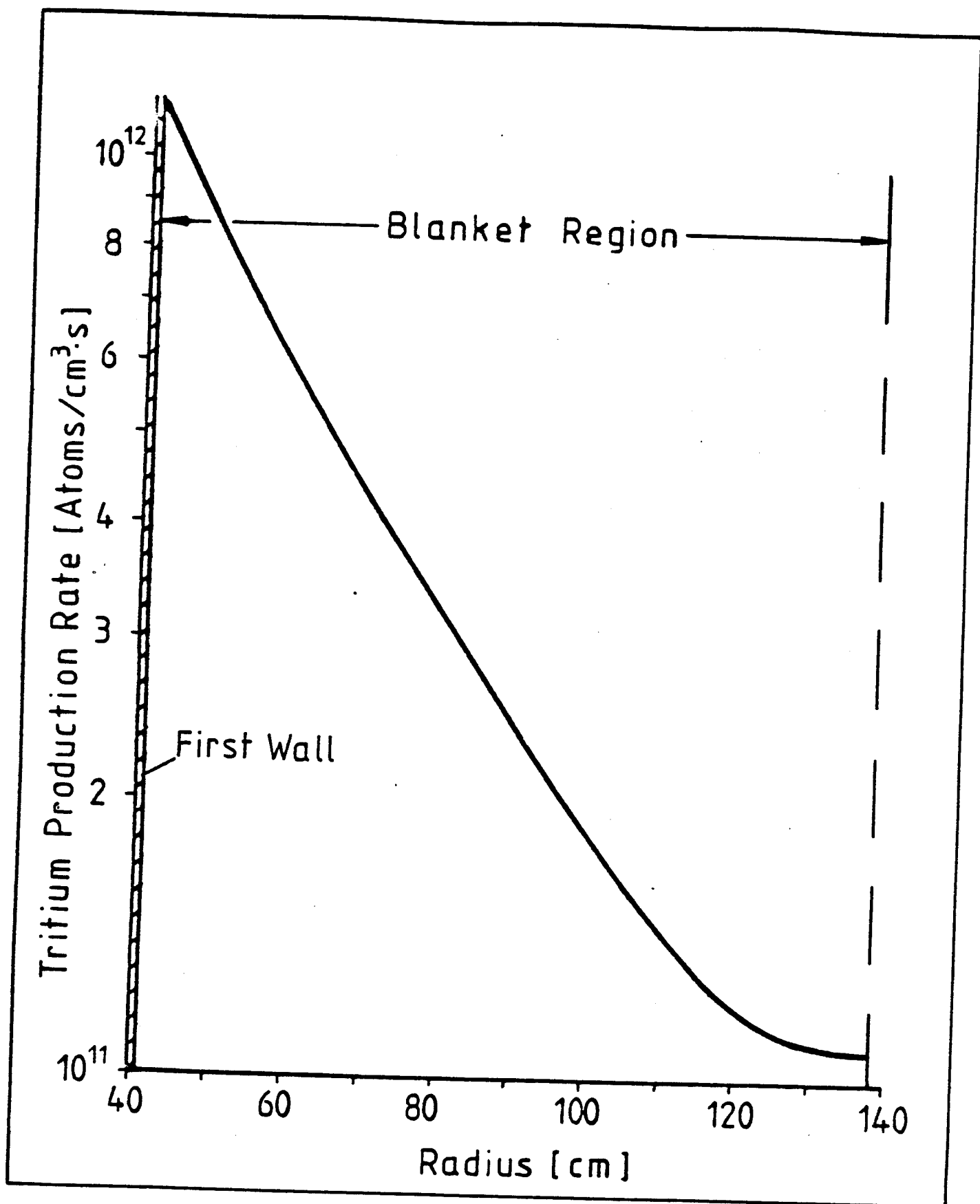


Fig. VII.3-13 Radial Breeding Rate Distribution  
at  $1 \text{ MW}/\text{m}^2$  Neutron Wall Loading

#### VII.3.2.4.2 Tritium Extraction

A concept was designed for an extraction process which applies the separation of hydrogen isotopes from sodium loops using cold traps to lithium systems. For sodium this process has already been tested on a large technical scale. The calculations show that such a process is promising. The majority of the tritium produced in the blanket is separated out in the cleaning section of the lithium loop.

It is necessary to interpose a sodium loop between the lithium and water/steam loop as a tritium barrier.

The following contains a survey of the tritium balance within the heat transport system and an explanation of the extraction method.

As shown in Fig. VII.3-14, the heat transport system consists of a lithium primary loop, a sodium secondary loop and a water/steam loop.

##### VII.3.2.4.2.1 Tritium Balance in the Lithium Primary Loop

According to Chapter VII.3.2.4.1 the tritium production rate was calculated as being  $1.8 \cdot 10^{18}$  T-atoms/s corresponding to about  $3 \cdot 10^6$  Ci/a.

The tritium concentration in the lithium is determined by the cold trap operated at 185 °C. According to Fig. VII.3-15, the total solubility S (saturation concentration) of the hydrogen isotopes at this temperature is  $3.06 \cdot 10^{-2}$  mol % for hydrogen and  $1.12 \cdot 10^{-2}$  mol % for deuterium / 5 /.

An additional external injection of hydrogen, which is higher by a factor of 10 than the tritium production rate, will lead to a tritium concentration of

$$C_T = 4.75 \text{ ppm} = 9.44 \cdot 10^{17} \text{ atoms T/g}_{Li}$$

in the lithium loop. This is due to the simultaneous precipitation of hydrogen and tritium in the cold trap and takes into account the mass dependence of the solubility.

The resulting hydrogen concentration is:

$$C_H = 2.5 \cdot 10^{19} \text{ atoms H/g}_{Li} = 42.8 \text{ ppm}$$

The  $T_2$ -partial pressure which is decisive for the tritium losses from the lithium loop is given by:

$$\sqrt{p_{T_2}} = \frac{C_T}{K_T}$$



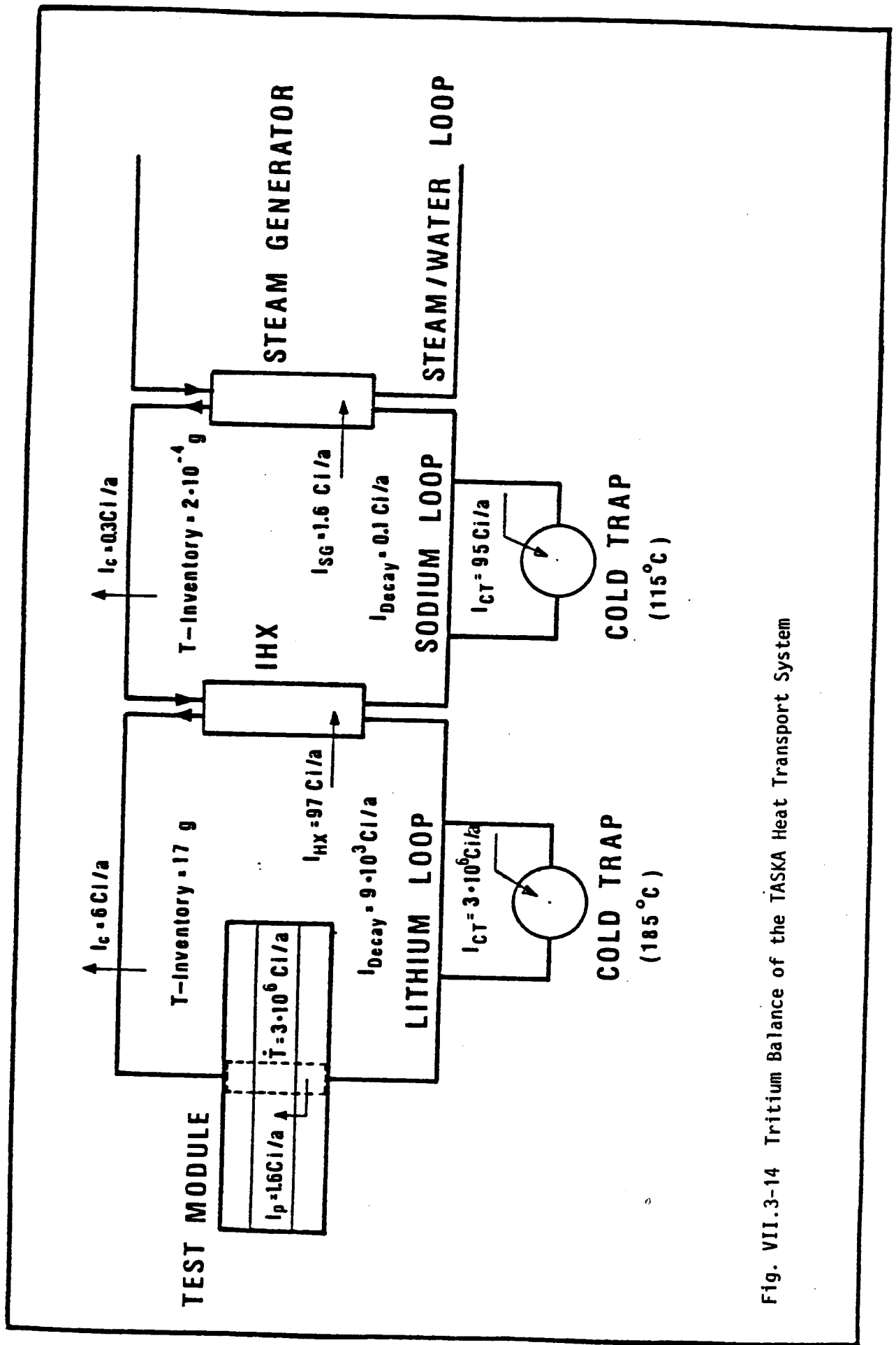


Fig. VII.3-14 Tritium Balance of the TASKA Heat Transport System

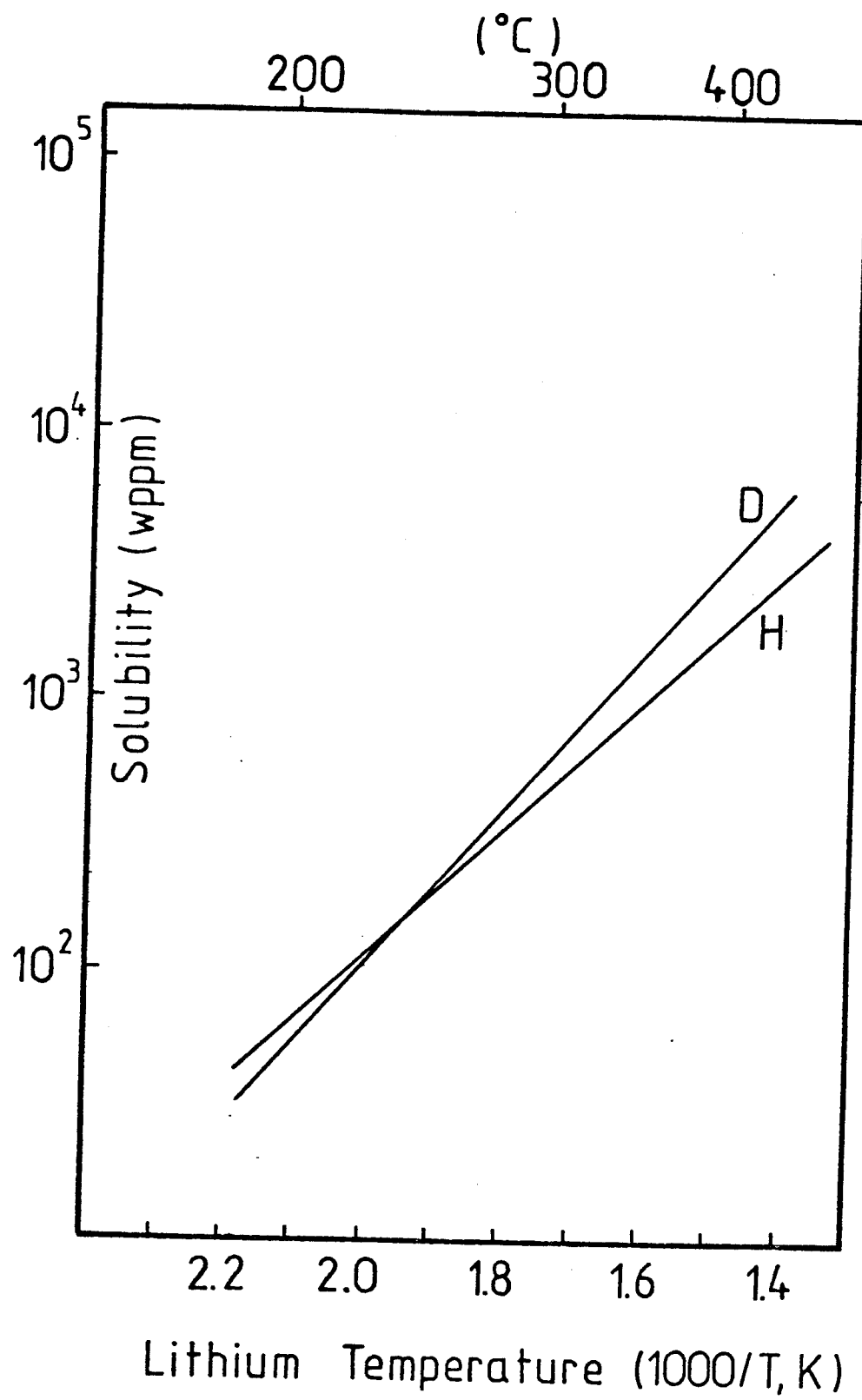


Fig. VII.3-15 Solubility of Hydrogen Isotopes  
in Lithium

$C_T$  : tritium concentration, ppm

$K_T$  : Sievert's constant, ppm . mbar<sup>-1/2\*</sup>

Assuming an average lithium temperature of 450 °C,

$$K_T = 4.64 \cdot 10^4 \text{ ppm} \cdot \text{mbar}^{-1/2} \quad / 6 /$$

As a result,

$$\begin{aligned} \sqrt{p_{T_2}} &= 1.02 \cdot 10^{-4} \text{ mbar}^{+1/2} \\ p_{T_2} &= 1.05 \cdot 10^{-8} \text{ mbar} \end{aligned}$$

In case of the calculated tritium partial pressure of  $1.05 \cdot 10^{-8}$  mbar, the tritium losses via the steam generator (SG) would attain non-permissible high values if the lithium loop were directly coupled to a water/steam loop. At present, approximately 2 Ci/a . MWe may be considered as a recommendation for the tritium discharge in the water/steam loop / 7 /. When transferred to the available reactor concept, the tritium loss rate per test module may amount to approximately 4.6 Ci/a in the water/steam loop. If the lithium loop and the water/steam loop are directly coupled, this value is exceeded by about a factor of 8. For this reason, the interposition of a sodium loop is deemed a suitable means of reducing the tritium discharge rate.

Under equilibrium conditions, the residual tritium inventory  $m_T$  in the lithium is:

$$\begin{aligned} m_T &= m_{Li} \cdot C_T \\ &= 16.6 \text{ g} \quad \approx \quad 2.62 \cdot 10^5 \text{ Ci} \end{aligned}$$

The main tritium losses from the primary loop are:

- Tritium losses from the blanket tubes, headers and coolant pipes in the test module
- Tritium permeation into the sodium loop through the intermediate heat exchanger (IHx)
- Tritium losses through the pipe walls of the primary loop
- Tritium losses due to radioactive decay

The tritium losses due to permeation are calculated according to the equation:

$$I_{i \rightarrow j} = \frac{\emptyset \cdot F}{X} \left( \sqrt{p_{T_2, i}} - \sqrt{p_{T_2, j}} \right) \quad (\text{VII.3-1})$$

whereby  $\emptyset$  = permeability of the component material  
 $F$  = permeation area  
 $X$  = wall thickness  
 $p_{T_2}$  = tritium partial pressure in the medium  
 $i$  = discharge medium  
 $j$  = inlet medium

A value of 250 Ci/a was calculated for the tritium losses through the walls of the blanket tubes, headers and main pipes in the test module.

This value is 2.7 times higher than the permeation rate through the intermediate heat exchanger.

This is an unsatisfactory result, as, according to the present state of the concept, the permeation flow through the tube walls of the blanket area would not enter the plasma but rather the intermediate vacuum area of the casing.

We therefore do not suggest evacuating this space. Instead, it is proposed that a pressure balance between the tritium and the hydrogen in this space and the pressures in the lithium be permitted. After the pressure balance, a  $T_2$ -partial pressure of

$$p_{T_2} = 1.05 \cdot 10^{-8} \text{ mbar}$$

would result in the vacuum space.

If one takes the simultaneous hydrogen permeation and recombination to HT into consideration, the resulting tritium inventory present in this vacuum space amounts to 68  $\mu\text{Ci}$  (available as  $T_2$  and HT), and is negligible. It is therefore not necessary to pump the tritium out of the vacuum space. At the same time, this means that the tritium can now diffuse into the plasma. The total pressure in the vacuum space amounts to  $3 \cdot 10^{-7}$  mbar.

The tritium flow into the plasma via the first wall and the side walls of the casing was estimated as being approximately 1.6 Ci/a. The area of the casing walls relevant for tritium permeation ( $T > 300\text{ }^{\circ}\text{C}$ ) was estimated as being  $6\text{ m}^2$ . The tritium losses as a result of permeation from the entire primary loop amount to 105 Ci/a. Including the losses due to radioactive decay, the total tritium losses are  $9 \cdot 10^3$  Ci/a (Table VII.3-6).

Fig. VII.3-14 presents a survey of the tritium losses from the primary loop.

Table VII.3-6 Tritium Losses in the Primary Loop

| Components                  | $\frac{\phi}{\text{T-atoms/s.cm.mbar}^{1/2}}$ | I / Ci/a /        |
|-----------------------------|-----------------------------------------------|-------------------|
| Test module                 |                                               |                   |
| First wall                  | $3.98 \cdot 10^{11}$                          | 1.3               |
| Casing wall                 | $6.1 \cdot 10^{10}$                           | 0.28              |
| Main coolant pipes          | $3.98 \cdot 10^{11}$                          | 6.16              |
| Intermediate heat exchanger | $3.98 \cdot 10^{11}$                          | 96.8              |
| Radioactive decay           | -                                             | $8.9 \cdot 10^3$  |
| Total                       | -                                             | $9.00 \cdot 10^3$ |

Table VII.3-7 Tritium Losses in the Secondary Loop

| Components         | $\frac{\phi}{\text{T-atoms/s.cm.mbar}^{1/2}}$ | I / Ci/a / |
|--------------------|-----------------------------------------------|------------|
| Steam generator    | $2.4 \cdot 10^{11}$                           | 1.6        |
| Main coolant pipes | $2.4 \cdot 10^{11}$                           | 0.3        |
| Radioactive decay  | -                                             | 0.1        |
| Total              |                                               | 2.0        |

The quantity of tritium to be separated in the cold trap of the lithium loop results from the difference between the tritium production rate  $\dot{T}$  and the tritium loss rate and amounts to 99.7 % of the tritium production rate or 280 g/a. Whether this separation rate can be achieved still has to be clarified experimentally. At present, we are not aware of any criteria which would negate this, especially as it is possible to connect more than one cold trap in series.

#### VII.3.2.4.2.2 Tritium Balance in the Sodium Intermediate Loop

As a result of comparisons of the primary and secondary coolant masses in sodium cooled reactors, one can assume that a sodium mass of

$$m_{\text{Na}} = 3 \cdot 10^6 \text{ g}$$

will be necessary for this concept. In this order of magnitude, this value can be considered relevant. On the whole, however, this is not a critical parameter for the tritium balance.

At an average sodium temperature of 420 °C, the Sievert's constant for tritium is:

$$K_T = 9.627 \text{ ppm} \cdot \text{mbar}^{-1/2} \quad / 8 /$$

Due to corrosion of the steam generator (SG) on the water side, a hydrogen flow enters the sodium loop from this side. As a result of comparison with the conditions in sodium-cooled reactors, one can assume that, in the case under question, the ratio of the flows of hydrogen (via SG) to tritium (via IHX) entering the sodium loop is about  $10^3/1$ , whereby this can be considered a conservative assumption.

The cold trap in sodium is operated at 115 °C. According to Fig. VII.3-16, the total solubility of the hydrogen isotopes at this temperature is:

$$S = 2.357 \cdot 10^{-2} \text{ ppm} \quad / 9 /$$

The following tritium and hydrogen concentrations result from this:

$$C_T = 7.03 \cdot 10^{-5} \text{ ppm} = 1.397 \cdot 10^{13} \text{ atoms T/g}_{\text{Na}}$$

$$C_H = 2.35 \cdot 10^{-2} \text{ ppm} = 1.397 \cdot 10^{16} \text{ atoms H/g}_{\text{Na}}$$

The tritium partial pressure is therefore:

$$\sqrt{p_{T_2}} = 7.302 \cdot 10^{-6} \text{ mbar}^{1/2}$$

$$p_{T_2} = 5.33 \cdot 10^{-11} \text{ mbar}$$

The residual tritium inventory in the sodium is:

$$\begin{aligned} m_T &= m_{\text{Na}} \cdot \epsilon_T \\ &= 3 \cdot 10^6 \cdot 7.03 \cdot 10^{-11} \text{ g} \\ &= 2.11 \cdot 10^{-4} \text{ g} = 2.02 \text{ Ci} \end{aligned}$$

and is therefore negligible in contrast to the tritium inventory in lithium.

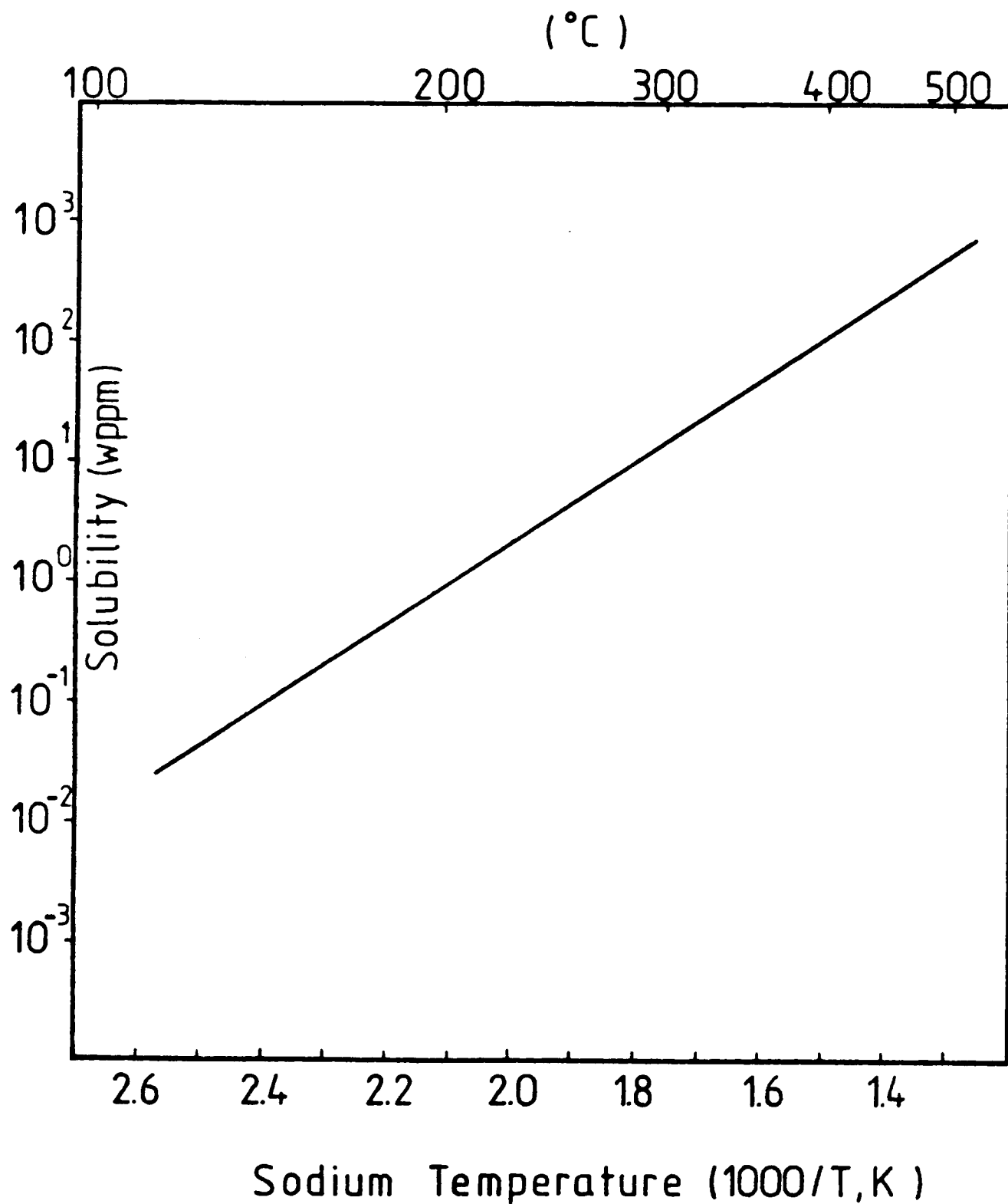
When calculating the tritium losses from the secondary loop, no attention is paid to oxide barriers on the water side of the steam generator. Firstly, it is well known that the tritium barrier effect produced by oxidation on the secondary side is not very large and, secondly, such a barrier effect cannot be guaranteed for all operating conditions of the reactor.

The tritium losses from the intermediate loop result from the tritium concentration in sodium determined by the cold trap.

As in the outer lithium loop, all components are manufactured in austenitic steel.

The tritium losses from the sodium secondary loop comprise the following:

- Permeation losses via the steam generator
- Permeation losses from the pipes
- Losses due to radioactive decay



**Fig. VII.3-16 Solubility of hydrogen isotopes in sodium**



The loss rates listed in Table VII.3-7 were calculated in accordance with the calculation method given in equation VII.3-1. The total tritium losses from the secondary loop amount to approximately 2 Ci/a.

The tritium overflow rate of 1.6 Ci/a into the water/steam loop is an acceptable value for a fusion reactor.

Fig. VII.3-14 contains a survey of the tritium losses from the secondary loop.

The tritium separation in the cold trap of the sodium loop results from the difference between the tritium flow via the intermediate heat exchanger, and the tritium losses, and is:

$$6.25 \cdot 10^{13} \text{ atoms T/s} = 94.8 \text{ Ci/a}$$

This is approximately 98 % of the tritium flow entering the intermediate loop.

#### VII.3.2.4.2.3 Tritium Recovery from Cold Traps

The tritium recovery from the cold traps of the primary and secondary loop is effected after the dismantling of the cold traps. In principle, it is possible to reprocess the cold trap and therefore to extract the tritium gained before the maximum capacity of the cold trap has been attained, thus obtaining a quasi-continuous tritium extraction. For this reason, only the quantity of tritium necessary for reactor operation until the next extraction interval should be stored.

As the extraction intervals for TASKA have not been established to date, it is assumed that the tritium will be recovered after the maximum capacity of the cold trap has been attained.

The capacity of cold traps is determined by the hydrogen source strengths: Operating experience gained with sodium systems shows that the capacity of cold traps for hydrogen amounts to about 10-20 % of the active volume. If one assumes that the capacity of cold traps in lithium is determined by the same effects as in sodium, a comparison with a sodium cold trap having a volume of  $1 \text{ m}^3$  leads to the assumption that the capacity of a lithium cold trap with the same volume corresponds to a separated hydrogen mass of about 2 kg.

Taking the primary tritium and hydrogen source strengths

$$S_T = 1.8 \cdot 10^{18} \text{ atoms/s}$$

$$S_H = 1.8 \cdot 10^{19} \text{ atoms/s}$$

the cold trap in the lithium loop has a service life of about 590 full load days  $\cong$  1.5 FPY. The tritium activity separated in this period is  $4.5 \cdot 10^6$  Ci.

Due to the lower source strengths of the hydrogen isotopes in the secondary system, the service life of the sodium cold trap is greater than the service life of the reactor itself. The tritium activity to be regenerated amounts to  $2 \cdot 10^3$  Ci for a service life of approximately 20 full load years. At this point in time, only a small proportion of the capacity of the cold trap is exhausted.

These considerations presuppose normal operating conditions. Disturbances of normal operation, which are linked with an increase in the hydrogen inlet rate, shorten the service life of the cold traps.

With reference to the lithium cold trap, greater requirements are made on the regeneration system than in case of the secondary cold trap. At present, no experimental experience is available on the regeneration of lithium cold traps, but one can assume that the same processes are suitable for these as for sodium cold traps.

The cleaning of sodium cold traps has been tested on a large technical scale. There are two ways of extracting the hydrogen isotopes from the coolant. This is either effected by continually evacuating the gas space above the coolant or by blowing argon scrubber gas over the surface. In both methods, the hydrogen partial pressure above the coolant is greatly decreased, and, as a result, metal hydride precipitated in the cold trap continuously dissolves and becomes gaseous. The kinetics of this process have been investigated in detail for sodium / 10, 11 /.

The efficiency of both processes increases greatly with increasing temperature. The vacuum extraction is more efficient than the scrubber gas extraction. The efficiency is increased if turbulences are generated in the coolant.

In the lithium systems, the gas discharge to be attained is lower by several orders of magnitude than in sodium at the same temperature due to the lower equilibrium pressure of the hydrogen isotopes. In order to obtain comparable extraction efficiencies, higher lithium temperatures and larger free surfaces are required. By extrapolating the data from / 10, 11 /, it is to be expected that, when these are transferred to lithium, bearing in mind the lower steam pressure, it will be possible to regenerate the lithium cold trap with a hydrogen extraction rate of about  $30 \text{ cm}^3 \text{ (STP)/s}$ . (This value is based on a lithium temperature of  $650^\circ\text{C}$  and a  $1 \text{ m}^2$  free surface in the cold trap).

The resulting regeneration time amounts to approximately 400 h.

Depending on the inventory, it is possible to regenerate the sodium cold trap in a shorter time.

The gas which occurs during the regeneration of the cold traps consists of tritium and hydrogen. In order to close the fuel cycle, it is necessary to separate these two isotopes. The most promising separation method is considered to be a permeation process in the gaseous phase, in which the mass-dependent diffusion after the diffusion step causes an enrichment of the hydrogen in relation to tritium. In order to attain sufficiently high separation factors, it is necessary to apply a cascade separation with several subsequent permeation steps. The strength of the occurring gas flows necessitates the use of materials with high permeability and working temperatures around  $500^\circ\text{C}$ . For example, niobium with a permeability of  $9.78 \cdot 10^{-5} \text{ cm}^3 \text{ (STP) /cm.s.mbar}^{1/2}$  is suitable for hydrogen at  $500^\circ\text{C}$ . In order to achieve the permeation of a gas flow of  $30 \text{ cm}^3/\text{s}$  (gas production when regenerating the lithium cold trap) the surface area of a 2 mm thick permeation diaphragm must amount to approximately  $1900 \text{ cm}^2$ .

#### VII.3.2.4.2.4 Tritium Storage

The tritium recovered from the cold traps of the primary loop and, if necessary despite the low inventory, of the secondary loop must be stored externally. In principle there are two methods of doing this. One method is to directly confine the gaseous tritium in a steel tank. The other method is to bind the tritium to hydride forming metals, e. g. titanium sponge.

If one assumes one regeneration process of the cold traps per operating year, 52 mol tritium from the lithium loop must be stored externally.

In the interest of safety, the storage of tritium in a steel tank should be effected at slight underpressure. At 0.1 bar underpressure in the tank this must have a volume of  $1.4 \text{ m}^3$  for storage at room temperature. Due to the high activity inventory in this tank ( $3 \cdot 10^6 \text{ Ci}$ ), however, the second form of storage is more advantageous in the interests of safety, as any leakages in the tank are not directly connected with serious radiological consequences due to the considerably lower tritium pressure above the titanium sponge.

The partial pressure of the tritium above the getter material is  $< 10^{-3} \text{ mbar}$ , if the getter material is cooled down to room temperature after being loaded with tritium at  $300^\circ\text{C}$ .

10 kg titanium are required to store the tritium at titanium sponge.

#### VII.3.2.4.3 Safety Considerations

The use and handling of tritium require special safety measures, as it has a very good diffusivity and permeates through steel at high temperatures. This applies to the lithium primary loop in particular, as a constant tritium level has already formed in the entire loop after a short time.

The components of the primary loop must have as good integrity as possible for the tritium containment. Connections should be welded and tested using helium leak tests.

In order to improve the tritium enclosure, the application of double-walled pipes and permeation inhibiting pipe coatings could be investigated. However, double-walled pipes are only prudent at high tritium discharge rates, as the high manufacturing costs for the pipes and the extra costs for scrubber systems could make up a large part of the construction costs.

Similar, the use of permeation-inhibiting coatings is only of slight importance, as only complete coating of the loops leads to verifiable tritium retention. It is, however, impossible to guarantee complete coating at the junction points for example, or because of cracks in the coating as a result of different thermal loadings.

Furthermore, it is impossible to prevent tritium contamination of the room atmosphere, as, for example, tritium diffuses through the flange sealings of the module or permeates through the walls of the coolant pipes.

More important, however, are the losses from the reactor fuel supply system and the plasma heating system which also contribute to room contamination. In order to keep the tritium discharge to the environment as low as possible, those parts of the plant transporting tritium must be enclosed in a second containment. This second containment is formed by the rooms of the plant, which are hermetically sealed off and for which the following measures must be provided:

- Monitoring of the room atmosphere with detectors
- Scrubbing the gas inventory of the rooms
- Build-up of an underpressure in tritium contaminated rooms
- Sealing the walls of the containment to impede tritium diffusion

The rooms must be supplied with locks for the transportation of components.

As the handling processes are essentially determined by the activation of the components and, for the most part, are remotely controlled, no further essential measures are necessary with regard to the external radiation exposure due to tritium. If contaminated rooms are entered for service or repair work, respiratory equipment should be worn, or if manipulator cabins are used, these should be equipped with a separate ventilation system in order to prevent a non-permissible high radiation exposure of the personnel due to the incorporation of tritium. Nevertheless, all free volumes of dismantled components should be checked for a possible tritium contamination before being removed from the containment.

### VII.3.2.5 Cost Estimates

With reference to the costs which are up to completion of a test module for:

- planning
- manufacturing design
- manufacturing plus test and TÜV acceptance costs
- material costs,

only those costs for the supply of material can be realistically estimated.

The material costs listed in Table VII.3-8 were determined on the basis of 1981 prices. Based on a test module with liquid metal blanket, they amount to a total of approximately 700 TDM, excluding the material costs for the shield. These material costs essentially depend on the specification of the B<sub>4</sub>C layer. The costs for B<sub>4</sub>C can be estimated as amounting to approx. 150 DM/kg according to Chapter XI.

Table VII.3-8: Estimated Material Costs for a Test Module  
with Liquid Metal Blanket (basis 1981)

| Component              | Quantity      | Material         | Unit Value | Total Costs<br>TDM |
|------------------------|---------------|------------------|------------|--------------------|
| Blanket tubes          | 320 m         | 1.4970           | 170 DM/m   | 55                 |
| Nozzles and<br>headers | approx. 0.5 t | 1.4970           | 25 DM/kg   | 12                 |
| Cooling lines          |               |                  |            |                    |
| Casing                 | approx. 8 t   | 1.4970           | 30 DM/kg   | 240                |
| Reflector              | approx. 12 t  | 1.4970           | 30 DM/kg   | 360                |
| Shield                 | approx. 20 t  | 1.4970           | 30 DM/kg   |                    |
|                        |               | Pb               | 5 DM/kg    |                    |
|                        |               | B <sub>4</sub> C | 150 DM/kg  |                    |
| Total (without shield) |               |                  |            | 660                |

At this point in the investigation, it is only possible to make tendentious estimates regarding the anticipated planning costs and the manufacturing documents which have to be prepared up to construction maturity.

The cost expenditure can be considerably reduced if it is possible to make use of already available knowledge, as, in this case, the sodium technology.

At present, however, this new technology is accompanied by problem points caused by the difficulties of the lithium technology, which cannot be summarized as yet, the compatibility of liquid lithium with the austenitic steel 1.4970 planned as structural material and the suitability of this material as structural material at high neutron fluxes and high temperature due to the unfavourable fatigue behaviour. If one considers the experience gained in the sodium technology with reference to the costs for the planning of comparable components, for example heat exchangers, the costs for planning and manufacturing design will probably be in the order of magnitude of the material costs estimated here.

At this point in the concept, it is very difficult to estimate the manufacturing costs without the costs for licensing and TOV acceptance. After a first estimate, however, these should be approximately twice the material costs. Further detaillling of the concept would be necessary for more exact cost data.

Table VII.3-9 shows an overview of the cost-estimates for one test module.  
Table VII.3-9 Survey of the Costs for a Test Module (basis 1981)

|                                             |         |          |
|---------------------------------------------|---------|----------|
| Planning                                    | approx. | 200 TDM  |
| Manufacturing design                        | approx. | 800 TDM  |
| Manufacturing without<br>test and TOV costs | approx. | 2000 TDM |
| Material costs                              | approx. | 700 TDM  |

(excluding shield)

---

|       |         |          |
|-------|---------|----------|
| Total | approx. | 4000 TDM |
|-------|---------|----------|

### VII.3.2.6 Operating and Other Peripheral Systems

#### VII.3.2.6.1 Operating Systems

A large number of auxiliary systems, instrumentation and control systems as well as measuring systems are required to ensure authorized operation of the test module. The survey compiled here essentially includes those systems in the primary loop mainly necessary for normal operation. It does not take into account any protective systems which may be additionally requested during the necessary licensing procedure or those which are additionally required for accident control. In addition to the main cooling and tritium systems already described in the preceding Chapters VII.3.2.3 and VII.3.2.4, the following auxiliary and secondary systems are also necessary for normal operation.

#### VII.2.6.1.1 Cooling Systems

##### A Cooling loop for the reflector and shield plug

###### Task

Heat removal from the reflector and shield structures by means of water cooling.

###### Requirements

- Provision of the necessary quantity of coolant with the values specified for pressure and temperature
- Installation of devices to monitor the tritium contamination and the contamination with activated corrosion products

##### B Emergency cooling systems

###### Task

Heat removal mainly from activated structures after failure of the main cooling systems

###### Requirements

- Design for the removal of decay heat even after accidents such as explosions or earthquakes
- Redundancy and diversity of the systems
- Energy supply ensured even in the case of an accident



#### VII.3.2.6.1.2 Evacuation Systems

##### A Evacuation system for the flange sealings

###### Task

Evacuation of the space in the double sealing to prevent leakages into the plasma volume and the casing.

###### Requirements

- Provisions of the pumping capacity required for evacuation
- Provisions for monitoring of leakages and tritium contamination

##### B Evacuation system for the module casing

###### Task

- Evacuation of the volume of the casing to approximately  $10^{-3}$  Pa
- Injection of hydrogen to lower tritium permeation from the blanket into the casing

###### Requirements

- Provision of the pumping capacity required for evacuation
- Connection to a hydrogen supply system
- Monitoring of the tritium inventory in the casing possible
- Integration of a tritium cleaning system or connection to the fuel cycle cleaning system

#### VII.3.2.6.1.3 Cover Gas Systems

##### A Cover gas system

###### Task

- Guarantee of a cover gas atmosphere above free surfaces in components carrying lithium e.g. Li-pumps, Li-tanks
- Build-up and control of the system pressure in the tanks
- Loading of sealing gas and pressure gas at seals
- Supply of the handling stations

###### Requirements

- Provision of the necessary quantity of cover gas even under accident conditions
- Monitoring of the tritium inventory in the cover gas possible
- Installation of a tritium and lithium aerosol cleaning loop

## B Inerting System

### Task

Inerting and cooling of certain main and secondary facilities such as buildings or handling systems, if this is required by precise safety requirements or specifications from the regulatory commissions.

### Requirements

- Installation of gas loops to scrub and reinert containment cells
- Installation of cooling loops to control the temperature in the rooms and to cool the gas inventories of certain handling components

## VII.3.2.6.1.4 Component Heating Systems

### Task

Maintenance of a given lithium temperature  $> 180\text{ }^{\circ}\text{C}$  in components such as leak pans, cleaning pipes etc.

### Requirements

- Separate gas heating (e.g. He) to heat components such as blanket, headers, intermediate heat exchanger before filling with lithium
- Electrical heating of the components (trace heating)
- Connection to emergency power supply

## VII.3.2.6.2 Measuring and Control Systems

### VII.3.2.6.2.1 Measuring Systems of the Test Module (Survey)

- Neutron flux

Radial and axial distribution of sensors which in conjunction with a slide-in technique could be replaced without dismantling the entire test module.

- Temperature

Measurement of the temperatures of:

- Blanket structure
- First wall
- Reflector and shield
- Casing walls
- Coolants
- Breeding material

The sensors must exhibit good radiation resistance due to the high-energy neutron flux.

- Monitoring of lithium leakage inside the test modules in connection with the evacuation system of the casing.
- Mass flow  
Measurement of the mass flow quantity of breeding material and coolant. Due to the magnetic field of the fusion plant, it is difficult to apply inductive measuring techniques. Only hydraulic measuring methods can be used.
- Magnetic field  
Measurement of the magnetic field in axial and radial distribution by means of inductive measuring methods.

#### VII.3.2.6.2.2 Measuring Systems Primary Loop

- Pressure  
Measurement of the coolant pressure using diaphragm, pressure and differential pressure measuring techniques
- Mass flow  
Measurement of the lithium mass flow using hydraulic flow meters, or permanent magnetic flow meters at a sufficient distance from the magnetic field of the plant
- Temperature  
Measurement of the coolant temperature by means of sensors
- Impurities  
Determination of the impurity level of the lithium in the cleaning circuit using plugging meters
- Leakage  
Determination of leakages from pipes and components by means of
  - electrical short circuit technique
  - photo diodes
  - measurement of the radiation level

#### VII.3.2.6.2.3 Control Systems

- Temperature  
Control of the coolant outlet temperature of the test module within the range 400 - 500 °C by controlling the pumping capacity

- After heat

The structure temperature is limited by control of the mass flow within the emergency cooling system in the case of failure of the main cooling system.

#### VII.3.2.6.3 Handling Systems

The requirements for the handling systems of the test modules are essentially determined by the type of activities to be performed, the radiation level in the working area and the room conditions.

All activities must be performed using remote handling due to the high radiation level mainly resulting from the activated structures and considering the tritium contamination.

With reference to the test modules, the main activities to be performed with the handling facilities are:

- Insertion and withdrawal of the test modules
- Unscrewing of screwed connections
- Disconnecting and connecting of the supply lines of the modules
- Insertion and dismantling of the reflector and shield plug
- Transport of the modules
- Visual inspection of the module casing and blanket area after removal of the reflector and shield plug
- Cutting and welding of the pipes in the blanket area

These activities can either be performed using remotely controlled manipulator facilities equipped with television cameras, or using manipulators attached to shielded cabins.

In general, the handling methods should be based on the following provisions:

- All handling processes are carried out after the reactor has been shut down
- Free lithium surfaces are covered with a protective gas atmosphere, e.g. argon, in order to prevent the lithium reacting with the air in the room
- If cooling of the components to be handled is necessary, this is carried out by forced circulation of the cover gas or by means of heat conduction and thermal radiation.

#### VII.3.2.6.4 Buildings and Infrastructure

The requirements for the buildings are essentially determined by the distribution of the facilities and components to the rooms. Fundamentally, the requirements of the valid safety regulations and, in the case of operation as a nuclear plant in the Federal Republic of Germany according to § 7 AtG or as a plant according to § 11 AtG, those of the Radiation Protection Regulation must be fulfilled. The following comments therefore only provide an overall view and do not include any additional requirements compiled by the licensing and surveillance authorities.

For example, the following requirements for buildings can generally be taken into consideration:

- Sealing of rooms containing activity to prevent penetration or leakage of water
- Gas-proof penetrations of pipes and cables
- Decontaminable paintwork in rooms containing activity
- Ventilation facilities designed to remove large quantities of heat (accident)
- Installation of fire-protection facilities, in particular in rooms containing lithium and sodium
- Separation of rooms into rooms possibly containing activity and rooms with conventional plant components

Rooms containing activity can be grouped together into one control area.

This comprises for example:

- Reactor hall
- Plant rooms of the primary loop
- Plant rooms of the tritium system
- Rooms for the repair of activated components

Furthermore, it should be planned to separate rooms containing facilities with lithium or sodium from those containing water-bearing facilities.

The reactor hall is subjected to special safety requirements.

The walls of the hall (radiation protection walls) must limit the radiation level outside the hall to the value permitted for that area (for example, operational surveillance area 500 mrem/a).

Tritium contamination of the atmosphere of the hall is to be expected. This requires gas-proof enclosure of the volume of the hall (containment). Accesses to adjoining rooms must be provided with locks. The atmospheric pressure in the hall should be lower than the ambient pressure to prevent a gas flow to the outside. The atmosphere of the reactor hall must be monitored for a non-permissible high tritium level, and the gas volume must be regularly scrubbed.

The ventilation facilities must be designed to remove any possible activity leakages by suction.

The infrastructure refers to the site of the plant and its supply with connections required for operation. Based on the operating concept of the plant as a test facility, the following supply connections must generally be available

- Access to the road network or to an equivalent waterway via a well-constructed communication road for the delivery of large components weighing several tons
- Cooling, domestic and waste water
- Mains electric connection for the energy supply to operate the plant and for the general supply
- Connections to communication systems (telephone, telex)
- Possibilities for the removal and final storage of waste material (radioactive and conventional)
- Facilities for monitoring radioactive emissions (meteorological stations, activity measurements in the settling tank etc.)

#### VII.3.2.7 Problems Regarding Development and Technology

The considerations of the problems arising in the development and technology of tritium breeding blankets with liquid lithium as breeding material and coolant may be divided into general points, which are valid for all liquid metal blankets and specific problems concerning the blanket concept applied here.

##### VII.3.2.7.1 General

The general problem points include the reliability of the cross section data which are at present available for natural lithium for the calculation of the tritium breeding rate, the compatability of lithium with austenitic steel (corrosion rate, mass transport) on a large technical scale and the improvement of austenitic steel for use in fusion plants.

When calculating the tritium breeding ratio for fusion facilities, which use natural lithium as breeding material, a deviation of 20 to 30 % was assumed for one-dimensional calculations of the tritium breeding ratio, and, as a result, a breeding factor of 1.2 to 1.3 was sufficient to obtain a facility breeding factor of 1.

With reference to the Li7 cross sections, which made up approximately 30 % of the total breeding factor in the TASKA calculations, newer estimations / 12 / resulted in values in the relevant energy range which were approximately 26 % lower than the ENDF/B data used to date. As a result of the neutron multiplication effect of the Li7 ( $n, n'\alpha$ ) T-reaction, this does not only influence the Li7 contribution to the breeding factor but also the tritium ratio from Li6.

At present, the compatability of lithium with austenitic steel is only being examined in small loops on bench scale. In addition to lithium corrosion, the experimental programme also includes mass transport and the effect of impurities in lithium.

It is only fundamentally possible to transfer the experimental results to the requirements and effects in plants with larger dimensions, higher temperatures and, above all, using materials under neutron irradiation. Detailed investigations of this point are therefore necessary and should be performed before a reactor-relevant blanket concept is determined in test facilities.

The same is valid for the application of austenitic steel in fusion reactors. Taking the special lithium characteristics into consideration, it is possible to make use of the experiences gained from the liquid metal technology in breeder reactors when selecting the material for loop components. In addition to the problems found in the liquid lithium technology, problems caused by radiation damages (expansion, He-production) also arise in areas with high neutron fluxes. These have a considerable effect on the operating behaviour (temperatures approximately 450 °C, low plant efficiency), and therefore call for the improvement of austenitic steels by means of suitable manufacturing processes and compositions or the development of new materials, whereby the time required to develop new materials up to production on a large technical scale (approximately 30 years) must be taken into consideration.

#### VII.3.2.7.2 Special Problems Concerning the Blanket Concept and Tritium Extraction

The special problems of the blanket concept refer to:

- The cooling of the first wall and the side walls of the casing
- The welds of the blanket tube nozzles
- The welding design of the casing
- The coolant distribution on the blanket tubes

Concept proposals for the cooling of the first wall and the side walls of the module casing were presented in Chapter VII.3.2.2.

As regards the cooling system of the first wall, the problems are not so much to be found in the choice of coolant - in addition to lithium, it is also possible to use helium - as in the temperature gradient between the inner side of the first wall and the cooling tubes, which is due to the power production in the first wall. For this reason, it is necessary to optimize the first wall in a further concept step with reference to the wall thickness necessary for the stability and the lowest possible power production. Due to the radial temperature gradient in the first wall, the temperature differences between the cooling tubes and the first wall, and the resulting diverse expansion, welding of the cooling tubes to the first wall, which could achieve better heat transmission should, nonetheless, be rejected.

The pressing of the cooling tubes on to the first wall calls for the examination of suitable materials, in order to improve the heat conductivity between cooling tubes and the first wall with the available contact pressure.



The cooling system of the side walls calls for further examinations of the power production in the structure and the heat transport out of the first wall area, as well as the stress-free welding of the guiding dents to the wall structure.

The welding design of the casing and blanket areas should be detailed within the framework of further consideration of the concept.

This includes the welding of the nozzles between blanket tubes and the headers, in particular with reference to remotely-controlled maintenance and repair in the blanket area at a later date, as well as the design and production of the shielding edges of the casing for vacuum tight welding and stable reinforcement of the casing structure.

Another problem point requiring further consideration is the coolant distribution in the blanket tubes, taking into consideration the radial gradient in the power production of the blanket. Due to the low hydraulic pressure loss in the blanket tubes, the sole use of restrictors and similar throttle devices would not be able to attain the required mass flow distribution, in particular because of the influence exerted by the magnetic field on the coolant flow. Examinations using flow models are therefore considered necessary.

Some important problem points regarding the extraction and storage of tritium have already been dealt with in Chapter VII.3.2.4. To summarize, one can say that the extraction concept for tritium using cold traps is theoretically very promising, and that lithium purification by means of cold trap operation is being investigated / 13 /. It is, however, necessary to carry out further extensive experimental investigations, in particular with reference to the separating effect of the cold trap before this concept is used on a large technical scale.

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## Appendix

### TASKA Test Module

#### Parameter list for the liquid lithium blanket

Table I

##### I Casing

1. Geometric data:

|                                             |                             |
|---------------------------------------------|-----------------------------|
| Length<br>(incl. reflector and shield plug) | 3730 mm                     |
| Width                                       | max. 1160 mm<br>min. 910 mm |
| Graduations                                 | 50 mm                       |
| First wall diameter                         | 800 mm                      |
| First wall length                           | 1010 mm                     |
| First wall area                             | 2.54 m <sup>2</sup>         |
| Width of coating and clearance resp.        | 1800 mm                     |
| Wall thicknesses:                           |                             |
| First wall                                  | approx. 10 mm               |
| Casing wall                                 | 20 mm                       |
| Base plate                                  | 40 mm                       |

2. Flange construction

|                  |                                                |
|------------------|------------------------------------------------|
| Height           | 3250 mm                                        |
| Width            | 1520 mm                                        |
| Depth            | 300 mm                                         |
| Flange width     | horizontal 180 mm<br>vertical 200 mm           |
| Composition      | Double O-ring seal with inter-space evacuation |
| Sealing material | Viton or Inconel vapour deposited with silver  |

3. Pressure conditions

|                                 |                             |
|---------------------------------|-----------------------------|
| Casing inner pressure           | approx. 10 <sup>-3</sup> Pa |
| Max. loading of the casing wall | 0.1 MPa                     |

4. Material Austenitic steel 1.4970  
(X10 Cr Ni Mo Ti B 15 15)

5. Weight of the casing approx. 8 t

## II Blanket

### 1. Composition

Structural material: Austenitic steel 1.4970  
(X10 Cr Ni Mo Ti B 15 15)  
Coolant Liquid lithium  
Breeding material Natural lithium (92.5 % Li7,  
7.5 % Li6)  
Blanket composition 73 % Li<sub>nat</sub>  
7 % steel  
20 % void

### 2. Dimensions

Length  $r = 48 \text{ cm}$  920 mm  
 $r = 138 \text{ cm}$  820 mm  
Thickness 960 mm  
First wall radius 400 mm  
First wall area  $2.54 \text{ m}^2$

### 3. Design

General rows of tubes arranged round  
the first wall cylinder

Number of blanket tubes 74

Dimensions of the blanket tubes

Outer diameter 101.6 mm

Wall thickness 2.6 mm

Length min. 3300 mm

max. 5500 mm

Total length of the blanket tubes 320 m

Tube material Austenitic steel 1.4970  
(X10 Cr Ni Mo Ti B 15 15)

Rupture strength of 1.4970 for

$t = 20.000 \text{ h}$ ,  $T_{\max} = 550 \text{ }^{\circ}\text{C}$   $110 \text{ N/mm}^2$

Weight of the blanket tubes 6.4 kg/m

4. Power balance
 

|                                        |                        |
|----------------------------------------|------------------------|
| Fusion power                           | 86 MW                  |
| Neutron wall loading                   | 1.53 MW/m <sup>2</sup> |
| First wall heat loading                | 0.8 W/cm <sup>2</sup>  |
| Blanket energy multiplication (theor.) | 1.02                   |
| Power production of the first wall     | 0.3 MW                 |
| Blanket power                          | 3.2 MW                 |
| Blanket volume                         | 3.6 m <sup>3</sup>     |
| Max. power density in the blanket      | 3 W/cm <sup>3</sup>    |
| Energy output per 14.1 MeV-neutron     |                        |
| Blanket:                               |                        |
| Neutrons                               | 11.47 MeV              |
| Gammas                                 | 2.84 MeV               |
| Total                                  | 14.3 MeV               |
  
5. Cooling
 

|                      |        |                         |
|----------------------|--------|-------------------------|
| Coolant temperature  |        |                         |
|                      | inlet  | 300 °C                  |
|                      | outlet | 450 °C                  |
|                      |        | (max. 500 °C)           |
| Coolant velocity     |        |                         |
| Inlet pipe           |        | max. 2 m/s              |
| Blanket tube         |        | approx. 7 cm/s          |
| Pressure loss        |        | approx. 0.1 MPa         |
| (without MHD losses) |        |                         |
| Coolant pressure     |        | 0.5 MPa                 |
| Coolant inventory    |        |                         |
| per loop             |        | 3.5 · 10 <sup>6</sup> g |
  
6. Tritium
 

|                   |                                         |
|-------------------|-----------------------------------------|
| Breeding material | nat. lithium (92.5 % Li7;<br>7.5 % Li6) |
| Mass flow         | 6.0 kg/s                                |
| Temperatures      |                                         |
| Lower boundary    | 300 °C                                  |
| Upper boundary    | 500 °C                                  |
| Average value     | 450 °C                                  |

|                                       |                                             |
|---------------------------------------|---------------------------------------------|
| Breeding material in the blanket      | 1.4 . 10 <sup>6</sup> g                     |
| Breeding factor                       | 1.32                                        |
| Production rate                       | 1.8 . 10 <sup>18</sup> atoms/s<br>= 282 g/a |
| Tritium concentration in the lithium  | 4.75 ppm                                    |
| Tritium inventory in the primary loop | approx. 16.6 g                              |
| Extraction process                    | Purification loop with cold traps at 185 °C |

#### 7. Costs (Base 1981)

|                      |                                                |
|----------------------|------------------------------------------------|
| Material             | Austenitic steel 1.4970                        |
| Blanket tubes        | approx. 170,-- DM/m                            |
| Header               | approx. 25,-- DM/kg<br>= 3000,-- DM/header     |
| Primary coolant pipe | approx. 350,-- DM/m                            |
| Casing walls         | approx. 30,-- DM/kg                            |
| Reflector material   | Austenitic steel 1.4970<br>approx. 30,-- DM/kg |

### III Reflector

|                             |                                              |
|-----------------------------|----------------------------------------------|
| Design                      | see Chapter VII.3.2.2                        |
| Material                    | Austenitic steel 1.4970                      |
| Coolant                     | H <sub>2</sub> O                             |
| Reflector power             | 0.12 MW                                      |
| Average power density       | 0.12 W/cm <sup>3</sup>                       |
| Reflector temperature       | max. 200 °C                                  |
| Coolant mass flow           | approx. 1 kg/s                               |
| Coolant pressure            | 0.5 MPa                                      |
| Coolant temperature         | Inlet approx. 70 °C<br>Outlet approx. 100 °C |
| Average reflector thickness | 350 mm                                       |
| Weight                      | approx. 12 t                                 |

#### IV Shield

Design

see Chapter VII.3.2.2

Material

Austenitic steel 1.4970

Coolant

Pb  
B<sub>4</sub>C  
H<sub>2</sub>O

Temperatures:

Structure

max. 200 °C

Coolant

Inlet

approx. 70 °C

Outlet

approx. 100 °C

Coolant pressure

0.5 MPa

Weight

approx. 20 t

#### V Radiation exposure

##### 1. During operation

Dose rate at a distance of 1 m

due to neutrons  $4 \cdot 10^3$  rem/h

due to secondary

gammas 1 rem/h

##### 2. Dose rate from structure activation of reflector and shield

Operating period

2.5 years

Load factor

0.8

Dose rate radial

at the surface 11.6 mrem/h

at a distance of 1 m 5.6 mrem/h

Isotopes considered

Co 60

Ta 182

Fe 59

Nuclide concentrations

500 ppm Co 59

500 ppm Ta 181

3100 ppm Fe 58

##### 3. Radiation damage

First wall  
(at 1 MW/m<sup>2</sup>)

8.61 dpa/a



Table II: Physical Data of Lithium from Ref. 14

|                                                |                                                                                                    |
|------------------------------------------------|----------------------------------------------------------------------------------------------------|
| Composition of nat. Lithium                    | 92.5 % Li7; 7.5 % Li6                                                                              |
| Melting Point                                  | 180.6 °C                                                                                           |
| Boiling Point                                  | 1342 °C                                                                                            |
| Density / g/cm <sup>3</sup> /                  | $\rho = 0.515 - 1.01 \cdot 10^{-4} (t-200)$<br>(t / °C / = temperature)                            |
| Heat Capacity / J/g . °C /                     | $c_p = 4.53 - 8.37 \cdot 10^{-4} t$<br>t < 422 °C<br>$= 4.21 - 7.24 \cdot 10^{-5} t$<br>t > 422 °C |
| Thermal Conductivity<br>/ J/s . m . °C /       | $\lambda = 42.3 + 12.3 \cdot 10^{-3} t$                                                            |
| Electrical Resistivity<br>/ $\mu\Omega$ . cm / | $R = 18.33 + 3.339 \cdot 10^{-2} t$<br>$- 6.795 \cdot 10^{-6} t^2$                                 |

## VIII End Plug Design

### VIII.1 General Features

The end plug region consists of the barrier, "c", and yin-yang magnets, their support structure and dewar, shielding penetration for neutral beam injection and ECRH heating, neutral beam dumps and the electron and ion thermal dumps at the end walls. There is also provision for testing of direct convertor modules. The details of the magnet design are given in Chapter V. Three-dimensional Monte Carlo calculations of neutron streaming through the various penetrations of the end plug shield are presented in Section VIII.2. The beam dumps for the various neutral beams are discussed in Section VIII.3. A conceptual design of a single stage plasma direct convertor module is given in Section VIII.4. The thermal hydraulics aspects of the thermal dumps is treated briefly in Section VIII.5. The surface erosion aspects of the direct convertor and thermal dumps is discussed in Section VIII.6. The overall arrangement of the various components is treated in Section VIII.7. Assembly and maintenance considerations are discussed in Chapter X.

## VIII.2 Neutron Environment

A primary objective of this work is to determine the neutron and gamma-ray streaming into the NBI and ECRH ducts. This will be of great importance in evaluating the radiation effects in these penetrations.

Evaluation of the radiation flow into the ducts requires an accurate knowledge of the neutron source, and the geometrical configuration of the shield, as well as the size of the ducts.

The neutron source distribution in the reactor is shown in Fig. VIII.2-1. The D-T reaction rate drops to a negligible value at the center of the end-plug region. The distribution is characterized by two relative peaks in the barrier and plug regions. The first peak occurs in the barrier region at a distance of 16.5 m from the center of the reactor, and the second peak is in the end-plug region at a distance of 25 m. Fortunately, the NBI and ECRH ducts are not in direct line of sight of the source neutrons at these peaks.

In the shield design of the barrier and end-plug regions, the shield wall follows the plasma envelope. The vertical and horizontal plasma envelopes are shown in Fig. VIII.2-2.

The distance between the plasma and the wall differs from one region to another according to the magnetic field strengths, and can easily be calculated as explained below.

The total magnetic flux is, in general, given by

$$F = \pi r^2(z) B(z) \sqrt{1-\beta(z)} + \pi [r_w^2(z) - r^2(z)] B(z) \quad (\text{VIII.2-1})$$

where  $r$  is the plasma radius,  $B$  is the vacuum magnetic field,  $r_w$  is the wall radius, and  $\beta$  is the ratio between the plasma and magnetic field pressures.

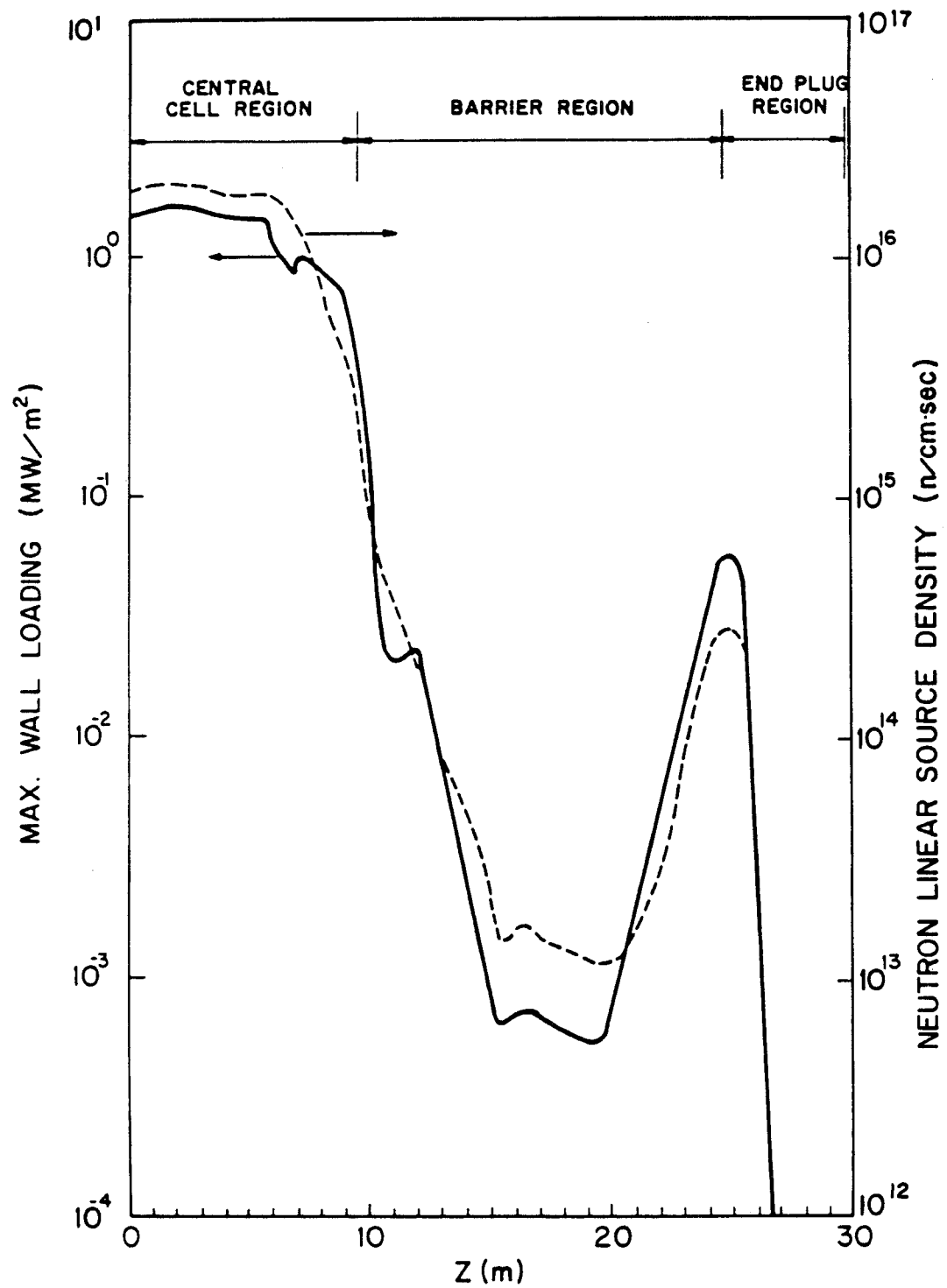


Fig. VIII.2-1 Variation of the neutron linear source density and maximum neutron wall loading along the z axis.

FRAME 8, GFUN LOSN1 DATE 10/29/81 TIME 23:59:08 RUN FOR BLANKET (HONG.)

34 CONDUCTOR ELEMENTS **0** IRON ELEMENTS  
Z-Y PLANE

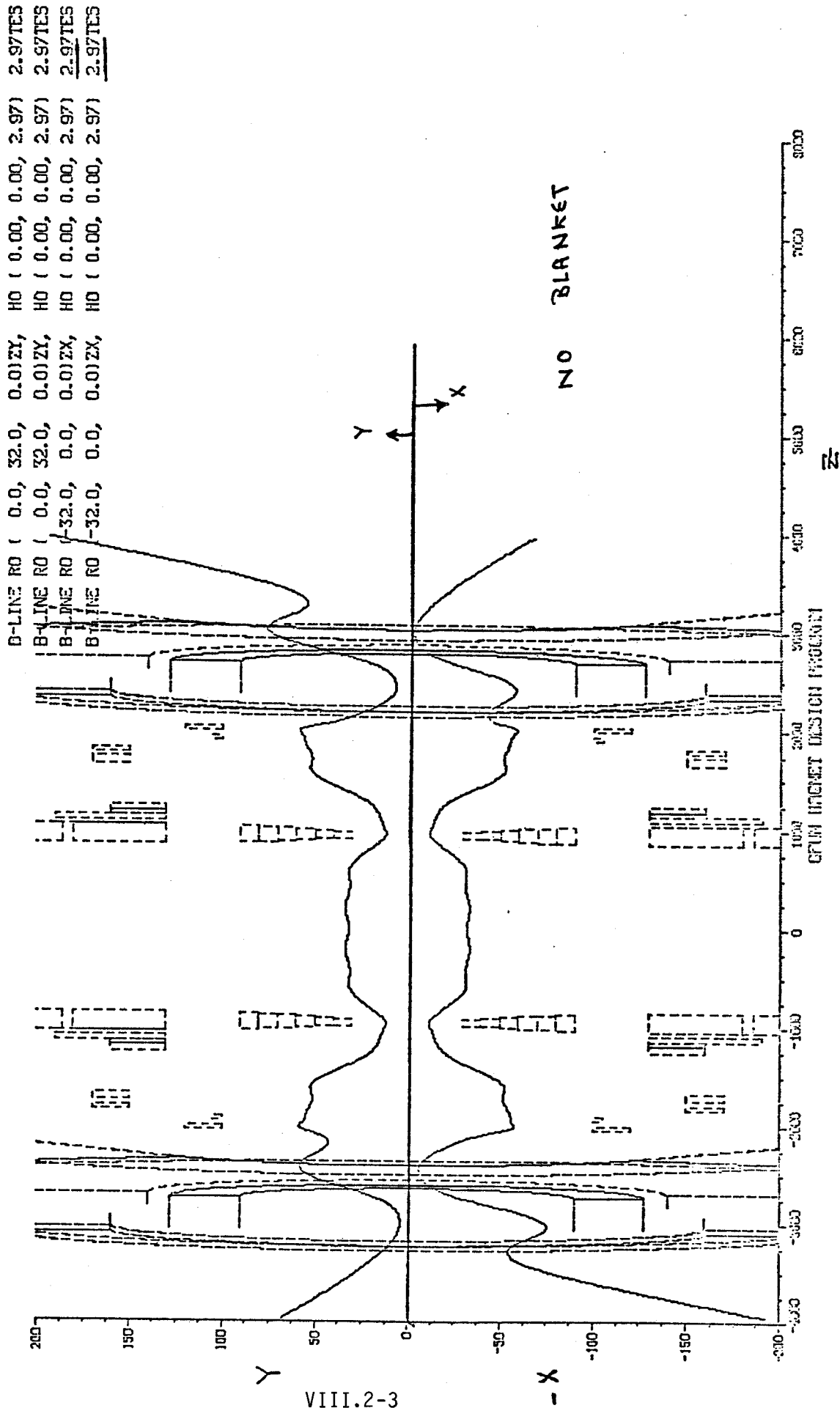


Fig. VIII.2-2 Vertical and horizontal plasma envelopes along the z axis.

The first term on the right-hand side is the contribution from the plasma, and the second is from the halo region.

By flux tube conservation, the plasma radius for a circular flux tube is given by

$$r(z) = r_c \left( \frac{B_c \sqrt{1-\beta_c}}{B(z) \sqrt{1-\beta(z)}} \right)^{1/2} \quad (\text{VIII.2-2})$$

where  $r_c$  is the central cell (cc) plasma radius,  $B_c = 2.7$  T is the cc magnetic field, and  $\beta_c = 0.5$  is the cc beta.

From Eqs. VIII.2-1 and VIII.2-2 the distance between the plasma and the wall ( $r_w - r(z)$ ) can be calculated at any point along the z-axis. The distance has an average value of 10 cm in the end-plug region and a maximum value of 30 cm in the barrier region.

The shield thickness in the barrier and end-plug regions has been determined from one-dimensional calculations. In these calculations, the peak source density in the appropriate region was used. In order to reduce the radiation damage in the field shaping, transition, and yin-yang coils to an acceptable level, the shield has been designed to have the thicknesses of 20-30 and 60 cm in the barrier and end-plug regions, respectively. Also, a 20 cm thick shield is used in the direct convertor region.

To deal with the three-dimensional geometrical configuration of the shield, the Monte Carlo code MCNP<sup>(1)</sup> was used for the treatment of the problem. Figures VIII.2-3 and VIII.2-4 are elevation and plan views, respectively, of the MCNP model from the midplane of the barrier to the end of the reactor. The model contains the NBI and ECRH ducts and the figure illustrates the geometrical relationship of these ducts to the reactor.

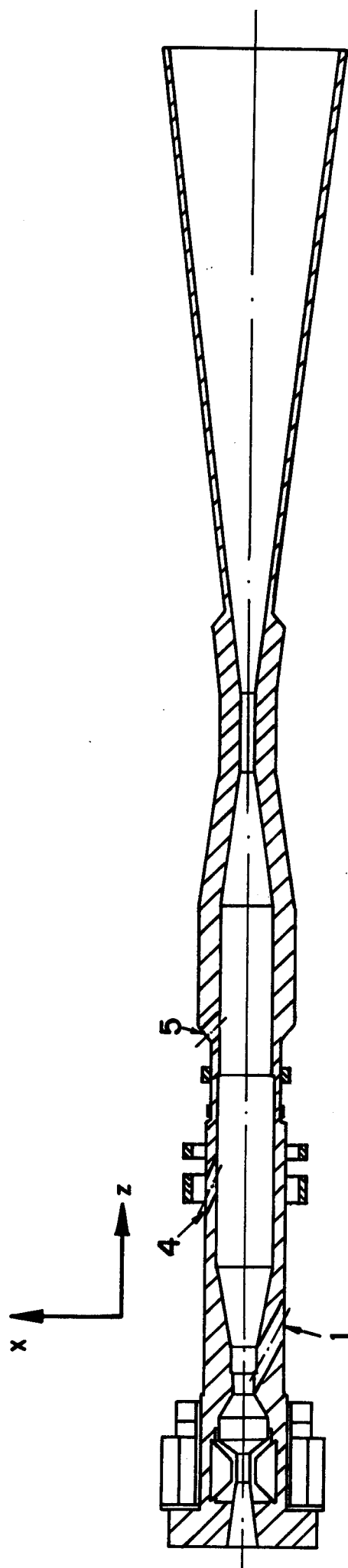


Fig. VIII.2-3

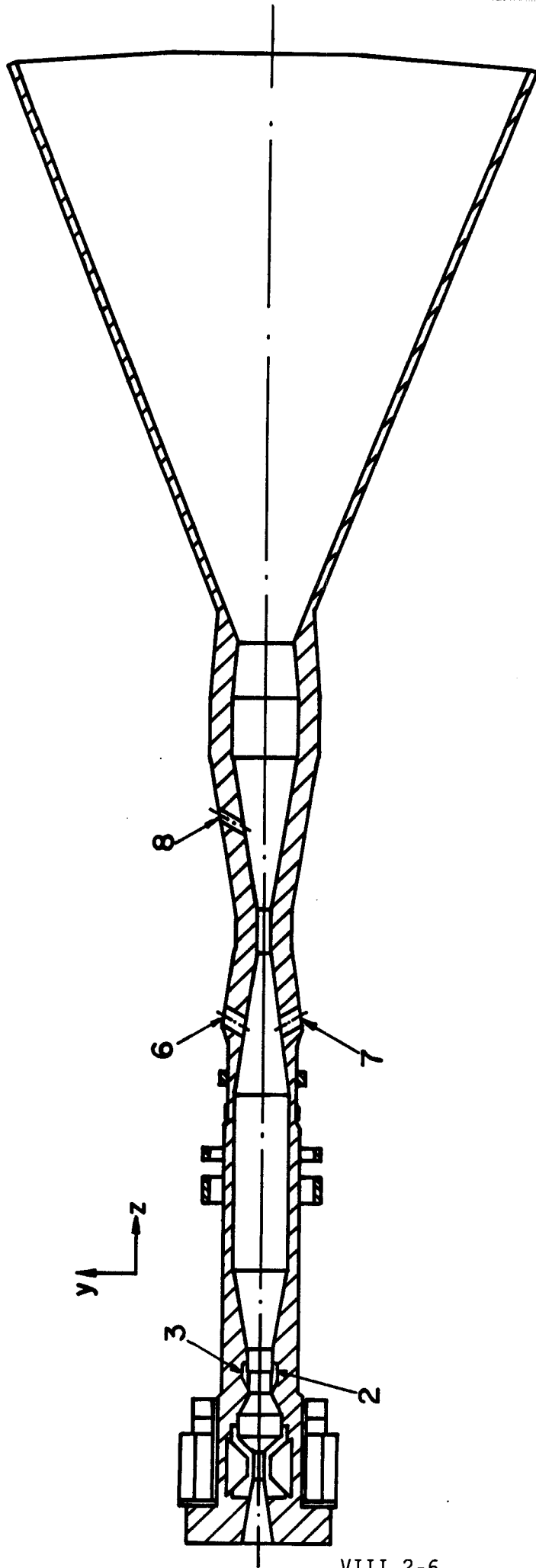


Fig. VIII.2-4



Details of the geometrical model in the vertical plane are given in Figs. VIII.2-5, VIII.2-6, VIII.2-7, VIII.2-8 for the axial zones 9.5-20.30, 20.30-31.10, 31.10-41.90, 41.90-50 m, respectively. The geometrical model details in the horizontal plane are given in Figs. VIII.2-9 - VIII.2-11, for the axial zones 9.5-20.30, 20.30-31.10, 31.10-38.30 m, respectively. Figure VIII.2-12 shows, on a different scale, the axial zone 34.4-50 m of the horizontal plane. This model will be used to evaluate radiation effects in these regions.

The model shown on these figures is entered into the MCNP input. Trapping surfaces are located at the entrance surfaces to eight ducts shown on Figs. VIII.2-3 and VIII.2-4. At these surfaces all particles entering the ducts are counted according to angle and energy bins. The results are stored to serve as source distributions in later modeling of the ducts themselves. These results were obtained by sampling the source distribution separately in the central cell and barrier and end plug regions. The source strengths are given in Table VIII.2-1. As the source in the barrier and plug are dominant, a run was made using 20,000 source particle histories from that part of the source distribution between 9.2 m and 14.5 m from the reactor midplane. A second run of 20,000 histories are sampled from axial position 14.5 m to 26.7 m. The data from all runs are combined according to a total source of  $1.47 \text{ E}19$  neutrons/sec to axial position 9.2 m and  $2.71 \text{ E}17$  and  $8.33 \text{ E}16$  neutrons/sec in the two regions described above.

The number of particles crossing each trapping surface is given in a per fusion and on a per second basis along with the energy per fusion in Table VIII.2-2.

The tabulated results can be interpreted qualitatively by observing the energy current density in  $\text{MW/m}^2$  in the right hand column. This corresponds to

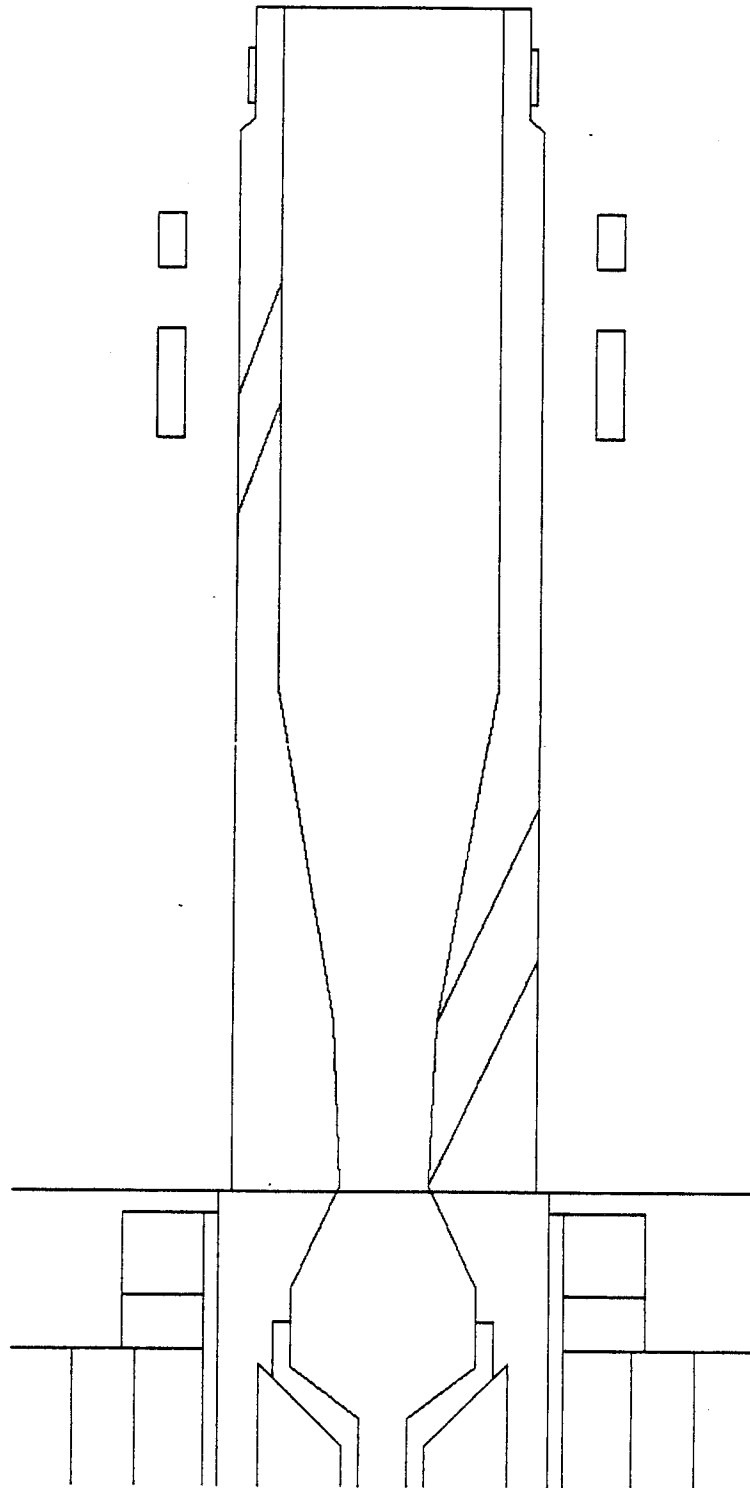


Fig. VIII.2-5 Vertical plane,  $z = 9.5 - 20.3$  m.

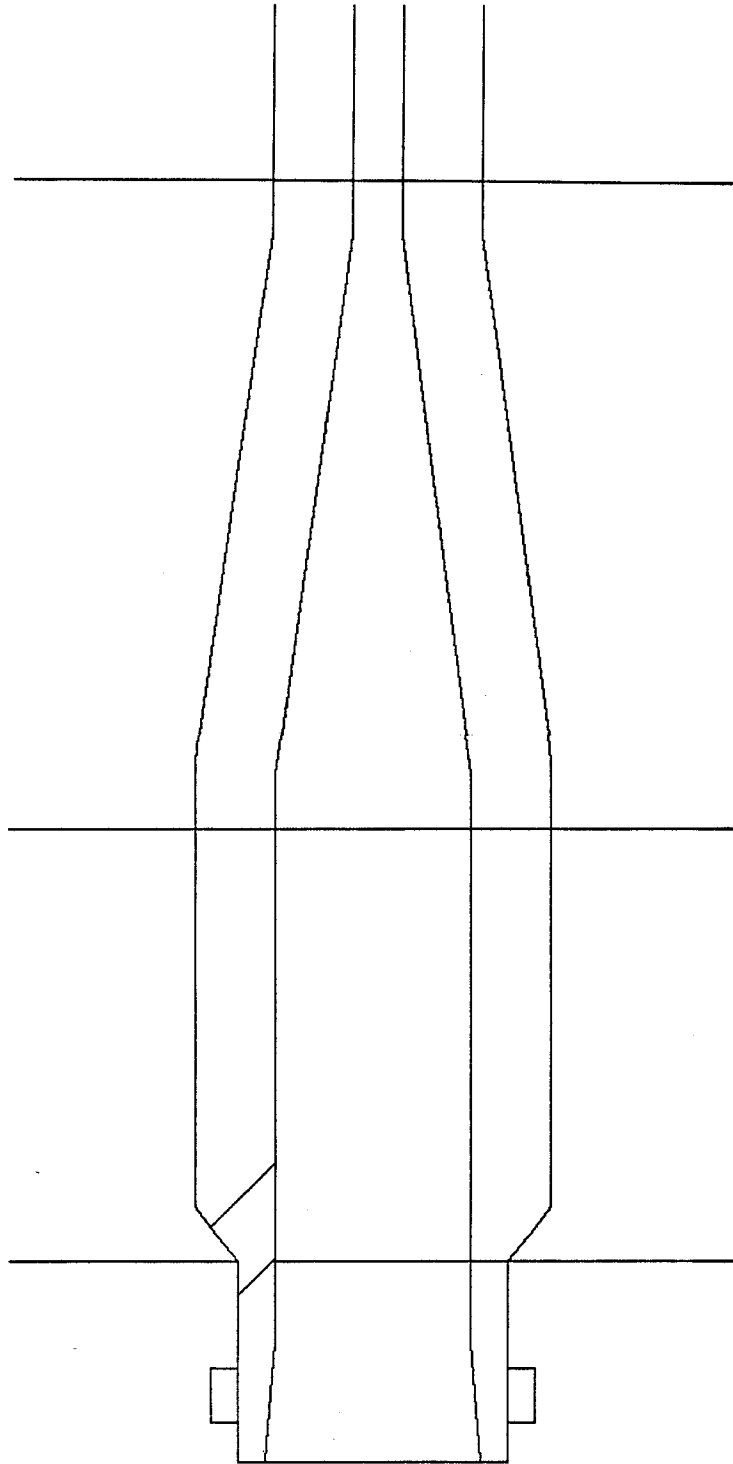


Fig. VIII.2-6 Vertical plane,  $z = 20.3 - 31.1$  m.

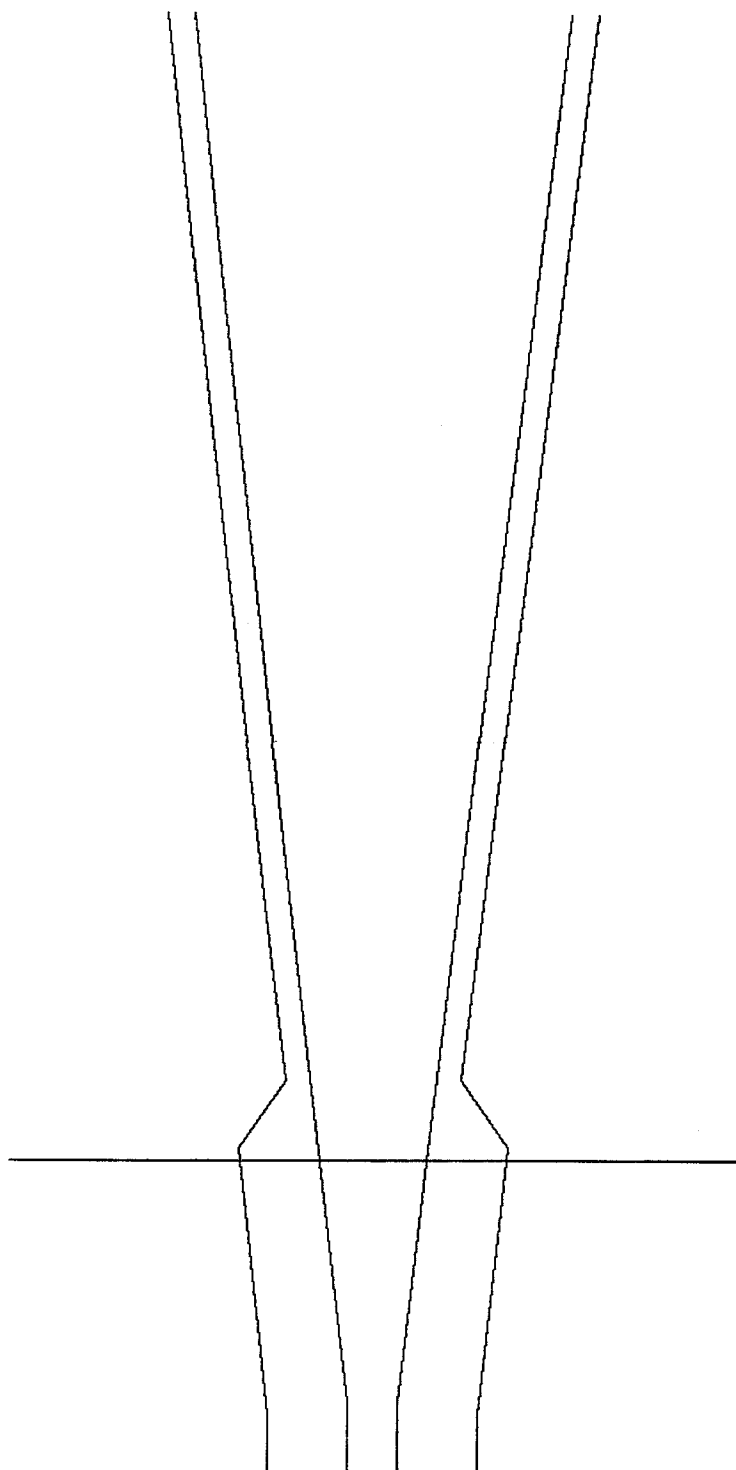


Fig. VIII.2-7 Vertical plane,  $z = 31.1 - 41.9$  m.

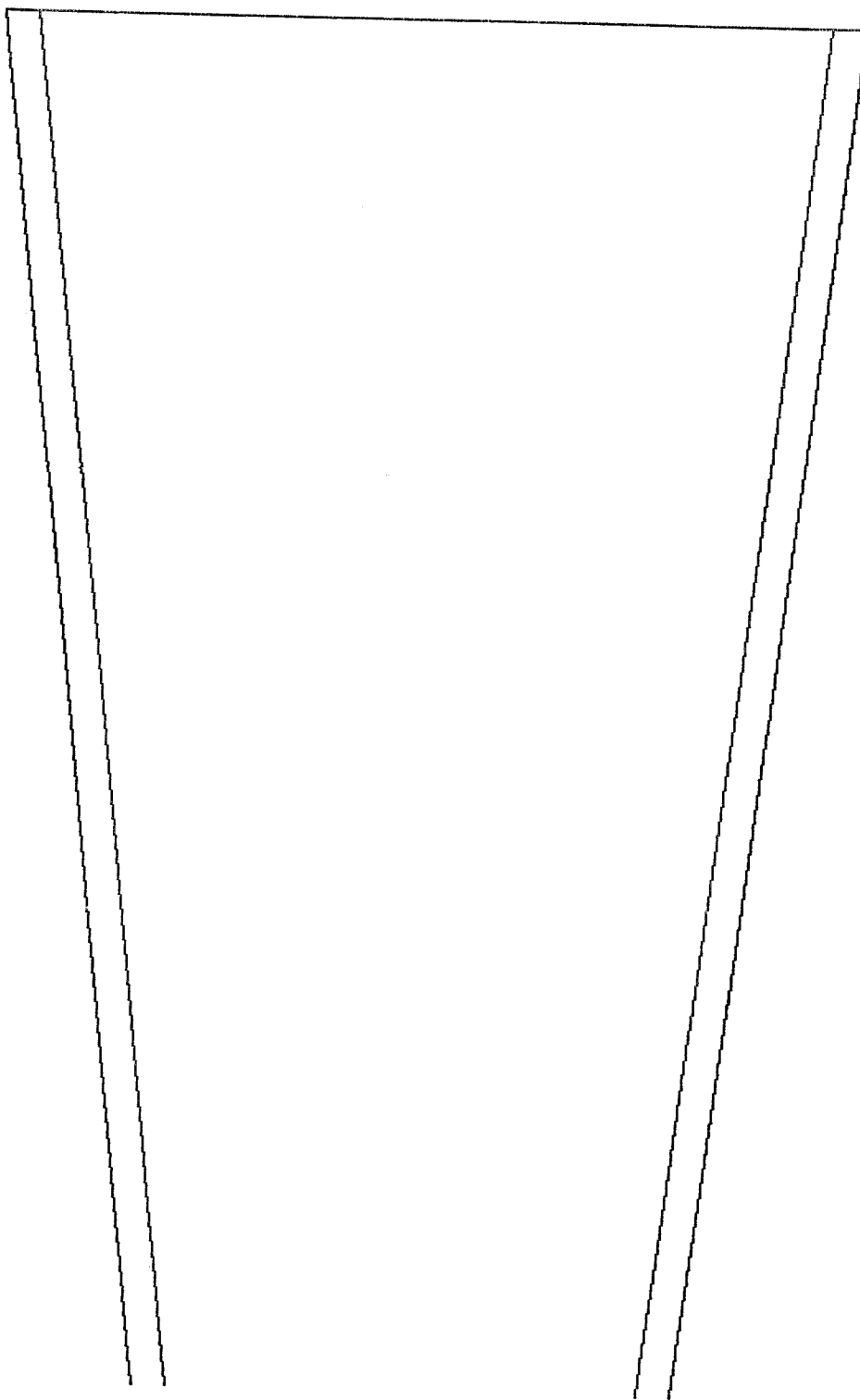


Fig. VIII.2-8 Vertical plane,  $z = 41.9 - 50$  m.

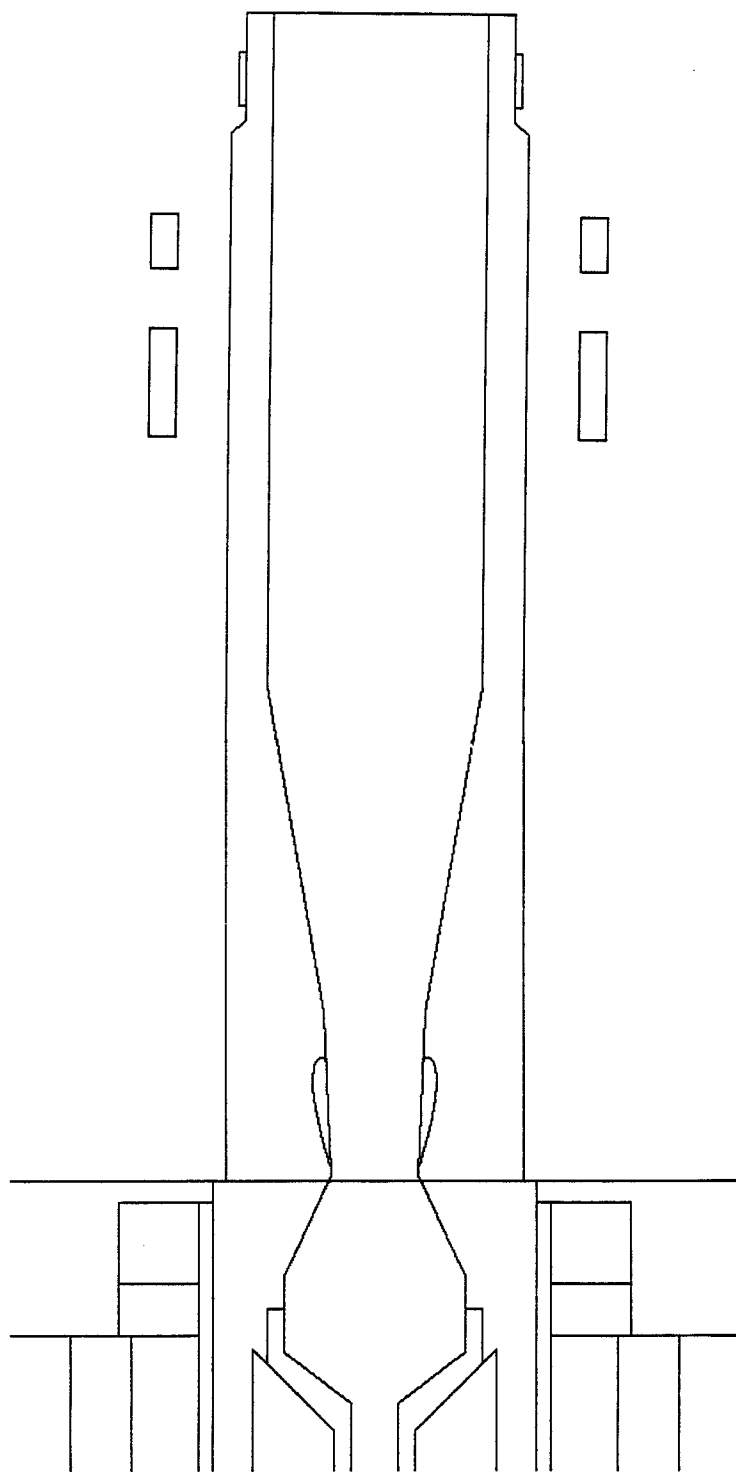


Fig. VIII.2-9 Horizontal plane,  $z = 9.5 - 20.3$  m.

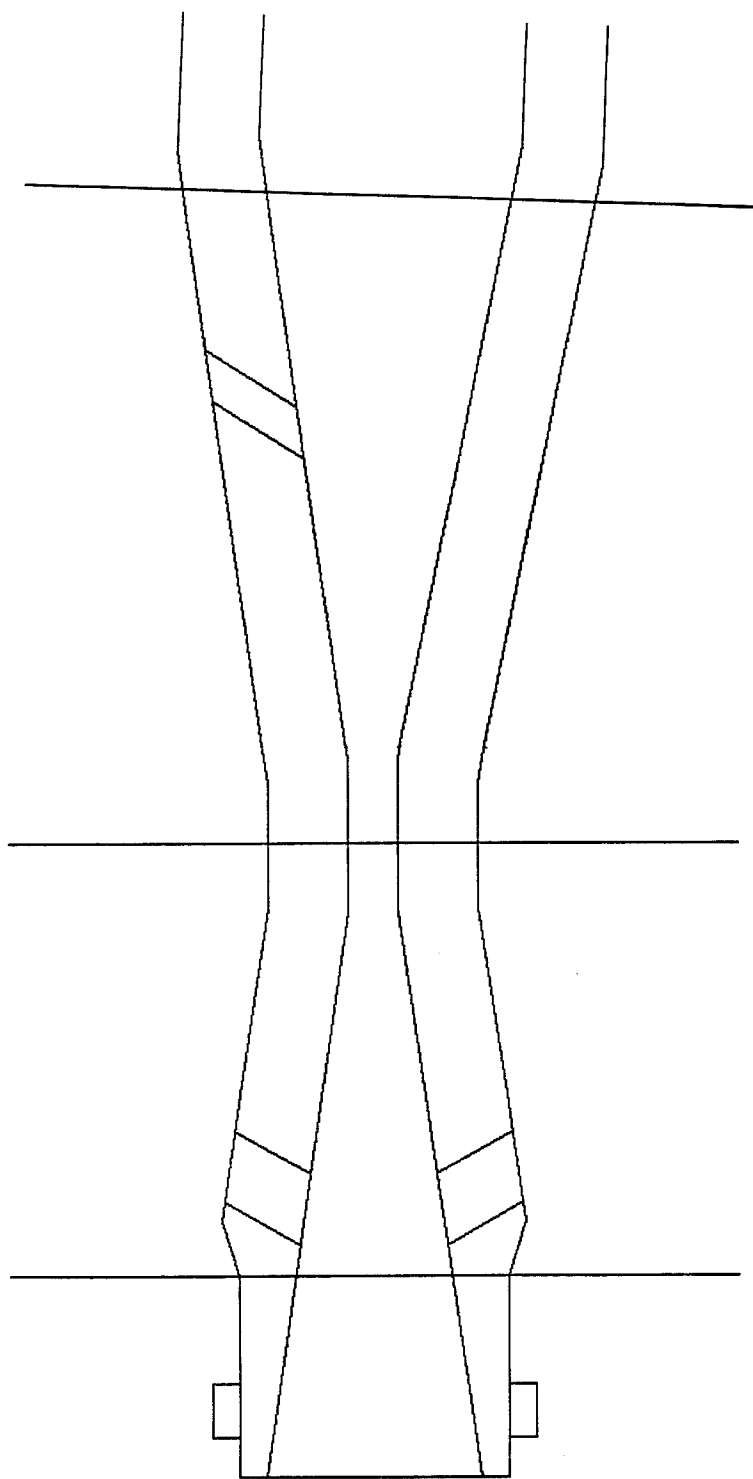


Fig. VIII.2-10 Horizontal plane,  $z = 20.3 - 31.1$  m.

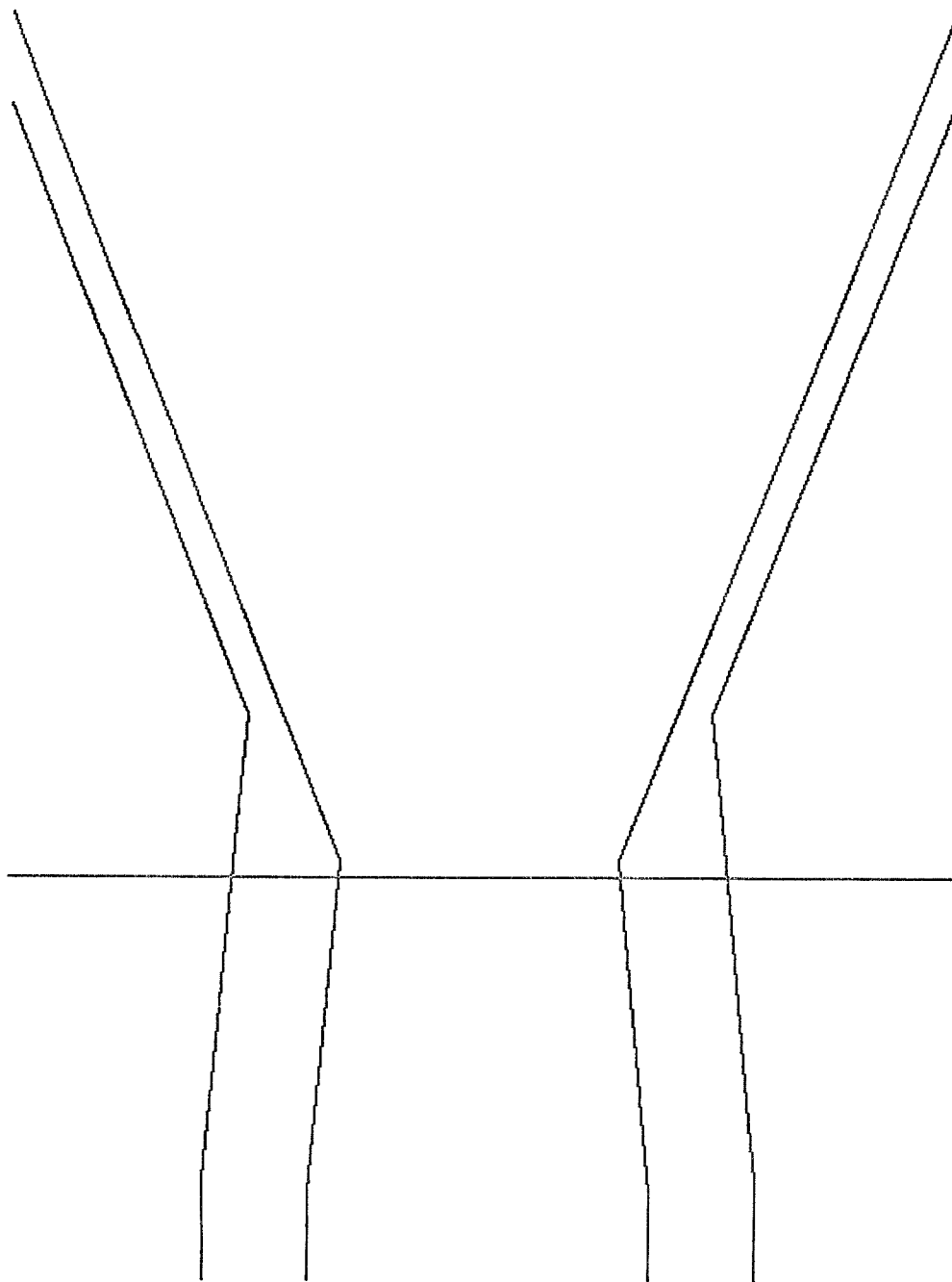


Fig. VIII.2-11 Horizontal plane,  $z = 31.1 - 38.3$  m.



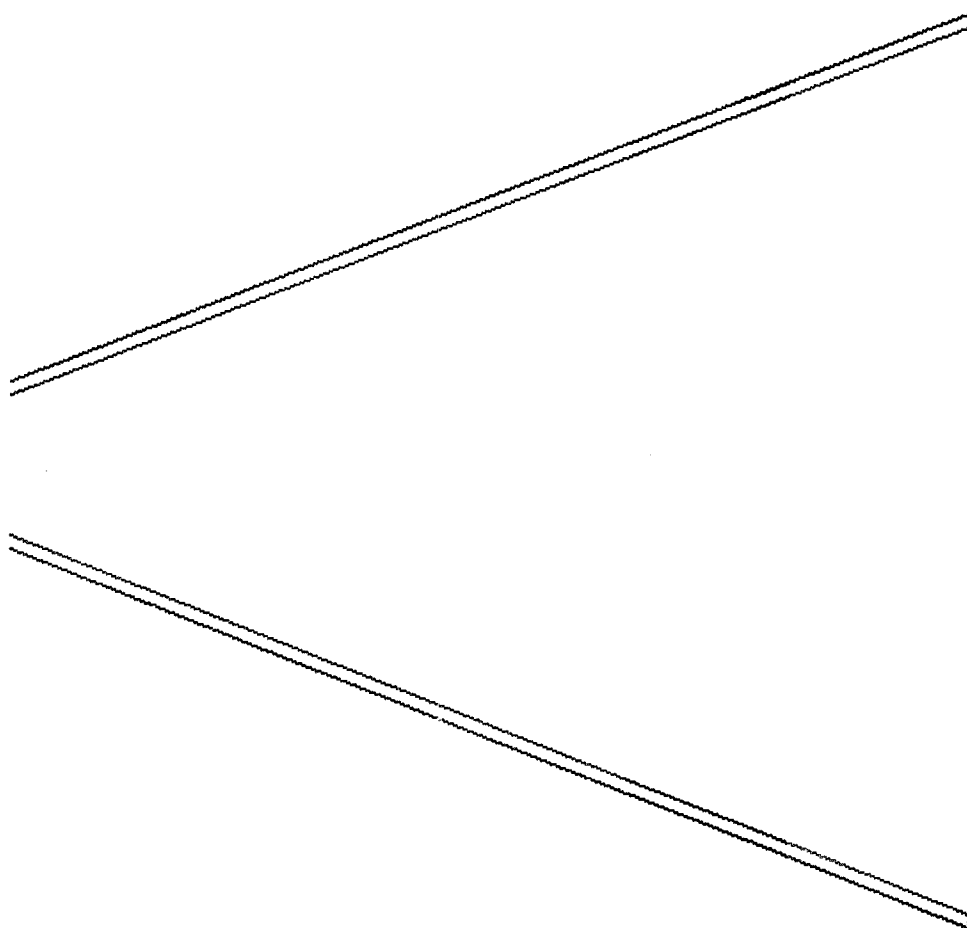


Fig. VIII.2-12 Horizontal plane,  $z = 34.4 - 50$  m.

Table VIII.2-1

| Region   | Source (n/sec) | Power (MW) |
|----------|----------------|------------|
| C.C.     | 2.94 E19       | 82.914     |
| B&EP (2) | 7.08 E17       | 1.998      |
| Total    | 3.01 E19       | 84.912     |

Table VIII.2-2

| Duct<br>Number         | Neutrons   |          | Gammas     |          | Energy     |        | Energy<br>Current<br>(MW/m <sup>2</sup> ) |
|------------------------|------------|----------|------------|----------|------------|--------|-------------------------------------------|
|                        | Per Fusion | Per Sec  | Per Fusion | Per Sec  | keV/Fusion | kW     |                                           |
| 1                      | 2.96 E-4   | 8.90 E15 | 4.79 E-5   | 1.44 E15 | 2.101      | 10.136 | 2.236 E-2                                 |
| 2                      | 2.77 E-4   | 8.39 E15 | 4.66 E-5   | 1.40 E15 | 1.833      | 8.842  | 1.950 E-2                                 |
| 3                      | 2.75 E-4   | 8.27 E15 | 4.40 E-5   | 1.32 E15 | 2.029      | 9.788  | 2.159 E-2                                 |
| 4                      | 5.48 E-6   | 1.65 E14 | 8.80 E-7   | 2.65 E13 | 0.042      | 0.202  | 9.779 E-4                                 |
| 5                      | 2.80 E-5   | 8.43 E14 | 5.32 E-6   | 1.60 E14 | 0.190      | 0.917  | 3.302 E-3                                 |
| 6                      | 1.94 E-5   | 5.84 E14 | 4.13 E-6   | 1.24 E14 | 0.122      | 0.587  | 2.587 E-3                                 |
| 7                      | 2.15 E-5   | 6.47 E14 | 4.68 E-6   | 1.41 E14 | 0.162      | 0.781  | 3.444 E-3                                 |
| 8                      | 1.89 E-6   | 5.69 E13 | 4.54 E-7   | 1.37 E13 | 0.005      | 0.025  | 2.677 E-4                                 |
| Total into<br>16 Ducts | 1.85 E-3   | 5.57 E15 | 3.08 E-4   | 9.25 E15 | 12.968     | 62.556 |                                           |

$\sim .02 \text{ MW/m}^2$  or less compared to  $\sim 2 \text{ MW/m}^2$  for the first wall of a fusion chamber or about 2 orders of magnitude less at the entrance to our penetrations. Further, the energy is less directed here; so the actual source to the back of the ducts will probably be at least another order of magnitude less than in similar fusion chamber penetrations.

The neutron and gamma energy spectra and singular distributions are obtained and since duct 1 is the most severe case and duct 6 is representative of the other ducts their spectra are shown in Figs. VIII.2-13 to VIII.2-18. An inspection of Fig. VIII.2-15 shows the angular distribution of the source particles peaks at normal incidence while the center line is appreciably off normal and most particles will go into the shield of the duct near the first wall. This is also true of duct 6.

Duct 1 is the medium energy neutral beam injector and is modeled as shown in Fig. VIII.2-19. The radiation problem of greatest concern here is the effect on the insulating electrode support structure which is a cylindrical shell, designated 1 on the figure, of the ceramic material MACOR. The composition of MACOR and other materials in the model are given in Table VIII.2-3. The model is an idealization and adaptation of a neutral beam system design provided us by John Osher.

The neutral beam duct model problem employs neutron and gamma surface sources at the duct opening of  $8.9 \times 10^{15}$  and  $1.44 \times 10^5$  particles/second respectively, as obtained from the previous model. Only the central ion source was considered as radiation damage effects in adjacent ion sources are expected to be less severe. An MCNP run of 50,000 histories was carried out resulting in relative standard deviations of less than 10% for the quantities of interest. The energy deposition and dose rate in the insulating electrode

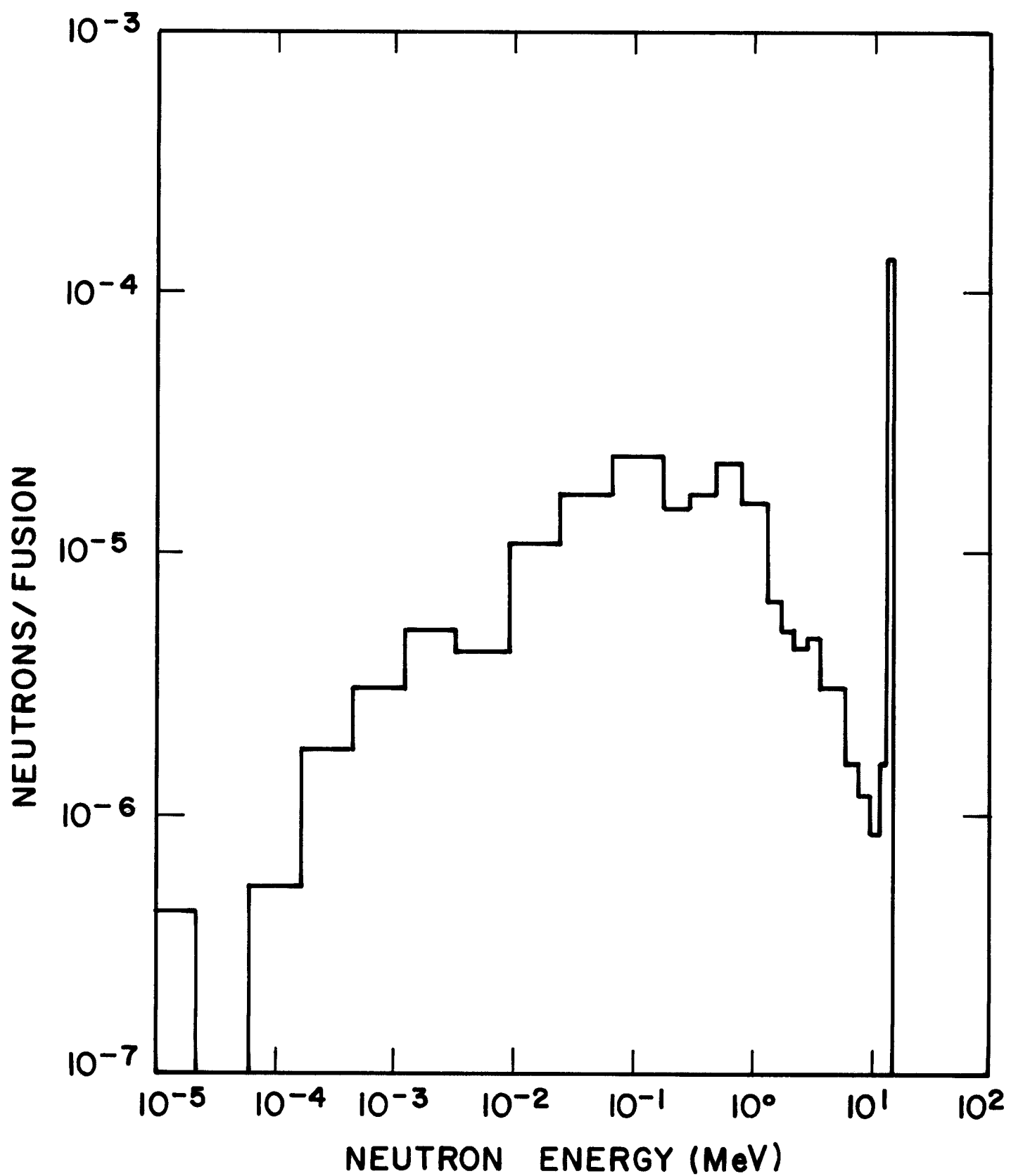


Fig. VIII.2-13. Neutron energy spectrum entering Duct 1.

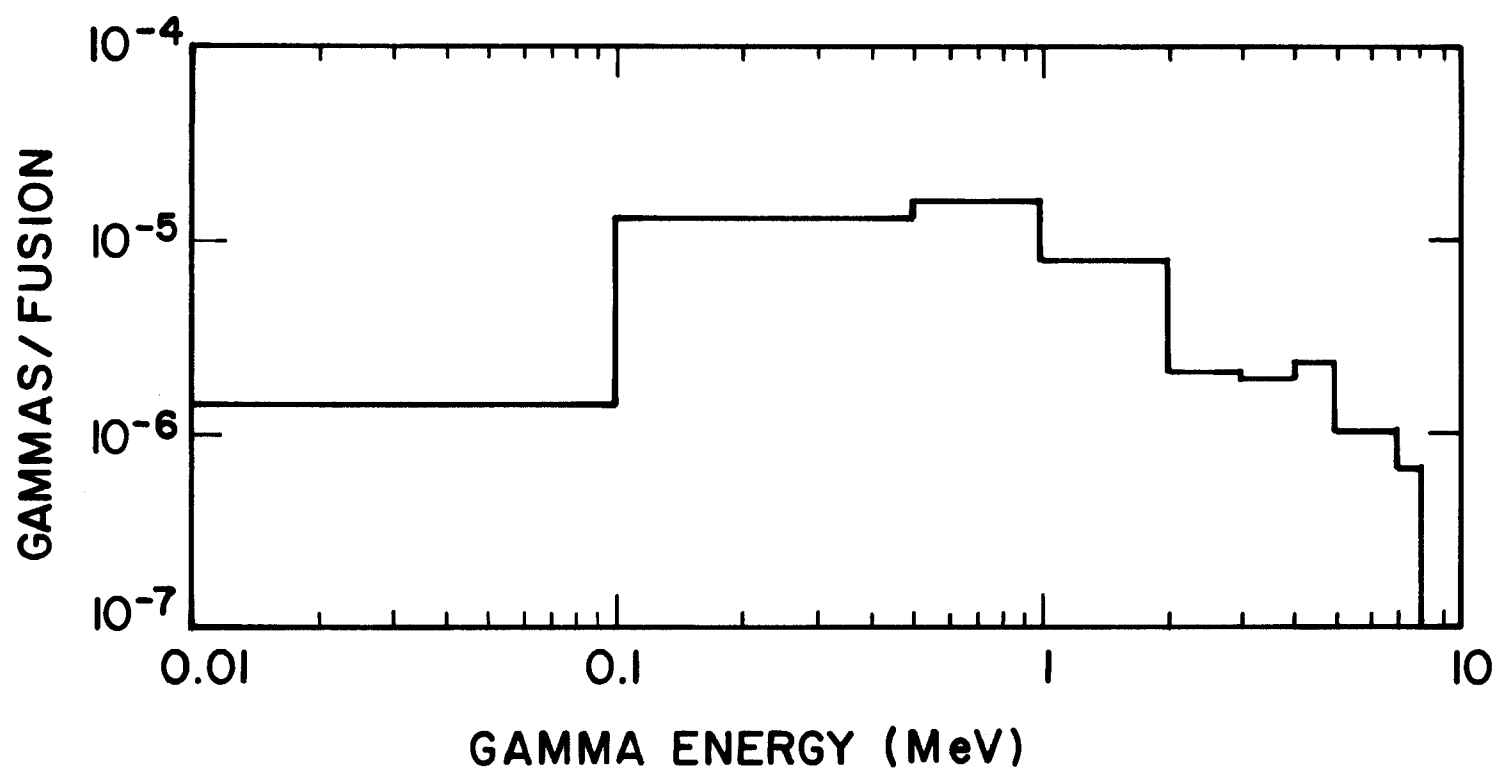


Fig. VIII.2-14. Gamma energy spectrum entering Duct 1.

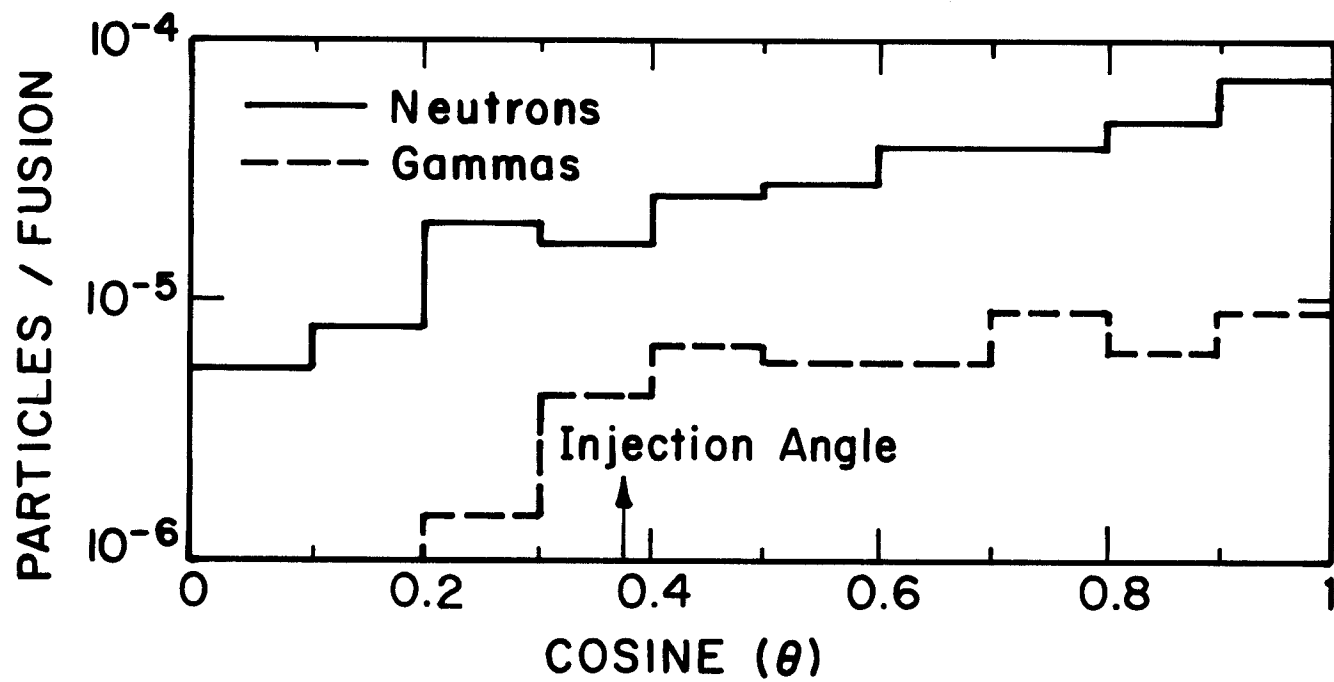


Fig. VIII.2-15. Angular distributions of neutrons and gammas entering Duct 1.

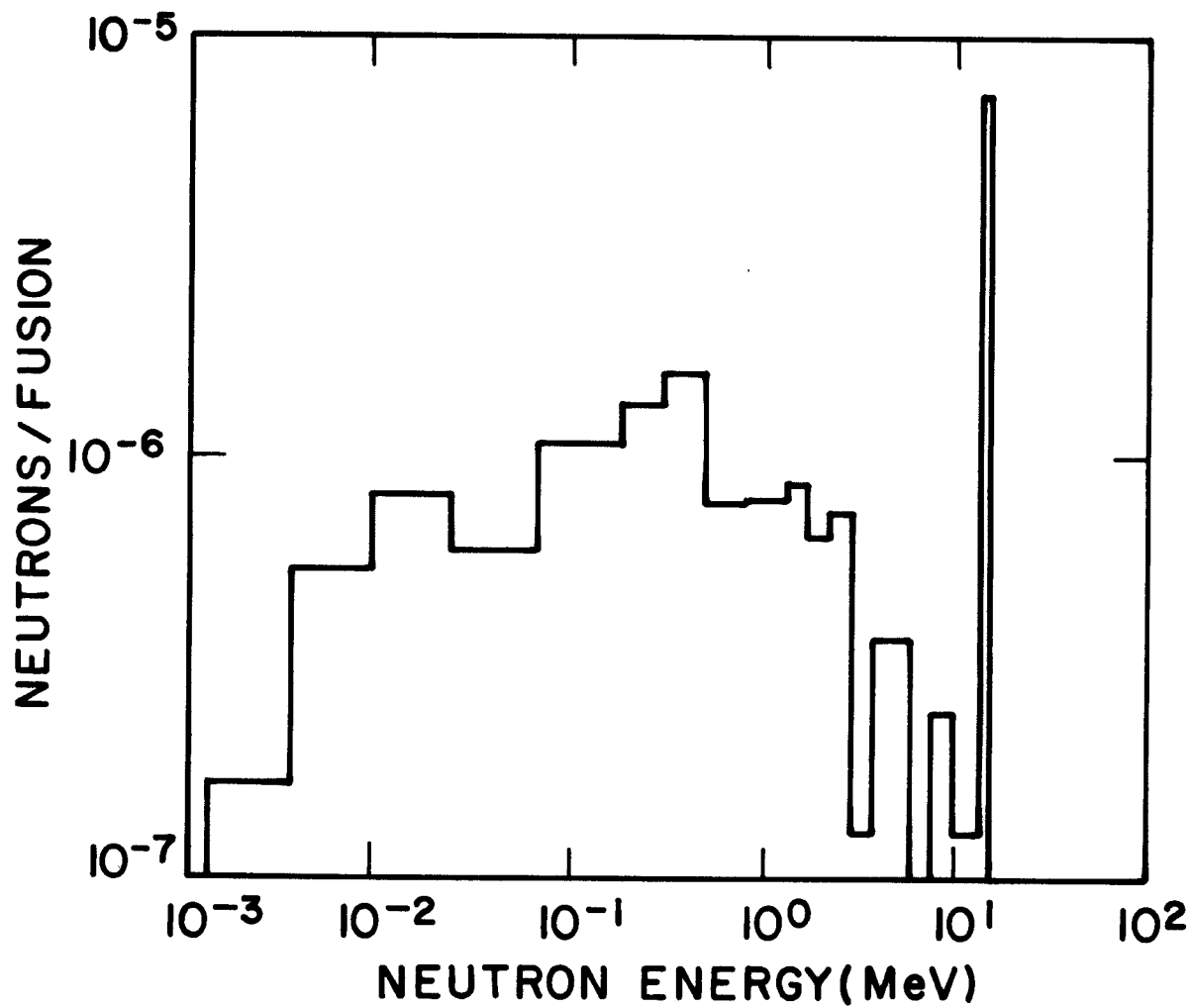


Fig. VIII.2-16. Neutron energy spectrum entering Duct 6.



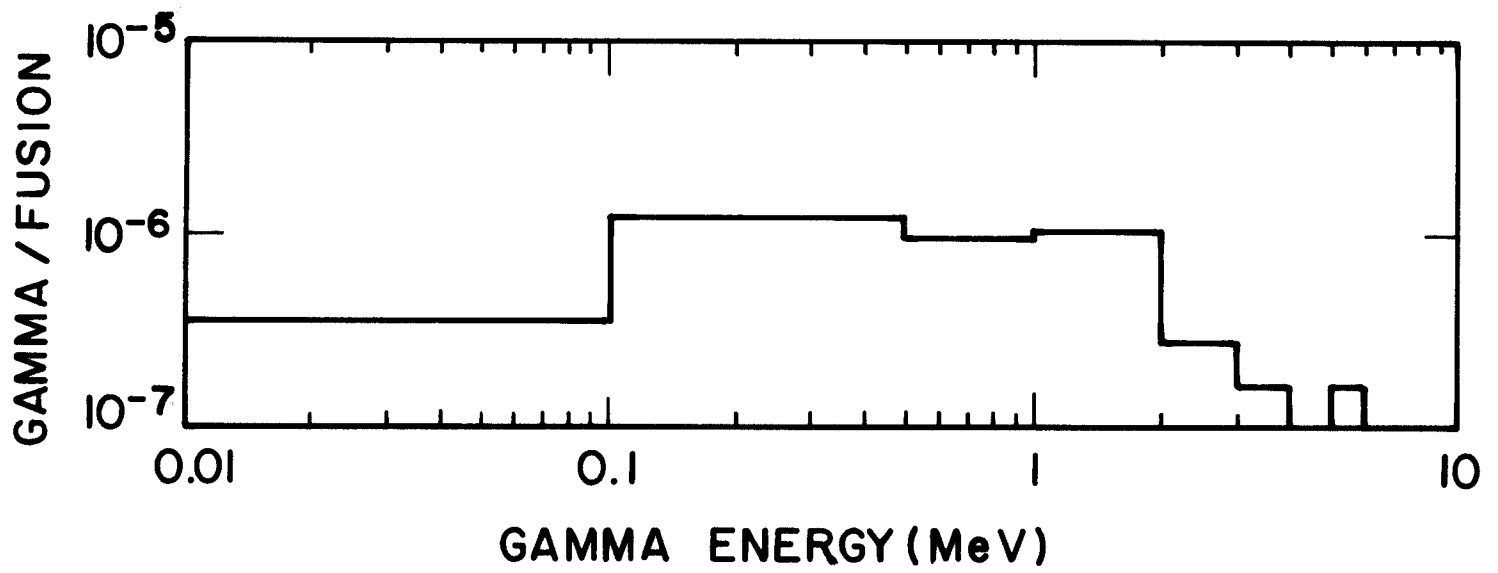


Fig. VIII.2-17. Gamma energy spectrum entering Duct 6.

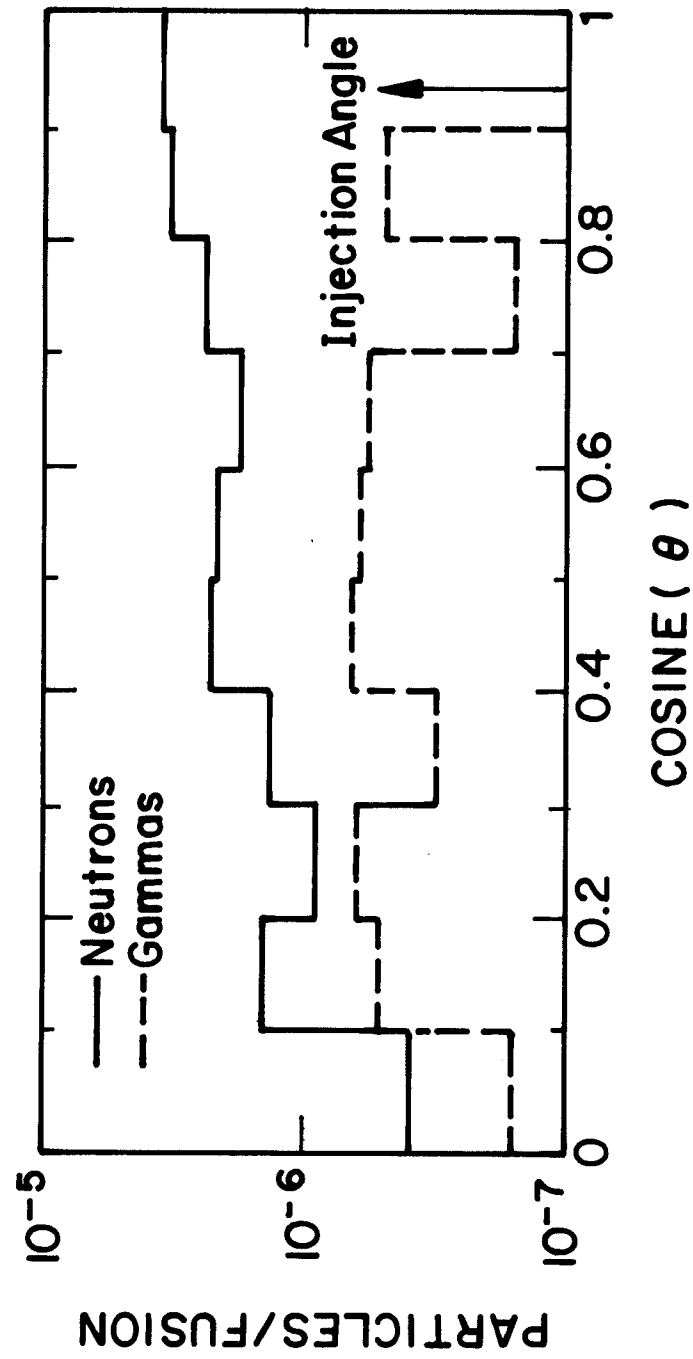
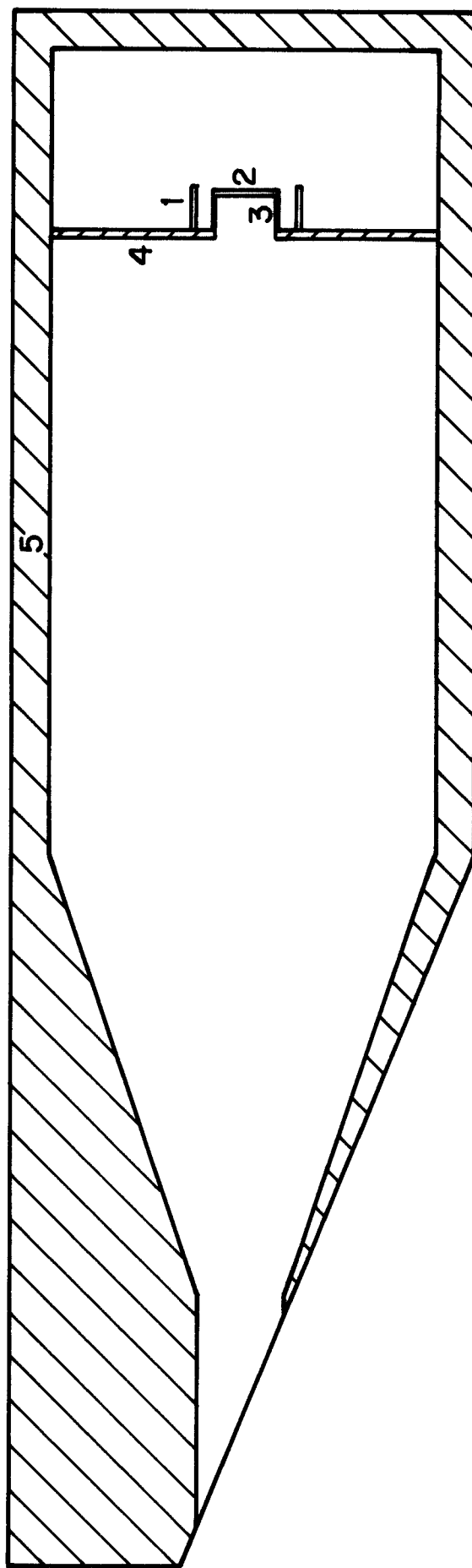


Fig. VIII.2-18. Angular distribution of neutrons and gammas entering Duct 6.



VIII.2-25

Fig. VIII.2-19

Table VIII.2-3. Composition of Region in Model of the NBI Duct 1

| Region | Thickness (cm) | Material | Composition                                                                                                                                  |
|--------|----------------|----------|----------------------------------------------------------------------------------------------------------------------------------------------|
| 1      | 1.5            | MACOR    | 46 w/o SiO <sub>2</sub><br>16 w/o Al <sub>2</sub> O <sub>3</sub><br>17 w/o MgO<br>10 w/o K<br>4 w/o F<br>7 w/o B <sub>2</sub> O <sub>3</sub> |
| 2      | 2              | Copper   | Cu                                                                                                                                           |
| 3      | 0.5            | Copper   | Cu                                                                                                                                           |
| 4      | 4              | SS-316   | 68 w/o Fe<br>17 w/o Cr<br>12 w/o Ni<br>3 w/o Mo                                                                                              |
| 5      | 20             | Shield   | 56 v/o Fe 1422<br>26 v/o B <sub>4</sub> C (87%)<br>14 v/o Pb<br>4 v/o H <sub>2</sub> O                                                       |

support structure due to gammas, neutrons, and totals are given in Table VIII.2-4. The flux (and heating rate) at the end of the insulator is higher at the end than elsewhere by the order of 50% due to direct streaming from the opening.

An experimental study of radiation effects in MACOR (Ref. VIII.2.2) indicated slight damage by 14 MeV neutrons at a fluence of  $10^{22}$  neutrons/m<sup>2</sup> which was estimated to correspond to a dose of  $6 \times 10^8$  Rad. Comparing this with the result of  $5 \times 10^7$  Rad per full power year from the table, it would appear that the insulators would last 10 FPY's as far as radiation is concerned. Note however that a fusion chamber source strength two orders of magnitude higher would indicate possible problems after about one month.

This points up the need for more definitive damage data in insulators. The data used were post-irradiation and give no indication of performance under irradiation where electrical conductivity and electric breakdown phenomena may be important. Thus our results, while favorable regarding insulator performance, must be used with some caution.

#### Reference for Section VIII.2

1. "MCNP", Los Alamos National Laboratory Report LA-7396-M, April 1981.

Table VIII.2-4. Energy Deposition and Deposition Rate in the MACOR  
Insulating Electrical Support Structure in NBI Duct 1

|                                                       | Neutron   | Gamma     | Total     |
|-------------------------------------------------------|-----------|-----------|-----------|
| Energy deposition per 14.1 MeV<br>neutron (eV/fusion) | 2.176 E-2 | 1.152 E-2 | 3.328 E-2 |
| Nuclear heating rate (w/cm <sup>3</sup> )             | 4.051 E-5 | 2.144 E-5 | 6.195 E-5 |
| Dose rate (rad/sec)                                   | 1.608     | 0.851     | 2.459     |
| Dose Rate (rad/FPY)                                   | 5.070 E7  | 2.683 E7  | 7.753 E7  |

### VIII.3 Beam Dumps

#### VIII.3.1 Introduction

In the process of neutral beam heating of the plasma, the trapping of the various incident neutral species is not, unfortunately, one hundred percent efficient. It is the purpose of the neutral beam dumps to provide an effective heat sink for these residual neutral beam fractions which are not absorbed by the plasma. It is interesting to note that, although the beam dumps comprise only a small part of the overall engineering design of TASKA, they nevertheless present some novel technology problems which warrant equally novel solutions.

The salient features of these technology problems may be summarized as follows:

##### 1. Power Absorption - Heat Transfer

The maximum power density of the TASKA neutral beams at the plasma footprint approaches a steady state value of  $\sim 30 \text{ kW cm}^{-2}$  for the case of the medium energy D/T beam into the barrier. This should be compared with power densities encountered in current state-of-the-art designs of dumps for high heat fluxes, where maximum power densities of only  $\sim 4 \text{ kW cm}^{-2}$  can be tolerated, and then only for short periods of  $\leq 0.5 \text{ s}$  [see, for example, Harbaugh,<sup>(1)</sup> Moir, et al.<sup>(2)</sup>]. It can be seen, therefore, that the power absorption - heat transfer aspects of the beam dump design require careful study in order to reduce the residual beam power densities to acceptable limits.

##### 2. Sputtering of the Dump Surfaces From Beam Impact

For the range of beam energies encountered from the neutral beam injectors in TASKA (i.e., 5 to 250 keV), the sputtering yield of

materials considered as candidates for the dump surfaces is typically in the range  $10^{-1}$  to  $10^{-4}$  atoms per incident particle [see, for example, Roth, et al.<sup>(3)</sup>]. This has two important consequences. Firstly, the phenomenon of surface erosion must be considered with regard to the lifetime of each beam dump. Secondly, the prevention of this sputtered high-Z surface material from entering the plasma region dictates some complex pumping requirements.

### 3. Tritium Absorption and Diffusion Into the Dump Coolant

The 50 keV medium energy pump beam in the barrier coil also performs the central cell fueling by virtue of its mixed D/T composition. However, this latter feature results in a large residual tritium particle flux to the dumps. Under equilibrium conditions, there is a steady state distribution of tritium within the dump matrix yielding a constant flow rate of tritium from both the front and rear (coolant) surfaces of each dump. This can result in an unacceptably large tritium inventory within the dump coolant itself.

### 4. DD and DT Neutron Generation Within the Dump Surfaces

As will be demonstrated later, the beam dumps for all four types of neutral beams in TASKA will contain, under equilibrium conditions, an internal steady state distribution of their respective incident neutral species. In the case of the dumps for the high and low energy beams into the barrier region, a pure deuterium distribution results. Subsequent D neutrals incident on these dumps are, therefore, capable of generating DD neutrons at any depth from the surface down to their implantation depth. In the case of the dumps for the mixed D/T medium energy beams into the barrier coil region, subsequent D and T neutrals



are similarly capable of generating DD and DT neutrons from D on D, D on T and T on D reactions. This, therefore, requires consideration of neutron source intensities and angular distributions from the beam dumps and attention to resulting shielding requirements.

#### 5. Vacuum Pumping of the Dump Chambers

In addition to pumping considerations for the high-Z sputtered surface material, attention must also be given to the re-emission of cold neutral gas from the dump surfaces. Although in the steady state there is a constant flow of neutral particles from both the front and rear (coolant) surface of each dump, it will be demonstrated later that the flow rate from the front surface is much larger than that from the rear surface and approaches the incident neutral beam implantation rate. Sophisticated vacuum techniques are, therefore, required to prevent this large backflow of cold gas from entering the plasma chamber.

In considering the above points, it is interesting to note that the ion dumps internal to the neutral beam injectors are subject to similar but usually more severe operating constraints. In the case of the medium energy D/T beam, for example, approximately 20% of the original charged particle beam from the ion source escapes neutralization and must be directed to an internal ion dump within the injector. The external beam dump, on the other hand, is only exposed to that fraction of the neutral beam which is not trapped in the plasma. For this medium energy beam, the ratio of the ion beam power on the internal dump to the neutral beam power on the external dump is approximately a factor of five.

### VIII.3.2 Design Philosophy

Table VIII.3-1 illustrates those aspects of the TASKA neutral beam parameters which are of relevance to the engineering design of the beam dumps. The parameters of prime importance in this respect are the neutral beam power and the fraction of the beam absorbed in the plasma (i.e., the trapping fraction). In the case of the 250 keV H beam into the plug, for example, only 21% of the incident beam is absorbed in the plasma. Therefore, the dump for the residual beam must be capable of dissipating 79% of the original beam power!

Current designs for high heat flux dumps employ a suitable metal substrate (e.g., copper or a refractory metal) in either tubular form or in the form of a thin plate with etched channels on the rear surface. High pressure water is then forced through these tubular or etched-channel passages. Maximum (pulsed) power densities for these designs are  $\sim 4 \text{ kW cm}^{-2}$ , above which the nucleate-boiling to film-boiling transition occurs resulting in catastrophic burnout of the dump.

It is important to note that current dump designs envisaged for use up to  $\sim 4 \text{ kW cm}^{-2}$  are at present only being designed for, and tested under, a pulsed mode of operation. Thus, in their "Thermoskan" design for Lawrence Berkeley Laboratory, United Technologies<sup>(4)</sup> has developed a photo-etched copper dump plate with dimensions of  $\sim 50 \times 50 \text{ cm}$ , capable of absorbing  $2 \text{ kW cm}^{-2}$  for 20,000 shots. In a similar design,<sup>(1)</sup> RCA has achieved successful test results for  $2.5 - 3 \text{ kW cm}^{-2}$  over 1000 cycles (0.5 seconds on, 0.5 seconds off). At present, the  $36 \times 110 \text{ cm}$  tubular-array beam dumps for the Mirror Fusion Test Facility-B at Lawrence Livermore National Laboratory are being designed<sup>(4,5)</sup> for  $2 \text{ kW cm}^{-2}$ . If currently planned tests are successful, the

Table VIII.3-1  
TASKA Neutral Beam Parameters  
of Relevance to Beam Dump Design

| LOCATION                           | PLUG<br>NEUTRAL BEAM | HIGH ENERGY<br>NEUTRAL BEAM | MEDIUM ENERGY<br>NEUTRAL BEAM | LOW ENERGY<br>NEUTRAL BEAM |
|------------------------------------|----------------------|-----------------------------|-------------------------------|----------------------------|
| NEUTRAL SPECIES                    | Plug                 | Barrier                     | Barrier coil                  | Barrier                    |
|                                    | H                    | D                           | T (56%)<br>D (44%)            | D                          |
| NUMBER OF NBI IN<br>EACH END CELL  | 1                    | 1                           | 3                             | 1                          |
| NUMBER OF ION<br>SOURCES PER NBI   | 1                    | 1                           | 3                             | 1                          |
| BEAM ENERGY (keV)                  | 250                  | 76                          | 50                            | 5                          |
| BEAM POWER PER NBI (MW)            | 2.7                  | 3.3                         | 8.27                          | 0.25                       |
| RADIUS OF PLASMA<br>FOOTPRINT (cm) | ~ 8                  | ~ 12                        | ~ 9.3                         | ~ 38 (unfocussed)          |
| PLASMA<br>TRAPPING<br>FRACTION     | 0.21                 | 0.42                        | 0.95                          | 0.99                       |

design will be upgraded to  $\sim 4 \text{ kW cm}^{-2}$ . Although MFTF-B will be a pulsed device,<sup>(6)</sup> the relatively long burn time of 30 s means that the beam dumps will be operated in a quasi-steady state mode (or, at least, with a beam-on time which is much greater than the thermal time constant of the dump material).

The main engineering problems in the design of high power density beam dumps are the very large pumping powers required for efficient coolant flow and the structural integrity of the dump substrate. Thus, for power densities in excess of those above  $2\text{-}4 \text{ kW cm}^{-2}$ , there is a rapid and unacceptable increase in pumping power (values in the region of hundreds of kW per  $\text{m}^2$  of dump area are required) coupled with a rapid decrease in the structural safety factor. It is possible to withstand very high power densities by utilizing very short test sections coupled with very high pressure drops. Gambill and Green,<sup>(7)</sup> for example, report a high heat-flux dump consisting of a 2.3 cm long tube with a 0.6 cm inside diameter and 3.4 MPa differential pressure drop, capable of absorbing  $11 \text{ kW cm}^{-2}$ . However, such a performance cannot be achieved for long tubes and reasonable dump areas due to excessive pressure drop and associated excessive pumping power requirements. Further considerations of pumping power requirements for high heat flux surfaces may be found in a recent paper by Moir et al.<sup>(2)</sup>

Recently, there have been several novel designs of large area high heat-flux dumps for operation at around  $10 \text{ kW cm}^{-2}$ , including a liquid lithium metal dump<sup>(8)</sup> and a high-velocity-flow circumferentially-cooled dump of semi-circular cross section.<sup>(2)</sup> However, besides presenting volume limitation constraints, such dumps are currently only in the conceptual stage and could not, therefore, be considered for use in the present TASKA design.

### VIII.3.3 Engineering Design

Further reference to Table VIII.3-1 indicates that maximum incident beam power densities at the plasma footprint are  $\sim 30 \text{ kW cm}^{-2}$ , with maximum residual power densities for the untrapped beam fractions of  $\sim 10 \text{ kW cm}^{-2}$ . However, reference to the present dump designs described above implies that maximum attainable power densities for such dumps are  $\sim 4 \text{ kW cm}^{-2}$  and then only for quasi-steady state operation. Accordingly, the following design criterion was defined: the neutral beam dumps for TASKA must be capable of withstanding a maximum steady state power density of  $1 \text{ kW cm}^{-2}$  under normal operation of full plasma density.

Figure VIII.3-1 is a schematic of the representative design for the TASKA beam dumps. The  $1 \text{ kW cm}^{-2}$  maximum power density criterion has been accommodated by inclining the dump surface at an angle relative to the beam direction so that the beam is dispersed over a larger effective area. Furthermore, the lower section of the dump has been inverted to form a V-shape. This yields two important advantages. Firstly, sputtering of the dump surface is reduced due to the phenomenon of "cross-sputtering", whereby a fraction of the beam-sputtered material from one plate will now tend to plate-out on the opposing plate (see Section VIII.3.5). Secondly, the overall volume occupied by the dump is reduced by a factor of two. This latter factor is especially important for the medium energy beam dumps situated in the barrier coil, where there are severe space constraints. Further reduction in volume would be possible by employing a multiple-V design.<sup>(9)</sup>

Two candidates were examined for employment as the dump substrate. The first consisted of a copper plate of approximately 3.5 mm thickness with an explosively-bonded molybdenum layer on the front surface. The rear surface of

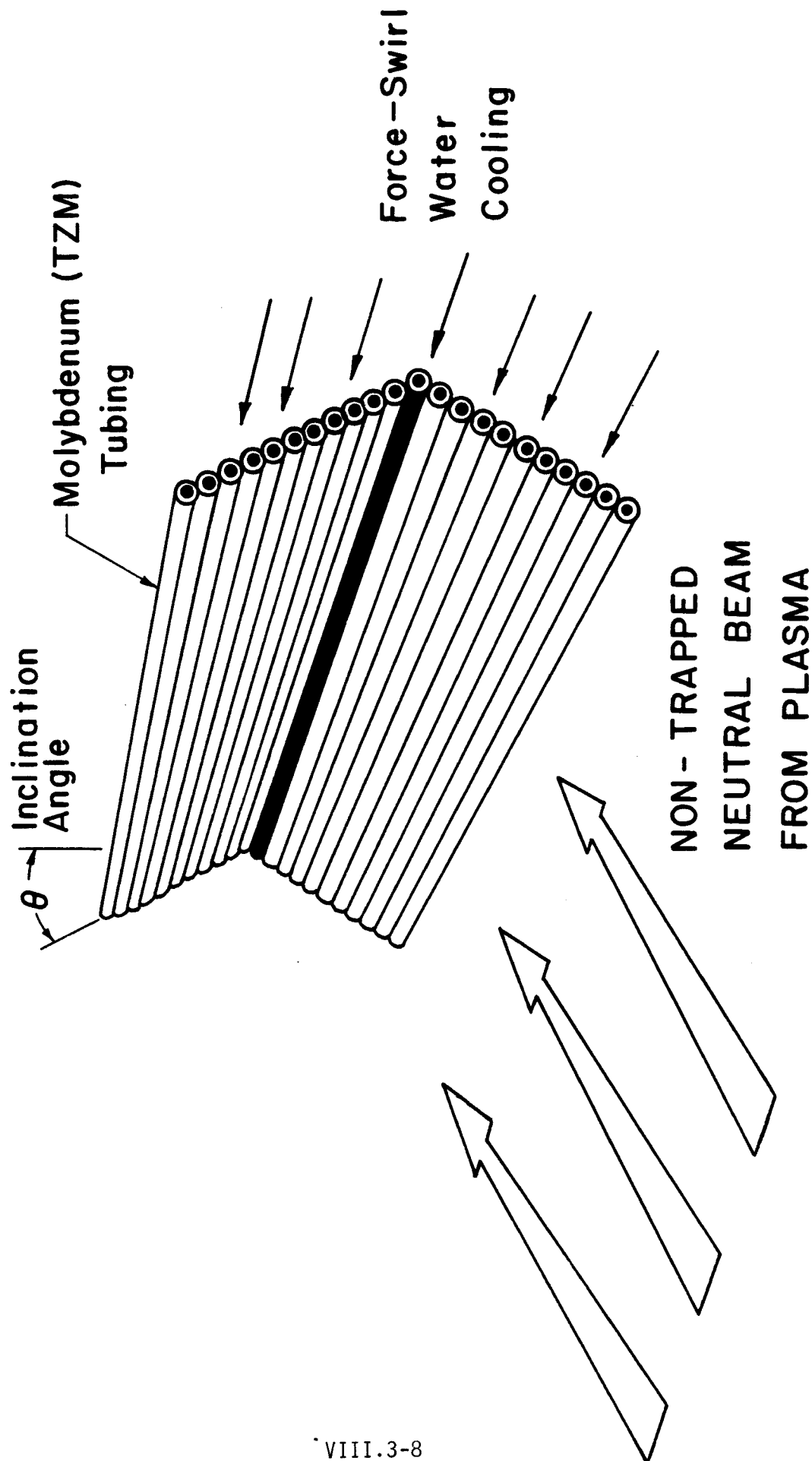


Fig. VIII.3-1 Schematic design of the V-configuration beam dump.

the copper plate contained machined channels of approximately 1.5 mm depth for improved water cooling.

The second dump substrate design, and the one chosen for use in TASKA, consists of a parallel array of TZM molybdenum alloy<sup>(10)</sup> tubes of approximately 10 mm OD and 2 mm wall thickness, the precise value of the latter parameter depending on the particular beam dump under consideration - see later. The tubular dump design is shown in Fig. VIII.3-1 and was chosen in preference to the copper plate design mainly from considerations of ease of construction and of current operating experience with this type of dump design.<sup>(4)</sup> The tubular dump is cooled by the "force-swirled" technique, by which water is forced along the tube length in a swirl vortex motion induced by an internal twisted tape. Such a motion of the water yields virtually forced-convection cooling of the tube inner wall and increases the heat transfer coefficient of the system over that of a conventional flow system by up to 1.5. Typical coolant conditions for this system are a pressure of 5.4 MPa, a flow rate of  $5.14 \times 10^4$  g/minute per square meter of surface area and an outlet temperature of 150°C.

The selection of TZM molybdenum alloy (Mo, 0.5 w/o Ti, 0.08 w/o Zr, 0.02 w/o C) as a dump material is attributable to four main factors: it is commonly available in thin walled tubing; it is a refractory metal and can, therefore, withstand high temperatures; it has a long lifetime with respect to stresses; it has relatively low sputtering properties.

Table VIII.3-2 gives the major design parameters for the TASKA neutral beam dumps. Fuller details of these design values are given in an associated publication.<sup>(9)</sup> It can be seen that no actual dump is required for the low energy barrier beam since the opposing first wall is easily capable of

Table VIII.3-2  
TASKA Beam Dump Parameters

|                                                                       | PLUG<br>NEUTRAL BEAM<br>(250 keV p) | BARRIER<br>HIGH ENERGY<br>NEUTRAL BEAM<br>(76 keV D) | BARRIER<br>MEDIUM ENERGY<br>NEUTRAL BEAM<br>(50 keV D/T) | BARRIER<br>LOW ENERGY<br>NEUTRAL BEAM<br>(5 keV D)                                          |
|-----------------------------------------------------------------------|-------------------------------------|------------------------------------------------------|----------------------------------------------------------|---------------------------------------------------------------------------------------------|
| DUMP POWER DENSITY<br>- NORMAL OPERATION<br>(kW cm <sup>-2</sup> )    | 1.0                                 | 0.58                                                 | 0.2                                                      | 4x10 <sup>-4</sup> (at<br>first wall)                                                       |
| DUMP POWER DENSITY<br>- NO PLASMA CONDITION<br>(kW cm <sup>-2</sup> ) | 1.27                                | 1.0                                                  | 4.0                                                      | 4x10 <sup>-2</sup> (at<br>first wall)                                                       |
| DUMP CONFIGURATION                                                    | Rectangular-V                       | Rectangular-V                                        | Circular-V                                               | No dump required.<br>(Opposing first wall<br>is capable of with-<br>standing power density) |
| DUMP DIMENSIONS (cm)                                                  | 29.4x44.1x62.5<br>(hwxwd)           | 50.1x71.6x47.5<br>(hwxwd)                            | 60.1,23.0,33.5<br>(or,ir,d)                              |                                                                                             |
| INCLINATION ANGLE $\theta$                                            | 76.7°                               | 61.9°                                                | 58.1°                                                    |                                                                                             |
| DISPLACEMENT FROM<br>Z-AXIS (cm)                                      | 125.4                               | 587.1                                                | 99.5                                                     |                                                                                             |
| MOLYBDENUM SURFACE<br>THICKNESS (cm)                                  | 0.2                                 | 0.5                                                  | 0.6                                                      |                                                                                             |
| OUTER SURFACE<br>TEMPERATURE (°C)                                     | 510                                 | 495                                                  | 277                                                      |                                                                                             |
| INNER SURFACE<br>TEMPERATURE (°C)                                     | 350                                 | 265                                                  | 190                                                      |                                                                                             |
| EFFECTIVE BEAM<br>AREA OVER DUMP<br>(cm <sup>2</sup> )                | 2131                                | 2068                                                 | 3300                                                     |                                                                                             |



withstanding the beam power density. However, this opposing first wall region will require recessing due to gas pumping constraints - see below.

It should also be noted from Table VIII.3-2 that the beam dumps for the plug H beam and the high energy barrier beam are single rectangular-V structures as shown in Fig. VIII.3-1, whereas the medium energy barrier coil beam dump is a one piece dump having circular symmetry about the z-axis and accommodates all three beams directed into each barrier coil. Dimensional safety factors have been included for each dump in that their linear dimensions are 50% greater than the expected beam imprints under normal operation.

The dump surface temperatures in the table were obtained from heat transfer calculations on the basis of forced convection and conduction through the substrate. Any effects of burn-out, vibration and erosion on the heat transfer coefficient were not included in these calculations.

It can also be seen from Table VIII.3-2 that the beam dumps have been designed to withstand a maximum steady state power density of  $1 \text{ kW cm}^{-2}$  under the normal operating conditions of full power density and appropriate plasma trapping fractions shown in Table VIII.3-1. If the plasma should be lost, the power densities of the plug beam and the high energy barrier beam at their respective dumps are still in the region of the maximum design criterion. However, during such an excursion, the power densities for the medium energy beams on the barrier coil dump would rise to  $4 \text{ kW cm}^{-2}$  which is unacceptable under steady state conditions. Therefore, a significant loss of plasma density in the barrier region would necessitate that these latter neutral beam injectors trip out. Typical shut-off time for the beam injectors is  $\sim 0.5 \text{ ms}$  and, therefore, is much less than the passive thermal inertia of the dump.

Steady state testing of the medium energy barrier coil beams could be accomplished in an external test stand with associated large area beam dumps. In situ testing of the beam optics, etc., under plasma conditions would be performed in short (e.g., 10-50 ms) pulses, which again can be accommodated by the thermal inertia of the dump.

#### VIII.3.4 Dump Location and Vacuum Pumping Considerations

Although methods for vacuum pumping of the beam dump spaces have been described elsewhere in Section IV.5, it is necessary to consider how the pumping requirements can be ameliorated by strategic location of the dumps. In this connection, impact of the residual neutral beams with the dump surfaces has two detrimental effects. Firstly, neutral beam impact causes sputtering of the dump surface with the resulting emission of molybdenum atoms. Secondly, under equilibrium conditions, there is a large backflow of cold neutral gas from the front surface of each dump.

With regard to the sputtered molybdenum atoms, their eventual fate is not clear. They will be emitted from the dump surface as neutral atoms with certain angular and energy distributions. The region in front of each dump will contain a cold plasma region of uncertain density. Therefore, due to either this cold plasma or subsequent incident neutral particles, the molybdenum atoms will eventually undergo ionization. The resulting  $\text{Mo}^{n+}$  ion will then be directed along a field line in the system which will act as a guiding center for the particle. Depending on the original mean free path (mfp) for ionization, there are now three possibilities. For a short mfp, the molybdenum ion will impact the vacuum wall in the region of the dump, thus initiating the release of impurities from the wall. For a larger mfp, the molybdenum atom will escape the dump space region and, on ionization, will either be

directed to the end of the machine due to the self-cleaning action of the tandem mirror or may become trapped as a deleterious high-Z impurity in the main plasma.

Therefore, with regard to both sputtered Mo atoms and cold gas backflow from the dumps, it is necessary to locate the dumps as far back from the plasma as possible. In the case of the dumps for the plug beam and the high energy barrier beam, this has been accomplished by recessing them a distance of  $\sim 1$  m into the opposite vacuum wall. In the case of the low energy barrier beam, no separate dump is required as described above. However, that portion of the opposite vacuum wall impacted by the residual low energy beam is also recessed by 1 m, thus forming a re-entrant cavity which can be conveniently pumped.

Location of the dump for the three medium energy beams directed into the barrier coil presents a much more complex problem due to the severe space constraints in this region. Figure VIII.3-2 is a cross section of the barrier coil region and shows the location of this dump. The section taken at B - B' is shown as an end-elevation in this figure and illustrates the intersection of the three medium energy beam lines and the dump vacuum pumping ports with the reactor vessel. The section taken at A - A' in Fig. VIII.3-2 is shown in end-elevation in Fig. VIII.3-3; note that the dump is a one piece construction with circular symmetry about the z-axis. If this dump were to be positioned nearer the plasma, pumping of the dump space becomes very difficult; in addition the three individual beam imprints will overlap in the x-y plane, thus increasing the power density at the dump. However, positioning the dump any further back means that the radiation shielding for the normal conducting copper insert coil becomes seriously jeopardized; as it is tungsten shielding



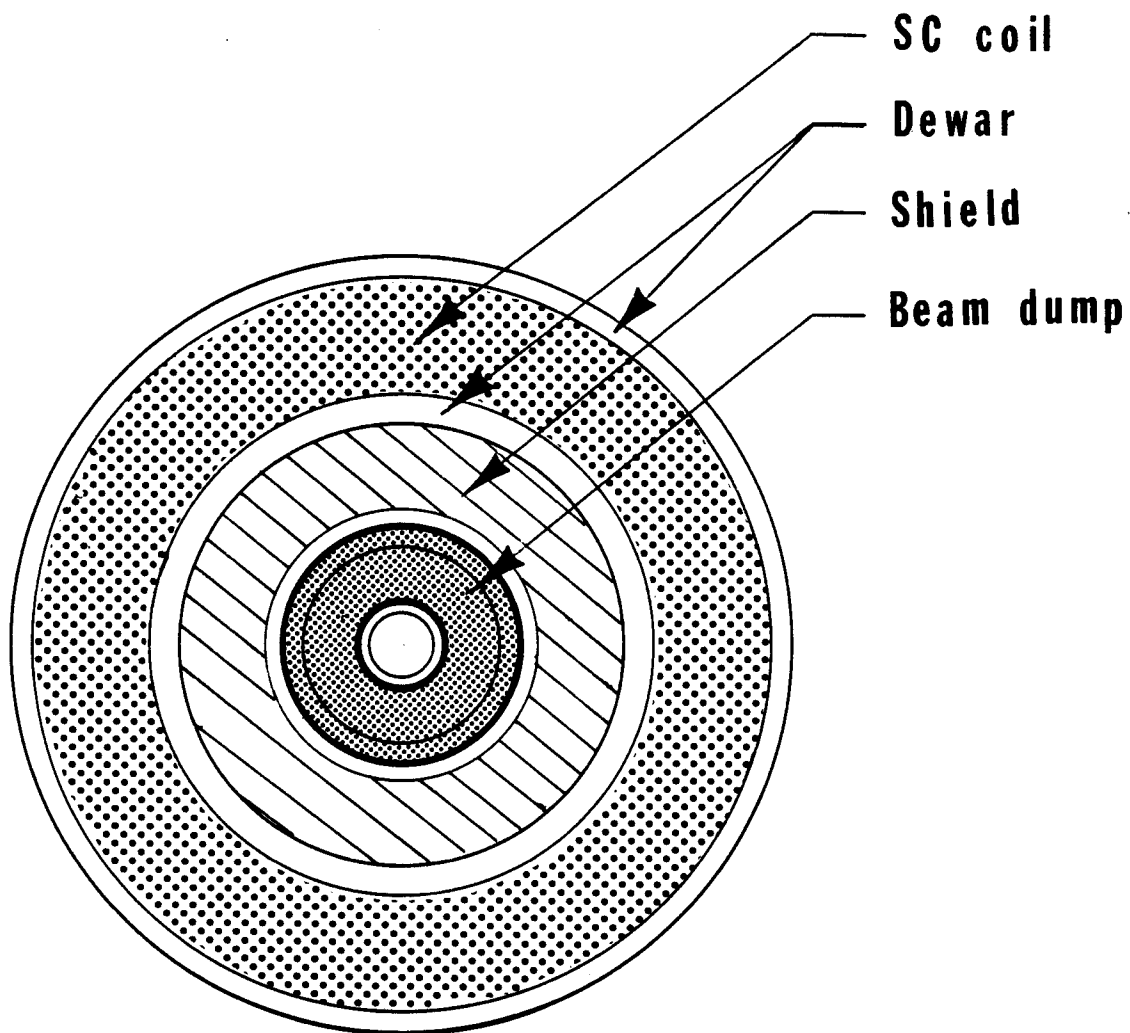


Fig. VIII.3-3 Cross section of the barrier coil region showing the circular-V medium energy neutral beam dump. This is section A-A' from Fig. VIII.3-2 .

must be employed in this region. The location of this dump is, therefore, a compromise between two opposing constraints.

#### VIII.3.5 Sputtering and Lifetime Considerations

Continual bombardment of the beam dumps by the incident neutral beams produces a gradual erosion of the dump surfaces due to sputtering of the molybdenum substrate. Requirements for estimates of dump lifetimes therefore arise. This is especially important for the medium energy beam dump in the barrier coil since disassembly of the coil assembly due to dump failure is a complex operation with an associated long machine downtime.

Figure VIII.3-4 illustrates the energy dependence of the sputtering yield (in atoms/ion) of molybdenum for H, D, T and  $^4\text{He}$  ions of normal incidence. The data for H, D and  $^4\text{He}$  below 10 keV were taken from Roth et al.,<sup>(3)</sup> while the data above this energy came from an, as yet, unpublished source.<sup>(11)</sup> Values for tritium were taken from Bay, et al.<sup>(12)</sup> These curves are not a function of the temperature of the dump surface, since the sputtering of elements by light ions does not increase with temperature providing no chemical reactions occur.<sup>(13)</sup>

It should be stressed that the data in Fig. VIII.3-4 are for normal incidence. As seen above, the beam dumps are inclined at an angle  $\theta$  with respect to the incident beam in order to reduce the beam power density by a factor of  $\cos\theta$ . However, as the angle of incidence  $\theta$  between the incident beam and a normal to the surface is increased to values greater than zero, a significant increase in sputtering yield occurs over those values for normal incidence shown in Fig. VIII.3-4. Sigmund<sup>(18)</sup> has determined the dependence of the sputtering yield  $Y(\theta)$  on  $\theta$  to be

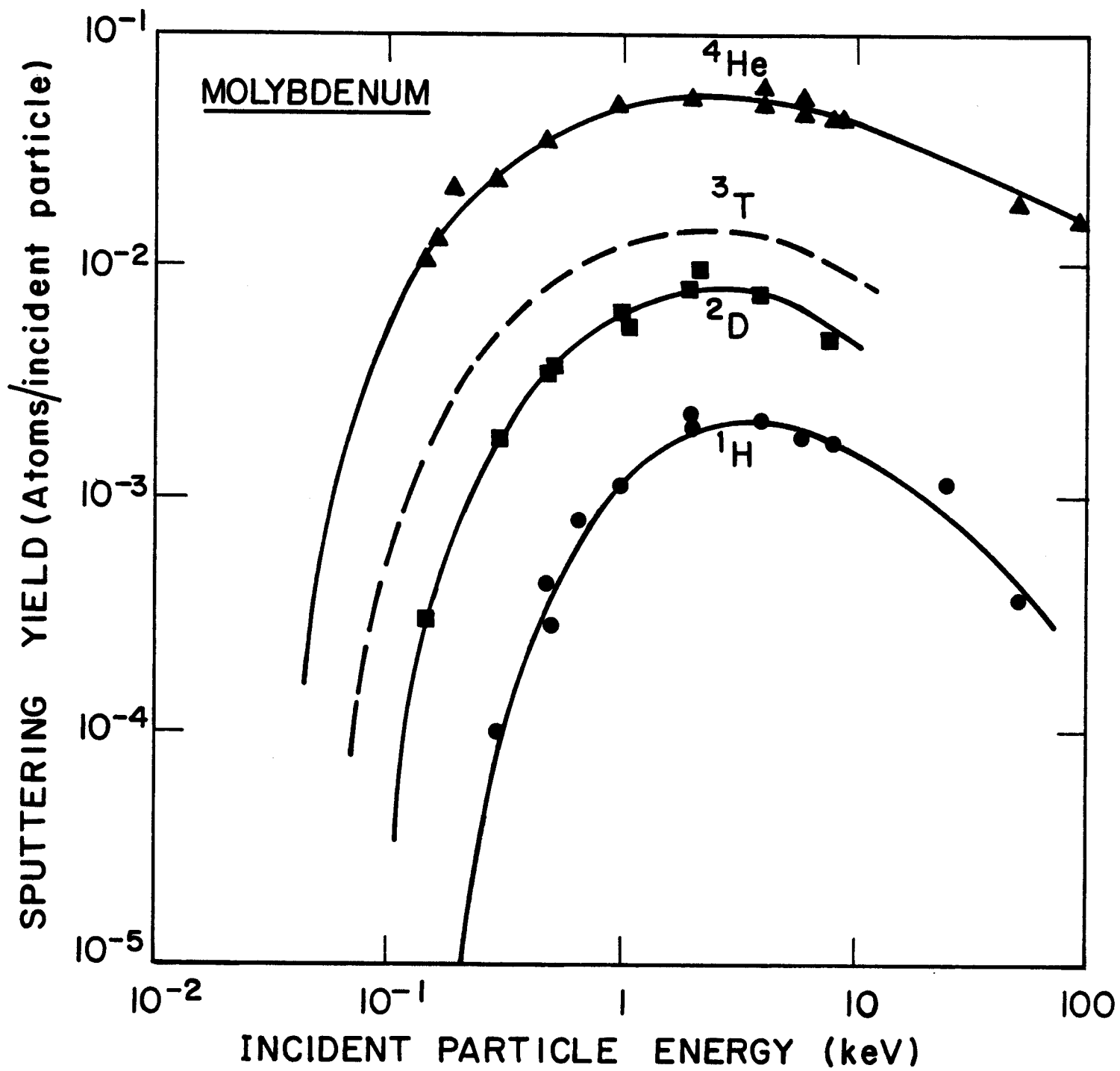


Fig. VIII.3-4 Energy dependence of the sputtering yield for molybdenum at normal incidence.

$$Y(\theta) = Y(o) \cos^{-n}\theta \quad (\text{VIII.3-1})$$

where:  $\theta$  = angle of incidence

$Y(o)$  = sputtering yield at normal incidence

and where  $n$  is a factor between 1 and 2 and is a function of the mass ratio of the incident and sputtered particles.

In addition to increasing the sputtering yield, impact of the neutral beam with the dump surface at non-normal incidence has an interesting effect on the angular distribution of the sputtered molybdenum atoms. At normal incidence, the angular distribution of sputtered atoms is diffuse and follows a cosine distribution [actually slightly "over-cosine" due to the contribution of reflected incident ions.<sup>(13)</sup>]. As the incident beam moves away from normal incidence, the sputtered angular distribution changes from diffuse and becomes increasingly specular with increasing angle of incidence. The peak in the sputtered product distribution, therefore, occurs at an angle of "reflection" approximately equal to the angle of incidence. It can be seen, therefore, that increasing the angle of inclination  $\theta$  of the V-configuration beam dump, with respect to the incident beam, causes a greater portion of the sputtered molybdenum from one V-plate to be emitted in the direction of the adjacent plate. Thus, in this phenomenon of "cross-sputtering", sputtered material from one surface will tend to plate out on the adjacent surface and thus reduce the overall erosion of the dump surfaces. Note, however, that it is not clear that the redeposited material is as good as the original surface in respect of the latter's relatively low sputtering properties.

It should be noted that self-sputtering yields for molybdenum on molybdenum are considerably greater than the corresponding yields for light ion



sputtering shown in Fig. VIII.3-4. In fact, for incident energies greater than  $\sim 1$  keV, the molybdenum self-sputtering yield becomes greater than unity.<sup>(14)</sup> In such a case, cross-sputtering would actually be detrimental to dump lifetime. Fortunately, in the energy regime of the TASKA neutral beams, sputtered Mo atoms are emitted in the eV range of energies,<sup>(9)</sup> thus cross-sputtering enhancement of dump lifetime will occur.

One serious obstacle to an accurate calculation of the beam dump lifetimes due to the sputtering process is that, at angles away from normal incidence, the data base for sputtering is rather deficient, especially at higher energies  $\gtrsim 10$  keV. The paucity of data for angular and energy distributions of sputtered products is particularly marked. For this reason two calculations of the estimated dump lifetimes were performed. Firstly, a worst case lower bound for each dump lifetime was computed by assuming that the increase in sputtering due to non-normal incidence followed a  $\cos^{-2}\theta$  dependence. This is the upper limit of current estimates since the factor  $n$  in Eq. (VIII.3-1) lies between 1 and 2. In addition, no allowance was made for cross-sputtering in that all initial atoms sputtered from the dump surfaces were assumed to be lost from the system.

In the second, more formal, analysis the current (and rather sparse) data for the actual angular dependence of sputtering yields were taken into account. Also, data on angular and energy distributions of emitted atoms were included, so that some estimation of cross-sputtering enhancement could be obtained. Further details of these calculations may be found in an associated publication.<sup>(9)</sup>

Table VIII.3-3 documents the results of the above calculations, showing the sputtering rates and estimated lifetimes for the TASKA beam dumps. The

TABLE VIII.3-3  
Beam Dump Sputtering  
Rates and Estimated Lifetimes

|                                          | ESTIMATION FROM FORMAL ANALYSIS                                           |                                                                    |                                                       |                                                                           | WORST CASE LOWER BOUND                                             |                                                       |
|------------------------------------------|---------------------------------------------------------------------------|--------------------------------------------------------------------|-------------------------------------------------------|---------------------------------------------------------------------------|--------------------------------------------------------------------|-------------------------------------------------------|
|                                          | DEPTH SPUTTERING<br>RATE OF Mo<br>(cm/s)                                  | DEPTH OF Mo<br>SPUTTERED OVER<br>MACHINE LIFE <sup>1</sup><br>(cm) | ESTIMATED DUMP<br>LIFETIME <sup>2</sup><br>(FP YEARS) | DEPTH SPUTTERING<br>RATE OF Mo<br>(cm/s)                                  | DEPTH OF Mo<br>SPUTTERED OVER<br>MACHINE LIFE <sup>1</sup><br>(cm) | ESTIMATED DUMP<br>LIFETIME <sup>2</sup><br>(FP YEARS) |
| PLUG<br>NEUTRAL BEAM                     | 5.73x10 <sup>-10</sup>                                                    | 0.096                                                              | > machine life                                        | 1.59x10 <sup>-9</sup>                                                     | 0.27                                                               | 3.14                                                  |
| BARRIER<br>HIGH ENERGY<br>NEUTRAL BEAM   | 1.34x10 <sup>-9</sup>                                                     | 0.22                                                               | > machine life                                        | 4.79x10 <sup>-9</sup>                                                     | 0.80                                                               | 2.65                                                  |
| BARRIER<br>MEDIUM ENERGY<br>NEUTRAL BEAM | 8.53x10 <sup>-10</sup>                                                    | 0.14                                                               | > machine life                                        | 2.91x10 <sup>-9</sup>                                                     | 0.49                                                               | 4.36                                                  |
| BARRIER<br>LOW ENERGY<br>NEUTRAL BEAM    | No dump required.<br>(Lifetime of opposing first wall is > machine life). |                                                                    |                                                       | No dump required.<br>(Lifetime of opposing first wall is > machine life). |                                                                    |                                                       |

<sup>1</sup>Machine life = 5.3 FP years

<sup>2</sup>Time required to reduce surface thickness to 20% of its original value

lifetime of a particular dump has been defined as the time required to reduce the surface thickness of the dump, by sputtering, to 20% of its original value. This is a rather arbitrary definition of lifetime, but it gives a reasonable safety margin on dump performance, especially since the inner diameter of the molybdenum tubing cannot be assumed to be perfectly regular.

From the table, it is seen that all three dumps will outlast the life of the machine (5.3 full power years) under the results from the formal analysis. In the case of the worst case lower bound estimates, the lifetime of the medium energy beam dump in the barrier coil is seen to be 4.36 which is slightly less than the machine life. However, this is certainly longer than the estimated changeout period for the normal conducting insert coil in the barrier (2.3 FPY) and is, therefore, satisfactory.

The worst case lower bound lifetime estimates for the plug beam dumps and the high energy beam dumps are 3.14 and 2.65 FPY, respectively. Therefore, under this estimation they would require renewing at least once during the operating life of the machine. However, unlike the medium energy beam dump in the barrier coil, they are easily accessible. Lifetimes of this order are, therefore, acceptable.

### VIII.3.6 DT and DD Neutron Generation From the Dump Surfaces

#### VIII.3.6.1 Nature of the Problem

The production of fusion neutrons from the dump surfaces themselves arises from the interaction of incident D or mixed D/T neutral beam particles with the equilibrium deuterium or deuterium/tritium distributions in the respective dump substrates. The neutral species in the plug heating beams are pure protons, thus considerations of neutron production from these dumps clearly do not apply. The high (76 keV) and low (5 keV) energy beams into the

barrier region consist of pure deuterium and thus are capable of DD neutron generation at their dump surfaces. The three medium energy (50 keV) beams into the barrier coil region each comprise a mixed deuterium-tritium species with atomic proportions of 44% and 56%, respectively. Thus, interaction of the D/T beams with the steady state D/T distributions in the dump gives rise to neutron generation via the separate reactions of D on D, D on T and T on D, respectively.

Neutrons resulting from the interaction of incident deuterons with tritium generated in-situ by the other branch of the DD reaction (i.e.,  $D(d,p)T$ ) may be neglected. This is due to the low equilibrium concentration of such tritium compared to the main equilibrium deuterium and tritium concentrations implanted by the incident beams.

Neutrons will also be generated in the medium energy beam dumps via the TT reaction ( $T(t,2n)^4\text{He}$ ,  $Q = 11.33 \text{ MeV}$ ). Preliminary studies show<sup>(19)</sup> that the magnitude of the TT neutron-producing cross section in this energy range (i.e., tens of keV) is comparable to that for the DD reaction -- for example, at 50 keV  $\sigma_{DD}/\sigma_{TT} \sim 1.34$ . However, due to the current paucity of data for this reaction, especially with regard to neutron angular distributions, TT neutron generation was not formally computed in this study; approximate estimates of TT yields will be presented in Section VIII.3.6.3. Work is continuing on this effect.

Accurate computation of neutron production from the dump surfaces, including source intensities and angular distributions of emission, consists of two distinct stages. Firstly, it is necessary to determine the steady state distribution of the neutral D/T species within each dump substrate from consideration of temperature and concentration profiles in the dump surface.

Secondly, computation of neutron production is performed by integration of all DD, DT and TD incremental neutron sources, ranging from the dump surface down to the implantation depths for each D or T neutral particle incident on the dump. Physically, this results in a very complex problem due to the nature of the processes occurring in the dump surface.

Accordingly, formal solution for neutron generation from the dumps required extensive computational effort and resulted in the evolution of the computer code "BDUMP". The code BDUMP computes neutron source and angular distributions resulting from the interaction of D, T or D/T beams with solid or liquid surfaces. It can also be used to compute neutron emission spectra from conventional tritiated or deuterated neutron-producing targets. BDUMP will shortly be available for general use on the U.S. National Magnetic Fusion Energy Computer System at Lawrence Livermore National Laboratory, and further information can be obtained in references 15 and 16.

A brief discussion of the physical mechanisms contributing to neutron generation within the dumps will now be given. Complete details will be given in an associated publication<sup>(15)</sup>, while a concise overview may be found in Ref. 20.

#### VIII.3.6.2 Substrate Equilibrium Particle Distributions and Tritium Loss Rates

The incident D neutrals, or D/T neutrals for the medium energy barrier coil beams, slow down in the dump substrate and stop at their respective implantation depths. This latter parameter is a function of the energy of the particles and their specific energy loss ( $dE/dx$ ) in the medium. For example, the implantation depth of 50 keV deuterium atoms in pure molybdenum is  $\sim .5 \mu\text{m}$ . Once brought to rest, the D or D/T atoms diffuse, both forward and backwards, from their implantation site under the action of the resulting concentration

gradients. However, the resulting concentration profiles cannot be accurately computed from a simple one region diffusion problem for two reasons. Firstly, for high neutral beam intensities, the stopping power for incident particles depends on the  $dE/dx$  parameters for both the substrate atoms and those neutral atoms already implanted in the dump plate. Thus, as the number density of implanted atoms becomes comparable to that of the substrate, the implantation depths for subsequent incident particles tends to decrease, moving towards the surface. Note that, in addition, the implantation depths for the incident D and T particles in the barrier coil dump will be different due to the different energy per unit mass of these ions. Secondly, under normal operation there is an appreciable temperature gradient between the front and rear (coolant) surfaces of each dump; therefore, the diffusion coefficient  $D_j(x)$  of each particle type  $j$  is a function of position in the material.

Once equilibrium has been reached, there will be a steady state profile of D or D/T within each dump and a constant flow rate of cold D or D/T particles from both the front and rear surfaces equal to the particle implantation rate from the beam.

The formal solution to the diffusion equation for this medium, where the diffusion coefficient is a function of position, is given in the associated publications.<sup>(9,15)</sup> It is shown there that the ratio  $R_j$  of the flow rate of particles of type  $j$  from the rear coolant surface of the dump to the flow rate from the front surface of the dump is given by

$$R_j = \frac{|F(\frac{k_j}{m\delta_j+c}) - F(k_j/c)|}{F(\frac{k_j}{m(\delta_j-t_1)+c}) - F(k_j/c)} \quad (\text{VIII.3-2})$$

where

$$F(x) = -\frac{e^x}{x} + \ln x + \sum_{n=1}^{\infty} \frac{x^n}{n \cdot n!}$$

$$m = (T_f - T_r)/t_1 ,$$

$$c = T_f + \frac{\delta_j}{t_1} (T_r - T_f)$$

$$R_j = \Delta H_j/R$$

and where

$t_1$  = dump thickness

$\delta_j$  = implantation depth of particle type j

$T_f$  = absolute temperature of front surface of dump

$T_r$  = absolute temperature of rear (coolant) surface of dump

$R$  = gas constant

$\Delta H_j$  = diffusion activation energy for particle type j (assuming that the temperature dependence of the diffusion coefficient follows  $D_j(T) = D_{0j} \exp (-\Delta H_j/RT)$ ).

Computing the tritium flow-rate ratio to the rear (coolant) and front surfaces of the barrier coil beam dumps yields a value of  $R_j = 2.09 \times 10^{-5}$ . This translates to a total tritium loss to the dump coolant of 5.68 kCi per year which is very large. It is interesting to note that had the relatively small temperature gradient across the plate ( $T_f = 277^\circ\text{C}$ ,  $T_r = 190^\circ\text{C}$ ) been neglected, the resulting tritium flow to the coolant would have been overestimated by a factor of  $\sim 4$ .

It should also be noted here that the inclusion of a thin metal surface of low tritium diffusivity between the molybdenum substrate and the coolant can promote a drastic improvement in the tritium retention of the dump. Aluminum is a candidate material for this "tritium-buffer zone" and could possibly be applied by vapor deposition or similar process. If the positional/temperature dependence of the diffusion coefficients within each zone is neglected here for simplicity, then the equation for the flow rate ratio becomes

$$R_j = \frac{\delta_j}{(D_{1j}(x)/D_{2j}(x)) t_2 + t_1 - \delta_j} \quad (\text{VIII.3-3})$$

where:

$\delta_j$  = implantation depth of particle type j

$t_1$  = thickness of dump surface

$t_2$  = thickness of tritium-buffer material

$D_{1j}(x)$ ,  $D_{2j}(x)$  = diffusion coefficients of particle j in regions 1 and 2.

Thus, neglecting the positional dependence of  $D_{1j}$  and  $D_{2j}$ , a backing region of aluminum of thickness in the range 10-100  $\mu\text{m}$  would reduce the tritium diffusion to the coolant in the range 0.36% - 0.036% of its former



value at a dump surface temperature of  $\sim 200^{\circ}\text{C}$ . This is a significant decrease!

### VIII.3.6.3 Neutron Production

Once the equilibrium particle distributions have been ascertained, the BDUMP code proceeds to determine the neutron production distributions. These are computed by discretizing the implantation range in the dump substrate for each incident particle type into a finite number of slowing down bins. The differential neutron yield at any specified energy and emission angle (i.e., neutrons  $\text{cm}^{-2} \text{ s}^{-1}$  per unit solid angle) originating from each incremental depth/energy bin is then a function of the local equilibrium D or D/T number density and the respective differential cross sections for that bin. The total number of neutrons per steradian which originate from energy bin  $E_i$  and emerge from the dump at a laboratory angle  $\theta$  with energies between  $E_n$  and  $E_n + dE_n$  per incident (D or T) particle is

$$\sum_{i,j} N_{ij}(\theta, E_n) dE_n = \sum_{i,j} \frac{\frac{d\sigma_{ij}}{dw'}(E_i, \theta') \frac{dw'}{dw}(E_i, \theta) n_j(E_i) \frac{dE_i}{dE_n}(E_n, \theta) dE_n}{-dE_i/dx}$$

(VIII.3-4)

- where  $i$  = incident particle (D for high and low energy beams, D or T for medium energy beams)
- $j$  = target particle (D for high and low energy beams, D or T for medium energy beams)
- $E_i$  = instantaneous energy of incident particle  $i$  at a specific depth in surface
- $\theta$  = neutron emission angle in lab frame

- $\theta'$  = neutron emission angle in cm frame  
 $d\sigma_{ij}/dw'$  = differential cross section in cm for reaction of particle i with j  
 $dw'/dw$  = ratio of solid angles in cm and lab frames  
 $n_j(E_i)$  = number density of target particle j in dump where incident particle has energy  $E_i$   
 $dE_i/dx$  = specific energy loss for incident particle i at energy  $E_i$  in surface (contributions from both dump substrate material and target particle distribution).

The neutron energy  $E_n$  at a lab emission angle  $\theta$  is related to the instantaneous incident particle energy  $E_i$  at a specific depth by considerations of the kinematics of the  $j(i,n)$  neutron-producing reaction. The total neutron yield at a given lab angle  $\theta$  is then given by the integral of Eq. (VIII.3-4) with respect to the instantaneous energy  $E_i$  of the incident particle, from its maximum value at the surface of the dump down to zero at the implantation depth. In BDUMP, this is equivalent to summing the angular yields from each incremental energy/depth bin in the implantation range. Further details of these processes may be found in Ref. 15.

Figure VIII.3-5 illustrates the BDUMP computed neutron yields arising from the interaction of each medium energy neutral beam (56% T, 44% D) with the beam dump in the barrier coil. Neutrons are seen to result from the DD reaction (D on D) and the DT reaction (D on T and T on D). The yields shown in Fig. VIII.3-5 should be multiplied by three for the total yield from each dump. The neutron yields for each reaction type are plotted as a function of the surface temperature of the dump. The strong dependence of the neutron source strength on this latter parameter is attributable to the temperature

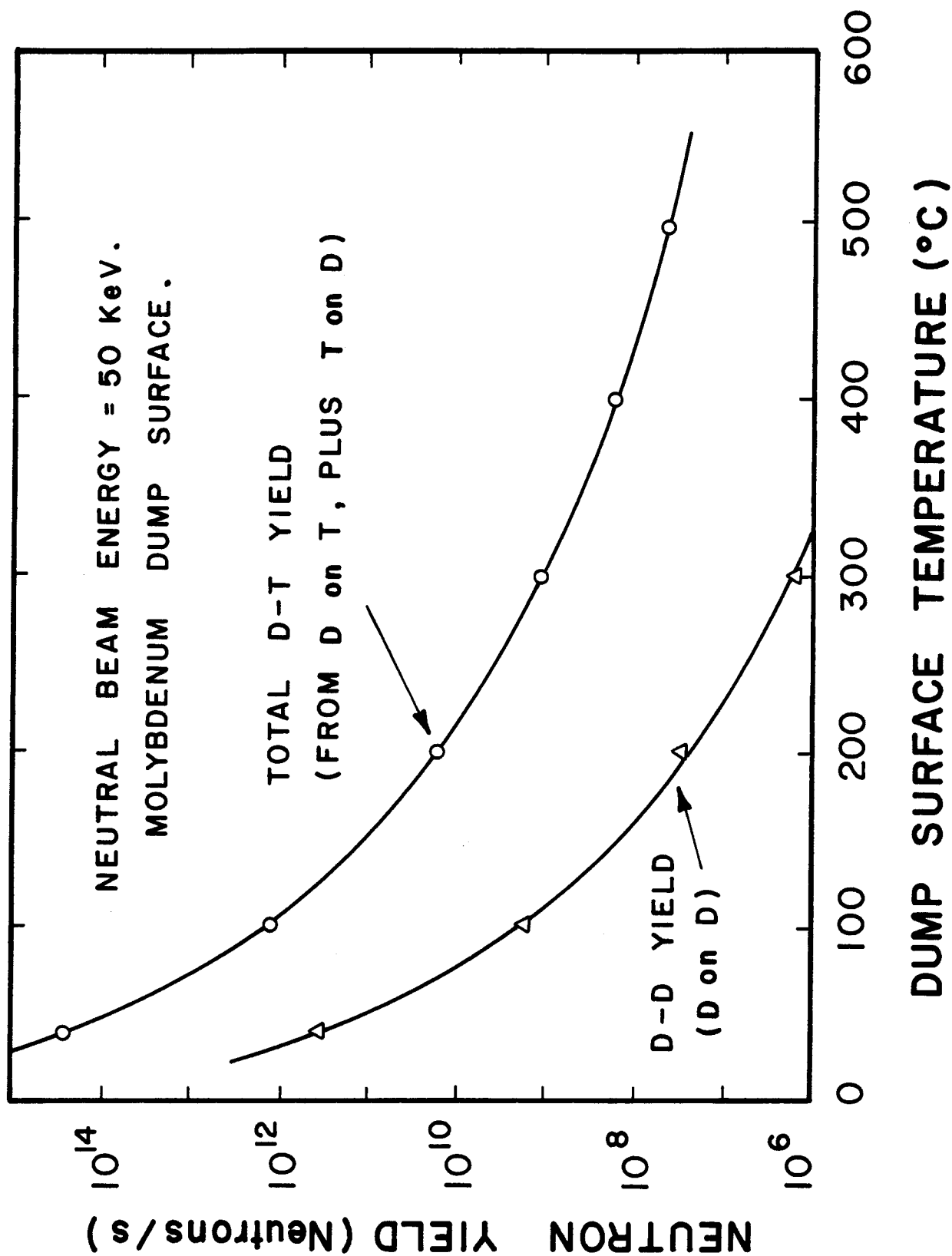


Fig VIII.3-5 Beam-induced neutron production from the medium energy neutral beam dump (via the code BDUMP). The incident beam current and energy are 8.27 A and 50 keV, respectively, and consist of 56% T and 44% D neutrals.

dependence of the diffusion coefficients for tritium and deuterium in the molybdenum substrate. Note that for a fixed substrate equilibrium D/T number density, the neutron yield would be proportional to the incident beam current. However, this equilibrium D/T particle density in the dump is also proportional to the beam current. Therefore, the neutron yield from the beam dump is proportional to the square of the incident neutral beam current. See Refs. 15, 16, and 20 for further details.

As expected from Fig. VIII.3-5, the DD neutron yield is about two orders of magnitude smaller than the DT yield at any surface temperature. This is due to the relative magnitudes of the DD and DT cross sections for incident neutral D or T particles at these energies (50 keV at the dump surface through zero at the implantation depth).

Production of neutrons from the dump via the  $T(t,2n)^4\text{He}$  reaction was discussed above in Section VIII.3.6.1. In view of the fact that the magnitude of the TT cross section is  $\sim 0.6$  of the DD cross section at 50 keV<sup>(19)</sup> and that two neutrons are emitted per reaction, it may be expected that the TT yield for this dump is comparable to the DD yield shown in Fig. VIII.3-5. Since the TT reaction ( $Q = 11.33$  MeV) involves a 3-body breakup, the reaction kinematics for the output neutrons are complex. In addition, angular distributions for this reaction are poorly known. For this reason, the reaction is not, at present, included in the BDUMP code. Work is, however, continuing on this effect.

In the case of the high (76 keV) and low (5 keV) energy deuterium neutral beams into the barrier, there is only the DD reaction to consider. For the high energy beam, the equivalent deuteron current to each dump is only approximately half that for each of the medium (50 keV) energy beams into the

barrier coil dump. However, the cross section for the DD reaction at 76 keV is  $\sim$  twice that at 50 keV and neutrons are produced for all deuteron energies ranging from these values down to zero at the respective implantation depths. Therefore, the total DD neutron yields from the high energy neutral beam dumps will be close to those shown in Fig. VIII.3-5 for the medium energy dumps.

In the case of the low energy deuterium neutral beam, the value of the DD cross section at the energy of the incident deuterons at the surface of the dump (i.e., 5 keV) is several orders of magnitude less than the cross section at 50 keV. Neutron yields are, therefore, similarly reduced by about this factor and may be safely neglected when compared with the contribution from the main fusion plasma in the end cell.

Given the results of Fig. VIII.3-5, it is interesting to compare the resulting neutron fluxes with those stemming from the main fusion plasma in this region. The barrier coil dump will be operated with a surface temperature in the range 200-300°C. Accordingly, at the edge of the tungsten shield facing the circular beam dump, the flux of 14 MeV neutrons arising from neutral beam production within the dump is  $\sim 1.5 \times 10^7 \text{ cm}^{-2} \text{ s}^{-1}$ . To compute the equivalent flux due to the plasma, a one-dimensional neutron transport calculation was performed across the mid-plane of the barrier coil employing the discrete-ordinates code ONEDANT<sup>(17)</sup> in cylindrical geometry. At the outer surface of the normal-conducting insert coil in the barrier (an x-y displacement from the z axis comparable to that for the beam dump), integrated neutron fluxes above a neutron energy  $E_n$  were determined to be  $\sim 2 \times 10^8 \text{ cm}^{-2} \text{ s}^{-1}$  for  $E_n > 1 \text{ MeV}$  and  $\sim 6.4 \times 10^7 \text{ cm}^{-2} \text{ s}^{-1}$  neutrons per second for  $E_n > 10 \text{ MeV}$ .

It can be seen, therefore, that, in the vicinity of the dump space, the neutron production from neutral beam dumps can be within an order of magnitude

of the neutron production from the main plasma itself, especially in the case of mixed deuterium-tritium beams. In general, therefore, in the case of neutral beam dumps for deuterium and deuterium-tritium neutral beams which are situated outside of the shielding for the main fusion neutron/gamma ray fluxes, appropriate shielding precautions should be taken against DD and DT neutrons emanating from the dumps themselves.

In conclusion, the following question should be posed: Given a D or D/T neutral beam of fixed specifications (i.e., fixed energy, power, trapping fraction, etc.), how can the resulting neutron production at the beam dump(s) be minimized? One very effective way to achieve this is to operate the dump at the highest possible surface temperature consistent with heat transfer constraints. From Fig. VIII.3-5, for example, increasing the surface temperature of the medium energy beam dump from 200°C to 400°C (i.e., a factor of two) decreases the DT neutron yield by more than two orders of magnitude.

A second method of reducing the neutron yield is to select a substrate material possessing a high tritium diffusion coefficient in that temperature range envisaged for dump operation. It was shown in Eq. (VIII.3-3), that the tritium flow rate from the dump surfaces is a function of the variation of the diffusion coefficient with the temperature gradient across the dump. The magnitude of the equilibrium D, T or D/T number density  $n_j$ , on the other hand, is a strong function of the absolute value of the diffusion coefficient for the substrate (see the associated publications in Refs. 9 and 15). Therefore, from Eq. (VIII.3-4), so is the resulting neutron yield.

Figure VIII.3-6 illustrates the above dependence of neutron yield on the tritium diffusion coefficient of the dump substrate where the neutron yields from the medium energy beam dumps are shown for the cases of a molybdenum

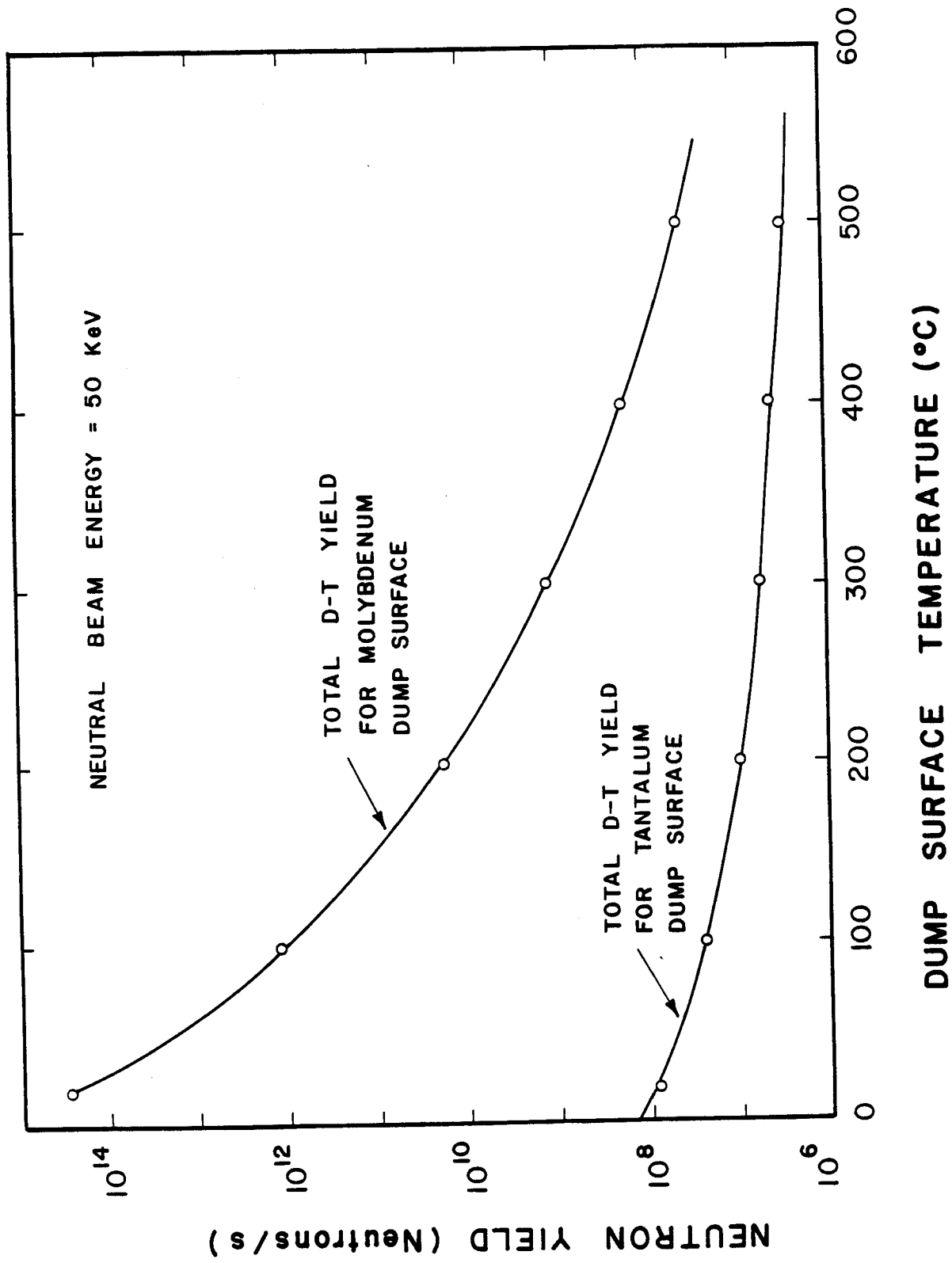


Fig. VIII.3-6 A comparison of beam-induced DT neutron production from molybdenum and tantalum dump surfaces (via the code BDUMP).

(reference design case) and tantalum dump surface. Tantalum has a very high tritium diffusivity relative to molybdenum and can be seen to result in a reduction in neutron yield of almost three orders of magnitude.

Although tantalum is also a refractory metal and thus has high heat load and stress capabilities, it was not considered to be a suitable material for the barrier coil beam dump in place of molybdenum for three reasons. Firstly, it has inferior sputtering (and thus lifetime) properties; secondly, the action of a tritium-buffer layer such as aluminum is much less successful when incorporated with a dump substrate material of high intrinsic tritium diffusivity; thirdly, there is little or no current operating experience of high heat flux dumps employing tantalum substrates. In addition, the barrier coil dump is situated in a well-shielded environment and thus high neutral-beam-induced neutron yields are of less consequence than in the case of a remote dump. An example of the latter situation can be seen in the case of the ion dumps internal to the medium energy neutral beam injectors. Here, careful thought should be given to the choice of the dump material, especially with regard to neutron production.



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#### VIII.4 Direct Converter

A single-stage plasma direct converter module has been incorporated into the TASKA design to provide direct conversion of the escaping ion kinetic energy to electricity. A direct converter is required only at one end since the ion end loss can be directed out one end. This is accomplished by lowering the potential of one plug slightly relative to the other plug. As central cell ions upscatter in energy, they escape out the lower potential plug before acquiring enough energy to escape out the other plug. The electron end loss can be directed out the end opposite to the direct converter by biasing that end wall slightly positive relative to the ground potential at the entrance of the direct converter. The main benefit of directing the electrons out the end opposite to the direct converter is that the heat load on the entrance grid of the direct converter is drastically reduced; it only has to handle the power carried by the small fraction of ions intercepted by the grid and not the entire electron power.

The obvious way to design a tandem mirror reactor with a direct converter is then to put the direct converter at one end and to use a simple thermal dump for the electron heat load at the other end. This configuration was used in the Livermore tandem mirror reactor design<sup>(1)</sup> and in WITAMIR-I.<sup>(2)</sup> A unique feature of the tandem mirror is the narrow energy distribution of the ions entering the direct converter. This is because they have fallen through the potential  $\phi_c + \phi_e$  (see Section III.2) which is many times the ion temperature in the central cell. Consequently, a single stage plasma direct converter (PDC) provides adequate efficiency.

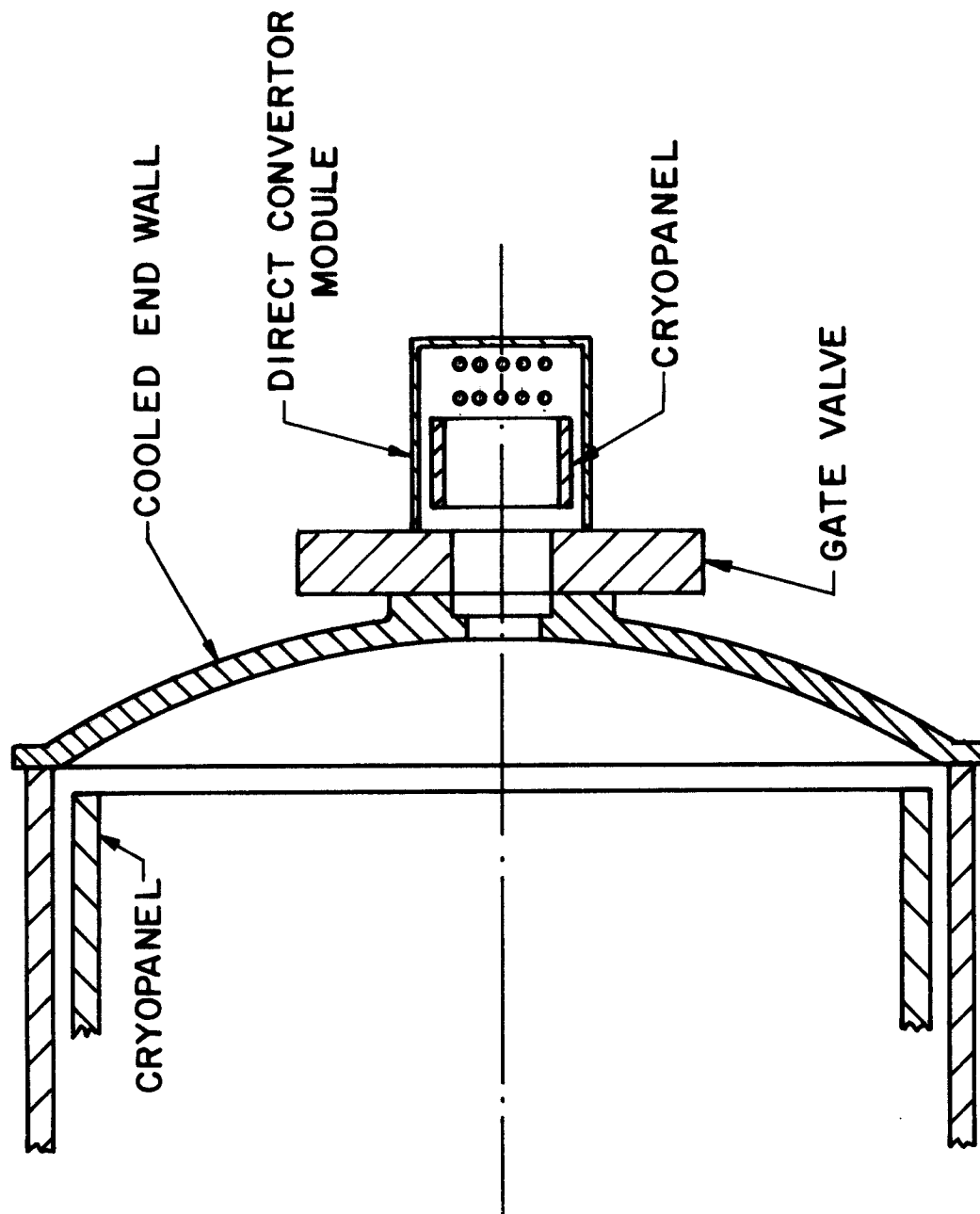
The purpose of the direct converter module is to provide testing of direct converters; the direct converter is not required for operation of TASKA.

A single module which intercepts only a fraction of the magnetic flux at the end wall is used. The module can be attached to the end wall by a gate valve and removed without disturbing the vacuum in the main chamber. When the module is removed, the port hole is covered by a cooled plate. This plate and the rest of the end wall receive a heat flux of  $100 \text{ W/cm}^2$  and must be cooled. A schematic is shown in Fig. VIII.4-1. An on-axis port allows testing the direct convertor in the radiation environment due to neutrons streaming down the axis. An off-axis port allows testing in a reduced radiation environment. The cooled plate at the same end as the direct convertor is referred to in this section as the ion thermal dump. The cooled end wall at the other end of the machine is referred to as the electron thermal dump.

A contributor to energy loss in the PDC is the high energy alpha particles. About 5% of the alpha particles are born inside the loss-cone of the central cell and immediately escape out the ends. Because their energy is much greater than the potential, they go equally out both ends. Approximately 30% of the fusion-born alphas scatter into the loss-cone while downscattering in energy; these also go equally out both ends unless one makes the peak field of the barrier at one end higher than that at the other end. Of the alpha particles entering the direct convertor, only a small amount of their energy is converted to electric power; they enter the PDC with a kinetic energy substantially greater than twice the PDC voltage.

The TASKA direct convertor is described in the following sections. The composition of the power incident on the thermal dump and entering the direct convertor is given in Section VIII.4.1. The details of the various grids and collector plates in the direct convertor are described in Section VIII.4.2.

# DIRECT CONVERTOR MODULE



VIII.4-3

Fig. VIII.4-1

Finally, a comparison of the module to the WITAMIR-I direct convertor is made in Section VIII.4.3.

#### VIII.4.1 Particle and Energy Flow to the Direct Convertor and Thermal Dumps

The basic parameters describing the central cell plasma, the end loss rates, and the electrostatic potentials are given in Section III.4. These parameters determine the species mix and energy spectrum of the particles entering the direct convertor at one end and incident on the electron and ion thermal dumps.

The central cell ions escaping through the end plug have essentially zero parallel kinetic energy as they pass over the plug potential and are accelerated by the potential ( $\phi_c + \phi_e$ ) before entering the direct convertor. They also carry with them a certain amount of perpendicular kinetic energy which is roughly  $T_{ic}$ , the central cell ion temperature. Consequently the mean energy of the central cell ions entering the PDC is 139 keV, while the minimum energy is 109 keV. This relatively narrow energy spectrum is advantageous for the direct convertor. The mean energy coupled with the ion loss rate gives a power of 78 MW approaching the ion end wall. In addition, the plug ions (protons) leave the plug with an energy of about 198 keV. This contributes another .6 MW to the power approaching the ion end wall.

The remaining ionic species are the fusion born alpha particles. These see a mirror ratio (determined by the barrier peak field and the beta-corrected central cell field) of 10. Since the alphas are born isotropically in velocity space, 5% are in the loss-cone and leave the central cell immediately with an energy of 3.5 MeV. They are then accelerated by the electric field and approach the ion end wall with an energy of 3.7 MeV, which corresponds to .4 MW of power. The remaining 95% of the alphas are magnetically

trapped in the central cell and slow down, first by electron drag and then by scattering off ions when their energy is lower. Approximately 31% are estimated to enter the loss-cone with an energy of approximately 680 keV. These alpha particles contribute 1.2 MW. One half of the alpha particles are assumed to exit at the ion end and the other half to go to the thermal dump. The remaining 64% of the alphas are electrostatically as well as magnetically contained in the central cell. They thermalize with the central cell ions and are lost axially. Because of the relatively low value of  $\phi_c/T_{ic} \approx 1.46$ , the axial confinement time of thermal alphas is sufficiently short so that their equilibrium fraction in the central cell is  $\approx 1\%$ .

The central cell electrons are confined axially by the potential  $\phi_e$ ; those electrons reaching the dump have little parallel kinetic energy and a perpendicular kinetic energy of about  $T_{ec}$ . This represents a power of 6.6 MW incident on the electron thermal dump. The plug electrons enter the central cell electron population upon upscattering in energy to above their confining potential of  $\phi_c + \phi_b$ ; this is not an additional loss from the system.

In addition to the energy carried by charged particles out the ends, energy leaves the plasma by energetic neutral atom transport, mostly in the barrier, and by synchrotron and bremsstrahlung radiation. This is estimated to contribute 9 MW. This is relevant for the blanket and shield design, but not for the direct convertor or thermal dumps, since they would receive only a minute portion of that power.

The power approaching the ion end wall is given by the sum of the central cell ion power, half the plug ion power, and half the energetic alpha power, which adds up to 81 MW. The electron thermal dump receives the central cell electron power, and the other half of the energetic alpha and plug ion power.

The total power incident on the electron thermal dump is then 8.2 MW. The electrical current approaching the ion end wall is given by the particle loss rate and charge for each species entering the PDC. The total current is 569 amps. The efficiency of the direct convertor is calculated in Section VIII.4.4. These results are summarized in Tables VIII.4-1 and VIII.4-2.

Table VIII.4-1 Input Power and Current to Ion End Wall

|                   | <u>Power, MW</u> | <u>Average Energy, keV</u> | <u>Current, A</u> |
|-------------------|------------------|----------------------------|-------------------|
| Central Cell Ions |                  |                            |                   |
| (50% D-T)         | 78.3             | 139                        | 563.              |
| Plug Ions         | .6               | 307                        | 2.1               |
| Prompt Alphas     | .4               | 3,630                      | .1                |
| Scattered Alphas  | .6               | 808                        | .7                |
| Thermal Alphas    | <u>.8</u>        | 248                        | <u>3.1</u>        |
| TOTAL             | 80.7             |                            | 569.              |

Table VIII.4-2 Input Power and Current to Electron Thermal Dump

|                        | <u>Power, MW</u> | <u>Average Energy, keV</u> | <u>Current, A</u> |
|------------------------|------------------|----------------------------|-------------------|
| Central Cell Electrons | 6.6              | 11.5                       | -576.             |
| Plug Ions              | .6               | 307                        | 2.1               |
| Prompt Alphas          | .4               | 3,630                      | .1                |
| Scattered Alphas       | <u>.6</u>        | 808                        | <u>.7</u>         |
| TOTAL                  | 8.2              |                            | -573.             |



#### VIII.4.2 Grid and Collector System

The minimum ion energy entering the direct convertor is 109 keV; hence the collector voltage relative to ground (the thermal dump) is set just below this value to collect all ions. We chose 100 kV in order to allow some loss of energy between the end plug and the collector; this could be due to scattering by neutrals or deflection by the non-planar electric fields near the grids. In front of the collector plate are the entrance grid and electron repeller grid, as shown in Fig. VIII.4-2. The potential of the ion thermal dump and entrance grid are taken to be zero; this defines the machine ground potential. The potential of the electron thermal dump at the opposite end of the machine is set to be slightly positive ( $\sim 1$  kV) in order to assure electron collection at that end. The electron repeller grid prevents low energy electrons near the entrance grid from reaching the collector plate. If they were allowed to reach the collector plate, the electrons would hit it with high energy. This would generate heat and reduce the collected current. The second grid (electron repeller grid) is set more negative ( $-9$  kV) to correct for the non-planar electrostatic potential surfaces near the grids. The system is analogous to a triode vacuum tube, as noted by Barr.<sup>(4)</sup> The repeller grid voltage is set so that the system operates in the cut-off condition; electrons reaching the entrance grid are prevented by the repeller grid from reaching the collector plate.

The grids intercept a fraction of the ion current entering the PDC. This produces a heat load which must be removed by cooling. The ion end wall also experiences the same heat load. Its cross sectional area is set so that the peak power density does not exceed  $150 \text{ W/cm}^2$ . This gives a radius of 415 cm and requires expansion of the magnetic field to 110 G at the end wall. The

## GRIDS AND COLLECTOR IN THE DIRECT CONVERTOR

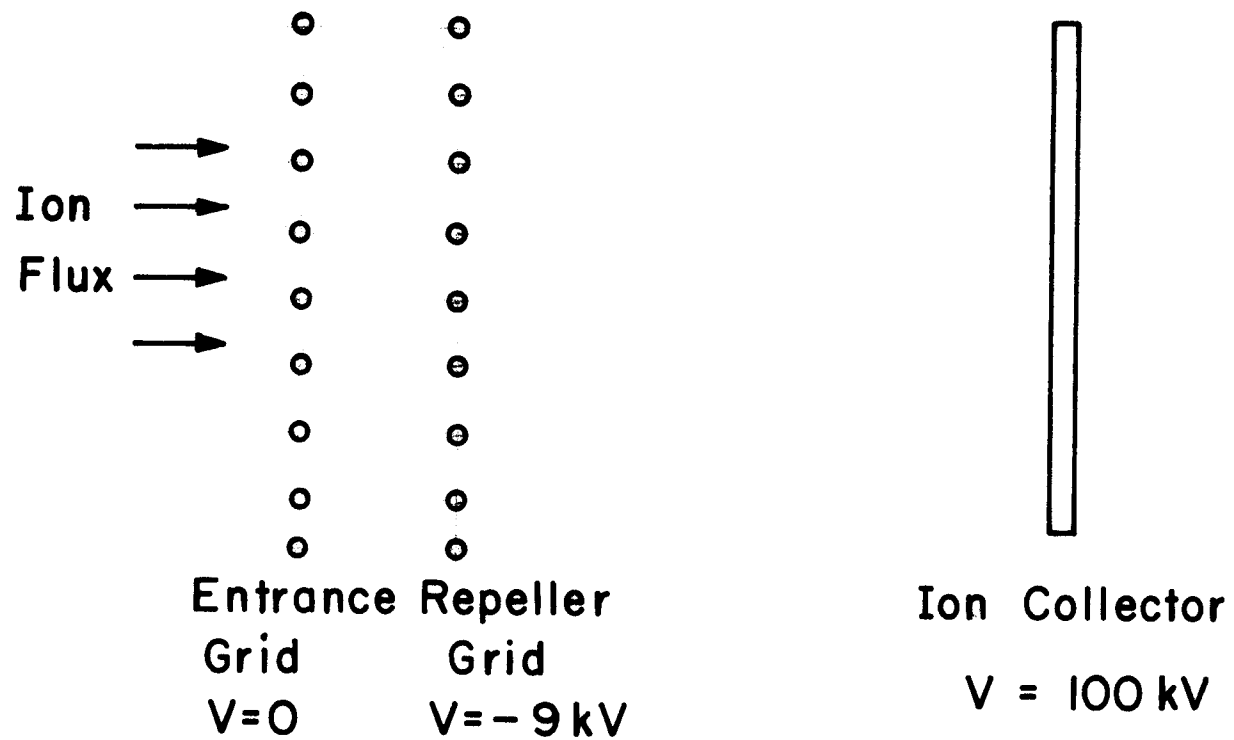


Fig. VIII.4-2

ion current density is then 10.5 amps/m<sup>2</sup>. The grid-collector spacing in the PDC is set by the requirement that the current density be less than the space-charge-limited current density,  $J_{\ell}$ , in a plane diode,<sup>(3)</sup>

$$J_{\ell} = \frac{4\epsilon_0}{9} \sqrt{\frac{2q}{m}} \frac{V^{3/2}}{d^2}, \quad (\text{VIII.4-1})$$

where  $V$  is the voltage across the diode of spacing  $d$ ,  $q$  is the charge,  $m$  is the mass, and  $\epsilon_0 = 8.85 \times 10^{-12}$  farads/meter. For the repeller grid-collector voltage, this gives a maximum  $d$  of 31 cm. We choose this spacing to be 25 cm. Equation (VIII.4-1) assumes the ions have zero speed at the collector. Since the ions approach the collector with finite kinetic energy (average = 39 keV), using Eq. (VIII.4-1) to set an upper limit on  $d$  is actually conservative.

The spacing,  $a$ , between the tubes forming the grids is chosen to be small compared with  $d$ , so that the potential surfaces are reasonably planar. We take  $a = .2 d$ ,<sup>(4)</sup> so that the spacing is 5 cm. The tube diameter then determines the transparency of the grids and hence the total heat load on the grids. We choose a tube diameter of .5 cm, which yields a transparency per grid of 90%. The radius of the PDC then determines total power incident on the PDC and the length of the grids, assuming they are not segmented in the module. For a 1 MW incident power, the module radius is 46 cm. Hence, the longest tube is ~ 100 cm, which is 200 times its diameter. This tube "aspect ratio" is similar to that in the direct convertor for a reactor. In a reactor, the tubes forming the grid are typically 1 cm in diameter and ~ 2-5 m long.<sup>(1,2)</sup>

In a full size power reactor, the grid system is segmented with each segment electrically isolated to minimize the electrical energy that can be

dissipated in a spark. The typical design value for the electrostatic energy per segment is  $\approx 10$  Joules. In the PDC module above, the total electrostatic energy is .1 Joules, so segmentation is not required.

Each grid is 90% transparent. Hence, a total of 20% of the ion current entering the PDC is intercepted by the grids. In addition, secondary electron emission at the electron repeller grid produces a beam of electrons which travel back through the plasma and strike the electron thermal dump at the other end of the reactor. Assuming three secondary electrons per incident ion, the -9 kV power supply for the electron repeller grid needs to supply 2.8 amps, or 25 kW. Of this power, 3/4 is deposited as heat at the electron thermal dump and 1/4 at the grid by ion bombardment. A total of 100 kW of heat is generated at each grid and 224 kW at the collector plate because the ions are collected with finite kinetic energy. These parameters are summarized in Table VIII.4-3. Table VIII.4-4 gives the fluxes of various energy components of ions incident on the electron and ion thermal dumps, the grids, and the collector plate. In calculating the particle flux to the electron thermal dump, it has been assumed that the radius ( $= 1.4$  m) has been chosen so that the heat load there is also  $150 \text{ W/cm}^2$ . One could place the electron thermal dump further back and increase its area to reduce the particle flux incident on it if blistering due to energetic alpha particle bombardment is a problem. The wall erosion aspects of the direct convertor and thermal dumps are discussed in Section VIII.6. The heat transfer aspects are discussed in Section VIII.5. The vacuum pumping aspects of the direct convertor and the thermal dumps are discussed in Section IV.4. The required neutral pressure is taken to be  $10^{-5}$  torr<sup>(4)</sup> in the direct convertor in order to minimize

Table VIII.4-3 Heat Generation in the PDC and Thermal Dumps

|                       |        |
|-----------------------|--------|
| Entrance Grid         | 100 kW |
| Repeller Grid         | 100 kW |
| Collector Plate       | 224 kW |
| Ion Thermal Dump      | 80 MW  |
| Electron Thermal Dump | 9 MW   |

Table VIII.4-4 Ion Energy and Particle Flux Incident  
on PDC Components and Thermal Dumps

| Particle<br>Species        | Mean<br>Energy<br>keV | Energy Spread<br>keV | Particle Flux<br>#/cm <sup>2</sup> -sec |
|----------------------------|-----------------------|----------------------|-----------------------------------------|
| Grids and Ion Thermal Dump |                       |                      |                                         |
| D                          | 139                   | 109-170              | $3.2 \times 10^{15}$                    |
| T                          | 139                   | 109-170              | $3.2 \times 10^{15}$                    |
| p                          | 307                   | 150-600              | $2.4 \times 10^{13}$                    |
| He                         | 248                   | 218-280              | $3.6 \times 10^{13}$                    |
| He                         | 808                   | 300-1,500            | $8.9 \times 10^{12}$                    |
| He                         | 3,630                 | 2,700-4,300          | $1.4 \times 10^{12}$                    |
| PDC Collector Plate        |                       |                      |                                         |
| D                          | 39                    | 9-70                 | $3.2 \times 10^{15}$                    |
| T                          | 39                    | 9-70                 | $3.2 \times 10^{15}$                    |
| p                          | 207                   | 50-500               | $2.4 \times 10^{13}$                    |
| He                         | 48                    | 18-80                | $3.6 \times 10^{13}$                    |
| He                         | 608                   | 100-1,300            | $8.9 \times 10^{12}$                    |
| He                         | 3,430                 | 2,600-4,200          | $1.4 \times 10^{12}$                    |
| Electron Thermal Dump      |                       |                      |                                         |
| p                          | 307                   | 150-600              | $2.2 \times 10^{14}$                    |
| He                         | 808                   | 300-1,500            | $8 \times 10^{13}$                      |
| He                         | 3,630                 | 2,700-4,300          | $1.2 \times 10^{13}$                    |

interactions between the escaping plasma and the neutral gas. The neutral pressure near the thermal dumps is taken to be  $5 \times 10^{-5}$  torr.

#### VIII.4.3 Comparison to a Full Scale Reactor Direct Convertor

In order to be a viable test of direct convertor technology, the module has to have characteristics similar to direct convertors for full size tandem mirror power reactors. The incident power density is  $150 \text{ W/cm}^2$ , which is half the design value for WITAMIR-I.<sup>(2)</sup> The incident DT ion flux in the module is  $6 \times 10^{15} \text{ cm}^{-2}\text{sec}^{-1}$ , which compares favorably with the WITAMIR-I value of  $4.6 \times 10^{15} \text{ cm}^{-2}\text{sec}^{-1}$ . The DT particle flux is higher in the module because the average energy of the ions approaching the PDC (139 keV) is less than in WITAMIR-I (358 keV). Consequently, one can expect a somewhat higher rate of sputtering of the grids in the module. The average energy at which the DT ions bombard the plate is the same (39 keV) in both devices.

The most significant difference with the particle-surface interactions between these two devices is the energetic helium flux to the PDC. The fractional burnup in TASKA is much less than in WITAMIR-I, but the "thermal" alpha component was assumed to be absent in the WITAMIR-I direct convertor. This is because the value of  $\phi_c/T_{ic}$  in WITAMIR-I was much higher; consequently the thermal alphas are well-confined axially in WITAMIR-I and had to be "pumped" from the central cell by some other means. In TASKA, the fusion-born alpha particles can escape axially since  $\phi_c/T_{ic} \approx 1.4$ . The fluxes for the various energy groups of alpha particles reaching the direct convertor are compared in Table VIII.4-5 for TASKA and WITAMIR-I. The module has a much lower flux of the more energetic ( $E > 600 \text{ keV}$ ) alphas, but has a substantial thermal ( $E = 250 \text{ keV}$ ) component. The total helium flux into the TASKA module

Table VIII.4-5 Comparison of Helium Fluxes at PDC Entrance

| Device  | Mean<br>Energy<br>keV | Energy Spread<br>keV | Flux<br>#/cm <sup>2</sup> -sec |
|---------|-----------------------|----------------------|--------------------------------|
| TASKA   | 248                   | 218-280              | $3.6 \times 10^{13}$           |
| WITAMIR | ---                   | ---                  | ---                            |
| TASKA   | 808                   | 300-1,500            | $8.9 \times 10^{12}$           |
| WITAMIR | 1,000                 | 650-2,000            | $4.9 \times 10^{13}$           |
| TASKA   | 3,630                 | 2,700-4,300          | $1.4 \times 10^{12}$           |
| WITAMIR | 3,950                 | 3,200-4,800          | $3.4 \times 10^{13}$           |

is about half the flux into WITAMIR-I. This will have an impact on wall erosion due to blistering.

Mechanically, the TASKA module and the WITAMIR-I PDC are similar. The tubes forming the grids in the module have a smaller diameter (1/2 cm vs. 1 cm) but similar length to diameter ratio. The feeder pipes and support structure needed to segment the WITAMIR-I PDC are not exposed to the plasma beam in the TASKA module, since it is not segmented.

The efficiency of the PDC module is computed to be 53%, which compares with the 65% value for the WITAMIR-I PDC. The sources of the various losses are somewhat different. The grid loss is higher in TASKA because of the reduced transparency, but the energetic alpha contribution to the loss is less because of the reduced fractional burnup.



In most respects the module and a reactor PDC are sufficiently similar so that the module should be a meaningful test of plasma direct convertor technology.

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## VIII.5 Ion and Electron Thermal Dumps

### VIII.5.1 Thermal Hydraulics

The power approaching the ion end wall is given by the sum of the central cell ion power, half of the plug ion power, and half of the energetic alpha power, which adds up to 81 MW. The power is distributed to an area of  $54 \text{ m}^2$ , resulting in a heat flux on the dump of  $150 \text{ W/cm}^2$ . This heat flux is picked so that forced convection water can be used as the coolant. Forced convection is more desirable than boiling heat transfer because of its larger safety margin. The thermal hydraulic and tritium diffusion aspects of the power dump are discussed in this section. The sputtering and blistering problems caused by the high energy ions are discussed in Section VIII.6.

A forced convection water stream with a velocity of 2.5 m/sec is used as the coolant of the power dump. The inlet and exit coolant temperatures are  $60^\circ$  and  $110^\circ\text{C}$ , respectively. With a coolant tube diameter of 0.5 cm, a coolant heat transfer coefficient of  $1.6 \text{ watt/cm}^2 \text{ }^\circ\text{C}$  is available. The film temperature drop is  $90^\circ\text{C}$  for a heat flux of  $150 \text{ watt/cm}^2$ . The front tube wall is .4 cm thick, which results in a temperature difference across the wall of  $346^\circ\text{C}$ . Therefore, at the exit end, the coolant wall temperature is

$$T = 865 x + 473^\circ\text{K} \quad (\text{VIII.5-1})$$

in which  $x$  is the distance from front surface. This temperature equation will be useful in the tritium diffusion calculations.

The total thermal power to the power dump is 81 MW. The coolant flow rate is, therefore,  $1.4 \times 10^6 \text{ kg/hr}$ . The coolant flow length is 1 m. The pressure drop is .04 MPa. The pumping power required is 1 MW. The important parameters of the power dump are listed in Table VIII.5-1.

Table VIII.5-1 Thermal Hydraulic Parameters for Power Dump

|                          |                             |
|--------------------------|-----------------------------|
| Total Power              | 81 MW                       |
| Energy Flux              | 150 W/cm <sup>2</sup>       |
| Coolant                  | H <sub>2</sub> O            |
| Structural Material      | 316 SS                      |
| Coolant Tube Diameter    | .5 cm                       |
| Coolant Tube Length      | 1 m                         |
| Coolant Velocity         | 2.5 m/sec                   |
| Coolant Inlet Temp.      | 60°C                        |
| Coolant Exit Temp.       | 110°C                       |
| Coolant Flow Rate        | 1.4 x 10 <sup>6</sup> kg/hr |
| Coolant Pumping Power    | 1 MW                        |
| Maximum Structural Temp. | 546°C                       |
| Minimum Structural Temp. | 60°C                        |

#### VIII.5.2 Tritium Considerations

High energy tritium ions impinge on the surface of the power dump area. The tritium will penetrate to a certain distance,  $\delta$ , from the front surface and will either diffuse backward to the vacuum cavity or forward to the coolant. Since water is used as the coolant, it is important to calculate the tritium leakage rate to the coolant.

The penetration depth of 139 keV tritium is calculated to be  $.4 \times 10^{-4}$  cm. It can be assumed that the tritium is deposited uniformly within this depth. It is further assumed that the back diffusion rate is much higher than

the forward diffusion rate. Therefore, the back diffusion can be calculated by a stationary diffusion system with uniform mass deposition, or

$$C(x) = \frac{\dot{M}}{2D} [\delta^2 - (\delta-x)^2] \quad (\text{VIII.5-2})$$

$$\text{and } \dot{M} = m/\delta \quad (\text{VIII.5-3})$$

in which C is concentration, g/cm<sup>3</sup>

$\dot{M}$  is mass generation rate, g/cm<sup>3</sup>-sec

D is diffusivity, cm<sup>2</sup>/sec

$\delta$  is the penetration depth, cm

x is distance from the front wall, cm

m is the particle flux, g/cm<sup>2</sup>-sec

Diffusivity of hydrogen in 304 SS is reported to be<sup>(1)</sup>

$$D = 1.8 \times 10^{-2} e^{(-14000/RT)} \quad (\text{VIII.5-4})$$

Within the penetration depth of  $\delta$ , the temperature is almost constant and has been calculated to be 546°. The incoming tritium particle flux is  $3.2 \times 10^{15}$ /sec-cm<sup>2</sup>. The maximum tritium concentration at  $x = \delta$  is, therefore,  $10^{-7}$  g/cm<sup>3</sup>.

The tritium concentration beyond  $x = \delta$  can be calculated by the solution for an infinite plate facing a constant concentration in front. The mass transfer can be calculated by

$$G = - D \frac{dc}{dx} \quad (\text{VIII.5-5})$$

or

$$\frac{C(o) - C(x_o)}{G} = \int_0^{x_o} \frac{dx}{D} \quad (\text{VIII.5-6})$$

in which  $G$  is the mass transfer rate,  $\text{g/cm}^2 \text{ sec}$ .

The temperature profile is

$$T = 865 x + 473 \text{ } ^\circ\text{K} \quad (\text{VIII.5-7})$$

and

$$D = 1.8 \times 10^{-2} e^{-14000/RT} . \quad (\text{VIII.5-8})$$

The left side of eq. VIII.5-6 can be integrated numerically from 0 to  $x_0 = .4$  and equals  $6.72 \times 10^6 \text{ sec/cm}$ . The mass transfer rate can then be calculated by assuming  $c(x_0) = 0$  and  $C_0 = 10^{-7} \text{ g/cm}^3$ :

$$\begin{aligned} G &\approx \frac{C_0}{6.72 \times 10^6} \\ &= 1.5 \times 10^{-14} \text{ g/cm}^2 \text{ sec} . \end{aligned}$$

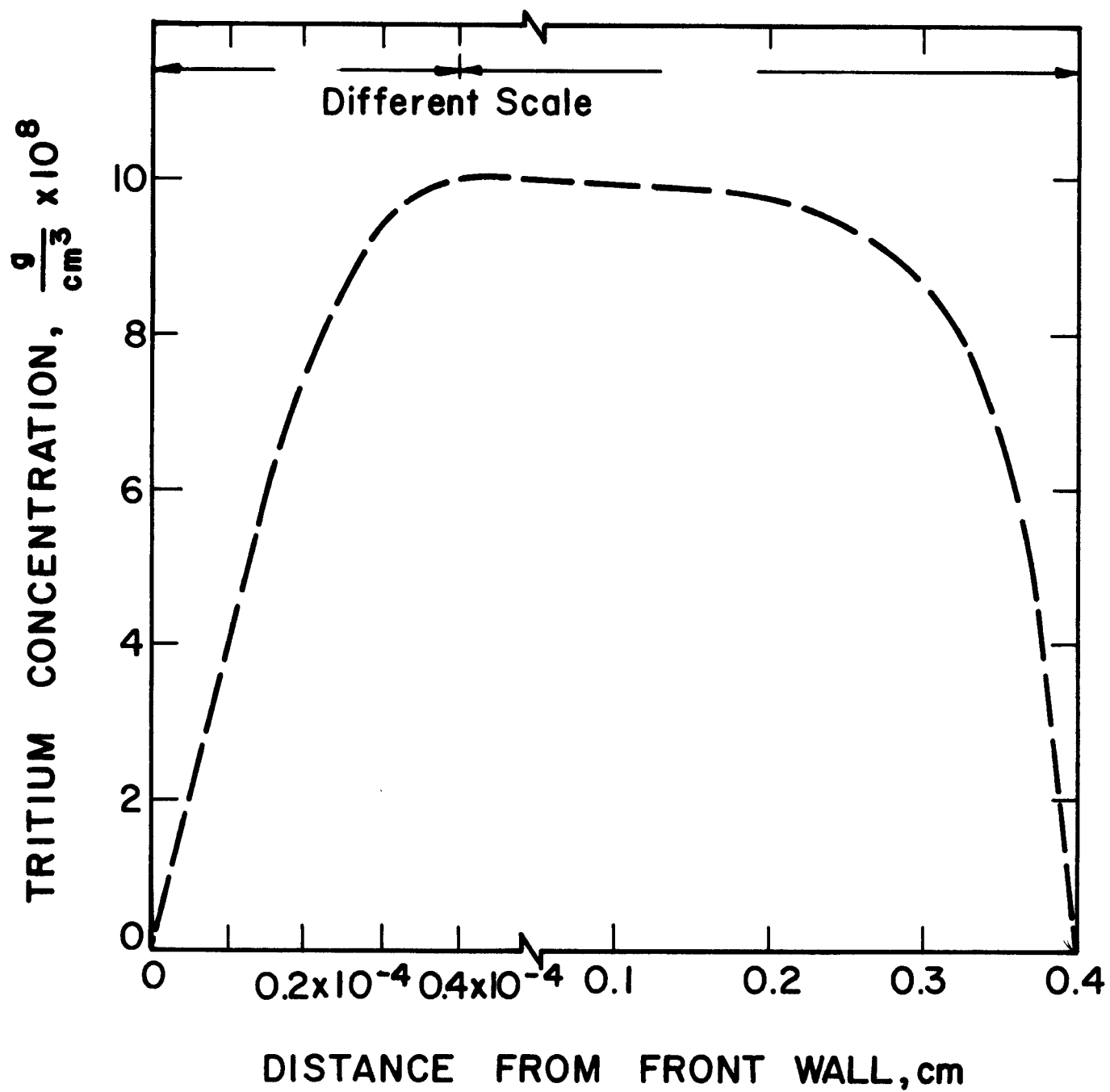
The total dump area is  $54 \text{ m}^2$ . Therefore, the total tritium leakage rate to the coolant is 7 curies/day. With any effect of oxide layers as a diffusion barrier, the tritium leakage rate can be reduced to  $<1 \text{ curie/day}$ .

The integration of  $\int_0^{x_0} \frac{dx}{D}$  can be transferred into a summation of  $\sum_{i=1}^{10} \frac{\Delta x}{D(\bar{T})}$ . Each term in the summation indicates the resistance to the diffusion of tritium for this particular zone. The magnitude of the resistances are shown on Table VIII.5-2. It is clear that the last zone provides almost 2/3 of the total resistance. The tritium concentration in the wall is shown on Fig. VIII.5-1. The total tritium inventory in the structure is less than .05 g.

Table VIII.5-2 Resistance to Tritium Diffusion from Each Zone

| Zone No. | Resistance $[\frac{1}{D(\bar{r})}]$ , $\frac{\text{sec}}{\text{cm}^2} \times 10^{-5}$ |
|----------|---------------------------------------------------------------------------------------|
| 1        | 3.64                                                                                  |
| 2        | 5.42                                                                                  |
| 3        | 8.36                                                                                  |
| 4        | 13.47                                                                                 |
| 5        | 22.80                                                                                 |
| 6        | 40.91                                                                                 |
| 7        | 78.58                                                                                 |
| 8        | 163.61                                                                                |
| 9        | 375.25                                                                                |
| 10       | 967.62                                                                                |

Fig. VIII -5 -1 TRITIUM CONCENTRATION IN  
POWER DUMP WALL



References for Section VIII.5

1. H. K. Perkins et al., "Tritium Hold Up Due to Coatings on the First Wall of Fusion Reactors," Radiation Effects and Tritium Technology for Fusion Reactors, Gatlinburg, TN (Oct. 1975).



## VIII.6 Surface Erosion on Components of the Plasma Direct Convertor and Thermal Dump

### VIII.6.1 Introductory Remarks

The energies carried by plasma particles which impinge on the components of the plasma direct convertor (PDC) are in the range where both sputtering and blistering are of utmost concern. Blistering, if it occurs repeatedly, can lead to particularly high erosion rates. Whether or not repetitive blistering is possible, depends on a number of factors which have only recently been recognized. These factors are of central importance in our following assessment of surface erosion, and they will therefore be briefly reviewed.

There are four basic conditions for blistering or flaking to occur as a result of ion implantation in solids:

- A critical concentration of the implanted gas must be reached within the solid of the order of 10 to 50% of the metal atom density, depending on temperature and material (see Fig. VIII.6-1).
- Erosion by sputtering must not remove more material during the time to reach a critical gas concentration than the ion range.
- The depth distribution of the implanted gas must be such that the critical concentration is first reached below the surface rather than at the surface.
- The creep strength of the material must be sufficiently high to contain within small bubbles a gas pressure of the order of  $10^{-3}$  \* shear modulus of the solid.

The last two conditions provide ways to avoid blistering altogether. The surface temperature of the solid must be kept above a temperature of about half the melting point.<sup>(1)</sup> In this case, only blister holes form, and the gas

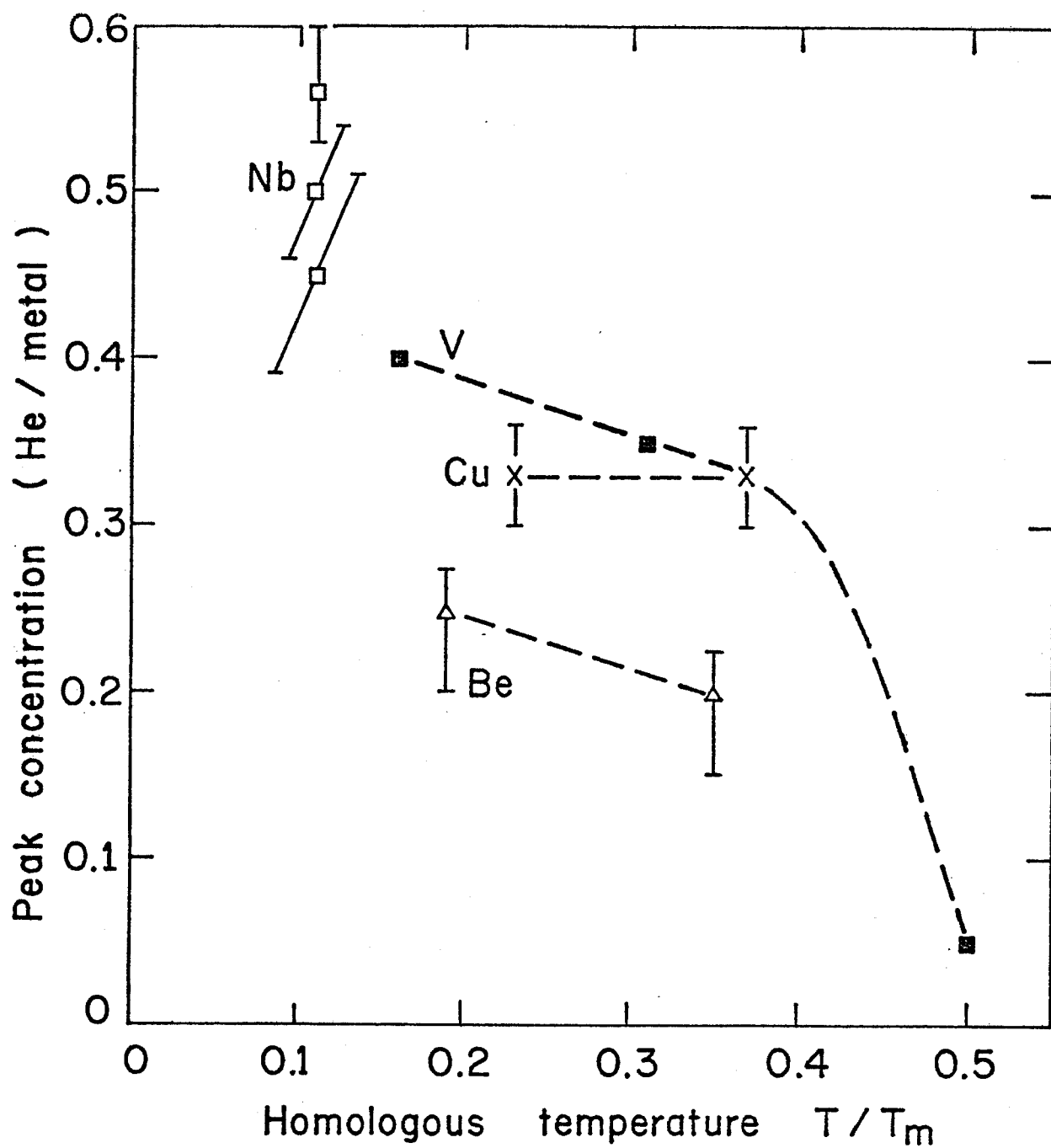


Fig. VIII.6-1. Critical helium concentration for blistering.

is released without further erosion. Or, the other strategy is to implant the solid with ions having a sufficiently wide spectrum of energies that a broad gas distribution is obtained in the solid which intersects the surface. For this strategy it must be noted that sputtering may erode a sufficient amount of material so that the critical concentration is first reached at the receding surface.

Terreault<sup>(2)</sup> has recently studied the repeated appearance of blistering or flaking upon monoenergetic helium bombardment in greater detail and derived the following empirical rules. If  $R_p$  is the projected range of ions in the solid, and FWHM the full width at half maximum of the implantation concentration profile, then

- repetitive blistering or flaking occurs when  $\text{FWHM}/R_p < 0.7$ ;
- blistering gradually disappears when  $\text{FWHM}/R_p$  is between 0.7 and 1.0;
- blistering occurs only once when  $\text{FWHM}/R_p \gtrsim 1.0$ ;
- blistering is likely to happen not at all when  $\text{FWHM}/R_p \gtrsim 1.5$ .

Figure VIII.6-2 from Ref. (2) shows the range of implantation energies and atomic numbers for which the various blistering processes take place.

Multiple energy implantations of helium ions into nickel and stainless steel have recently been carried out by Jaeger and Roth<sup>(3)</sup> in order to investigate the reasons why blistering does not occur. They found that a continuous distribution of small helium bubbles is formed under multiple energy implantations which reaches from the (receding) surface into the material. Upon reaching the critical gas concentration, interbubble fracture and bubble coalescence lead to an interconnected porosity up to the surface which provides microscopic channels for gas release. By implication, it can then be concluded that for monoenergetic ion implantation at sufficiently high

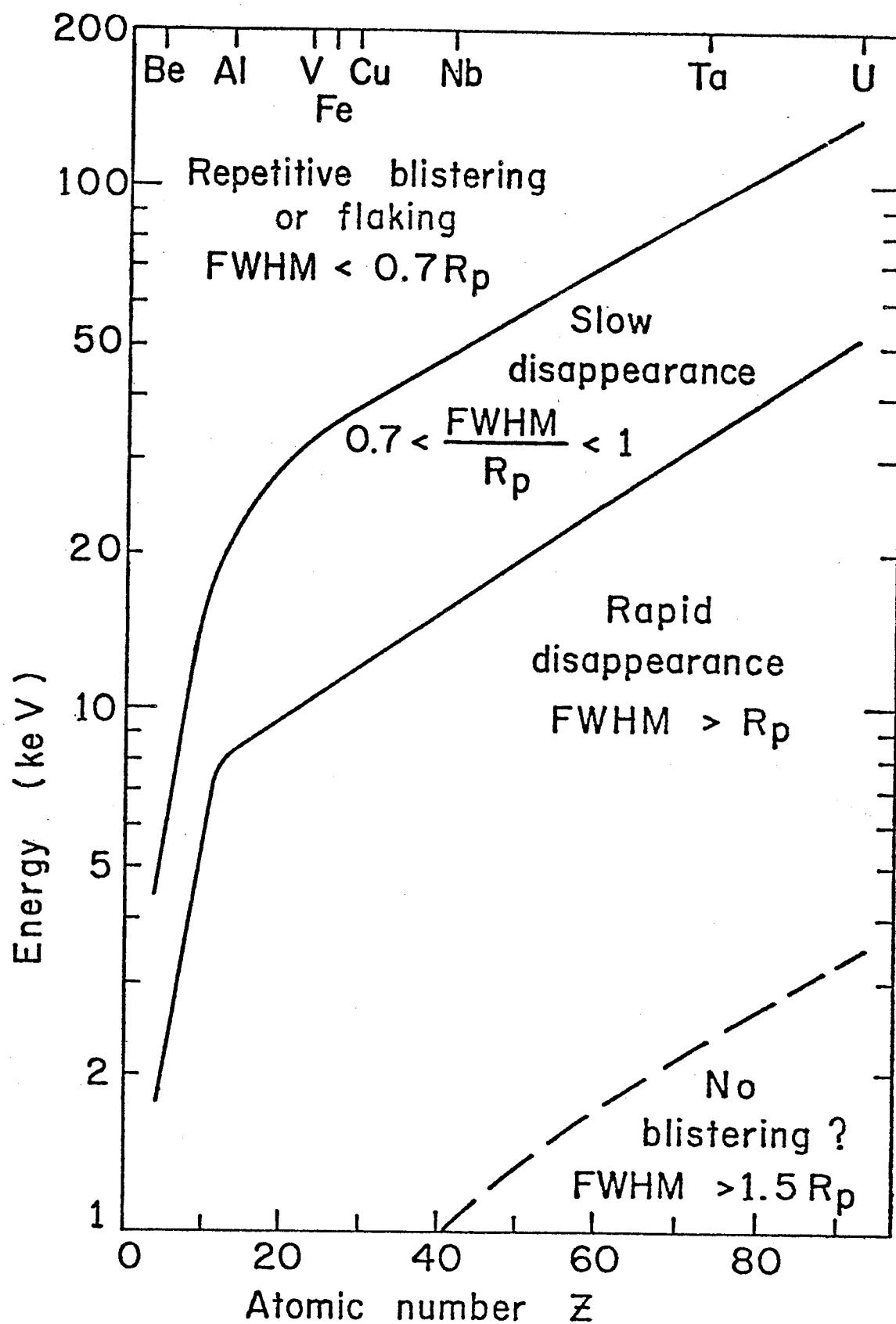


Fig. VIII.6-2. Blistering conditions.

energies such that  $\text{FWHM}/R_p < 0.7$ , the interconnected porosity does not reach the surface, and blistering or flaking will ensue.

#### VIII.6.2 Concentration Profiles of Implanted Ions

The collector plate, grids, and ion thermal dumps are exposed to ions which can be divided in six different groups. Each group is characterized by a range of energies listed in Tables VIII.6-1 and VIII.6-2. Their implantation ranges covers therefore certain depth intervals which were obtained from the compilation in Refs. (4,5) and which are also listed in Tables VIII.6-1 and VIII.6-2 for stainless steel. If we assume a uniform energy distribution within each group, the concentration of implanted ions within the corresponding depth interval will build up uniformly at rates given by the last columns in Tables VIII.6-1 and VIII.6-2. Figures VIII.6-3 and VIII.6-4 show the depth distribution of ions for both helium and hydrogen, assuming that no diffusion occurs. This assumption is certainly incorrect for hydrogen at elevated temperatures, but it does apply to helium trapped in vacancy clusters produced in connection with the ion bombardment.

In order to assess the possibility of blistering we assume that a critical concentration for implanted helium of 20% of the metal atom density must be reached.

For the case of the collector plate, the low energy group (18-80 keV) of the helium ions accumulates at the fastest rate, and a critical concentration is reached in about 1415 seconds.

However, both the low energy group of the helium ions and the flux of D and T give rise to sputtering erosion. Using an average sputtering yield of 0.03 for D and T, and a sputtering yield of 0.1 for the helium ions with energy between 18 and 80 keV, we find a sputtering erosion distance of  $0.0325 \mu\text{m}$

Table VIII.6-1. Flux of Particles, Range Distribution,  
and Rate of Gas Concentration Buildup in the Collector Plate

| Particle Species | Energy Spread (keV) | Flux $\text{cm}^{-2}\text{s}^{-1}$ | Depth Range ( $\mu\text{m}$ )<br>Without/With Straggling | Rate of Gas Buildup<br>(atom fraction/sec) |
|------------------|---------------------|------------------------------------|----------------------------------------------------------|--------------------------------------------|
| D                | 9-70                | $3.2 \times 10^{15}$               | 0.065-0.4                                                | } $2.25 \times 10^{-3}$                    |
| T                | 9-70                | $3.2 \times 10^{15}$               | 0.065-0.4                                                |                                            |
| H                | 50-500              | $2.4 \times 10^{13}$               | 0.27 -2.5                                                | $1.27 \times 10^{-6}$                      |
| He               | 18-80               | $3.6 \times 10^{13}$               | 0.09 -0.3                                                | $2.02 \times 10^{-5}$                      |
| He               | 100-1300            | $8.9 \times 10^{12}$               | 0.2 -2.0                                                 | $5.83 \times 10^{-7}$                      |
| He               | 2600-4200           | $1.4 \times 10^{12}$               | 4.4 -8.0                                                 | $4.58 \times 10^{-8}$                      |

Table VIII.6-2. Flux of Particles, Range Distribution, and Rate of Concentration Buildup in Grids and Ion Thermal Dumps

| Particle Species | Energy Spread (keV) | Flux $\text{cm}^{-2}\text{s}^{-1}$ | Depth Range ( $\mu\text{m}$ )<br>Without/With Straggling | Rate of Gas Buildup<br>(atom fraction/sec) |
|------------------|---------------------|------------------------------------|----------------------------------------------------------|--------------------------------------------|
| D                | 109-170             | $3.2 \times 10^{15}$               | 0.6 -0.8                                                 | } $3.77 \times 10^{-3}$                    |
| T                | 109-170             | $3.2 \times 10^{15}$               | 0.6 -0.8                                                 |                                            |
| H                | 150-600             | $2.4 \times 10^{13}$               | 0.65-2.5                                                 | $1.53 \times 10^{-6}$                      |
| He               | 218-280             | $3.6 \times 10^{13}$               | 0.6 -0.7                                                 | $4.24 \times 10^{-5}$                      |
| He               | 300-1500            | $8.9 \times 10^{12}$               | 0.7 -2.5                                                 | $5.83 \times 10^{-7}$                      |
| He               | 2700-4300           | $1.4 \times 10^{12}$               | 4.5 -8.0                                                 | $4.72 \times 10^{-8}$                      |
|                  |                     |                                    | 4.4 -8.2                                                 |                                            |

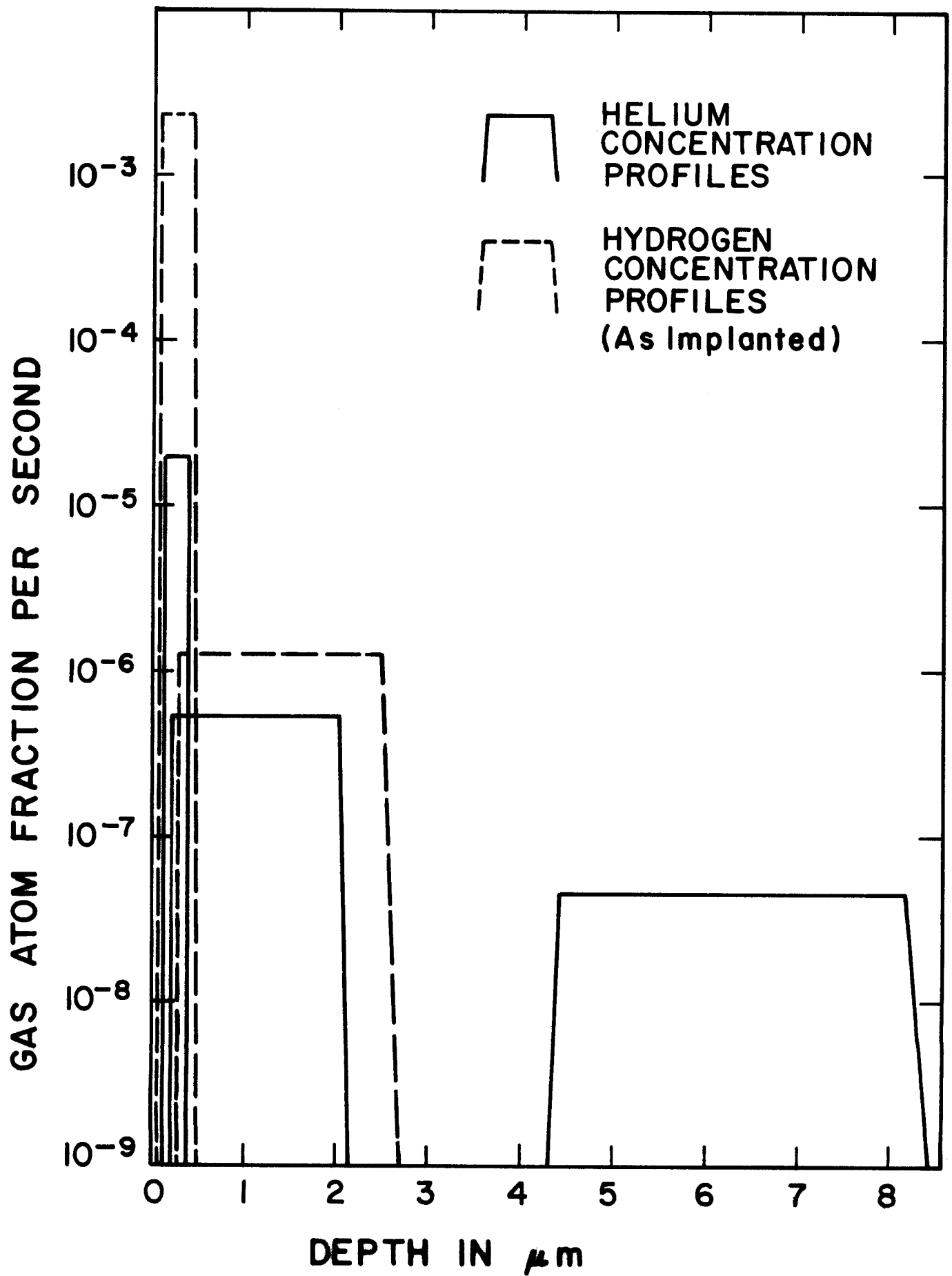


Fig. VIII.6-3. Depth distribution of implanted ions in the collector plate.



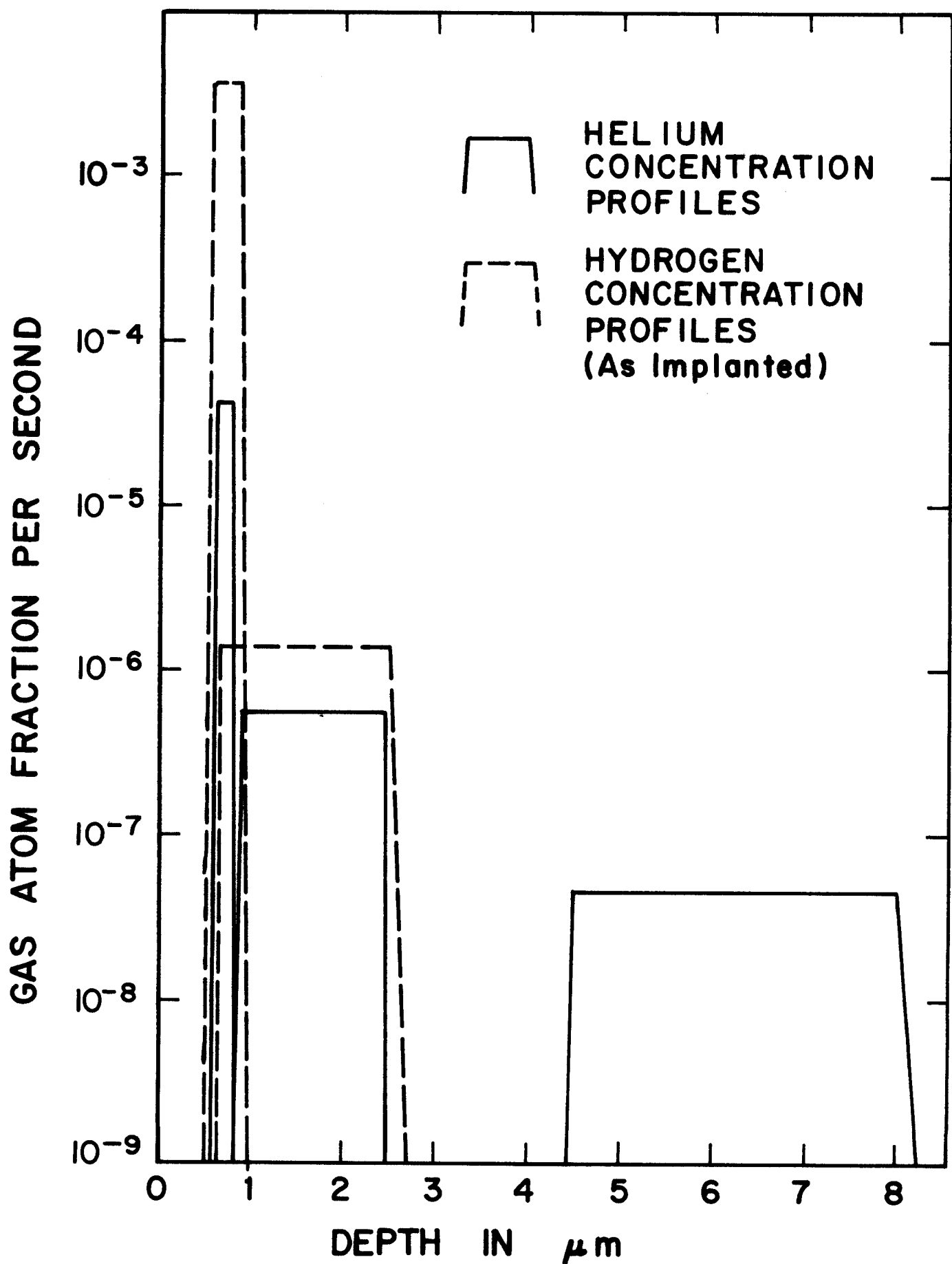


Fig. VIII.6-4. Depth distribution of implanted ions in grids and thermal dumps.

within the time interval of 1415 seconds. Since the helium depth distribution starts now at a distance of only  $0.0275\text{ }\mu\text{m}$  below the surface (taking straggling into account) blistering is not expected to occur, or at most only once.

The second energy group of implanted helium (ranging from 100 to 1300 keV) overlaps with the first one, and reaches its critical concentration after the first one. Therefore, an interconnected network of pores will already exist for continuous helium release for this second group by the time it has reached the critical concentration. Hence, the second group will not lead to blistering either.

The third group, whose implantation depth extends initially from 4.3 to  $8.2\text{ }\mu\text{m}$ , would require an implantation time of  $4.8 \times 10^6$  seconds to reach a critical concentration. By that time, however, sputter erosion by the flux of D and T will have resulted in the removal of about  $100\text{ }\mu\text{m}$  of material, i.e. many times more than the implantation depth of the high energy  $\alpha$ -particles. Consequently, the critical concentration is never reached.

In conclusion, blistering or flaking by the implantation of helium into the collector plate will not occur. Instead, sputter erosion from the low energy flux of D and T is the dominant erosion mechanism, resulting in a total erosion rate of about  $0.73\text{ mm/FPY}$ .

The situation with regard to erosion of the grids and thermal dumps for the PDC is somewhat different. As seen from Table VIII.6-2, the six different groups have energy spreads larger by about 100 keV. As a result, the implantation range for the low-energy D and T particles extends now from 0.45 and  $0.95\text{ }\mu\text{m}$ .

The low energy group of helium ions is implanted between 0.6 and  $0.7\text{ }\mu\text{m}$ , in which range a critical concentration of 20 at.% is reached within a time of

$4.7 \times 10^3$  seconds. The thickness of the eroded surface layer by sputtering with the D and T particles is only  $0.07 \mu\text{m}$  for this time interval, and therefore insufficient to expose the layer of helium saturated material. As a result, blistering will occur, removing about  $0.7 \mu\text{m}$ .

The medium energy group of helium ions, whose implantation depth extended originally from  $0.6$  to  $2.6 \mu\text{m}$ , has accumulated to a concentration of only  $0.27$  at.%, a negligible amount compared to the  $20$  at.% at saturation. Therefore, after the first blister generation, the low-energy group of helium ions will rapidly build up a second implantation profile, leading to the second blister generation after an additional time of  $4.7 \times 10^3$  seconds.

Repetitive blistering will continue up to six generations or a depth corresponding to the original range of the high-energy group of  $\alpha$ -particles. When this implantation profile is exposed, helium has been accumulated up to about  $0.13$  at.%. Again, this is negligible in comparison to  $20$  at.%, required to create the interconnected porosity necessary for gas release. It must therefore be concluded that repetitive blister erosion by the low-energy group of helium ions will occur, unless other remedial actions are taken.

As discussed above, there are two possibilities. The PDC could be designed such that the grids and the thermal dumps have surface temperatures in excess of  $0.5 T_m$  ( $T_m$  is the melting temperature of the material in K) on all surfaces exposed to the particle flux.

The second choice is to precondition these components by implanting helium at energies ranging from  $10$  to  $100$  keV up to the critical concentration prior to using these components in the TASKA reactor. It may be possible to precondition a second set of grids first at the collector plate and then exchange them with the first set of grids. These preconditioned components

would release helium upon further bombardment in the same manner as it is released from the collector plate. The maximum rate of erosion would then be by sputtering, i.e. at a rate of 0.73 mm/FPY.

The electron thermal dump is bombarded by three groups of particles whose fluxes and depth distributions are listed in Table VIII.6-3. The medium energy helium flux reaches its critical implantation concentration after a time of  $3.8 \times 10^4$  s. The sputtering erosion by both the hydrogen and the helium flux is estimated to be about  $3 \times 10^{-7}$  cm, i.e. much shorter than the implantation depth. As a result, blistering by the medium energy helium flux is again the dominant erosion mechanism unless the electron thermal dump is preconditioned or run at a surface temperature of about half the melting point or higher.

It should be noted that in the present assessment of blistering, it was assumed that the energy spectrum in each particle group is uniform. If, however, the spectrum is peaked at the low energies of the energy spread, the critical concentration would first be reached at the minimum range for that group. For example, if the medium energy helium flux is peaked at 300 keV, the blister thickness would be 0.7  $\mu\text{m}$ , leaving a depth region of 1.3  $\mu\text{m}$  where the helium concentration is high. After the first generation of blisters have been produced, the subsequent helium concentration buildup in the underlying material would be a superposition of the remainder of the old distribution plus the new one. The total distribution reaches now from the new surface into the material, and the next critical concentration may very well be produced at the surface. In this case, no further blistering would occur.

A detailed analysis of both the energy spectra of the plasma particle fluxes and the accumulated depth distributions of helium will be required to obtain a more reliable assessment of blistering. It appears that repetitive

Table VIII.6-3. Flux of Particles, Range Distribution,  
and Rate of Concentration Buildup in the Electron Thermal Dump

| Particle Species | Energy Spread (keV) | Flux $\text{cm}^{-2}\text{s}^{-1}$ | Depth Range ( $\mu\text{m}$ )<br>Without/With Straggling | Rate of Gas Buildup<br>(atom fraction/sec) |
|------------------|---------------------|------------------------------------|----------------------------------------------------------|--------------------------------------------|
| H                | 150-600             | $2.2 \times 10^{14}$               | 0.65-2.5    0.55-2.7                                     | $1.40 \times 10^{-5}$                      |
| He               | 300-1,500           | $8.0 \times 10^{13}$               | 0.7 -2.5    0.6 -2.6                                     | $5.24 \times 10^{-6}$                      |
| He               | 2,700-4,300         | $1.2 \times 10^{13}$               | 4.5 -8.0    4.4 -8.2                                     | $4.05 \times 10^{-7}$                      |

blistering can be avoided in a more detailed analysis or with proper tailoring of the energy spectra of the impinging plasma particles.

#### References for Section VIII.6

1. S.K. Das and M. Kaminsky, "Radiation Blistering in Metals and Alloys," Chapter 5, in Radiation Effects on Solid Surfaces, ed. by M. Kaminsky, American Chem. Soc., (1976).
2. B. Terreault, J. Nucl. Matls. 93&94, (1980), 707-712.
3. W. Jaeger and J. Roth, J. Nucl. Matls. 93&94, (1980), 756-766.
4. H.H. Andersen, J.F. Ziegler, "Hydrogen, Stopping Powers and Ranges in all Elements," Pergamon Press, New York, (1977).
5. J.F. Ziegler, "Helium, Stopping Powers and Ranges in all Elements," Pergamon Press, New York, (1977).

### VIII.7 End Plug Coils Arrangement

The end plug coils are composed of two transition coils and a pair of yin-yang coils. All are superconducting coils and therefore inserted in a dewar vessel.

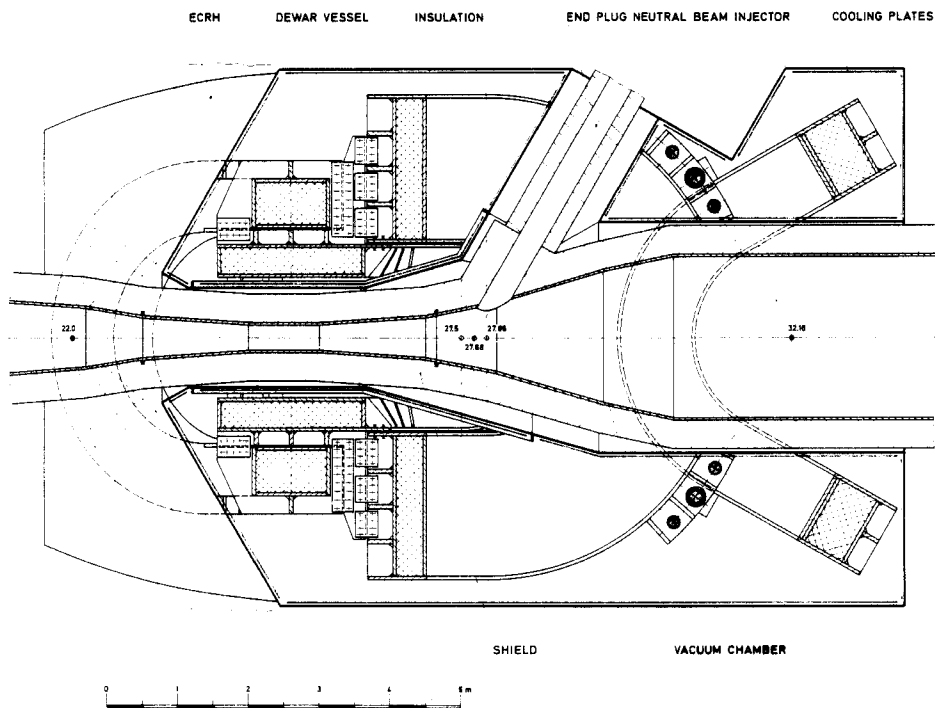
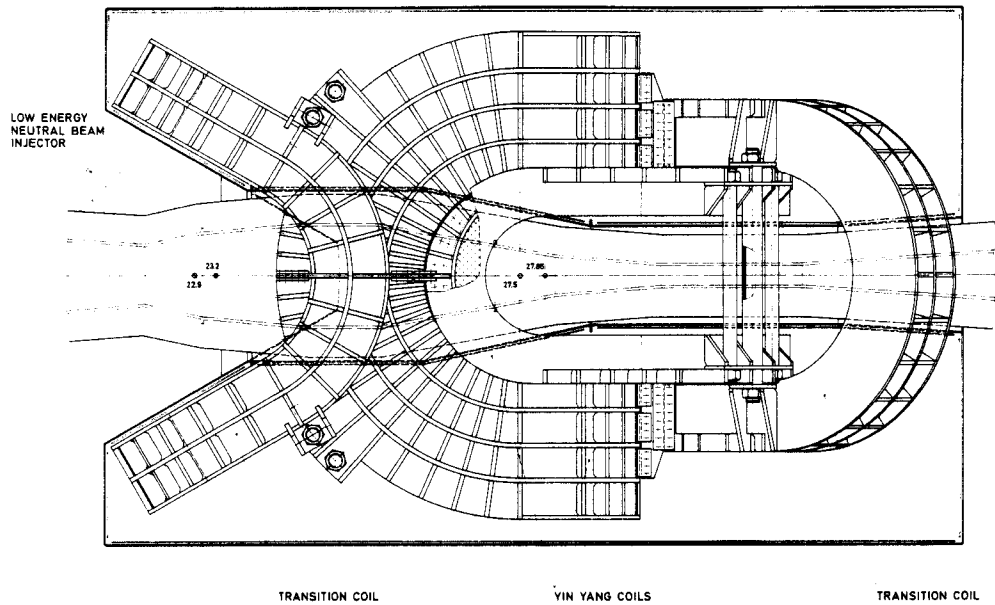
As can be seen in Fig. VIII.7-1, the middle of the first transition coil is arranged at a distance of 23.2 m from the center of the test facility. The yin-yang coils have no common coil center and are located at the 27.5 and 27.86 m points; the outer transition coil is located at 32.16 m. Also located in the end plug region are the ECRH units and the end plug neutral beam injector, 22.0 m and 27.68 m distant from the center of the test facility center. Figure VIII.7-1 shows also that all coils are surrounded by a single common dewar vessel. The reasons for this concept are:

- the proximity of the magnets to each other and to the vacuum chamber which in addition is enclosed by the shield;
- a common dewar vessel renders possible a favorable support structure of the coils in relation to the magnetic forces acting upon the particular magnets;
- the cold support structure between the coils eliminates heat transfer from supports external to the dewar.

The dewar vessel design pays regard to assembling and dismantling problems of coils and heating units. The difficulties result from the special shape of the plasma and the shape of the vacuum chamber and shield as well as the lack of space between the shield and the magnets. Therefore the vessel is composed of three obliquely divided parts welded and clamped together. Moreover, the inner and outer shells of the dewar vessel are divided in the longitudinal direction too. Channels for the neutral beam injector ducts, the microwave transport system, and the end parts of the vessel connect the outer

Fig. VIII.7-1

End plug coils -- arrangement.



VIII.7-2



and inner shell, while the ECRH gyrotrons and reflectors are located outside and do not penetrate it. Insulation is attached to the inner side of the shell and cooling plates with liquid nitrogen for cooling are arranged between the magnet support structure and the shell. The pressure differential across the vessel is assumed to be 760 torr. This determines the vessel thickness of 2 cm, under the assumption that the shell is reinforced by fins. The dewar vessel material was chosen to be austenitic steel.

To support the end plug unit a special support structure, moveable on rails, will be used. The necessary thermal insulation of the unit resting on the structure will be obtained by application of carbon fiber synthetic resin. The assembly of the vacuum chamber, magnets and dewar vessel has to be performed in one assembly schedule, because a subsequent mounting as well as a later displacement of components along the chamber axis will be prevented by the chamber shape.

The maintenance and replacement of end plug components have been checked. The problems are solvable but further design investigations will be necessary.

## IX Operations

### IX.1 Plasma Startup

#### IX.1.1 Introduction

The discussion of the plasma physics aspect of TASKA in Chapter III has been concerned with the steady state operating condition. Obviously, consideration has to be given to the problem of initiating the plasma and, by heating and fueling, bringing it to the desired operating condition. The startup sequence in tandem mirror experiments is to inject into the plugs a plasma stream from plasma guns in the end wall. The plug neutral beams are then turned on and the plug plasma is built up to its required density and mean ion energy. Gas is then injected into the central cell to build up the central cell density.

The startup sequence in a barrier tandem mirror is complicated by the additional sources from barrier pumping, and ECRH, and perhaps ICRF heating. Electron and ion power losses from the central cell or plug plasma scale as the appropriate collision frequency which, in turn, scales as  $n/T^{1.5}$ , where  $n$  and  $T$  represent the appropriate density and temperature. This suggests that, if one tries to heat the plasma at the full density, the power loss is greater when the plasma is cold. It takes more power to heat the plasma than to sustain it at the final temperature. The power sources, however, have been sized for the required steady state loads. It is expensive to have additional power available that is only required for startup. The solution to this is to start with a low density plasma and raise the density while heating the plasma in such a way as to not exceed the power available.

An additional complication arises from the variation of the neutral beam trapping efficiency with density. At too low a density, the beam is poorly

trapped and the plasma heats too slowly. Furthermore, the plug mirror ratio varies with beta, which affects the hot ion confinement time; this effect is stronger for a sloshing ion plug than for 90° injection. In addition, the absorption of the ECRH power is poor (for the 0-mode) for electron temperatures less than a few keV. One could use the X-mode to heat the electrons at low  $T_e$ , but switching modes as the plasma heats complicates the ECRH delivery system. One possible scenario is to heat the plug plasma at moderate density with the neutral beam until the electron temperature gets to a few keV. One then applies ECRH to heat the electrons; whether one can simultaneously control the plug ion density in the appropriate manner is not yet established. Once a sufficiently hot and dense plug plasma is established, the central cell plasma can be built up by gas injection into the central cell; the hot ion plasma leaking out of the plug would act as a "seed" for ionization in the central cell. While the central cell plasma is building up, beam injection into the barrier region to establish the thermal barrier would be initiated.

We have developed a rate code to study various scenarios for plasma startup. This code solves the time dependent rate equations for the central cell and plug plasmas and includes the effect of the thermal barrier. The code is described in Section IX.1.3 and some results are given in Section IX.1.3.3.

#### IX.1.2 Initial Target Plasma

Each time TASKA is restarted after an interruption in operation, a startup sequence will need to be followed. One possible startup scenario is given here; other approaches are possible and may have advantages for barrier formation using ECRH.

Step 1. The machine is assumed to be under good vacuum and the magnetic field magnets have been energized. If the plasma has been opened to contaminants, then a glow discharge cleaning cycle to remove  $H_2O$ ,  $CO_2$ ,  $CH_4$ , etc. surface impurities is recommended. This entails insertion of several high power probe anodes, filling the machine with  $H_2$  (or  $D_2$ ) to a pressure of a few m torr, establishing a several hundred volt glow discharge to the chamber (cathode) walls at a current density of a few  $\mu A/cm^2$ , and pumping with sufficient  $H_2$  gas flow to flush out the described impurity molecules. This should be continued for  $\sim 20$ -100 hours for surface cleaning to a depth of  $\sim 1000 \text{ \AA}$ . All cryopanel are to be warm and getter panels cool during the glow discharge cleaning cycle.

Step 2. The cryopanel are then cooled and the getter panels warmed to achieve the maximum  $H_2$  pumping speed and specified low background pressure ( $\ll 10^{-6}$  torr).

Step 3. The fast plasma startup sequence is then initiated. A "cold" target plasma stream is created using approximately 10 MFTF-B type stream guns. The plasma stream becomes the target for "startup" neutral beam injection into the plugs. The required stream plasma is roughly 40 cm in order to provide a plasma halo beyond the required 28 cm hot plasma radius in the plugs. The required line density is  $\sim 2 \times 10^{14} \text{ cm}^{-2}$  in order to have at least 10% trapping of the startup neutral beams. The plasma stream is only required to last a few msec. Successful startup with a cold plasma stream injected into one plug was accomplished in TMX. This mode of startup is assumed here.

Step 4. A set of startup beams is injected into both plugs to rapidly replace the cold target plasma formed in Step 3. A high initial neutral beam current is required in each plug at an appropriate energy ( $\sim 15 \text{ keV}$ ) such that

the hot ion trapping rate over the full radius of interest is at least equal to the charge exchange loss rate from cold impinging neutral gas. The startup neutral beam energy must also be high enough so that the collisional loss time of the hot plasma is longer than the plasma buildup time with the available neutral beams. For reasonable vacuum assumptions, this requires approximately 500 A at 15 keV in each plug. This can be accomplished by an array of ten 50 A beams split into three beam lines, with a fourth beam line reserved for the steady state 250 keV plug beam. A startup neutral beam pulse time  $\sim 10$  msec appears adequate.

Step 5. The final step is the transition from a  $\sim 15$  keV plug plasma to the desired operating conditions fueled by the steady state 250 keV  $H^0$  beams, barrier pump beams on, ICRF heating in the central cell on, and ECRH heating of the plug plasma. The buildup of the plasma to this phase is described in the next section.

#### IX.1.3 Rate Code Calculations

A code based on a set of six coupled rate equations has been developed in order to make numerical simulations of the time evolution of the density and temperature of each species in the central cell and plug, as well as the time dependence of the central cell, plug, and barrier potentials. It is assumed in this code that a target plasma has already been established; the task now is to bring the plasma to the desired operating point. The code is written for the TASKA configuration of an inboard thermal barrier with minimum-B plugs. The objective is to study scenarios for startup and to demonstrate a technique for attaining the steady-state operating conditions of the TASKA device by using a suitable programmed set of injected powers and particle

sources subject to constraints in the amount of sources required. This code is zero-dimensional but fully time dependent.

#### IX.1.3.1 Physics Model

The six coupled rate equations considered here describe the time evolution of density, mean ion energy, and electron temperature in the plugs, and the density, ion temperature, and electron temperature in the central cell region. The processes that have been taken into account in these equations are described below. Only those expressions that have not been specifically used and given in the power balance model (see Section III.2) are detailed here.

The plug ion density is given by

$$\frac{dn_p}{dt} = S_p - \frac{n_p^2}{(n\tau)_{ip}}$$

where the first term on the right side is the ion source due to ionization of the neutral beam, and the second represents the end losses.

The plug electron density evolution is given by

$$\frac{dn_{ep}}{dt} = S_p + \frac{n_{ep}n_b}{(n\tau)_{ep}} \left(\frac{T_{ep}}{T_{ec}}\right)^{\nu_c} \exp\left(\frac{\phi_b + \phi_c}{T_{ep}}\right) - \frac{n_{ep}^2}{(n\tau)_{ep}}$$

where the first term on the right side is the electron source due to ionization of the plug neutral beam; the second gives the source due to central cell passing electrons scattered into the plug; and the third simulates the end losses to the central cell over the potential barrier  $(\phi_b + \phi_c)$ . Also,

$$n_{ep} = n_{ew} + n_{e,passing} \approx n_{ew} ;$$

this model neglects the central cell electron passing density in the plugs.

The central cell ion density evolution is given by

$$\frac{dn_c}{dt} = S_c + J_{\text{TRAP}} \frac{2V_b}{V_c} \frac{\langle \sigma v \rangle_{\text{ionb}}}{\langle \sigma v \rangle_{\text{cxb}}} - \frac{n_c^2}{(n\tau)_{ic}} - \frac{n_c^2}{(n\tau)_{xfi}} - \frac{1}{2} n_c^2 \langle \sigma v \rangle_{\text{DT}}$$

where the first term on the right-hand side corresponds to an external ion source which is only needed for startup; the second term accounts for the ionization of the barrier pumping neutral beams; the third represents the end losses over the potential barrier  $\phi_c$ . The fourth gives the cross field diffusion; and the fifth is the sink due to fusion reactions.

The central cell electron density is given by

$$\frac{dn_{ec}}{dt} = S_c + J_{\text{TRAP}} \frac{2V_b}{V_c} \frac{\langle \sigma v \rangle_{\text{ionb}}}{\langle \sigma v \rangle_{\text{cxb}}} - \frac{n_c^2}{(n\tau)_{ec}} - \frac{n_c^2}{(n\tau)_{xfe}} - \left( \frac{dn_{ep}}{dt} - S_p \right) \frac{2V_p}{V_c}$$

where the first term on the right-hand side represents any external source; the second comes from ionization of barrier pumping neutral beams; the third gives the end losses over the potential barrier  $\phi_e$ ; the fourth simulates the cross field diffusion; and the fifth and sixth give the transfer of particles between the plug and central cell populations.

By quasineutrality, the ion and electron densities and their derivatives are equal everywhere. Therefore, only the ion density rate equations are integrated to obtain the time evolution of the plug and central cell densities. The integration of the electron density rate equations does not provide any new information. By equating the ion and electron currents in the central cell and plugs, two new independent equations are obtained which allow for the calculation of the confinement electrostatic potentials. The central electron

confinement potential follows from the equation

$$\frac{dn_{ec}}{dt} = \frac{dn_c}{dt} \quad .$$

The central cell ion confinement potential follows from the equation

$$\frac{dn_{ep}}{dt} = \frac{dn_p}{dt} \quad .$$

In order to evaluate the confinement parameters  $(n\tau)_{ic}$ ,  $(n\tau)_{ec}$ , and  $(n\tau)_{ep}$ , generalized Pastukhov expressions are used. The Logan-Rensink model is used to calculate the plug ion confinement parameter  $(n\tau)_{ip}$ .

The four temperature rate equations are of the form

$$\frac{dT_j}{dt} = \frac{2}{3n_j} (S_j - L_j) - \frac{T_j}{n_j} \frac{dn_j}{dt}$$

where the subindex  $j$  indicates species;  $S_j$  and  $L_j$  are the sources and sinks of power as described in Section III.2; and  $dn_j/dt$  have been given above.

The expressions for the central cell beta, plug beta, plug radius, barrier parameters (potential, density, radius, beta, ion trapping current, and energy and angle required for the pumping neutral beams), as well as the formulas to calculate the neutral beam absorption fractions, fusion power, powers absorbed and injected into the plasma and amplification factor  $Q$ , are those described in Section III.2.

Plug fueling and heating are provided by neutral beam injection and ECRF power. The central cell fueling and heating are done by ionization of the barrier pumping neutral beams and ICRH power to ions. This time dependent



code also includes external fueling and auxiliary power to central cell electrons to be used during startup. The other source of power taken into account is the heating of central cell ions and electrons due to the thermalization of the alpha particles, but the code does not account for the detrimental effect on the ion density due to the accumulation of alpha particles.

The rate code approach to study transient phenomena has the benefit of simplicity and speed but contains the uncertainties inherent in the models. The electron and ion end loss expressions as well as the Logan-Rensink model used to find the plug ion confinement time are reasonably valid over the range of parameters used in this study. But the expressions for the barrier ion trapping rate are valid only for  $\phi_b/T_{ic} \gg 1$  and intermediate values of the pumping parameter,  $g_b$ . No model has yet been developed which gives the barrier ion trapping current for all ranges of  $\phi_b/T_{ic}$  and  $g_b$ . Unfortunately, during startup  $\phi_b/T_{ic}$  can be less than one. The code assumes that the formula for the barrier trapping ion rate is valid for all obtainable values of  $\phi_b$ , and a constant pumping parameter  $g_b$  is imposed. The same applies to the expression for the barrier passing ion density, which is only valid for  $\phi_b/T_{ic} > 1$ .

#### IX.1.3.2 Method of Solution

The system of six coupled rate equations can be written in the form

$$\frac{dy_i}{dt} = g_i(y_j, \phi) + a_i(y_j, \phi) S_i \quad i, j = 1 \text{ to } 6 \quad (\text{IX.1-1})$$

where  $y_1, y_2, y_3$  are the density, mean ion energy and electron temperature of the plug plasma and  $y_4, y_5, y_6$  are the density and ion and electron temperatures of the central cell plasma.  $S_i$  represents the external sources and  $\phi$

represents the barrier, plug, and central cell potentials, which are given by instantaneous nonlinear equations of the form

$$f(y_j, \phi) = 0 \quad .$$

The solution of this system of equations is calculated using one of the three possible modes of operation of the code:

1. Steady-state calculations in which, for a given set of  $y_i$ , the code calculates the sources  $S_i$  such that  $dy_i/dt = 0$ .
2. Starting with a set of initial conditions  $y_{i0}$ , the code calculates the time evolution of  $y_i$  for given time dependent sources injected into the plasma. To integrate the system, a predictor-corrector method is used.
3. Starting with a set of initial conditions  $y_{i0}$ , the code calculates the sources needed to maintain a certain kind of time variation of  $y_i$ . The profiles chosen are growing exponentials with constant asymptotic values and variable growth rate such that the sources can be kept within some specified limits. A schematic flow chart is given in Fig. IX.1-1.

In the steady state calculation it takes only a few time steps for a set of sources to converge to a stationary value. The convergence is not obtained in the first iteration since some of these sources influence the instantaneous equations which give the potentials at each time step, thus altering the coefficients of the rate equations. If care is used in choosing the input values for the set  $y_i$ , the obtained sources are also reasonable and the convergence is very fast. These calculations are useful in doing parametric studies and trying to obtain a reference case in which the power handled by every fuel and heating system is acceptable.

# FLOW DIAGRAM FOR THE RATE CODE

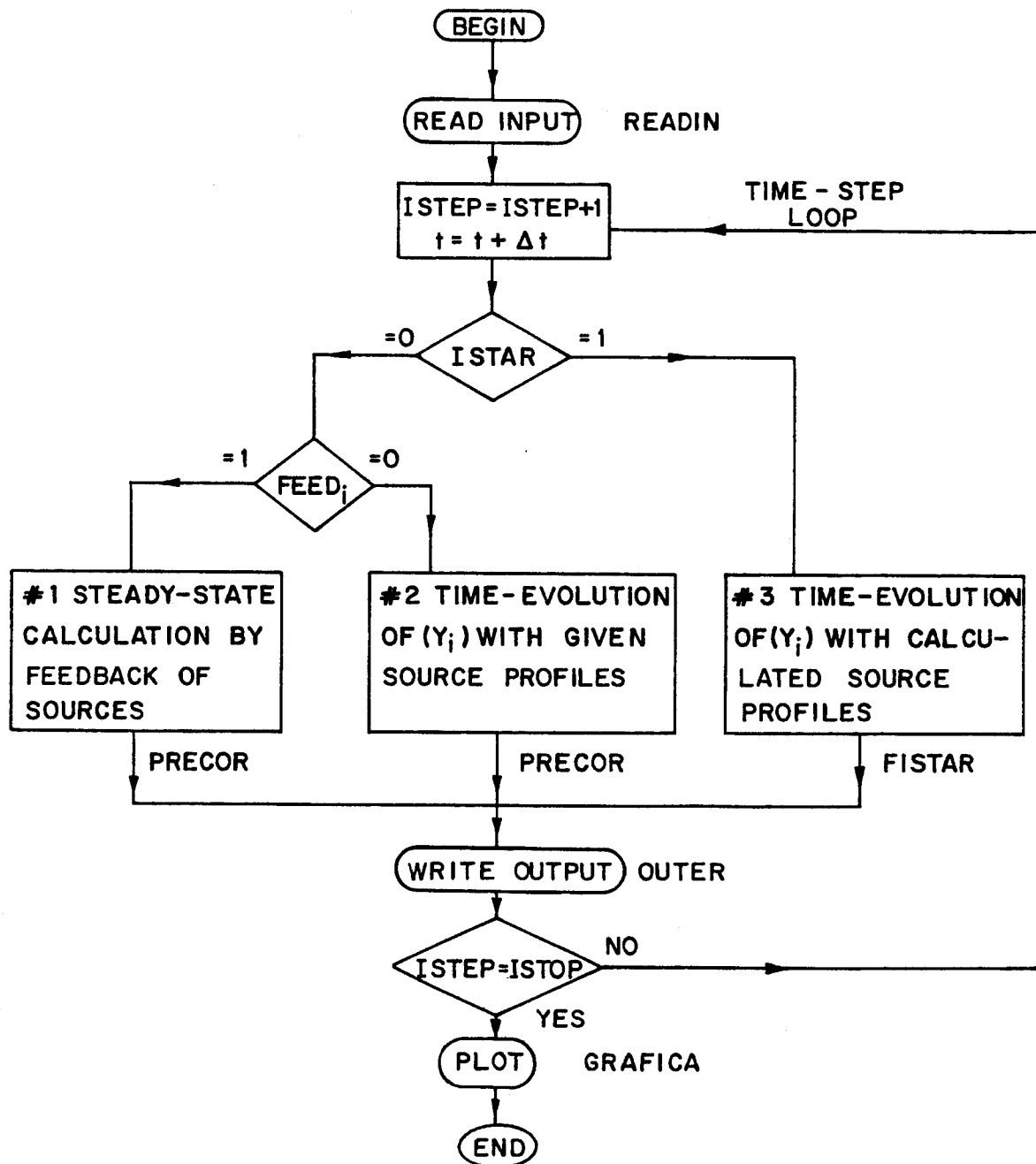


Fig. IX.1-1

The predictor-corrector method used in the second mode of operation is Heun's method. Figure IX.1-2 shows the scheme of the calculation. The predictor is Euler's formula and the corrector assumes that  $y_i(t)$  is quadratic. The iterative procedure between the corrector formula and the corresponding differential equation continues until the convergence criterion is satisfied for all six equations simultaneously. There is a trade-off between the length of the time step and the number of corrector iterations. If  $\Delta t$  is chosen too large, an excessive number of iterative cycles may be needed for convergence or even the algorithm may not converge at all. Large differences between predictor and corrector outputs suggest reduction of the interval. On the other hand, very small differences between predictor and corrector outputs suggest increasing the time step length and perhaps speeding up the computation. The convergence criterion used is  $\epsilon = 0.001$  with maximum number of iterations ICMAX=10. If this number is reached the time step length is reduced and the time step repeated.

The time step length can be either specified or calculated at the beginning of each step. The code calculates characteristic times  $\tau_i$  for the variation of each variable and the subsequent time step length is limited to 10% of the shortest characteristic time for density or temperature variations up to a specified maximum time step size as equilibrium is approached.

The numerical stability of this scheme has been studied by choosing a set of initial conditions and external sources known to form a stationary point, and running the code with different fixed time step sizes. It was found that for time steps  $\leq 0.1$  ms, the differences obtained with respect to the stationary point, after a long enough simulation, were not significant. For time steps sizes  $> 1$  ms, the departure from the stationary point was significant.

# CALCULATIONAL SCHEME OF MODE # 2.

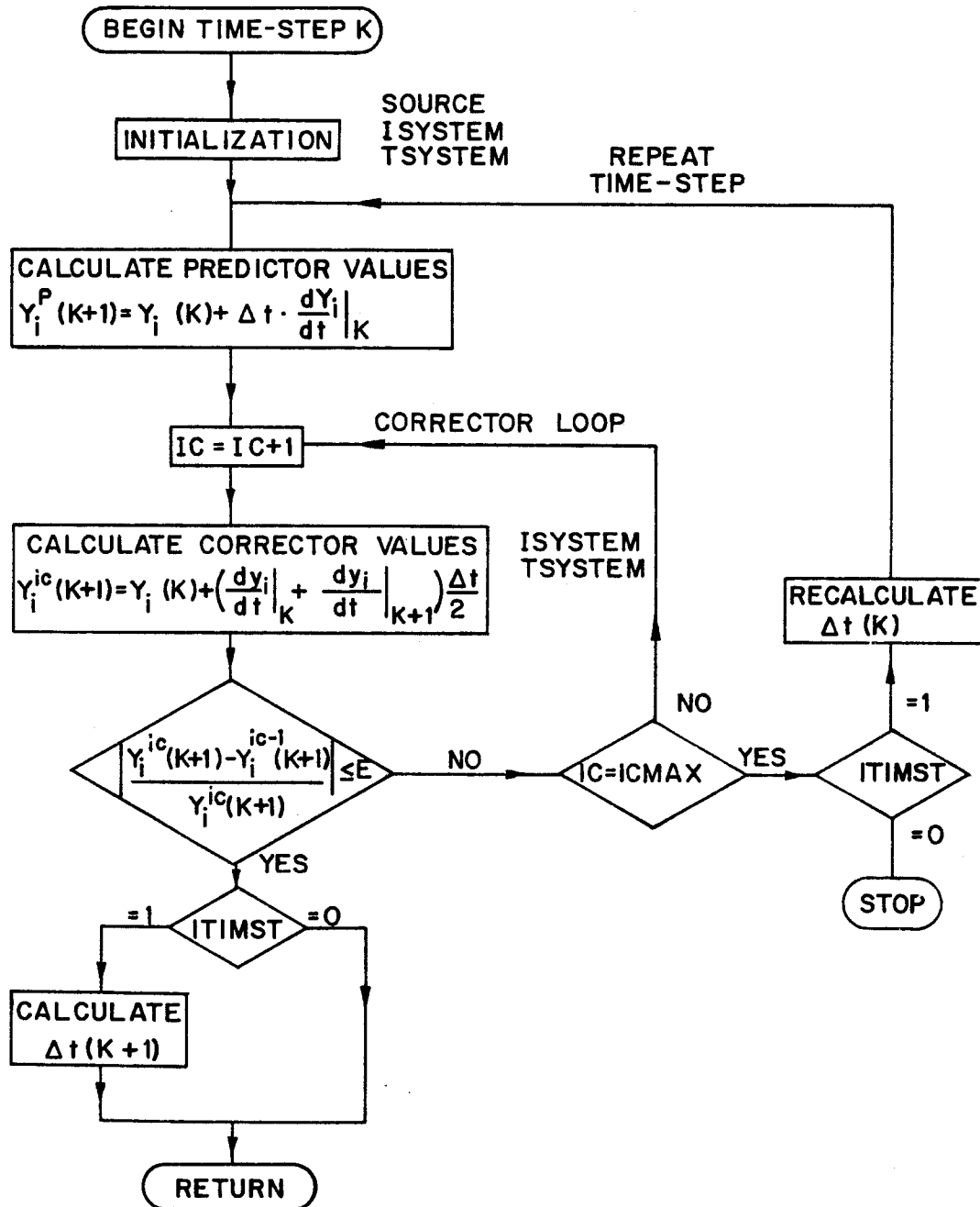


Fig. IX.1-2

Therefore,  $\Delta t = 1 \text{ ms}$  is the largest time step used in calculations near steady state values. For situations far from steady state a very small time step is initially taken ( $\sim 1 \text{ } \mu\text{sec}$ ) and the option of variable time step is used.

The predictor-corrector method can be used in combination with steady state calculations in finding the time evolution of some of the variables and the sources required to keep the other variables constant. This method is helpful for doing steady state calculations in which any of the required powers becomes negative. This power can be set equal to zero and the corresponding variable is allowed to change. Also it can be used for startup calculations in which one of the cells (e.g., the plug) is initially built up and it is required to hold it in steady state.

The scheme of calculation of the third mode is represented in Fig. IX.1-3. At time step  $k$ , the iteration is started using the values of the exponential growth rates  $\alpha_i^k$  and external sources  $S_i^k$  found in the preceding time step. The values of the variables at time  $t + \Delta t^k$  are found from the equations

$$y_i^{k+1} = y_{iA} - (y_{iA} - y_i^k) e^{-\alpha_i^k \Delta t^k} \quad i = 1 \text{ to } 6$$

where  $y_{iA}$  are the asymptotic values of the variables which coincide with the steady state operating point.

Now,

$$\frac{dy_i^{k+1}}{dt} = (y_{iA} - y_i^k) \alpha_i^k e^{-\alpha_i^k \Delta t^k} \quad (IX.1-2)$$

However, from the rate Eq. (IX.1-1) the time derivative has the expression

# CALCULATIONAL SCHEME OF MODE # 3.

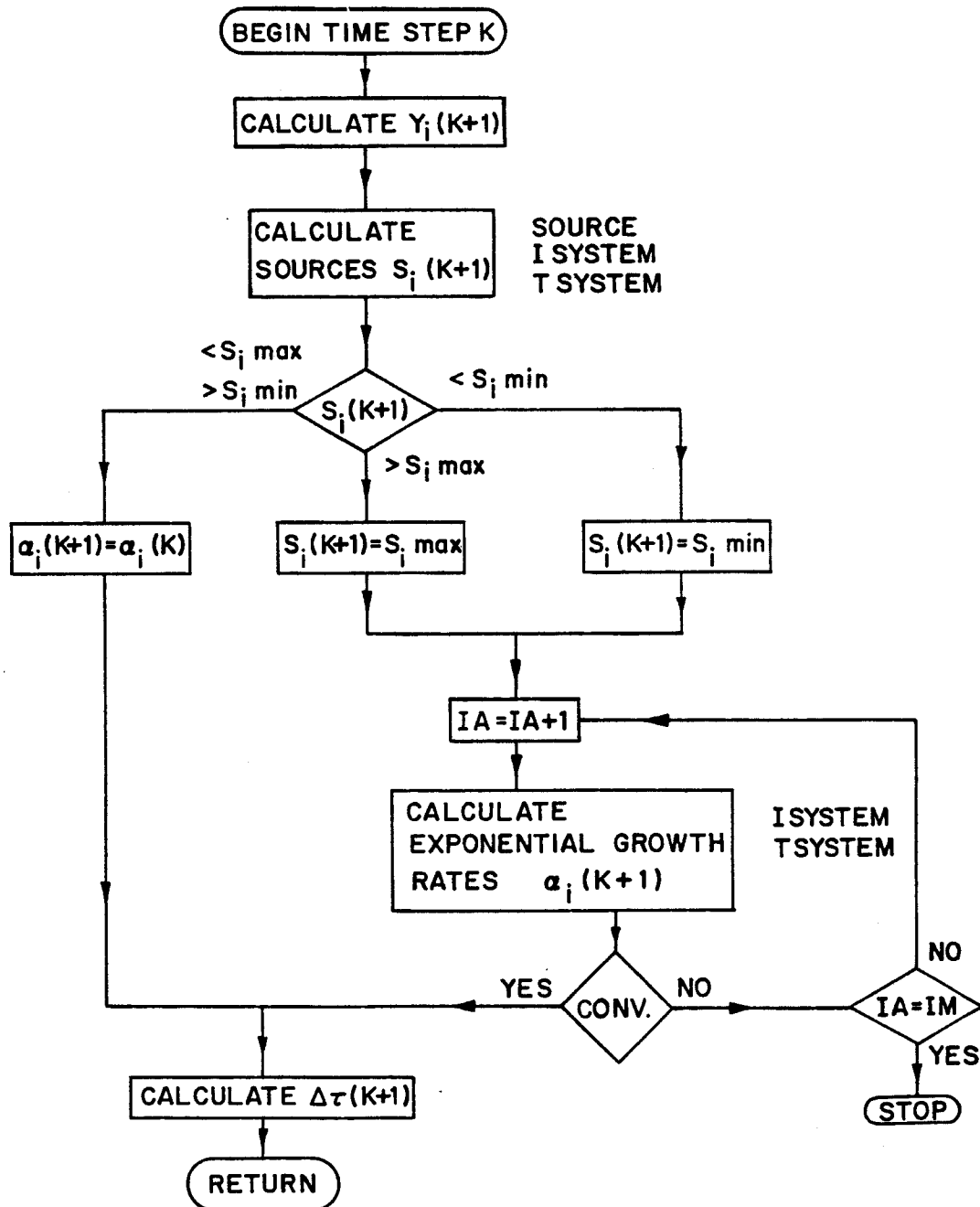


Fig. IX.1-3

$$\left(\frac{dy_i^{k+1}}{dt}\right) = g_i(y_j^{k+1}, \phi) + a_i(y_j^{k+1}, \phi) \cdot S_i^{k+1} \quad j = 1 \text{ to } 6 . \quad (\text{IX.1-3})$$

This can be solved for the source strength,

$$S_i^{k+1} = \frac{\frac{dy_i^{k+1}}{dt} - g_i(y_j^{k+1}, \phi)}{a_i(y_j^{k+1}, \phi)} . \quad (\text{IX.1-4})$$

This source is now compared with specified maximum ( $S_{i\max}$ ) and minimum ( $S_{i\min}$ ) values of the source given as input data. If  $S_{i\min} < S_i^{k+1} < S_{i\max}$ , the values of  $y_i^{k+1}$  and  $S_i^{k+1}$  are those given by Eqs. (IX.1-2) and (IX.1-4), respectively, and the code takes  $\alpha_i^{k+1} = \alpha_i^k$ . If  $S_i^{k+1} < S_{i\min}$ , the code sets  $S_i^{k+1} = S_{i\min}$  and the new values  $\alpha_i^{k+1}$  are the solution of a nonlinear system of six coupled equations

$$(y_{iA} - y_i^k) \alpha_i^{k+1} e^{-\alpha_i^k \Delta t^k} = g_i(y_j^{k+1}, \phi) + a_i(y_j^{k+1}, \phi) S_{i\min}$$

where

$$y_j^{k+1} = y_{jA} - (y_{jA} - y_j^k) e^{-\alpha_j^{k+1} \Delta t^k} .$$

It should be noted that all  $\alpha_j^{k+1}$  are involved in the calculation of  $\alpha_i^{k+1}$ . The loop is iterated until the convergence in all six  $\alpha_i^{k+1}$  is obtained simultaneously. If  $S_i^{k+1} > S_{i\max}$ , the same procedure is used, but now putting  $S_i^{k+1} = S_{i\max}$ .

Once the values of  $y_i^{k+1}$ ,  $\alpha_i^{k+1}$ , and  $S_i^{k+1}$  are calculated, the code obtains the characteristic time  $\tau_i$  for the variation of each unknown,



$$\tau_i^{k+1} = \frac{1}{\alpha_i^{k+1}} \left( \frac{y_{iA}}{(y_{iA} - y_i^k) e^{-\alpha_i^{k+1} \Delta t^k}} - 1 \right)$$

and the new time step is estimated as one-tenth of the minimum of these characteristic times.

The condition to reach the asymptotic values is that  $\alpha_i^{k+1}$  be positive. If at any time step a negative  $\alpha_i$  value is found the code changes it to the positive value that is specified as input data, and recalculates the source needed to sustain this growth rate, even if this implies going out of the allowed interval for the variation of sources. This feature has been included in the code in order to avoid the value of the variable going to  $-\infty$  and the calculation being stopped due to numerical errors.

Another source of numerical error is due to those variables which reach values very near to the steady state values. This is corrected by assuming that when a variable reaches a value close enough to the desired asymptotic one, both are equal and the calculations related with that particular variable are skipped.

Finally, for this method to work, the initial set of  $\alpha_i$ ,  $S_{imin}$ , and  $S_{imax}$  must be chosen with care, since the loop that calculates  $\alpha_i$  is highly non-linear, and has included in it the calculation of potentials which are also given by nonlinear equations. By using the predictor-corrector method of operation simulating a short excursion, a good estimation of the initial values of  $\alpha_i$  and of the limiting values of the sources can be obtained.

Next the instantaneous calculations carried out at every time step are described. The code evaluates several parameters either for output purposes or for use in obtaining the coefficients for the rate equations. This

calculation is common for the three ways of solving the rate equations. The scheme is as follows. Magnetic field profile, central cell radius and length, and barrier length are held constant. First central cell beta, plug beta, and plug radius are calculated using densities and temperatures of the preceeding time step. The code now proceeds to find the barrier parameters by solving one of two possible barrier models which are based on the same equations, the difference between them being the input and output variables. In the first model, the pumping parameter  $g_b$  and the fraction of total pumping done for each one of the three pumping beams is given. Then, barrier beta, potential and density are calculated in a loop. Next the code obtains the trapping current and the pumping neutral beam currents. Injection energy and angle of the neutral beams can be either input or calculated; if they are input data the code checks that the injection point is in the appropriate region of velocity space. In the second model, the beam current profiles are given and the pumping parameter calculated. This last model seems more realistic in the sense that the beam currents can be externally controlled. The problem with this approach is that even though any kind of time dependent profile for the neutral beams can be specified, it is difficult to guess one that will obtain reasonable barrier parameters when compared with the parameters of other regions of the machine. The code also includes the possibility of doing the barrier pumping in two stages. In this option, the high energy pumping neutral beam does not pump ions directly to the central cell but rather it changes their energy in such a way that the resulting ions can be pumped by a very low energy neutral beam. After the barrier calculation the code obtains the central cell ion confinement potential, solves the Logan-Rensink model for

the plug and finally it calculates the central cell electron confinement potential.

In doing startup calculations, there is usually no difficulty in obtaining the time evolution using either one of the two integration schemes. But when the code is solving the instantaneous equations, it may occur, and often does, that for certain values of densities and temperatures the nonlinear equations that give the potentials do not have roots or they are out of the valid ranges. This happens more often with sets of values far from the stationary point or for situations in which one or more of the variables have grown at a much faster rate than the others. If this occurs, the calculation is stopped. This suggests that the physics model used only allows for the simultaneous buildup of the three regions of the machine.

The exponential growth method provides an easier way of simulating startup sequences than the predictor-corrector method does. In the latter method the sources specified at every time step have strong influence in the rate equations. Therefore, the set of densities and temperatures obtained as a result of the time integration may be such that the roots of the equations which give the potentials cannot be found. The exponential growth method does not have this inconvenience because the sources are not input data to the time integration. In order to obtain a complete startup sequence by the predictor-corrector method it would be necessary to adjust the input sources every few steps. The output consists of the time evolution of the most relevant parameters of the machine. This code is written to run on the MFECC CDC-7600 machine. Running time is approximately twenty seconds per one thousand time steps.

### IX.1.3.3 Results

Startup studies are aimed at demonstrating a technique for attaining steady state plasma parameters starting from a set of initial conditions and trying to achieve this goal, while minimizing the amount of injected power. An ideal case would be to reach the stationary point using only the power supply needed at steady state operation. Unfortunately this goal seems impossible to accomplish, at least within the constraints of the physics model used here.

The results presented in this section are an example of the estimates of the amount of power and particle currents that are expected to be necessary to reach a steady state operating point in the machine. They have been obtained using the exponential growth method. The asymptotic values and input parameters correspond to the reference case (see Section III.4). The density and temperature of the initial target plasma are assumed to be one-tenth of the asymptotic values.

Graphs that give the time evolution of the more relevant plasma parameters and the required external sources are included. Densities are given in  $\text{cm}^{-3}$ ; temperatures, potentials and energies in keV; currents in  $\text{sec}^{-1}$ ; powers in MW; and time in seconds.

Figure IX.1-4 shows the time evolution of plug density, plug ion mean energy, plug electron temperature, central cell density, central cell ion temperature, and central cell electron temperature. Plug electron temperature is the first variable to reach steady state even though the ECRH power to plug electrons is less than nominal during startup. The rest of the variables, except the plug ion mean energy, reach the stationary point in  $\sim 300$  msec. The plug ion mean energy reaches its steady state value at time  $\sim 1$  sec.

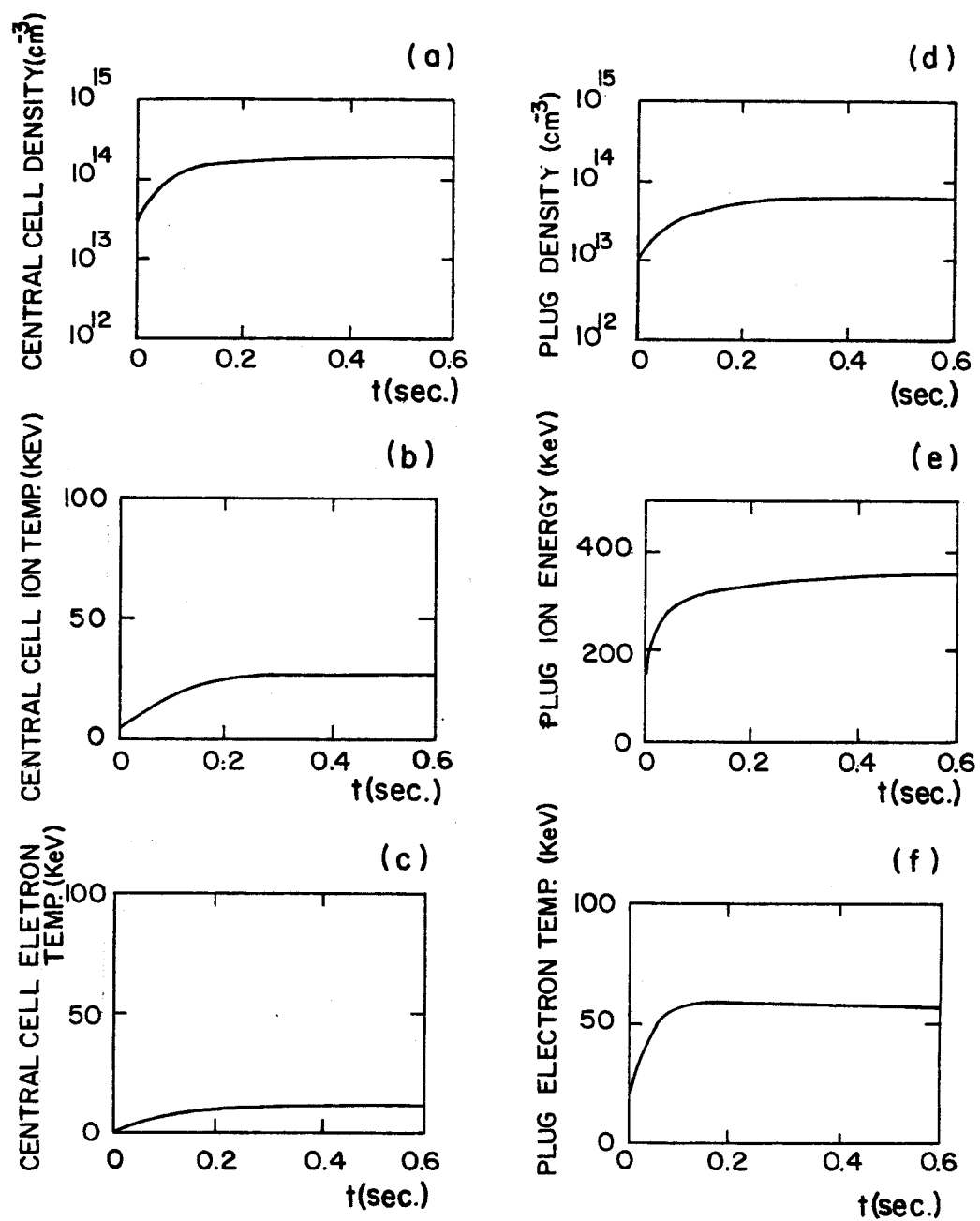


Fig. IX.1-4. Time-evolution of densities and temperatures in central cell and plug.

Figure IX.1-5 shows the evolution of barrier, plug, and central cell potentials, and central cell, barrier and plug betas. The plug beta reaches the steady state value in  $\sim 1$  sec. The other variables are stationary at time  $\sim 300$  msec.

Figure IX.1-6 describes the medium energy barrier pumping neutral beam showing the time evolution of the power and current absorbed by the plasma; injection energy (held constant during this simulation); power and current injected into plasma and absorption fraction, respectively. The absorbed power needed for startup is smaller than that at steady state. This fact is compensated by the smaller absorption fraction; the end result is that the total injected power remains near the steady state value. Similarly, Fig. IX.1-7 describes the high energy barrier pumping neutral beam, and Fig. IX.1-8 shows the time evolution of the low energy barrier pumping neutral beam.

Figure IX.1-9 shows the evolution of the plug neutral beam: neutral beam power and current absorbed by the plasma, energy of the beam, neutral beam power and current injected into plasma and absorption fraction of the beam. During startup the absorbed plug neutral beam power is slightly above its steady state value. But when the absorption fraction is taken into account, the injected power needed grows dramatically.

Figure IX.1-10 shows the time evolution of the plug ECRH power and the central cell sources: ICRF heating to ions, particle source and auxiliary electron heating. The time evolution of the absorption coefficients for the several RF heating systems are not yet modeled in the code so total absorption is considered. ECRF heating of the plug must be supplied gradually or else the plug electrons overheat. Finally, Fig. IX.1-11 shows the total absorbed power in the plasma, amplification factor of the absorbed power, fusion power

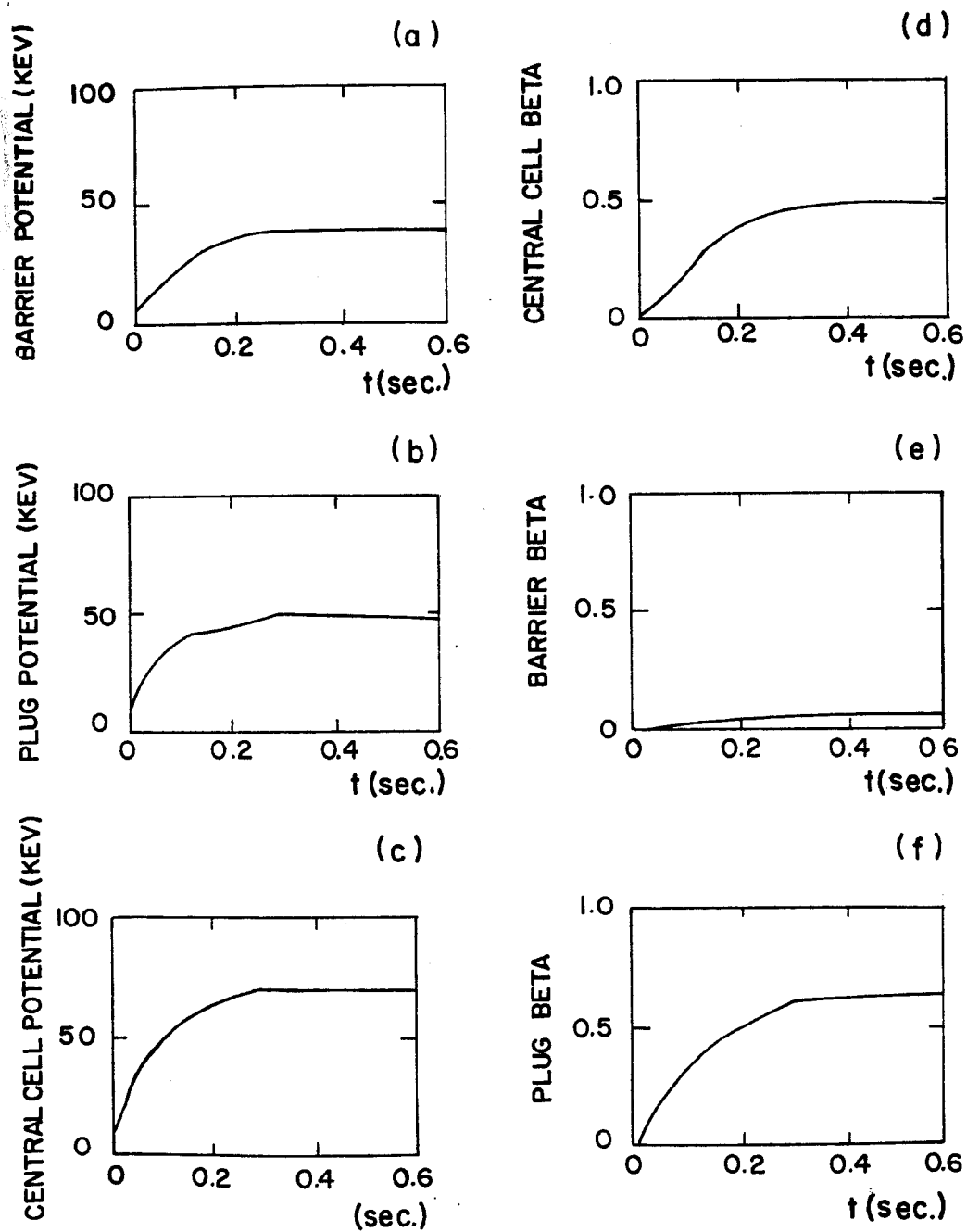


Fig. IX.1-5. Time-evolution of potentials and betas.

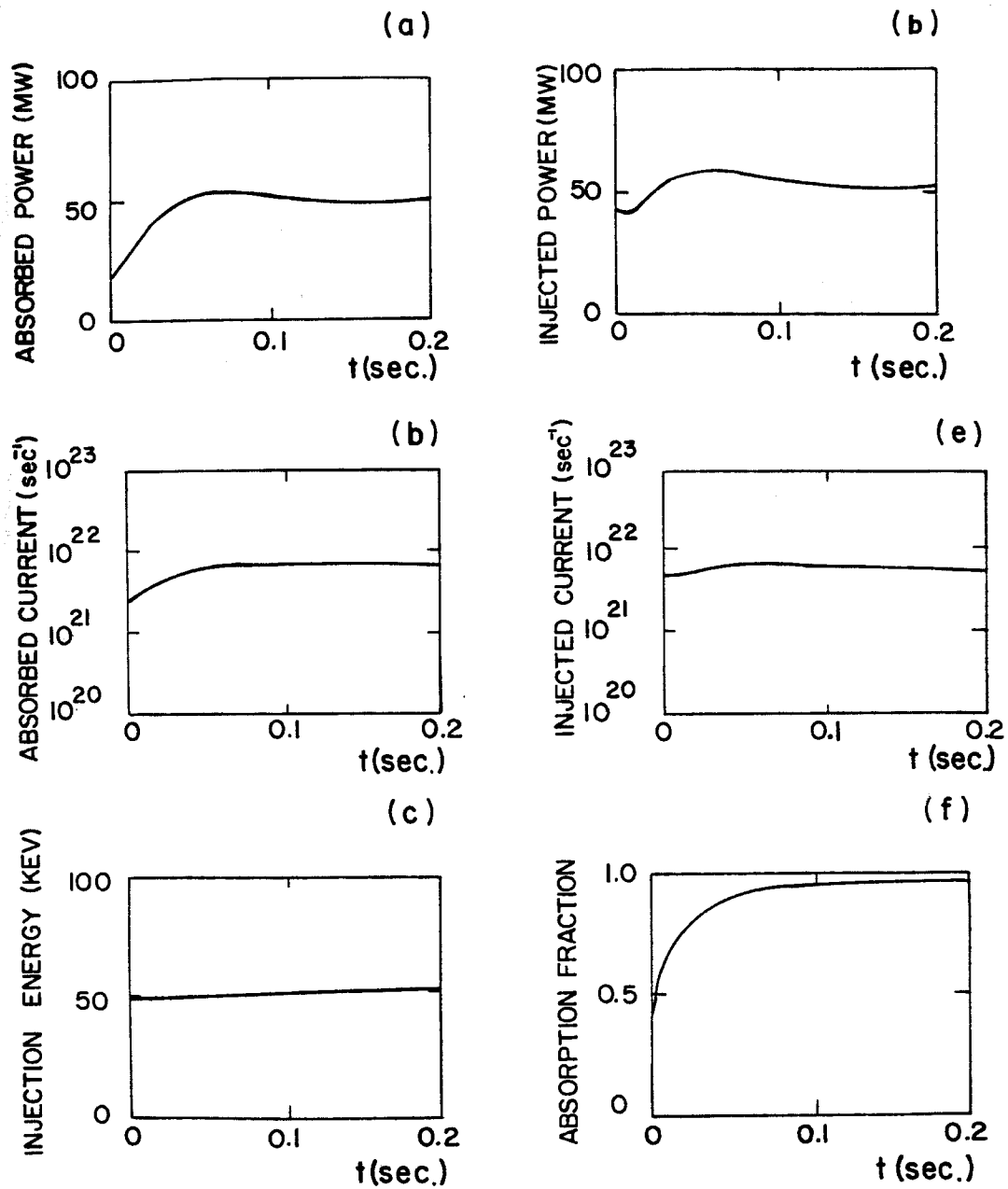


Fig. IX.1-6. Time-evolution of the medium-energy barrier pumping neutral beam.



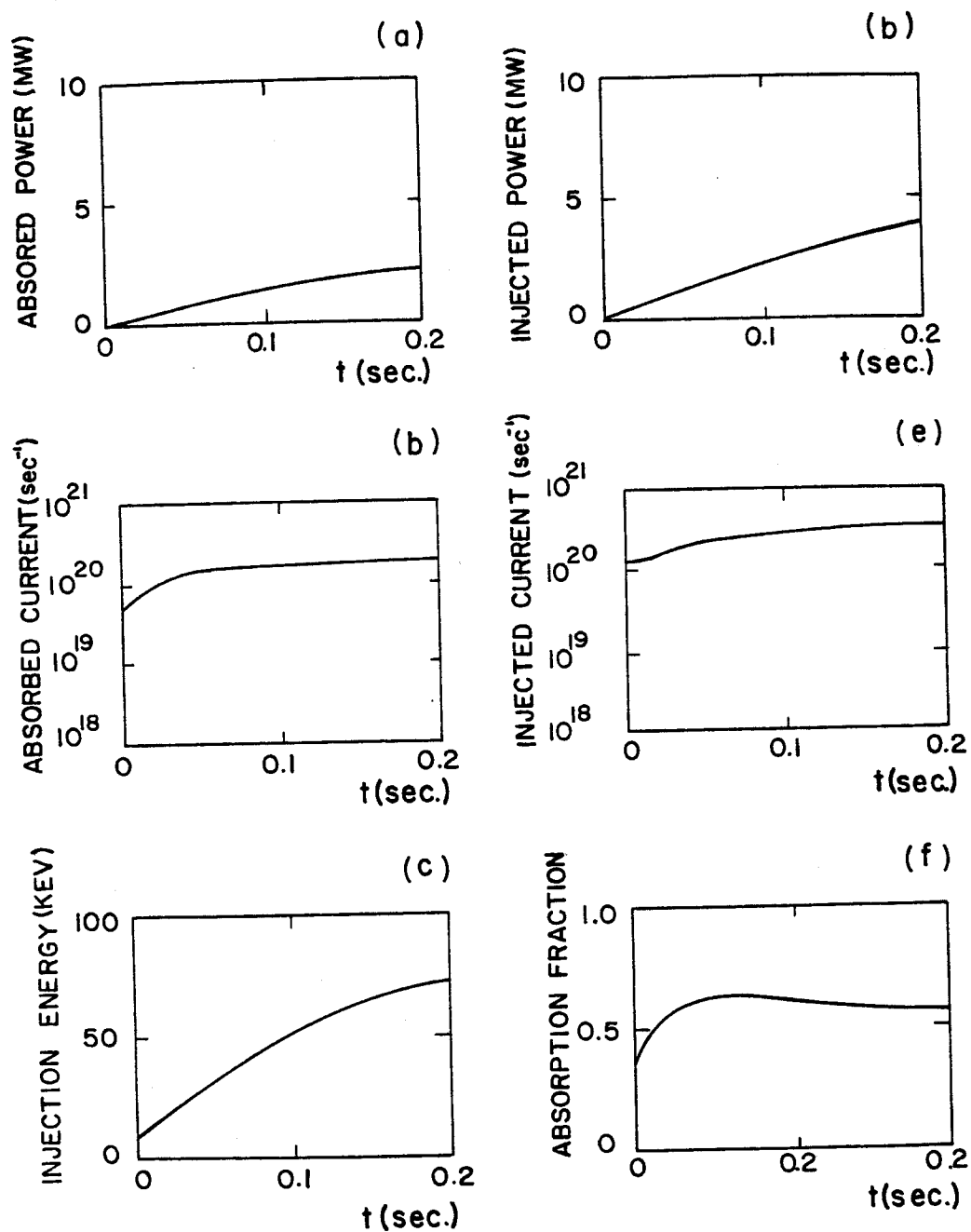


Fig. IX.1-7. Time-evolution of the high-energy barrier pumping neutral beam parameters.

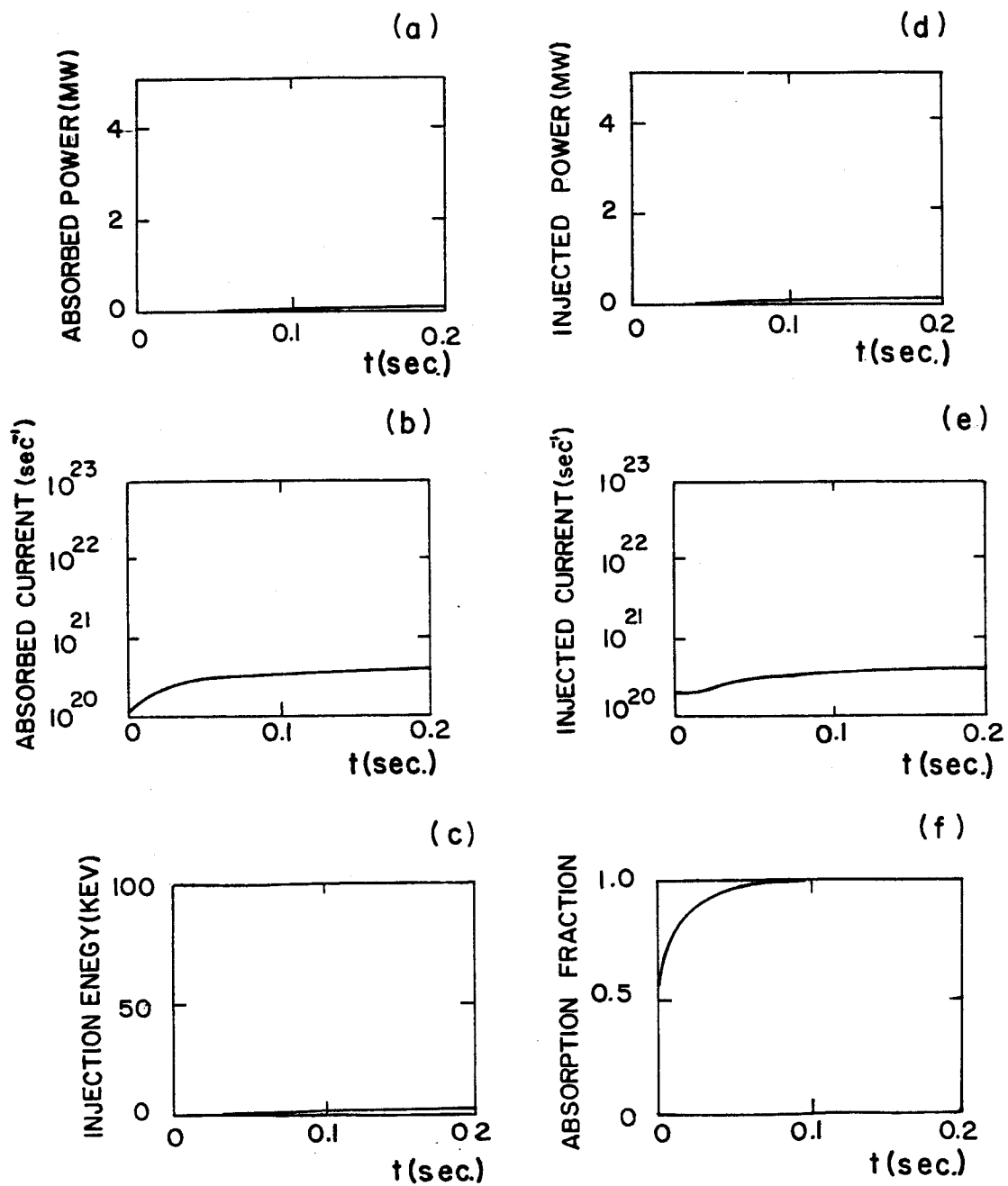


Fig. IX.1-8. Time-evolution of the low-energy barrier pumping neutral beam parameters.

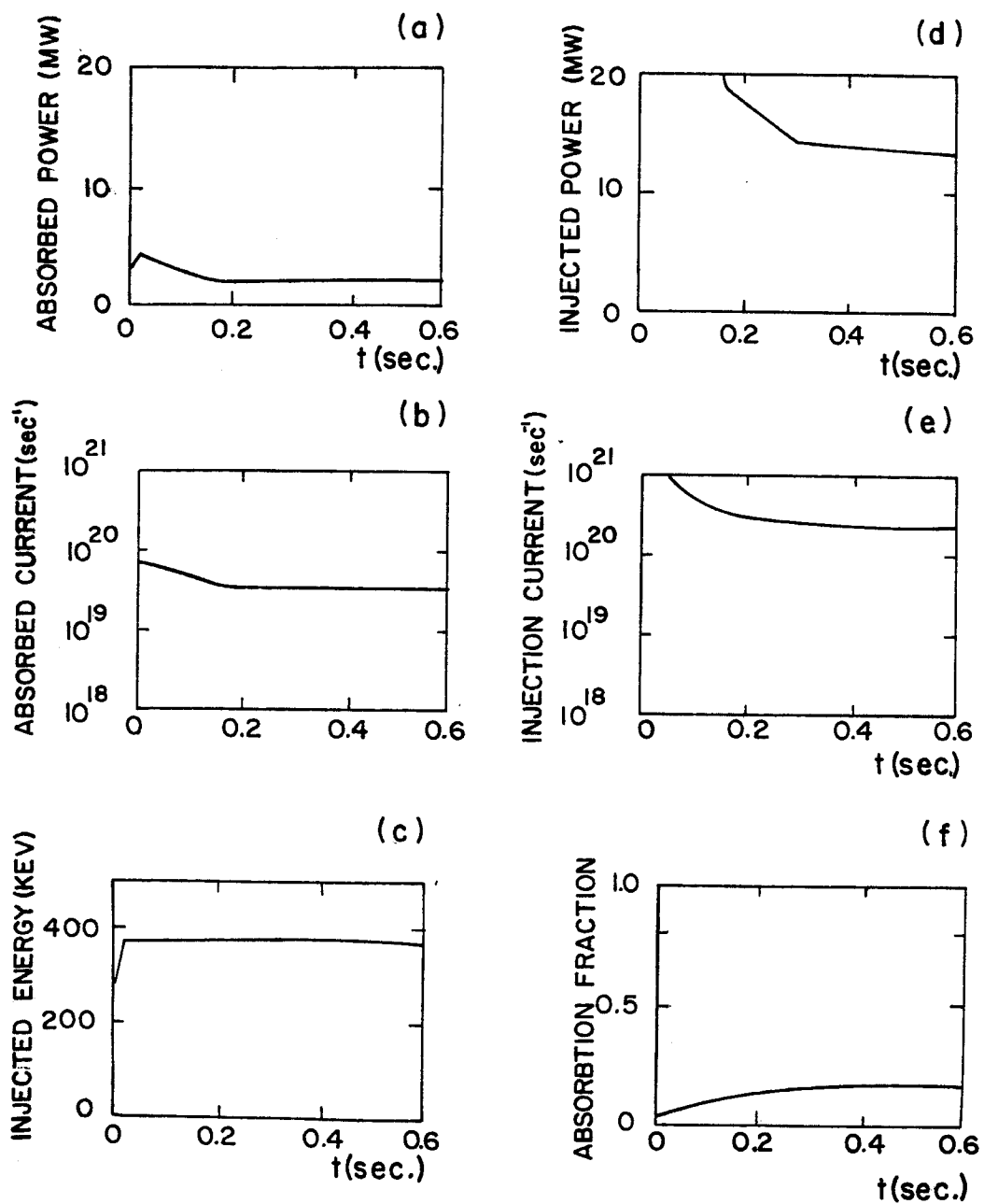


Fig. IX.1-9. Time-evolution of the plug neutral beam parameters.

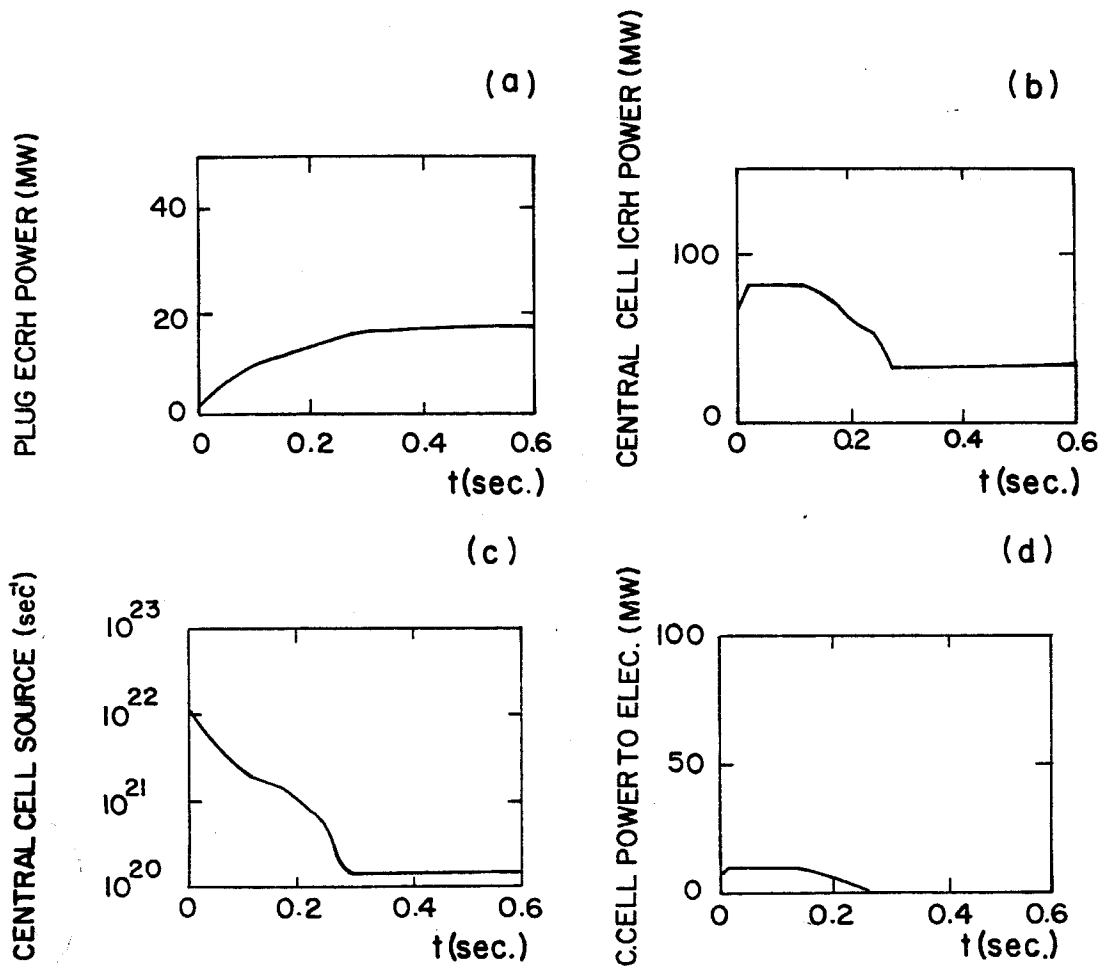


Fig. IX.1-10. Time-evolution of:

- (a) ECRH power absorbed in the plug.
- (b) ICRH power absorbed in the central cell.
- (c) Central cell particle source.
- (d) Auxiliary power absorbed by central cell electrons.

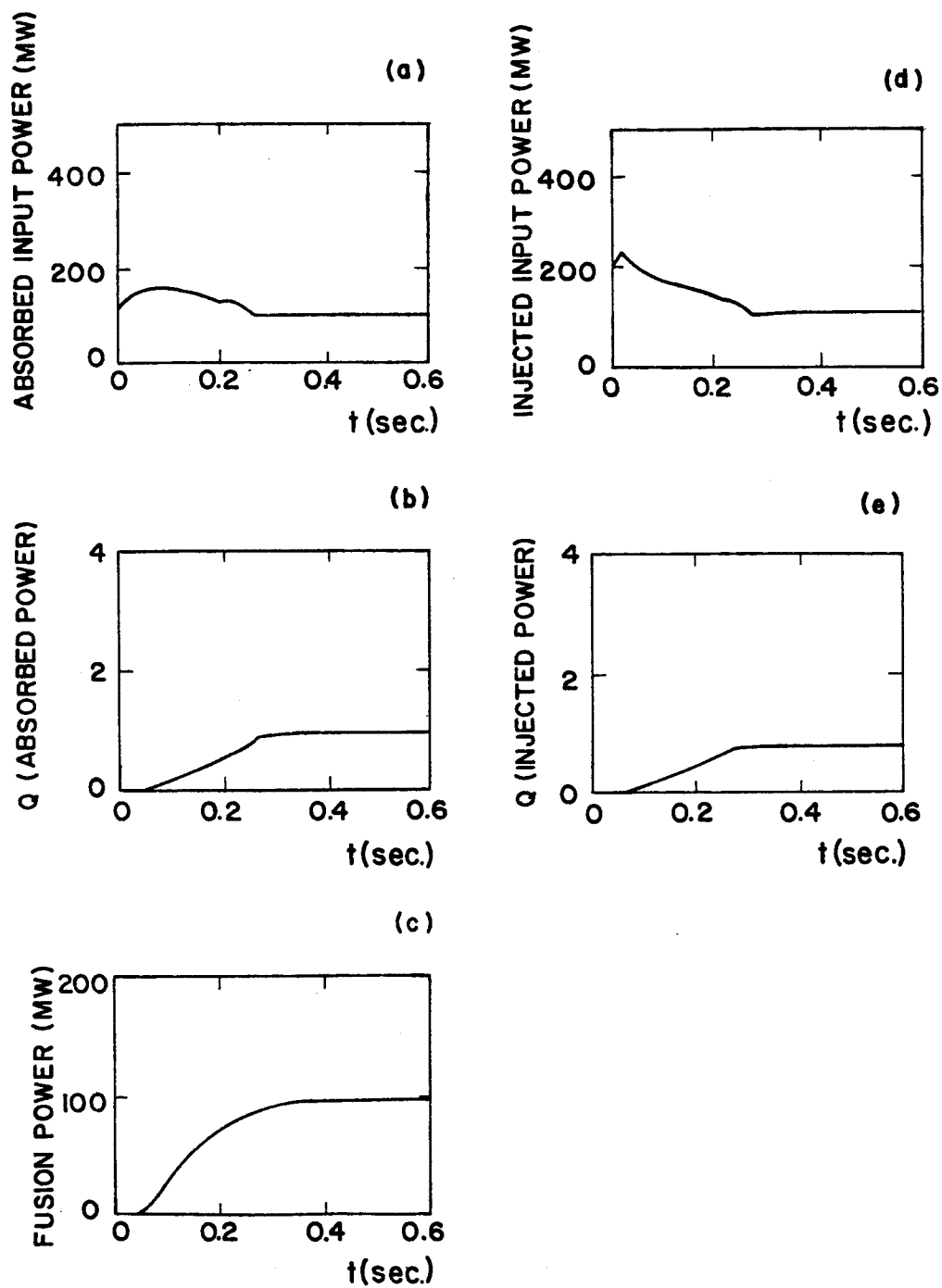


Fig. IX.1-11. Time evolution of: (a) total absorbed input power; (b)  $Q$  (absorbed power); (c) fusion power; (d) total injected power; (e)  $Q$  (injected power).

produced, total injected power in the plasma and amplification factor of the injected power. With respect to the central cell results one can see the large amount of extra heating and fueling, with respect to steady state values, that are needed to obtain the buildup of the central cell plasma. For this startup sequence the total injected power in the first stage of the buildup of the plasma, more than doubles the steady state value.

#### IX.1.4 Conclusions

From the results presented and from other possible startup scenarios that have been considered, there are some conclusions that can be made; their generality, however, is constrained by the validity of the physics model used.

1. In startup studies, a steady state plasma with the desired parameters can be achieved, but only with careful programming of the particle sources and injection power.
2. In steady state operation, the fueling of the central cell is provided by the passing ions produced by the ionization of the barrier pumping neutral beams. During startup the barrier neutral beams are not working at full power. Their strengths must be ramped in such a way to provide the necessary pumping power to create an increasing dip in the barrier potential. (An effective ramp of the neutral beam power can be obtained by repetitive pulsed operation with increasing duty factor.) Because of this, the central cell is underfueled and it needs an additional particle source until its buildup is completed.
3. Central cell heating in steady state operation is due to both direct ion heating by ICRF and power transferred by passing ions coming from the thermal barrier region. The latter is reduced during startup and a larger amount of ICRF power must be injected to raise the ion temperature. For a

design without any direct central cell ion heating, as was considered for the TASKA design in an earlier stage, all the central cell heating in steady state operation is provided by the barrier passing ions. This kind of design worsens the problem of lack of power to the central cell during startup since it increases substantially the extra power needed to reach the steady state central cell ion temperature and it implies the necessity of an independent heating system to be used only during startup.

4. Full RF power to the plug electrons is not needed during startup. If the nominal value of ECRH power is absorbed in a low density plasma, the rate of growth of the plug electron temperature is larger than that of any other species in the central cell and plug. This causes the plug electrons to overheat, leading to a very lossy plug plasma and eventually to the failure of the plug model.

The preceding considerations about the power required for startup apply to those powers absorbed by the plasma. The actual powers that need to be injected depend on the absorption coefficients which change dramatically with the plasma conditions. This effect influences the power injected as follows:

1. The injected power required for the barrier pumping neutral beams turns out to be approximately that of the steady state situation. This is because the power absorbed by the plasma and the absorbed fraction of the injected power increase as the barrier buildup progresses. In some sense, the barrier pumping neutral beams ramp themselves.
2. The plug neutral beam injection power needed in the first stage of startup increases by an order of magnitude. This suggests the necessity of studying other approaches. For example, initially creating a cold target

plasma of high density in the plug and using the neutral beams to heat this plasma is one possibility.

3. It has been seen that the heating characteristics of the plug region during startup are such that the absorbed ECRH power should increase gradually. Otherwise, it is known that the absorption of ECRH power improves as electron temperature increases. Therefore, it seems reasonable to expect that the ECRH system might naturally ramp itself. This will not be applicable to the RF power needed in the central cell since they're twice the steady state value of ICRF absorbed in plasma is required at startup.

It appears possible that a revision of the rate at which the central cell is built up will reduce this initial surge in the required ICRF power. Further work on various startup scenarios is required in order to be more definitive about this possibility. Further work on starting at lower initial values, i.e. more appropriate to stream gun plasma conditions, is also needed in order to model the entire startup process.



## IX.2 Diagnostics, Instrumentation, and Control

### IX.2.1 Introduction

The requirements for diagnostics, instrumentation, and control for TASKA can be conveniently divided into three aspects, namely General Instrumentation and Control, Neutron Diagnostics and Plasma Diagnostics. Requirements for each of these categories follow in the next three sections.

A detailed listing of each of the transducers, sensors and control instruments required by this device will not be given here; indeed such a description would be well beyond the scope of this report. However, a broad examination of the requirements will demonstrate the wide variety of diagnostic and control instrumentation necessary for a machine of this complexity, and will indicate which aspects of the instrumentation will require attention between now and the envisaged time frame for TASKA (i.e., the 1990's). In this respect, it should be noted that whereas the status of instrumentation for both general control monitoring and neutron diagnostics is in a fairly comfortable situation, the same is not true of plasma diagnostic instrumentation. This is because there are few current plasma diagnostic techniques which can be successfully integrated with the severe reactor-like environment of a fusion device such as TASKA. There appears, therefore, to be an urgent requirement for the development of fully engineered and hardened plasma instrumentation capable of sustained operation in a high power D-T environment.

### IX.2.2 General Instrumentation and Control

Individual aspects of the general instrumentation and feedback control requirements for TASKA do not present the same degree of sophistication of complexity as those for the neutron and plasma diagnostics. This is due to

the fact that the majority of these requirements will be met by conventional process instrumentation which is currently available. In addition, the wealth of experience stemming from the design and application of instrumentation and control systems for fission reactors will be of direct applicability to TASKA.

In this connection, two areas of concern for the operation of conventional instrumentation in TASKA should be highlighted. Firstly, instrumentation positioned in the inner regions of the blanket or shield near the plasma will be subjected to neutron and gamma radiation fields of much higher mean energies than those encountered in fission reactors. Thus, conventional instrumentation systems in this region (e.g., temperature monitoring thermocouples, etc.) will be expected to sustain increased levels of radiation damage (see, for example, reference 1). Secondly, this instrumentation must be able to function successfully in a magnetic field environment where  $B$  will attain several teslas. Attention should, therefore, be focussed on the replacement of these susceptible hardwired signal channels and their associated transducers and preamplifiers, etc., with data transmission via light, IR or microwave techniques.

Although more or less conventional process/BOP instrumentation and control systems will be employed, TASKA will, nevertheless, be a highly instrumented device. In this connection, a total number of  $\sim 10,000$  data channels may be assumed. A redundancy factor of the order of three will be incorporated for those data channels originating from, or directed to, inaccessible regions of the machine, thus reducing maintenance and downtime requirements. It is also envisaged that digital signal multiplexing will be employed, by which signals are digitally encoded near their source and then transmitted simultaneously to control room destinations via a few main data

highways. Such a system can be seen to result in a large reduction in wiring penetrations and installation costs.

A summary list of general instrumentation and control requirements for TASKA follows below in Table IX.2-1. As indicated above, much of the parametric data transmission will be by light/fiber-optic, IR, or microwave technology, a field which is currently undergoing rapid evolution. Microprocessor-based control, of at least one order of magnitude greater capability than present day, may be assumed.

Table IX.2-1  
General Instrumentation and  
Control Requirements for TASKA

1. NEUTRAL BEAMS

NBI ion source parameters; beam diagnostics and calorimetry; injected power distribution; internal ion dump temperature and coolant conditions; vacuum conditions; getter pump cycling control; tritium injection and recycle monitoring; magnetic shielding conditions.

2. NEUTRAL BEAM DUMPS

Residual NB diagnostics; dump temperatures; coolant conditions and flow rates; burnout detection; tritium leakage.

(Table IX.2-1 continued)

3. ECRH

All ECRH input parameters; microwave injected power and input distributions; gyrotron coolant conditions; gyrotron magnet cooling; waveguide and horn temperatures and coolant conditions; SF<sub>6</sub> gas parameters; vacuum conditions.

4. ICRH

ICRH input parameters to RF oscillators; RF power to antennas; RF power and distribution to plasma; antenna coolant conditions; coaxial line temperature profiles; SF<sub>6</sub> gas conditions.

5. MAGNET SYSTEMS

All cryogenic parameters for He, N<sub>2</sub>, and H<sub>2</sub>O including cryo-levels, temperatures, flow rates, boiling detection; refrigeration plant instrumentation; superconducting coil parameters including voltage, currents, temperatures; coil quench detection; coil shorted turn detection; general magnet protection feedback system; general power supply instrumentation and quench protection; extensive field monitoring.

(Table IX.2-1 continued)

6. VACUUM VESSEL - FIRST WALL

Vacuum conditions in each region; first wall temperatures and coolant conditions; neutron fluence levels for radiation damage assessment; structural loading.

7. BLANKET

Temperature profiles; neutron flux levels; gamma flux levels; Pb-Li coolant conditions (pressure, temperatures, flow rates); power flow; breeding ratio; tritium containment and leak detection; induced radioactivity; decay heat conditions.

8. REFLECTOR AND SHIELD

Temperature profiles, neutron/gamma leakage; tritium leak detection; induced radioactivity; decay heat conditions.

9. REGAT MODULE

General test module instrumentation; temperature profiles; neutron/gamma fluence gradients; coolant parameters; tritium leakage and contamination.

(Table IX.2-1 continued)

10. VACUUM SYSTEM

General pumps - pressures; flow rates; tritium containment; leak detection.

Cryopumps - temperature; cryo-conditions; saturation recycling monitoring; backing pump instrumentation; tritium containment and leak detection.

Getter panels - temperatures; saturation recycling monitoring; backing pump instrumentation; tritium containment and leak detection.

11. TRITIUM FUELLING AND REPROCESSING

Tritium inventories for liquid, gas and solid phases; tritium loss and leak detection; external tritium leakage and contamination; all remote handling controls; decay heat monitoring; fuelling flow rates; impurity assay.

12. OUTER SUPPORT STRUCTURES

Temperatures; stress profiles; operating radiation levels; induced radioactivity; tritium contamination.

(Table IX.2-1 continued)

13. INNER CONTAINMENT BUILDING

Temperatures; pressures; air flow conditions; operating radiation levels; radioactivity levels; tritium leakage and contamination.

14. HEAT TRANSFER CYCLE/ELECTRICITY PRODUCTION

Pb-Li - pressure; temperatures; flow rates; tritium inventories; radiation levels; decay heat conditions.

HB-40 intermediate loop - pressure; temperatures; flow rates; tritium leakage and contamination.

Steam generators/turbines - conventional instrumentation; tritium leakage and contamination.

IX.2.3 Neutron Diagnostics

The neutron diagnostic requirements for TASKA present a degree of sophistication somewhat between those for the conventional process control and the diagnostic instrumentation for the other plasma parameters. Neutron spectrometers, by definition, are designed to operate in neutron radiation fields. However, the D-T neutron environment of TASKA presents flux levels

which are several orders of magnitude greater than those encountered in other applications of neutronics instrumentation to date (e.g., accelerator environments, neutron benchmark experiments, neutron cross section determinations, etc.).

The three main neutron diagnostic parameters of interest are the total neutron yield, the spatial variation of neutron emission intensity across the plasma and the energy distributions of neutrons originating from localized regions of the plasma.\* The first of these measurements is required in the determination of the overall energy balance of the machine (i.e., system Q), while the second provides information on the spatial density variation of the plasma profile. The third neutron diagnostic parameter, i.e., high resolution spectrometry, leads to the determination of local ion temperatures from the Maxwellian-broadened modulation of output neutron line spectra.

Problems which will be encountered in TASKA and which must be surmounted for successful neutron measurements follow below; in this connection it is interesting to note that there are apparently no radiation damage problems for neutron diagnostic instrumentation:

1. Very high radiation fields can lead to signal "pile-up" in active neutron detectors. Stringent collimation is thus required (see also point 8 below) (continued).

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\* Time dependent neutron spectrum measurements are not considered to be important here since TASKA is a steady state device; neutron detectors which give a fast time response may, however, be useful during startup.



2. Many active neutron detectors are equally sensitive to gamma radiation.<sup>(4)</sup> Some form of effective pulse-shape discrimination is, therefore, required.
3. Active detectors, especially those containing photomultipliers, must be effectively shielded from the high magnetic fields.
4. Similarly, detectors must be shielded from induced voltages resulting from dB/dt effects during startup.
5. Neutron spectrometers must be well shielded from room-return and scattered neutrons. Again, effective collimation must be employed together with non-radiatively absorbing shield material, e.g.,  $\text{Li}_2\text{Co}_3$  (see also point 8 below).
6. Sensitivity of active detectors to mechanical vibration. This is especially important for ion chambers and  $^3\text{He}$  proportional counters.
7. Spurious D-D and D-T neutron production from the neutral beam dumps (see Section VI.3). Such neutron production presents a serious problem from the point of view of spectral measurements. It is, however, of no consequence in a total neutron yield determination, since it represents just another output term in the determination of the overall Q of the device.
8. Access and penetrations. Neutron spectrometer systems in a DT reactor environment such as this will require shielding enclosures comprising concrete,  $\text{Li}_2\text{O}_3$ , paraffin wax, etc., with a total mass in the region of several tonnes, together with collimation paths of the order of meters. Thus, the detector systems must be situated well back from the z-axis of the machine and, for the central cell, must be located outside of the superconducting magnets. This, therefore,

results in a considerable number of line-of-sight penetrations\* to the plasma, especially in the case of the profile monitor system where a typical design will require an array of approximately twenty radial penetrations in the x-y plane (see for example Jarvis, et al.<sup>(5)</sup> and Hendel and Seiler<sup>(6)</sup>).

Table IX.2-2 summarizes the neutron diagnostic requirements for TASKA and indicates the appropriate instrumentation. Such instrumentation is either currently available or under development. In this respect, neutron diagnostic instrumentation is considerably further advanced than diagnostic instrumentation for other plasma parameters (see the section following).

It should be noted in connection with Table IX.2-2, that the range of neutron fluxes encountered in TASKA will span  $\sim 8$  orders of magnitude, from low power startup to full power. Neutron yields from the end cells of the machine during normal operation will always be at least an order of magnitude below those in the central cell.

#### IX.2.4 Plasma Diagnostics

By contrast to both the conventional process instrumentation and the neutron diagnostic instrumentation for TASKA, the requirements for the plasma diagnostics present a much more formidable problem. During the preliminary testing stage of TASKA, a pure hydrogen fueled phase is envisaged. This phase would be able to make use of the extensive plasma diagnostic techniques

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\* Although these penetrations must be line-of-sight, they do not, of course, require evacuating and may be sealed by the vacuum vessel or by stainless steel windows of  $\sim 1$  cm thickness.

Table IX.2-2  
Neutron Diagnostic Instrumentation for TASKA

| NEUTRON MEASUREMENT TYPE                                                                    | SUGGESTED INSTRUMENTATION                                                                                                                                                                                                                                                                                                                                                   |
|---------------------------------------------------------------------------------------------|-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| 1. Total time integrated yield<br>(total fusion reaction rate<br>and Q determination)       | $^{235}\text{U}/^{238}\text{U}$ fission chambers;<br>foil activation with<br>pneumatic rabbit system;<br>$\text{BF}_3$ long counters                                                                                                                                                                                                                                        |
| 2. Time dependent neutron flux<br>(startup reaction rate monitor)                           | Fission chambers; $\text{BF}_3$ long<br>counters; (time resolution~<br>~ 1 ms)                                                                                                                                                                                                                                                                                              |
| 3. Radial profile determination<br>(spatial profile of reaction<br>rate)                    | Collimated x-y array of<br>either fission chambers<br>or NE 213 scintillators.                                                                                                                                                                                                                                                                                              |
| 4. High resolution*<br>neutron spectrometry<br>(localized ion<br>temperature distributions) | $\text{D}(\text{d},\text{n})^3\text{He}$ neutron spectrum: -<br>$^3\text{He}$ proportional counter<br>(2%); NE 213 scintillator<br>(6%); $^3\text{He}/\text{Si}$ -surface<br>barrier sandwich (2%);<br>proton recoil telescope<br>(3%)<br><br>$\text{T}(\text{d},\text{n})^4\text{He}$ neutron spectrum: -<br>NE 213 scintillator (4%);<br>proton recoil telescope<br>(1%). |

\* Figures in brackets after each spectrometer here indicate the best resolution obtained with that detector type.

which are currently available and employed on present day test devices. However, the D-T phase of operation will require a considerable advance in the "hardening" aspects of diagnostic instrumentation. Many types of current plasma diagnostic instruments would not survive the reactor-like environment of TASKA while others simply would not work (e.g., soft X-ray measurements for purposes of plasma electron temperature or impurity determination would be swamped by the large low energy gamma flux present in the system).

The major aspects of the TASKA fusion environment which place severe constraints on the plasma diagnostic instrumentation can be summarized as follows:

1. Very high neutron and gamma first wall fluxes of the order of  $8 \times 10^{13} \text{ cm}^{-2} \text{ s}^{-1}$  (14 MeV) and  $10^{13} - 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$  ( $\sim 0.5$  MeV), respectively, with attendant severe radiation damage and activation problems.
2. Very high X-ray fluxes of  $\sim 10^{15} - 10^{16} \text{ cm}^{-2} \text{ s}^{-1}$  in the region of 10-100 keV and 0.5-2.5 keV resulting from bremsstrahlung and soft X-ray processes in the plasma, respectively.
3. Charged fusion products (mainly 3.5 MeV alpha particles) and charge exchange neutral particles with intensities  $\sim 3 \times 10^{10} \text{ cm}^{-2} \text{ s}^{-1}$  and  $\sim 10^{15} - 10^{16} \text{ cm}^{-2} \text{ s}^{-1}$  respectively. (These are only a problem to direct line-of-sight plasma instrumentation with evacuated penetrations).
4. High in-situ temperatures of up to 440°C.
5. High direct heating rates from incident particle fluxes (greater problem for direct line-of-sight instrumentation).
6. Very high magnetic fields in the region of several teslas.
7. Beta-decay from tritium inventory causing serious perturbation of

charged-particle sensitive instrumentation (only a problem for instrumentation requiring a vacuum access to the plasma).

It is interesting here to note that plasma diagnostic instrumentation requirements for INTOR, a fusion device of approximately the same time frame as TASKA, have recently been reported.<sup>(2)</sup> Again, the limitations of present day instrumentation in D-T power environments were emphasized. In addition, it was expected that the rather limited list of suitable instruments which are currently available would remain basically unchanged until experimentation in current fusion devices (e.g., TFTR, ISX, JET, JT-60, etc.) begin to add to the list. The results from such experiments will also have direct applicability for improved specification of TASKA plasma instrumentation.

Diagnostic requirements for MFTF-B have also been recently outlined.<sup>(3)</sup> Even though this machine will only be D-D fueled and will not include a D-T phase of operation, proposed plasma diagnostics which will be situated near the plasma must withstand power fluxes of  $\sim 100 \text{ W cm}^{-2}$ , enough to seriously damage many materials.

Table IX.2-3 summarizes the major plasma diagnostic requirements for TASKA. The various problems associated with the operation of each of the suggested diagnostic methods in the D-T environment of TASKA are documented in footnotes at the end of the table. The diagnostic instrumentation which is most at risk is seen to be those methods which require a direct line-of-sight vacuum access to the plasma (e.g., charge exchange analyzers, bolometers, UV spectrometers, ion beam probes., etc.) or which must be placed on, or across, system field lines (e.g., electrostatic end loss analyzers, etc.). Instrumentation operating via "optical" signals (e.g., laser scattering

Table IX.2-3  
Plasma Diagnostic Requirements for TASKA

(Footnotes documenting the problems of operating the suggested diagnostic techniques in the D-T reactor-like environment of TASKA are included at the end of the table).

| <u>PLASMA PARAMETER</u>                                                                            | <u>POSSIBLE DIAGNOSTIC METHODS</u>                                                                                                       |
|----------------------------------------------------------------------------------------------------|------------------------------------------------------------------------------------------------------------------------------------------|
| 1. Ion temperature profile -<br>central cell and end plugs                                         | Charge-exchange analyzer <sup>A,F,G,H</sup> ;<br>reaction ion analyzer <sup>B,F,G,H</sup> ;<br>neutron methods (see above);              |
| 2. Plasma density profile -<br>central cell, barriers<br>and end plugs                             | Reaction ion analyzer <sup>B,F,G,H</sup> ;<br>neutral beam attenuation<br>detector <sup>A,E,H</sup> ; neutron techniques<br>(see above); |
| 3. Electron temperature pro-<br>file - central cell,<br>barriers and end plugs                     | Doppler shift of Thomson-scattered<br>laser light <sup>C</sup> ; continuum soft X-ray<br>detector <sup>D</sup> ;                         |
| 4. Electron density profile<br>- central cell, barriers<br>and end plugs                           | Microwave and far-IR interferometry <sup>C</sup>                                                                                         |
| 5. Electrostatic potential<br>profiles - barriers and<br>end plugs relative to the<br>central cell | Ion-beam probes <sup>A,G,H</sup> ;                                                                                                       |

(Table IX.2-3 continued)

|                                                                                |                                                                       |
|--------------------------------------------------------------------------------|-----------------------------------------------------------------------|
| 6. Plasma beta - central cell and plugs                                        | Diamagnetic loops <sup>H</sup>                                        |
| 7. Radiated power (for Q determination) - central cell, barriers and end plugs | Neutron methods (see above); bolometers; radiometers <sup>A,F</sup> ; |
| 8. Plasma impurity content - central cell, barriers and end plugs              | Line X-ray detector <sup>D</sup> ; UV spectrometers <sup>A,F</sup> ;  |
| 9. Spectra of ions to direct convertor/end dump                                | Electrostatic end loss analyzer <sup>B,F,G,H</sup> ;                  |
| 10. Plasma instabilities - central cell, barriers and end plugs                | Radiofrequency probes <sup>A,H</sup> ;                                |
| 11. Background gas density - central cell, barriers and end plugs              | Residual gas analyzer <sup>F,G</sup> ;                                |

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A. Requires direct line-of-sight vacuum access to plasma.

(Table IX.2-3 continued)

- B. Requires positioning on, or across, system field lines - high particle impact rate.
- C. Direct line-of-sight not required here but associated mirrors and windows subject to high neutron, gamma and X-ray fluxes; attendant problems of high heat loads and radiation darkening.
- D. Soft X-ray detector will be swamped by low energy gamma ray flux from system; additional contamination of continuum spectrum by line radiation from impurities.
- E. High incident neutral beam power density - high heat loads.
- F. Signal perturbation due to tritium beta decay radiation.
- G. Large neutron/gamma noise background for ion spectrometers.
- H. High direct particle heat fluxes; radiation damage problem.

methods, far-IR interferometers, etc.) can take advantage of stepped-port penetrations with reflection mirrors at junctions and vacuum seal entry windows. Although these latter folded-path techniques have the dual advantage of much less neutron and X-ray streaming to the detectors and no tritium beta ray contamination, there will be considerable damage and heat loads to the mirrors, windows and seals which face the plasma chamber. An additional consideration for direct-path instrumentation is that it will probably be integrated with each blanket or system module and will, therefore, require lifetimes consistent with the module changeout time. By contrast, the primary transducers in the folded-path diagnostic systems can be situated in shielded enclosures, well away from the main plasma and radiation environments. Maximum fluence or heating rate specifications for such instrumentation would, therefore, be comparable to, or less than, those for the superconducting



magnet systems, with the added advantage of ease of access and replacement.

As a summary to this section it can be seen that, at present, there are few plasma diagnostic techniques which can be successfully integrated with the reactor-like environment of TASKA. There is thus an urgent need for the development of fully-engineered and hardened plasma instrumentation in order that diagnostic requirements for the next generation of D-T fusion devices (e.g., TASKA, INTOR, FED, TMNS, NET, etc.) can be satisfied.

### References for Section IX.2

1. U.S. INTOR Group "U.S. INTOR - the U.S. Contribution to the International Tokamak Reactor Phase-1 Workshop", U.S.A. INTOR/81-1, Chapter XII, 13-15 (1981).
2. Ibid, Chapter XII, 1-21.
3. See, for example, Simonen, T.C. and Leppelmeier, G.W., "Diagnostic Instrumentation for Mirror Confined Plasmas", Energy and Technology Review, UCRL-52000-80, Lawrence Livermore National Laboratory (1980).
4. See, for example, Perkins, L.J., "The Application of Pulse Shape Discrimination in NE 213 to Neutron Spectrometry", Nucl. Instr. Meth. 166, 451 (1979).
5. Jarvis, O.N., Uttley, C.A., and Swinhoe, M.T., "Neutron Diagnostics Design Report - JET Diagnostic Design Study 10.2/3(a)", AERE G1644, UK Atomic Energy Research Establishment, Harwell, UK (1980).
6. Hendel, H.W. and Seiler, S, "TFTR Diagnostic Physics Review", Neutron Diagnostics, p. 120, Princeton Plasma Physics Laboratory (16-17 November 1978).

## X. Maintenance

### X.1 General Aspects and Plant Layout

Successful operation of TASKA can only be achieved if the shutdown time required for maintenance operations, as well as the radiation hazard for the maintenance personnel are kept to a minimum. As is usual in nuclear plants, both requirements lead in most cases to opposite technical solutions and management procedures. This is due to the fact that remote handling operations of a medium or high degree of complexity are up to one order of magnitude more time consuming than contact operations.

The environmental conditions in the vicinity of TASKA after shutdown are characterized by

- low neutron induced radioactivity of the outer structures;
- possible airborne tritium contamination;
- possible residual magnetic fields.

It is anticipated that these conditions allow limited personal access to the major part of the components one day after shutdown if they are in place.

In addition to this, hot areas have to be anticipated in the course of time resulting from

- neutron streaming through openings (vacuum, coolant, injection, diagnostic, etc.) and unavoidable mounting interstices in the shields;
- radioactive deposits in the coolant and auxiliary circuits after long operation or accidental coolant contamination;
- accumulated outer surface contamination in the facility after repeated handling of highly activated components or after unexpected activity release incidents.

Experiences from nuclear fission reactor operation show that these factors tend to yield additional restrictions for direct personnel access.

Finally, there are two additional important aspects which have to be taken into account for achieving a sufficiently high availability of the plant:

- The reactor hall has to be evacuated of personnel when parts of the shielding are to be removed or failed radioactive components are to be replaced.
- Personnel access to the building is restricted after accidents with a high radioactivity release which cannot be removed by repeated air exchange or confined by remotely placed shielding.

In existing nuclear fission reactors and fuel cycle facilities similar problems can often be limited by subdividing the total system into groups of components and placing them into various separately sealed rooms or shielded cells. In the case of TASKA with its size and shape and its multitude of directly attached components this is hardly possible. Thus, the accessibility of the whole plant suffers considerably if scheduled or unscheduled high radiation fields exist or occur in the reactor hall during maintenance or repair periods.

As a result the maintenance concept for the facility has to be based on

- maximum contact procedures;
- option of remote repair or replacement.

The latter principle dominates as far as the general layout of the plant, the design of the components and the maintenance equipment. The first is important for the shielding thickness and some less decisive plant characteristics.

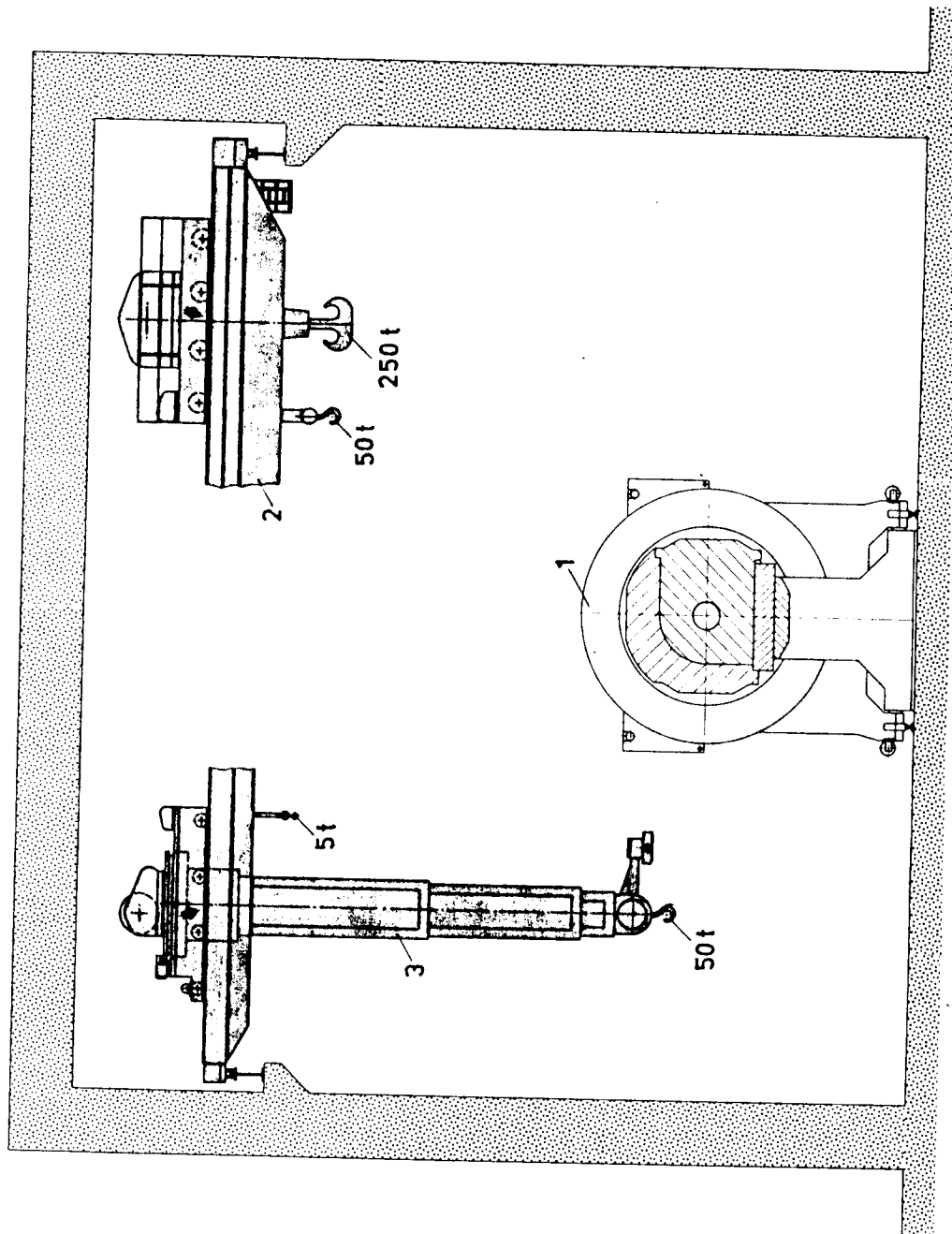
In order to meet the requirements for at least the necessary remote operations, the reactor building for TASKA has to have typical features of a large hot cell (Fig. X.1-1) for indirect remote handling and viewing.

The building walls have to reduce the radiation level to such an extent so as to permit the continuous presence of personnel outside, both under normal operation as well as under high active component replacement or accident conditions. In addition they have to protect the outside environment against accidental tritium release and mechanical impacts resulting from potential accidents in the plant.

A remotely controlled overhead bridge crane is scheduled for handling the heavy items. A second crane bridge with a telescoping mast and a cantilever arm at its lower end acts as a handling and positioning system for manipulators and components. It is capable of exerting the considerable horizontal forces required for extraction and insertion of plug-in devices, e.g. blanket modules. Typical design parameters for both main handling machines are listed in Tables X.1-1 and X.1-2. The carriages can be driven off the bridge through a shielded door into a maintenance position outside the reactor hall for maintenance during plant operation.

The TASKA reactor and its components are arranged on the floor of the reactor hall in such a way as to permit maximum access by the handling machines from above and both sides. They have to be designed in such a way that the main handling operation can be performed as much as possible by a sequence of simple vertical and horizontal movements. The required high precision of the positioning procedures can be achieved by carefully controlling operating speeds and exerted forces.

- 1 Test facility
- 2 Bridge crane
- 3 Component handling machine



**Fig. X.1-1 TASKA handling system**

Table X.1-1. Typical Design Parameters for the Overhead Bridge Crane

|                             |  |       |
|-----------------------------|--|-------|
| - Load carrying capacities: |  |       |
| main lifting device         |  | 250 t |
| auxiliary lifting device    |  | 50 t  |
| - Lifting/lowering heights: |  |       |
| main lifting device         |  | 25 m  |
| auxiliary lifting device    |  | 25 m  |

Table X.1-2. Typical Design Parameters for the Component Handling Machine

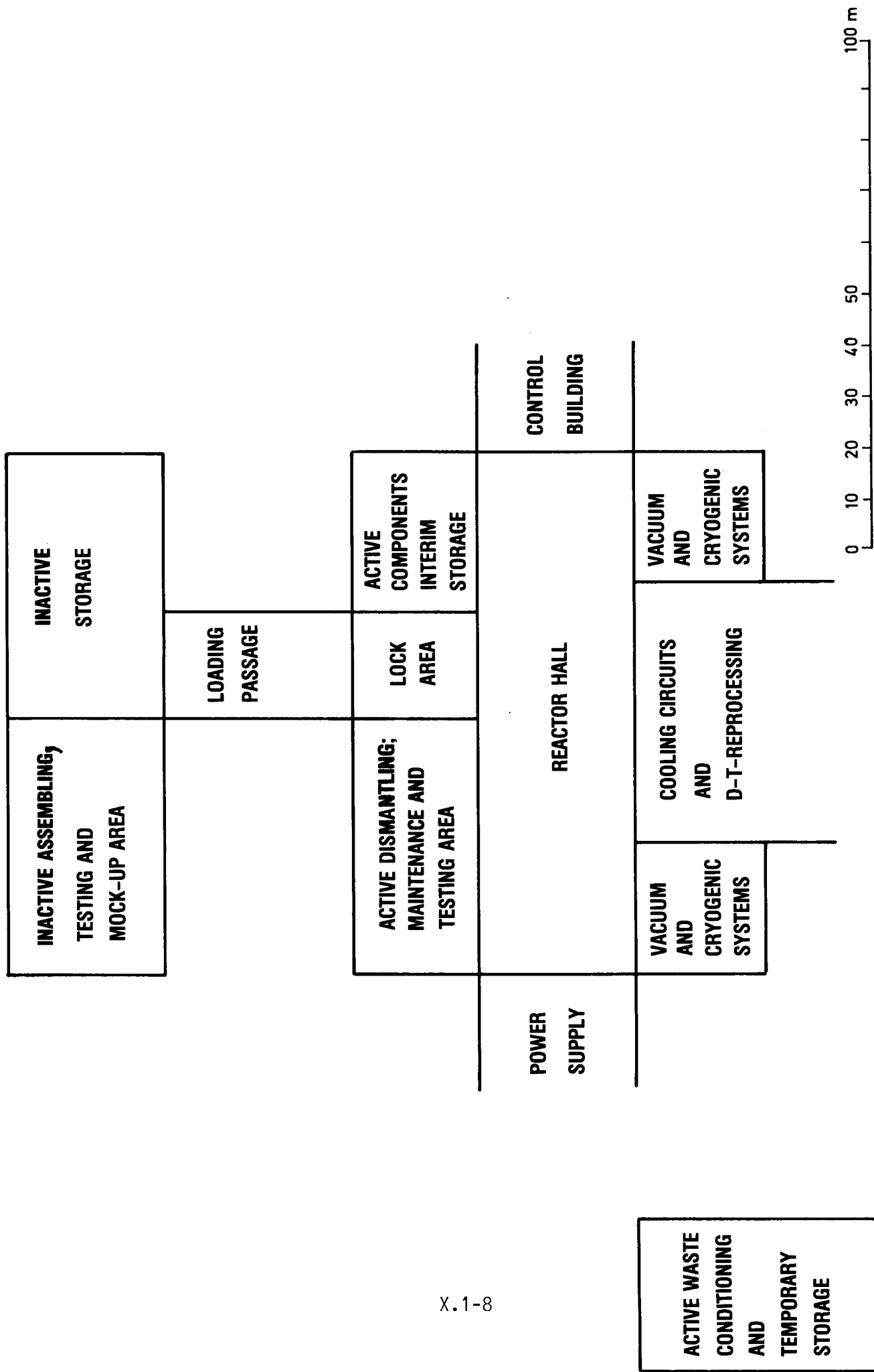
|                                                      |  |        |
|------------------------------------------------------|--|--------|
| - Load carrying capacities:                          |  |        |
| lifting telescope                                    |  | 50 t   |
| auxiliary lifting device                             |  | 5 t    |
| - Maximum torque at the lower telescope tilting head |  |        |
|                                                      |  | 10 t-m |
| - Lifting/lowering heights:                          |  |        |
| telescope                                            |  | 15 m   |
| auxiliary lifting device                             |  | 25 m   |



For other remote operations such as bolting, screwing and delicate handling, powered wrenches and manipulators are available.

In order to facilitate the necessary remote connecting and disconnecting procedures nondestructive techniques are to be applied as much as possible. These operations are principally well known from remotely maintained hot fission fuel processing plants; however, extensive development work is still needed to meet the high precision and reliability requirements. Furthermore, while destructive disconnecting operations such as cutting by mechanical, thermal or electric means should not present principal difficulties, remote welding, including edge machining and weld inspection, are probably the most important maintenance and repair techniques still to be developed.

If one tries to analyze the large variety of various inspection, repair and post-repair testing operations required, it appears that only a comparatively small part of the extensive maintenance procedures should or can be performed in-situ or inside the reactor hall. Replacement by spare components, or dismantling, repair outside the reactor hall and reintroduction will be a more suitable maintenance management. Therefore, a well equipped hot cell facility with several compartments, mechanical master-slave manipulators and direct viewing through shielding windows is needed for radioactive components, adjacent to the reactor hall. In Fig. X.1-2 these cells are scheduled to be located in the active dismantling, maintenance and testing area. In addition to this a shielded interim storage area is required for reusable items. Loading and unloading for both shielded zones can be performed via a common lock area which is connected with the reactor hall and serves in its upper part as a semi-remote or contact maintenance cell for the components of the reactor hall remote maintenance system. It is estimated that the floor space



**Fig. X.1-2 Reactor plant**

needed for this active component maintenance and storage complex is at least about one-half that of the reactor hall.

If a plant like TASKA is not situated near a component development and testing establishment, another building complex has to be provided which serves for developing, assembling and testing of inactive components. It comprises the inactive mockup area, too, which is needed for remote procedures development work, and will be utilized during plant operation for inactive equipment maintenance and storage.

## X.2 Handling Equipment for Repair and Operation

### X.2.1 Generalities

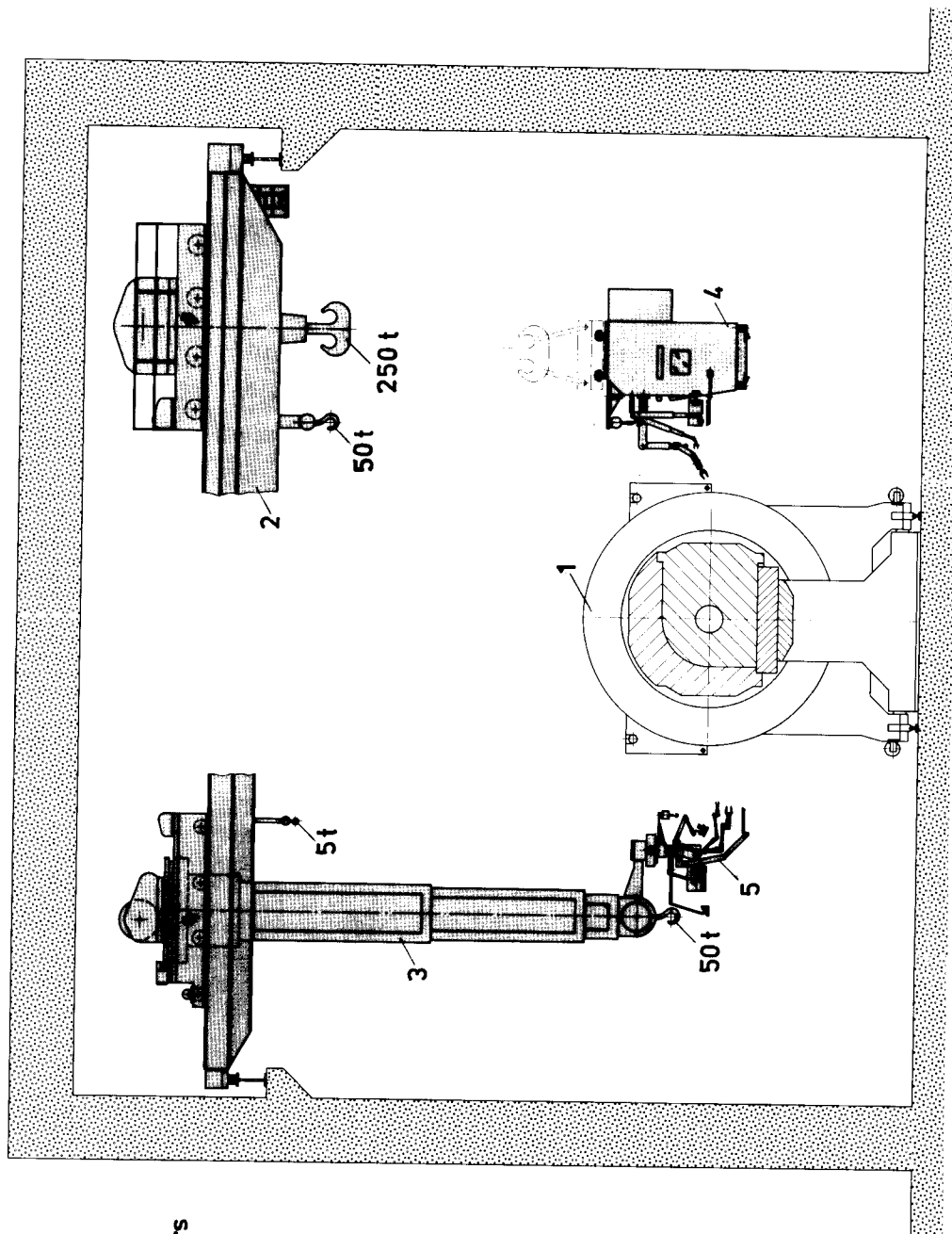
In order to perform repair work and operation in a test facility such as TASKA a great variety of handling tasks must be carried out in the reactor hall under greatly varying environmental conditions.

As long as the reactor shielding is intact provision can be made by appropriate dimensioning and design that hands-on work can be done in-situ by the staff. Assuming a period of e.g., ten hours per month and ten months per year of working time, the mean local dose rate according to the German Radiation Protection Ordinance must not exceed 100  $\mu\text{Sv/h}$  (10 mrem/h). In case the shield must be opened at the work station, the dose rates will attain values which will require the use of remotely handled equipment.

For these reasons a handling system was conceived which comprises tools to be used under both working methods. Since many different tasks with a relatively low frequency of repetition will have to be fulfilled at widely separated places and since unexpected situations must also be anticipated, general-purpose, high-performance tools will be provided for remotely controlled work. The handling of objects varying in weight between many tons and less than 1 kg calls for several tools having highly differing capabilities. Moreover, a limited working capability in high magnetic fields should at least be aimed at in short periods of shutdown. The system (Fig. X.2-1) therefore consists of the following main components:

- for in-situ work by the staff
  - an elevated working platform from which the tools can be operated
  - temporary shielding (not shown on the figure)

- 1 Test facility
- 2 Bridge crane
- 3 Component handling machine and manipulator carrier
- 4 Shielded cabin with manipulators
- 5 Manipulator unit
- 6 Raisable working-platform



**Fig. X.2-1 TASKA handling system**

- for work by staff in-situ or by remote control
  - a movable 250 t bridge crane      for more details see
  - a component handling machine      Section X.1
- for remotely controlled work
  - a remotely controlled manipulator unit (carried by the component handling machine)
  - a shielded cabin with manipulators.

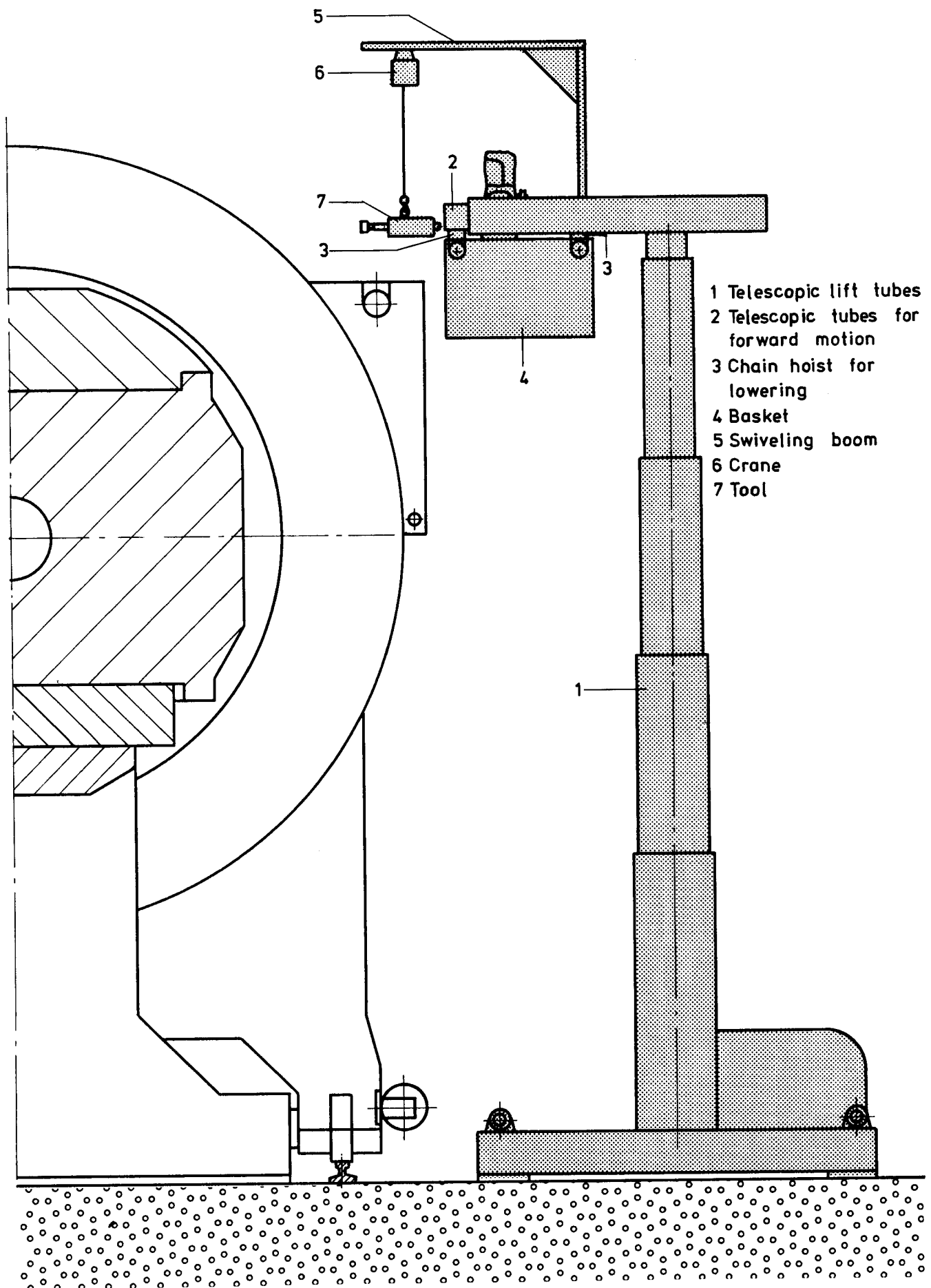
#### X.2.2 Elevated Working Platform

The elevated work platform is transferred by the bridge crane to the respective work station and lowered there.

The platform has been designed on the following principle (Fig. X.2-2): On a floor space a fourfold lifting telescope has been provided; at its upper end a double telescope for forward movement and at the front end of the double telescope a basket can be provided from which one or two people can work. The basket can be lowered in addition by means of chain hoists so that in the lower position of the lifting telescope the working space can be utilized down to the floor.

Practically all types of conventional tools as well as special purpose tools allowing personnel to work at some distance from the reactor can be used by the staff. To balance the weight of heavy tools a swiveling boom with a light lifting device has been provided at the basket.

Hydraulic drives will be provided for the telescopes and an electric motor drive for the light lifting device.



**Fig. X.2-2 Raisable working platform**

### Technical Data

|                           |                          |         |
|---------------------------|--------------------------|---------|
| - Load carrying capacity: | basket                   | 1000 kg |
|                           | light lifting device     | 50 kg   |
| - Reaches of motion:      | lifting telescope        | 8 m     |
|                           | advance                  | 4 m     |
|                           | basket lowering capacity | 2 m     |

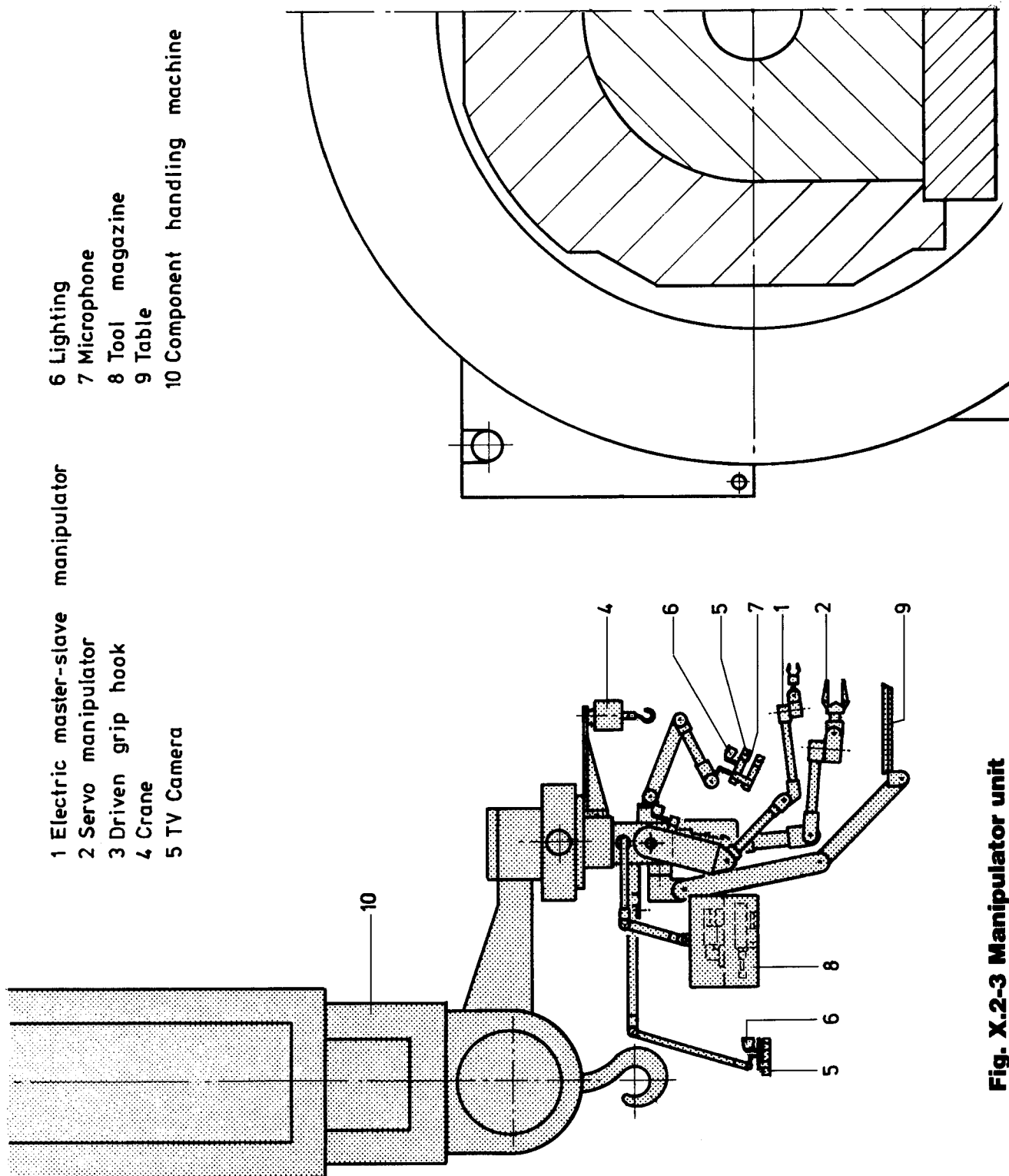
Due to the potential of tritium contamination of the surroundings, the staff working in the reactor hall must wear full protective suits with breathing equipment. It should be taken into account that the associated inconveniences considerably extend the required working times as compared with similar work performed under conditions usually encountered in a workshop.

### X.2.3 Manipulator Unit

When maintenance work is done in a large-scale facility, the situation regarding remotely controlled work, in contrast to work performed in the classical hot cells, is characterized by the fact that instead of the objects being transferred to a work station, the manipulators and the necessary auxiliaries must be brought to the work stations. Since several small carrier systems cannot be provided in a room in which all three dimensions are large, all components must be combined into a single unit in order to become operational by one carrier system. Even simple activities in most cases require at least two manipulators. It is reasonable to provide two arms of different load carrying capacities so that a large scale of different work can be handled.

The manipulator unit (Fig. X.2-3) includes the following components with the following design:





**Fig. X.2-3 Manipulator unit**

- a slave arm of a master-slave manipulator with electric power transmission, arranged on the right side
- a servo manipulator, arranged on the left side
- a driven gripping hoop, placed between the manipulators so as to allow the transport of rather heavy components
- a lifting device fixed at a swiveling boom
- television cameras allowing observation of the work station and lights as well as stereoscopic microphones for noise transmission
- television cameras for viewing the scene and their respective lights
- two swiveling and extendable booms with lights for indirect lighting
- a store installed to an articulated boom to accommodate tools which can be used together with the manipulators
- a table attached to an articulated boom on which equipment devices or components can be carried and deposited.

The manipulator unit is kept ready for use in a holder and can be coupled by remote control to the plate accommodating the component handling machine or deposited again into this holder. All components of the unit can be quickly dismounted and remounted. All drives are equipped with electric motors. Should working in very high radiation fields become necessary the radiation sensitive components would have to be made radiation resistant. The current is supplied from the component handling machine and the signals will be transmitted by radio. The handling and control devices are installed in a separate remote handling room.

#### Load Carrying Capacities of the Equipment:

- |                                     |        |
|-------------------------------------|--------|
| - Electric master slave manipulator | 45 kg  |
| - servo manipulator                 | 150 kg |

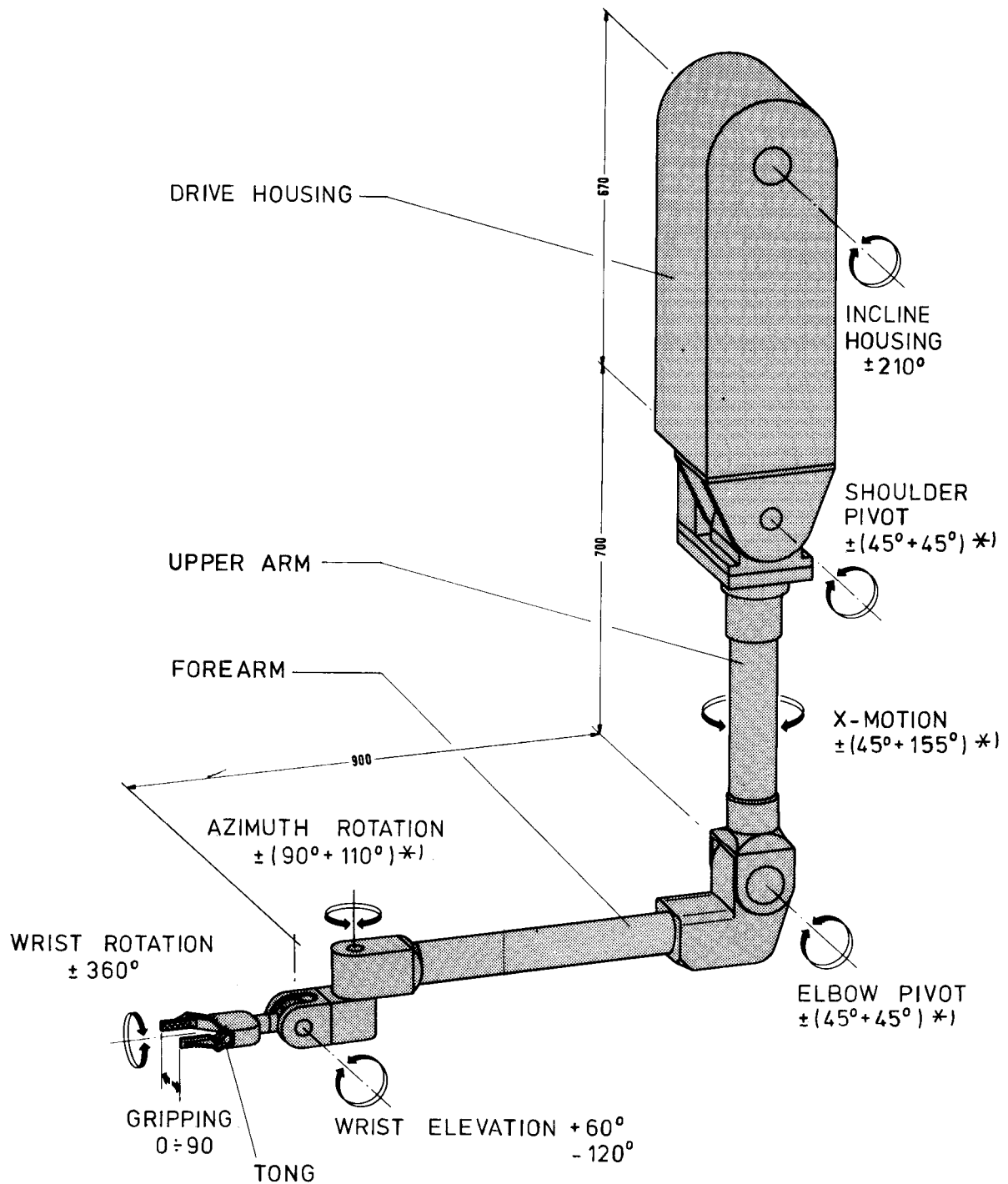
- driven gripping hook 2000 kg
- lifting device 2000 kg

An electric master-slave manipulator (Fig. X.2-4) is particularly suited for complicated and delicate work and for the use of adapted hand tools. The equipment possesses bilateral position control loops for all motions by which force reflection is achieved, which means that the operator gets a true feeling of pressure in his hand corresponding to the forces applied. In the upper load range the forces acting at the slave arm are amplified ten times at the maximum as compared with the master arm. The necessary dead weight compensation is realized by electric motors. The length of the master arm is about 70% that of the slave arm. The tong and the jaws can be replaced by remote handling. Free programmability can be provided as the second mode of operation, e.g., with a view to welding operations.

The servo-manipulator will be used to handle rather heavy objects and for working with heavy-adapted tools. It is an improvement of the switch controlled power manipulators with speed control, intended to increase substantially the working speed. Moreover, the arm has been largely adapted to the electric master-slave manipulator in its basic design so that the possibility of good coordination is ensured. Position controls and a handle for operation will be provided. The tong can likewise be replaced by remote handling.

Both manipulators have seven position-controlled motions each. To extend the reach the driver housings can be inclined by switch control. The electric tools are supplied with current from plugs installed near the wrists.

The effective working speed is much slower when work is done with manipulators as compared to work carried out by staff under conditions prevailing in



\* ) SLAVE AND MASTER ARMS SYNCHRONOUSLY  
+ DISPLACEMENT OF SLAVE ARM RELATIVE  
TO MASTER ARM

**Fig. X.2-4 EMSM-WA electric master-slave manipulator,  
slave arm**

a workshop. If master-slave manipulators are used, four times the time is required on an average and with power manipulators 50 times the time; scattering is high, independent of the degree of difficulty of the work. With the servo-manipulator equipped with the control and handling device about ten times the time will presumably be needed.

The television system and the tools will be dealt with in Sections X.2.5 and X.2.6, respectively.

#### X.2.4 Shielded Cabin With Manipulators

A shielded cabin will offer considerable advantages as an alternative and supplementary system to the remotely controlled manipulator unit. The application of different principles for the manipulators and means of viewing provides a different characteristic in terms of the advantages and limits in application. Subject to the clarification of the effects exerted by magnetic fields on the performance of electric motors, television cameras, etc. and hence of the application of the manipulator unit, the equipment in the shielded cabin can most likely be designed to operate in intense magnetic fields.

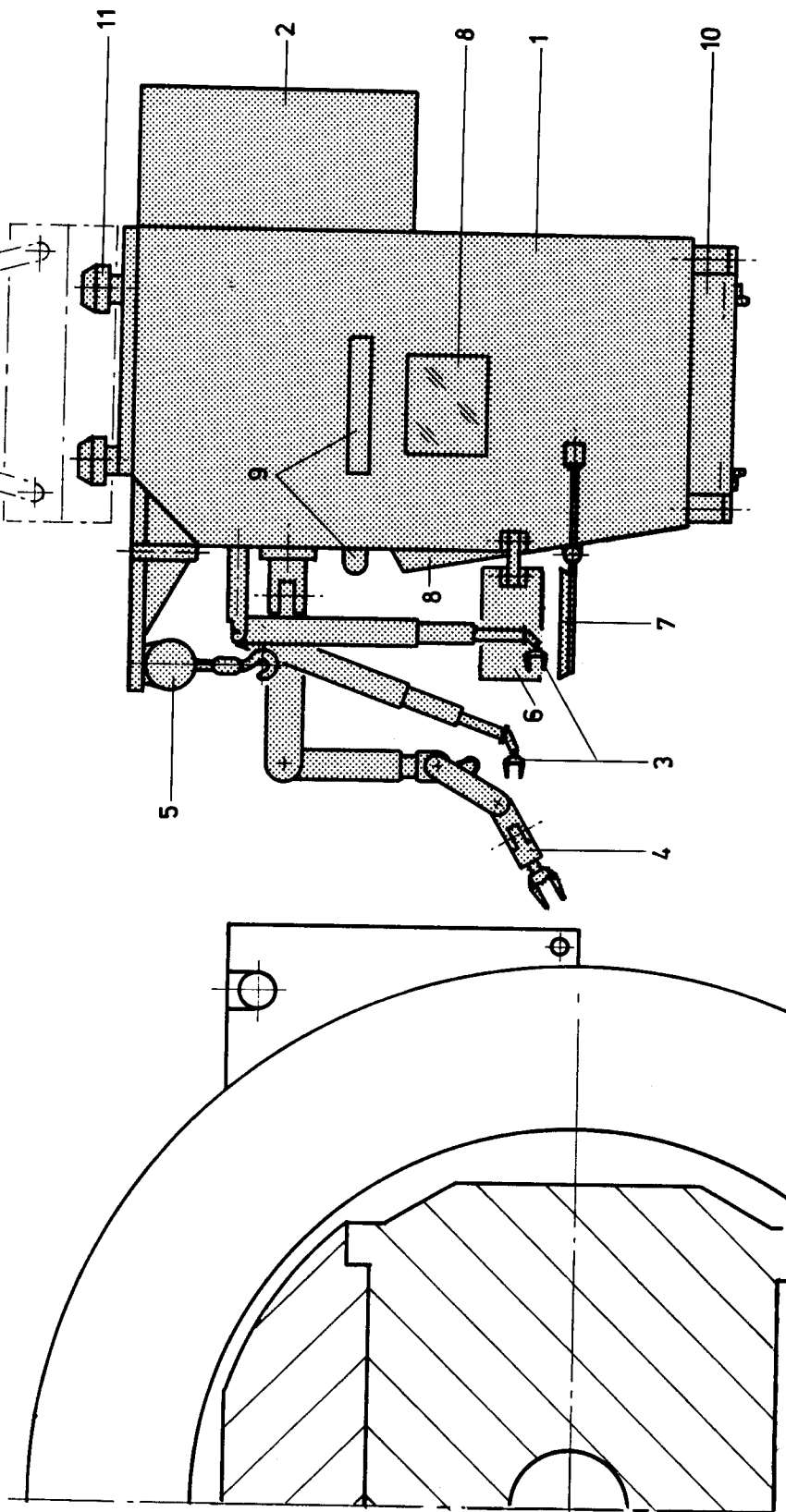
The cabin will be used in the reactor hall where it will be suspended from the bridge crane. In other shielded rooms it can also be moved on rails run on the ground. The effectiveness of the cabin is generally comparable with that of the manipulator unit and roughly corresponds to that available in the conventional hot cells.

The equipment of the cabin (Fig. X.2-5) consists of the following main components:

- two master-slave manipulators with mechanical power transmission
- one power manipulator

1 Shielded cabin  
2 Supply unit  
3 Mechanical master-slave  
manipulators  
4 Power manipulator  
5 Crane  
6 Tool magazine

7 Table  
8 Shielding window  
9 Lighting  
10 Carriage  
11 Attachment device



**Fig. X.2-5 Shielded cabin with manipulators**

- one lifting device fixed to a swiveling boom
- four shielding windows (arranged on the front, left and right sides and in the floor) and the respective lights
- one tool store installed to an articulated boom
- one folding table which can be displaced.

#### Load Carrying Capacities of the Equipment

- |                                                   |            |
|---------------------------------------------------|------------|
| - mechanical master-slave manipulators, tong/hook | 20/50 kg   |
| - power manipulator, tong/hook                    | 250/400 kg |
| - lifting device                                  | 2000 kg    |

#### Technical Data of the Cabin

- |               |                                  |                |
|---------------|----------------------------------|----------------|
| - Shielding:  | material                         | lead           |
|               | thickness of lateral faces, rear |                |
|               | wall and ceiling                 | 150 mm         |
| - Other data: | crew                             | one or two men |
|               | total weight                     | about 95 t     |

Four mushroom shaped suspension heads for the crane gear have been provided at the ceiling of the cabin. The motion drive is via an electric motor with redundancy. Considering the mode of operation on rails, a power supply via storage batteries is preferred. The staff enter the cabin and leave it in the service cell. The door has been provided on the rear side. There is an intercom from and to the control center in the remote handling room. The ventilation system ensures a slight overpressure in the cabin in order to prevent contamination from entering it. Optionally, television cameras also can be used as an additional means of observation.

Mechanical master-slave manipulators serve the same purposes as the manipulators with electric power transmission. Each of the two manipulators

possesses seven motions with force reflection which allow transmission of the motions of the operator's arms. In three of these motions the slave arm can be adjusted in addition by electric motor control by switches, contrary to the master arm, in order to be able to extend the reach of work. Arms of rugged design will be provided in which the torques will be transmitted by shafts and tooth gears. The tongs and the jaws can be replaced remotely.

The power manipulator possesses nine motions with speed control. It is operated via master switches and the speeds can be varied continuously. The gripping force of the tong and the torque of the hand rotation motion can be preselected. The tong can be replaced by remote handling.

Both manipulators will be equipped with hydraulic drives for work in intense magnetic fields.

#### X.2.5 Television Camera Systems

To be able to work rapidly and safely with a remotely controlled manipulator unit the production of stereoscopic images is indispensable. The stereoscopic systems should comprise two channels in order to avoid the vertical image resolution or the horizontal angle of viewing being cut in half. Color television systems will be provided since they allow much better visibility than black and white systems. Color systems of the German standards operate at 625 lines, 50 Hz and in the PAL-B mode.

The system for observing the work station possesses one pair of stereoscopic television cameras and two additional monoscopic cameras for enlarging the viewing field in the horizontal direction. The survey system furnishes a picture of the whole scene so that the manipulators can be moved safely over large spaces and also in case of restricted space. This system likewise comprises one pair of stereoscopic cameras. The cameras of both systems are



mounted on articulated booms so that their positions relative to the manipulators can be varied in order to select the desired viewing angle and the distance of observation.

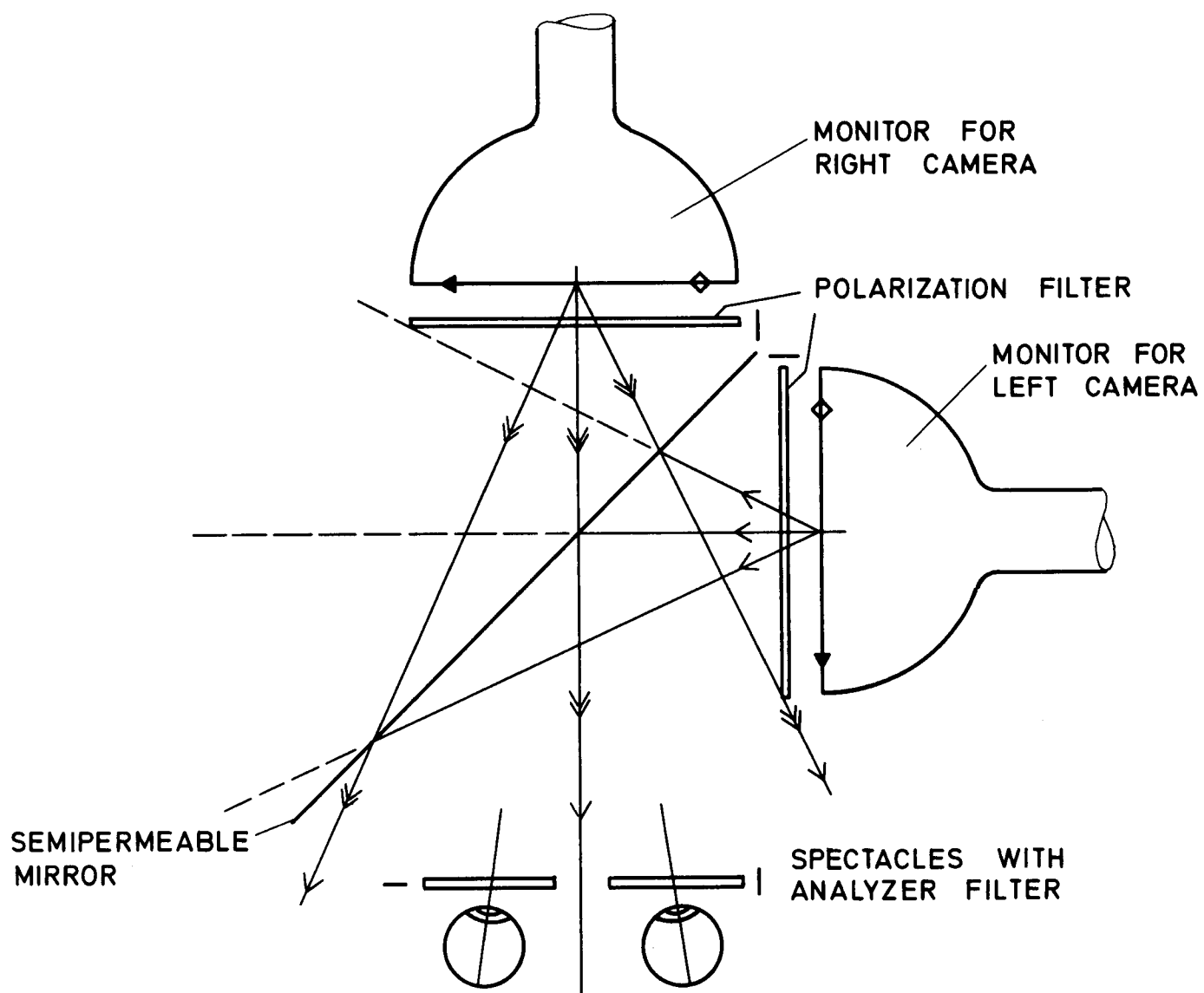
With a view to the dimensions, single-tube color cameras will be provided. The diameter of the camera tubes is 1" and the stereoscopic base is 130 mm. All cameras will be equipped with zoom lenses with a maximum horizontal viewing angle of 45° and a focal length ratio of 1:6.

It is proposed to use monitors with a diagonal image of at least 51 cm.

To display the stereoscopic images the polarization method was selected in the test because it produced the best image quality (Fig. X.2-6). Moreover, this method requires few special auxiliaries. The monitors have been arranged at right angles to each other. Filters have been placed in front of the screens which act as vertical and horizontal polarizers, respectively. Between the monitors a diagonal semi-permeable mirror has been installed which passes the image from the monitor installed in front of the operation and reflects the image of the other monitor into the direction of view of the operator. The operator wears spectacles with appropriate polarizing filters so that the left eye sees only the picture taken by the left camera and the right eye only the picture taken by the right camera.

#### X.2.6 Tools for Manipulators

Together with the manipulators described, a great variety of adapted tools can be used which must be actuated together with a manipulator. Moreover, almost all types of driven tools can be used such as impact wrenches, drilling machines, compass saws, circular saws, angle sanders, cutting tools or hammers. Depending on whether these tools will be used together with the manipulator



**Fig. X.2-6 Viewing equipment for stereo television**

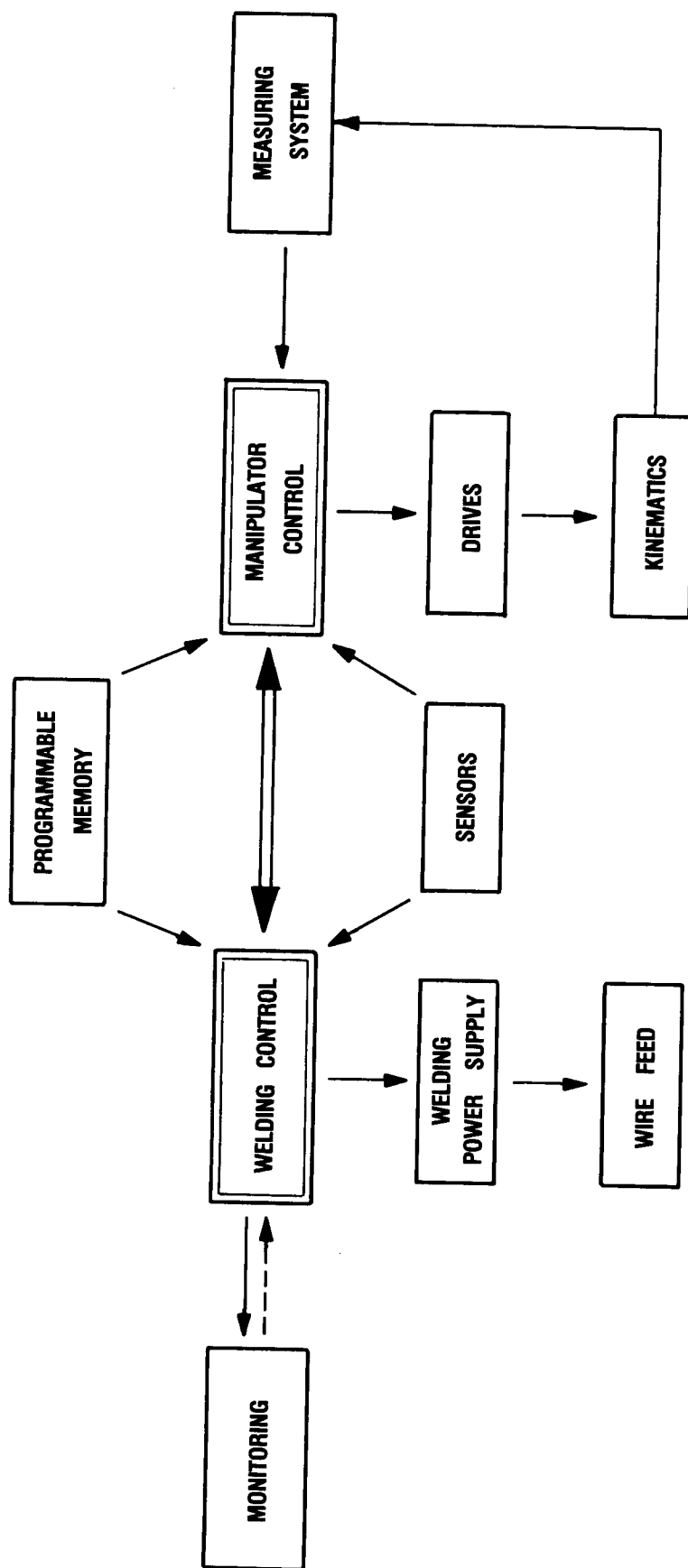
unit or with the manipulators of the cabin the drives are by electric motor or hydraulic.

In cases where a connection of plant components through a bolted flange is not possible, seam welding must be performed remotely. Four main working steps are required to replace a welded component:

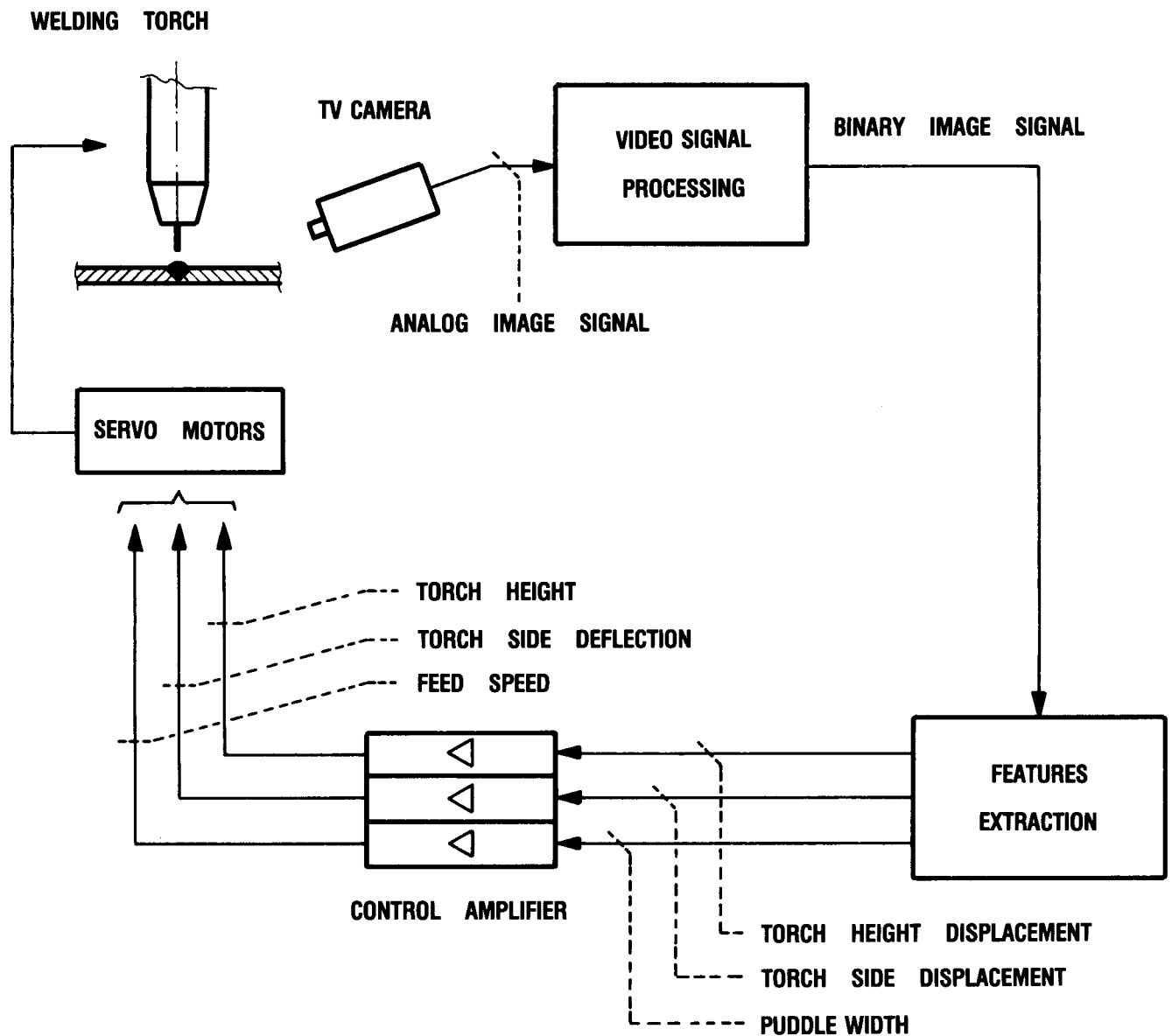
- for disconnecting, careful mechanical or thermal separation, and
- for connecting, and preparation of the joint, welding of the seams, and, if applicable, inspection of the welded seam.

Of the four operations, seam welding is the most difficult. To obtain satisfactory quality, the welding process must be automated. Two methods are applicable in principle. Either a device performing the welding process automatically is transferred to the work station via a manipulator and fixed there or the manipulator is freely programmable and moves the welding head along the seam.

In the first method, the welding device for large components could be, for example, a small carriage approaching the component on rails. For welding pipework rotating heads and tongs with a rotating unit made with the conventional technology are currently used. In the second method the manipulator control would have to be backfitted accordingly for automated welding (Fig. X.2-7). Because of the required tolerances of the component on the one hand, and the accuracy of positioning of the welded head of about  $\pm 1$  mm, on the other, satisfactory results can only be obtained with correction of the track provided by sensors. For this purpose, tactile sensors, induction sensors and television cameras have so far been used. Use of a television camera (Fig. X.2-8) has the advantage that several sensors are not required for the sizes to be covered and the design of the plant components can be much less



**Fig. X.2-7 Components of a free programmable manipulator with equipment for arc welding**



**Fig. X.2-8 Block diagram for welding control circuit  
with TV camera as sensor**

sophisticated at the work station. The molten pool and the wire end are observed with the camera. The image is evaluated by methods of pattern recognition in order to obtain the correction signals. Depending on the deviations from the nominal geometries, the welding head height and the lateral position as well as the rate of advancement are controlled.

#### X.2.7 Concluding Remarks

To realize the necessary equipment for remotely controlled work in repair and operation of a facility such as TASKA, one can rely on a broad spectrum of technologies both from the nuclear and the conventional areas. However, since the task is a new one, adaptations and improvements are required. The central topics of development work would include the kinematic systems of the manipulators with drives by electric motors, the control system for the electric master-slave manipulator, and the automated seam welding device. So far no problem can be recognized which does not seem soluble.

In case the proposed versatile remote handling system will be realized and, moreover, the components will be so designed as to meet the needs of remote operation, guarantee convenient access to the active zones and be capable of backfitting, settlement of the problems at hand will be ensured.

### X.3. Maintenance Classification

To get an overview of maintenance tasks as well as necessary equipment major subsystems and components are categorized into the four general classes as shown in Tables X.3-1 and X.3-2. The classes range from operating for the design life of the system without scheduled replacement to short lifetime. Between these categories there are two additional classes with expected lifetimes of more and less than two full power years (fpy). Beyond the distinction dependent on the lifetime we also differentiate between scheduled replacement because of radiation damage and unexpected replacement or repair due to random component failures. In addition to this classification, masses and sizes of main components are shown in Table X.3-3.

To demonstrate the practicability of maintenance tasks four typical cases have been selected; i.e., the exchanges of:

- blankets
- neutral beam injector components
- central cell coil and
- barrier coil, especially the barrier insert coil (see Fig. X.3-1).

Reasons for this selection were the frequency of required replacements in the first two cases and difficulty and complexity of the removal in the latter two cases.

Table X.3-1. Classification of Components With Respect to Lifetime

- Class 1 components (full lifetime)  
Hands-on or remote repair/replacement not necessary, e.g.: building-  
and support structures, parts of the vacuum vessel
- Class 2 components (lifetime  $\geq$  3 FPY)  
Remote repair/replacement complicated but possible during long shutdown  
periods, e.g.: magnet coils
- Class 3 components (average lifetime)  
Remote repair/replacement foreseen during regular shutdown periods,  
e.g.: blanket and material test modules, injector components
- Class 4 equipment (short lifetime)  
Remote replacement foreseen immediately after shutdown or during  
operation, e.g.: handling and experimental equipment

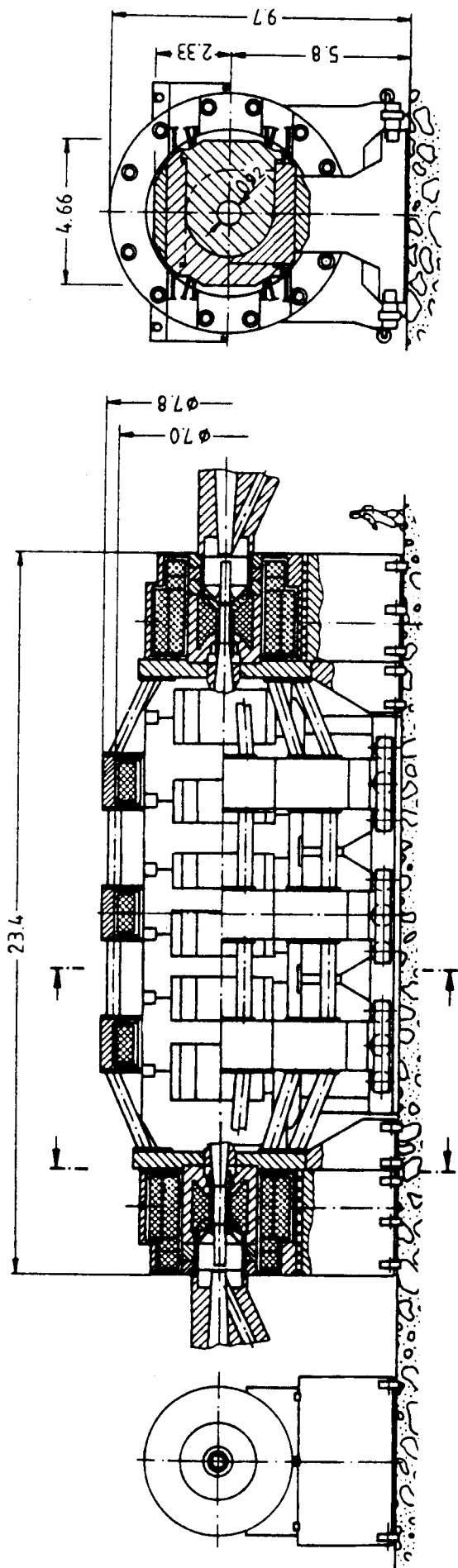


Table X.3-2. Categories of Maintenance With Respect to Lifetime<sup>(1)</sup>

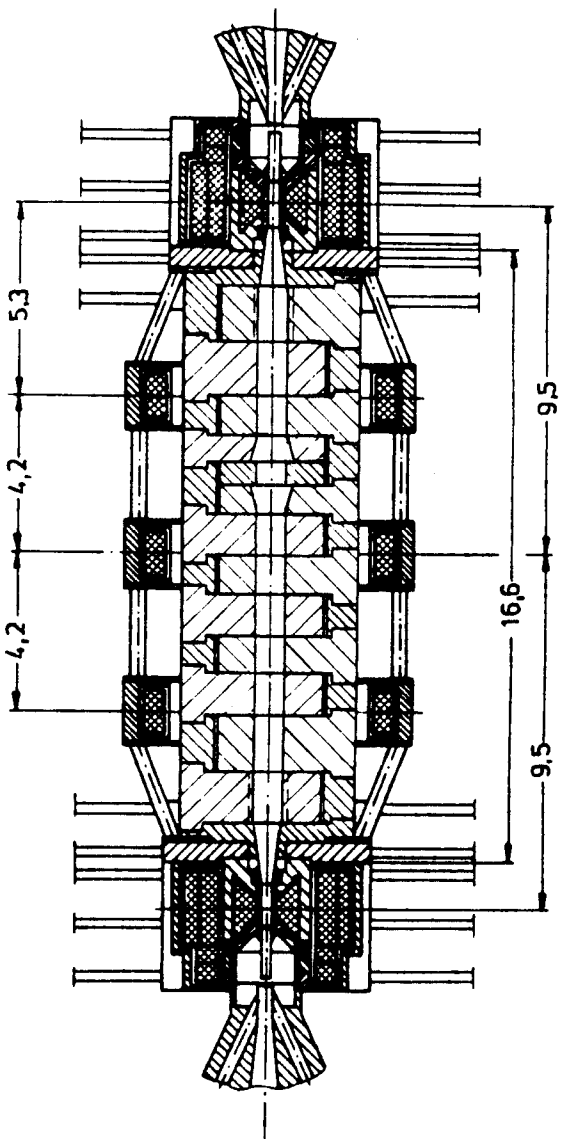
| <u>System</u>                    | <u>Class 1<br/>Full<br/>Lifetime</u> | <u>Class 2<br/>Lifetime<br/>≥ 3 FPY</u> | <u>Class 3<br/>Average<br/>Lifetime<br/>&lt; 2 FPY</u> | <u>Class 4<br/>Short<br/>Lifetime</u> |
|----------------------------------|--------------------------------------|-----------------------------------------|--------------------------------------------------------|---------------------------------------|
| Building and support structure   | X                                    |                                         |                                                        |                                       |
| Shield                           | X                                    |                                         |                                                        |                                       |
| Vacuum Vessel                    | X                                    |                                         |                                                        |                                       |
| Breeder Test Blanket             |                                      |                                         | S                                                      |                                       |
| Permanent breeder blanket        | X                                    | R                                       |                                                        |                                       |
| Material test blanket            |                                      |                                         | S                                                      |                                       |
| Samples of material test blanket |                                      |                                         |                                                        | S                                     |
| Coils                            | X                                    | R                                       | R                                                      |                                       |
| Insert coils                     |                                      | S                                       | R                                                      |                                       |
| Cryopumps                        | X                                    | R                                       |                                                        |                                       |
| Neutral beam injectors total     |                                      | R                                       |                                                        |                                       |
| NBI - ion source                 |                                      |                                         | S                                                      |                                       |
| NBI - ion dumps                  |                                      | R                                       | R                                                      |                                       |
| NBI - getter panels              |                                      | S                                       | R                                                      |                                       |
| ECRH                             |                                      | R                                       | R                                                      |                                       |
| ICRH                             |                                      | S                                       | R                                                      |                                       |
| Beam dumps                       |                                      | S                                       | R                                                      |                                       |
| Diagnostics                      |                                      | SR                                      | SR                                                     |                                       |
| Heat exchanger                   |                                      | R                                       |                                                        |                                       |
| T-system                         |                                      | R                                       |                                                        |                                       |
| Maintenance equipment            | X                                    | R                                       | R                                                      |                                       |

S - scheduled replacement

R - replacement in case of failure



X.3-4



0 1 2 3 4 5 6 7 8 9 10m

Figure X.3-1 Central Cell

Table X.3-3. Main Components - Estimated Sizes and Weights

|                           | <u>Size (m)</u> | <u>Weight (t)</u> |
|---------------------------|-----------------|-------------------|
| Vacuum chamber (section)  | 3.0 x 3.0 x 6.0 | 17                |
| Permanent breeding module | 3.0 x 3.0 x 1.5 | 35                |
| Breeding test blanket     | 3.0 x 3.0 x 1.4 | 50                |
| Central cell coil         | 7.2(dia.) x 1.6 | 127               |
| Barrier coil              | 4.7(dia.) x 3.5 | 250               |
| Transition coil           | 6.8 x 5.0 x 4.0 | 135               |
| Yin-yang coil             | 8.0 x 3.5 x 5.5 | 230               |
| ECRH                      | 1.5(dia.) x 6.0 | 47                |
| Neutral beam injector     | 3.5 x 3.5 x 7.5 | 160               |

References for Section X.3

1. J.N. Doggett et al., Proc. of the 28th Conference on Remote Systems Technology, CRSTBJ 26 1-460 (1978).

#### X.4 Short Lifetime Components

Besides samples inserted in material test blankets which can be exchanged comparatively easily by suitable methods during the operation phase, all the other components in immediate connection with the facility have to be replaced during shutdown periods. This is not only due to radiation, but also due to the high magnetic field in the coils, making access impossible and largely preventing the application of electrical machines.

The most frequently replaced components of the central cell are the test blankets, material test blankets as well as breeding test blankets and of the plug region, the neutral beam injectors, especially the ion sources and the getter panels.

##### X.4.1 Blankets

The blankets inside the vacuum chamber surround the plasma and face the plasma directly. They moderate the highly energetic neutron flux and convert it to thermal heat. The first wall in particular is subjected to extreme thermal and radiation loads. Therefore, it is possible that besides the test blankets which have to be replaced more frequently according to the maintenance schedule during regular shutdown periods in order to perform checks, repairs and test material sample investigations, the permanent breeder blanket modules are designed for the full lifetime of the machine and have to be exchanged in the event of failures.

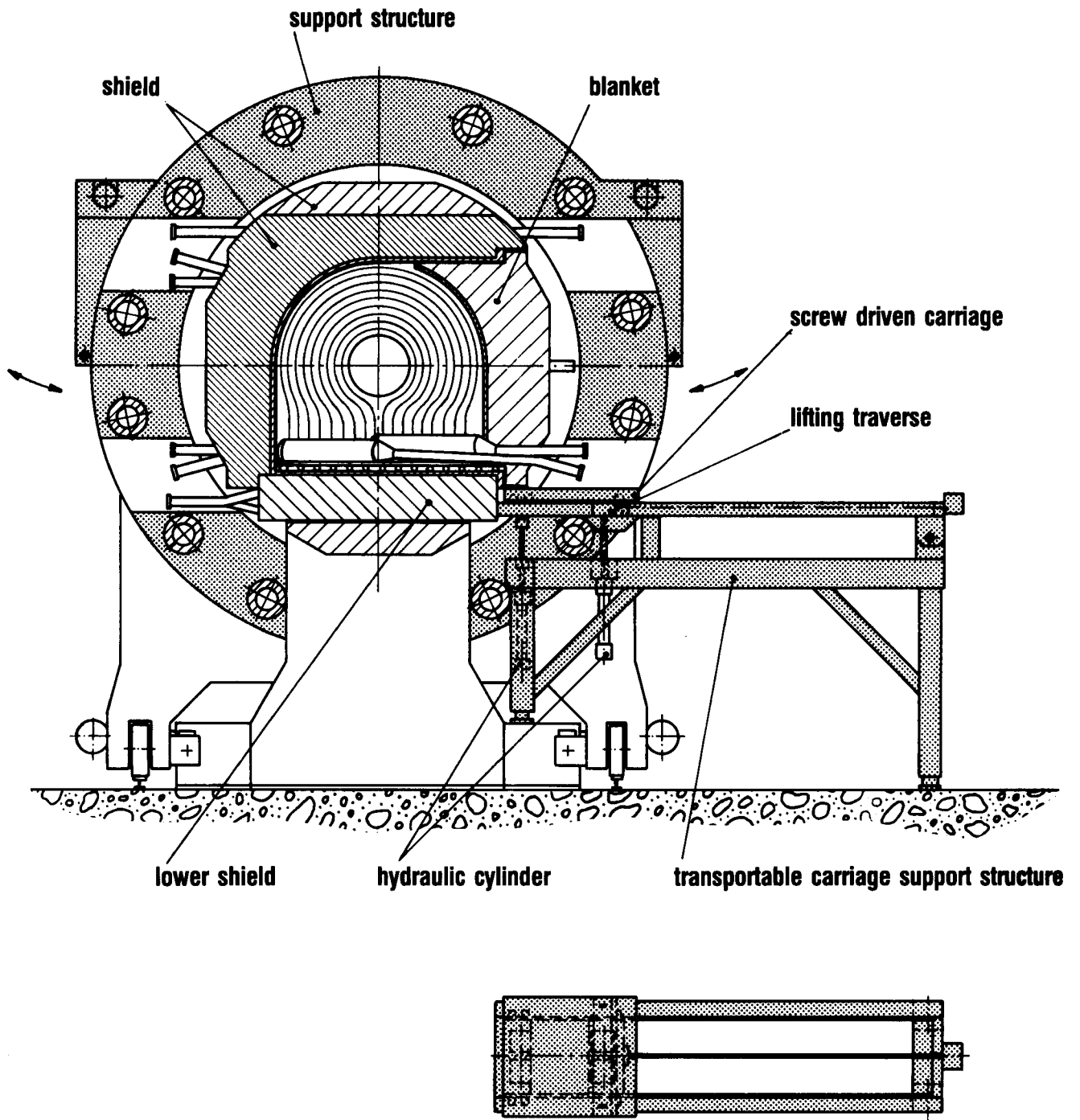
The replacement of blanket modules raises two questions:

- the dose rate at the surface of the blanket one day after shutdown is  $\sim 1000$  rem/h.
- the weight of  $\sim 50$  t for each blanket module makes handling difficult.

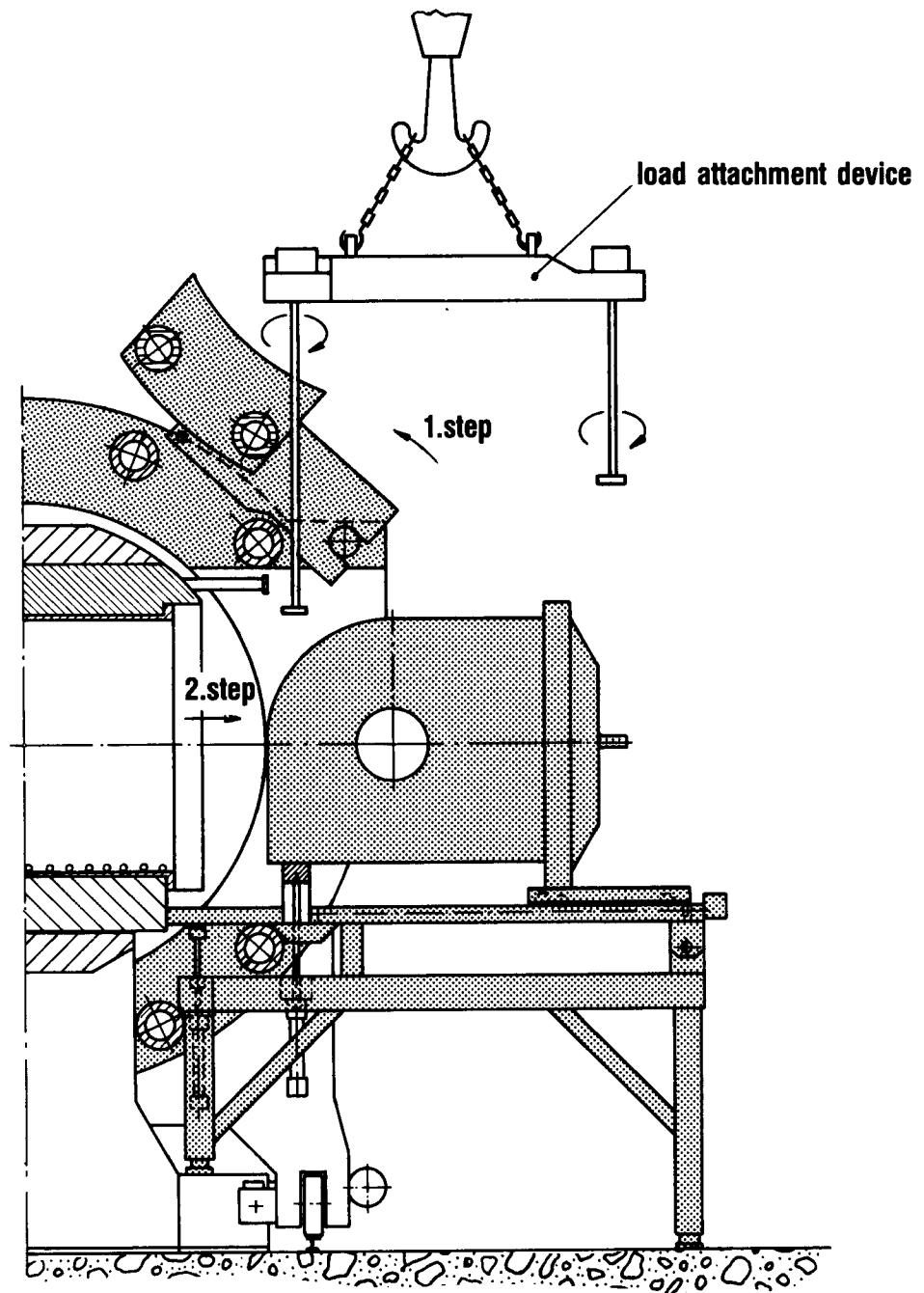
As a result of the first point, hands-on techniques are only possible as long as the shield is intact. In this case the biological dose rate one day after shutdown is less than 20 mrem/h. Additional transportable shield walls protect personnel against scattered radiation from other regions where the immediate radiation is higher. But all subsequent operations performed after removal of the blanket flange bolts have to be performed by remote handling techniques.

To facilitate the handling of blankets, roller rails are installed at the bottom of the vacuum chamber and a special transportable carriage support structure can be adapted at the test facility (see Fig. X.4-1). This carriage support structure is composed of an adjustable frame with linked tilting carriage tracks lifted by hydraulic cylinders, a screw driven carriage and a separate lifting traverse also raised by hydraulic cylinders to support the blanket after withdrawal. This traverse serves simultaneously as a blanket auxiliary structure in combination with the load attachment device shown in Fig. X.4-2 to transport blankets. The device itself is equipped with three revolving rods driven by electrical motors. The front parts of the rods are shaped as bayonet joints matching with a lug at the front of the blanket and the lifting traverse. By remotely rotating the rods attachment device or the blanket module, the lifting traverse is automatically engaged.

As mentioned in Chapter VI.1.2 blanket exchange is possible without dismantling the shield and reflector. The superconducting coils are deenergized and the coolant has to be drained. Subsequently the supply connections will be disconnected and the support structure swung away. After unscrewing the blanket flange bolts (shown in Fig. X.4-2) the carriage support structure will be adapted and the carriage track lifted hydraulically until the front part of



**Fig. X.4-1 Blanket exchange equipment**



**Fig. X.4-2 Blanket exchange**

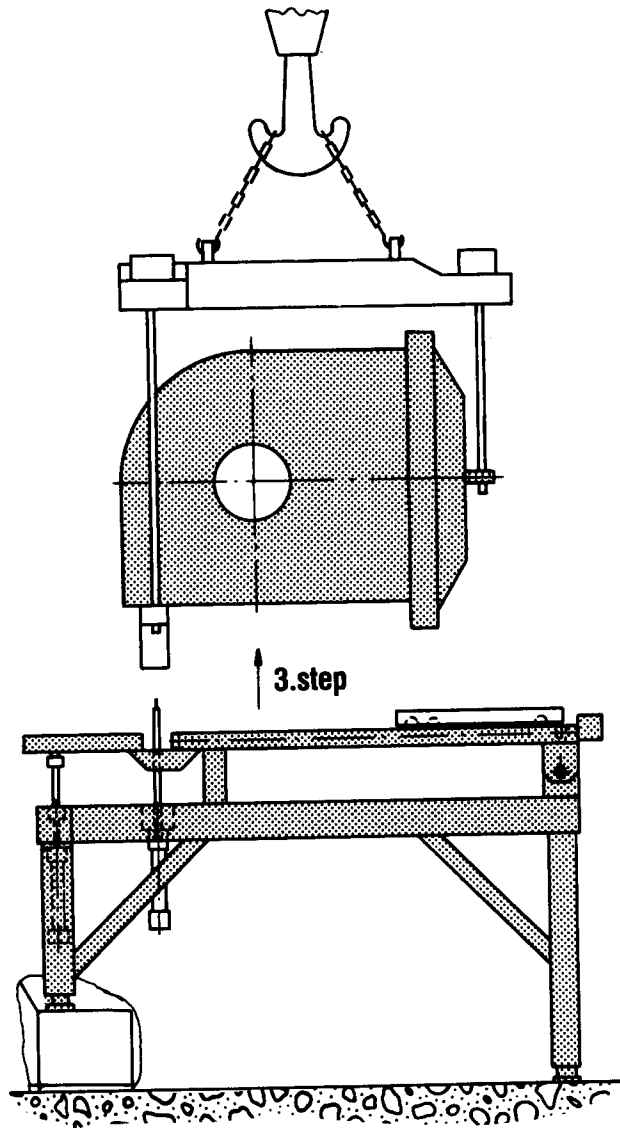
the blanket rests on the carriage. The blanket can then be withdrawn by means of the carriage. Before the blanket is totally withdrawn from the vacuum chamber, the lifting traverse is raised hydraulically to support the blanket until it will be taken over by the 250 ton overhead crane with the aid of the load attachment device (see Fig. X.4-3). For further operation it can then be transported into the temporary storage or hot cell. Blanket assembly can be performed in a reverse sequence.

#### X.4.2 Neutral Beam Injectors

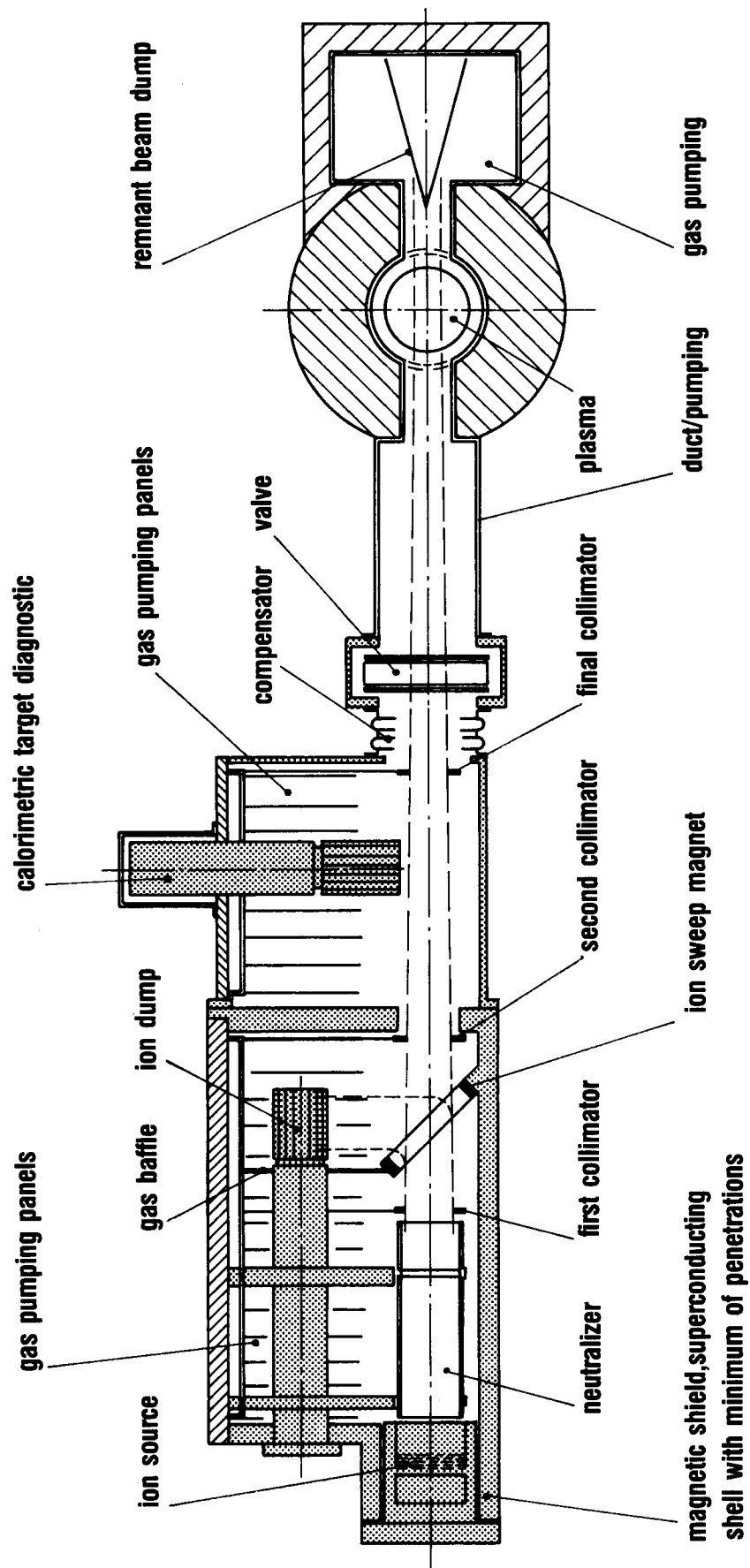
Altogether the plug regions will have 12 neutral beam injectors. As can be seen from Table X.3-1, parts of the neutral beam injectors are the most frequently replaced components in the facility. Figure X.4-4 shows a cross section of an injector where the main components are the ion source assembled near the cover, ion sweep magnet, ion dump, calorimetric target diagnostics, neutralizer, gas pumping panels and collimators. To provide the capability for adjusting the neutral beam box relative to the vacuum chamber a bellows will be needed between the box and the valve. The injectors operate at high voltages. To avoid failures caused by vacuum and water leaks and to minimize replacement time, an attempt has been made to combine components with units which have the fewest penetrations to be sealed. Thus, gas pumping panels, including bearing and panel drives, as well as the neutralizer and collimators are sealed as a unit with a single flange. Further, all supply tubes and cables were also attached to the same flange.

To protect the ion source and the beams from stray magnetic fields from adjacent coils it is necessary to shield them. Therefore, the box as well as the covers consist of approximately 20 cm thick shielding material, making these components heavy and difficult to handle.





**Fig. X.4-3 Blanket transport**



**Fig. X.4-4 Neutral beam injector**

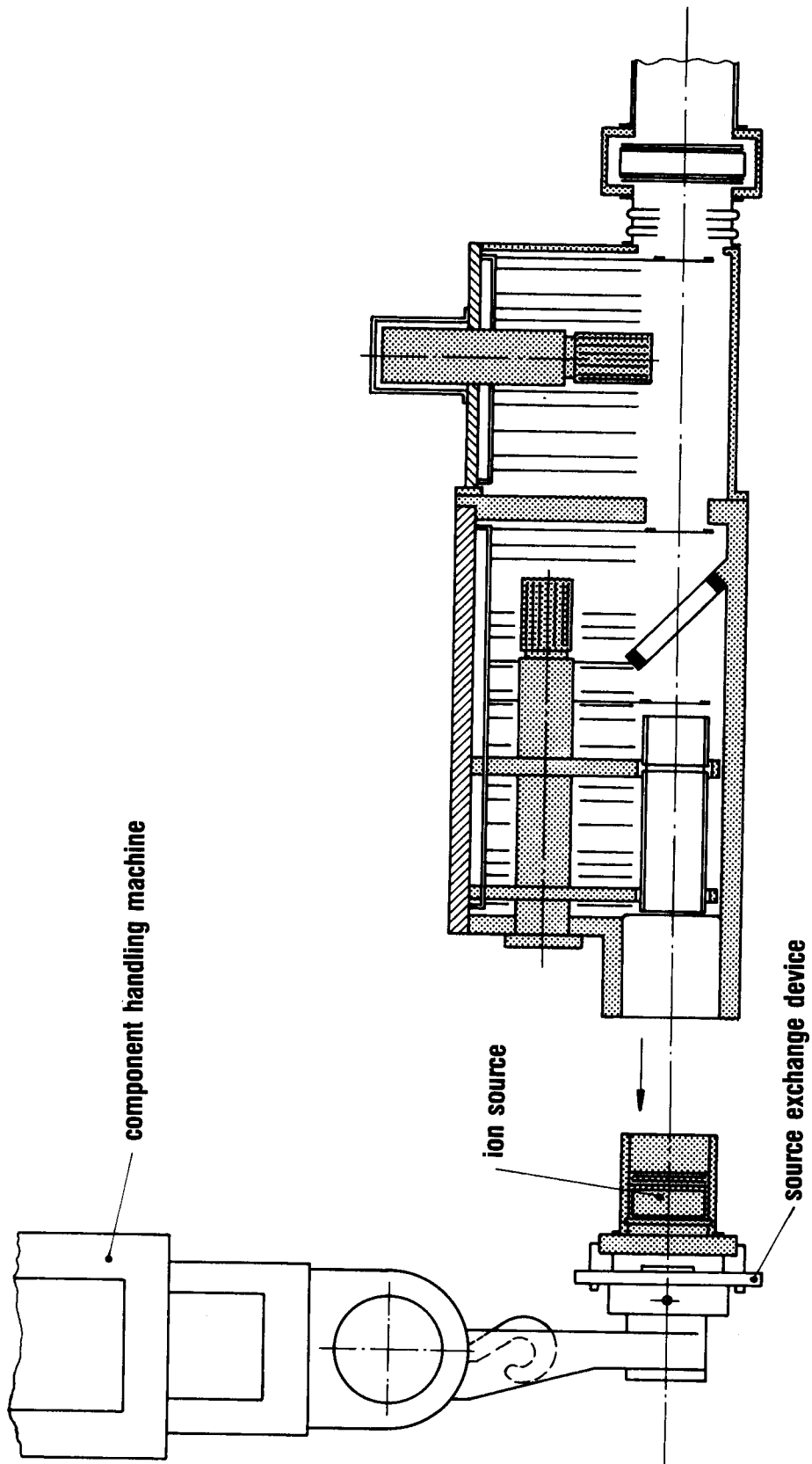
#### X.4.2.1 Replacement of Ion Sources (Fig. X.4-5)

In order to isolate the vacuum chamber from the neutral beam box during replacement operations the valve between the neutral beam box and duct will be closed. Subsequently the supply connections will be broken and after unscrewing the flange bolts and attaching the component handling machine with the ion source exchange device to the flange, it can be withdrawn along with the source. The assembly of a new one can be accomplished in the reverse manner.

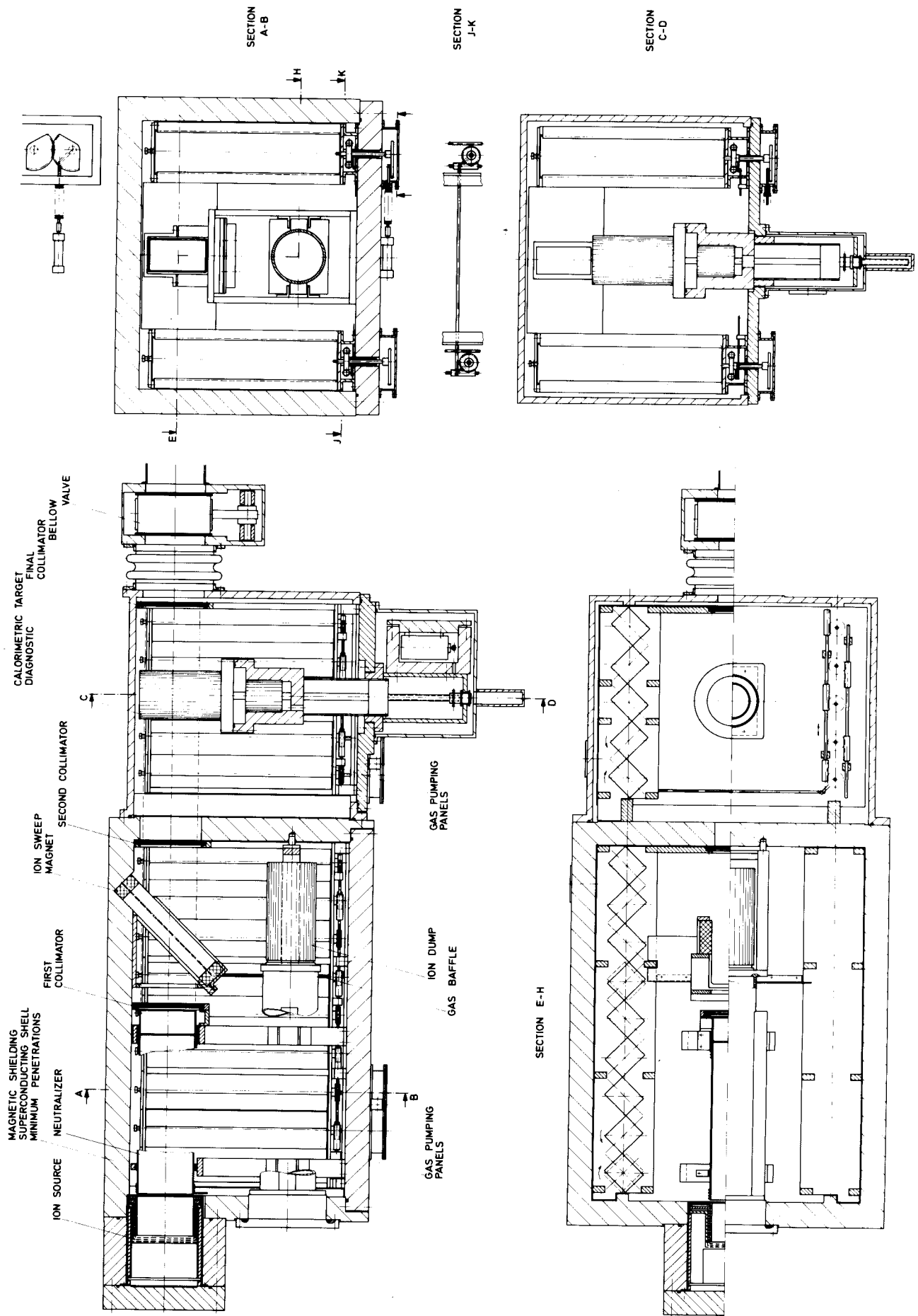
#### X.4.2.2 Replacement of Getter Panels

Gas pumping panels foreseen to operate in a DC-mode are assembled as blocks of getter panels. Every two panels are assembled in a common frame back-to-back. The frames themselves are arranged in a sawtooth fashion (see Fig. X.4-6) with the consequence that simultaneous pumping and regeneration can take place. The front panels facing the ion stream pump at temperatures of  $\sim 200^{\circ}\text{C}$ , while the rear panels facing the box walls are heated to temperatures of  $\sim 700^{\circ}\text{C}$  and the released gas will be pumped out by turbomolecular pumps. To change pumping and regeneration sides the frames with inserted panels are simultaneously reversed by cog wheels and toothed rack transmission drives mounted at the cover.

To replace the panels (see Fig. X.4-7) the cover has to be removed after the duct valve is closed, the ion dump is removed in a similar manner as the ion source, the supply tubes and cables disconnected and the flange bolts are unscrewed. Removal of the cover will be performed by means of a crane. For neutral beam injectors arranged in another orientation as shown in Fig. X.4-7 it will be necessary after the valve is closed to disconnect the box at the bellows, displace it away from the facility and then rotate it by means of a turnover machine to a position for lifting the covers from above.

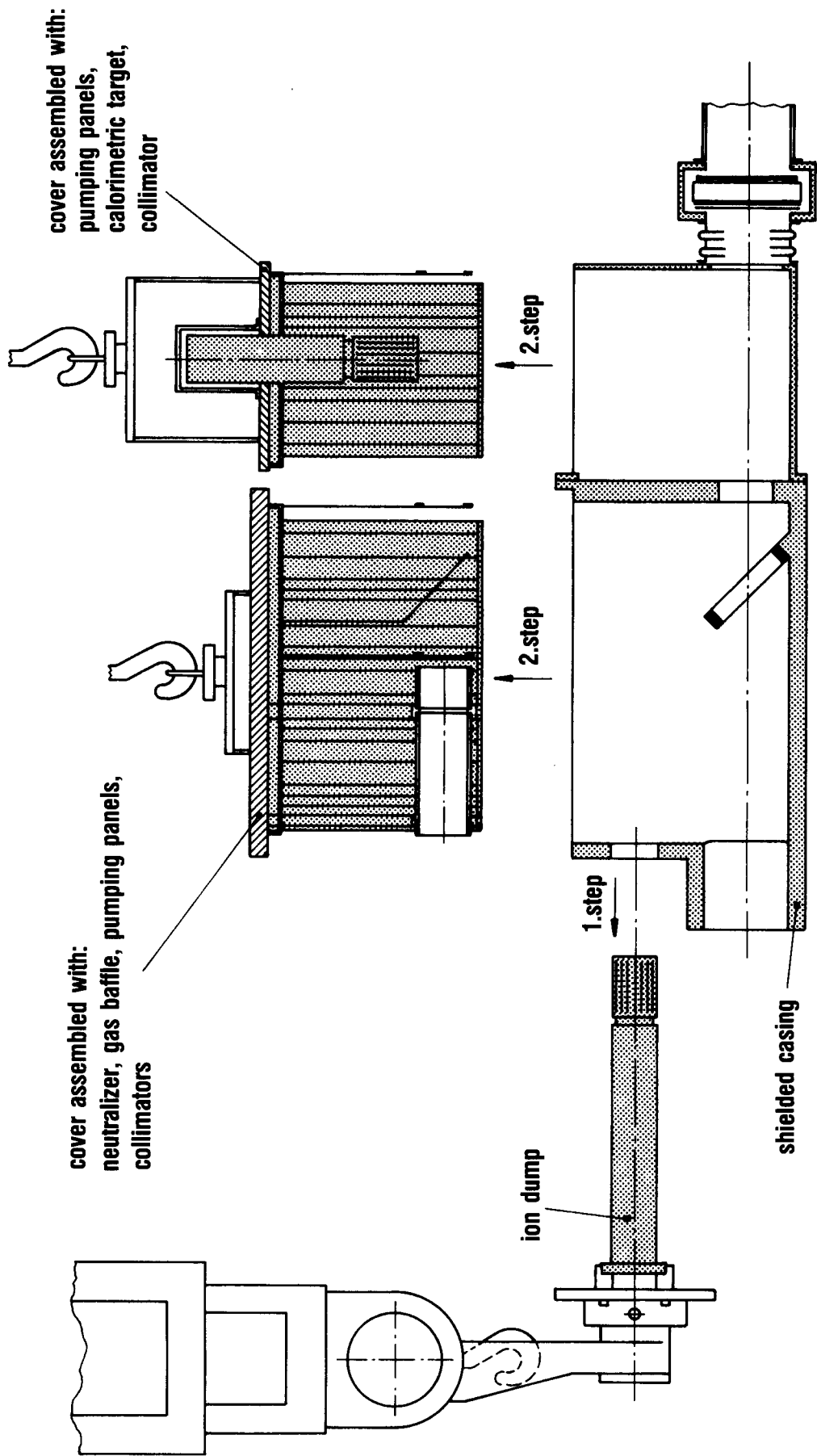


**Fig. X.4-5 Neutral beam injector - exchange of the ion source**



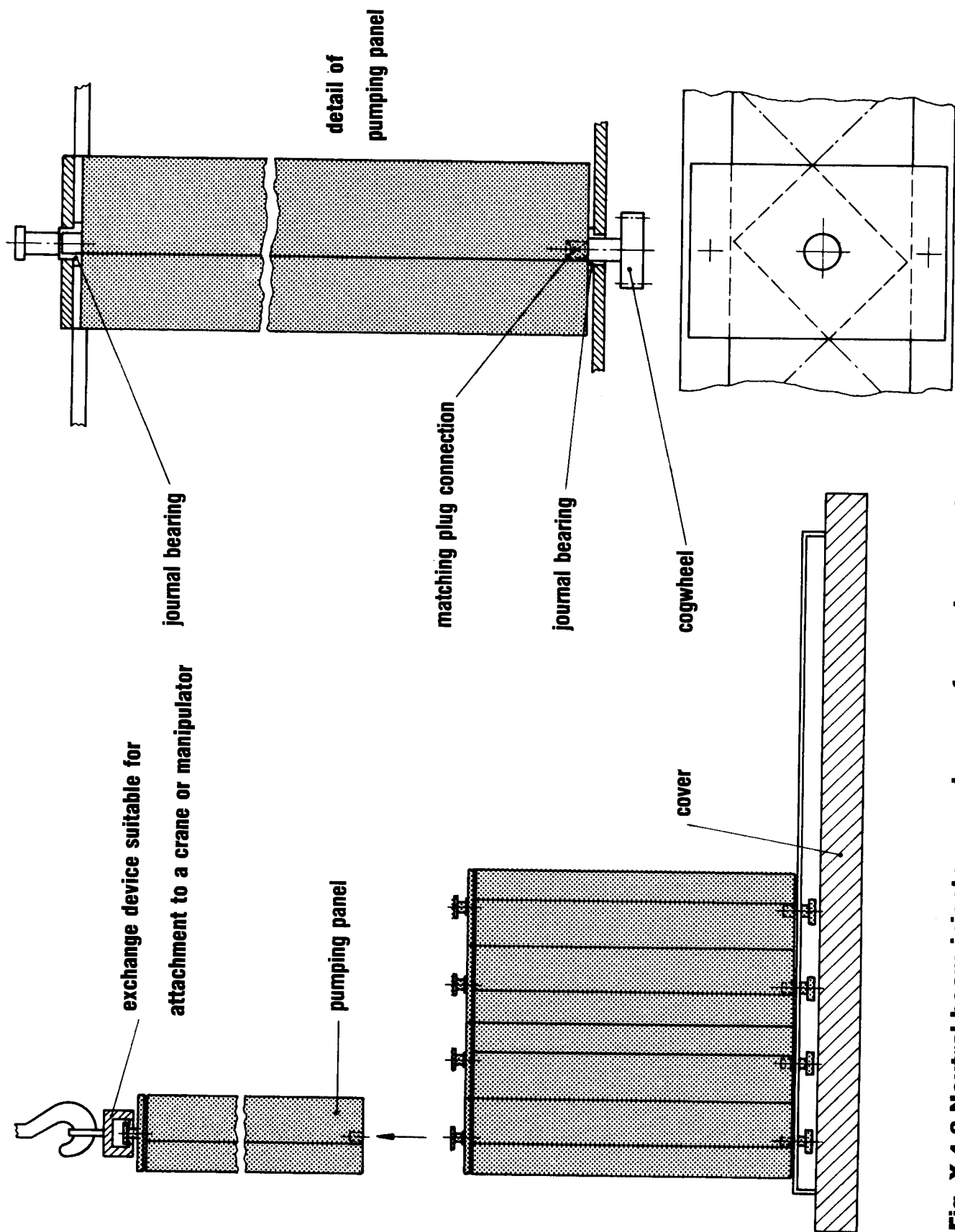
X.4-10

**Fig. X.4-6 High energy neutral beam injector**



**Fig. X.4-7 Neutral beam injector - exchange of the cover**

After removal the cover has to be attached to a turnover plate and rotated to a position shown in Fig. X.4-8. To transmit the torque from the cog wheels to the panels a matching plug connection is provided. On the other hand the panels are pivoted in a plate fixed with two screws at the frame for the getter panels. Removing the two screws allows the withdrawal of a set of panels with relative ease. The procedure is reversed for assembly of the panels.



**Fig. X.4-8 Neutral beam injector - exchange of pumping panels**



## X.5 Large Sized Components

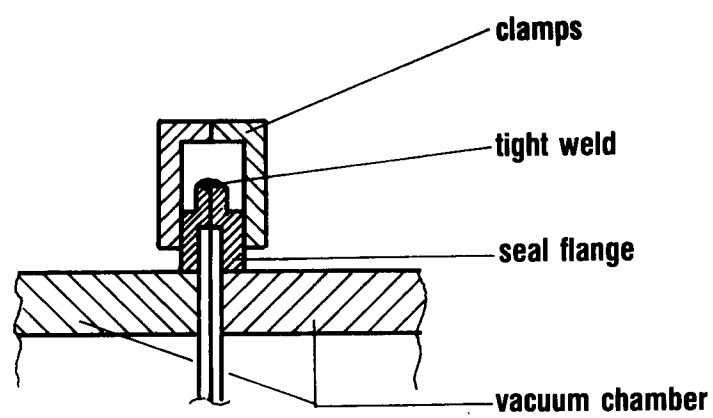
Besides the routine operations, such as blanket exchange and neutral beam injection component replacement, there is another group of more difficult operations that must be investigated. The difficulties arise from the weight of the components, and the remote handling techniques needed. For example, it is impossible to replace the central cell coils or barrier coils without cutting into the vacuum chamber sections. Therefore, the design of the facility must take into account the separation of the vacuum chamber and integrate it into the overall system. In view of the needed vacuum chamber tightness it does not seem possible to connect the vacuum vessel sections only by mechanical gaskets and clamps. A solution seems to be a combination of tight welds and mechanical clamps (shown in Fig. X.5-1). While the clamps would transmit thermal and mechanical loads, the weld joints would provide the leak tightness. To equalize the variation in the dimensions of the sections the flanges can be made flexible to a certain degree.

### X.5.1 Central Cell Coils

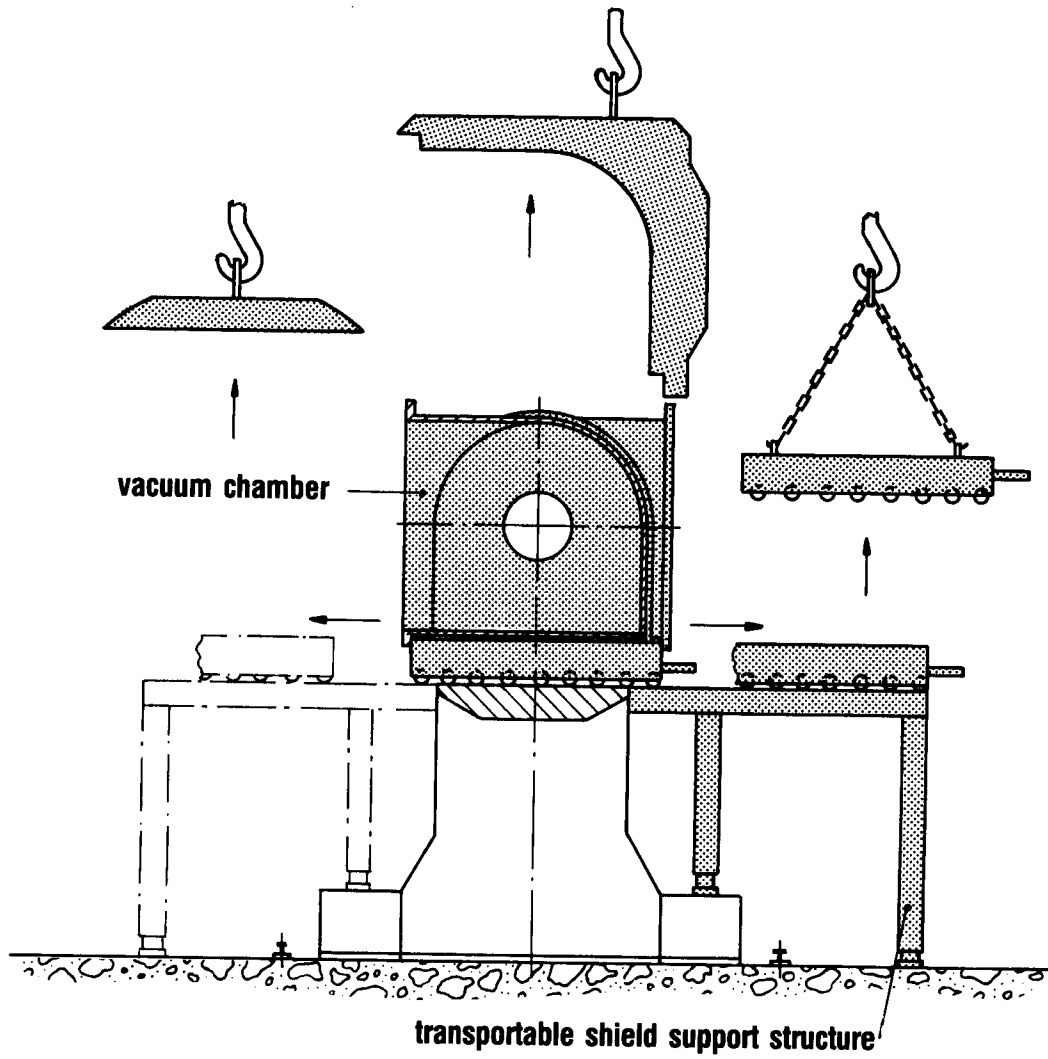
An exchange of coils has to be performed in the case of a possible coil failure. In this case, a longer shutdown period will be necessary and the complexity of the operation requires certain additional special equipment.

Figure X.5-2 shows a cross section of the central cell in which blanket and shield items have been removed, with the exception of the lower shield. For withdrawal this shield is equipped with roller rails such that it can be transferred to a transportable shield support structure attached to the reactor.

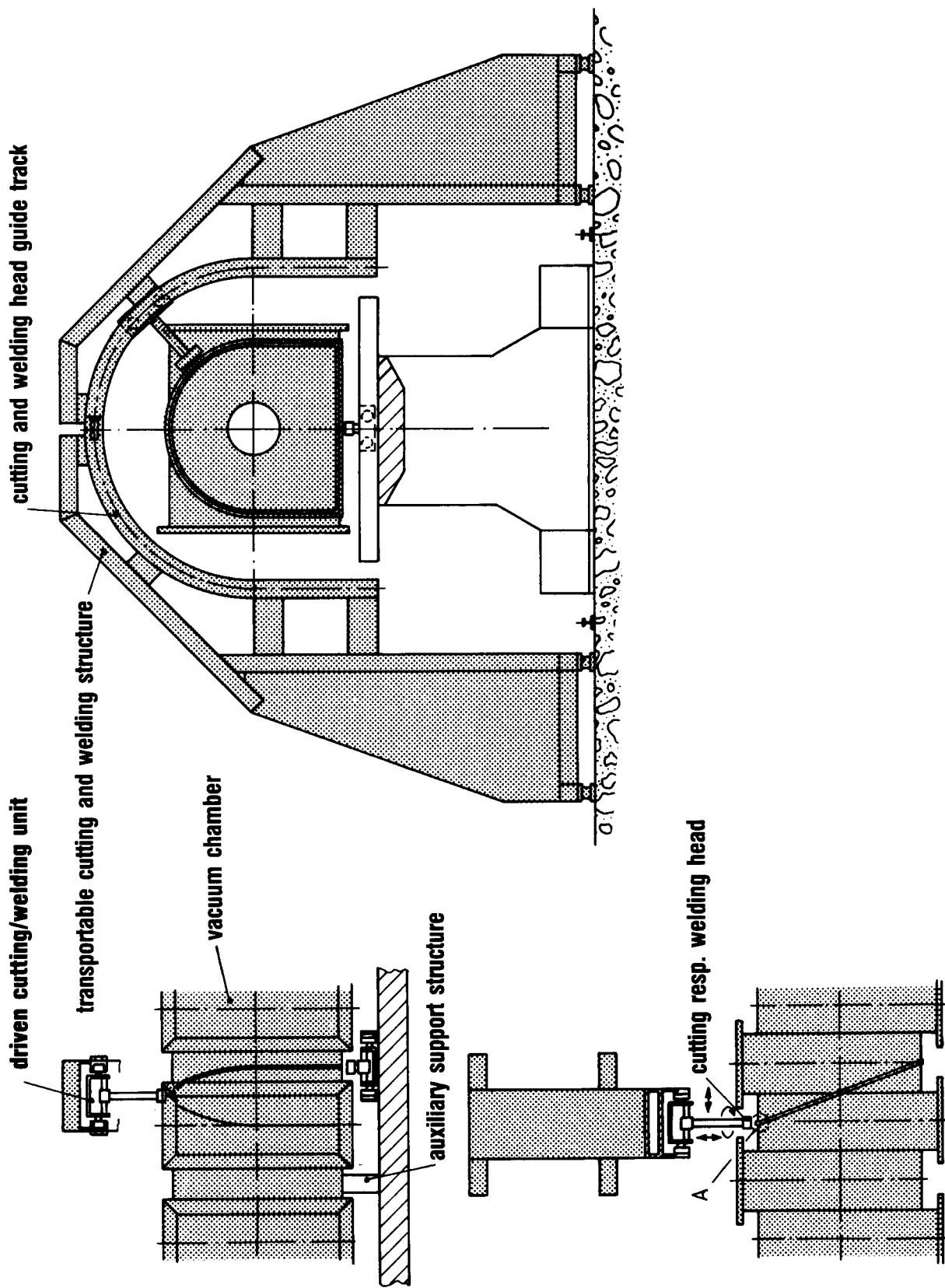
The most critical operation will be the rewelding of the vacuum chamber. A cutting and welding machine (see Fig. X.5-3), assembled with a cutting and



**Fig. X.5-1 Vacuum chamber connection**



**Fig. X.5-2 Exchange of central cell coils - removal of the shield**



**Fig. X.5-3 Exchange of central cell coil - welding of vacuum chamber**

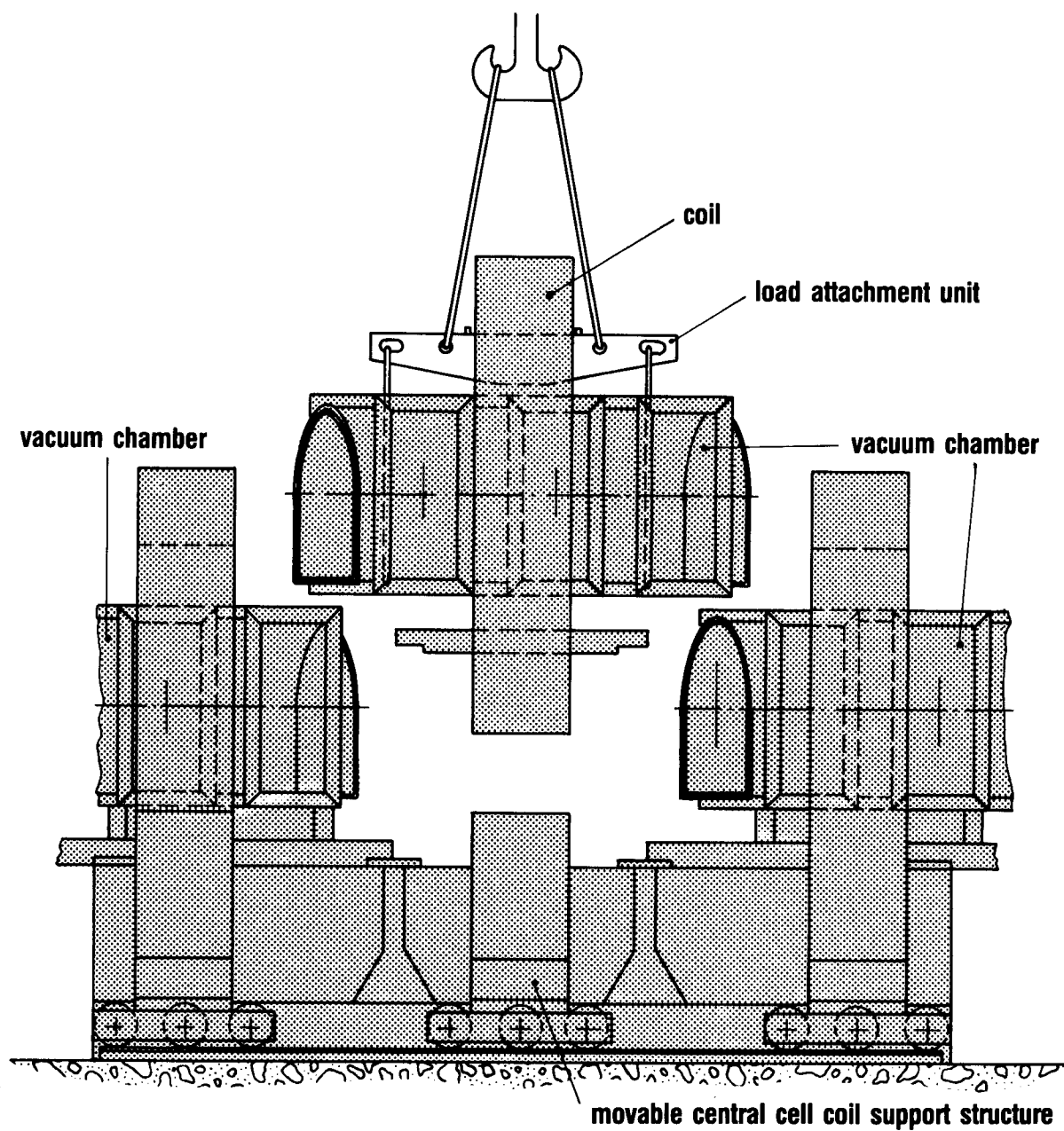
welding heat guide track, serves as a basic unit where either a welding or a cutting head can be mounted. The cutting head as well as the welding head is remotely controlled and can be tilted or rotated. A guide track is mounted for welding the vacuum chamber bottom plate. The weld joints are made in a manner that would not require a refinishing operation of the flanges used to seal the blanket module closures.

To replace a central cell coil, the coils are deenergized and the blankets withdrawn as shown in Section X.4.1. Subsequently the reflector and shield are drained and the supply lines disconnected. After removal of the support structure, reflector and shield units (see Fig. X.5-2) the cutting and welding machine can be attached (Fig. X.5-3). A shift of the corresponding central cell coil will be necessary. After dismantling the mechanical clamps the weld joint can be cut. A second cut on the other side of the coil makes it possible to remove the coil, chamber unit and lower support structure at the same time by means of the crane and a load attachment unit (Fig. X.5-4).

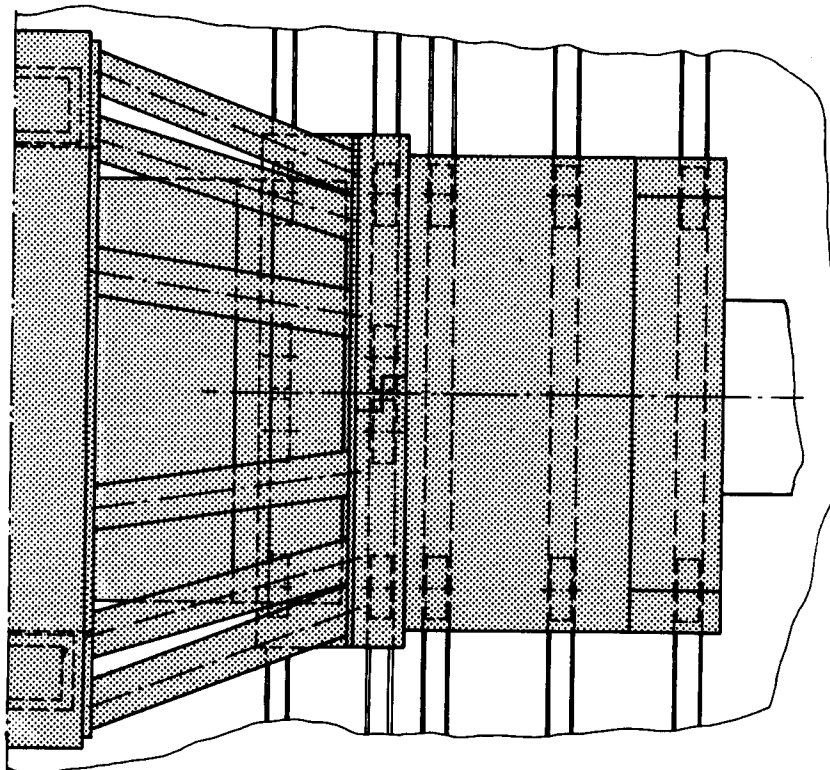
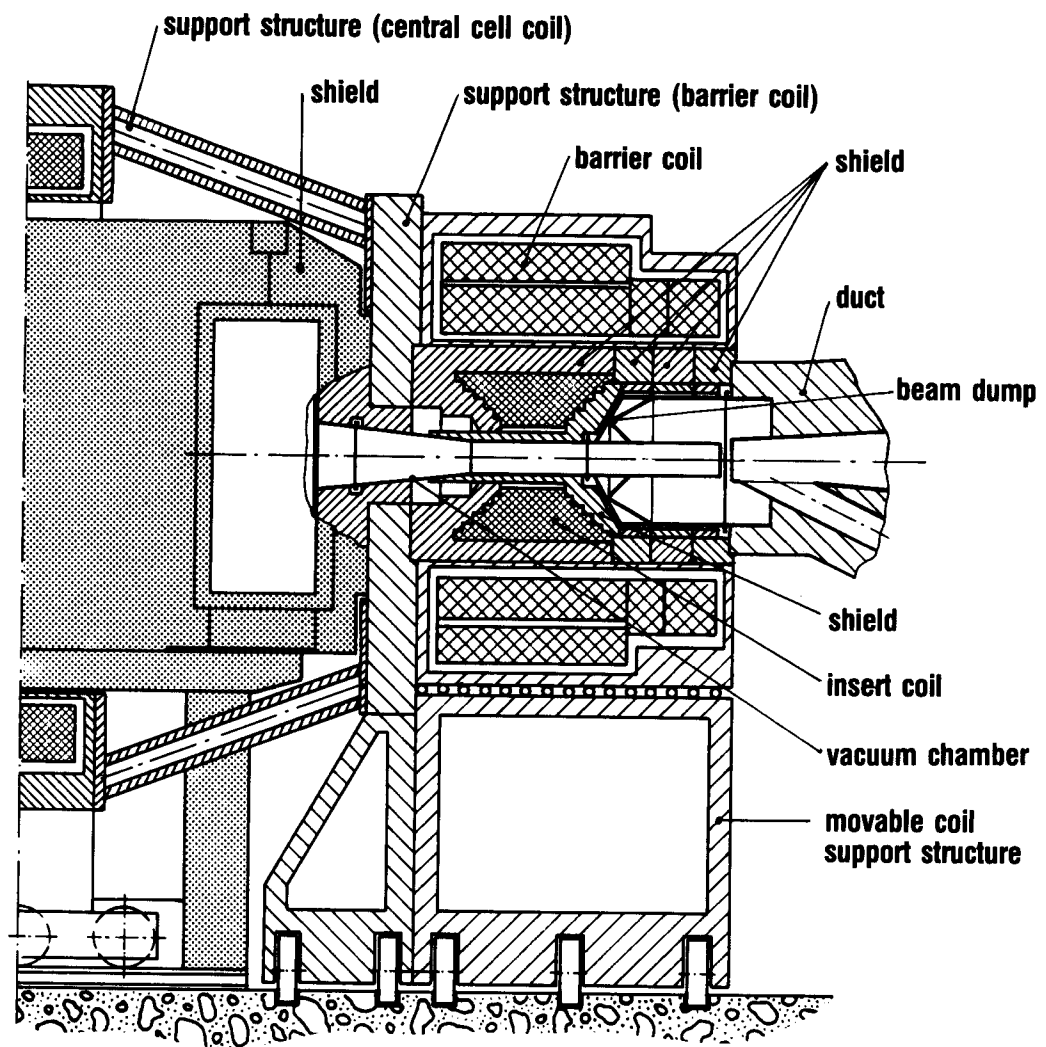
Before a new coil/chamber unit can be put into place, a check of the vessel dimensions and preparation of the weld joint must be made. After the new parts have been adjusted their position will be checked by optical measurements and the vacuum chamber welded and clamped. Additional work must follow the same separation sequence.

#### X.5.2 Barrier Coil (Fig. X.5-5)

Barrier coil replacement will be necessary at least every three years due to radiation damage of the barrier insert coils and sputtering of the medium neutral beam dump.



**Fig. X.5-4 Exchange of central cell coil - removal of coil and vacuum chamber**



**Fig. X.5-5 Barrier region**

X.5-7

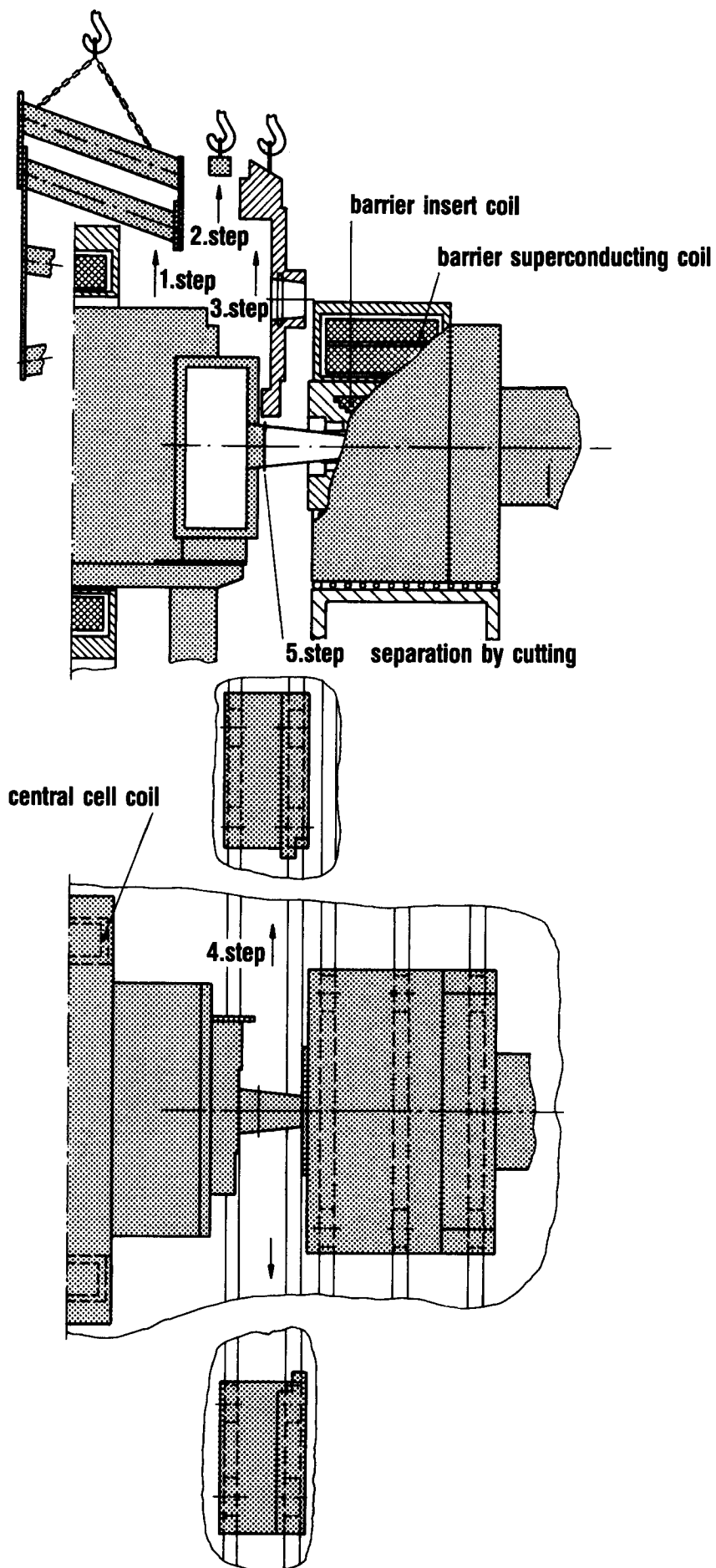
The preliminary work of preparing a barrier coil for replacement is shown in Fig. X.5-6, namely the disconnection of supply lines and the removal of support structure, the support plate which is movable on tracks and finally the shield. This gives access to the weld joint. After dismantling the clamps and mounting the cutting guide track around the vacuum chamber, the weld joint can be cut. Subsequently an auxiliary support structure will be adapted to the base structure of the barrier coil (Fig. X.5-7) and the coil equipped with roller rails can be shifted toward the central cell for cutting the weld joint on the opposite side of the barrier coil in the same manner. While the coil is being shifted, additional shield segments located inside the coil will be withdrawn. After the second cut is made the barrier coil will be shifted to its original position whereupon it can be removed together with its support structure out of the reactor (Fig. X.5-8). Because of the weight of the barrier coil, it should be dismantled in the reactor hall before parts of it within the capability of the crane can be transported into the lock area.

Reassembly will be performed in a reverse sequence and in a similar manner as the central cell coil.

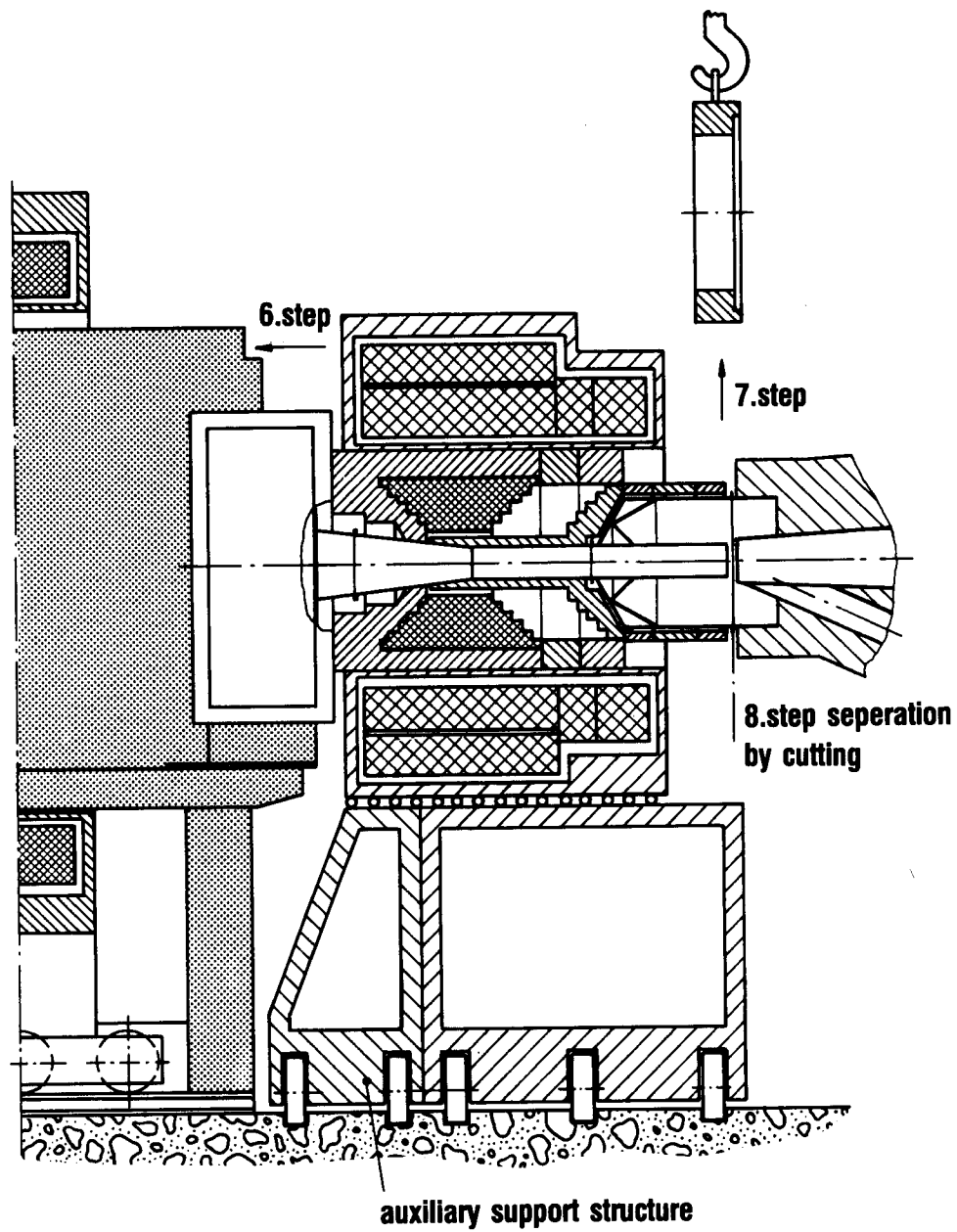
### X.5.3 Conclusion

It was only possible to consider a few critical maintenance operations and the objective was to check the possibility of performance. There are numerous maintenance tasks which impact on the component design but it can also be assumed that the maintenance problems are soluble.

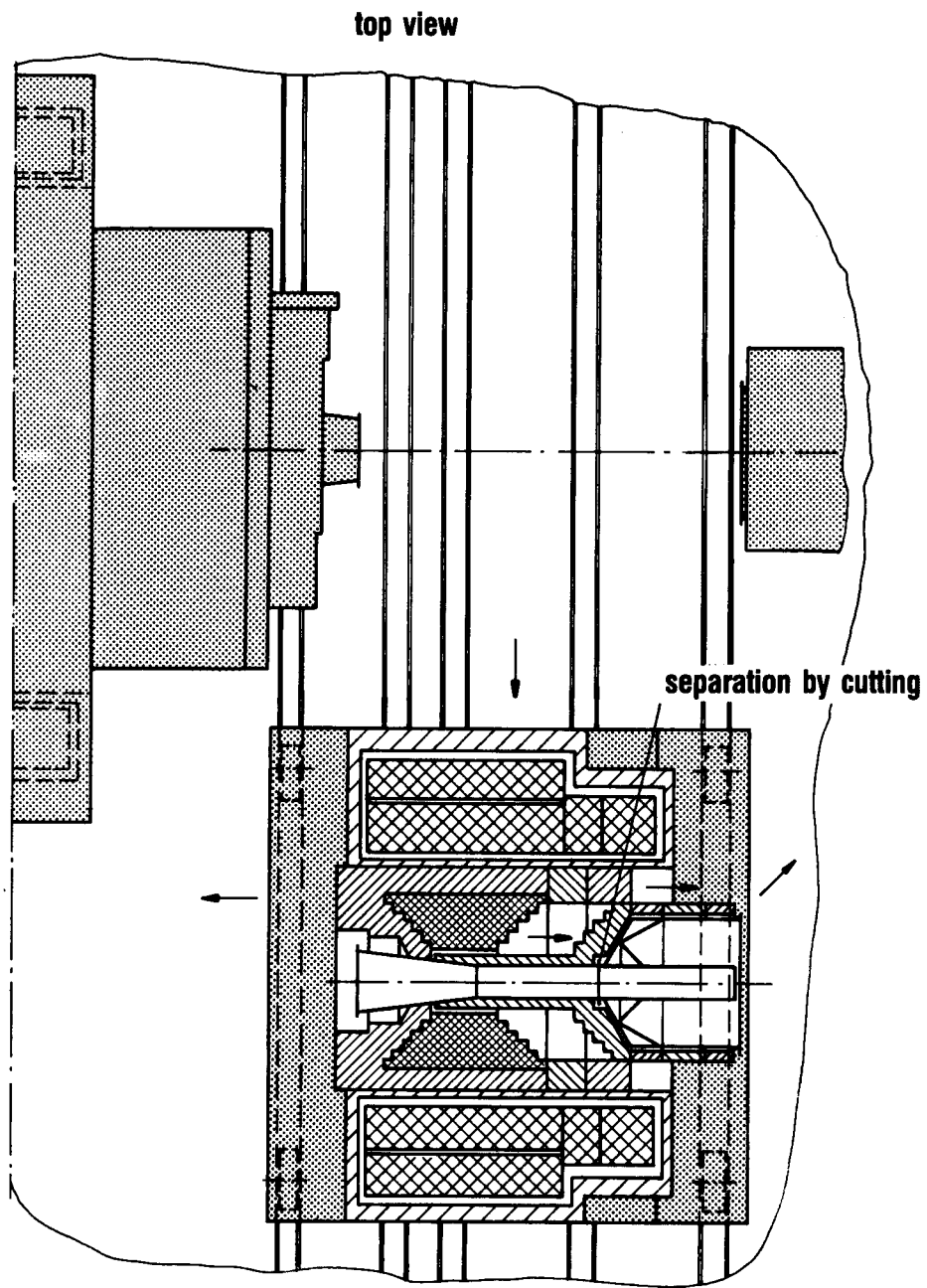




**Fig. X.5-6 Components exchange at the barrier region - preliminary works**



**Fig. X.5-7 Components exchange at the barrier region - vacuum chamber separation**



**Fig. X.5-8 Components exchange at the barrier region - components dismantling**

## X.6 Barrier Coil

### X.6.1 Requirement and Objective

The requirement is to replace the normal conducting insert coils and barrier pump neutral beam dumps in the barrier coil assemblies without simultaneously replacing their superconducting coils, and to do so by remote control without manual assistance.

The objective is to evolve design features and assembly/disassembly procedures to satisfy the above requirement, using criteria of cost effectiveness, operational risk, simplicity, reliability, practicality, and inspectability for quality control purposes.

### X.6.2 Approach

TASKA is envisioned to produce neutron wall loadings near  $1 \text{ MW/m}^2$  during more than 10 years of operation at machine availabilities from 15 to 50%. Hence, remote maintenance provisions approaching those of a power reactor become mandatory.

Early emphasis of remote maintenance requirements is made difficult by the fact that many individual component reliabilities and lifetimes are not well established. Due to the scope of TASKA studies, remote maintenance considerations have been restricted to the most important machine components rather than governing all configuration and design decisions from the beginning. In Sections X.1, X.2, X.3, central cell, plugs, and the exchange of test modules were discussed. In this section, replacement of a module consisting of the barrier pump neutral beam dumps (henceforth called "beam dumps") and normal conducting insert coils (henceforth called "normal coils") will be addressed. The lifetime of these coils is expected to be limited to about 5 years (at 50% availability) due to radiation damage to their ceramic

insulation. Lifetime limitation of the beam dumps, due to erosion, is expected to be comparable.

It is assumed (see Fig. X.6-1), that the central cell (station B to C) and the entire end-plug assemblies of the machine (to the left of station A, and to the right of station B) stay in place during all operations associated with replacement of the normal coils and beam dumps. The approach is to cut the vacuum vessel at two locations in the vicinity of, say, C and D, and to move the entire module between stations C and D horizontally normal to the machine axis, thus becoming accessible from its ends allowing replacement of the normal coil and the beam dumps. For reassembly, the sequence is reversed. Individual steps are explained in Section X.6.5.

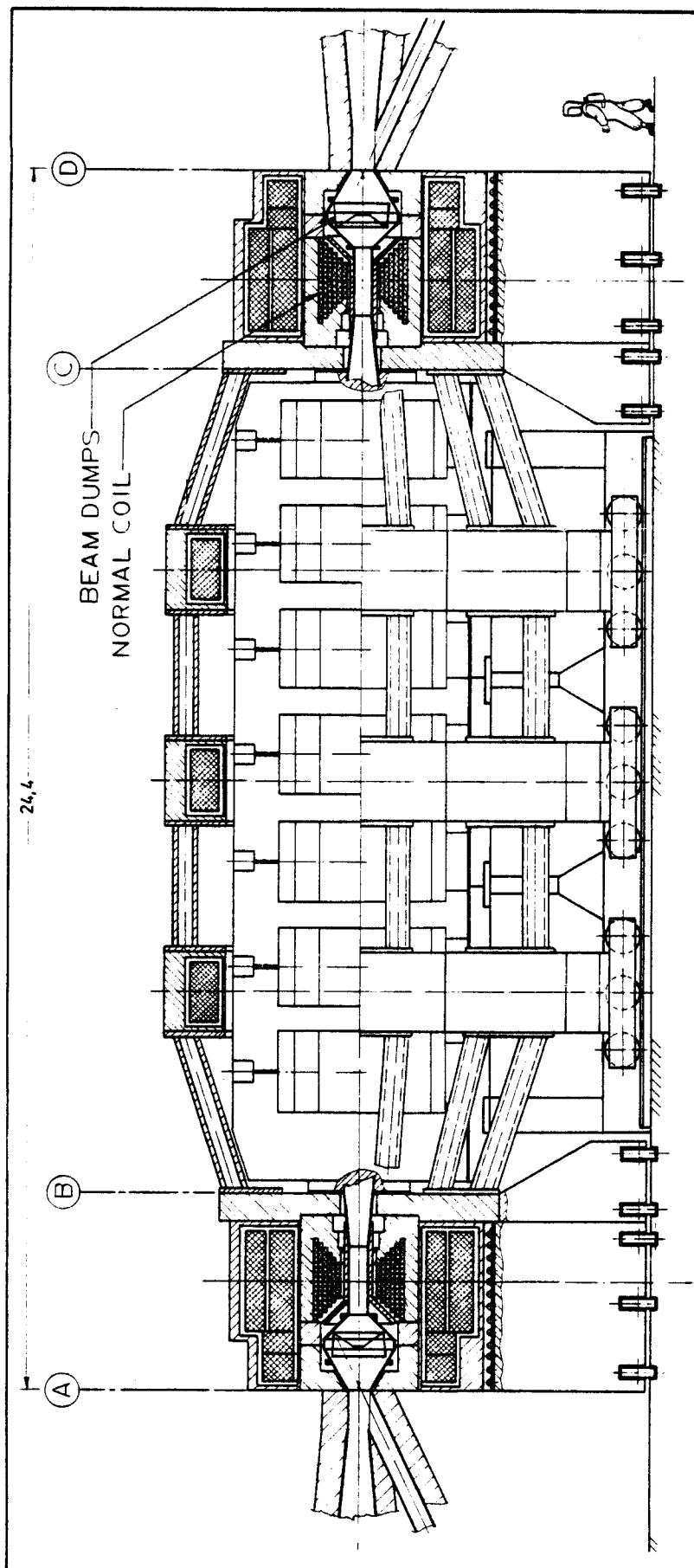
To simplify the above procedure, a number of design features have been incorporated, as described conceptually in Section X.6.3.

Several alternate approaches have been considered as discussed in Section X.6.4.

### X.6.3 Configuration Description

#### X.6.3.1 General Description

The barrier coil module consists of a superconducting coil which encircles a normal coil which, in turn, encircles a segment of the vacuum vessel. The beam dumps (and associated cooling components) are located within the vacuum vessel. Three neutral beam injection tubes intersect the vacuum vessel, each making a 25 degree angle with the plasma axis. The tubes are spaced at 120 degree intervals around the plasma axis. Units of shielding are positioned between the vacuum vessel and the coils.



Elevation View of the Central Cell and Barrier Coils

Fig. X.6-1

The normal coil is encircled by a stainless steel band approximately 15 centimeters thick. The cryostat of the superconducting coil is enclosed in a structural stainless steel case varying in thickness from 5 to 20 centimeters.

A massive support plate, 50 centimeters thick, encircling the vacuum vessel, is located between the barrier coil and the central cell. Intercoil supports connect the support plate to the structural supports of the central cell solenoid coils.

Figure X.6-1 shows the relationship of the two barrier coil areas to the central cell. Figure X.6-2 presents a cross sectional view of the barrier coil module and Fig. X.6-3 illustrates its major components in an exploded view.

#### X.6.3.2 Remote Maintenance Configuration

To ensure that barrier coil maintenance can be accomplished within reasonable limits of time, cost and operational risk, some modification to the configuration described above has been made and is discussed below.

The structural case of the superconducting coil interfaces with and is supported by a track mounted barrier coil carriage. The carriage consists of an upper and a lower section. The upper section cradles the superconducting coil structural case. Three sets of rollers are positioned between the upper and lower carriage sections to permit at least 60 centimeters of axial translation of the barrier coil along the plasma axis for maintenance purposes.

The lower section of the barrier coil carriage is supported by floor mounted tracks in such a way that barrier coil horizontal translation normal to the plasma axis is possible. It also incorporates adjustment capability for barrier coil alignment to adjacent TASKA components. Suitable devices must also be provided to secure the carriage sections to each other (during

Fig. X.6-2

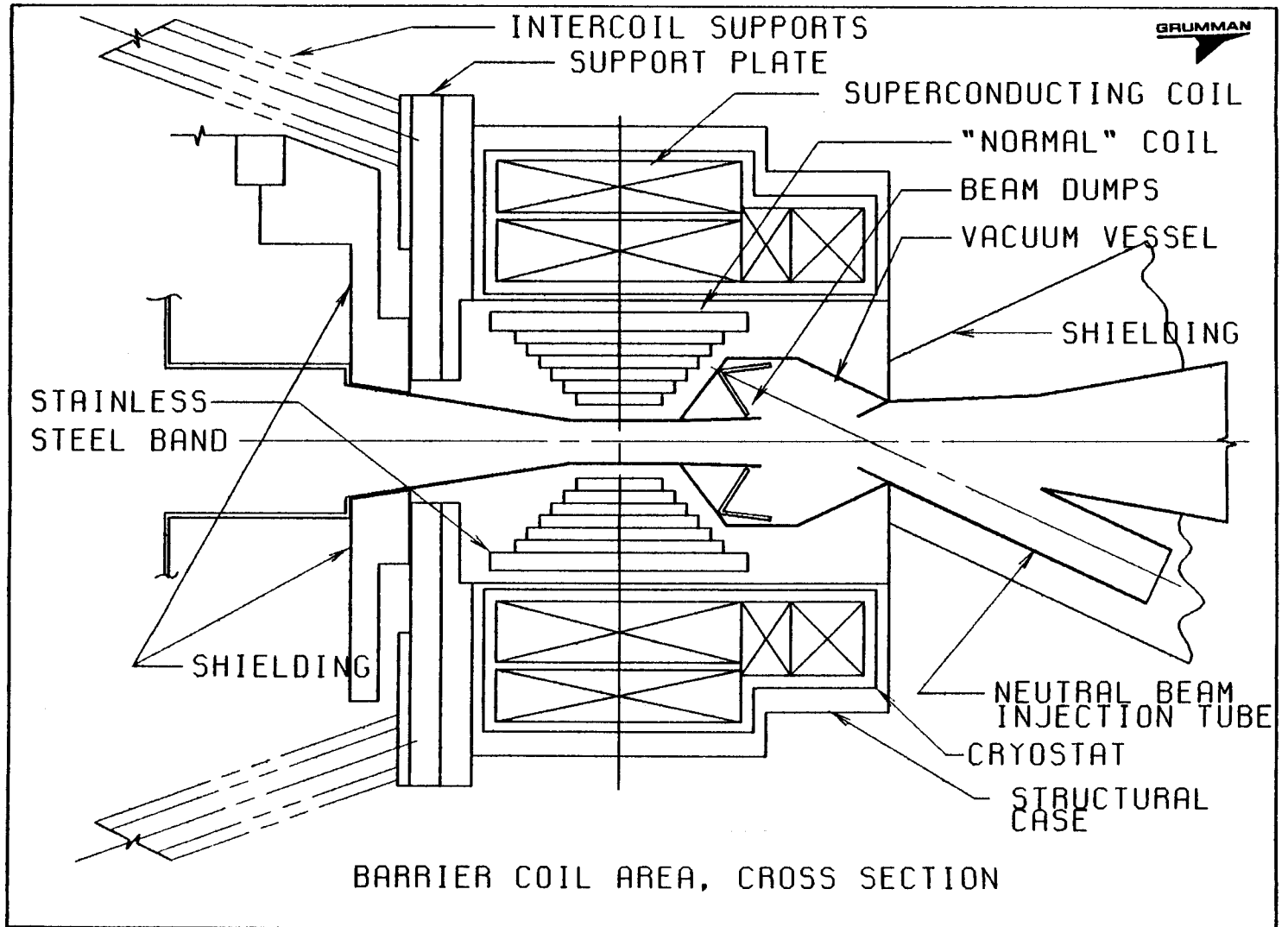
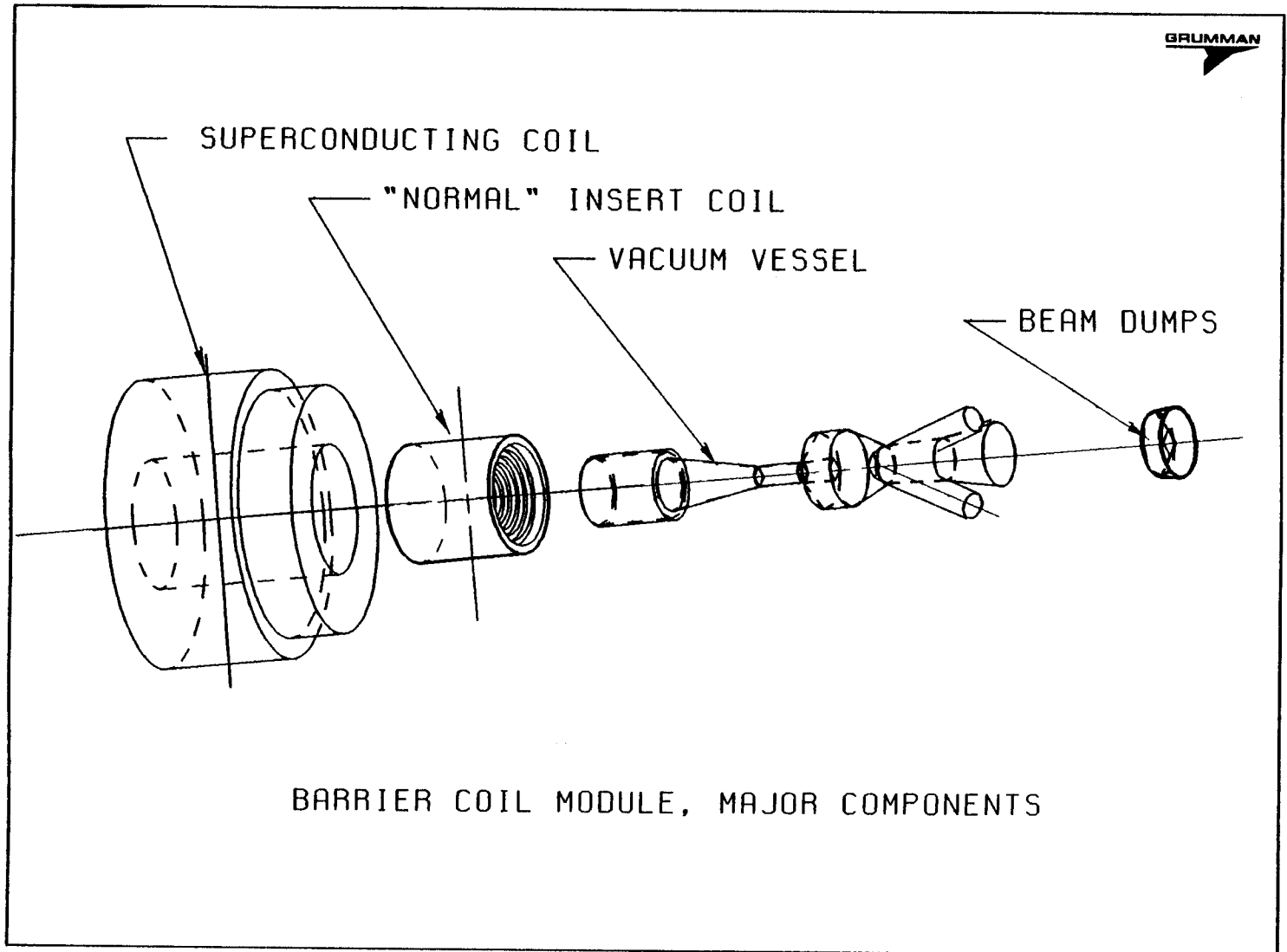




Fig. X.6-3



maintenance procedures) and to the facility floor during TASKA operation. Positioning devices and stops are incorporated in the barrier coil carriage and in the floor mounted tracks to simplify remote maintenance operations.

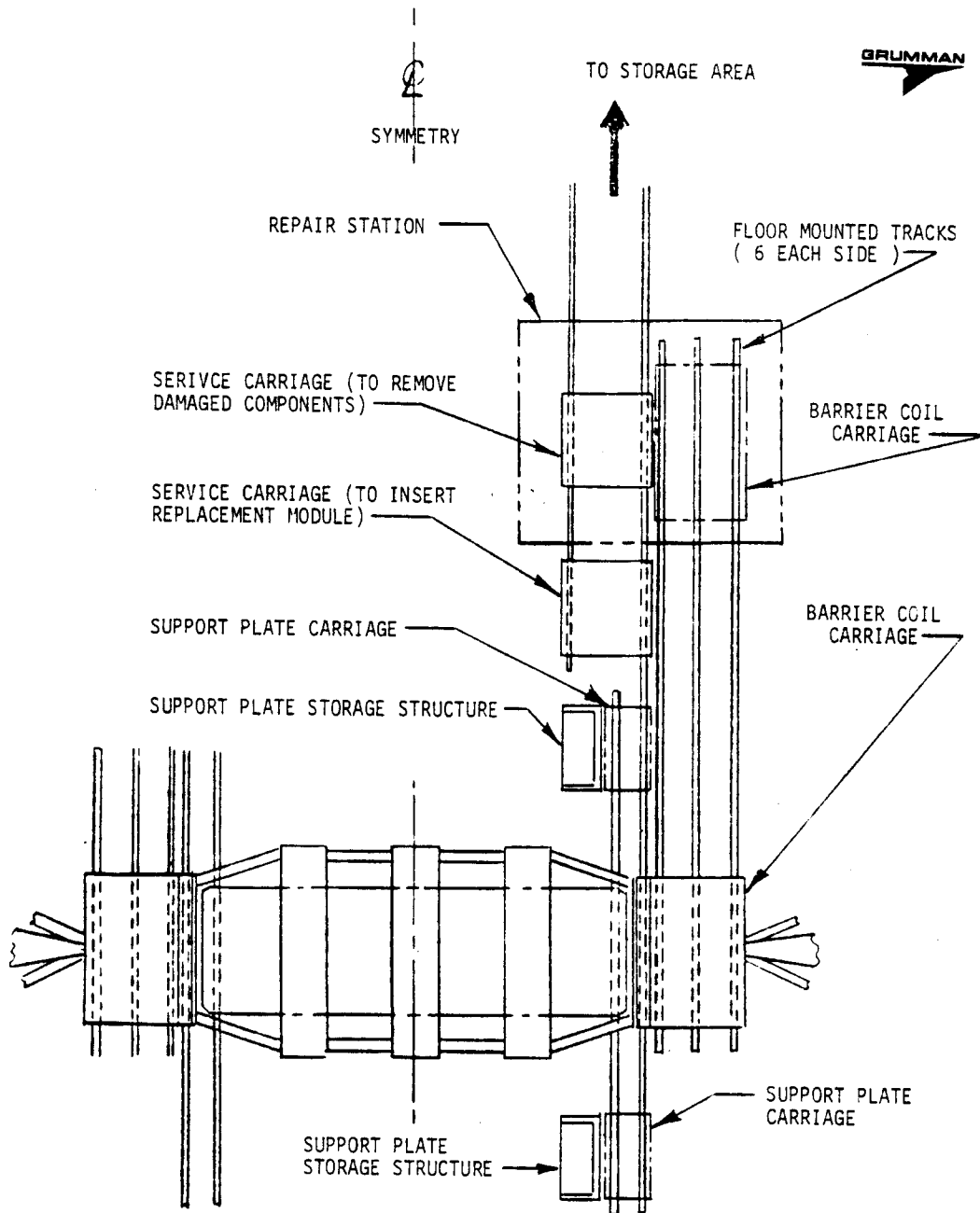
Figure X.6-4 shows the barrier coil carriage in relationship to the TASKA machine and to the facility. It is also illustrated in Fig. X.6-8 and X.6-9. Figure X.6-11 shows its axial motion (along the plasma axis) capability.

The massive support plate encircling the vacuum vessel is designed as two interlocking halves joined on the vertical centerline, enabling its removal to gain access to the vacuum vessel parting flange. Each half is supported by a carriage mounted on tracks on the floor which permit horizontal translation normal to the plasma axis. As with the barrier coil carriage, positioning devices and stops are required in the support plate carriage and its associated tracks simplify remote maintenance. The support plate carriage is illustrated in relationship to TASKA and the facility in Fig. X.6-4 and is also shown in Fig. X.6-8. Its remote maintenance usefulness is shown in Fig. X.6-9.

Although not illustrated, it is recommended that the support plate carriage incorporate provisions to stabilize and secure the support plate halves during removal procedures. These provisions may be added during maintenance procedures when the support plate halves are partially removed from their normal positions.

Two sets of parting plane flanges located to each side of the barrier coil must be incorporated into the design of the vacuum vessel so that removal of the barrier coil may be accomplished. Their recommended locations are illustrated in Fig. X.6-6. The flanges are sealed by welds at their outer circumference to provide vacuum integrity. The flanges are designed so that the seal welds may be removed by a side mill remote cutter and subsequently

Fig. X.6-4



FACILITY LAYOUT, PLAN VIEW

NO SCALE

Fig. X.6-5

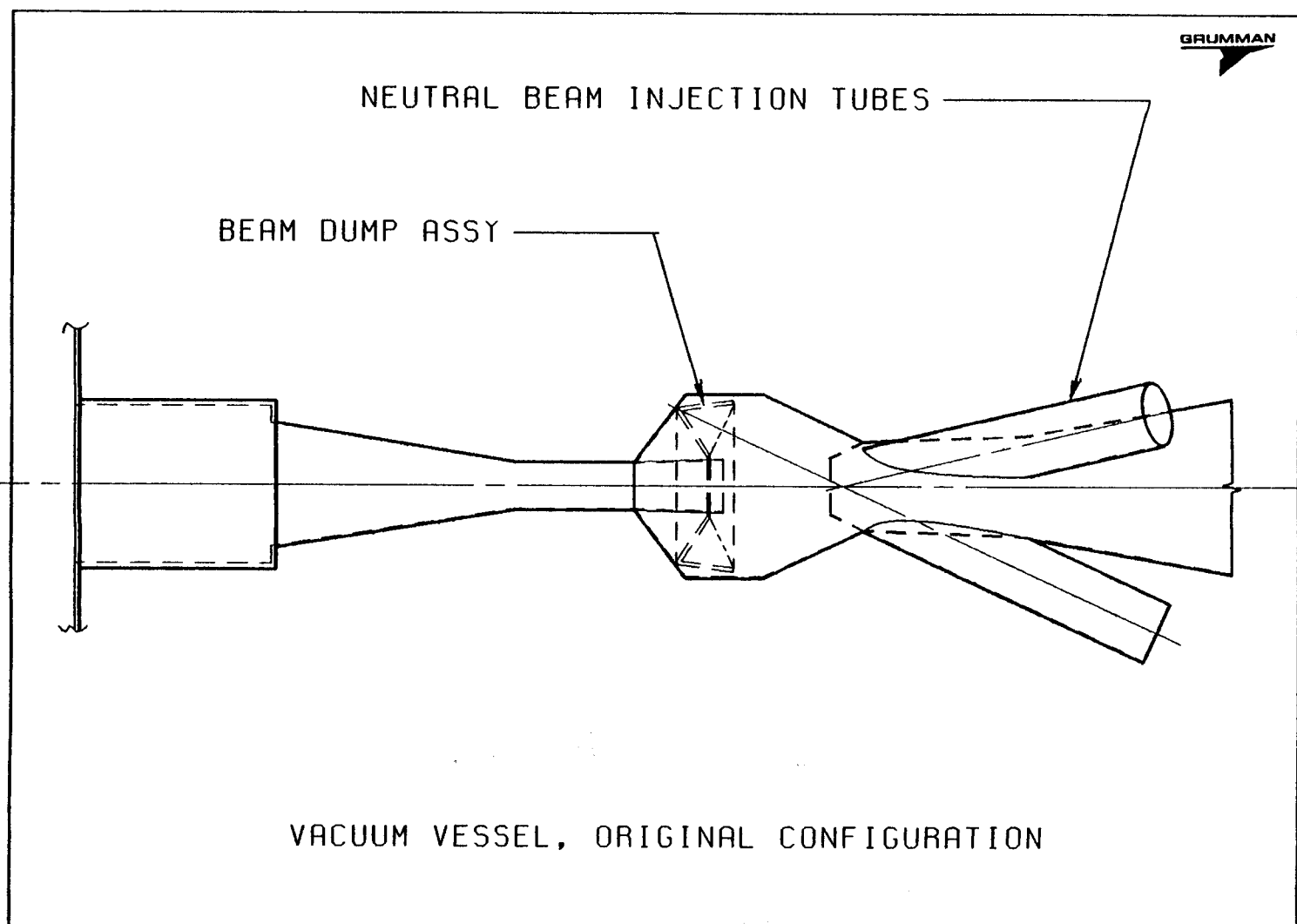


Fig. X.6-6

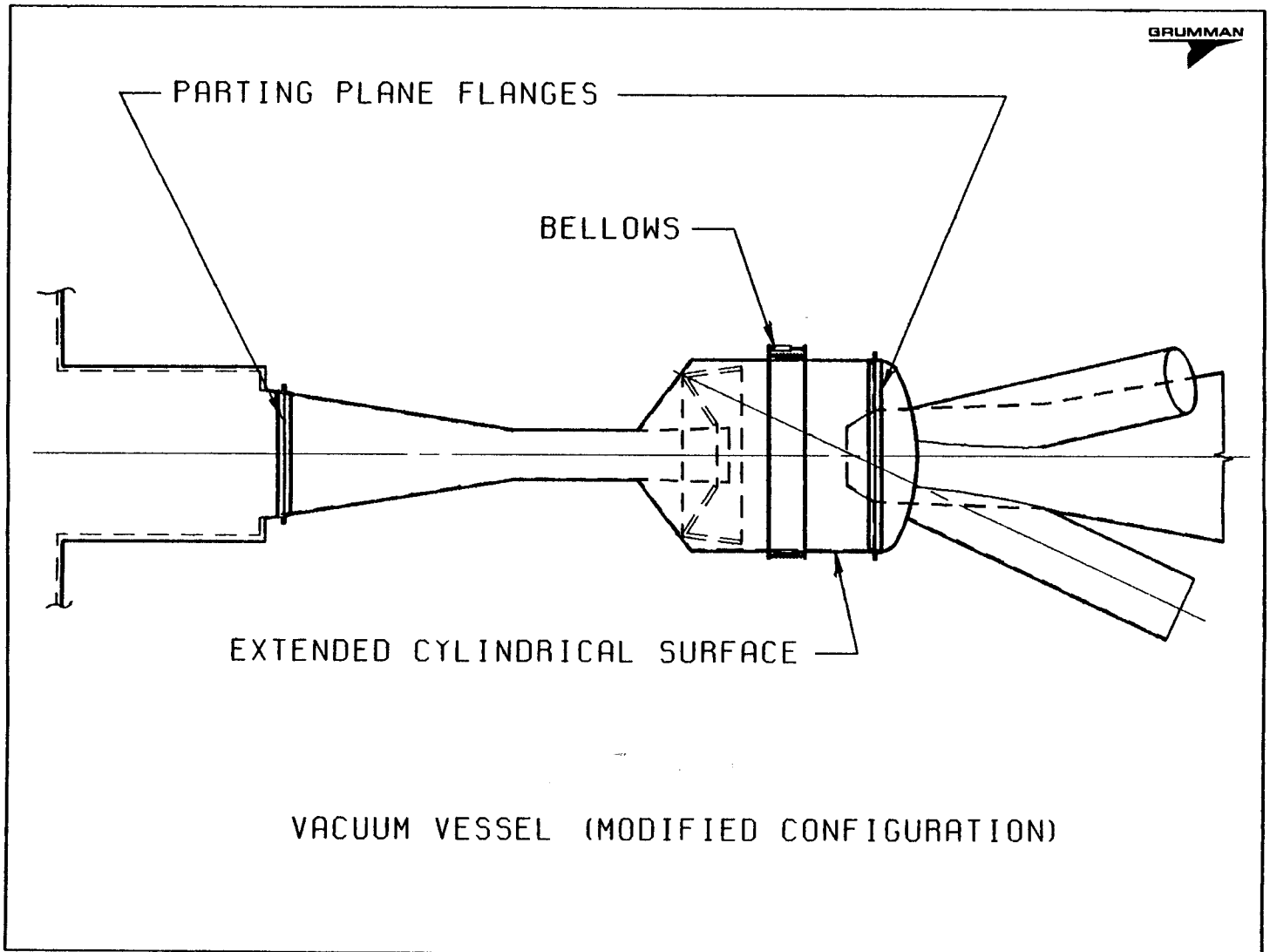


Fig. X.6-7

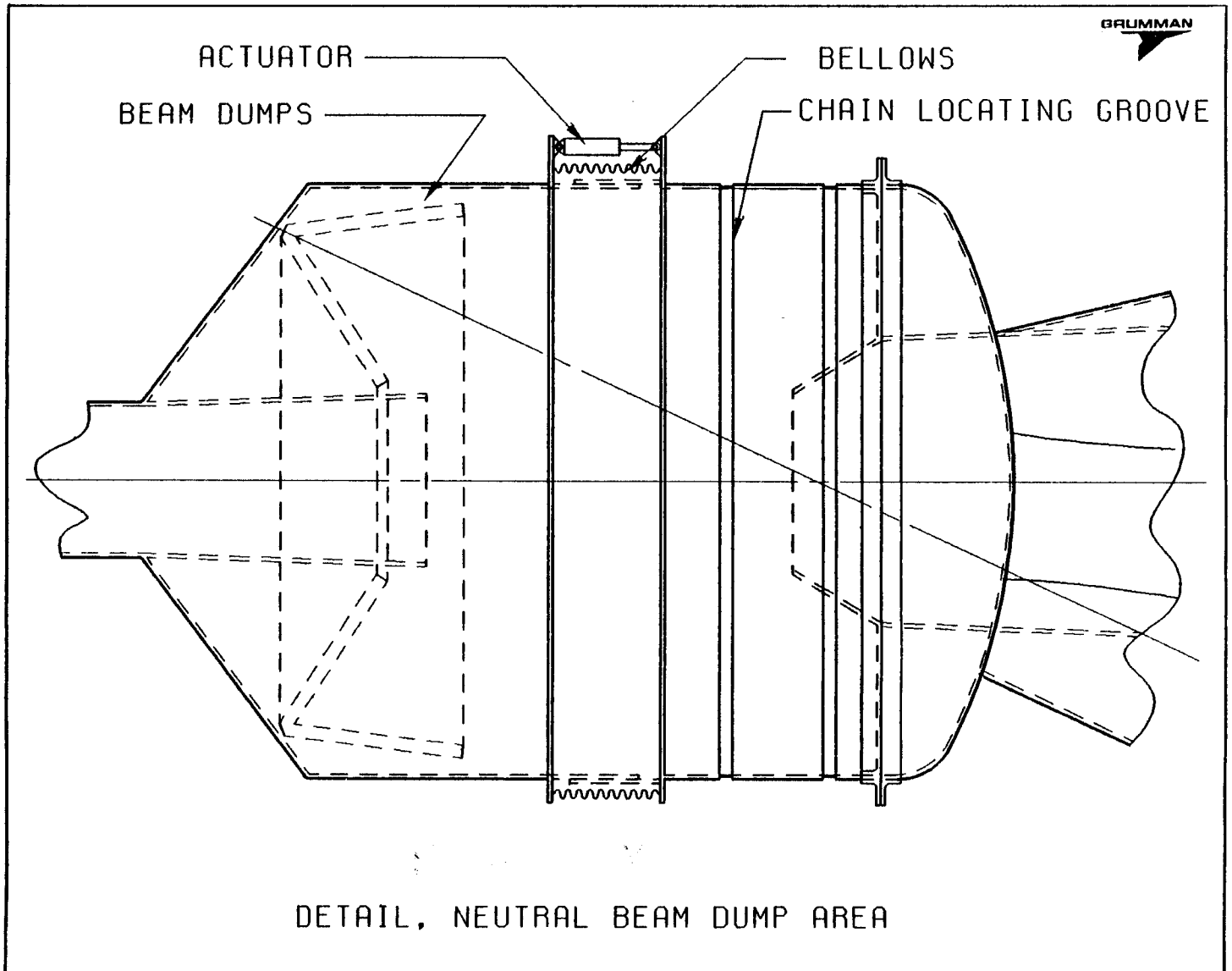


Fig. X.6-8

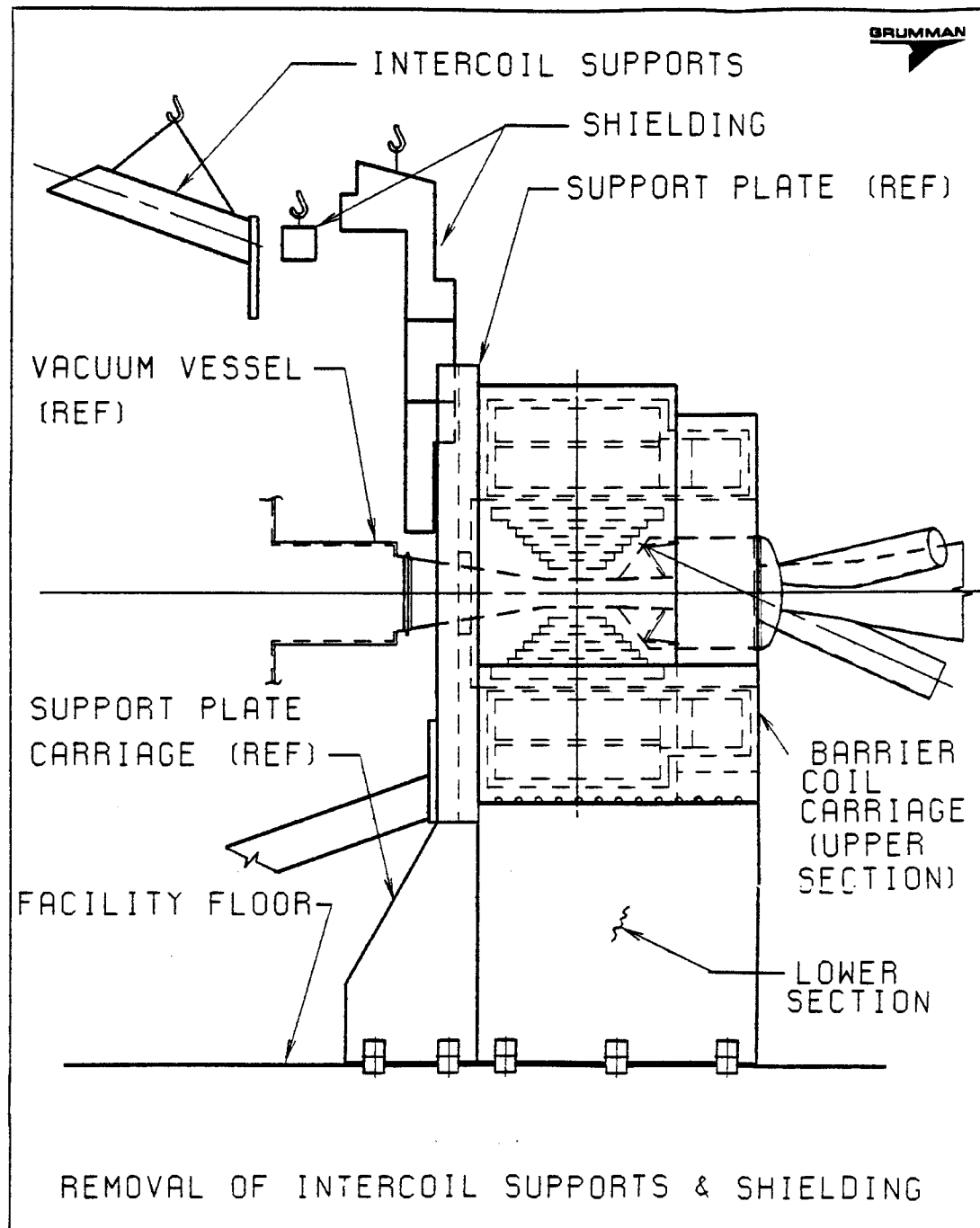


Fig. X.6-9

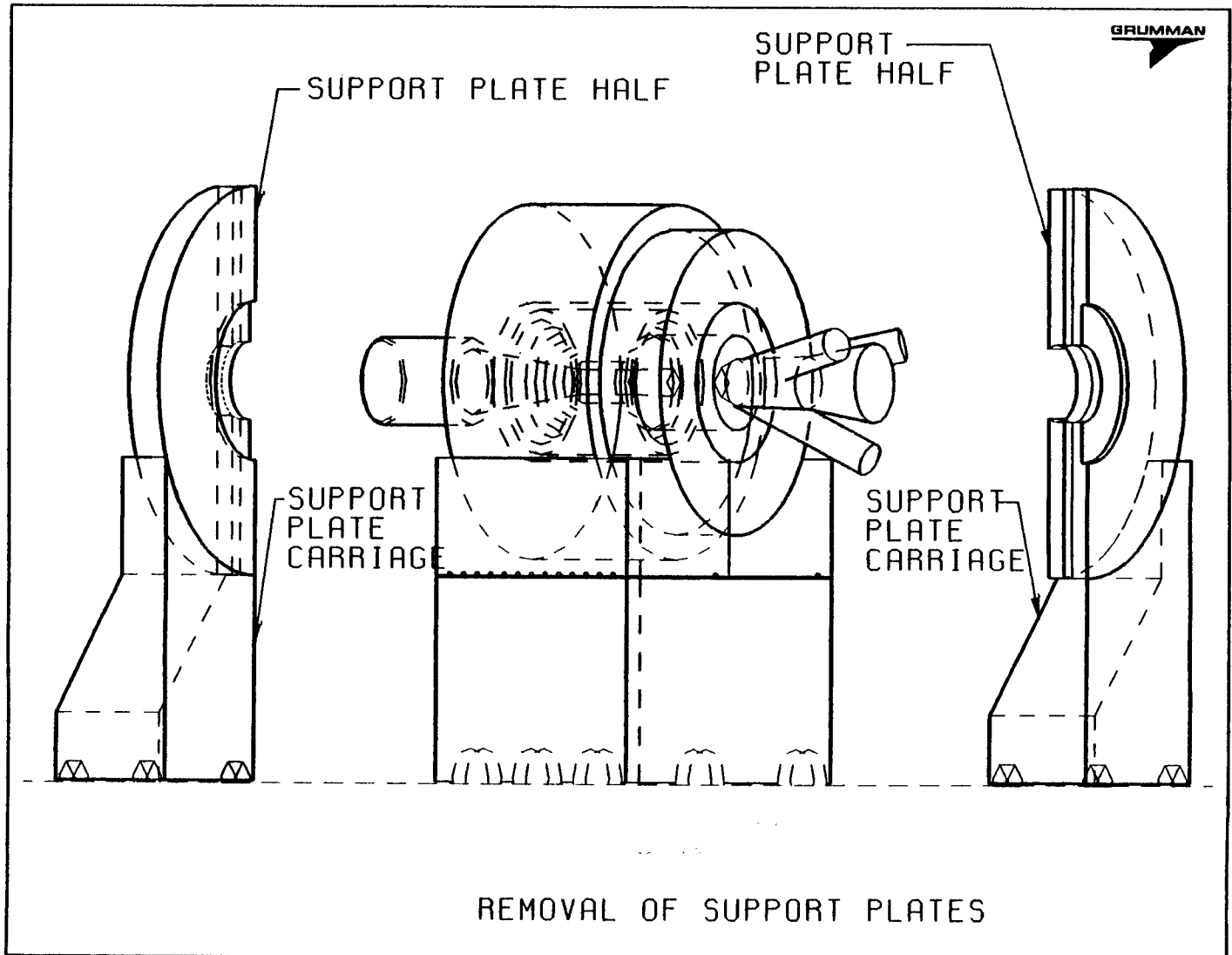




Fig. X.6-10

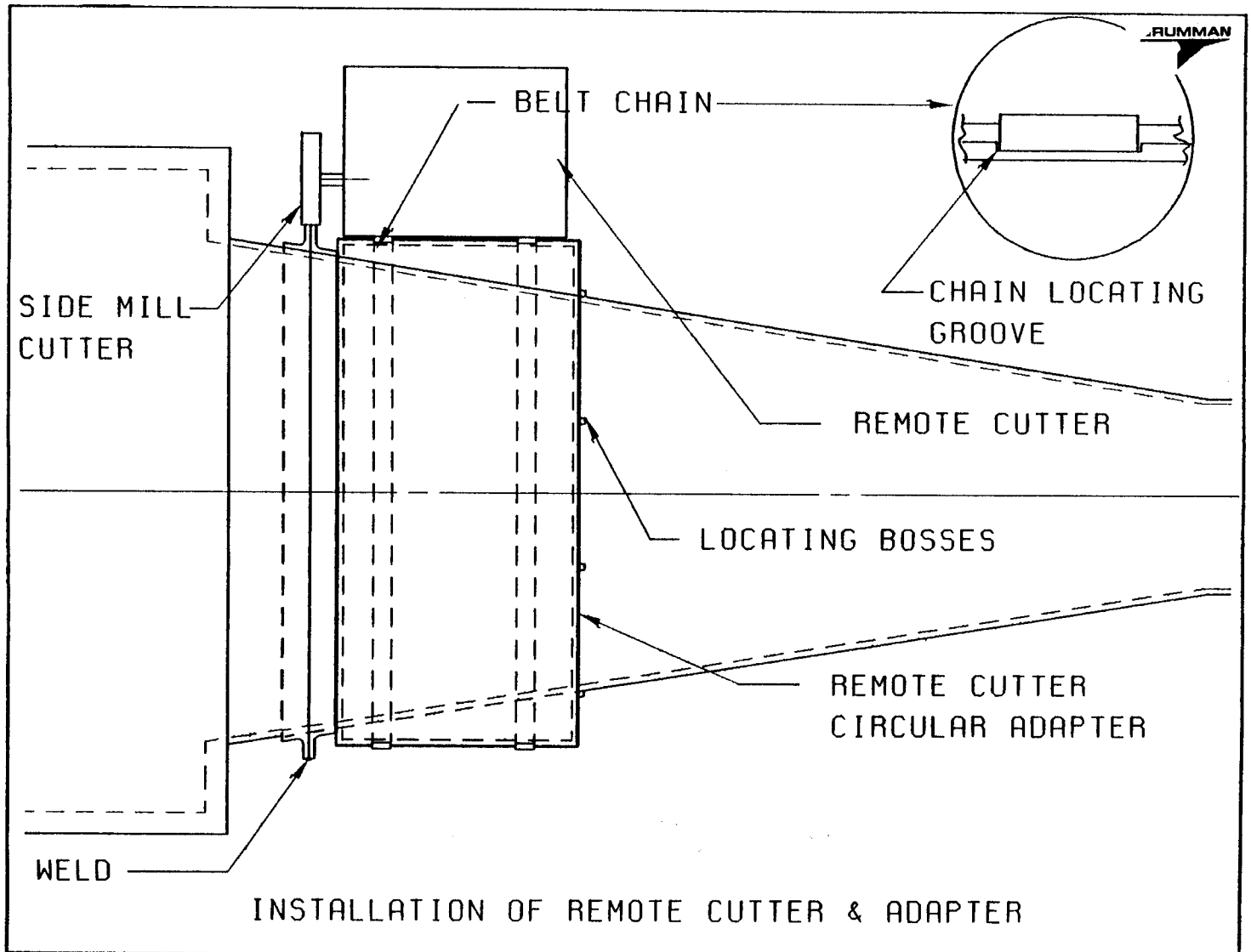
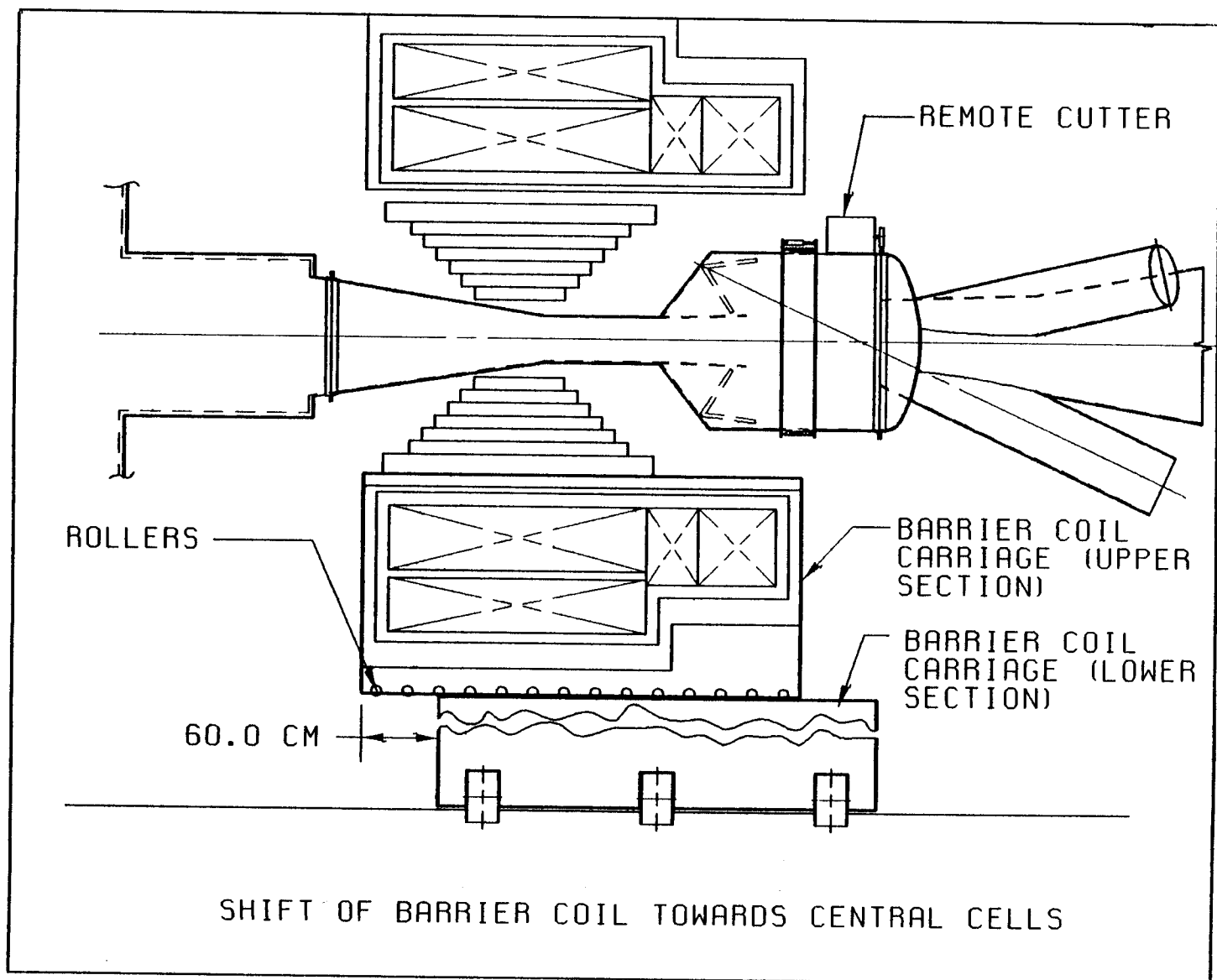


Fig. X.6-11



remotely rewelded three times (three times the estimated need). Figure X.6-7 shows one of the parting plane flanges in somewhat greater scale. Refer to Section X.6.4 for a discussion of alternate parting plane and vacuum seal designs.

A construction joint in the vacuum vessel at the barrier coil throat will be required for initial TASKA assembly.

Belt chains are used to attach commercial remote pipe cutters, remote welding machines and other types of remote maintenance equipment. Therefore, locating grooves and bosses in the vacuum vessel are required to ensure the tracking accuracy of these devices and to simplify their installation and removal by remote means. Examples of these design modifications are depicted in Fig. X.6-10.

The configuration of the vacuum vessel in the beam dump area requires modification. The cylindrical vacuum vessel surface here has been lengthened and terminated by a domed end. The conical section shown in Fig. X.6-5 has been eliminated. These modifications, which are illustrated in Fig. X.6-6, provide a secure surface upon which the remote cutter and welder can travel and track circumferentially around the vacuum vessel.

As mentioned in Section X.6.2, the approach to barrier coil maintainability is to cut the vacuum vessel in the vicinity of stations A and B, C and D and to horizontally move the entire barrier coil module normal to the machine axis to an appropriate nearby repair station where the damaged barrier coil components can be removed and replaced. Milling away the seal weld from the vacuum vessel flanges while structurally decoupling the barrier coil vacuum vessel segment from its adjacent sections will leave zero clearance at the parting plane flanges. Adequate clearance must be provided to translate

the barrier coil module to its repair station and even more importantly to re-insert, align and adjust a repaired module into its position in the machine. Any system to provide clearance for module removal must be reversible to allow accurate realignment for welding. Similar requirements exist if a mechanical joint is used instead of the seal welded joint assumed here.

A bellows section, with associated actuators (to extend and contract the bellows), has been incorporated into the vacuum vessel segment to provide the necessary clearance. At least three actuators will be required to provide smooth operation. The bellows and actuators are shown in several figures but are presented in somewhat enlarged scale in Fig. X.6-7. Alternate design approaches are discussed in Section X.6.4.

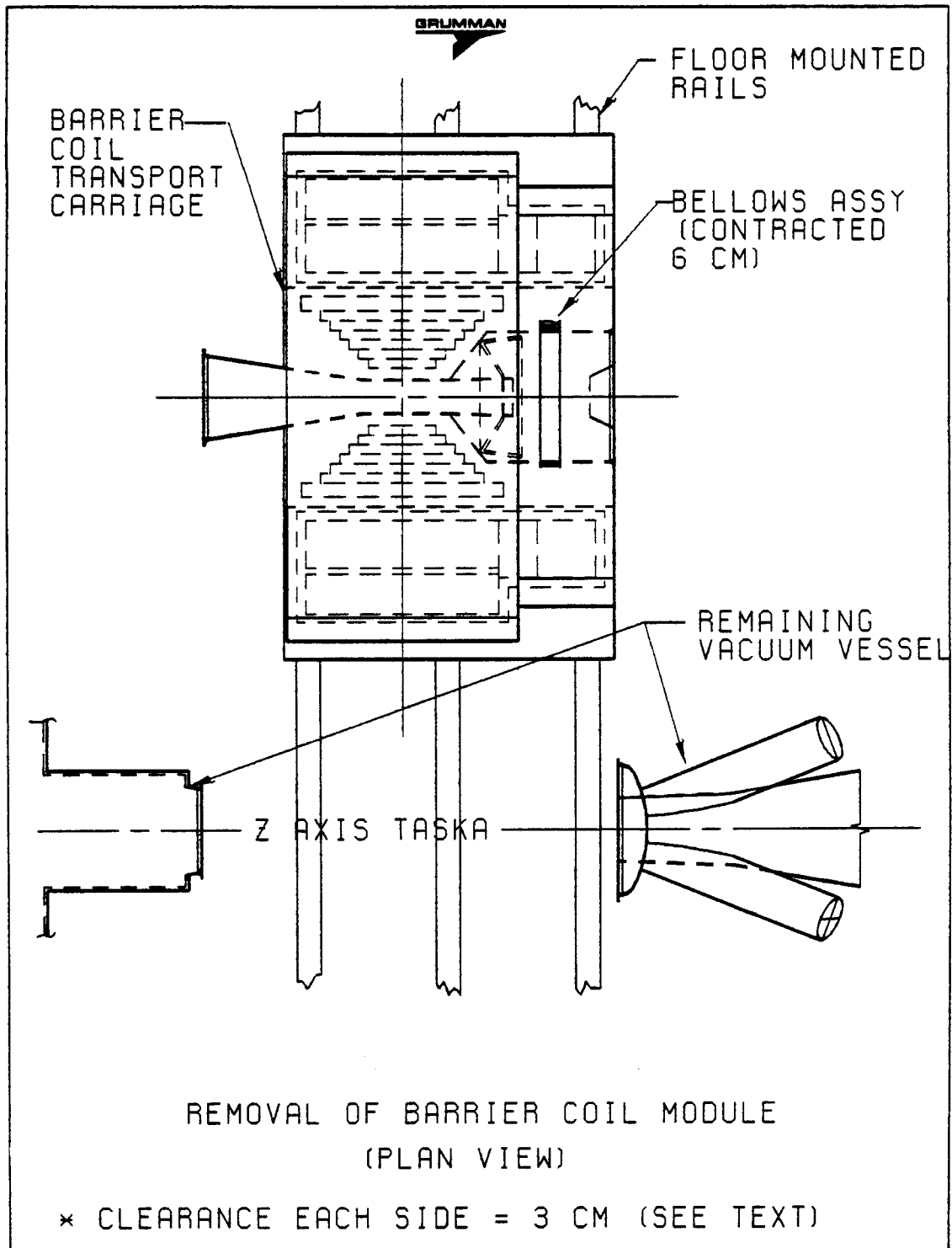
The shielding that envelopes the vacuum vessel can and must be designed in suitably sized, removable, interlocking units. "Interlocking" is used in the sense of: no straight lines of sight between shielding units will be permitted -- in order to minimize neutron streaming. No difficult design problems are envisioned in this area, however, several design features are important. The size and configuration of each shielding unit must be governed by the limitations of remote handling equipment with particular regard for weight, moment-arm and "reach" capacity. Recessed "hand holds" in each shielding unit provide gripping interfaces with remote handling equipment.

The superconducting coils are expected to satisfactorily operate throughout the TASKA lifetime. They are also extremely costly components. Therefore, a desirable maintainability goal is to retain the superconducting coils and replace only the less durable and less expensive normal coils. The lifetime of the beam dumps is expected to roughly correspond to the normal coil lifetime. It does not appear to be cost or "risk" effective to attempt to re-

use the severed vacuum vessel segment. Therefore, the following is recommended: (1) provide at least one spare, preassembled replacement module consisting of a new normal coil and a new vacuum vessel segment together with its associated internal beam dump assembly, (2) withdraw the damaged components from the superconducting coil and insert the spare replacement module and (3) accomplish the above at the time of failure of either the normal coil or the beam dumps - whichever occurs first.

The weight of the normal coil is borne by the structural case of the superconducting coil (as is the weight of the vacuum vessel, once severed). The estimated twenty-five tonnes combined weight will present a high frictional resistance to withdrawal of the damaged components from the superconducting coil and/or the associated shielding. "Binding" due to possible misalignment during withdrawal (and reinsertion) must also be considered. Three built-in tracks at the interface between the superconducting coil structural case and the normal coil's stainless steel band (and the associated shielding) are proposed). The tracks may incorporate integral track rollers or accommodate remotely attachable wheels. The function of the tracks is to provide removal and insertion clearances and to position the spare replacement module until the interfacing shielding is replaced. Correspondingly, prealigned tracks will be incorporated on each of two separate service carriages: one dedicated to the withdrawal of the damaged components, the other to the insertion of the spare replacement module. The relationship of these service carriages to the repair station is shown in Fig. X.6-4. Figure X.6-12 addresses the removal of the damaged components from the superconducting coil. Section X.6.7 discusses the design and usage of the service carriage.

Fig. X.6-12



#### X.6.4 Alternate Designs and Procedures

##### X.6.4.1 History and Candidate Approaches

The desirability of following certain design directions became apparent early in the barrier coil maintenance study. It was preferable, for example, to conceive a scheme whereby the barrier coils could be removed from the TASKA machine and transported laterally to a repair station. The barrier coil module thus becomes accessible, freed from its ends, permitting replacement of the damaged components. A separate room for the repair station was at first conceived but soon abandoned because of the needless duplication of cost and equipment.

Considered from the viewpoint of economy of time, cost, operational risk, simplicity and reliability, the practicality of treating the damaged components as an entity (rather than piece by piece repair) became apparent. Given the above concepts and considering the enormous mass of the barrier coil, it was a natural and logical conclusion to conceive the barrier coil module to be permanently mounted on carriages which would serve, during TASKA operation, merely as structural supports and during maintenance procedures as movable transport devices. Mounting the carriages on tracks logically followed. Thus, the need for costly, dedicated bridge cranes or air bearing plates with their inherent inaccuracies and high operational risk was avoided.

Some major areas for study remained: how to gain access to the vacuum vessel to cut it, how to provide the required removal and reinsertion clearances and how to withdraw the damaged components from the superconducting coil with simplicity, and reliability within the limitations inherent in remote maintenance.

Solutions to these questions are discussed in the body of this text. Some alternate approaches are given below. It is interesting to note that almost identical barrier coil maintenance procedures were derived almost simultaneously by two separate groups studying independently in two different countries.

#### X.6.4.2 Alternate Design Candidates

##### X.6.4.2.1 Vacuum Vessel Parting Flange Seals

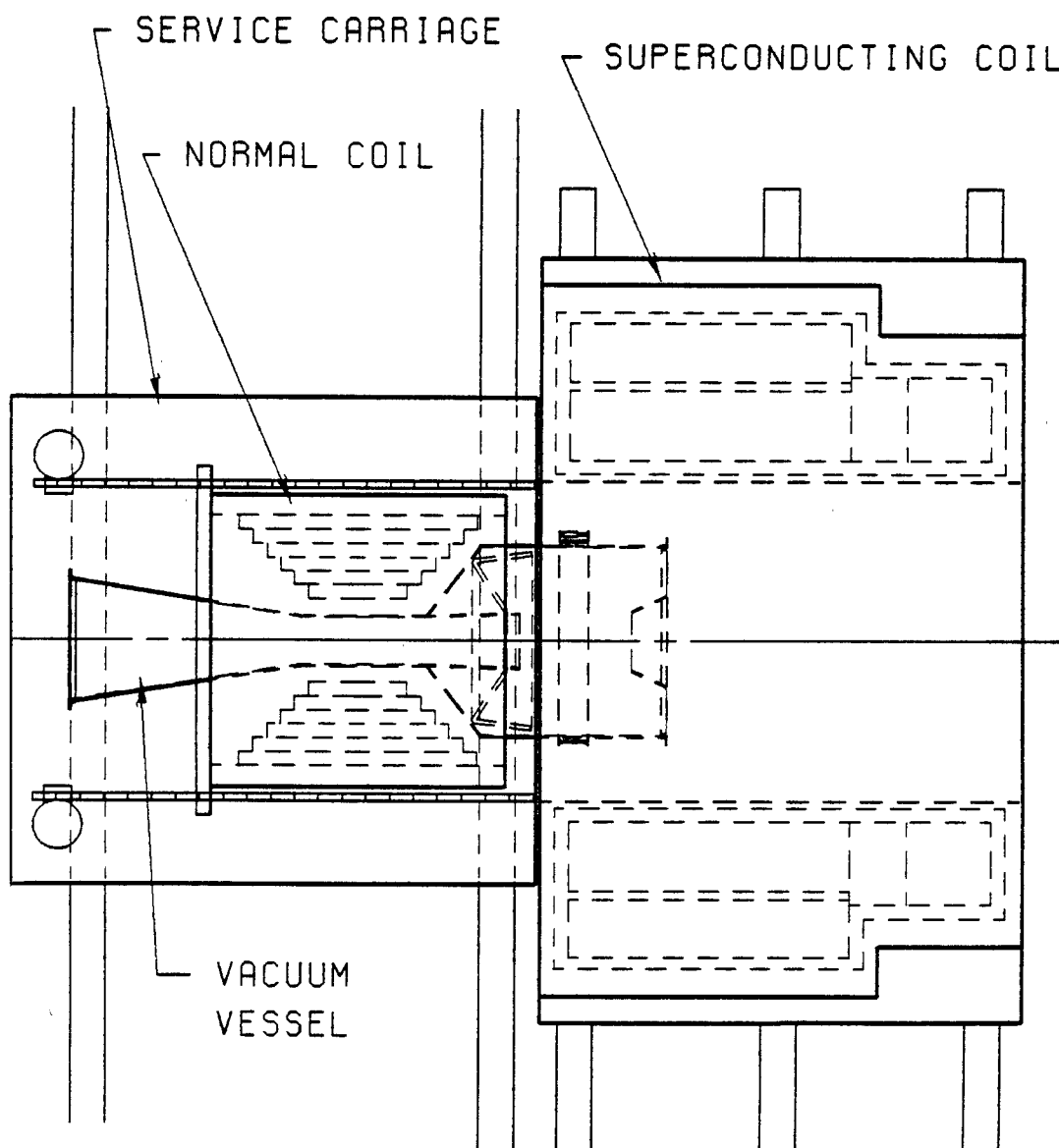
The approach selected is to seal the vacuum vessel flanges by circumferential welding. It was selected for its inherently higher reliability in comparison to mechanical seals with their susceptibility to temperature gradient induced "creep" and careful torquing and handling requirements. Nonetheless, seal welding too has its difficulties -- among them the need to acquire a remote cutter and a remote welder. It is believed that additional study should be directed to this area. It is therefore listed in Section X.6.8 "Recommendations for Further Study".

##### X.6.4.2.2 Vacuum Vessel Parting Flange Clearance Design

The method to achieve adequate clearance at the vacuum vessel parting flanges discussed in the text is to incorporate a bellows section in the vacuum vessel wall. Actuators are employed to contract and expand the bellows. A bellows section has the advantage of serving as a thermal expansion joint and the disadvantage of vulnerability to radiation bombardment. An alternate welded sleeve joint design with built-in clearance is illustrated in Fig. X.6-13. It has considerable merit and is therefore listed in Section X.6.8 "Recommendations for Further Study".



Fig. X.6-13



WITHDRAWAL OF DAMAGED COMPONENTS

#### X.6.5 Description of Selected Procedure

Replacement of the normal coil assembly may be accomplished by the following procedure.

1. Remove the high energy neutral beam (NB) injectors and store. This step is necessary only if the NB injectors are mounted in the vicinity of the barrier coil module.
2. Disconnect service lines to the barrier coil module.
3. Remove the intercoil structure using the overhead crane for lifting and transporting each section, and a general purpose manipulator unit for removal of bolts.
4. Remove the central cell shield adjacent to the barrier coil module using the overhead crane and general purpose manipulator unit as discussed above.
5. Disconnect two halves of the support plate and translate laterally.
6. Install the cutting machine adapter to the vacuum vessel segment as shown in Fig. X.6-10. The adapter is constructed of two 180° segments which clamp to the vacuum vessel. Two grooves on its outer surface are used to secure the cutting machine guide chain.
7. Install cutting machine on the adapter, connect lines to control unit and power supply.
8. Place temporary supports between the vacuum vessel segment and the coil support module. This is required to carry the vacuum vessel weight when the flange is cut. The temporary supports must allow for 60 cm axial movement between vacuum vessel and coil support.
9. Cut the vacuum vessel flange.
10. Remove the cutting machine and adapter.

11. Translate the barrier coil axially 60 cm toward the central cell using the axial motion mechanism provided in the coil support module. This motion provides access for cutting the remaining vacuum vessel flange.
12. Remove shielding around the vacuum vessel segment at the end plug side.
13. Install cutting machine as shown in Fig. X.6-11. Grooves for the cutting machine drive chains are machined into the vacuum vessel surface.
14. Cut the vacuum vessel flange.
15. Remove cutting machine.
16. Translate barrier coil axially 60 cm to original position.
17. Retract the bellows section of the vacuum vessel (see Fig. X.6-6).
18. Connect a tow unit to the barrier coil module and transport it to the maintenance area.
19. Position an "insert" replacement module at the maintenance area. This module contains a replacement normal coil (NC) assembly, and has provision for receiving the damaged NC assembly after removal. Also, it contains linear actuators for removal and installation of the NC assembly from the superconducting coil (SC) assembly. Three rails on the insert replacement module lineup with similar rails in the SC assembly.
20. Remove the damaged NC assembly and install a replacement NC assembly.
21. Connect service lines between NC assembly and coil support module.  
Perform checkout tests on NC assembly.
22. Transport barrier coil module to its operational location.
23. Align vacuum vessel segment with the plasma axis using adjustments provided on the coil support module.
24. Assemble the barrier coil module to adjacent modules using a reverse order of the above procedure. Flange welding is performed by a welding

machine which mounts to the vacuum vessel similar to the cutting machine described above.

#### X.6.6 Facility Requirements

The following list presents the facility equipment which (in addition to the requirements of previous sections) will be specifically required to maintain the barrier coils. Figure X.6-4 presents a plan view of the suggested facility layout.

- Six floor mounted tracks extend laterally from each barrier coil. Their function is to transport the barrier coil carriage from the TASKA machine to the repair station and back again, to transport the support plate carriages to and from their appropriate temporary storage structures and lastly, to transport both service carriages from the storage area to the repair station and back again. Floor mounted tracks were selected in lieu of overhead bridge mounted cranes because of their inherent capacity to support huge weights, for easier and more accurate guidance and for their capability to incorporate stops and positioning devices.
- Two storage structures for each barrier coil for the temporary safekeeping of the support plate "halves". These structures are secured to the facility floor immediately adjacent to the floor mounted tracks which are dedicated to support plate removal. They are constructed of standard catalog structural members. It is planned that each support plate half will remain mounted on its appropriate carriage throughout the barrier coil maintenance procedures until reinstallation. The storage structures serve only as "hitching posts" to temporarily secure the heavy mass of the support plate halves.

- Storage areas for the spare barrier coil replacement modules and associated support equipment.
- "Hot" storage area for safekeeping of removed and damaged components.
- Standard leak detection equipment.
- At least two spare replacement modules (one for each barrier coil) consisting of a normal coil and a vacuum vessel segment together with its internal beam dump.

#### X.6.7 Special Equipment Requirements

The following items are required for the procedure described in Section X.6.5.

##### X.6.7.1 Remote Cutter Assembly

Existing remote pipe cutters used in the petroleum industry are suitable for the proposed procedure. A belt chain (similar to a "silent" chain) is attached circumferentially to the vessel to be cut. The cutting mechanism is pre-attached to the chain. A pneumatic self-contained motor drives the mechanism circumferentially around the vessel and powers a slide milling cutter. Various sizes and configurations of cutters are available "off-the-shelf". Various size chains are available to meet TASKA needs. The chains include a "quick release" device and tension adjustment is incorporated. A temporary attachment fixture to support the cutter on the vacuum vessel while the chain is being installed will be provided.

The commercial remote cutter assembly may need to be modified to incorporate provisions for cooling gas to be directed toward the cutter teeth to reduce heat buildup (due to cutting the vacuum vessel flanges without the benefit of lubricating fluids) and to remove cuttings from the cutter teeth. Experience with TFTR has shown that chips tend to weld themselves (albeit

somewhat loosely) to the cutter teeth and to accumulate in small masses which subsequently loosen and find their way between the teeth and material being cut. An unsatisfactory cut may result. A transparent plastic shield or hood to momentarily contain and collect cuttings must be incorporated, as well as a vacuum source to remove cuttings from the plastic hood and vacuum vessel area. Additionally the remote cutter will be modified for interface with remote handling equipment. Reliability requirements on the manufacture may require strengthening.

#### X.6.7.2 Remote Welder Assembly

Commercial tungsten inert gas welders and associated controls are available and are suitable for the proposed procedure without extensive modification. The only modifications envisioned will be for interface with remote handling equipment (e.g., remote manipulators).

#### X.6.7.3 Barrier Coil Carriage

The barrier coil carriages (one is required for each barrier coil) are the structural supports for the 2 barrier coil modules. They have the added functions of adjusting and transporting the coils in various directions for maintenance purposes. In fact, each barrier coil carriage has four functions: it supports and positions the barrier coil module in the TASKA machine - reacting the gravity loads to floor mounted tracks; it adjusts the barrier coil module to adjacent TASKA components; it translates the barrier coil module along the plasma axis and lastly, it transports the barrier coil module to the repair station where it will interface with both service carriages.

The carriage is designed in an upper and a lower section. The upper section cradles and supports the superconducting coil structural case. It interfaces with the lower section through three sets of rollers which permit

at least 60 centimeters of translation of the barrier coil module along the plasma axis for improved access to the vacuum vessel parting flanges on the end plug side of the barrier coil module. Figure X.6-11 shows the upper section translated sixty centimeters towards the central cell.

The lower section is mounted on wheels which, in turn, are set into the floor mounted tracks illustrated in Fig. X.6-4. The primary function of the lower section is to transport the upper section, together with the barrier coil module, from the TASKA machine to the repair station and, eventually, back again. It also includes provisions to drive the upper section horizontally along the plasma axis and to adjust and align the barrier coil module with respect to adjacent TASKA components. The lower section incorporates measures to secure each to the other and to control the relative motions between them.

Section X.6.3.2 discusses the need to provide clearance at the vacuum vessel parting flanges and proposes a device to provide six centimeters total clearance. Advantage will be taken of the barrier coil carriage's axial translation capability to "split-the-difference", thereby assuring equal clearance on each side of the severed vacuum vessel.

#### X.6.7.4 Service Carriages

There are two service carriages. They are similar in design but each is dedicated to a specific task. Their relationship to the facility, and to the barrier coil carriage is illustrated in Fig. X.6-4. The service carriages will be used to maintain both barrier coils.

It is preferable that prior to the removal of TASKA shielding and while "hands on" operations are possible, the first service carriage, containing a spare replacement module, be moved to the position shown in Fig. X.6-4. The

second service carriage will follow behind the first and be secured to the facility floor at a preselected position at the repair station - ready to remove the damaged barrier coil components and subsequently transport them to the "hot" storage area.

The service carriages will incorporate equipment (e.g. power screws) to withdraw the damaged components from the barrier coil module and insert replacements. An adapter plate (integral with the withdrawal equipment) will interface with and secure to the stainless steel band encircling the normal coil (previously described in Section X.6.3.1).

Tracks are incorporated on the service carriages which correspond to and will mate with the "built-in" tracks of the barrier coil module which were previously discussed at the end of Section X.6.3.2. The function of these tracks is to support and guide the coil components during the removal and insertion processes. The service carriages include provisions to adjust their tracks to those of the barrier coil module and it is planned that all necessary adjustments and alignments will be accomplished manually prior to the first TASKA operational cycle.

The service carriages include provisions for them to be secured to the facility floor and to the barrier coil carriage as well as devices to correctly position them with respect to the barrier coil module.

Figure X.6-12 shows the role of a service carriage during the withdrawal of damaged components. An alternate design is discussed in Section X.6.5.

#### X.6.7.5 Support Plate Carriages

Because it is necessary to remove the support plate (to gain access to a vacuum vessel parting flange) it is designed in two vertical interlocking halves and each half is cradled and supported by a support plate carriage so



that horizontal translation normal to the plasma axis is possible. In turn, the support plate carriage interfaces, through wheels, with floor mounted tracks. Positioning devices and stops are required in the support plate carriage and its associated tracks to simplify remote maintenance. The support plate carriage is illustrated in relationship to TASKA and the facility in Fig. X.6-4 and is also shown in Fig. X.6-8. Its remote maintenance usefulness is shown in Fig. X.6-9.

#### X.6.7.6 Remote Cutter/Welder Mounting Adapter

The conical section of the vacuum vessel adjacent to the central cell is not suitable for mounting a remote cutter and welder. Therefore, an adapter is required to encompass the conical section and to provide a secure cylindrical surface upon which the cutter and welder can travel and track circumferentially around the vacuum vessel.

The adapter contains the necessary positioning devices to simplify remote installation of the cutter and welder. It is constructed in two interlocking halves with quick release latching devices and remote handling grips for ease of installation. It is positioned on the vacuum vessel between the parting flanges and locating bosses incorporated on the vessel wall. These bosses are designed to ease adapter installation under remote handling conditions.

#### X.6.7.7 Vacuum Vessel Temporary Support

Temporary supports will be required to support and position the vacuum vessel segment at the parting flanges during the cutting operations. Their purpose is to assure cutting accuracy and prevent damage to the vacuum vessel segment and to the cutting tools. At present, they are envisioned as straps attaching the vacuum vessel segment to the superconducting coil structural case.

#### X.6.8 Recommendations for Further Study

Further study is highly recommended with the effort directed toward revision of the present baseline design in order to simplify maintenance procedures and reduce costs.

##### X.6.8.1 Vacuum Vessel Parting Flange Seals

A number of good mechanical seals are currently available. Their design is within state of the art. Particular attention must be devoted to the effects of thermal gradients on structural flange design. A successful conclusion of this study would eliminate the need for a remote cutter and welder and, of course, the need to install these devices under remote handling conditions.

##### X.6.8.2 Vacuum Vessel Parting Flange Clearance Design

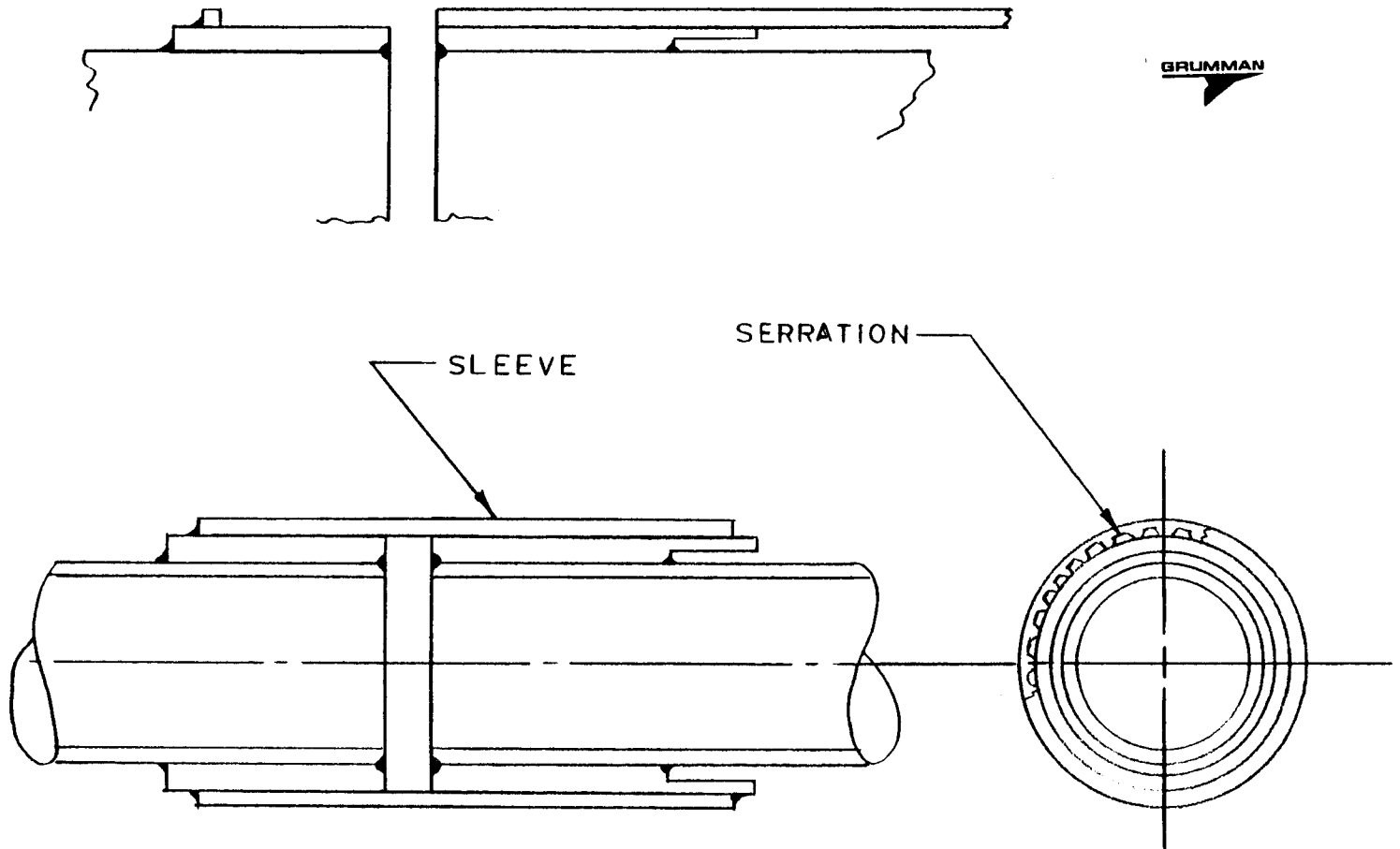
The alternate design shown in Fig. X.6-14 has considerable merit with the advantage of eliminating the bellow-actuator section of the vacuum vessel segment. Additional study is required to develop procedures to axially translate the sleeve without binding under remote conditions. A stress analysis of the loads in the welds will be required.

##### X.6.8.3 Relocation of Vacuum Vessel Parting Flange

When additional design definition is available, it is recommended that an indepth study be conducted to determine if the vacuum vessel parting flange can be axially relocated towards the end plug side of the barrier coil module thus eliminating the need for axial translation of the barrier coil module and reducing the complexity of the barrier coil carriage. This approach, if possible, would result in a much easier remote maintenance task.

The installation of a remote cutter circular adapter similar to that shown in Fig. X.6-10 might accomplish the same purpose.

Fig. X.6-14



ALTERNATE DESIGN OF PARTING FLANGE

The baseline approach described in the body of the text, however, can be successfully employed for all configurations of the vacuum vessel and superconducting coil considered so far during the TASKA study effort.

## X.7 Availability and Maintainability

### X.7.1 Introduction and Summary

TASKA steady state availability has been analyzed. The system has been broken down into several main subsystems and a logic diagram of the device has been put together. A computer code using the Monte Carlo method has been used to find the availability that can be expected.

Since the computer code is still under development, a very simplistic model resulted. Therefore, the results would probably tend to be on the optimistic side. However, both routine maintenance and unexpected random failures were taken into account. Redundancy and spares of certain subsystems and overdesign (i.e., more subsystems operating than the minimum needed for successful operation) have also been incorporated.

Due to lack of data, only subsystem levels have been treated. This means that no attempt has been made yet to assess the availability of each subsystem using the failure rate and repair rate data of constituent components. Instead, we employed the failure rate and repair rate data estimates for each subsystem. Large spreads in estimates have been observed, hence three cases have been run: one, using the pessimistic set of data, which yielded an overall steady state availability of 24-30%; secondly, the median set of data, yielding 40-48% availability; and the optimistic set of data yielding an average availability above 70%. In order to improve availability, it is suggested that either the quality of the magnets be increased and/or their number decreased. To a lesser extent, the same recommendations apply to the neutral beam and the RF heating subsystems.

### X.7.2 Main Subsystems in TASKA and Their Logic Interconnection

The main subsystems in TASKA are presented in Table X.7-1. Some simplifications are evident in this table; for instance, both the permanent and the test blanket have been lumped under the term "blanket". The more frequent replacement of the test blanket is assumed to take place during the annual shutdown period.

Table X.7-1 also provides information on the total number of each kind of subsystem operating, the minimum number necessary for operation of the plant and number of redundant subsystems (meaning idle subsystems that can be switched on instantaneously to substitute for a failed subsystem of the same kind). Note that, in order to increase the availability, a certain amount of liberty was used. For example, redundancy was provided for the 250 keV neutral beam subsystem, even though the design does not specifically call for it, because sufficient space exists for installing another 250 keV injector.

The logic diagram of TASKA is given in Fig. X.7-1; only the most important interconnections have been shown.

### X.7.3 The Computer Code for Analysis of Availability

The computer code we used to analyze the availability of TASKA employs the Monte Carlo method. This means that the reliability  $R_i$  of the  $i$ th subsystem in a time interval  $\Delta t$ ,  $R_i = \exp(-\lambda_i \Delta t)$ , is compared to a random number  $N_n$  (between 0. and 1.) generated by the computer. If  $N_n < R_i$ , no failure of that subsystem occurs in this time interval; if  $N_n > R_i$ , a failure occurs in this time interval.

$\lambda_i$  is the failure rate of subsystem  $i$ . It can be constant in time (random failures, which was used here) or it can have a general dependence on time of the form:

Table X.7-1. Main Subsystems in TASKA

| Subsystem                   | Number of Units | Redundancy | Minimum Number |
|-----------------------------|-----------------|------------|----------------|
| Shield                      | 1               | 0          | 1              |
| Blanket                     | 1               | 0          | 1              |
| CC Coils (1 side)           | 3               | 0          | 2              |
| CC-TB Coils                 | 2               | 0          | 2              |
| Plasma Cross Section Coils  | 2               | 0          | 2              |
| Yin-Yang Coils              | 2               | 0          | 2              |
| Recirc. Coils               | 2               | 0          | 2              |
| TB Coils                    | 8               | 0          | 8              |
| Cryosystem and Vacuum       | 1               | 0          | 1              |
| Neutral Beams (one side)    |                 |            |                |
| 250 keV                     | 1               | 1          | 1              |
| 50 keV                      | 1               | 0          | 1              |
| 76 keV                      | 1               | 0          | 1              |
| 2 keV                       | 1               | 0          | 0              |
| ECRH (one side)             | 2               | 0          | 2              |
| ICRH                        | 16              | 0          | 13             |
| Direct Convertor (one side) | 1               | 0          | 1 either side  |
| Instrumentation             | 1               | 0          | 1              |
| Balance of Plant            | 1               | 0          | 1              |
| Tritium System              | 1               | 0          | 1              |



Fig. X.7-1. LOGIC DIAGRAM OF TASKA



$$\lambda(t) = \lambda_c \beta t^{\beta-1}$$

where  $\lambda_c$  is a constant.

If this relationship is plotted with  $\beta$  as a parameter, Fig. X.7-2, a bathtub curve results, which is a typical time dependence of  $\lambda$  over the life of a component. For  $\beta < 1$ , we have a decreasing failure rate in time. For obvious reasons, this period is called the "burn-in" or "debugging" period. It is expected that this will last about 7 years for TASKA, or about the first half of the plant lifetime. The burn-in period is usually followed by a "random failure" period where  $\lambda$  is constant and  $\beta \approx 1$ . The last portion of the curve is the "wear out" period, where  $\beta > 1$ , and the failure rate is increasing in time. We have concentrated on the random failure period of each subsystem for simplicity, assuming that the worn out components will be replaced during the 28-day annual scheduled maintenance period.

The random number generator employs the recursive congruential method. This means that the random number is related to the remainder of division of the previous random number by a constant.

Basically, the computer code works as follows. The input data is read in; this includes the length of the time step, number of time steps, data for each subsystem (failure rate, mean time to repair or MTTR, number of identical units, redundancy, minimum number necessary for operation), logic interconnections (number of AND and OR gates, identification of subsystems and gates tied to the inputs of each gate).

The clock is then started and the reliability of  $R_i$  of each subsystem is computed and compared to a random number. In case of failure, the subsystem is down until the elapsed time is greater than the MTTR of that subsystem.

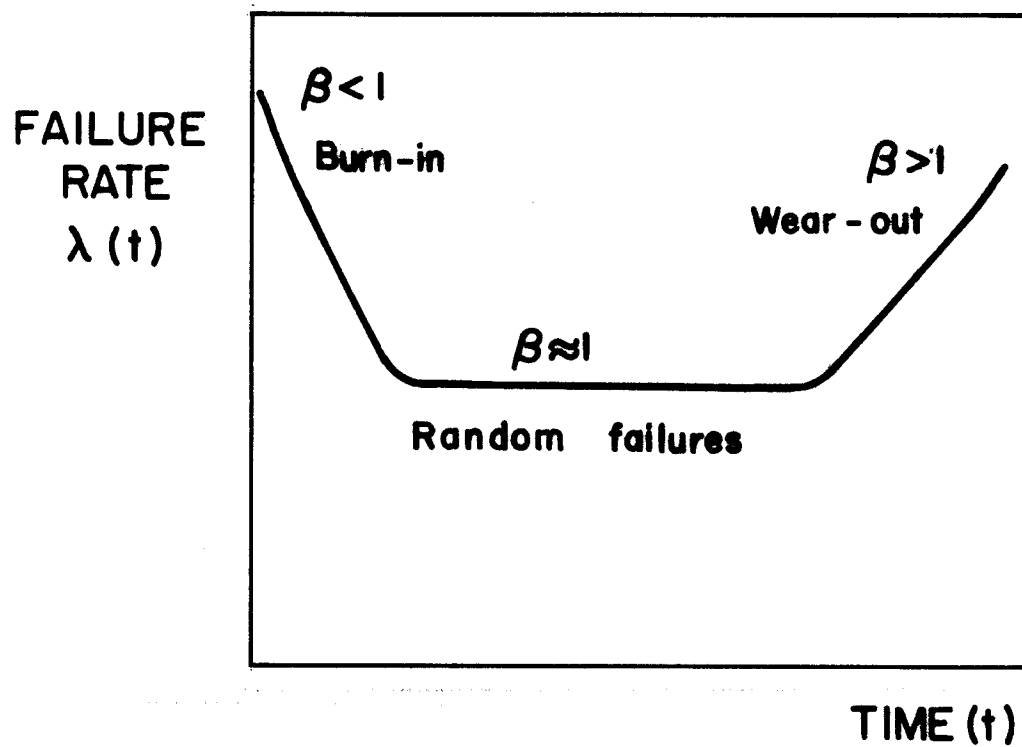


Fig. X.7-2. The bath tub curve.

The availability of each subsystem during a mission is computed as:

$$a_i = \frac{\text{total up time during mission}}{\text{total mission time}} .$$

The status of each gate is computed from the states of all its inputs as follows:

for an AND gate with inputs  $i=1, \dots, N$ :

$$A_{\text{AND}} = \prod_{i=1}^N a_i$$

where  $a_i$ 's are either availabilities at the inputs (for time independent analysis) or the states of the inputs (0's or 1's).  $A_{\text{AND}}$  is then the average availability of the gate or its state during a particular time step (0 or 1).

For an OR gate:

$$A_{\text{OR}} = 1. - \prod_{i=1}^N (1. - a_i) .$$

The availability of the system is then equal to the availability at the output of the last logic gate in the logic diagram of the system.

At this time, the computer code can handle spares, redundancy, overdesign (greater number of subsystems than the minimum necessary), immediate repair.

The assumptions built into the code at this time are that there is a 28-day shutdown period for each year of operation of the plant, in order to accomplish routine maintenance. In case of shutdown, no residual failures are assumed to occur. Only constant failure rates are treated at this time.

#### X.7.4 Data

For obvious reasons, only crude estimates of the data needed exist. Due to the spread in data, three types of runs were made: one, using the most pessimistic data (both for the failure rates and the MTTR's); secondly, using the most optimistic data; and thirdly, using most reasonable data which are similar to the data employed in the ETF availability estimate.<sup>(2)</sup> The data have been found in recent fusion design reports and papers which are listed in the Reference section, and also from discussions with the people in the field.

The data used are presented in Table X.7-2.

#### X.7.5 Results

The results for the steady state availability under the above assumptions and assuming a 28-day scheduled annual shutdown period, are as follows:

24-30% for the most pessimistic set of data;

40-48% for the most reasonable set of data;

74% for the most optimistic set of data.

Due to the simplifying assumptions employed, the true values may be lower.

The availability drivers are magnets, because there are so many of them and because their MTTR is relatively long (~ 40 days), due to the long time for warm up and cool down required for these cryogenic components. In order to improve the overall availability, the quality of magnets must be increased. The same holds true, to a smaller extent, for the neutral beam and the RF heating subsystems.

#### X.7.6 Maintainability

Essentially the same philosophy of maintainability will be followed as in previous fusion plant designs.<sup>(6,1)</sup>

Table X.7-2. Failure Rates and Mean Times to Repair for  
Various Subsystems in TASKA

| Subsystem             | $\lambda(\text{hr}^{-1})$ | MTTR (hr)            |
|-----------------------|---------------------------|----------------------|
| Shield                | 2.3E-5, 1.9E-5, 5.7E-6    | 24., 240.            |
| Blanket               | 1.7E-4, 1.1E-4, 1.9E-5    | 120., 240.           |
| All Coils             | 3.8E-5, 4.6E-6, 2.9 E-6   | 1056., 3600., 26298. |
| Cryosystem (Vacuum)   | 5.7E-5, 4.1E-5, 2.3E-5    | 24., 432., 4824.     |
| All Neutral Beams     | 1.4E-3, 2.3E-4, 5.7E-5    | 24., 240., 864.      |
| All ECRH              | 1.0E-3, 4.6E-4, 1.1E-5    | 240., 480.           |
| ICRH                  | 1.1E-4, 5.7E-5            | 240.                 |
| All Direct Convertors | 2.3E-4, 1.1E-4, 2.3E-5    | 950.                 |
| Instrumentation       | 5.7E-5, 2.3E-5            | 144.                 |
| Balance of Plant      | 1.1E-4, 5.7E-5            | 240.                 |
| Tritium System        | 2.3E-5                    | 60.                  |

The mirror reactors pose fewer accessibility problems than the tokamaks. The central cell can be built out of modular sections, which can be replaced using an overhead gantry crane. An approach similar to that in the WITAMIR-I reactor can be employed here to provide maintenance of the blanket, coils and the shield sections.

#### References for Section X.7

1. "STARFIRE - A Commercial Tokamak Fusion Power Plant Study," Argonne National Laboratory, ANL/FPP-80-1 (1980).
2. "Preliminary Availability Assessment and Apportionment of the Engineering Test Facility," Final Report, prepared for the Oak Ridge National Laboratory by Grumman Aerospace Corporation.
3. E.R. Hager, "Remote Maintenance of FED Neutral Beam Injectors," General Atomic Company.
4. P.T. Spampinato, "Considerations for Replacing PF Coils on FED," Grumman Aerospace Corporation.
5. "FRG Contributions, TASKA Review Meeting," (November 1981).
6. B. Badger et al., "WITAMIR-I, A University of Wisconsin Tandem Mirror Reactor Design," University of Wisconsin Fusion Engineering Program Report UWFD-400, (1980).

## X.8 Conclusion

To achieve successful operation of TASKA with high availability, the maintenance of the whole facility as well as its major components has been considered. A steady state availability has been analyzed, where both routine maintenance and unexpected random failures were taken into account. The results show that the availability drivers are magnets and, to a smaller extent, neutral beam and RF heating subsystems.

To demonstrate the practicability of maintenance tasks, four typical cases have been selected; i.e. the exchange of blanket modules, neutral beam injector components, central cell and barrier coils. Reasons for this selection were either the frequency of required replacement or the difficulty and complexity of the exchange. The latter necessitated a more detailed approach to the barrier coil replacement.

Due to the great variety of handling tasks to be performed under greatly varying environmental conditions, a handling system was conceived which comprises hands-on devices and tools as well as remotely handled equipment.

The consideration of critical maintenance operations impacts on component design but it can also be assumed that the maintenance problems are soluble.

## XI Costs

### XI.1 Introduction

The direct costs developed for TASKA follow the format adopted for INTOR. Thus, the accounts are divided into the following categories:

- 1.1 Blanket and Shield
- 1.2 Magnets
- 1.3 Plasma heating
- 2.0 Reactor Support systems
- 3.0 Buildings

The Reactor Support Systems in turn include the following:

- 2.1 Electrical
- 2.2 Auxiliary Cooling
- 2.3 Instrumentation and Control
- 2.4 Fuel Handling and Storage
- 2.5 Maintenance Equipment
- 2.6 Primary Heat Transport
- 2.7 Secondary Heat Transport
- 2.8 Reactor Vacuum System
- 2.9 Radwaste Treatment
- 2.10 Thermal Dumps
- 2.11 Direct Convertor
- 2.12 Reactor Support Structure.

Wherever possible, unit costs and cost algorithms were obtained from the "INTOR - Summary of Cost/Schedule/Manpower" FEDC-M-81-SE-062. The costs which were not available from INTOR were taken from PNL-2987 "Fusion Reactor Design



Studies: Standard Cost Estimating Rules," issued by Pacific Northwest Labs and DOE; the "Unit Cost Documentation" WFPS-TN-057 issued by ORNL with Westinghouse and industrial sources. The sources of the unit costs will be identified as INTOR, PNL/DOE, WFPS and industry.

## XI.2 Costs in Individual Accounts

In the following section, the accounts will be broken down into their constituents and tabulated wherever possible as to the quantities, unit cost, cost source and total cost.

### 1.1 Blanket/Shield

|                    | Material                          | Quantity<br>(10 <sup>3</sup> kg) | Unit Cost<br>\$/kg | Source   | Total<br>\$ x 10 <sup>6</sup> |
|--------------------|-----------------------------------|----------------------------------|--------------------|----------|-------------------------------|
| Vacuum Chamber     | 316 SS                            | 144.47                           | 45                 | INTOR    | 6.50                          |
| Blanket            | HT-9                              | 39.87                            | 45                 | INTOR    | 1.79                          |
| Breeding Material  | Pb <sub>83</sub> Li <sub>17</sub> | 600.                             | 10.8               | PNL/DOE  | 6.48                          |
| Reflector          | 316 SS                            | 273.9                            | 10                 | INTOR    | 2.74                          |
| Shield             | 1442 Steel                        | 1481.2                           | 10                 | INTOR    | 14.81                         |
|                    | B <sub>4</sub> C                  | 191.76                           | 50                 | INTOR    | 9.59                          |
|                    | Pb                                | 538.35                           | 4                  | INTOR    | 2.15                          |
|                    | W                                 | 33.45                            | 60                 | Industry | 2.01                          |
| Neutral Beam Dumps | TZM                               | 1.095                            | 200                | Industry | 0.22                          |
| TOTAL              |                                   |                                  |                    |          | 46.29                         |

### 1.2 Magnets

Here the costs of recent big projects such as LCT and MFTF have been used as the basis for cost estimates.

On this basis the following reference values have been adopted for the TASKA-magnets.

1. Superconductor costs

NbTi cryostable, reinforced 14.5\* \$/kA·m at 6 T

Nb<sub>3</sub>Sn, cryostable, reinforced 34.5 \$/kA·m at 10 T

2. Winding costs

(equipments, insulation, labor) 46 \$/m of wound conductor

3. Coil case 46 \$/kg of final steel case

4. Intercoil structure 13.8 \$/kg of steel

5. Normal conducting coils 55.2 \$/kg of winding

6. Pretesting and special cryogenic engineering 20% of items 1 to 3

Based on these specific values the costs for the several magnets are:

| Magnet                 | CC1-CC3 | C 2  | Barrier   |           | TC-RC | Yin-Yang | FSC5+6+7+8 |
|------------------------|---------|------|-----------|-----------|-------|----------|------------|
|                        |         |      | S.C. Part | N.C. Part |       |          |            |
| Conductor              | 2.67    | 1.98 | 20.29     |           | 3.86  | 4.60     |            |
| Winding                | 0.83    | 0.6  | 4.46      | { 1.10    | 1.06  | 1.24     | { 1.37     |
| Case                   | 1.29    | 1.2  | 4.74      | ---       | 1.93  | 2.71     | ---        |
| Total per coil         | 5.75    | 4.50 | 35.42     | 1.10      | 8.23  | 10.26    | 1.37       |
| Specific costs (\$/kg) | 50.6    | 46   | 127.9     | 55.2      | 65.3  | 58.4     | 55.2       |

Cost breakdown for the different TASKA-magnets (values in \$ x 10<sup>6</sup>)

\* (1 DM = 0.46\$)

Thus the total magnet system costs are:

|                                          | <u>\$ x 10<sup>6</sup></u> |
|------------------------------------------|----------------------------|
| Central Cell                             | 16.01                      |
| Barrier Coils (including normal inserts) | 73.05                      |
| End Plug Coils                           | 73.97                      |
| Field Shaping coils                      | 2.74                       |
| Intercoil/Support Structure              | <u>62.10</u>               |
|                                          | 227.87                     |

### 1.3 Plasma Heating

#### NEUTRAL BEAMS

|                       | Energy<br>(keV) | Power<br>(MW) | Unit Cost<br>\$/W | Source   | Total<br>\$ x 10 <sup>6</sup> |
|-----------------------|-----------------|---------------|-------------------|----------|-------------------------------|
| Low Energy Barrier    | 5               | 0.25          | 1.80              | INTOR    | 0.45                          |
| Medium Energy Barrier | 50              | 49.6          | 1.80              | INTOR    | 89.28                         |
| High Energy Barrier   | 76              | 6.6           | 1.80              | INTOR    | 11.88                         |
| V. High Energy Plug   | 250             | 5.4           | 2.50              | Industry | 13.50                         |

#### RF HEATING

Freq.

|                   |  |      |   |          |        |
|-------------------|--|------|---|----------|--------|
| ECRH (Plug)       |  | 14.9 | 5 | Industry | 74.5   |
| ICRH Central Cell |  | 40   | 2 | Industry | 80.00  |
| TOTAL             |  |      |   |          | 269.61 |

## 2.0 Reactor Support Systems

### 2.1 Electrical

The magnet electrical system and the substation system were taken from INTOR as  $\$25 \times 10^6$ . A detailed estimate will be performed when all the components are identified.

### 2.2 Auxiliary Cooling

| System                                 | Type          | Power                 | Basis     | Total Source | \$ x 10 <sup>6</sup> |
|----------------------------------------|---------------|-----------------------|-----------|--------------|----------------------|
| Magnet and Vacuum System Cooling       | Cryogenerator | 7.5 MW <sub>e</sub>   | ---       | KfK          | 12.33                |
| Ion & Electron Thermal Dump            | Water         | 89 MW <sub>th</sub>   | Algorithm | INTOR        | 1.33                 |
| Neutral Beam Dump                      | Water         | 10.6 MW <sub>th</sub> | Algorithm | INTOR        | 0.24                 |
| Reflector                              | Water         | 5.1 MW <sub>th</sub>  | Algorithm | INTOR        | 0.13                 |
| Shield                                 | Water         | 11.3 MW <sub>th</sub> | Algorithm | INTOR        | 0.25                 |
| ICRH Antennae                          | Water         | 11 MW <sub>th</sub>   | Algorithm | INTOR        | 0.25                 |
| Normal Inserts and Field Shaping Coils | Water         | 39 MW <sub>th</sub>   | Algorithm | INTOR        | 0.35                 |
| Cooling Tower for all Above            | Water/Air     | 174 MW <sub>th</sub>  | Algorithm | INTOR        | 4.43                 |
| TOTAL                                  |               |                       |           |              | 19.31                |

### 2.3 Instrumentation and Control

For the present  $\$25 \times 10^6$  has been used. However, an attempt will be made to estimate I&C better in the future.

## 2.4 Fuel Handling

| Item                   | Capacity<br>g/day        | Costing<br>Basis | Source  | Total<br>\$ x 10 <sup>6</sup> |
|------------------------|--------------------------|------------------|---------|-------------------------------|
| Piping                 | 1500                     | Algorithm        | INTOR   | 1.320                         |
| Isotopic Separation    | 1500                     | Algorithm        | INTOR   | 1.390                         |
| Glove Box              | 1500                     | Algorithm        | INTOR   | 5.790                         |
| Miscellaneous          | 1500                     | Algorithm        | INTOR   | 1.70                          |
| Pellet Fuel            |                          | Algorithm        | INTOR   | 0.13                          |
| Emergency Detritiation | 18,000 m <sup>3</sup> /h | Algorithm        | INTOR   | 6.30                          |
| T <sub>2</sub> Storage | 5000 g                   | \$80,000/unit    | PNL/DOE | 0.080                         |
| TOTAL                  |                          |                  |         | 16.71                         |

## 2.5 Maintenance Equipment

| Item                                             | Source | Total<br>\$ x 10 <sup>6</sup> |
|--------------------------------------------------|--------|-------------------------------|
| Overhead bridge crane                            | KfK    | 1.20                          |
| Component handling machine                       | KfK    | 2.00                          |
| Shielded cabin                                   | KfK    | 0.90                          |
| Manipulator unit                                 | KfK    | 1.00                          |
| 2 work platforms                                 | KfK    | 0.20                          |
| Temporary shielding                              | KfK    | 0.20                          |
| NBI turnover machine                             | KfK    | 0.50                          |
| Blanket carriage/barrier coil support structures | KfK    | 0.40                          |
| Cutting/welding unit                             | KfK    | 0.90                          |
| Auxiliary remote equipment                       | KfK    | 2.50                          |
| 20 hot cell work stations                        | KfK    | 4.00                          |
| Active service area bridge crane                 | KfK    | 0.80                          |
| Overhead manipulator system                      | KfK    | 2.20                          |
| Supplementary tools and devices                  | KfK    | 3.50                          |
| TOTAL                                            |        | 20.30                         |

## 2.6 Primary Heat Transport

| Item                | Material | Capacity<br>Size       | Unit Cost   | Source   | Total<br>\$ x 10 <sup>6</sup> |
|---------------------|----------|------------------------|-------------|----------|-------------------------------|
| Piping              | HT-9     | D = 40 cm<br>L = 50 m  | \$120,000/m | PNL/DOE  | 6.00                          |
|                     | HT-9     | D = 22 cm<br>L = 30 m  | \$65,000/m  | PNL/DOE  | 1.95                          |
|                     | HT-9     | D = 20 cm<br>L = 30 m  | \$60,000/m  | PNL/DOE  | 1.80                          |
| Pump                | ---      | 10 <sup>7</sup> kg/hr  |             | PNL/DOE  | 3.30                          |
| IHX                 | ---      |                        |             | B&W      | 3.26                          |
| Cleanup System      | ---      | 6 x 10 <sup>5</sup> kg | \$2/kg      | PNL/DOE  | 1.20                          |
| Dump Tanks & Piping | Steel    | 6 x 10 <sup>5</sup> kg |             |          | 3.00                          |
| Valves              | Steel    | 60                     | 25,000 ea   | Industry | 1.50                          |
| TOTAL               |          |                        |             |          | 22.01                         |

## 2.7 Secondary Heat Transport

| Item                 | Material | Capacity<br>Size            | Unit Cost | Source  | Total<br>\$ x 10 <sup>6</sup> |
|----------------------|----------|-----------------------------|-----------|---------|-------------------------------|
| Intermediate Coolant | HB-40    | 1.4 x 10 <sup>4</sup> kg    | \$0.8/kg  | *       | .01                           |
| Piping               | Steel    | D = 36 cm<br>L = 100 m      | \$3,500/m | PNL/DOE | 0.35                          |
| Pump                 | Steel    | 6.1 x 10 <sup>5</sup> kg/hr |           | PNL/DOE | 0.73                          |
| Preboiler            | Steel    |                             |           | B&W     | 0.67                          |
| Evaporator           | Steel    |                             |           | B&W     | 1.20                          |
| Superheater          | Steel    |                             |           | B&W     | 0.62                          |
| Steam Drum           | Steel    |                             |           | B&W     | 0.12                          |
| Misc. Piping         | Steel    |                             |           | ---     | 0.50                          |
| Cooling Tower        | ---      | 30 MW                       | Algorithm | INTOR   | 1.29                          |
| TOTAL                |          |                             |           |         | 5.49                          |

\* Whiteshell Nuclear Research Establishment, Canada.

## 2.8 Reactor Vacuum

| System         | Type          | Capacity                  | Cost Basis | Source | Total<br>\$ x 10 <sup>6</sup> |
|----------------|---------------|---------------------------|------------|--------|-------------------------------|
| Helium Pumping | Cryosorption  | 1 x 10 <sup>6</sup> l/s   | Algorithm  | INTOR  | 0.520                         |
| DT Pumping     | Cryocondens.  | 2.4 x 10 <sup>7</sup> l/s | Algorithm  | INTOR  | 7.2                           |
| Regeneration   | Roots Blowers | 10 <sup>5</sup> l/s x 10  | Algorithm  | INTOR  | 0.400                         |
| Valves         | Hard Seal     | D = 1.2 m<br>N = 50       | Algorithm  | INTOR  | 6.95                          |
| Ducting        | Steel         | ---                       | ---        | ---    | 1.00                          |
| TOTAL          |               |                           |            |        | 16.07                         |

## 2.9 Radwaste Treatment

According to PNL/DOE the radwaste treatment equipment costs \$850/MW<sub>th</sub>. For TASKA (~ 50 MW<sub>th</sub>) this is \$42,500.

## 2.10 Thermal dumps

The neutral beam dumps were included in the Blanket/Shield account. The ion and electron thermal dumps must dissipate ~ 90 MW of power. They are constructed of water cooled stainless tubes and are estimated to require ~ 7.35 x 10<sup>3</sup> kg including the manifolding. Using a unit cost of \$45/kg we get \$0.33 x 10<sup>6</sup>.

## 2.11 Direct Convertor Module

The direct convertor module intercepts 1.2% of the ion thermal dump current or ~ 0.65 m<sup>2</sup> in area. We assume the extended area PDC is ~ 1.0 m<sup>2</sup>. According to UCRL-86026, such a single stage PDC costs \$7,500/m<sup>2</sup>. The cryo-surfaces were included in the auxiliary cooling system.



### 2.12 Reactor Support Structure

The intercoil and support structure for the magnets was included in the magnet account. For the remainder of the system we estimate ~ 900 MT at ~ \$15/kg, and get  $\$13.5 \times 10^6$ .

### 3.0 Buildings

| Building                                   | Volume/Area           | Unit Cost            | Source | Total<br>\$ x 10 <sup>6</sup> |
|--------------------------------------------|-----------------------|----------------------|--------|-------------------------------|
| Reactor Hall                               | 78,000 m <sup>3</sup> | Algorithm            | INTOR  | 47.04                         |
| Lock Area & Active Comp. Inter.<br>Storage | 15,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 4.05                          |
| Active Dismant. Maint. & Testing           | 15,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 4.05                          |
| Active Waste Cond. & Temp.<br>Storage      | 15,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 4.05                          |
| Inactive Assem. Testing                    | 15,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 4.05                          |
| Inactive Storage                           | 15,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 4.05                          |
| Vac. & Cryo. Systems                       | 13,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 3.51                          |
| Cooling & DT Reprocess.                    | 25,000 m <sup>3</sup> | \$270/m <sup>3</sup> | INTOR  | 6.75                          |
| Control                                    | 4,500 m <sup>3</sup>  | \$270/m <sup>3</sup> | INTOR  | 1.21                          |
| Administr.                                 | 15,000 m <sup>3</sup> | \$80/m <sup>3</sup>  | INTOR  | 1.20                          |
| Diesel Gen. Bldg.                          | 5,000 m <sup>3</sup>  | \$80/m <sup>3</sup>  | INTOR  | 0.40                          |
| TOTAL                                      |                       |                      |        | 80.36                         |

### XI.3 Summary of Accounts

|                                 | <u>\$ x 10<sup>6</sup></u> |
|---------------------------------|----------------------------|
| 1.1 Blanket/Shield              | 46.29                      |
| 1.2 Magnets                     | 227.87                     |
| 1.3 Plasma Heating              | 269.61                     |
| <u>Support Systems</u>          |                            |
| 2.1 Electrical                  | 25.00                      |
| 2.2 Auxiliary Cooling           | 19.31                      |
| 2.3 Instrumentation and Control | 25.00                      |
| 2.4 Fuel Handling               | 16.71                      |
| 2.5 Maintenance Equipment       | 20.30                      |
| 2.6 Primary Heat Transport      | 22.01                      |
| 2.7 Secondary Heat Transport    | 5.49                       |
| 2.8 Reactor Vacuum              | 16.07                      |
| 2.9 Radwaste Treatment          | 0.04                       |
| 2.10 Thermal Dumps              | 0.33                       |
| 2.11 Direct Convertor Module    | 0.01                       |
| 2.12 Reactor Support Structure  | 13.50                      |
| 3.0 Buildings                   | <u>80.36</u>               |
| Total Direct Costs              | 787.90                     |

In addition to the direct costs, certain percentages for indirect costs such as engineering, assembly and contingency have to be accounted for. As an example, in INTOR these consisted of 45% and 15% of the direct costs for engineering and assembly respectively, and 30% of the direct and indirect costs for contingency.

#### XI.4 Annual Operating Costs

The operating expenses for TASKA consist of the operation and maintenance costs and the power costs. O & M costs were taken as 3% of the direct costs or  $\$24 \times 10^6$ /year. Annual power costs depend on the availability which is taken to be the same as INTOR, namely 10% the first year, 15% the second and third year, 25% for the fourth-seventh years and 50% for the eighth-fifteenth years. Power consumption for TASKA is  $8.7 \text{ MW}_e$  continuously and an additional  $189 \text{ MW}_e$  during operation. We have taken 45 mills/ $\text{kW}_h$  as the cost of electricity. Table XI.4-1 summarizes the operating costs.

The levelized annual operating cost is  $\$54 \times 10^6$ . It is interesting to note that if the normal coils (barrier inserts and field shaping coils) cannot be deenergized during the downtime, the levelized annual operating cost rises to  $\$64 \times 10^6$ .

Table XI.4-1. Summary of Operating Costs

| Year | Avail. (%) | O & M<br>(\$ x $10^6$ ) | Annual Power<br>(\$ x $10^6$ ) | Annual Cost<br>(\$ x $10^6$ ) |
|------|------------|-------------------------|--------------------------------|-------------------------------|
| 1    | 10         | 24.0                    | 10.9                           | 34.9                          |
| 2-3  | 15         | 24.0                    | 14.6                           | 38.6                          |
| 4-7  | 25         | 24.0                    | 22.1                           | 46.1                          |
| 8-15 | 50         | 24.0                    | 40.8                           | 64.8                          |

## TASKA PARAMETER LIST

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| 3. Neutral beams and beam power dumps..... | A-8         |
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## 1. PHYSICS PARAMETERS

### 1.1 General Power Parameters

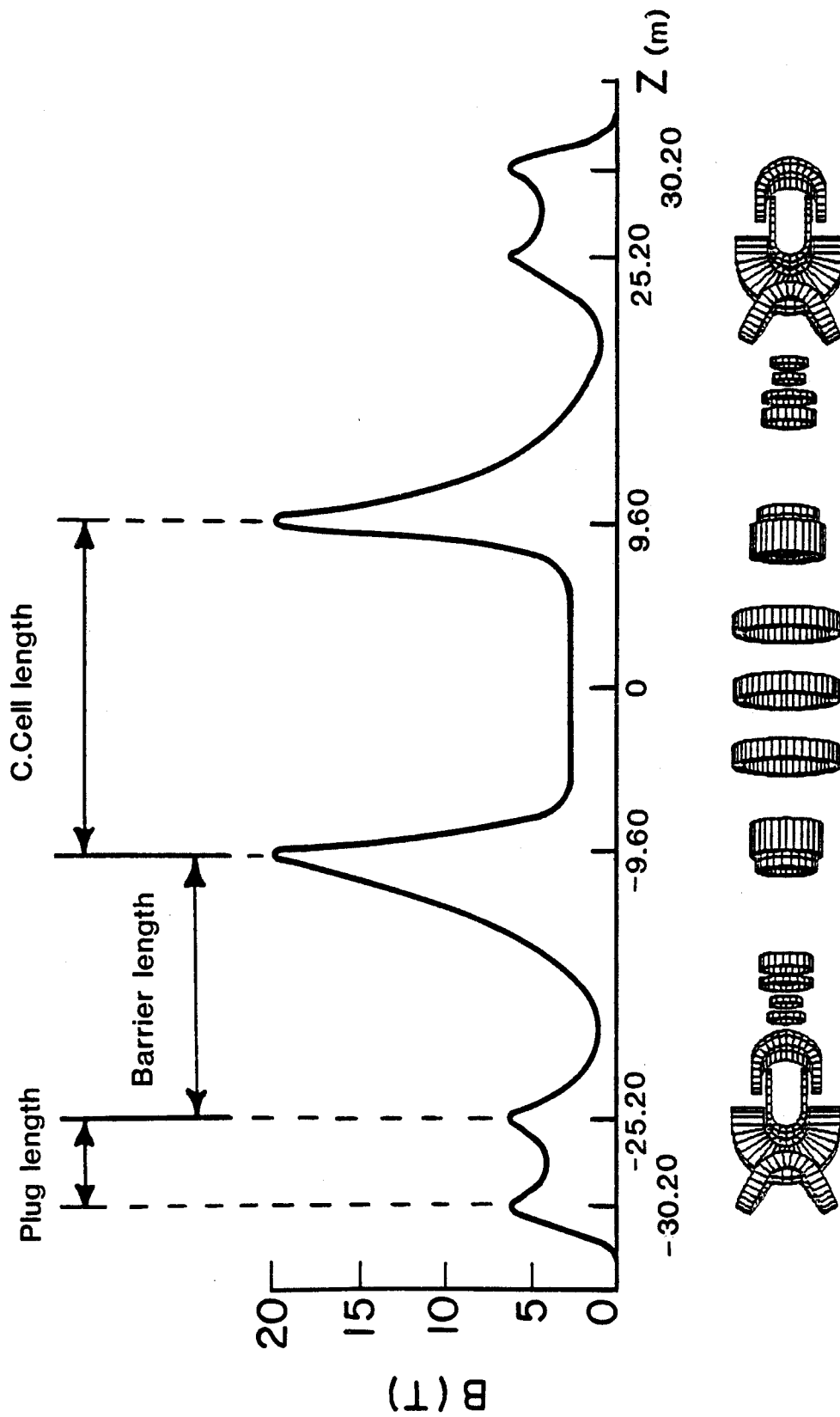
|                                                      |       |
|------------------------------------------------------|-------|
| Fusion Power, MW                                     | 86    |
| Q                                                    | 0.74  |
| Neutron wall loading, MW/m <sup>2</sup>              |       |
| Central cell                                         | 1.52  |
| Barrier                                              | 0.566 |
| Plug (max)                                           | 0.056 |
| Surface heat load, W/cm <sup>2</sup>                 |       |
| Central cell (max)                                   | 5     |
| Barrier (max)                                        | 38    |
| Plug (max)                                           | 44    |
| Central cell plasma power density, MW/m <sup>3</sup> | 17    |

### 1.2 Dimensions

|                                                      |       |
|------------------------------------------------------|-------|
| Central cell                                         |       |
| Plasma radius, m                                     | 0.32  |
| Minimum wall radius, m                               | 0.46  |
| Physics length, m                                    | 19.2  |
| (B field peak to peak -- see diagram)                |       |
| Barrier                                              |       |
| Minimum plasma radius, m                             | 0.099 |
| Average plasma radius, m                             | 0.20  |
| Maximum plasma radius, m                             | 0.50  |
| Physics length, m                                    | 15.6  |
| (barrier B peak to first plug B peak -- see diagram) |       |
| Effective length,* m                                 | 9     |
| Plug                                                 |       |
| Midplane radius, m                                   | 0.28  |
| Physics length, m                                    | 5.0   |
| (B field peak to peak -- see diagram)                |       |

---

\*Region over which barrier potential is less than central cell potential; for purposes of effective volume computation.



Length definitions for plasma physics purposes

Figure A-1

### 1.3 Magnetic Fields on Axis

|                         |      |
|-------------------------|------|
| Central cell average, T | 2.7  |
| Barrier maximum, T      | 20   |
| Barrier minimum, T      | 0.8  |
| Plug maximum, T         | 6.25 |
| Plug minimum, T         | 4    |

### 1.4 Heating

#### Plug neutral beam

|                        |                |
|------------------------|----------------|
| Neutral beam power, MW | 5.4            |
| Neutral species        | H <sup>0</sup> |
| At energy, keV         | 250            |
| At angle               | 60°            |
| At trapping fraction   | 0.21           |

#### Barrier ME neutral beam

|                           |                                            |
|---------------------------|--------------------------------------------|
| ME neutral beam power, MW | 49.7                                       |
| Neutral species           | T <sup>0</sup> (56%), D <sup>0</sup> (44%) |
| At energy, keV            | 50                                         |
| At angle                  | 25°                                        |
| At trapping fraction      | 0.95                                       |

#### Barrier HE neutral beam

|                           |                |
|---------------------------|----------------|
| HE neutral beam power, MW | 6.6            |
| Neutral species           | D <sup>0</sup> |
| At energy, keV            | 76             |
| At angle                  | 20°            |
| At trapping fraction      | 0.42           |

#### Barrier LE neutral beam

|                           |                |
|---------------------------|----------------|
| LE neutral beam power, MW | 0.2            |
| Neutral species           | D <sup>0</sup> |
| At energy, keV            | 2-5            |
| At angle                  | 45°            |
| At trapping fraction      | 0.99           |

#### RF Heating

|                     |      |
|---------------------|------|
| Plug ECRF power, MW | 14.9 |
| At frequency, GHz   | 56   |
| CC ICRF power, MW   | 40   |

|                                    |                      |
|------------------------------------|----------------------|
| At frequency, MHz                  | 30                   |
| <u>1.5 Heating Power Balance</u>   |                      |
| Plug neutral beams                 |                      |
| Total (wall plug), MW              | 6.75                 |
| Injected, MW                       | 5.40                 |
| Absorbed, MW                       | 1.13                 |
| Barrier HE neutral beams           |                      |
| Total (wall plug), MW              | 8.25                 |
| Injected, MW                       | 6.60                 |
| Absorbed, MW                       | 2.77                 |
| Barrier ME neutral beams           |                      |
| Total (wall plug), MW              | 62.13                |
| Injected, MW                       | 49.70                |
| Absorbed, MW                       | 47.22                |
| Barrier LE neutral beams           |                      |
| Total (wall plug), MW              | 0.25                 |
| Injected, MW                       | 0.20                 |
| Absorbed, MW                       | ~ 0.20               |
| ECRH                               |                      |
| Total (wall plug), MW              | 27.60                |
| Injected, MW                       | 14.90                |
| Absorbed, MW                       | 14.90                |
| ICRH                               |                      |
| Total (wall plug), MW              | 44.40                |
| Injected, MW                       | 40.0                 |
| Absorbed, MW                       | 32.0                 |
| Total                              |                      |
| Total (wall plug), MW              | 149.38               |
| Injected, MW                       | 116.8                |
| Absorbed, MW                       | 98.22                |
| <u>1.6 Particle Balance</u>        |                      |
| Axial loss rate, sec <sup>-1</sup> |                      |
| Deuterium                          | 1.8x10 <sup>21</sup> |
| Tritium                            | 1.8x10 <sup>21</sup> |
| Alphas                             | 3.1x10 <sup>19</sup> |



|                                    |                      |
|------------------------------------|----------------------|
| Protons                            | $2.6 \times 10^{19}$ |
| Fueling, $\text{sec}^{-1}$         |                      |
| By high energy NB (100% D)         | $3.2 \times 10^{21}$ |
| By medium energy NB (56% T, 44% D) | $1.6 \times 10^{20}$ |
| By low energy NB (100% D)          | $2.0 \times 10^{20}$ |
| By pellets                         | 0                    |
| By gas puffing                     | 0                    |
| Injected burn fraction,* %         | .87                  |
| Barrier trapping current, A        | 437                  |

### 1.7 Plasma Parameters

|                                               |                      |
|-----------------------------------------------|----------------------|
| Central cell                                  |                      |
| Density, $\text{cm}^{-3}$                     | $1.9 \times 10^{14}$ |
| Ion temperature, keV                          | 30                   |
| Electron temperature, keV                     | 11.5                 |
| Beta                                          | 0.5                  |
| $(n\tau)_{ic}$ , $\text{cm}^{-3} \text{ sec}$ | $5.4 \times 10^{13}$ |
| $(n\tau)_{ec}$ , $\text{cm}^{-3} \text{ sec}$ | $5.3 \times 10^{13}$ |
| Potential, $\phi_c$ , keV                     | 43                   |
| Barrier                                       |                      |
| Minimum density, $\text{cm}^{-3}$             | $6.8 \times 10^{12}$ |
| Passing electron fraction                     | 1                    |
| Beta                                          | 0.054                |
| Potential, $\phi_b$ , keV                     | 38                   |
| Pumping parameter, $g_b$                      | 2                    |
| Plug                                          |                      |
| Density, $\text{cm}^{-3}$                     | $6.3 \times 10^{13}$ |
| Species                                       | p                    |
| Mean ion energy, keV                          | 388                  |
| Electron temperature, keV                     | 59                   |
| Beta                                          | 0.64                 |

---

\*Number of T atoms burnt  $\text{s}^{-1}$  in the plasma divided by the number of T atoms injected  $\text{s}^{-1}$  into the reactor vessel.

$(n\tau)_{ip}, \text{ cm}^{-3} \text{ sec}$   
 $(n\tau)_{ep}, \text{ cm}^{-3} \text{ sec}$   
Potential,  $\phi_c + \phi_e, \text{ keV}$

$2.9 \times 10^{13}$   
 $7.5 \times 10^{11}$   
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## 2. PLASMA HEATING - EM WAVES

### 2.1 Electron Cyclotron Resonance Heating (ECRH)

|                                    |          |
|------------------------------------|----------|
| Location                           | Plug     |
| Total (wall plug) power, MW        | 27.6     |
| Injected power, MW                 | 14.9     |
| Absorbed power, MW                 | 14.9     |
| Frequency, GHz                     | 56       |
| Resonance B-field, T               | 2        |
| Propagation mode                   | Ordinary |
| Injection angle, deg               | 40-50    |
| Overall efficiency, %              | 54       |
| Cyclotron efficiency, %            | 60       |
| Launching structure losses, %      | 10       |
| Absorption efficiency, %           | 100      |
| Gyrotron output power per tube, MW | 0.35     |

### 2.2 Ion Cyclotron Range of Frequencies (ICRF)

|                                |                |
|--------------------------------|----------------|
| Location                       | Central cell   |
| Number of antennas (both ends) | 8              |
| Antenna radius (average), cm   | 46             |
| Antenna material               | Cu-plated HT-9 |
| Faraday shield material        | Mo             |
| Total (wall plug) power, MW    | 44.44          |
| Injected power, MW             | 40             |
| Absorbed power, MW             | 32.0           |
| Frequency, MHz                 | 30             |
| Overall efficiency, %          | 72             |
| Antenna efficiency, %          | 80             |

### 3. NEUTRAL BEAMS AND BEAM POWER DUMPS

(for neutral beam injector power requirements see 1.4 Heating)

#### 3.1 Plug Region

|                                  |                |
|----------------------------------|----------------|
| Neutral species                  | H <sup>0</sup> |
| Beam energy, keV                 | 250            |
| Number of beam lines             | 2              |
| Ion sources/NBI                  | 1              |
| Ion source type                  | Ehlers         |
| Extraction area, cm <sup>2</sup> | ~ 20 x 50      |
| Ion current/NBI, A               | 23             |
| Neutral current/NBI, A           | 10.8           |
| Neutral beam power/NBI, MW       | 2.7            |
| Pulse duration                   | DC             |
| Gas input, Torr ℓ/s              | ~ 20           |
| Gas efficiency,* %               | ~ 5            |

#### 3.2 Barrier Region - Medium Energy Beams

|                                  |                                            |
|----------------------------------|--------------------------------------------|
| Neutral species                  | T <sup>0</sup> (56%), D <sup>0</sup> (44%) |
| Beam energy, keV                 | 50                                         |
| Number of beam lines             | 6                                          |
| Ion sources/NBI                  | 3                                          |
| Ion source type                  | Bucket                                     |
| Extraction area, cm <sup>2</sup> | ~ 2700                                     |
| Ion current/NBI, A               | 121                                        |
| Neutral current/NBI, A           | 55                                         |
| Neutral beam power/NBI, MW       | 8.27                                       |
| Pulse duration                   | DC                                         |
| Gas input, Torr ℓ/s              | 49.5                                       |
| Gas efficiency,* %               | 30                                         |

---

\*Fraction of gas molecules handled which  
are injected as neutral particles

### 3.3 Barrier Region - High Energy Beams

|                                      |           |
|--------------------------------------|-----------|
| Neutral species                      | D°        |
| Beam energy, keV                     | 76        |
| Number of beam lines                 | 2         |
| Ion sources/NBI                      | 1         |
| Ion source type                      | Bucket    |
| Extraction area/NBI, cm <sup>2</sup> | ~ 20 x 45 |
| Ion current/NBI, A                   | 108.8     |
| Neutral current/NBI, A               | 43.4      |
| Neutral beam power/NBI, MW           | 3.3       |
| Pulse duration                       | DC        |
| Gas input Torr $\ell$ /s             | 15.1      |
| Gas efficiency,* %                   | 26        |

### 3.4 Barrier Region - Low Energy Beam

|                                      |           |
|--------------------------------------|-----------|
| Neutral species                      | D°        |
| Beam energy, keV                     | 2 to 5    |
| Number of beam lines                 | 2         |
| Ion sources/NBI                      | 1         |
| Ion source type                      | Bucket    |
| Extraction area/NBI, cm <sup>2</sup> | ~ 20 x 45 |
| Ion current/NBI, A                   | 97.5      |
| Neutral current/NBI, A               | 50.0      |
| Neutral beam power/NBI, MW           | 0.1-0.25  |
| Pulse duration                       | DC        |
| Gas input, Torr $\ell$ /s            | 13.0      |
| Gas efficiency,* %                   | 33%       |

### 3.5 Beam Dumps

|                                                               |               |
|---------------------------------------------------------------|---------------|
| Plug region                                                   |               |
| Dump power density (normal operation), kW cm <sup>-2</sup>    | 1.0           |
| Dump power density (no plasma condition), kW cm <sup>-2</sup> | 1.27          |
| Configuration                                                 | Rectangular-V |
| Material                                                      | Mo            |
| Inclination angle                                             | 76.7°         |
| Surface temperature, °C                                       | 510           |

|                                                               |                      |
|---------------------------------------------------------------|----------------------|
| Coolant                                                       | H <sub>2</sub> O     |
| Barrier region - high energy beams                            |                      |
| Dump power density (normal operation), kW cm <sup>-2</sup>    | 0.58                 |
| Dump power density (no plasma condition), kW cm <sup>-2</sup> | 1.0                  |
| Configuration                                                 | Rectangular-V        |
| Material                                                      | Mo                   |
| Inclination angle                                             | 61.9°                |
| Surface temperature, °C                                       | 495                  |
| Coolant                                                       | H <sub>2</sub> O     |
| Barrier region - medium energy beams                          |                      |
| Dump power density (normal operation), kW cm <sup>-2</sup>    | 0.20                 |
| Dump power density (no plasma condition), kW cm <sup>-2</sup> | 4.0                  |
| Configuration                                                 | Circular-V           |
| Material                                                      | Mo                   |
| Inclination angle                                             | 58.1°                |
| Surface temperature, °C                                       | 277                  |
| Coolant                                                       | H <sub>2</sub> O     |
| Beam-generated DT neutron yield, s <sup>-1</sup>              | 4x10 <sup>10</sup>   |
| Implanted tritium inventory, g                                | 0.50                 |
| Barrier region - low energy beams                             |                      |
| No dump required                                              |                      |
| (Opposing first-wall power density, kW cm <sup>-2</sup>       | 4x10 <sup>-4</sup> ) |

## 4. MAGNETS

### 4.1 Central Cell

|                                                                        |                   |
|------------------------------------------------------------------------|-------------------|
| Coil type                                                              | Solenoid          |
| Number of coils                                                        | 3                 |
| Function                                                               | Central cell      |
| Minor radius, m                                                        | 2.8               |
| Major radius, m                                                        | 3.38/3.3*         |
| Bundle cross section, m x m                                            | 1.2x0.58/1.2x0.5* |
| Mean turn length, m                                                    | 19.17/19.42*      |
| Volume of winding per coil, m <sup>3</sup>                             | 13.5/11.5*        |
| Weight of winding per coil, tonnes                                     | 86/72.5*          |
| Overall current density, A/cm <sup>2</sup>                             | 1350/1160*        |
| Ampere-turns, 10 <sup>6</sup> A-turns                                  | 9.4/6.96*         |
| Self-inductance per N <sup>2</sup> , 10 <sup>-6</sup> H/N <sup>2</sup> | 8.32/8.34*        |
| Stored self-energy, GJ                                                 | 0.37/0.2*         |
| Operating temperature, K                                               | 4.2               |
| Magnetic field on axis, T                                              | 2.7               |
| Maximum magnetic field on conductor, T                                 | < 6               |
| Maximum ripple, %                                                      | < 5               |
| Superconductor                                                         | NbTi              |
| Stabilizing material                                                   | Al                |
| Structure material                                                     | SS                |
| Maximum working strain, %                                              | 0.24              |
| Maximum working stress, MPa                                            | 500               |
| Coil separation (wind/wind), m                                         | 3.0               |
| Coil separation (mid/mid), m                                           | 4.2               |

### 4.2 Barrier Coil

|                 |                             |
|-----------------|-----------------------------|
| Coil type       | Hybrid, nested<br>solenoids |
| Number of coils | 2                           |
| Function        | Mirror                      |

---

\* Outer solenoids / center solenoid

## Dimensions

See Tables  
V.3-1 and  
V.3-2

|                                                        |                         |
|--------------------------------------------------------|-------------------------|
| Weight of winding per superconducting solenoid, tonnes | 200                     |
| Ampere-turns, $10^6$ A-turns                           | 57                      |
| Stored self-energy, GJ                                 | 2.67                    |
| Operating temperature, K                               | 4.2                     |
| Magnetic field on axis, T                              | 14 + 6                  |
| Maximum magnetic field on conductor, T                 | 15 20.7                 |
| Superconductor                                         | Nb <sub>3</sub> Sn/NbTi |
| Stabilizing material                                   | Cu/Al                   |
| Structure material                                     | SS                      |
| Maximum working strain, %                              | 0.24                    |
| Maximum working stress, MPa                            | 500                     |
| Coil separation (wind/wind), m                         | 18                      |
| Coil separation (mid/mid), m                           | 19                      |

## 4.3 Transition Coil

|                                                                 |                                           |
|-----------------------------------------------------------------|-------------------------------------------|
| Coil type                                                       | Reversed-C                                |
| Number of coils                                                 | 4                                         |
| Function                                                        | Change of flux<br>bundle cross<br>section |
| "Minor radius", m                                               | 1.6                                       |
| "Major radius", m                                               | 1.4                                       |
| Sweep angle, degree                                             | 60                                        |
| Bundle cross section, m x m                                     | 1.0 x 0.6                                 |
| Mean turn length, m                                             | 19.9                                      |
| Volume of winding per coil, m <sup>3</sup>                      | 11.94                                     |
| Weight of winding per coil, tonnes                              | 84                                        |
| Overall current density, A/cm <sup>2</sup>                      | 1900                                      |
| Ampere-turns, $10^6$ A-turns                                    | 11.4                                      |
| Self-inductance per N <sup>2</sup> , $10^{-6}$ H/N <sup>2</sup> | 7.5                                       |
| Stored self-energy, MJ                                          | 487                                       |
| Operating temperature, K                                        | 4.2                                       |
| Maximum magnetic field on conductor, T                          | < 8                                       |



|                             |             |
|-----------------------------|-------------|
| Superconductor              | NbTi        |
| Stabilizing material        | 3/4 hard Cu |
| Structure material          | 3/4 hard Cu |
| Maximum working strain, %   | 0.24        |
| Maximum working stress, MPa | 500         |

#### 4.4 Plug Yin-Yang

|                                                                        |                          |
|------------------------------------------------------------------------|--------------------------|
| Coil type                                                              | Yin-Yang                 |
| Number of coils                                                        | 2 + 2                    |
| Function                                                               | Mirror, magnetic<br>well |
| Minor radius, m                                                        | 0.9                      |
| Major radius, m                                                        | 1.4                      |
| Sweep angle, degree                                                    | 90°                      |
| Bundle cross section, m x m                                            | 2.0 x 0.38               |
| Mean turn length, m                                                    | 21.93                    |
| Volume of winding per coil, m <sup>3</sup>                             | 16.67                    |
| Weight of winding per coil, tonnes                                     | 117                      |
| Overall current density, A/cm <sup>2</sup>                             | 1630                     |
| Ampere-turns, 10 <sup>6</sup> A-turns                                  | 12.4                     |
| Self-inductance per N <sup>2</sup> , 10 <sup>-6</sup> H/N <sup>2</sup> | 5.35                     |
| Stored self-energy, MJ                                                 | 411                      |
| Operating temperature, K                                               | 4.2                      |
| Magnetic field on axis, T                                              | 6.25 to 4                |
| Maximum magnetic field on conductor, T                                 | < 8                      |
| Superconductor                                                         | NbTi                     |
| Stabilizing material                                                   | 3/4 hard Cu              |
| Structure material                                                     | 3/4 hard Cu              |
| Maximum working strain, %                                              | 0.24                     |
| Maximum working stress, MPa                                            | 500                      |

## 5. CENTRAL CELL PARAMETERS

### 5.1 Composition

|                             |                                                        |
|-----------------------------|--------------------------------------------------------|
| Blanket structural material | HT-9                                                   |
| Blanket coolant             | $\text{Li}_{17}\text{Pb}_{83}$                         |
| Breeding material           | $\text{Li}_{17}\text{Pb}_{83}$<br>(90% $\text{Li}^6$ ) |
| Blanket composition, v/o    |                                                        |
| Structure                   | 7                                                      |
| Coolant/breeder             | 73                                                     |
| Void                        | 20                                                     |
| Reflector composition, v/o  |                                                        |
| Structure                   | 90 (316 SS)                                            |
| Coolant                     | 10 ( $\text{H}_2\text{O}$ )                            |
| Void                        | 0                                                      |
| Shield material             |                                                        |
| Material 1                  | Fe-1422                                                |
| Composition, v/o            | 56                                                     |
| Material 2                  | $\text{B}_4\text{C}$ (87 v/o)                          |
| Composition, v/o            | 26                                                     |
| Material 3                  | Pb                                                     |
| Composition, v/o            | 14                                                     |
| Coolant                     | $\text{H}_2\text{O}$                                   |
| Composition, v/o            | 4                                                      |

### 5.2 Dimensions

|                                          |      |
|------------------------------------------|------|
| Central cell I.D., m                     | 0.92 |
| Non-testing first wall area $\text{m}^2$ | 25.5 |
| Testing zone area, $\text{m}^2$          | 7.8  |
| First wall thickness, <sup>+</sup> mm    | 3    |

---

<sup>+</sup> Thickness of first wall tubing

|                        |     |
|------------------------|-----|
| Blanket thickness, m   | 1   |
| Reflector thickness, m | .28 |
| Shield thickness, m    | .25 |

### 5.3 Neutronic Summary

(3-D results except where indicated by an asterisk)

|                                                                    |        |
|--------------------------------------------------------------------|--------|
| Fusion power, MW                                                   | 86     |
| 14 MeV neutron wall loading, MW/m <sup>2</sup>                     | 1.52   |
| Blanket multiplication <sup>+</sup>                                | 0.616  |
| Overall tritium breeding ratio                                     | 1.04   |
| Local tritium breeding ratio                                       | 1.65   |
| Power in first wall, MW                                            | 1.052  |
| Power in breeding blanket, MW                                      | 42.42  |
| Power in reflector, MW                                             | 5.08   |
| Power in shield, MW                                                | 11.304 |
| Max. first wall power density,* W/cm <sup>3</sup>                  | 10.4   |
| Max. blanket power density,* W/cm <sup>3</sup>                     | 11.24  |
| Max. reflector power density,* W/cm <sup>3</sup>                   | 0.1064 |
| Peak dose outside biological shield,*<br>during operation, mrem/hr | 1.23   |

### 5.4 Thermal Hydraulics

|                               |     |
|-------------------------------|-----|
| Blanket temperature, °C       |     |
| Coolant inlet                 | 300 |
| Coolant outlet                | 400 |
| Maximum structure temperature | 420 |
| Minimum structure temperature | 300 |
| Reflector temperature, °C     |     |
| Coolant inlet                 | 80  |
| Coolant outlet                | 140 |
| Maximum structure temp.       | 150 |

---

<sup>+</sup> Multiplication for breeding blanket only, defined by (power deposition in blanket)/(total fusion neutron power)

<sup>\*</sup> 1-D results.

|                                           |                      |
|-------------------------------------------|----------------------|
| Minimum structure temp.                   | 80                   |
| Shield temperature, °C                    |                      |
| Coolant inlet                             | 80                   |
| Coolant outlet                            | 140                  |
| Maximum structure temp.                   | 165                  |
| Minimum structure temp.                   | 80                   |
| Blanket coolant conditions                |                      |
| Maximum velocity, cm/s                    | 9.7                  |
| Maximum pressure drop,* MN/m <sup>2</sup> | 1.08                 |
| Maximum pressure, MN/m <sup>2</sup>       | 1.02                 |
| Total coolant flow, kg/hr                 | 1.05x10 <sup>7</sup> |
| Pumping power, kW                         | 500                  |
| Reflector coolant conditions              |                      |
| Maximum velocity, cm/s                    | 100                  |
| Maximum pressure drop, MN/m <sup>2</sup>  | .01                  |
| Maximum pressure, MN/m <sup>2</sup>       | .3                   |
| Total coolant flow, kg/hr                 | 7.2x10 <sup>4</sup>  |
| Pumping power, kW                         | 1                    |
| Shield coolant conditions                 |                      |
| Maximum velocity, cm/s                    | 100                  |
| Maximum pressure drop, MN/m <sup>2</sup>  | .01                  |
| Maximum pressure, MN/m <sup>2</sup>       | .3                   |
| Total coolant flow, kg/hr                 | 1.6x10 <sup>5</sup>  |
| Pumping power, kW                         | 2                    |

---

\*Including head pressure.

## 6. MATERIALS TEST MODULE

### 6.1 Gas Cooled REGAT Module

|                                                 |                  |
|-------------------------------------------------|------------------|
| Width (m)                                       | 0.7              |
| Thickness (m)                                   | 0.2              |
| Area (m <sup>2</sup> )                          | 2.02             |
| Module volume (liters)                          | 493              |
| Capsule volume -- 40% of module volume (liters) | 197              |
| Module structure and v/o                        | Ti-6Al-4V 20 v/o |
| Coolant and v/o                                 | Helium 40 v/o    |
| Specimen and capsule structure<br>and v/o       | 316 SS<br>20 v/o |
| Thermal contact material and v/o                | NaK 20 v/o       |
| Average dpa per MW-yr/m <sup>2</sup>            | 7.3              |
| Cumulative dpa•l (5.3 FPY operation)            | 11416            |
| Average He appm per MW-yr/m <sup>2</sup>        | 49.03            |
| Cumulative He appm•l (5.3 FPY operation)        | 76861            |
| Average power density (W/cm <sup>3</sup> )      | 3.27             |

### 6.2 Water Cooled REGAT Module

|                                            |                         |
|--------------------------------------------|-------------------------|
| Width (m)                                  | 0.5                     |
| Thickness (m)                              | 0.2                     |
| Area (m <sup>2</sup> )                     | 1.45                    |
| Module volume (liters)                     | 352                     |
| Capsule volume (liters)                    | 141                     |
| Module structure and v/o                   | 316 SS 20 v/o           |
| Coolant and v/o                            | H <sub>2</sub> O 15 v/o |
| Module void v/o                            | 25 v/o                  |
| Specimen and capsule structure and v/o     | 316 SS 20 v/o           |
| Thermal contact material and v/o           | NaK 20 v/o              |
| Average dpa per MW-yr/m <sup>2</sup>       | 5.14                    |
| Cumulative dpa•l (5.3 FPY operation)       | 5764                    |
| Average He appm per MW-yr/m <sup>2</sup>   | 44.0                    |
| Cumulative He appm•l (5.3 FPY operation)   | 49322                   |
| Average power density (W/cm <sup>3</sup> ) | 7.36                    |

7. NEUTRONICS  
(Based on 1 MW/m<sup>2</sup> Wall Load)

7.1 General

|                                                    |       |
|----------------------------------------------------|-------|
| Overall Breeding ratio                             |       |
| <sup>6</sup> Li                                    | 0.976 |
| <sup>7</sup> Li                                    | 0.064 |
| Total                                              | 1.04  |
| Energy deposition per 14.1 MeV neutron, MeV/fusion |       |
| Neutron                                            | 8.25  |
| Gamma                                              | 6.72  |
| Total                                              | 14.97 |
| Overall energy multiplication                      | 1.06  |

7.2 First Wall

|                                                    |       |
|----------------------------------------------------|-------|
| Radius, cm                                         | 46    |
| Volume, m <sup>3</sup>                             | 0.1   |
| Thickness, <sup>+</sup> cm                         | 0.3   |
| Av. power density, W/cm <sup>3</sup>               |       |
| Neutron                                            | 3.03  |
| Gamma                                              | 3.8   |
| Total                                              | 6.83  |
| Peak DPA per FPY <sup>*</sup>                      | 14.8  |
| Peak H production <sup>*</sup> , appm/FPY          | 347.6 |
| Peak He production <sup>*</sup> , appm/FPY         | 107.1 |
| Energy deposition per 14.1 MeV neutron, MeV/fusion |       |
| Neutron                                            | 0.095 |
| Gamma                                              | 0.12  |
| Total                                              | 0.215 |

---

<sup>\*</sup> 1-D results

<sup>+</sup> Thickness of first wall tubing

### 7.3 Breeding Blanket

|                                                    |       |
|----------------------------------------------------|-------|
| Radius, cm                                         | 46.3  |
| Volume, m <sup>3</sup>                             | 51.7  |
| Thickness, cm                                      | 100   |
| Av. power density, W/cm <sup>3</sup>               |       |
| Neutron                                            | 0.306 |
| Gamma                                              | 0.234 |
| Total                                              | 0.54  |
| Energy deposition per 14.1 MeV neutron, MeV/fusion |       |
| Neutron                                            | 4.91  |
| Gamma                                              | 3.77  |
| Total                                              | 8.68  |

### 7.4 Reflector

|                                      |        |
|--------------------------------------|--------|
| Radius, cm                           | 155    |
| Volume, m <sup>3</sup>               | 39.5   |
| Thickness, cm                        | 28     |
| Av. power density, W/cm <sup>3</sup> |        |
| Neutron                              | 0.0084 |
| Gamma                                | 0.057  |
| Total                                | 0.065  |

### 7.5 Shield (Central Cell)

|                                                    |       |
|----------------------------------------------------|-------|
| Radius, cm                                         | 183   |
| Volume, m <sup>3</sup>                             | 96.78 |
| Thickness, cm                                      | 25    |
| Av. power density, W/cm <sup>3</sup>               |       |
| Neutron                                            | 0.042 |
| Gamma                                              | 0.035 |
| Total                                              | 0.077 |
| Energy deposition per 14.1 MeV neutron, MeV/fusion |       |
| Neutron                                            | 1.26  |
| Gamma                                              | 1.05  |
| Total                                              | 2.31  |
| Percent of total energy in shield                  | 15%   |

#### 7.6 Barrier Insert Coil Shield

|                       |                                                            |
|-----------------------|------------------------------------------------------------|
| First wall radius, cm | 17                                                         |
| Shield thickness, cm  | 12                                                         |
| Shield composition    | 80 v/o W (90%)<br>10 v/o 316 SS<br>10 v/o H <sub>2</sub> O |

#### 7.7 Barrier Zone Shield

|                      |                                                                                        |
|----------------------|----------------------------------------------------------------------------------------|
| Shield thickness, cm | 20/30                                                                                  |
| Shield composition   | 56 v/o Fe 1422<br>26 v/o B <sub>4</sub> C (87%)<br>14 v/o Pb<br>4 v/o H <sub>2</sub> O |

#### 7.8 Plug Zone Shield

|                      |                                                                                        |
|----------------------|----------------------------------------------------------------------------------------|
| Shield thickness, cm | 60                                                                                     |
| Shield composition   | 56 v/o Fe 1422<br>26 v/o B <sub>4</sub> C (87%)<br>14 v/o Pb<br>4 v/o H <sub>2</sub> O |

#### 7.9 Direct Convertor Zone Shield

|                      |                                                                                        |
|----------------------|----------------------------------------------------------------------------------------|
| Shield thickness, cm | 20                                                                                     |
| Shield composition   | 56 v/o Fe 1422<br>26 v/o B <sub>4</sub> C (87%)<br>14 v/o Pb<br>4 v/o H <sub>2</sub> O |



## 7.10 Magnets<sup>+</sup>

### 7.10.1 Central Cell Coils

|                                       |           |
|---------------------------------------|-----------|
| Coil radius, cm                       | 280       |
| Peak dose in mylar, rad/FPY           | 1.85 E7   |
| Peak dose in epoxy, rad/FPY           | 1.61 E7   |
| Peak DPA in Al per FPY                | 1.05 E-5  |
| Peak power density, W/cm <sup>3</sup> |           |
| Neutron                               | 5.31 E-6  |
| Gamma                                 | 7.77 E-6  |
| Total                                 | 1.308 E-5 |

### 7.10.2 Barrier Coils

|                                       |           |
|---------------------------------------|-----------|
| Wall loading, MW/m <sup>2</sup>       | 0.566     |
| Normal conducting insert coils        |           |
| Coil radius, cm                       | 30        |
| Peak dose in MgO, rad/FPY             | 4.354 E11 |
| Peak power density, W/cm <sup>3</sup> |           |
| Neutron                               | 0.178     |
| Gamma                                 | 0.698     |
| Total                                 | 0.876     |
| Superconducting coils                 |           |
| Coil radius, cm                       | 130       |
| Peak dose in mylar, rad/FPY           | 1.307 E6  |
| Peak dose in epoxy, rad/FPY           | 1.047 E6  |
| Peak DPA in Al, dpa/FPY               | 8.624 E-7 |
| Peak DPA in Cu, dpa/FPY               | 2.935 E-9 |
| Peak power density in LI* coil        |           |
| Neutron, W/cm <sup>3</sup>            | 1.59 E-7  |
| Gamma, W/cm <sup>3</sup>              | 2.14 E-7  |
| Total, W/cm <sup>3</sup>              | 3.73 E-7  |
| Peak power density in LO* coil        |           |
| Neutron, W/cm <sup>3</sup>            | 6.27 E-10 |

---

<sup>+</sup> 1-D results

<sup>\*</sup> For diagram of superconducting coil modules see Chapter V.

|                          |           |
|--------------------------|-----------|
| Gamma, W/cm <sup>3</sup> | 2.359 E-9 |
| Total, W/cm <sup>3</sup> | 2.986 E-9 |

### 7.10.3 Field Shaping Coils

|                                       |          |
|---------------------------------------|----------|
| Max. wall loading, MW/m <sup>2</sup>  | 7.28 E-4 |
| Smallest coil radius, cm              | 100      |
| Peak dose in MgO, rad/FPY             | 3.87 E8  |
| Peak power density, W/cm <sup>3</sup> |          |
| Neutron                               | 2.03 E-4 |
| Gamma                                 | 4.4 E-4  |
| Total                                 | 6.43 E-4 |

### 7.10.4 Transition and Yin-Yang Coils

|                                       |          |
|---------------------------------------|----------|
| Max. wall loading, MW/m <sup>2</sup>  | 0.056    |
| Smallest coil radius, cm              | 90.0     |
| Peak dose in mylar, rad/FPY           | 2.03 E8  |
| Peak dose in epoxy, rad/FPY           | 1.92 E8  |
| Peak DPA in Cu, dpa/FPY               | 8.47 E-5 |
| Peak power density, W/cm <sup>3</sup> |          |
| Neutron                               | 3.69 E-5 |
| Gamma                                 | 4.13 E-5 |
| Total                                 | 7.82 E-5 |

## 8. RADIOACTIVITY

Operating period for radioactivity calculations, FPY 2

### 8.1 Activity at Shutdown ( $Ci/W_{th}$ )

|                      |                       |
|----------------------|-----------------------|
| Central cell (total) | 1.077                 |
| Blanket              | 1.005                 |
| Vacuum boundary      | $1.25 \times 10^{-2}$ |
| Reflector            | $5.84 \times 10^{-2}$ |
| Shield               | $4.41 \times 10^{-4}$ |

### 8.2 Percent Afterheat at Shutdown (%)

|                      |                        |
|----------------------|------------------------|
| Central cell (total) | $7.07 \times 10^{-1}$  |
| Blanket              | $6.36 \times 10^{-1}$  |
| Vacuum boundary      | $1.28 \times 10^{-4}$  |
| Reflector            | $0.575 \times 10^{-4}$ |
| Shield               | $6.57 \times 10^{-6}$  |

### 8.3 BHP at Shutdown (Air) $km^3(air)/kW_{th}$

|                      |                       |
|----------------------|-----------------------|
| Central cell (total) | $9.14 \times 10^1$    |
| Blanket              | $87.9 \times 10^1$    |
| Vacuum boundary      | $7.11 \times 10^{-1}$ |
| Reflector            | 2.79                  |
| Shield               | $2.27 \times 10^{-2}$ |

### 8.4 Dose Outside Shield 1 Day After Shutdown (mrem/hr)

|              |       |
|--------------|-------|
| Central cell | 1.004 |
| Barrier      | < 1.0 |
| Plugs        | < 1.0 |

## 9. TRITIUM SYSTEMS

### 9.1 Fueling

#### Central cell isotope injection rates<sup>\*</sup>

##### Barrier neutral beams

D (atoms/s)  $3.87 \times 10^{21}$

T (atoms/s)  $3.50 \times 10^{21}$

##### Fraction of tritium gas handled

in beam lines but not injected 0.70

#### Plug neutral beams<sup>\*</sup>

H (atoms/s)  $1.3 \times 10^{20}$

### 9.2 Plasma

Injected fractional burnup of T <sup>+</sup> .0087

T burned (atoms/s)  $3.0 \times 10^{19}$

D burned (atoms/s)  $3.0 \times 10^{19}$

### 9.3 Exhaust

#### Particles pumped from reactor

T (atoms/s)  $3.5 \times 10^{21}$

D (atoms/s)  $3.84 \times 10^{21}$

H (atoms/s)  $1.3 \times 10^{20}$

He (atoms/s)  $3.0 \times 10^{19}$

#### Thermal dump

T inventory due to implantation (g) 0.02

Coolant type  $H_2O(l)$

T loss to coolant (Ci/d) (T barrier factor X-100) 0.07

Plate material Stainless steel

---

<sup>\*</sup>Including atoms not trapped in plasma

<sup>+</sup> Tritium burn-rate / tritium injection-rate

|                                     |                     |
|-------------------------------------|---------------------|
| ME beam dump (D/T beam)             |                     |
| T inventory due to implantation (g) | 0.50                |
| Coolant type                        | H <sub>2</sub> O(l) |
| T loss to coolant (Ci/d)            | 0.16                |
| (T barrier factor x 100)            |                     |
| Plate material                      | Mo                  |

#### 9.4 Blanket

##### Breeder

|                                        |                                   |
|----------------------------------------|-----------------------------------|
| Material                               | Li <sub>17</sub> Pb <sub>83</sub> |
| Temperature (°C)                       |                                   |
| High                                   | 400                               |
| Low                                    | 300                               |
| Average                                | 350                               |
| Breeder inventory (kg)                 | 5.7x10 <sup>5</sup>               |
| Tritium                                |                                   |
| Overall breeding ratio                 | 1.0                               |
| Breeding rate (atoms/s)                | 3.0x10 <sup>19</sup>              |
| Concentration (wppm)                   | 0.032                             |
| Pressure at average temperature (torr) | 0.39                              |
| Inventory (g)                          | 18.2                              |
| Extraction scheme                      | In-situ diffusion                 |

##### Blanket coolant

|                              |                                   |
|------------------------------|-----------------------------------|
| Material                     | Li <sub>17</sub> Pb <sub>83</sub> |
| Flow rate (kg/s)             | 2900                              |
| Temperature (°C)             |                                   |
| Coolant in                   | 300                               |
| Coolant out                  | 400                               |
| Coolant inventory (kg)       | 5.7x10 <sup>5</sup>               |
| Tube area (m <sup>2</sup> )  | 1234                              |
| Tube thickness (mm)          | 2.27                              |
| Pressure (torr)              | 0.39                              |
| Tritium concentration (wppm) | 0.032                             |

### 9.5 Heat Exchanger Tubes in Intermediate Loop

|                                          |      |
|------------------------------------------|------|
| Composition                              | HT-9 |
| Thickness (mm)                           | 2.27 |
| Surface area (m <sup>2</sup> )           | 895  |
| Temperatures (°C)                        |      |
| Coolant in                               | 371  |
| Coolant out                              | 271  |
| T permeation to intermediate loop (Ci/d) | 370  |
| (T barrier factor x 150)                 |      |

### 9.6 Tritium Inventory (g)

|                                    |                   |
|------------------------------------|-------------------|
| Fueling and exhaust                |                   |
| Neutral beam pumps (Zr/Al getters) | 259               |
| Thermal and beam dump              | 0.52              |
| End cell and beam dump cryopumps   | 125               |
| (Time cryopumps on-line) (hr.)     | 2                 |
| Cryogenic distillation columns     | 28                |
| Fuel cleanup                       | 47                |
| Blanket                            |                   |
| Breeding material                  | 18.2              |
| Breeder processing                 | 0                 |
| Structural material and piping     | 0.12              |
| Heat exchange cycle                |                   |
| Inventory intermediate loop (g)    | 15.0              |
| Total active tritium inventory, g  | 493               |
| Total active tritium inventory, Ci | $4.9 \times 10^6$ |
| Storage (number of days/g)         | 5000              |

## 10. MATERIALS

### 10.1 Central Cell

#### 10.1.1 "Fixed" Blanket Structure

|                                      |                       |
|--------------------------------------|-----------------------|
| Material                             | HT-9                  |
| Neutron wall load, MW/m <sup>2</sup> | 1.52                  |
| dpa/s (Max)                          | 7.13x10 <sup>-7</sup> |
| dpa/FPY (Max)                        | 22.5                  |
| Peak H production, appm/FPY          | 528.4                 |
| Peak He production, appm/FPY         | 162.8                 |
| Operating temp. range, °C            | 300-420               |
| Number of cycles/y                   | ~ 100                 |

#### 10.1.2 Coolant

|                 |                                   |
|-----------------|-----------------------------------|
| Material        | Pb <sub>83</sub> Li <sub>17</sub> |
| Temp. range, °C | 300-400                           |
| Velocity, m/s   | .097                              |
| Pressure, MPa   | 1.02                              |

#### 10.1.3 Breeder

|                 |                                   |
|-----------------|-----------------------------------|
| Material        | Pb <sub>83</sub> Li <sub>17</sub> |
| Temp. range, °C | 300-400                           |
| Velocity, m/s   | .097                              |
| Pressure, MPa   | 1.02                              |

#### 10.1.4 Reflector

|                              |                  |
|------------------------------|------------------|
| Material                     | HT-9             |
| Max. dpa/FPY                 | 0.065            |
| Peak H production, appm/FPY  | 0.014            |
| Peak He production, appm/FPY | 0.003            |
| Coolant                      | H <sub>2</sub> O |
| Operating temp., °C          | 80-140           |

#### 10.1.5 Shield

|           |                        |
|-----------|------------------------|
| Materials | SS-B <sub>4</sub> C-Pb |
| Coolant   | H <sub>2</sub> O       |

|                              |          |
|------------------------------|----------|
| Operating temp., °C          | 80-140   |
| Max. dpa/FPY                 | 4.92E-03 |
| Max. He/FPY                  | 8.64 E-4 |
| Peak H production, appm/FPY  | 1.5 E-4  |
| Peak He production, appm/FPY | 3.8 E-5  |

#### 10.1.6 Magnets

##### Materials

|                              |                      |
|------------------------------|----------------------|
| Superconductor               | NbTi                 |
| Stabilizer                   | Al                   |
| Electrical insulation        | G-10                 |
| Thermal insulation           | Mylar                |
| Structure                    | SS                   |
| Dewar                        | SS                   |
| Max. dpa/FPY in stabilizer   | $1.6 \times 10^{-5}$ |
| Peak dose in mylar, rads/FPY | $2.81 \times 10^7$   |
| Peak dose in epoxy, rads/FPY | $2.45 \times 10^7$   |

#### 10.2 End Plug Region

##### 10.2.1 Barrier Coil

##### Shield

|                              |                        |
|------------------------------|------------------------|
| Insert shield                | W-SS                   |
| Zone shield                  | Fe-B <sub>4</sub> C-Pb |
| Heat flux, W/cm <sup>2</sup> | 38 (max)               |
| Coolant                      | H <sub>2</sub> O       |
| Temperature, °C              | 80-140                 |

##### Magnet

##### Superconducting solenoid

|                         |                         |
|-------------------------|-------------------------|
| Superconductor          | Nb <sub>3</sub> Sn/NbTi |
| Stabilizer              | Al/Cu                   |
| Electrical insulator    | G-10                    |
| Thermal insulator       | Mylar                   |
| Structure               | SS                      |
| Dewar                   | SS                      |
| Peak DPA in Al, dpa/FPY | $8.62 \times 10^{-7}$   |
| Peak DPA in Cu, dpa/FPY | $2.94 \times 10^{-9}$   |



|                             |                    |
|-----------------------------|--------------------|
| Peak dose in mylar, rad/FPY | $1.31 \times 10^6$ |
| Peak dose in epoxy, rad/FPY | $1.05 \times 10^6$ |
| Normal conducting insert    |                    |
| Conductor                   | OFHC (Cu)          |
| Insulator                   | MgO (compacted)    |
| Coolant                     | H <sub>2</sub> O   |
| Peak dose in MgO, rad/FPY   | $3.87 \times 10^8$ |

#### 10.2.2 Field Shaping Coils

|                           |                    |
|---------------------------|--------------------|
| Conductor                 | OFHC (Cu)          |
| Insulator                 | MgO (compacted)    |
| Coolant                   | H <sub>2</sub> O   |
| Peak dose in MgO, rad/FPY | $3.87 \times 10^8$ |

#### 10.2.3 Transition, Recirculating and Yin-Yang Coils

##### Shield

|                              |                        |
|------------------------------|------------------------|
| Material                     | Fe-B <sub>4</sub> C-Pb |
| Heat flux, W/cm <sup>2</sup> | 44 (max)               |
| Coolant                      | H <sub>2</sub> O       |
| Temperature, °C              | 80-140                 |

##### Magnet

|                             |                       |
|-----------------------------|-----------------------|
| Superconductor              | NbTi                  |
| Stabilizer                  | Cu                    |
| Electric insulator          | G-10                  |
| Thermal insulator           | Mylar                 |
| Structure                   | Cu/SS                 |
| Dewar                       | SS                    |
| Max. DPA in Cu, dpa/FPY     | $8.47 \times 10^{-5}$ |
| Peak dose in mylar, rad/FPY | $2.03 \times 10^8$    |
| Peak dose in epoxy, rad/FPY | $1.92 \times 10^8$    |

## 11. DIRECT CONVERTOR AND POWER DUMP

### 11.1 Direct Convertor - Grids

|                                                |                      |
|------------------------------------------------|----------------------|
| Material                                       | 316 SS               |
| Particle flux, $\text{cm}^{-2}, \text{s}^{-1}$ |                      |
| D                                              | $3.2 \times 10^{15}$ |
| T                                              | $3.2 \times 10^{15}$ |
| p                                              | $2.4 \times 10^{13}$ |
| He (low energy)                                | $3.6 \times 10^{13}$ |
| He (medium energy)                             | $8.9 \times 10^{12}$ |
| He (high energy)                               | $1.4 \times 10^{12}$ |
| Average particle energy, keV                   |                      |
| D                                              | 139                  |
| T                                              | 139                  |
| p                                              | 307                  |
| He (low energy)                                | 248                  |
| He (medium energy)                             | 808                  |
| He (high energy)                               | 3,630                |
| Heat flux, $\text{W}/\text{cm}^2$              | 150                  |
| Max temperature, $^{\circ}\text{C}$            | 546                  |
| Coolant                                        | $\text{H}_2\text{O}$ |
| Coolant velocity, m/s                          | 2.5                  |
| Coolant pressure, MPa                          | 1.0                  |
| Coolant temp., $^{\circ}\text{C}$              | 60-110               |

### 11.2 Direct Convertor - Collector Plate

|                                               |                      |
|-----------------------------------------------|----------------------|
| Material                                      | 316 SS               |
| Particle flux, $\text{cm}^{-2} \text{s}^{-1}$ |                      |
| D                                             | $3.2 \times 10^{15}$ |
| T                                             | $3.2 \times 10^{15}$ |
| p                                             | $2.4 \times 10^{13}$ |
| He (low energy)                               | $3.6 \times 10^{13}$ |
| He (medium energy)                            | $8.9 \times 10^{12}$ |
| He (high energy)                              | $1.4 \times 10^{12}$ |
| Average particle energy, keV                  |                      |
| D                                             | 39                   |

|                              |                  |
|------------------------------|------------------|
| T                            | 39               |
| p                            | 207              |
| He (low energy)              | 48               |
| He (medium energy)           | 608              |
| He (high energy)             | 3,430            |
| Heat flux, W/cm <sup>2</sup> | 34               |
| Maximum temperature          | 270              |
| Coolant                      | H <sub>2</sub> O |
| Coolant velocity, m/s        | 1.5              |
| Coolant pressure, MPa        | 1.0              |
| Coolant temperature, °C      | 60-100           |

### 11.3 Electron Power Dump

|                                                 |                      |
|-------------------------------------------------|----------------------|
| Material                                        | 316 SS               |
| Particle flux, cm <sup>-2</sup> s <sup>-1</sup> |                      |
| p                                               | 2.2x10 <sup>14</sup> |
| He (medium energy)                              | 8x10 <sup>13</sup>   |
| He (high energy)                                | 1.2x10 <sup>13</sup> |
| e <sup>-</sup>                                  | 5.8x10 <sup>16</sup> |
| Average particle energy, keV                    |                      |
| p                                               | 307                  |
| He (medium energy)                              | 808                  |
| He (high energy)                                | 3,630                |
| e <sup>-</sup>                                  | 11.5                 |
| Electron heat flux, W/cm <sup>2</sup>           | 121                  |
| Non-electron heat flux, W/cm <sup>2</sup>       | 29                   |
| Coolant                                         | H <sub>2</sub> O     |
| Coolant temp., °C                               | 60-110°C             |
| Coolant pressure, MPa                           | 1.0                  |
| Coolant velocity, m/s                           | 1.5                  |

12. REMOTE REPLACEMENT CHARACTERISTICS  
("Average lifetime" is near time between failures)

12.1 Vacuum Vessel (central cell)

|                                        |                       |
|----------------------------------------|-----------------------|
| Max. dimension, m                      | 5φx3                  |
| Max. weight, tonne                     | 30                    |
| Ave. lifetime, y                       | 15                    |
| Lifetime, min-max, y                   | 5-20                  |
| Max. activity 1d after shutdown, Ci/cc | 4.99x10 <sup>-2</sup> |

12.2 Test Blanket + Breeding Blanket

|                                        |           |
|----------------------------------------|-----------|
| Max. dimension, m                      | 1.7x3.2x1 |
| Max. weight, tonne                     | 30        |
| Ave. lifetime, y                       | 2         |
| Lifetime, min-max, y                   | 0.67-6    |
| Max. activity 1d after shutdown, Ci/cc | 0.239     |

12.3 Shield (central cell)

|                                            |            |
|--------------------------------------------|------------|
| Max. dimension, m                          | 3x1.5x0.84 |
| Max. weight, tonne                         | 30         |
| Ave. lifetime, y                           | 15         |
| Lifetime, min-max, y                       | 5-20       |
| Max. dose 1d after shutdown, mr/hr contact | 0.106      |

12.4 Central Cell Coils

|                                        |                    |
|----------------------------------------|--------------------|
| Max. dimension, m                      | 7.0x1.2            |
| Max. weight, tonne                     | 150                |
| Ave. lifetime, y                       | 3                  |
| Lifetime, min-max, y                   | 2-40               |
| Max. activity 1d after shutdown, Ci/cc | < 10 <sup>-5</sup> |

12.5 Barrier Coil

|                      |      |
|----------------------|------|
| Max. dimension, m    | 3.5  |
| Max. weight, tonne   | 350  |
| Ave. lifetime, y     | 3    |
| Lifetime, min-max, y | 2-40 |

|                                        |             |
|----------------------------------------|-------------|
| Max. activity 1d after shutdown, Ci/cc | $< 10^{-5}$ |
|----------------------------------------|-------------|

#### 12.8 Transition Coil

|                                                |             |
|------------------------------------------------|-------------|
| Max. dimension, m                              | 7x5         |
| Max. weight, tonne                             | 200         |
| Ave. lifetime, y                               | 3           |
| Lifetime, min-max, y                           | 2-40        |
| Max. activity 1d after shutdown, Ci/cc contact | $< 10^{-5}$ |

#### 12.9 Yin-Yang Coil

|                                            |             |
|--------------------------------------------|-------------|
| Max. dimension, m                          | 7.5x3       |
| Max. weight, tonne                         | 200         |
| Ave. lifetime, y                           | 3           |
| Lifetime, min-max, y                       | 2-40        |
| Max. activity 1/10 d after shutdown, Ci/cc | $< 10^{-5}$ |

#### 12.10 Neutral Beam Injector

|                      |        |
|----------------------|--------|
| Ave. lifetime, y     | 0.5    |
| Lifetime, min-max, y | 0.08-2 |

#### 12.11 ECRH Units

|                      |        |
|----------------------|--------|
| Ave. lifetime, y     | 0.5    |
| Lifetime, min-max, y | 0.11-5 |

#### 12.12 Direct Converter

|                      |       |
|----------------------|-------|
| Ave. lifetime, y     | 0.5   |
| Lifetime, min-max, y | 0.5-5 |

#### 12.13 Vacuum Equipment

|                      |     |
|----------------------|-----|
| Ave. lifetime, y     | 2   |
| Lifetime, min-max, y | 2-5 |

#### 12.14 Vacuum First Wall - End Cell

|                  |    |
|------------------|----|
| Ave. lifetime, y | 15 |
|------------------|----|

12.15 Auxiliary Equipment

|                                    |   |
|------------------------------------|---|
| Instrumentation - ave. lifetime, y | 5 |
| Heat exchangers - ave. lifetime, y | 2 |

### 13. POWER CONVERSION

|                                              |        |
|----------------------------------------------|--------|
| Blanket thermal power, MW(th)                | 42.42  |
| Generation cycle efficiency, %               | 31.0   |
| Net electrical output, MW(e)                 | 13.15  |
| IHX surface area, m <sup>2</sup>             | 639    |
| Intermediate coolant                         | HB-40  |
| Coolant temperature, °C                      | 271/37 |
| Coolant flow rate, kg/sec                    | 170    |
| Steam generator surface area, m <sup>2</sup> | 537    |
| Steam temperature, °C                        | 343    |
| Steam pressure, MPa                          | 6.6    |
| Water flow rate, kg/sec                      | 7.2    |