JAERI-M
8711

IAEA INTOR WORKSHOP REPORT, GROUPS 2,5,7,9,10 AND 15

—FIRST WALL/BLANKET/SHIELD, MAGNETICS, SYSTEMS INTEGRATION AND SUPPORT STRUCTURES, ASSEMBLY AND REMOTE MAINTENANCE, RADIATION SHIELDING AND PERSONNEL ACCESSIBILITY, SAFETY AND ENVIRONMENT—

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Fusion Research and Development Center

日本原子力研究所
Japan Atomic Energy Research Institute

IAEA INTOR Workshop Report, Groups 2, 5, 7, 9, 10 and 15

Fusion Research and Development Center,
Tokai Research Establishment, JAERI

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In order to prove scientific feasibility of magnetic confinement fusion, large fusion devices are under construction in several countries (JT-60 in Japan, T-15 in U.S.S.R., TFTR in U.S.A. and JET in EC). International Tokamak Reactor (INTOR) Workshop was organized by the International Atomic Energy Agency (IAEA) to identify roles, objectives and characteristics of the next generation fusion device.

This report is a compilation of the home task reports of six groups on INTOR engineering aspects by Japan Atomic Energy Research Institute for workshop sessions 2 and 3 held in 1979. Tasks of the respective groups are

group 2: first wall/blanket/shield,
group 5: magnetics,
group 7: systems integration and structure,
group 9: assembly and remote maintenance,
group 10: radiation shielding and personnel access,
group 15: safety and environment.

Keywords; INTOR Tokamak Reactor, First Wall, Blanket, Remote Control, Radiation Shielding, Safety, Environment, Magnetics, System Integration

IAEA INTOR ワークショップ検討報告書・
グループ2，5，7，9，10および15

日本原子力研究所東海研究所
核融合研究開発推進センター

（1980年1月24日受理）

磁気閉じ込め核融合を科学的に実証するために，現在，世界各国で大型核融合装置が建設中である。（日本のJT - 60，ソ連のT - 15，米国のTFTR，欧州のJET）。
これらの装置の次の世代の装置の果たすべき役割とその性格を明らかにする目的で，IAEAによって国際トカマク炉（INTOR）ワークショップが開催された。

この報告書は，原研において1979年に作成されたINTORの工学的分野の6グループの報告書をまとめたものである。各グループの分担項目は以下の通りである。

第2グループ　第1壁，ブランケット，しゃへい
第5グループ　マグネットおよび関連設計
第7グループ　システム総括および構造設計
第9グループ　組立および遠隔操作
第10グループ　放射線しゃへいと作業者近接評価
第15グループ　環境安全性

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1. Introduction

This report compiles the home task reports of six groups related to engineering aspects of the International Tokamak Reactor (INTOR) prepared in the Japan Atomic Energy Research Institute (JAERI)*. These reports are presented to the IAEA INTOR Workshop Sessions 2 and 3 held in June and October, 1979 respectively. The field of responsibility of each of the six groups are as follows:

Group 2 First Wall/Blanket/Shield
Group 5 Magnetics
Group 7 Systems Integration and Structure
Group 9 Assembly and Remote Maintenance
Group 10 Radiation Shielding and Personnel Accessibility
Group 15 Safety and Environment

The reports of the other 10 groups are published individually as JAERI-M reports.

In the following sections, the home task reports of the above six groups are compiled consecutively. Due to the limited time and man power available for the preparation of these reports on schedule, some of the reports are very much incomplete. For more detailed information about the requested home tasks, the minutes and documents produced in the Sessions 1 and 2 of the INTOR Workshop should be referred. Also the following four reports produced in JAERI before each of the four Sessions of the INTOR Workshop supplement the data base requested as the home tasks. These reports were produced based on the judgement that a systematic reactor design is the best way to assess the INTOR design data base, whereas isolated surveys for individual components tend to lack balance among them.

(1) Second Preliminary Design of JAERI Experimental Fusion Reactor (JXFR), JAERI-M 8286, June (1979)
(2) Engineering Aspects of the JAERI Proposal for INTOR (I), JAERI-M 8503, October (1979)
(3) Engineering Aspects of the JAERI Proposal for INTOR (II), JAERI-M 8518, November (1979)
(4) Evaluation of INTOR Reactor Size, JAERI-M 8710, January (1980)

* These reports were produced mainly by the members of Fusion Reactor System Laboratory, JAERI with the cooperations of the industries. As for the reports of Group 5 (Magnetics), the contribution of Drs. S. Shimamoto and H. Tsuji of Superconducting Magnet Laboratory was received.

2. Group 2 First Wall/Blanket/Shield*

The group 2 home task report was assigned to conform to the following format.

Prososed Table of Contents of Report for June Workshop

I. Introduction

(to be written at the Workshop)

II. INTOR Parameters

(parameters relevant to first wall/blanket/shield as defined by Steering Committee)

III. Design Options for INTOR

(alternative options including tritium breeding, electricity generation, divertor, location of vacuum wall)

* Report for Session 2 of IAEA INTOR Workshop

(Edited by H.Iida)

IV. Review of Previous Design Concepts

Preliminary design of JXFR finished on March, 1979. In this design protection wall is placed in front of blanket cell to suppress temperature change in first wall ("first wall" means the closest part of blanket cell to plasma).

The blanket structure consists of about 2,300 blanket cells with round cornered rectangular cross sections and is placed in a vacuum vessel. The blanket vessel contains Li_2O pebbles and blocks for tritium breeding.

The main structure of the shielding vessel is formed by the stainless steel panels and is filled up with heavy concrete (stainless steel-heavy concrete). About 10 v% of water channels are buried in the heavy concrete to remove the nuclear heat and to obtain the required shielding characteristics.

To form the vacuum barrier, stainless steel lining is fixed to the inside (blanket side) of the stainless steel-heavy concrete structure. Part of this lining consists of bellows to acquire the electrical resistivity (more than 0.5 mΩ) along the torus. Ceramic breaks are also inserted within the stainless steel-heavy concrete structure at the position corresponding to the bellows. Seal welding at the flange of each one-eighth sector is employed for vacuum seal.

PETF (Plasma Engineering Test Facility) is now being studied.

The PETF's plasma chamber consists of the vacuum vessel, the poloidal magnetic limiter, the low Z liner if necessary, and the toroidal material limiter.

The vacuum vessel has the double-walled structure of stainless steel cooled by helium gas and has one-turn electric resistance of 0.5 mΩ. It can be also baked up by flowing hot helium gas. It is supported by the primary shield at its inner, outer, top and bottom positions to make the thermal displacement of its center minimum. The suports have sliding mechanism to reduce the tangential thermal expansion.

The front surface (plasma side) of the chamber will be covered by low Z liner such as carbon to control plasma impurities.

The magnetic limiter plates are made of low Z material from the same reason as liner. The magnetic limiter coils are installed outside the vessel for easy assembly and disassembly.

The primary shield is installed between the vacuum vessel and the toroidal field coils. It stands on its own legs, and supports the vacuum vessel at four positions. It consists of the stainless steel layers cooled by borated water, the heavy concrete and lead if necessary. Its thickness is 60 cm in the inner part and 90 cm in the outer, upper and lower parts.

V. Design Concepts and Data Base for First Wall

A. Previous Studies

Table 2.1 shows the comparative first wall concepts of JXFR,JXFR upgrade and PETF. JXFR upgrade is one of the alternative ideas when a higher beta is available (beta in JXFR design is 2.5%). Heat removal from the first wall, limiter plate and liner will be a crucial problem of a first wall system design of Tokamak rector with high wall loading.

B. First wall coating, including fablication and stability

(H. Nakamura)

1. Objectives of coating the first wall

1 Attain a clean surface of the first wall.

2 Construct first wall surface with desirable material to plasma.

3 Compensate the loss of the first wall caused by erosion.

The objective 1 aims at reducing the plasma contamination by covering by in-situ coating the light impurities adsorbed on the first wall. It also aims at controlling the recycle of the light impurities. In present day tokamak experiments, the impurity influx is suppressed by Ti-gettering and the improvement of the plasma parameters were achieved.

The objective 2 is to coat desirable 1st wall material such as low Z material on the base structure. Pre-coating and in-situ coating are feasible for this accomplishment.

The objective 3 is to compensate by in-situ coating the erosion of the first wall due to sputtering, blistering and arcing.

2. Material

Titanium was used as coating material to aim at gettering of first wall before. However, recently many studies of low z materials have been carried out because of their favorable property as first wall material. Candidates as low Z coating material are compounds of TiN, TiB_2, TiC and B_4C as well as C or B.

Table 2.1 Comparative First Wall Concepts of JXFR, JXFR upgrade and PETF

	Operation Parameter			Compositions	Heat removal	Vacuum tight?	One-turn resistance
	Neutron Wall Loading	Pulse Length	Heat Flux at Full Power (Nucl. Heating)				
JXFR	0.17 (MW/m^2)	100 (sec)	11 W/cm^2 (2 W/cc)	Protection wall (TZM with carbon coating) Blanket cell (S.S.)	He 10 ata. Inlet Temp. 200°C Outlet Temp. 500°C (blanket)	No	-
JXFR UPGRADE	0.77	120	20 W/cm^2 (8 W/cc)	Protection wall (TZM with carbon coating) Cooling tube panel (S.S.) Blanket cell	He 20 ata. Inlet Temp. 200°C Outlet Temp. 500°C	No	-
PETF	0.5	30∼60	limiter plate 1 kW/cm^2 liner 5 W/cm^2 (7 W/cc)	Double-walled vacuum vessel (S.S.) Poloidal magnetic limiter (C) Toloidal material limiter (low Z material)	He Gas	Yes	0.5 mΩ

3. Coating method

o pre-coating
- . chemical vapor deposition
- . physical vapor deposition
- . electroplating
- . plasma spraying
- . cladding

o in-situ coating
- . vacuum deposition using a filament
- . sputter method by coaxial magnetron

4. Coating thickness required

Erosion of a first wall by fuel ion sputtering is estimated.

$$d = S_p \cdot \frac{nV}{\tau_p S} \cdot \left(\frac{A}{N \cdot \rho}\right) t \cdot D \cdot Av$$

d; depth of erosion (cm)
Sp; sputtering yield
n ; plasma density (cm^{-3})
V ; plasma volume (cm^3)
s ; surface area (cm^2)
τ_p; particle confinement time (sec)
A ; atomic weight of wall material
N ; Avogadro's number
ρ ; density of wall material (g/cm^3)
t ; time of irradiation (sec)
D ; duty factor
Av; availability

where,

τ_p = 2 sec, n = 1.2×10^{14} cm^{-3}, V = 200×10^6 cm^3
D = 0.5, Av = 0.25, N = 6×10^{23}, t = 1 year

$$\phi = \frac{nV}{\tau_p} = 1.2 \times 10^{22} \ /sec$$

$$\Gamma = \phi\ t\ D\ A_V = 4.7 \times 10^{28}$$

$$d = 4.7 \times 10^{28} \frac{S_p}{S} \left(\frac{A}{N\rho} \right)$$

$$= 7.8 \times 10^{4} \frac{S_p}{S} \left(\frac{A}{\rho} \right) \quad \text{(cm)}$$

Sps for molybdenum and carbon are as follows.

$$S_p = 10^{-4} \sim 10^{-3} \qquad A/\rho = 95.9/10.2 = 9.4 \text{ (for Mo)}$$

$$10^{-2} \sim 10^{-1} \qquad A/\rho = 12/2 = 6.0 \qquad \text{(for C)}$$

$$S = 300 \times 10^{-4}\ cm^2 \qquad \text{(for first wall)}$$

$$(5 \sim 20) \times 10^{4}\ cm^2 \quad \text{(for magnetic limiter plate)}$$

The loss caused by erosion in a year are estimated as follows

o	Mo	first wall	$d = 2.4 \times (10^{-4} \sim 10^{-5})$ cm
		M.L.P.	$d = 1.5 \times 10^{-2} \sim 3.7 \times 10^{-4}$ cm
o	C	first wall	$d = 1.6 \times (10^{-2} \sim 10^{-3})$ cm
		M.L.P.	$d = 0.94 \sim 0.02$ cm

5. In-situ coating of molybdenum by coaxial magnetron sputter method (1)

The major production mechanisms of light impurities are the particle-induced desorption and/or the thermal desorption of water and methane, etc., which are formed by the chemical reaction with incident hydrogen and surface impurities. Hence, attaining a clean surface of the first wall is a simple and useful countermeasure. In-situ coating is one of the most powerful method to attain a clean surface.

Vacuum deposition using a molybdenum filament has been investigated. The weakpoint of this method is that the coating layer will exfoliate when its thickness becomes over than 10 μm. In order to form more fine grained layer, in-situ coating by sputter method is suitable because of its energetic sputtered atoms.

Figure 2.1 shows a schematic drawing of the sputter device for molybdenum coating. Since a low pressure atomsphere is necessary to form a fine grained layer, RF discharge is more suitable than DC discharge as shown in Figs.2.2 and 2.3. Deposition rate is sufficiently large as shown in Fig. 2.4, which is 600 Å/min near the target. If the distance between the first wall surface and the target is 1-5 m, it will takes about 10 minutes to coat the first wall surface by 100 μm thickness.

The characteristics of the coating layer such as dependence of coating structure on substrata temperature and argon pressure, mechanical strength, resistance against evaporation and repeating heat load, etc., was investigated. The following results were obtained.

Mechanical strength of coating depends on the substrata temperature, but hardly depends on the deposition rate. The coating which had been deposited at 200 Å/min on the substratum of 500°C was very strong and did not exfoliate in spite of 126 times of thermal cycle (50°C ∿ 500°C) when its depth was under 45 μm. If substratum temperature is above 300°C caoting with the depth of more than 10 μm exfoliates without external force. The deposition rate is 3.8 Å/min for molybdenum, 2.8 Å/min for Titanium and 0.28 Å/min for carbon when input electric power is 0.5 KW and pressure of Ar gas 2.7×10^{-2} Pa.

D. Steady State Heat Flux Limitations

The heat flux limitations should be determined as follows.

a. Water : Burn out heat flux

b. Gas : Depends upon a structure

c. Radiation : Allowable temperature for the material and emissivity

F. Leak Detection by Remote Handling

Leak detections by remote handling are divided into two groups ; during reactor operation and shutdown.

The access to the reactor may be possible during reactor shutdown, while impossible during operation.

1. Leak detection during operation.
 a. Acoustic Emission method.
 The following items will be possible,
 (1) Find a small crack and its propagation.
 (2) Forecast the failure.
 b. Analysis of exhausted helium in vacuum pumps.

2. Leak detection during shutdown.
 a. Testing by pressurizing blanket.
 b. Acoustic emission method.
 c. Testing by ultrasonics.
 d. Testing by eddy current.

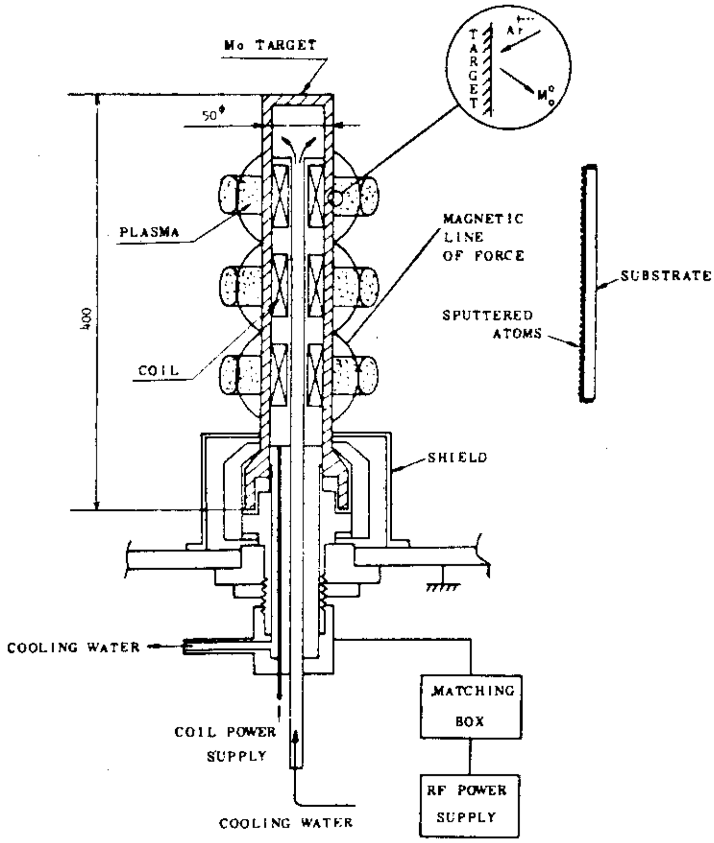


Fig.2.1 Schematic drawing of the sputter device for molybdenum plating.

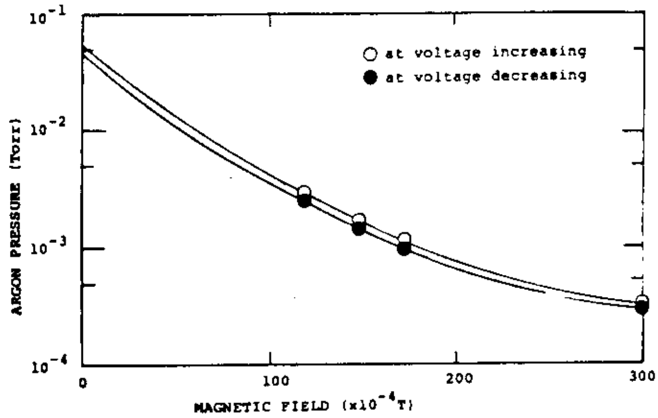


Fig.2.2 Arc burning relation between gas pressure and magnetic field at DC target voltage of 500V.

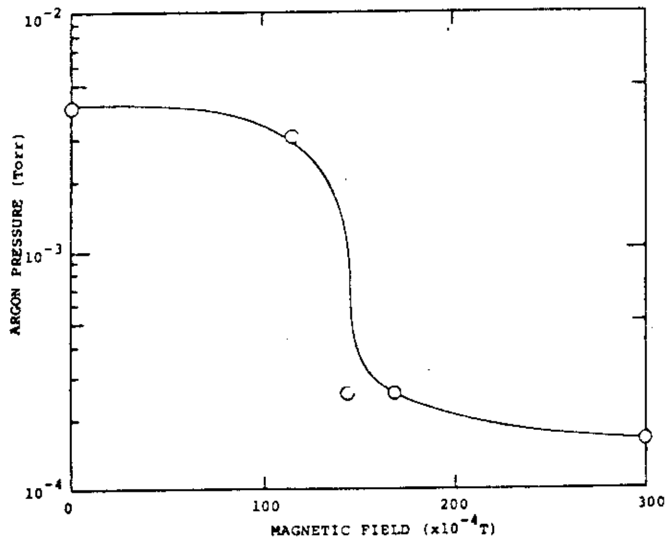


Fig.2.3 Arc burning relation between gas pressure and magnetic field with RF powder of 4 kW (13.56 MHz)

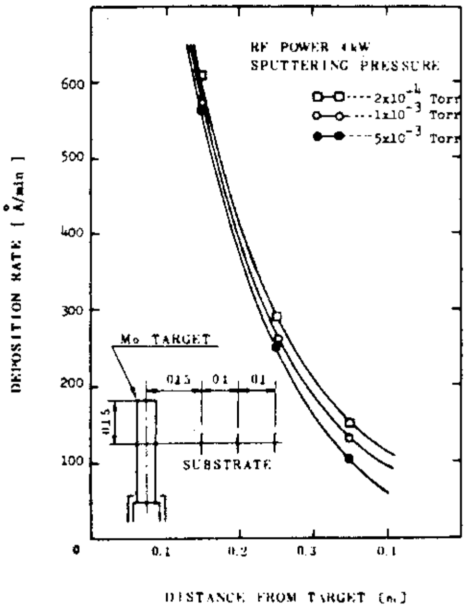


Fig.2.4 Deposition rate by RF sputtering (13.56 MHz) as function of distance from target.

G. Data base of design codes that could be applied to first wall analysis

o When first wall temperature is below $700^{o}F$.

ASME. Boiler and Pressure Vessel Code.

a. Sec. III, Div. 1-Subsection NB-class 1 Components.

b. Sec. VIII, Pressure Vessels-Div. 2-Alternative Rules.

Examples;

. Design load conditions - Sec.III, Div. 1 NB-3221.

. Operating load conditions - Sec.III, Div. 1 NB-3222.

o When first wall temperature is above $700^{o}F$ a cases of ASME. Boiler and Pressure Vessel Code, 1592, Class 1 Components in Elevated Temperature Service, Sec.III.

Examples;

· Deformation and Strain limits for structural integlity : T-1300.

· Creep-fatigue evaluation : T-1400.

Reference

(1) T. Tazima : JAERI-M 7717 (1978)

VI. Design Concepts and Data Base for In-board Shields of Minimum Thickness.

See home task Group 10 Appendix A.

VII. Design Concepts and Data Base for Blanket

A. Previous Studies

In JXFR design study main design parameters of blanket were determined as shown in Table 2.2. Figure 2.5 shows typical JXFR blanket cells. Outer blanket contains Li_2O as a tritium breeding material while inner blanket does not in order to sufficiently attenuate radiation dose rate.

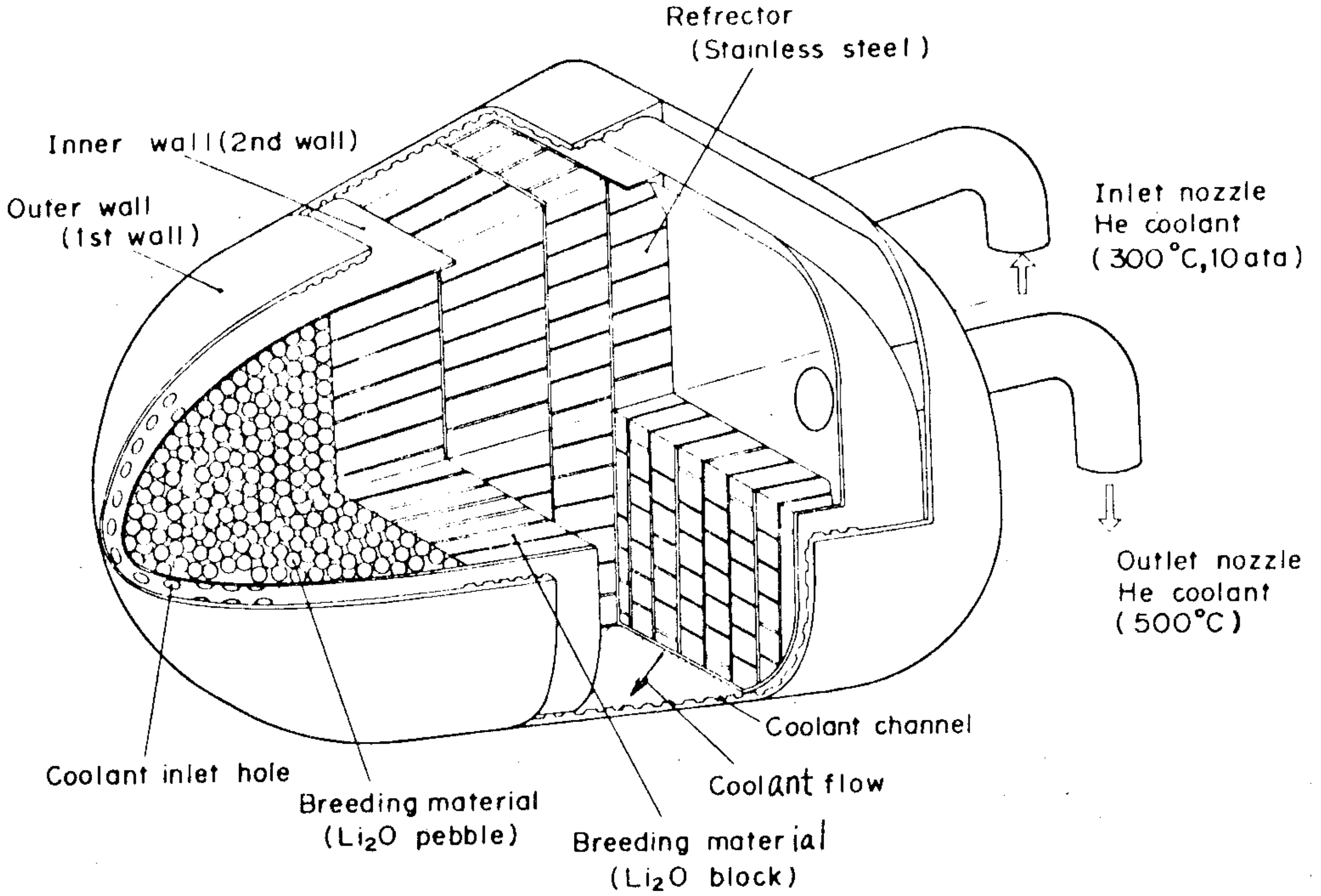


Fig. 2.5 A typical blanket vessel of JXFR

Table 2.2 JXFR blanket parameter

Cooling system	
Number of loops	4
Coolant, Pressure (kg cm^{-2})	He,10
Inlet/Outlet temp. (°C)	300/500
Flow rate (kg/s)	143
Nominal max. 1st wall temp. (°C)	540
Materials	
Structural material	316SS
Blanket fertile material	Li_2O
Reflector material	Stainless steel
Neutronics	
Neutron current at 1st wall (n $cm^{-2}s^{-1}$)	7.6×10^{12}
Neutron wall loading (MW m^{-2})	0.17
First wall displacement damage rate (dpa y^{-1})	1.1
Max. helium production rate in 1st wall (appm y^{-1})	13
Max. hydrogen production rate in 1st wall (appm y^{-1})	38
Tritium breeding ratio	0.9
Nuclear heating per DT neutron (MeV n^{-1})	16.1
Total induced activity at one hour after shutdown (Ci) (after one year operation)	7.3×10^{7}

B. Tritium Breeding Blanket

1. Tritium Breeding Blanket Concept for INTOR

Tritium breeding ratio of INTOR blanket needs not be greater than unity. The blanket should be designed to be able to operate in a fusion reactor condition without failure during INTOR lifetime.

(1) Blanket structure

Fig.2.6 shows a typical blanket cell. The blanket vessel is a double-walled and truncated conical shell made of Type 316 stainless steel with a spherical domed surface at the plasma side. A thick cover plates to which inlet and outlet coaxial ducts attached are placed at the opposite of the plasma. Ribs (5 x 5 mm) for the coolant channel are provided between inner and outer wall. For tritium breeding, the blanket cell contains pebbles and blocks of lithium oxide (Li_2O) which are packed in the stainless steel canning to prevent their deformations.

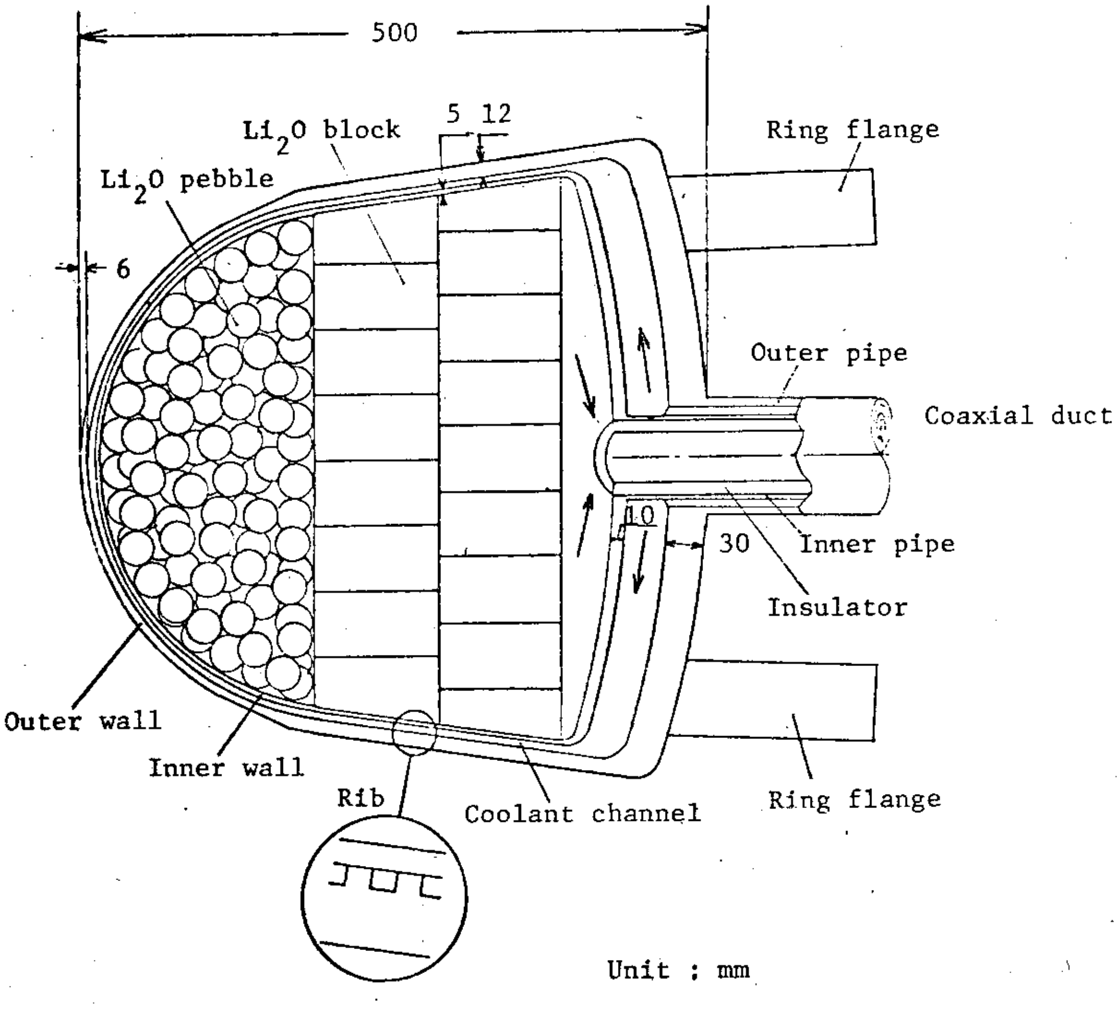


Fig.2.6 A typical blanket cell

(2) Tritium breeding ratio

To estimate tritium breeding ratio of INTOR blanket mentioned above, three cases of calculation are carried out.

Figure 2.7 shows the calculational model of case 1 which is performed by using one-dimensional Sn transport code ANISN. The case 2 and case 3 are three-dimensional calculation by Monte Carlo transport code MORSE-I. Figure 2.8 shows the calculation model of the case 2. In this case, torus geometry of plasma, blanket and shield, neutron source density distribution and the difference between the compositions of inner and outer blankets are taken into account while in one-dimensional calculation they are not. The case 3 is a three-dimensional blanket cell model which takes account of heterogeneity of the cell. Figures 2.9 and 2.10 show the calculational model of the case 3. Table 2.3 shows tritium breeding ratios estimated for the three cases. The case 1 and the case 3 give almost same value of tritium breeding ratio, so the effect of blanket cell heterogeneity must be very small in this case. The tritium breeding ratio the case 2 gives is about two thirds of that the case 1. It is because the case 3 takes account of the fact that the inner blanket does not contain breeding materials. It is almost impossible to load breeding materials in the inner blanket of a small-sized tokamak reactor such as INTOR because the inner blanket is required to be a high energy neutron shield rather than tritium breeding blanket in that case. The tritium breeding ratio of INTOR with blanket cells as shown in Figure 2.6 is estimated to be about 0.6 according to the case 2 calculation. The fusion power density distribution assumed in the case 2 is shown in Figure 2.11.

Table 2.3 Comparison of tritium breeding ratios of one- and three-dimensional calculations

	Tritium Breeding Ratio		
	^{7}Li	^{6}Li	Total
Case 1	0.157	0.751	0.908
Case 2	0.127	0.487	0.614
Case 3	0.158	0.734	0.892

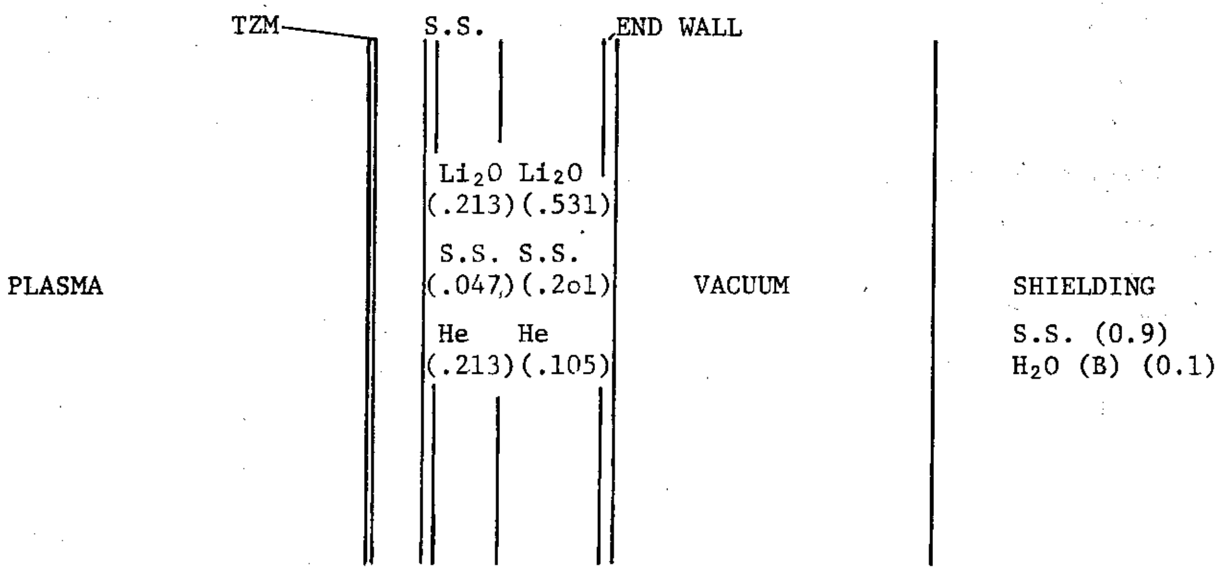
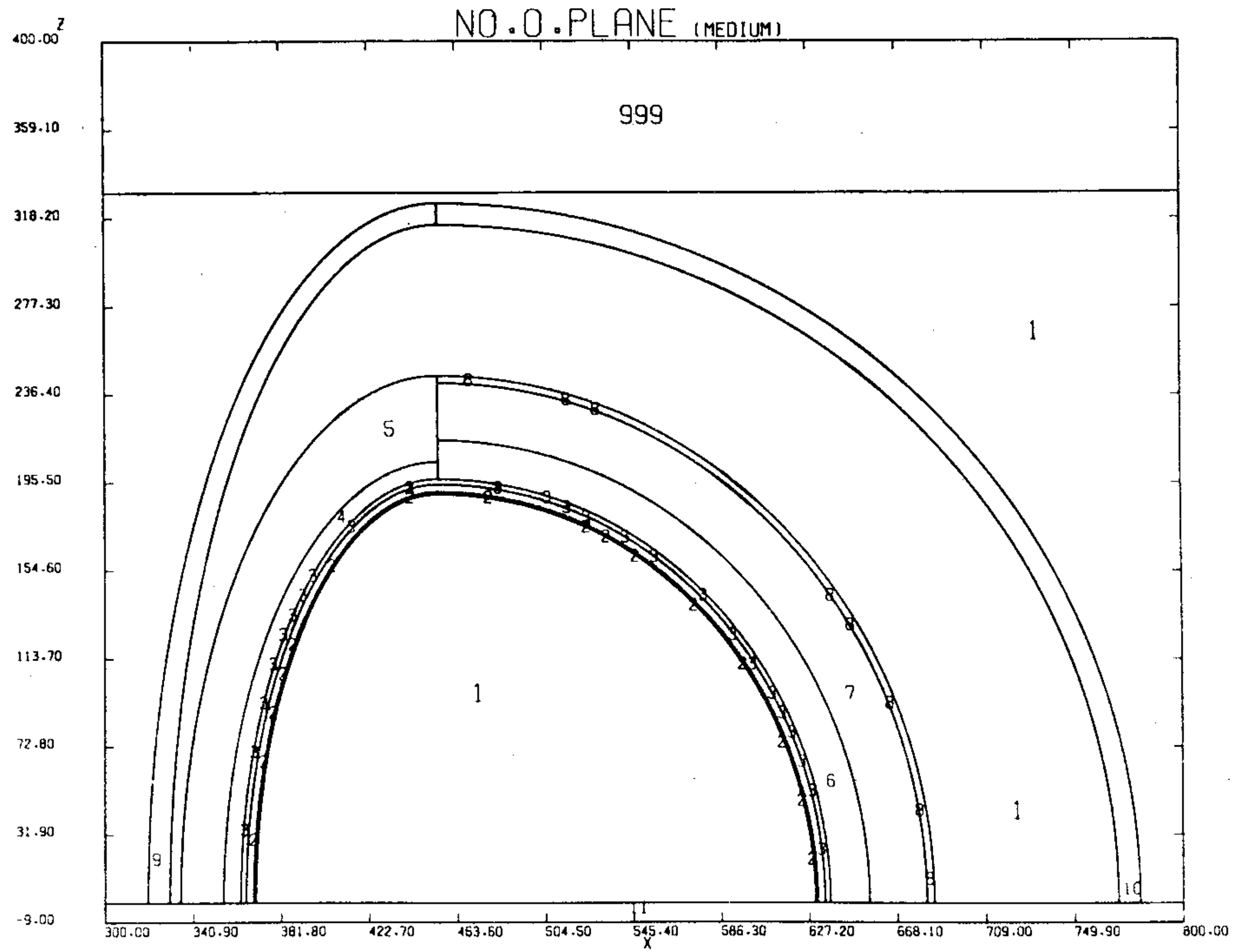


Fig.2.7 One-Dimensional Calculation Model of INTOR-J



1 Vacuum
2 Protection wall
3 First wall
4 S.S.(0.8) + H_2O(0.2)
5 S.S.(0.9) + H_2O(0.1)
6 Li_2O(0.213)+S.S.(0.047)
+He(0.213)
7 Li_2O(0.5305)+S.S.(0.2015)
+He(0.105)
8 End wall
9 Inner shield
10 Outer shield

Fig.2.8 Three-dimensional Calculation Model of INTOR-J

1 Vacuum

2 Protection wall

3 S.S.

4 Li_2O(0.45)+S.S.(0.1)
He(0.45)

5 Li_2O(0.6697)+S.S.(0.0837)
He(0.0837)

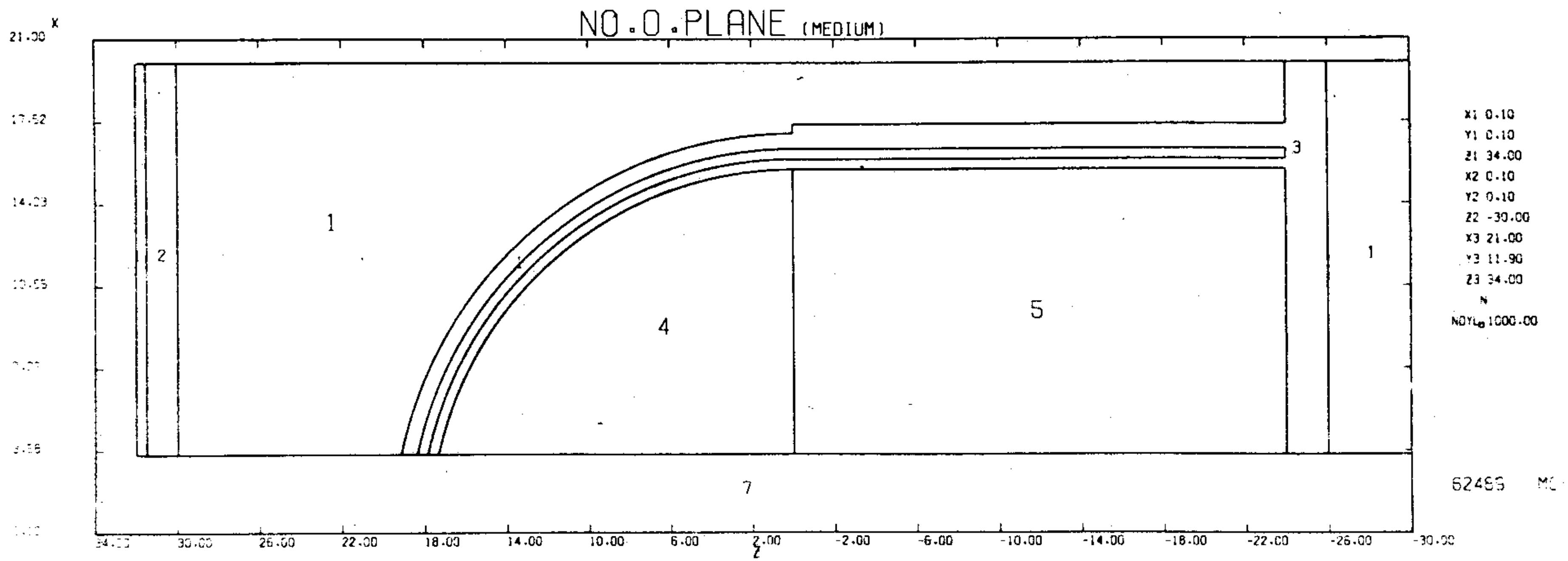


Fig.2.9 Three-dimensional Calculation Model of Blanket Cell

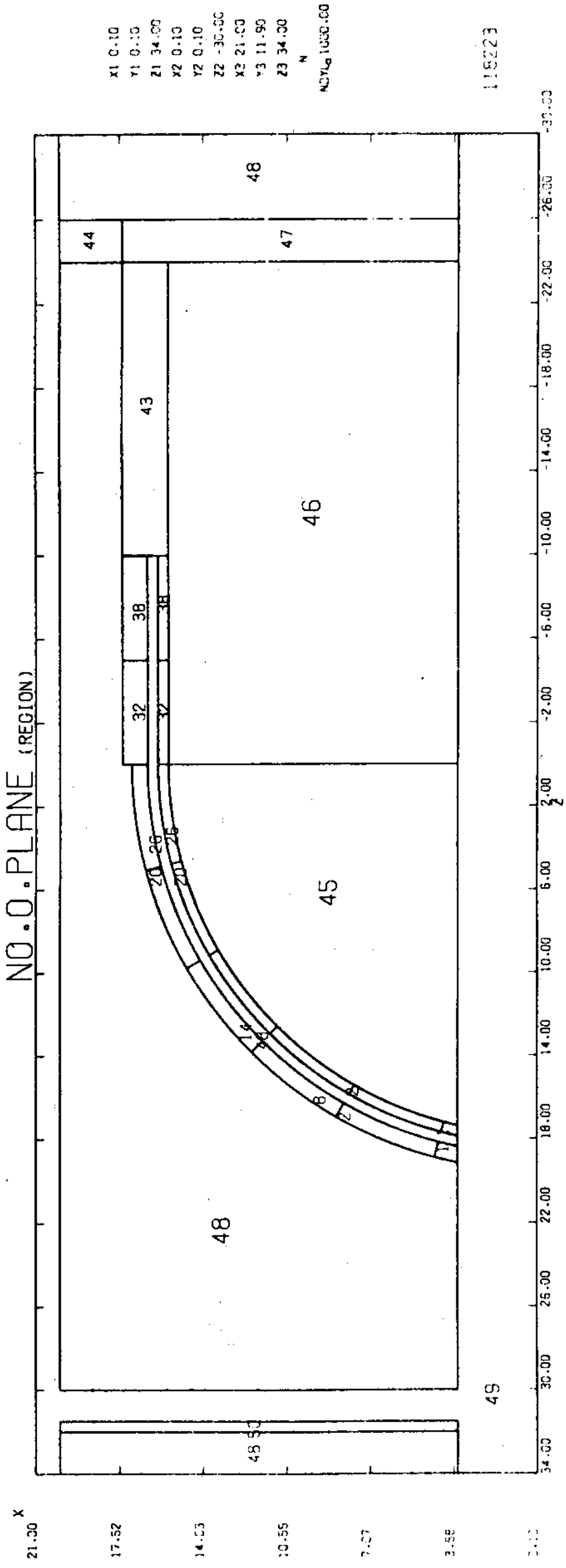


Fig.2.10 Nuclear Heating Density Editing Region for Blanket Cell

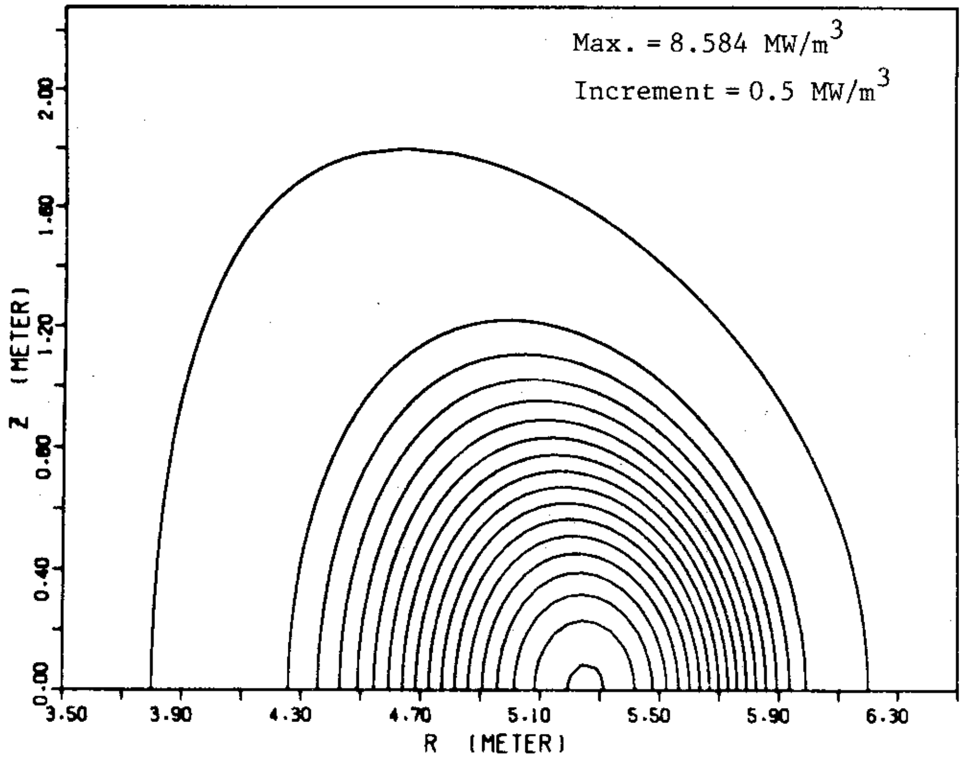


Fig.2.11 Power density contour

(3) Nuclear Heating

Nuclear heating in the first wall is a severe problem for blanket structural design when a neutron wall loading is large. In INTOR case the nuclear heating might be a severe problem. So, as well as tritium breeding ratio, nuclear heating densities are also calculated for the three cases previously mentioned.

Figure 2.12 shows the poloidal distributions of nuclear heating density in the first wall (case 2). The peaking factor is estimated to be 1.25 ~ 1.3 according to this figure. However, it should be noted that the average nuclear heating density in the first wall of the three dimensional calculation is not always equal to that of the one-dimensional calculation. Actually, the average nuclear heating in the case 2 is 7.6 w/cc while that in the case 1 is 10 w/cc. It is because of the difference between fusion power density distributions assumed in one-dimensional and three-dimensional calculation. In the case 1, uniform fusion power density distribution is assumed while in the case 2 the distribution shown in Figure 2.11 is assumed.

Figure 2.13 shows the calculational result of the case 3. For calculation of thermal stress of blanket vessel this nuclear heating density distribution will be used.

2. Tritium breeding ratio of blanket loaded with PbO and Li_2O.

Table 2.4 shows tritium breeding ratios when a part of Li_2O in JXFR blanket is substituted by PbO. Melting point and specific weight of PbO are 880°c and 9.5, respectively. The maximum tritium breeding ratio is obtained when the ratio of Li_2O substituted by PbO is 60% and the enrichment of ^6Li is 20~30%. The maximum ratio is larger than that of JXFR by 7.2%.

VIII. <u>Design Concepts and Data Base for Out-board Shielding</u>

A. }
B. } See Home task Group 10 Appendix A

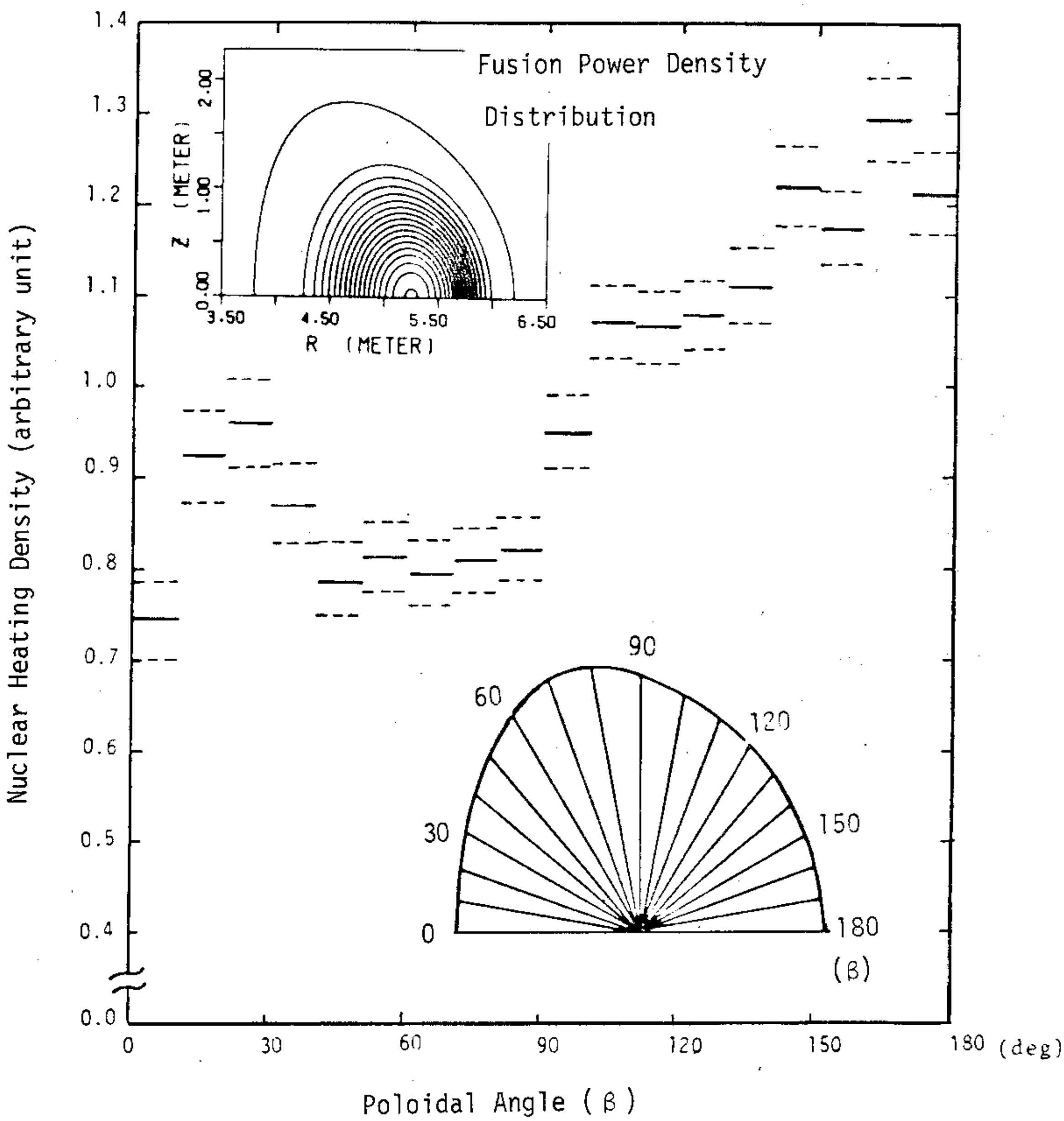


Fig.2.12 Nuclear Heating Density Distribution in the First Wall

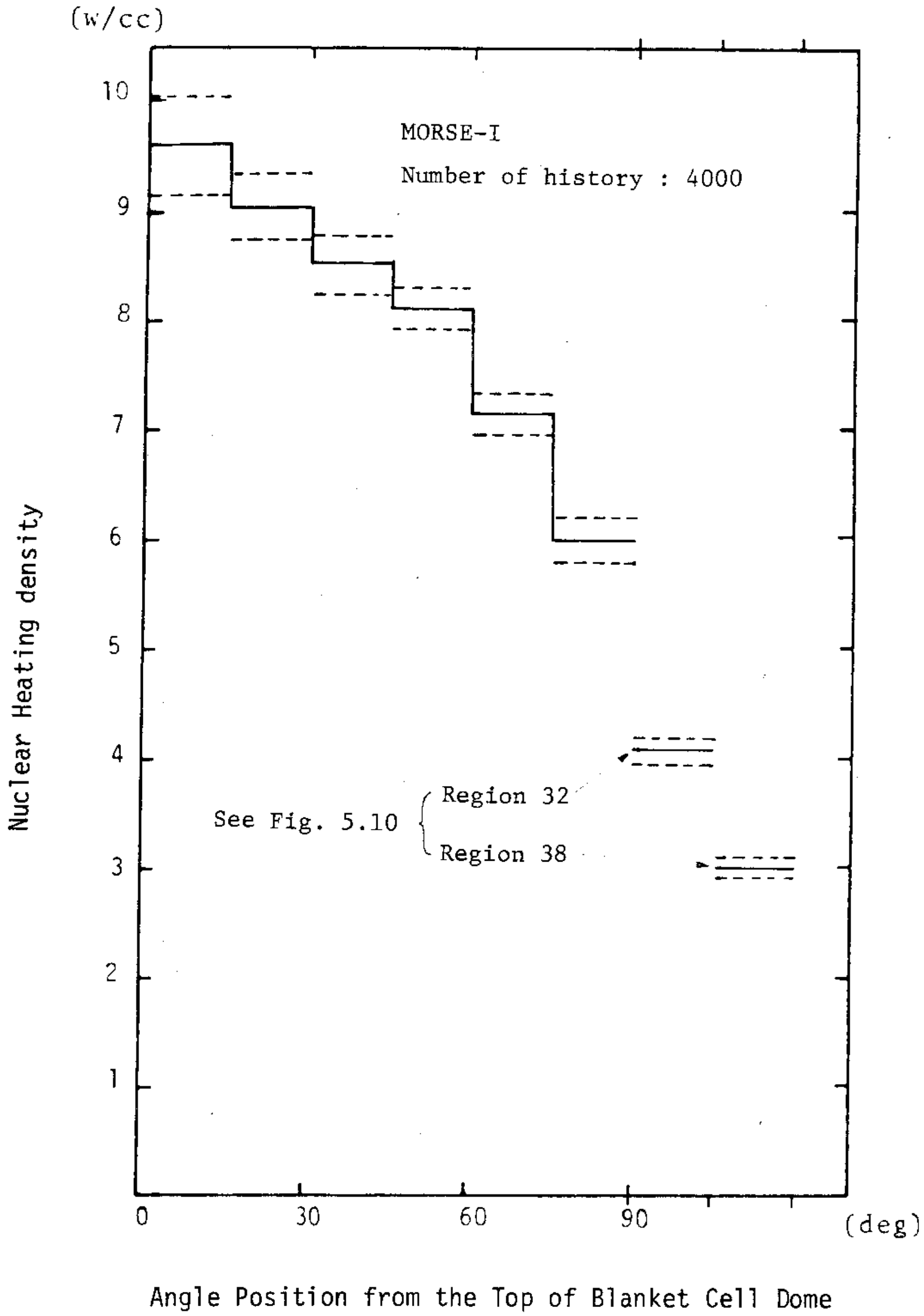


Fig.2.13 Nuclear Heating Density Distribution on the Blanket Dome

Table 2.4 Tritium breeding ratios of JXFR blanket loaded with PbO and Li_2O

	Ratio of Li_2O Substituted by PbO in Pebble Region	Enrichment of ^6Li in Block Region	Tritium Breeding Ratio		
			^7Li	^6Li	Total
Case 1	0 %	7.42 %	0.257	0.835	1.09_2
Case 2	20 %	7.42 %	0.225	0.874	1.09_9
Case 3	40 %	7.42 %	0.196	0.910	1.10_6
Case 4	60 %	7.42 %	0.169	0.942	1.11_1
Case 5	80 %	7.42 %	0.151	0.952	1.10_3
Case 6	100 %	7.42 %	0.120	0.984	1.10_4
Case 7	60 %	14.84 %	0.155	1.003	1.15_8
Case 8	60 %	22.26 %	0.141	1.030	1.17_1
Case 9	60 %	29.68 %	0.127	1.044	1.17_1
Case 10	60 %	37.10 %	0.110	1.053	1.16_3

C. Incorporation of Support Stracture for First Wall/Blanket/Shield

Figure 2.14 shows an example of blanket support. In this design blanket module has replaceable wheels on its lower part. In case of reactor assembly and disassembly the module will be inserted or withdrawn in the radial direction by using these wheels. During operation, however, the module will be supported with the support bloc which is set under the blanket module support.

D. Incorporation of Vacuum Vessel into Shielding

Requirements for a vacuum chamber is to keep its inside at a high vacuum and to heighten the one-turn resistance. Further in view of the evacuation the surface area of the chamber should be reduced as small as possible. The choice of the location of the chamber is important to meet these requirements.

We think the inner side of the bulk shield is the best location for the following reasons.

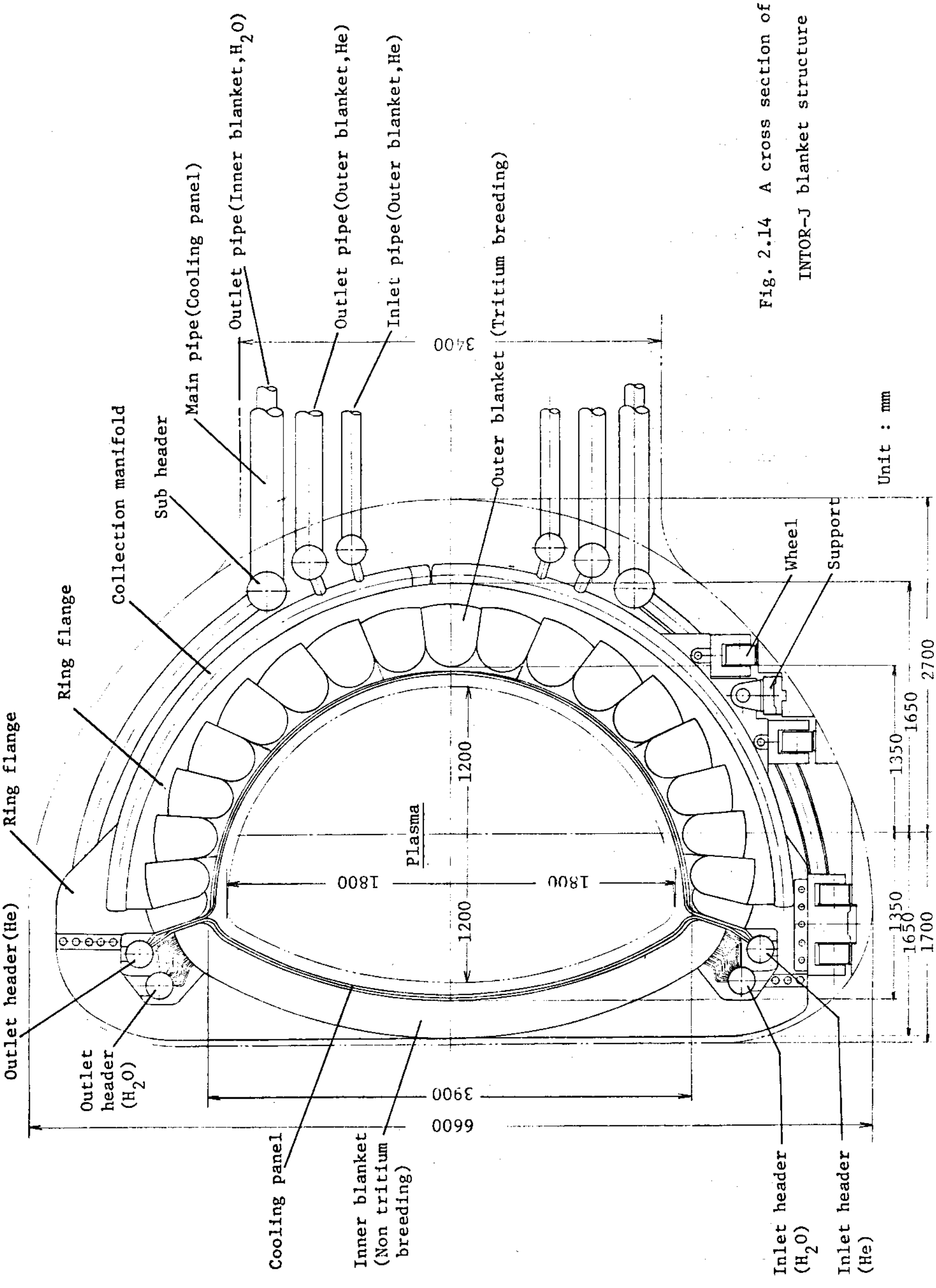


Fig. 2.14 A cross section of INTOR-J blanket structure

i) Locating the vacuum chamber further outward causes a difficulty in evacuation.

ii) The outside of the blanket is too complicated to obtain a vacuum-tight structure.

iii) Placing the vacuum chamber near a plasma is very difficult to design in view of large nuclear heating and a serious problem of electromagnetic force.

Further in cases of ii) and iii) the repair is very difficult.

X. Design Concepts and Data Base for Divertor Target

A. Anticipated Heat Fluxes in INTOR

Figure 2.15 shows the anticipated heatflux density,wall erosion rate ,power loss and plasma line density.

XI. Data Base for Assessing First Wall and blanket Lifetime

Design analysis shall consider the time dependent material properties and structural behavior by guarding against the main modes of failure.

a. Ductile rupture from short-term loading

b. Creep rupture from long term loading

c. Creep-fatigue failure

d. Gross distortion due to incremental collapse and ratcheting

e. Buckling due to short-term loading

The analytical tools available for assessing first wall and blanket lifetime by stress analysis are as follows.

o Stress analysis

(1) SAP-V (elastic)

(2) ANSYS (elastic,plastic and creep)

(3) ASKA (same as above)

(4) MARC (same as above)

(5) NASTRAN (elastic)

(6) NONSAP (elastic,plastic)

o Heat transfer analysis

(1) TAC-1D,2D,3D

(2) TRUMP

(3) ANSYS

(4) MARC-HEAT

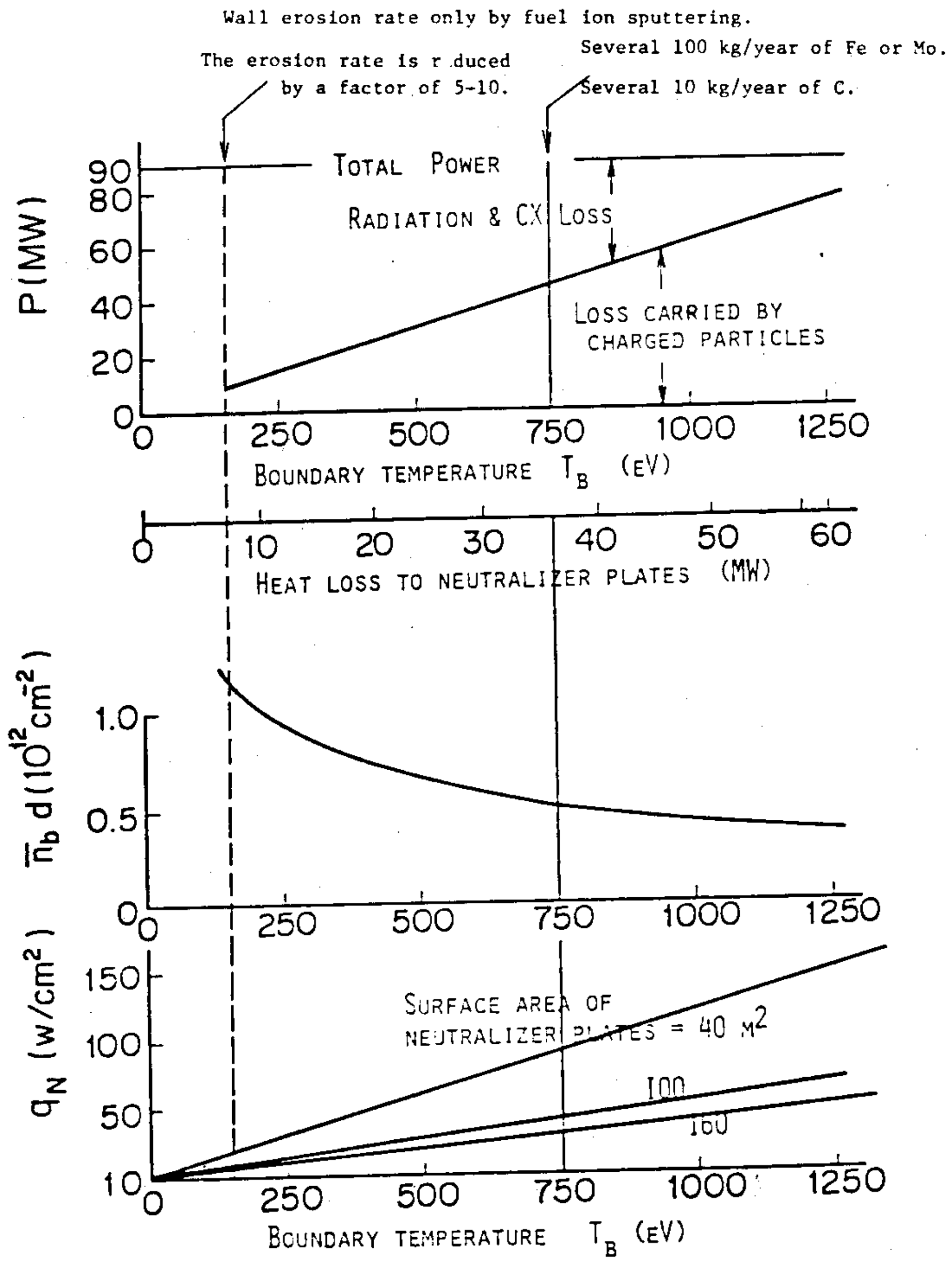


Fig.2.15 Relation between boundary temperature (T_b), power loss (P), heat flux density (q_N) wall erosion rate, and plasma line density ($\bar{n}_b d$)

(From JAERI-M 8294 , Y. Shimomura, 1979)

XII. Data Base for Field Fabrication of Large First Wall, Blanket and Shield Structure Similar to INTOR

C. Previous Examples

Many membrane panels similar to the cooling panel which might be required for INTOR first wall have been already fabricated. They have been used as boilers of petroleum power plant. The welding procedure of membrane panel is described below.

Flat Plate (Fin)

1 Material Check

2 Reforming

3 Check the Width

4 Cutting

Tube

1 Material Check

2 Cutting

3 Polishing

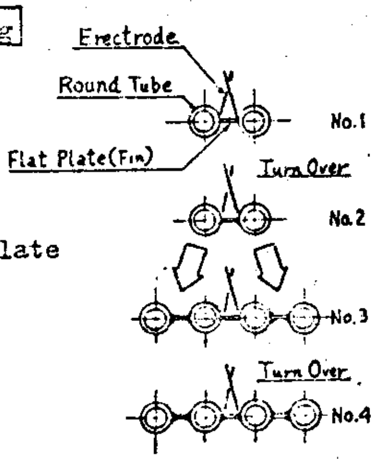
4 Check the Tube Dimension

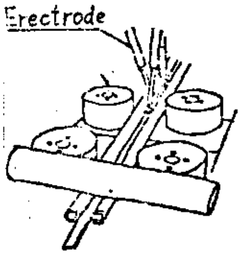
Fin Reforming M/C 3 sets

Tube Polishing Equip. 2 sets

Membrane Panel Welding

Twin-tube welding





5 Setting Roller Jig

6 Welding

7 Gas Cutting of Flat Plate

8 Panel Reforming

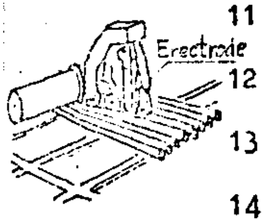
9 Turn Over

10 Welding

Twin-tube Welding Equip. 4 sets

Panel Reforming M/C 2 sets

Multi-tube welding



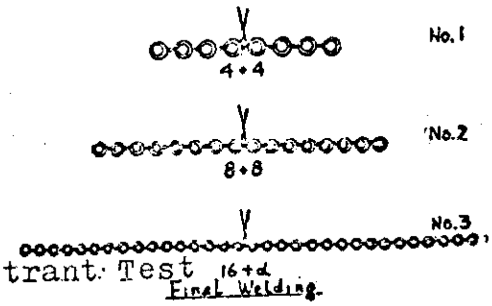
11 Setting Supporting

12 Arrangement of Twin – tubes & Flat – Plate

13 Welding

14 Turn Over

15 Welding



16 Liquid Penetrant Test

17 Repair Welding

18 Measurement and Reform

19 Annealing

↓

To Panel Sub Assembly

Multi-tube Welding Equip. 3 sets

Annealing Furnace

1. Tube polishing

The face of tube is previously painted with the rustproof stuff by tube maker. However, the restproof stuff does not fail to cause blow-holes in deposited metal unless it is eliminated before welding, so the face of tube must be polished. In order to attainour purpose, only the welding zone has to be polished, though actually we polish thoughout the circumference of tube.

Photo. 1 denotes the tube polishing equipment and photo. 2 tube polishing machine. Tubes are fed in, polished and fed away automatically.

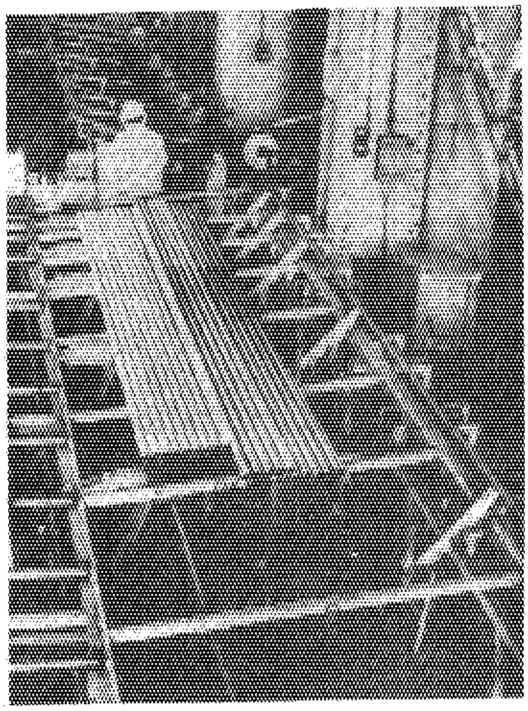


Photo. 1 Tube polishing Equipment

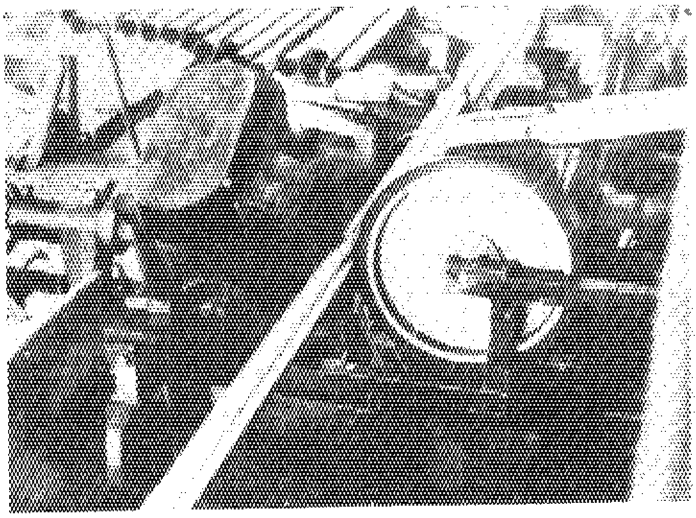


Photo. 2 Tube polishing machine

2. Fine reforming

The width reduction of steel plate by reforming is not less than 1.5 mm. We should straighten the steel plates at the thought of contraction after welding. In case of multi-tube (or panel) welding, the straightened steel plates, the length of which is previously decided, are cut down by the cutter.

3. Twin-tube welding

Polished tube and steel plate are welded together and completed into such tube panel that is shown in figure 1.

Fig. 1 Twin-tube

Photo. 3 demotes the welding installation. Tubes and steel plates are cramped and fed by rollers fixed above, below and on both sides. At the same time, welding by two electrodes is done at the one face and the other.

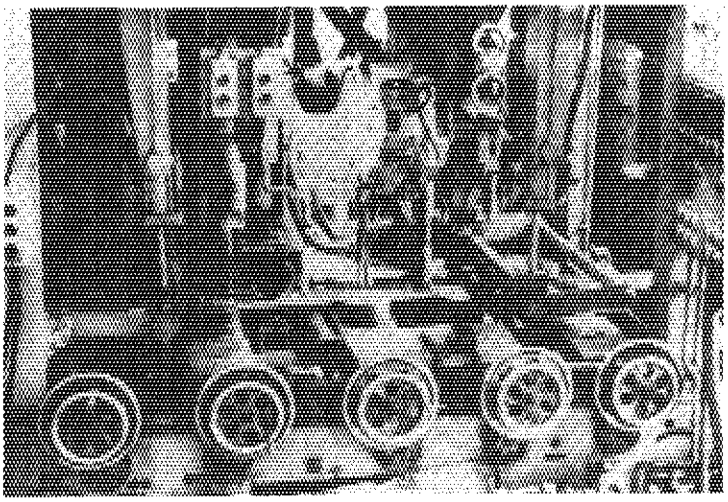


Photo. 3 Twin-tube welding Installation

By dint of pressing rollers, welding strain hardly occur, even if strain occurred.

Welding method is Sub-merged welding type, and the welding rate is 1,200 mm per minute at present. Photo shoes the inlet side of twin-tube welding machine. Tubes are connected in such a simple way as is shown in photo. 4.

After twin-tube welding, the tube panel are welded with another tube panel by twin-tube welding installation (No. 3, 4).

1twin-tube + 1twin-tube = 2.

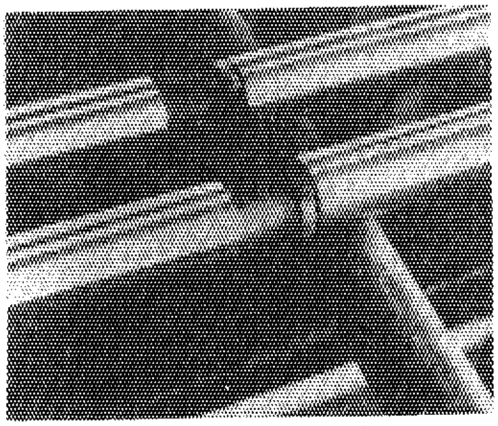


Photo. 4 Joint of tubes

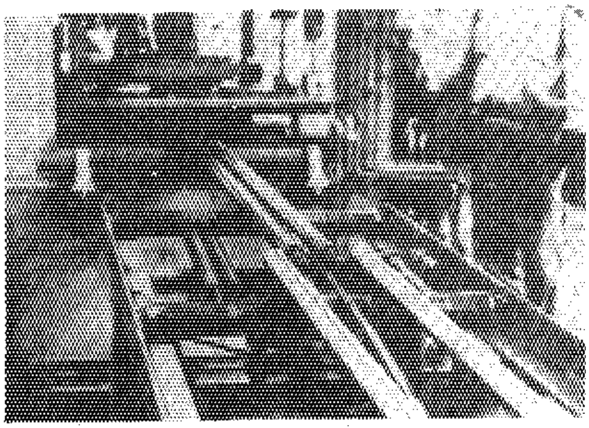


Photo. 5 Inlet side of twin-tube welding machine

Photo. 6 denotes the welding mechanism in detail. Final we cut at the joint as the tube panel pass through the outlet side of welding machine. Photo. 7 shows the cutting process.

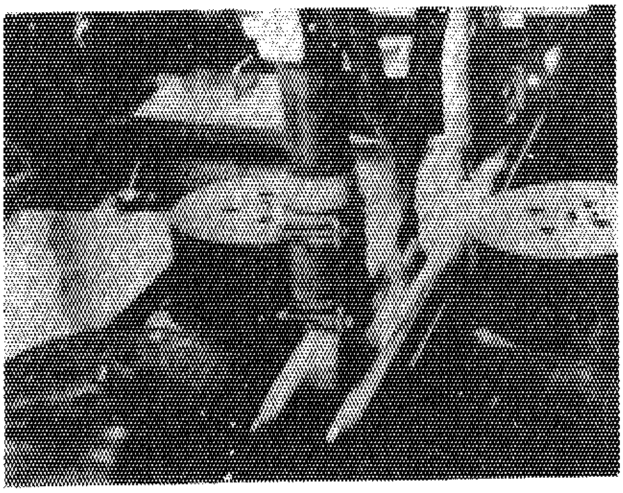


Photo. 6 Welding mechanism

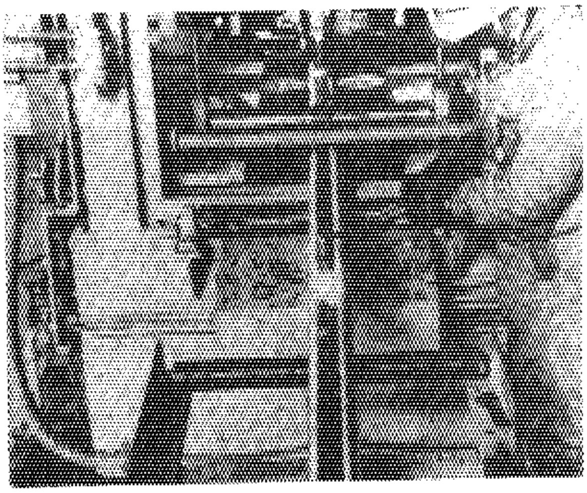


Photo. 7 The outlet side of welding machine

4. Multi-tube welding

After twin-tube welding, the tube panel are welded with another tube panel by multi-tube welding installation. The welding method is as follows.

twin-tube+
twin-tube = 4.

This 2twin-tube is put together with another 2twin-tube

2twin-tube + 2
twin-tube = 8.

And likewise,

4twin-tube + 4
twin-tube = 16.

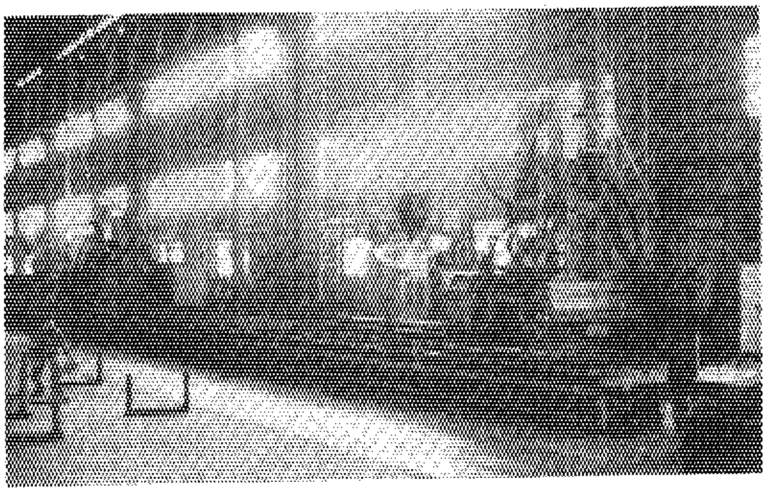


Photo. 8 Multi-tube welding installation

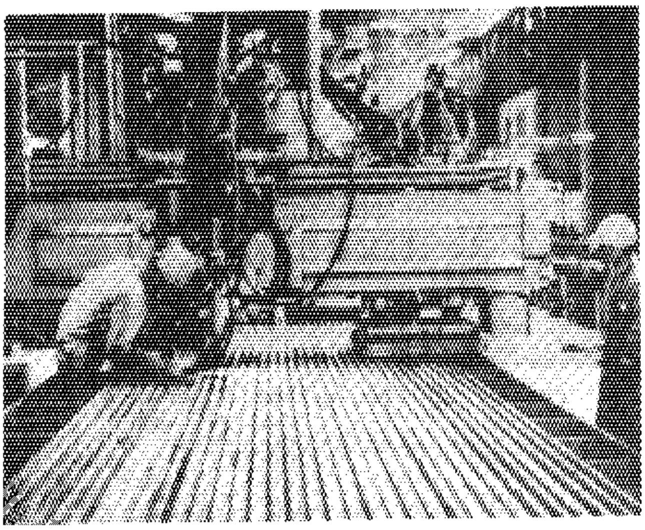


Photo. 9 Being welded

Photo. 10 Head portion of multi-tube welding.

The welding rate of it is 12,00 mm per minute at present. When the panel is welded, it is pressed from on it with the guiding rollers as shown in photo. 9 and photo. 10. The maximum compressive load is two tons.

3. Group 5 Magnetics*

A. Toroidal Coils

1. Requirements of the INTOR toroidal coils

a. What field is required on the axis?

b. What is the allowable field ripple?

5T at the plasma axis and ±1% allowable field ripple are settled, according to the guiding parameters, although R&D are required to realize the TF coils of these parameters.

c. What are the required tolerances of the systems?

The required tolerances should be decided in consideration of following items.

Requirements on accuracy

	To be considered
Plasma characteristics	Toroidal field ripple, error field (horizontal field)
Structure	Electromagnetic force support disassembly, assembly

Disassembly and repair

Activated material

↓

replace or repair

↓

Segmentation / Transportation design

d. What are the realistic alternatives in the structure design? How does maintenance affect the design?

(1) Comparison between disc structure and pancake winding

Two typical coils are compared with each other in some characteristics. Layer winding is not favourable, and excluded in this comparison. The quantitative comparison on basis of coil demonstration will be necessary for our choise of the appropreate

* Report for Session 2 of IAEA INTOR Workshop

coil type. Results of LCT (Large Coil Task) will provide engineering criteria on these types of coils.

Type	Disc	Pancake
Conductor	Electromagnetic force is not accumulated	Accumulated
	Weak jacket conductors are possible (bundle, strand)	Difficult
Current density /Conductor	High	Low
Total weight	Heavy	Light
Stress value	Low	High

(2) Affects of maintenance on coil design

The following items should be considered.

- . Less coil number and thinner coil are favourable for enough access.
- . Locally insufficient coil shielding is allowable in inevitable case.
- . Demountable structure (for example, sector segmentation) is preferred.
- . Support structure for centering force should be able to be assembled and disassembled without worker's access.

e. What heat loads (radiation, conduction, joule heating, etc.) are practical for the alternate designs?

How do access requirements (beams, machine assembly, diagnostics, test modules, maintenance, etc.) affect the design?

Heat load analysis in our INTOR design study has not been performed yet. Calculated values on JXFR are shown as follows.

(1) superconductor loss (1.30 kW)

(2) neutron and gamma ray heating (0.98 kW)

gamma ray heating is dominant

(3) eddy current loss at structure material (0.15 kW)
(4) cryostat loss (3.98 kW)
(5) current lead loss (7.49 kW)
* total (13.9 kW)

f. How does the fault analysis affect the design of the magnet and support structure?

In safety design of superconducting magnet for fusion reactors, it is necessary that occurrence of any disturbances (power break-down, fault of each apparatus, earthquake) at normal operation do not impair the safety for the public. Coil protection systems and other apparatus, required for keeping the safety, are equiped and must be surely operated.

In order to satisfy these conditions, the magnet should be designed considering following items.

(1) Assuming a simple fault, important system or apparatus should keep the ability of carrying out its function.
(2) The concept of "Defence in Depth" should be adopted.
(3) Automatic operation system is favourable.
(4) Even in the case of incorrect operation, apparatus should not go into dangerous way.
(5) Reasonable interlock should be used.
(6) Transmission of information should be highly reliable.
(7) Coil protection system should not work frequently by incorrect signals.

In safety design study for the superconducting magnet, it is needed to know various origins of fault and its effect on other systems and in addition, to add protection systems against the fault and the effect. Faults and its protection systems are tabulated in FMEA chart (Table 3.1).

Fault Tree Analysis (FTA) is shown in Fig. 3.1. Systems considered in the analysis are superconducting coil, cryostat and superconducting disc. Top event of the analysis is rupture of the superconducting magnet and secondary event is rupture of each system mentioned above. In this analysis, not only rupture of structural element in the components but also earthquake, fire, missile are considered.

Table 3.1 F M E A Chart

(1) Superconducting Coil

NO.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class*			Manage-ment
							1	2	3	
1	Superconducting Cable	Current Carrying	Incapable of Power Supply	Breaking of a Wire, Failure	Coil Breakage	Voltmeter	○			Operation of Coil Protection System
2	Solder	Soldering Superconducting Cable	Exfoliation	Excessive Heat Generation, Wire Movement	Failure of Superconducting Cable				○	
3	Stabilizer	Quench Protection	Incapable of Quench Protection	Excessive Stress, Failure, Insufficient Cooling	Loss of Superconducting State, Quench	Voltmeter	○			As Above
4	Slot of Superconducting Disc	Setting of Superconducting Cable	Dropdown of Superconducting Cable	Large Distortion of Slot, Failure of Suppression Plate, Wire Movement	Failure of Superconducting Cable			○		
5	He Passage	Coolant Flow Passage	No Coolant Flow	Blockage of Coolant Passage	In Capable of Conductor Cooling and Maintaining Superconducting State, Temperature Rise, Melting		○			As Above
6	Inter-Disc Insulator	Protection Against Eddy Current	Eddy Current	Failure of Insulator	Excessive Magnetic Force, Increace of Structure Weight and Inner Heat Generation			○		

Table 3.1 (Continued) F M E A Chart

(1) Superconducting Coil (Continued)

NO.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Manage-ment
7	Coil Suppression Plate	Suppressing Superconducting Cable	Large Distortion, Failure	Large Distortion of Slot.	Failure of Superconducting Cable			○		
8	Superconducting Disc	Support of Magnetic Force Acting on Superconducting Cable	Fracture of Disc	Excessive Magnetic Force, Defect at Welded Area	Coil Breakage, Failure of Superconducting Cable		○			

* Class 1: Heavy class fault and in principle operation should be stopped.
  Class 2: Middle class fault and to be repaired or exchanged of the apparatus.
  Class 3: The other faults.

Table 3.1
(Continued) F M E A Chart

(2) Cryostat

NO.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
1	Vacuum Chamber	Heat Insulation	Leakage	Breakage of Vacuum Pump, Failure of Valve	Pressure Increase in He Chamber by Temperature Rise	Vacuum Gauge		○		
2	Super Insulation	Lowering of Heat Conduction	Excessive Incoming Heat Conduction	Mechanical Fracture of Super Insulation	Temperature and Pressure Rise in He Chamber			○		
3	Vapour Shield	Heat Removal	Insufficient Heat Removal	Insufficiency of Coolant Flow, Failure of Vapour Shield	Excessive Load on Refrigerator	Flowmeter		○		
4	FRP Support	Support Between Vacuum and He Chamber	Failure of FRP Support	Large Distortion of Vacuum or He Chamber	Excessive Movement of Vacuum or He Chamber, Fracture of Super Insulation and Vapour Shield			○		
5	Rapture Disc	Rupture at Prescribed Pressure to Protect He Chamber	No Fracture, Crack	Excessive Design Pressure, Failure	Fracture of He Chamber, Fracture Below the Prescribed Pressure, Outflow of He		○			
6	He Chamber	Containment of He	Function Loss of He	Failure, No Coolant Flow	Quench, Temperature Rise of Conductor		○			Stop of Operation

Table 3.1 (Continued) F M E A Chart

(3) Support Structure

NO.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
1	Support Cylinder	Support of Centering Force of Coil	Incapable of Force Support	Breakage, Excessive Magnetic Force	Unbalance of Coil Position	Strain Gauge, Position Meter	○			Stoppage of Operation
2	Support Ring	As Above	As Above	As Above	Large Load on Support Cylinder	As Above	○			As Above
3	Key	Pressure Coil Interval (Insulator)	Excessive Distortion, Fracture	Fracture, Excessive Magnetic Force	Degradation of Accuracy of Coil Interval, Support Unbalance			○		
4	Support Beam	Lateral Force Support	Incapable of Force Support	Fracture, Excessive Stress	Coil Fracture, Function Loss of Coil	Strain Gauge, Position Meter	○			As Above
5	Insulator	Prevention of Electric Loop Formation	Electric Loop Formation	Fracture	Excessive Magnetic Force Generation, Bad Effect on Plasma Start Up			○		As Above
6	Bellows	Connecting of Inter-Coil Vacuum Chamber	Fracture of Connecter	Defect of Welded Area	Degradation of Vacuum Level	Vacuum Gauge		○		
7	Room Temperature Pin	Support of Coil Weight	Incapable of Coil Support	Fracture, Excessive Magnetic Force, Earthquake	Dropdown of Coil, Function Loss of Cryostat			○		

Table 3.1 (Continued) F M E A Chart

(3) Support Structure (Continued)

No.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
8	Cryogenic Temperature Pin	Support of Coil Weight	Incapable of Coil Support	Fracture, Excessive Magnetic Force	Dropdown of Coil, Function Loss of Cryostat			○		
9	Lever	Connecting of Pins and Coil Support	As Above	As Above	As Above	Strain Gauge		○		
10	Support Leg	Support of Coil Weight	As Above	As Above	Function Loss of Support Structure, Fracture of Cryostat	Strain Gauge	○			Stoppage of Operation

Table 3.1
(Continued) F M E A Chart

(4) Cooling System

No.	Compoment	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
1	Main Coolant System	Cool the Coolant	Function Loss	Failure of Component	Insufficient Magnet Cooling, Incapable of Cooldown	Thermometer		○		
2	Cryogenic Distributor	He Distribution to Each Cryostat	Function Loss of Distribution System	Failure of Component	Incapable of He Supply to Cryostat	Flowmeter		○		
3	Liquefaction System	Liquefaction of He	Incapable of Liquefaction	As Above	Insufficiency of Liquid He	As Above		○		
4	Liquid He Container	He Storage	Incapable of He Storage	Leakage, Breakage	Insufficient He Supply, Excessive Load on Refrigerator	Levelmeter		○		
5	He gas Recovery System	Collect He Gas	Loss of Recovery Function	Failure of Component	Incapable of He Gas Recovery	Flowmeter		○		
6	Heat Exchanger	Remove the Surplus Heat	Leakage, Breakage	Breakage, Injury of the Seal, Fatigue of the Parts with Long Term Use	Incapable of Surplus Heat Removal, Temperature Rise of Coolant	Thermometer		○		
7	Expansion Engine	He Cooling	Loss of its Function	Breakage of Component	Insufficient Magnet Cooling, Insufficient Heat Removal of Vapor Seal			○		

Table 3.1 (Continued) F M E A Chart

(4) Cooling System (Continued)

No.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
8	Gas Holder	Gas Storage	Leakage, Breakage	Injury of the Seal, Breakage by Unusual Force	Gas Release Outside of the System	Pressure Gauge			○	
9	Valve	Control of the Coolant Flow	As Above	As Above		Inspection			○	
10	Pipe	Supply or Collect the Coolant	Leakage, Choke	Injury of the Seal, Mixing of the Un-expected Material in the Coolant	Degradation of Cooling Ability	Flowmeter			○	

Table 3.1
(Continued)
F M E A Chart

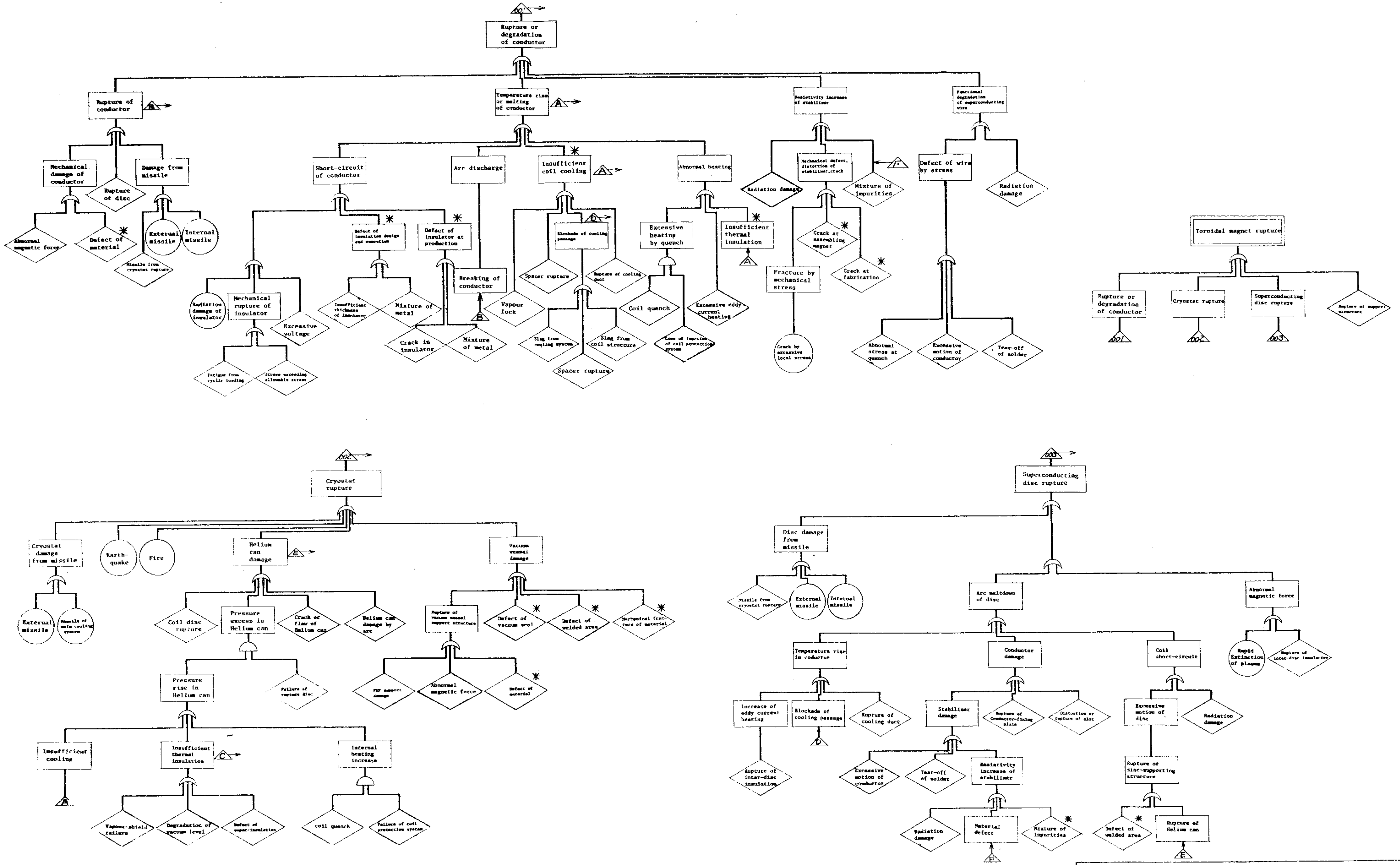
(5) Coil Excitation, Measurement

No.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
1	Power Supply	Current Supply for Coil	No Power	Disconnection, Short Circuit	Incapable of Coil Excitation	Ammeter Voltmeter		○		Exchange or Repair
2	Power Lead	As Above	As Above	As Above	As Above	As Above		○		As Above
3	Ammeter	Current Measurement	Incorrect Display, No Display	Short Circuit, Failure of Component	Incapable of Current Check, Undetectable of Coil Failure	Inspection			○	As Above
4	Voltmeter	Voltage Measurement	As Above	As Above	Undetectable of Coil Quench	As Above			○	As Above
5	Pressure	Pressure Measurement	As Above	As Above	Undetectable of Vacuum Level of Vacuum Chamber and Excessive Heat Generation of He Chamber	As Above			○	As Above
6	Strain (Stress) Gauge	Strain Measurement	As Above	As Above	Incapable of Check of Strain	As Above		○		As Above
7	Magnetic Flux Meter	Magnetic Flux Measurement	As Above	As Above	Incapable of Check of Magnetic Field	As Above			○	
8	Thermometer	Temperature Measurement	As Above	As Above	Incapable of Check of Over Heat of Leads	As Above			○	
9	He Level Gauge	He Level Measurement	As Above	As Above	Undetectable of Needed Coolant Flow	As Above		○		

Table 3.1 (Continued) F M E A Chart

(5) Coil Excitation, Measurement (Continued)

No.	Component	Function, Purpose	Fault Mode	Origin of Fault	Effect on System	Detection	Class 1	Class 2	Class 3	Management
10	Current Circuit Breaker	Breakage of Coil Current	Incapable of Operation	Failure of Parts	Coil Damage at Quench	Inspection		○		
11	Dumping Resistor	Release of Coil Energy at Quench	rupture	Short-Circuit, Failure of Parts	Coil Failure at Quench	As Above		○		
12	Quench Detection System	Detection of Coil Quench	Function Loss	Failure of Detection System	Undetectable of Coil Quench	As Above		○		



Rupture or degradation of conductor
Rupture of conductor
Mechanical damage of conductor
Rupture of disc
Damage from missile
Abnormal magnetic force
Defect of material
External missile
Internal missile
Missile from cryostat rupture
Temperature rise or melting of conductor
Short-circuit of conductor
Arc discharge
Insufficient coil cooling
Abnormal heating
Defect of insulation design and execution
Defect of insulator at production
Breaking of conductor
Radiation damage of insulator
Mechanical rupture of insulator
Excessive voltage
Fatigue from cyclic loading
Stress exceeding allowable stress
Insufficient thickness of insulator
Mixture of metal
Crack in insulator
Vapour lock
Blockade of cooling passage
Spacer rupture
Rupture of cooling duct
Slag from cooling system
Slag from coil structure
Excessive heating by quench
Insufficient thermal insulation
Coil quench
Loss of function of coil protection system
Excessive eddy current heating
Resistivity increase of stabilizer
Radiation damage
Mechanical defect, distortion of stabilizer, crack
Mixture of impurities
Crack at assembling magnet
Crack at fabrication
Fracture by mechanical stress
Crack by excessive local stress
Functional degradation of superconducting wire
Defect of wire by stress
Abnormal stress at quench
Excessive motion of conductor
Tear-off of solder
Toroidal magnet rupture
Cryostat rupture
Superconducting disc rupture
Rupture of support structure
Cryostat damage from missile
Earth-quake
Fire
Helium can damage
Vacuum vessel damage
Missile of main cooling system
Coil disc rupture
Pressure excess in Helium can
Crack or flaw of Helium can
Helium can damage by arc
Pressure rise in Helium can
Failure of rupture disc
Insufficient cooling
Insufficient thermal insulation
Internal heating increase
Vapour-shield failure
Degradation of vacuum level
Defect of super-insulation
Failure of coil protection system
Rupture of vacuum vessel support structure
Defect of vacuum seal
Defect of welded area
Mechanical fracture of material
FRP support damage
Abnormal magnetic force
Defect of material
Disc damage from missile
Arc meltdown of disc
Abnormal magnetic force
Rapid Extinction of plasma
Rupture of inter-disc insulation
Temperature rise in conductor
Conductor damage
Coil short-circuit
Increase of eddy current heating
Blockade of cooling passage
Rupture of cooling duct
Stabilizer damage
Rupture of Conductor-fixing plate
Distortion or rupture of disc
Excessive motion of disc
Radiation damage
Rupture of inter-disc insulation
Tear-off of solder
Resistivity increase of stabilizer
Rupture of disc-supporting structure
Material defect
Defect of welded area
Rupture of Helium can

Fig. 3.1

2. What knowledge, experience is required to design, construct and operate the toroidal magnetic system of INTOR? Extend, modify or add to these answers:
   a. Development technology of production of large Nb_3Sn and NbTi coils
   b. Construct and test several coils
   c. Development of a highly efficient liquefied and refrigeration system
   d. Safety analysis of superconducting system

   The following R&D are required.
   a. Demonstration of operation of high field and large coil with Nb_3Sn or NbTi
   b. Demonstration of large capacity He liquefier and refrigerator
   c. Refined system of automatic operation of He liquefier and refrigerator (large capacity compressor, large capacity gas collection and supply system)
   d. Good selection of materials and operational load (structure, insulator)
   e. Establishment of design criteria, specially stress problem (problem of modeling of winding which is composed of several materials)
   f. Fabrication and winding technique of large conductor

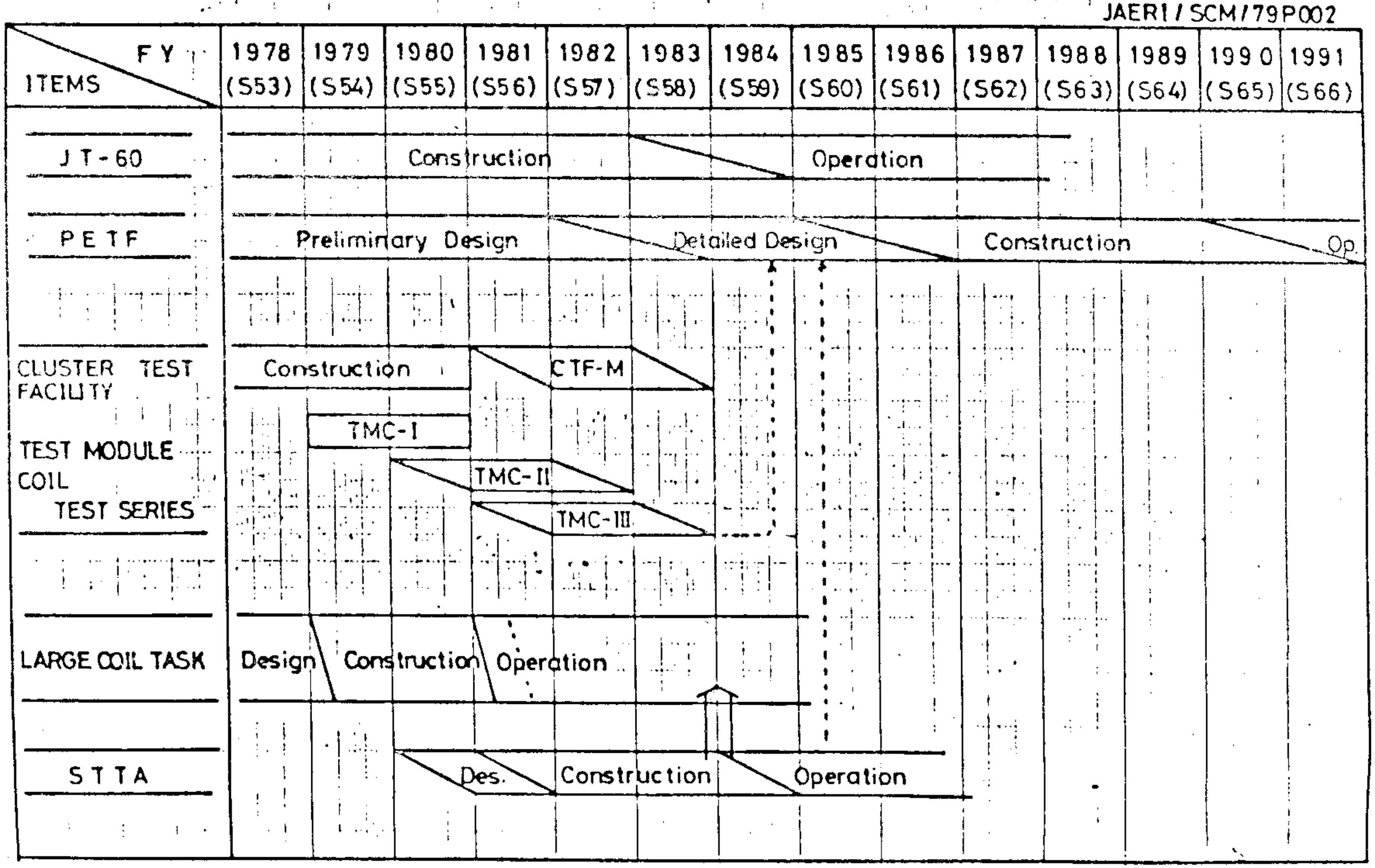
3. List and analyze existing or expected data base and add further information:

   e. Large Coil Program

   The development of superconducting toroidal coil is mainly performed in JAERI. Table 3.2 shows the superconducting toroidal coil development plan in JAERI. Two large projects, Cluster Test Facility and LCT are being carried out.

Table 3.2

Superconducting Toroidal Coil Development Plan



i) Cluster test facility and test module coil

Following the preliminary design work of cluster test facility in the fiscal year of 1976, we carried out the advanced design of the facility and test module coil in relation with IEA-Large Coil Task.

The design conditions are a little modified from the work of 1976. The modified conditions are followings.

..... The current in a test module coil is 10,000 12,000 A

..... The maximum field in a test module coil is 8.0 T

..... The configuration of a test module coil is similar to IEA-LCT coil.

The size is half of IEA-LCT coil.

The back ground coil, called as cluster test coil has 1.5 m mean diameter, the operational current of 2,150 A and the maximum field of 7 T in the winding with the operating test module coil.

The coil disposition and the dimension of the cluster test coil and the test module coil are shown in Fig. 3.3. In order to simulate the poloidal field effect on the test module coil, a separated system is designed as pluse poloidal coil, which is also superconducting. The principal designed values of the cluster test coil and the test module coil are shown in Table 3.3.

This facility requires a refrigerator of 250 W at 4.2 K. The total cooling weight is about 25 tons.

With the cluster test facility the principal experimental simulation of toroidal coil can be carried out on conductor, stability, insulator, mechanical reinforcement, cooling system and varying field effect. In order to generate a high stress in a test module coil, a high current density coil may be constructed and tested with a fixed ampere turns.

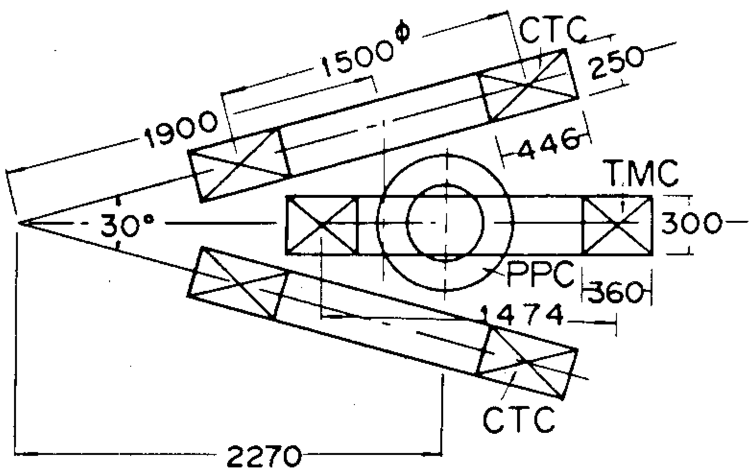


Fig. 3.3 Cluster test facility disposition

Table 3.3 Designed value of the test module coil and the cluster test coil

TMC

Coil Cross Section	(mm)	360×300
Coil Shape		Oval
Coil Size	(m)	1.47×1.97
Magnetic Field	(T)	8.0
Average Current Density	(A/mm^2)	30
Operating Current	(KA)	10-12
Stord Energy	(MF)	15
Centering Force	(ton)	1400
Winding Volume	(m^3)	0.589
Energy Density	(MJ/m^3)	30
Ampere-turns	(MAT)	3.24
Weight	(ton)	7
Minimum Winding Curvature	(m)	0.4
Number of Turn		3240
Wiere Length per Turn	(m)	5,53
All Wire Length	(m)	1792.
B_{max}/B_{min}		1.62

CTC

Magnetic Field	(G)	7.0
Coil Size	(m)	1.5
Ampere-turn	(MAT)	3.3×2
Current Density	(A/mm)	30
Total Energy	(MJ)	40

ii) Japanese LCT coil design

The Japanese LCT coil design was carried out in 1978 by JAERI superconducting magnet laboratory in collaboration with three electro-technical companies and three cable companies.

The design parameters are shown below.

JAPANESE LCT COIL

COIL BORE	2.5M X 3.5M
MAX. OPERATIONAL FIELD	8.0T
AMPERE TURNS	6.76×10^{6}AT
OPERATIONAL CURRENT	10,210A
S.C. MATERIAL	NB-TI
CONDUCTOR	ROUGH SURFACE COPPER STABILIZER SOLDERED CABLE SUPERCONDUCTOR
COOLING	POOL BOILING (4.2K, 1ATM)
WINDING	EDGE WISE, 2 GRADINGS 20 DOUBLE PANCAKES
STRUCTURAL MATERIAL	304L STAINLESS STEEL
ASSEMBLY	WELDED CASE WITH BOLT
WEIGHT	39 - 40TON

8T CONDUCTOR

* THERMAL ANALYSIS: IN CASE THAT THE SURFACE IS "THEMOEXCEL"

EFFECTIVE ELECTRICAL CROSS SECTION OF COPPER: 234.1MM^2
JOULE HEATING PER UNIT LENGTH: 2.76W/CM
COOLING CHARACTERISTIC PER UNIT LENGTH:
100% OF BOTH SURFACE IS EXPOSED,
Q_{MAX}= 4.02 - 5.04W/CM
Q_{MIN}= 2.18 - 3.27W/CM
ELECTRICAL RESISTIVITY AT 8T: 6 X 10^{-8}OHM-CM

* SUPERCONDUCTING CHARACTERISTICS:

CURRENT DENSITY IN SUPEROCNDUCTOR AT 8T: 6 X 10^4A/CM^2
PACKING FACTOR OF CABLE: 85%
OPERATIONAL CURRENT AT 8T: 10,200A
STRAND: 15 STRANDS, 2.3MMØ, CU RATIO 1.0
NB-TI: 50μØ X 1060 FILAMENTS, TWIST PITCH 30MM,
STRAND PITCH 200MM

5T CONDUCTOR

* THERMAL ANALYSIS: IN CASE THAT THE SURFACE IS "THERMOEXCEL"

EFFECTIVE ELECTRICAL CROSS SECTION OF COPPER: 181.1MM^2
JOULE HEATING PER UNIT LENGTH: 2.59W/CM
COOLING CHARACTERISTIC PER UNIT LENGTH:
100% OF BOTH SURFACE IS EXPOSED,
Q_{MAX}= 4.02 - 5.04W/CM
Q_{MIN}= 2.18 - 3.27W/CM
ELECTRICAL RESISTIVITY AT 5T: 4.5 X 10^{-8}OHM-CM

* SUPERCONDUCTING CHARACTERISTICS:

CURRENT DENSITY IN SUPERCONDUCTOR AR 5T: 14 X 10^4A/CM^2
PACKING FACTOR OF CABLE: 83%
OPERATIONAL CURRENT AT 5T: 10,200A
STRAND: 11 STRANDS, 1.9MMØ, CU RATIO 1.4
NB-TI: 50uØ X 600 FILAMENTS, TWIST PITCH 25MM,
STRAND PITCH 130MM

The manufacturing of the Japanese LCT coil will be started in 1979 and will be finished in 1981. The coil will be tested in Japan and will be transported to Oak Ridge National Laboratory at the end of 1981.

B. Poloidal Coils (inductor, control coils, divertor)

1. Requirement of the INTOR poloidal coils

a. Which are coils? Superconductor or not? If superconductor, is it Nb_3Sn or NbTi? Cryogenic or normal, if not superconductor?

Fig. 3.4 shows layout of poloidal coils, and Table 3.4 shows positions and maximum ratings of poloidal coils. No.1-8 are superconducting coils, and No.9 and No.11 are coils of copper conductor, but type of No.10 is not decided. The choice between Nb_3Sn and NbTi is not fixed at this time. Copper conductor is employed at room temperature.

b. Where are coils? Inside or outside the toroidal field coils?

All poloidal coils are outside the toroidal field coils.

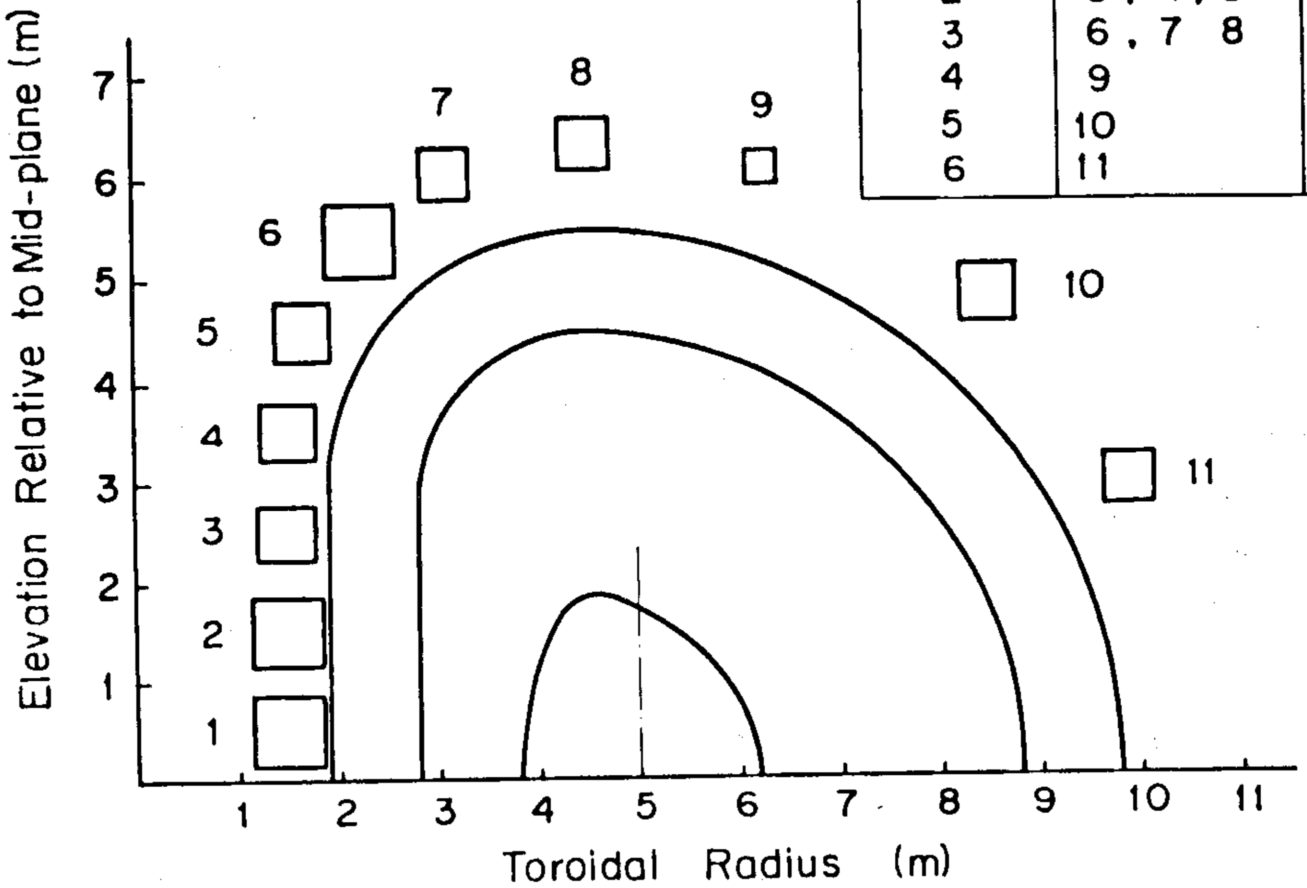
c. How to create and control the plasma? (Start, heating, working cycle and shutdown)

See other group's.

d. What are the realistic alternatives in the structure design? How does maintenance affect the design?

(1) Selection of superconducting material: NbTi or Nb_3Sn
Merits of NbTi: cycling mechanical load, easiness of wire production, low cost enough technical experience
Merits of Nb_3Sn: stabilization, margin of temperature rise, forced cooling system, higher magnetic field

(2) Selection of cooling system: pool boiling or forced cooling
Merits of pool boiling: stabilization for long time heat input, easiness of conductor production, simplicity and cost of cooling system
Merits of forced cooling: higher heat transfer coefficient to helium, rigidity of winding, easiness of coil winding



Block NO.	Coil NO.	Function
1	1 , 2	Hybrid
2	3 , 4 , 5	OH only
3	6 , 7 8	Hybrid
4	9	Hybrid
5	10	Hybrid
6	11	Hybrid

Fig. 3.4 INTOR Hybrid Poloidal Coil Locations

Table 3.4 INTOR Poloidal Field Coil Locations and Maximum Ratings

Coil No.	Block No.	R (M)	Z (M)	N #	Transformer Component (MAT)	Shaping Component (MAT)	Total (MAT)	Shaping Component (MAT)	Total (MAT)
1	1	1.5	0.5	1	-3.68	-2.34	-6.02	-1.95	-5.63
2	1	1.5	1.5	1	-3.68	-2.34	-6.02	-1.95	-5.63
3	2	1.5	2.5	1	-3.60	*****	-3.60	*****	-3.60
4	2	1.525	3.5	1	-3.60	*****	-3.60	*****	-3.60
5	2	1.675	4.5	1	-3.60	*****	-3.60	*****	-3.60
6	3	2.25	5.4	2	-1.72	10.0	8.28	8.0	6.28
7	3	3.1	6.05	1	-0.86	5.0	4.14	4.0	3.14
8	3	4.5	6.35	1	-0.86	5.0	4.14	4.0	3.14
9	4	6.25	6.1	1	-0.15	*****	-3.15	*****	-3.15
10	5	8.5	4.85	1	-0.10	-5.17	-5.27	-3.29	-3.39
11	6	9.9	3.0	1	-0.15	-1.98	-2.13	-2.87	-3.02
						with divertor		without divertor	

Plasma current I_p at the flat-top is 4.7 MA.

#.. N is a ratio of the coil turns in the same block.

(3) Selection of winding structure: layer or pancake winding
(in the case of pool boiling)
Merits of pancake winding: mechanical intensity for radial electromagnetic force, easy repairing of winding
Merits of layer winding: effective helium channels, easiness of grading

e. What heat loads (radiation, conduction, joule heating, etc.) are practical for the alternate designs?

Items of heat loads are as follows

(1) Losses generating in superconductors
  (1-1) hysteresis loss
  (1-2) coupling loss between superconducting filaments
  (1-3) coupling loss between superconducting strands
  (1-4) self field loss in superconducting filaments
  (1-5) eddy current loss in the material for reinforcement
  (1-6) eddy current loss in conductor matrix
(2) Neutron heating
(3) Eddy current loss in support and cryostat systems
(4) Heat leak from normal temperature
  (4-1) heat conduction
  (4-2) heat radiation

Next tables show the example of heat load in the case of JXFR.

Weight for initial cooldown

units[tons]

materials / parts	SUS	Cu	FRP
toroidal coils	2,988	1,207	99
poloidal coils	325	244	240
support structure	1,850	0	80
total	5,163	1,451	419

Heat load at the charge or discharge of toroidal coils

units[W]

parts	heat load
toroidal coils	158
power leads	107
total	265

Static heat load

units[w]

parts \ temperature	4.2K	80K
vacuum vessel	5,104	19,940
leggs of central support	1,086	4,543
total	6,190	24,483

Dynamical heat load at the each phase of poloidal coils

units [W]

parts \ phase		I 0.8sec	II 9.2sec	III 120sec	IV 10sec	V 20sec	VI 20sec
toroidal coils	coils	41,830	9,281	4	9,333	0	81
	power leads	322	322	322	322	322	322
poloidal coils	coils	353,789	6,983	294	8,647	0	3,992
	power leads	87	35	107	107	73	10
neutron heating		0	0	991	0	0	0
support structure		667,101	2,466	20	5,044	0	181
total [W]		1,063,129	19,087	1,738	23,453	395	4,586
amount of heating [J]		850,503	175,600	208,560	234,530	7,900	91,720

amount of heating for one period = 1,568,813 [J]

average heat load for one period = 8,715 [W]

f. How does the fault analysis affect the design of the magnet and support structure?

The answer is the same as mentioned in A.1.f.

2. What knowledge and experience is required to design, construct and operate the poloidal magnetic system of INTOR? Extend and add to these answers:

a. Study of optimal poloidal system
b. Development technology of construction of the poloidal coils inside and outside the toroidal coils
c. Study of mechanical and electrical behaviour of poloidal coils under heating loads, radiation fluence and pulsed fields
d. Safety analysis of the poloidal system

The following R & D are required :

a. Demonstration of operation of large and pulse coil with Nb_3Sn or NbTi
b. Establishment of a.c. loss analysis criteria in a large current pulse coil
c. Demonstration of large capacity He liquefier and refrigerator
d. Refined system of automatic operation of He liquefier and refrigerator (large capacity compressor, large capacity gas collection and supply system)
e. Establishment of design criteria on material fatigue by pulse field
f. Fabrication and winding technique of large conductor

3. List and analyze existing or expected data base and add further information:

JAERI is now planning a pulse poloidal coil system for the cluster test facility.

4. Group 7 Systems Integration and Support Structure*

Answers to Question No. 1 of Home Task are given in the report of Group 2.

In order to make sufficient answers tc the other questions a considerably detailed design study is required. In this sense we have carried out a preliminary study[1] based on the INTOR guiding parameters to provide materials for discussion.

Reference

1) K. Sako, et al. : "Engineering Aspects of the JAERI Proposal for INTOR (I)", JAERI-M 8503 (1979)

* Report for Session 2 of IAEA INTOR Workshop

5. Group 9 Assembly and Remote Maintenance*

(Edited by H.Iida)

A. Machine Assembly:

1. What machine layouts are most advantageous from the viewpoint of machine assembly? Why?

Machine layouts and schemes of reactor assembly should be as simple as possible. Since linkage of toroidal and poloidal coils makes them extreamly complex, it is not preferable if magnets should be repairable. Fig.5.1 shows the machine layout of JXFR which consists of eight reactor modules. Modular concept has the following advantages.

(1) sound machine integrity
(2) test of every components before installations into reactor possible.
(3) easier disassembly and reassembly.

2. What subassembly packages are best for ensuring timely assembly and orderly testing of reactor systems?

Each JXFR reactor module contains a pair of toroidal coils, primary shield, vacuum chamber, inner and outer blankets, a pair of vacuum pump units and neutral beam injector which can be easily dismounted from module. After setup of a complete module each components can be tested befor installation of the module into a reactor device.

3. What are reasonable placement tolerances for these subassemblies?

Requirements for controlling plasma shape and position, thermal deflection of each component and radiation streaming should be considered to determine the placement tolerances for each component.
If the gap is small at room temperature, blanket modules will contact each other during reactor operation and a large stress will be induced. The gap between blanket modules should be determined taking acount of thermal expansion during operation.

* Report for Session 2 of IAEA INTOR Workshop

On the other hand, if the gap is large, neutron streaming through the gap will unacceptably increase.

4. What are the requirements for assembly facilities?

Assembling of reactor requires the following facilities

(1) reactor module assembling room

(2) facility for test and inspection of reactor module

(3) transportation system of reactor module and its components.

Figure 5.2 shows JXFR reactor room and repair shop. Reactor can be assembled by using those facilities.

B. Remote Maintenance:

1. What components of the machine must be repairable?

In JXFR design following components have designed to be repairable or replaceable.

(1) protection wall

(2) blanket cells and primary cooling pipes

(3) vacuum chamber

(4) shielding

(5) neutral beam injector

(6) vacuum pumps

(7) toroidal field coils

(8) poloidal field coils

Above components should be considered to be repairable or replaceable also in INTOR design.

2. What components of the machine can be repaired in a reasonable time?

Neutral beam injectors, vacuum pumps and diagnostic instruments can be repaired or replaced in a short time. Repair or replacement of superconducting magnets requires rather long down time since in that case withdrawal of reactor modules is necessary and it takes long time to cool down magnets again. The down time for the maintenance of cooling tube panels and blanket cells strongly depends on the design, so it is very difficult to assess it at present.

3. What are the basic approach to remote maintenance?
What are the comparative advantages and disadvantages of the approaches?
How compatible are these approaches with reasonable capital cost?

Three basic approaches and their advantages and disadvantages are shown in Table 5.1 which is our revision of the attachment in group 9 minute of the last session. Approach A may not be adopted because of its large risk. The repair of device after D-T burning is almost impossible in case of approach A.
Possible approaches are B and C. Though modular concept (Approach C) requires large repair shop and transportation facility and may cost very high, it seems to be the most suitable approach in case of INTOR since it is the most reliable one. After the sufficient development of remote handling technique approach B can be suitable for tokamak maintenance.
Appendix A shows JXFR maintenance process and facilities required for it as a example of approach C.

Table 5.1 Maintenance Concept for Tokamaks

	Advantage	Disadvantage
Approach A Minimum size device	a. lowest initial cost. b. fewest number remote maintenance tasks.	a. repair inside device impossible. b. repair of TF- coils and PF-coils impossible. c. preventative maintenance difficult.
Approach B Clearance for in-situ maintenance	a. easy preventative maintenance. b. probably minimum down time.	a. larger TF-coils and shield required. b. repair of TF-coils and PF-coils impossible. c. more remote maintenance taskes.
Approach C Modular Concept	a. combines small size and good maintenability. b. in-situ maintenance minimized. c. most reliable	a. large repair shop and module transportation facility required. b. extra joint demounting required.

4. Which systems can be maintained or repaired remotely or semi-remotely and how?

All components highly activated should be so designed that they can be maintained or repaired remotely. In modular concept the following components should be repaired by remote handling.

(1) cooling tube panel
(2) blanket cell
(3) shield
(4) toroidal coil
(5) neutral beam injector
(6) vacuum pump

5. What are the implications of question 4 to overall machine down time? Are the problem of activation and remote maintenance compatible with the objectives of INTOR?

The development of highly reliable reactor components and equipments for quick remote maintenance are very important to shorten the overall machine down time. The problem of activation and remote maintenance is not easy to solve. It is necessary to carry out the design study of INTOR from the viewpoint of maintenance to point out the problematic items and R & D required for them.

6. What are the possible failure of the INTOR components, their probability and the consequences? Based on this analysis, what must be done to insure a 25% availability for INTOR? Is any development required?

Possible failures of the INTOR components are followings.

<u>Possible failures of blanket structure</u>

(1) Outer blanket	. leak of coolant at joints and elbows induced cyclic thermal stress
(2) Inner blanket	. leak of coolant at joints and elbows induced cyclic thermal stress

	. temperature increase due to separation of brazing material between cooling tube and block.
(3) Cooling panel	. leak of coolant

Possible failures of vacuum chamber

- o small leaks of coolant at bellows or of air at lip seal between modules.

Possible failures of shield

- o leak of coolant
- o damages of pipings and valves

Possible failures of superconducting magnet (toroidal and poloidal coil)

(see Group 5 Home Task)

- o damage of superconducting wires
- o damage of helium cooling channels
- o damage of insulators
- o damage of spacers
- o helium leak from the helium vessels
- o failure of rapture disk or safety valves
- o damage of cooling pipes
- o damage of magnet vacuum chamber
- o damage of power leaks
- o damage of cryostat
- o damage of coil support structure

Possible failures of nutral beam injector

(1) Ion sources	break of cathodes
	deformation of grids
	erosion of grids
	leak of coolant
(2) Ion collector (energy converter)	deformation of electrode
	elosion of electrode
	leak of coolant
(3) Pumping system	leak of coolant
	failure of vacuum valves
(4) Neutralizing cell and others	leak of coolant

Possible failures of vacuum pump

(1) cryopump	failure of gate valve
	failure of pipings and other valves
	failure and degradation of cryopanel
(2) back up and auxiliary pump	failure of T.M.P.
	failure of R.P.
	failure of pipings and valves
(3) other components	failure of tritium recovery system.

Possible failures of diagnostic

(See Group 16 Home Task)

At present, it is very difficult to assess the probabilities of above failures.

7. Where does components redundancy make sense for INTOR and where is it less effective or detrimental?

In case of modular concept, at least one reactor module must be prepared as a standby in order to decrease down time in case of failure. Ion source in neutral beam injector will be replaced very often because of its short lifetime. Many spares will be required for ion source. Other components which seem to fail frequently require redundancy.

8. Does remote maintenance require the development of specific new hardware such as vacuum seals, coolant or electrical connectors?

In modular concept reactor maintenance by remote handling requires development of the following equipment.

(1) Manipulator
More elaborate manipulator with many degree of freedom is necessary, and it must be small-sized and remote-controlled.

(2) Auto cutter/welder
The device for cutting and welding the vacuum seal lip between shield walls of modules.

(3) TV monitor

This system is used for watching the operation and inspection in high level radioactivity.

(4) Weld test

Establishment of nondestructive testing for welding of thin austenitic stainless steel plate

(5) Metal O-ring

Long lifetime for sealing helium gas at temperature of 300 - 500° C.

(6) Auto piping welder

Coolant pipings of blanket or cooling tube panel have to be completely welded without helium leak in the circumstance of high gamma dose level.

These equipments and methods are possible theoretically or conceptually. Some of them have succeeded experimentally in a simple form. Research and development aimed at actual size will be required immediately.

9. In the event of component failure within the reactor, what backup procedures might allow continued operation of the device with minimum impact on experimental objectives?

Any backup procedure does not seem to be able to allow continued operation of the device.

10. What position of the vacuum vessel is best for overall maintenance?

Inner surface of reactor room may be the best position of the vacuum vessel from a viewpoint of reactor overhaul since it is not necessary to break vacuum in case of disassembly. However great many complex equipments are contained in the vacuum vessel, so reliability of reactor will be decreased. Inner surface of shield may be the best position for overall maintenance.

(Toshiba Co. Ltd., Mitsubishi Atomic Power Industry Inc., Sumitomo Heavy Industries Ltd., Kawasaki Heavy Industries Ltd.)

Appendix A JXFR maintenance process and facilities required

(Sumitomo Heavy Industries Ltd.)

A.1
General description of overhaul procedure

The basic policy of the overhauling method is that a damaged module of the reactor is exchanged for a new or repaired one.
On the other hand, the damaged module is transported to the repairing shop beside the reactor room, then it is repaired and stored.

Each module is installed on a carriage which can be movable in radial direction.
Eight shield doors, in order to pull out the module, are furnished at the wall of the reactor room.
Circular railways are set along the outside of the reactor room, and a "rotating carriage" moves on the rail.
A damaged module pulled out to the outside through a shield door is mounted on the rotating carriage.

The rotating carriage transports the module in front of a repair shop which is laid out just beside the railway. The module is drawn into the repair shop, and a complete module is sent out from the shop instead of the damaged one.
The complete module is sent into the reactor room by the reverse order of the damaged one.

This process is effective for the repair of blanket, toroidal coil and shielding vessel.
The overhauling procedure for the blanket is shown in section 7.5.
As it is almost impossible to repair the toroidal coil, the toroidal coil must be exchanged for new one, if it is damaged. Damaged coil is transfered with the module (One module has 2 toroidal coils).
A NBI is dismounted from the module and then transfered independently.
We think that the shielding vessel is not almost damaged. But if it is damaged, it is similarly exchanged for new one. Replacement of the central poloidal coil is very troublesome.

In order to take out the central poloidal coil, each module are moved outwards about 2 m.
The central poloidal coil is put on the carriage by the polar crane and moved to the repairing shop by the foregoing order.

It is disassembled and the damaged part is exchanged for new one.
It is impossible to exchange the upper poloidal coils for new ones, because there is not wide entrance to bring the new coils.
If necessary, regarding the lower poloidal coils, spare coils are prepared in the pit.

A.2
Facilities for overhaul

Facilities for the repair and maintenance of the reactor are reactor module mounting carriages, the rotating carriage, overhead cranes, the auto welder/cutter, shield doors, remote handling devices, etc.
The followings are out conceptual designs of these facilities.

1. Reactor module mounting carriage

A segment mounting carriage consists of four parts (A, B, C, D).
The outline is shown in Fig. A.1.
For transporting the damaged module, a NBI (D carriage) will be dismounted from the reactor, and the carriage connected one another (A, B, C) should be drawned outwardly (Fig. A.2).
One module weighs about 3000 tons without NBI.
The driving equipment consists of the speed reduction gear system and 11 KW motor (Fig. A.3).
The carriage can travel at the rate of 20 mm/s.

2. Rotating carriage

The rotating carriage will be used for transporting the reactor module from the reactor room to the repair shop.
A large shield box mounted on the rotating carriage contains a damaged module and moves on the circular railway along the outside of the reactor room. The motor powers for the outside and inside circumferential drives are 110 KW and 55 KW respectively, and velocities are about 48 mm/s and 27 mm/s.
This carriage can revolve semicircle at an hour.
Radial movement is also necessary to put the reactor module into the shield box (see Fig. A.4). This can be accomplished by rollers mounted on the rotating carriage. Driving power required for rollers is 5.5 KW.

Air-bearing method might be useful as a driving system (refer to Fig. A.5) instead of above carriage method. There is no actual experience of carring such extreme heavy object as about 17,000 tons, so the carriage used wheels has been adopted.

3. Overhead cranes

Several kinds of overhead cranes should be furnished in the reactor room, the room of the rotating carriage and the repair shop.
The polar crane installed in the reactor room is mainly used to carry the central poloidal coil, the upper and lower poloidal coils.
The rotating overhead crane installed in the room of the rotating carriage can travel along the outside of the reactor room and is useful mainly for maintenance of the rotating carriage. The repair shop provides various capacity of the overhead cranes.

4. Auto welder/cutter

In JXFR, the vacuum wall is not the first wall but installed just inside the primary shield.
The conceptual design of the auto welder/cutter is shown in Fig. A.6.
Two thin stainless steel flanges are pressed by means of the rollers and the ends of the sealing plates are cut off to aim at complete welding.
TIG arc welding is done along the D shaped flange (see Fig.A.7).
In order to keep stable travelling, guide rollers are necessary. Total weight of the equipment is about 100 kg and power is 400 W.
Welding current is 100 - 150 A. Welding speed is about 200 mm/min.
Welding speed is adjusted by the program control.
Welding point is controlled to be right on the end of the sealing flange by means of the profiling equipment which uses differential transformer.
State-of-the-art technology is applied to these control system.

5. Shield door

The shield door is so heavy that movable equipment become a large size.
The shield doors of the reactor room should move up-and-down, while others are opened and shut sideways.
Therefore the winch type is adopted for the shield doors of the reactor room. For the others, hydraulic type, rack and pinion type, winch type are applied.

6. Remote control equipments

For overhauling and assembling the reactor, various operations such as fastening and loosening bolts, connecting and disconnecting electric wires, welding and cutting pipes, transporting heavy objects, various kinds of inspections are required. These works should be performed by the remote control.

The equipments described in Section 1 through 5 above are remotely controlled, too.

Appendix B Fabrication Technique Required for INTOR Blanket

(Kawasaki Heavy Industries Ltd.)

INTOR structures include many complicated components where difficulties are encountered in the fabrication. Therefore, some of the most problematical fabrication techniques required are herein picked up and explained in brief.

(1) Welding of Outer Nozzle and Connecting Pipe

This welding is to be carried out in all positions at the workshop. Furthermore, the gap between outer nozzle and inner one is strictly limited and the integrity of the welds must be assured. From this standpoint, automatic orbital TIG welding process is employed by means of controlling system of welding conditions of each welding position.

(2) Welding of Cooling Tube and Stub

Both cooling tube panel and inner blanket are small sized in the outside diameter and closely spaced. Therefore, welding of cooling tube and stub should be automatically carried out without filler metal. Meanwhile, the welds of these small-sized tubes are apt to accompany concavity on the outside surface resulting from the relation between the gravity of molten metal and its surface tension.

In this sense, automatic orbital TIG welding should be conducted by controlling device of shield gas pressure which can give slight pressure difference between inside and outside of the tube.

(3) Welding of Non-destruction Testing of Coolant Pipe

In the welding of coolant pipe, accessibility and vigibility during welding is not insured especially at the site. Furthere, the best quality of the welds must be assured. Therefore, welding performance should be remotely controlled and its non-destructive testing is to be automatically carried out either.

(4) Brazing of Inner Blanket Tube

Inner blanket tube is brazed to enhance the conductivity between blanket tube and block. This brazing is automatically performed by the electric furnace which is held at the temperature above both sensitizing temperature range of stainless steel and melting point of brazing material.

2. Critical Points to be Analyzed

All the welds of INTOR structures must withstand the severe service conditions. The quality of these welds must be assured perfectly. As mentioned above, non-destructive testing must be automatically performed at the site for some of the welds of coolant pipe. However, the technique of non-destructive testing for these coolant pipe welds leaves some difficulties. Therefore, these welds can be regarded as one of the most critical points in the structures.

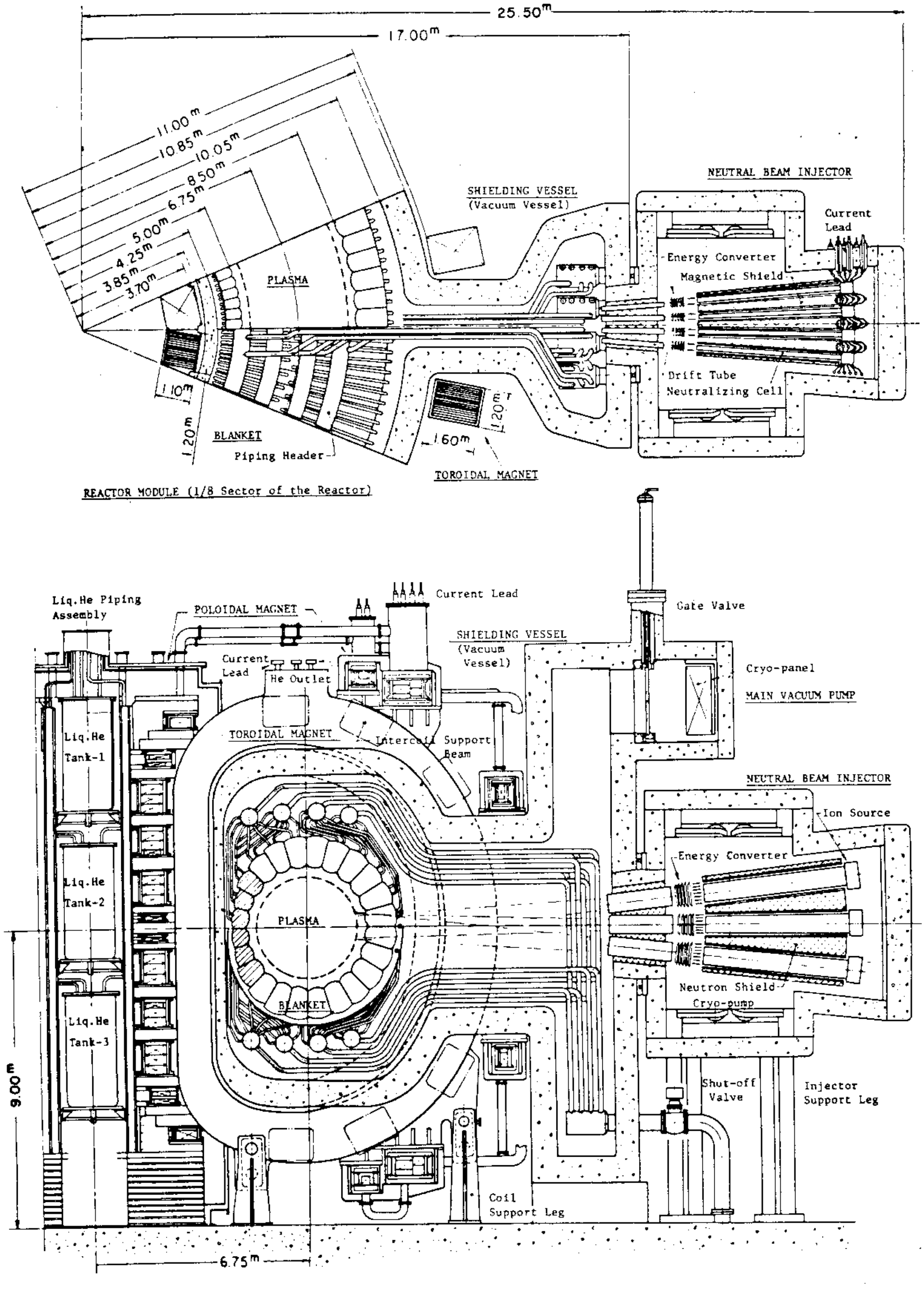


Fig.5.1 Machine layout of JXFR

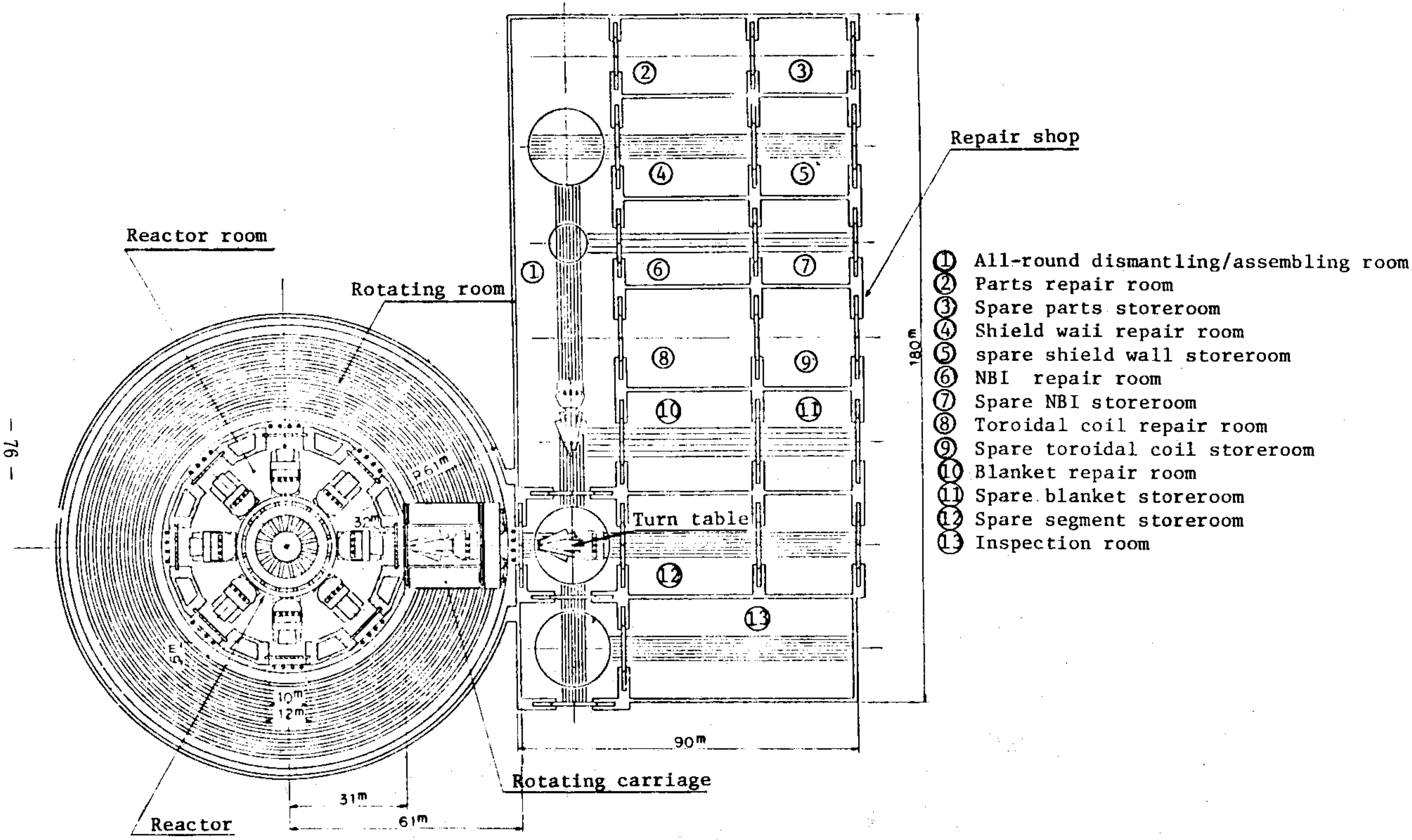


Fig. 5.2 JXFR reactor room and repair shop

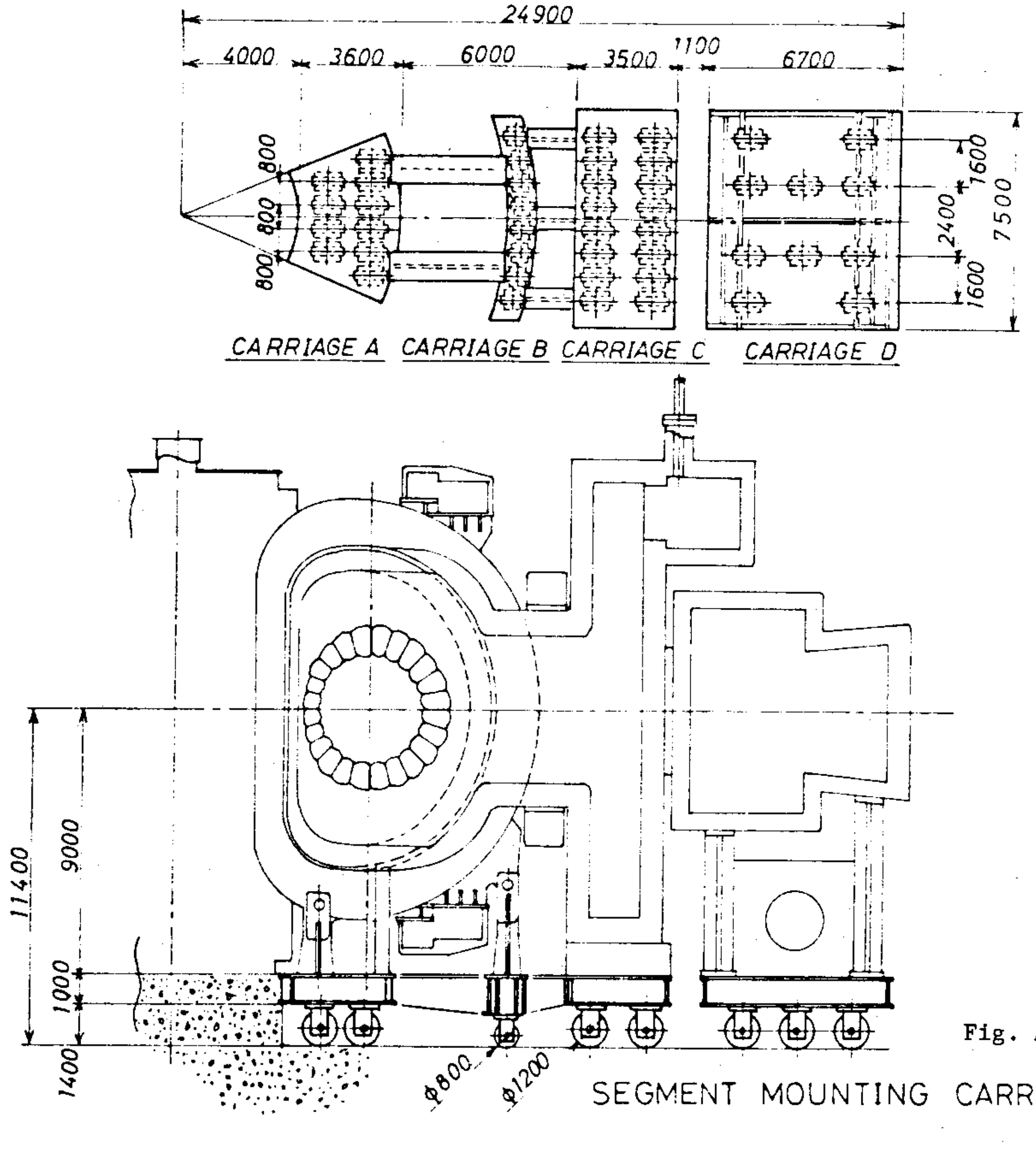


Fig. A.1

SEGMENT MOUNTING CARRIAGE

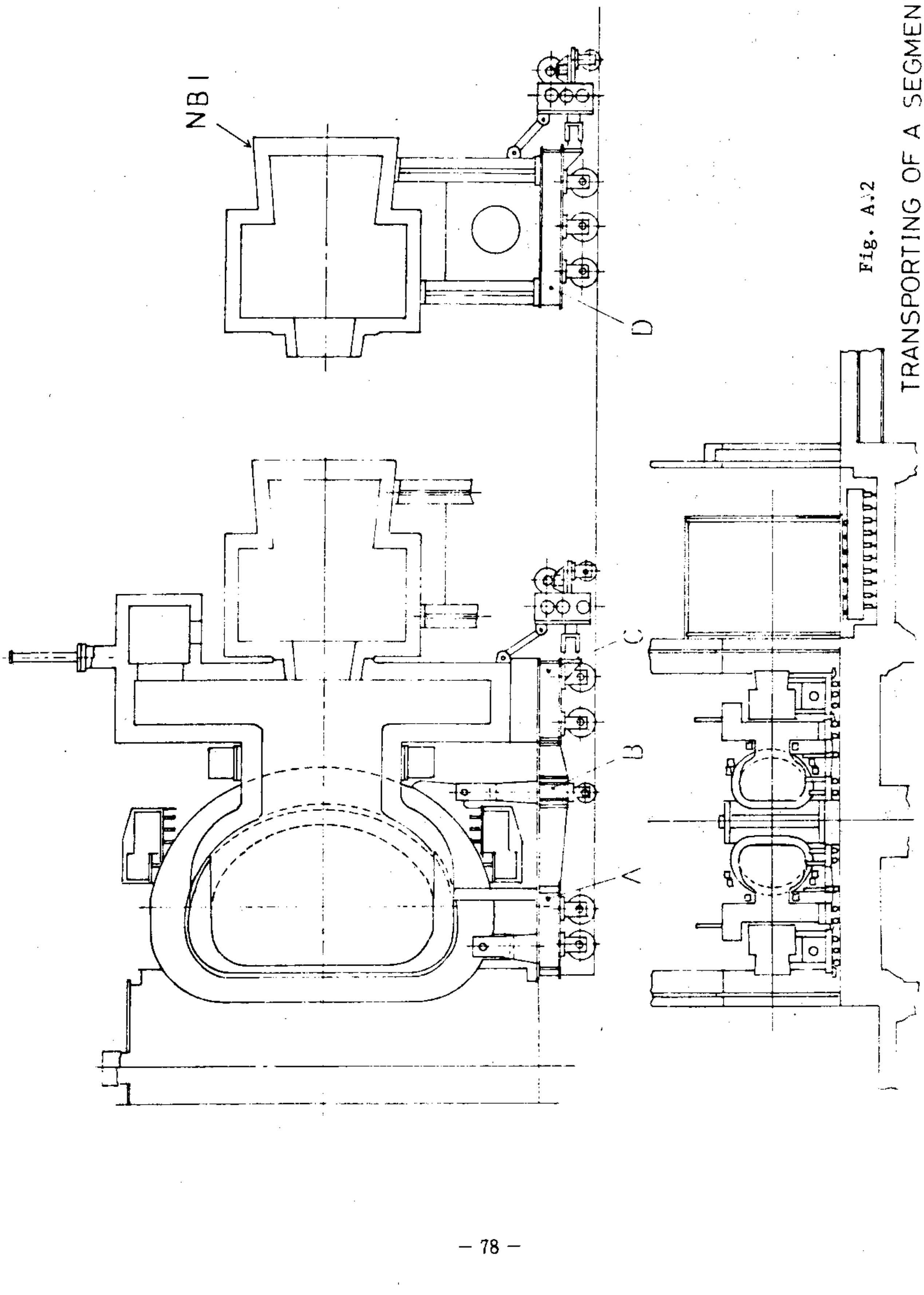


Fig. A.2 TRANSPORTING OF A SEGMENT

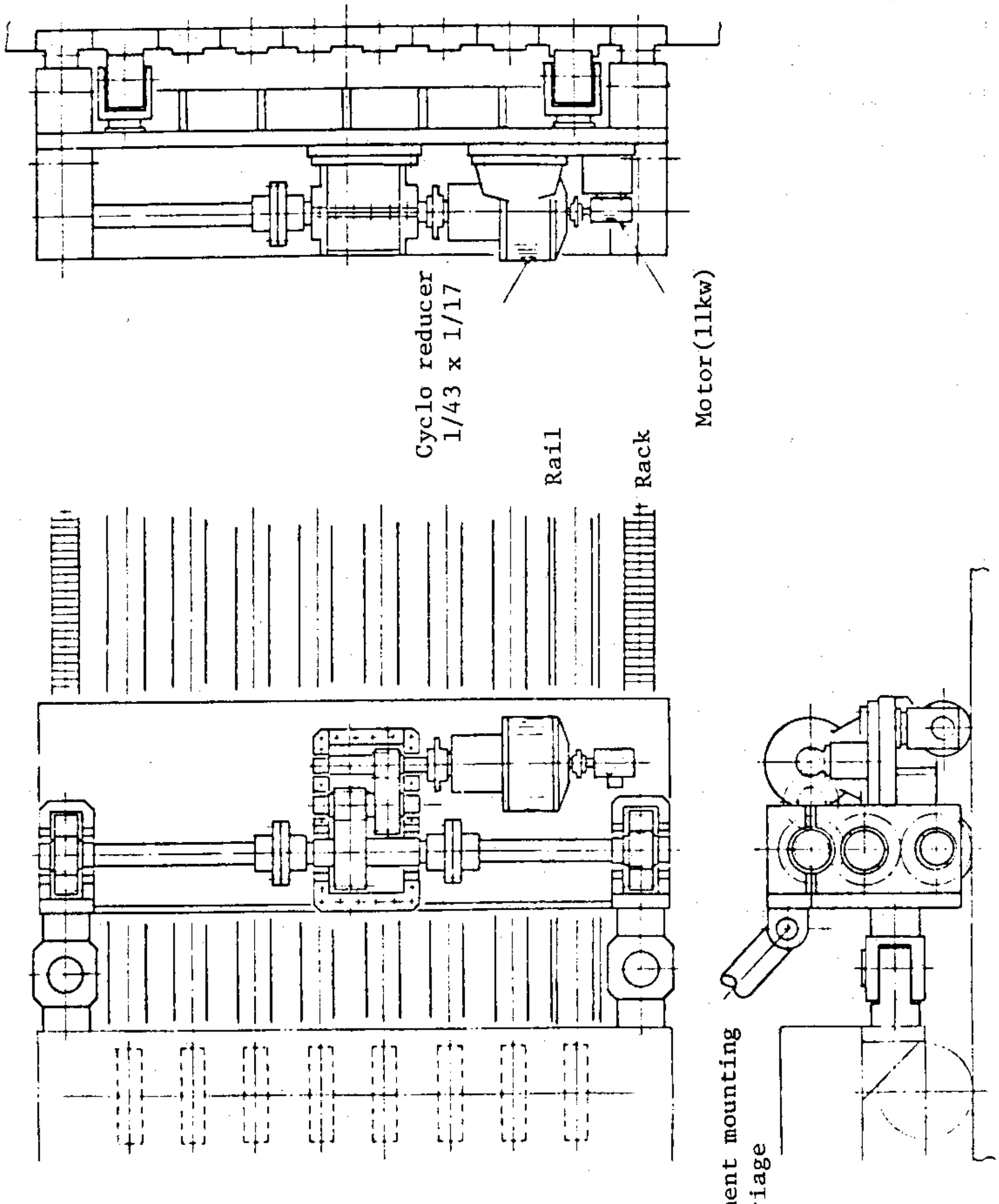


Fig. A.3 Driving equipment for a segment mounting carriage

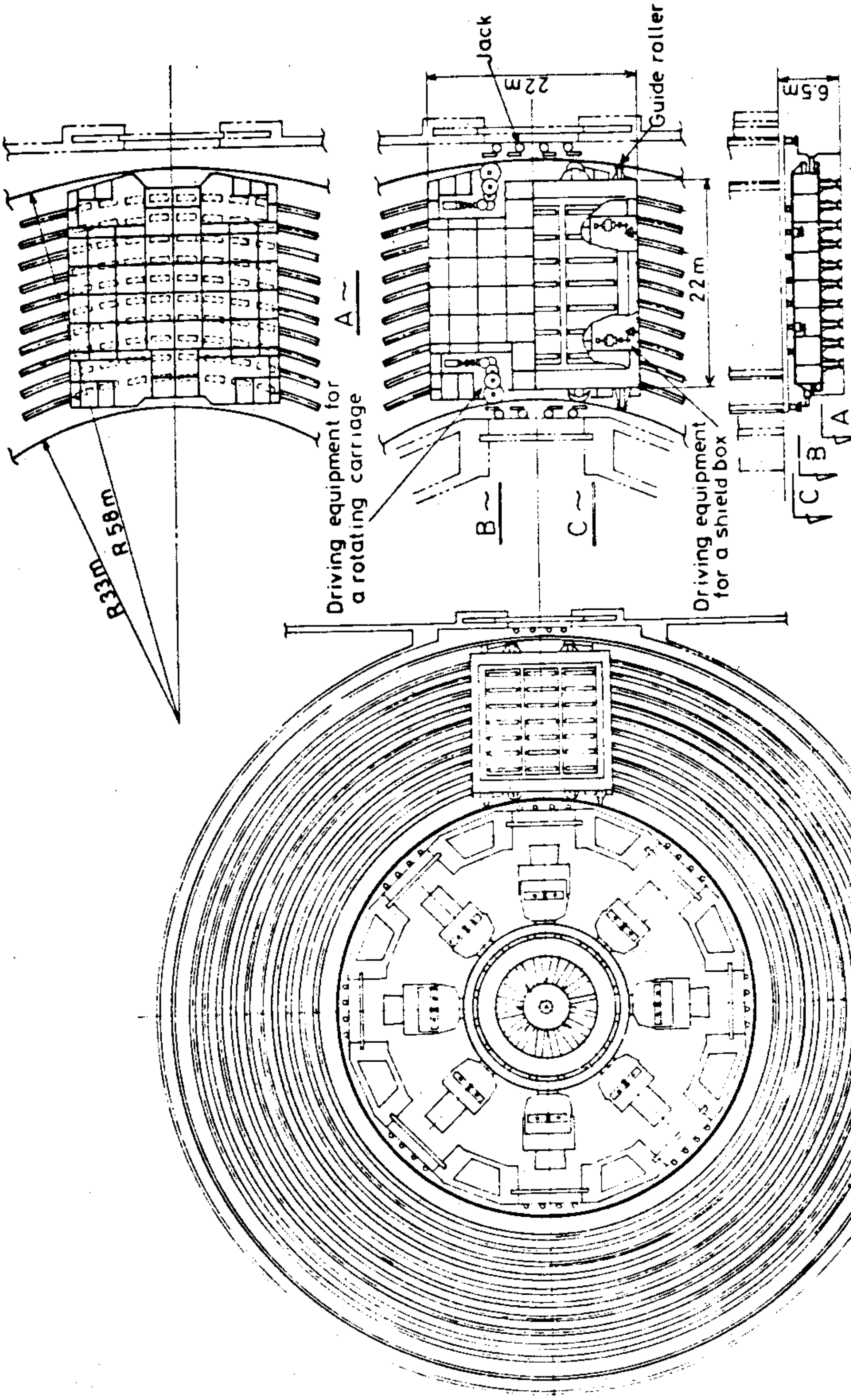


Fig. A.4 ROTATING CARRIAGE (WHEEL TYPE)

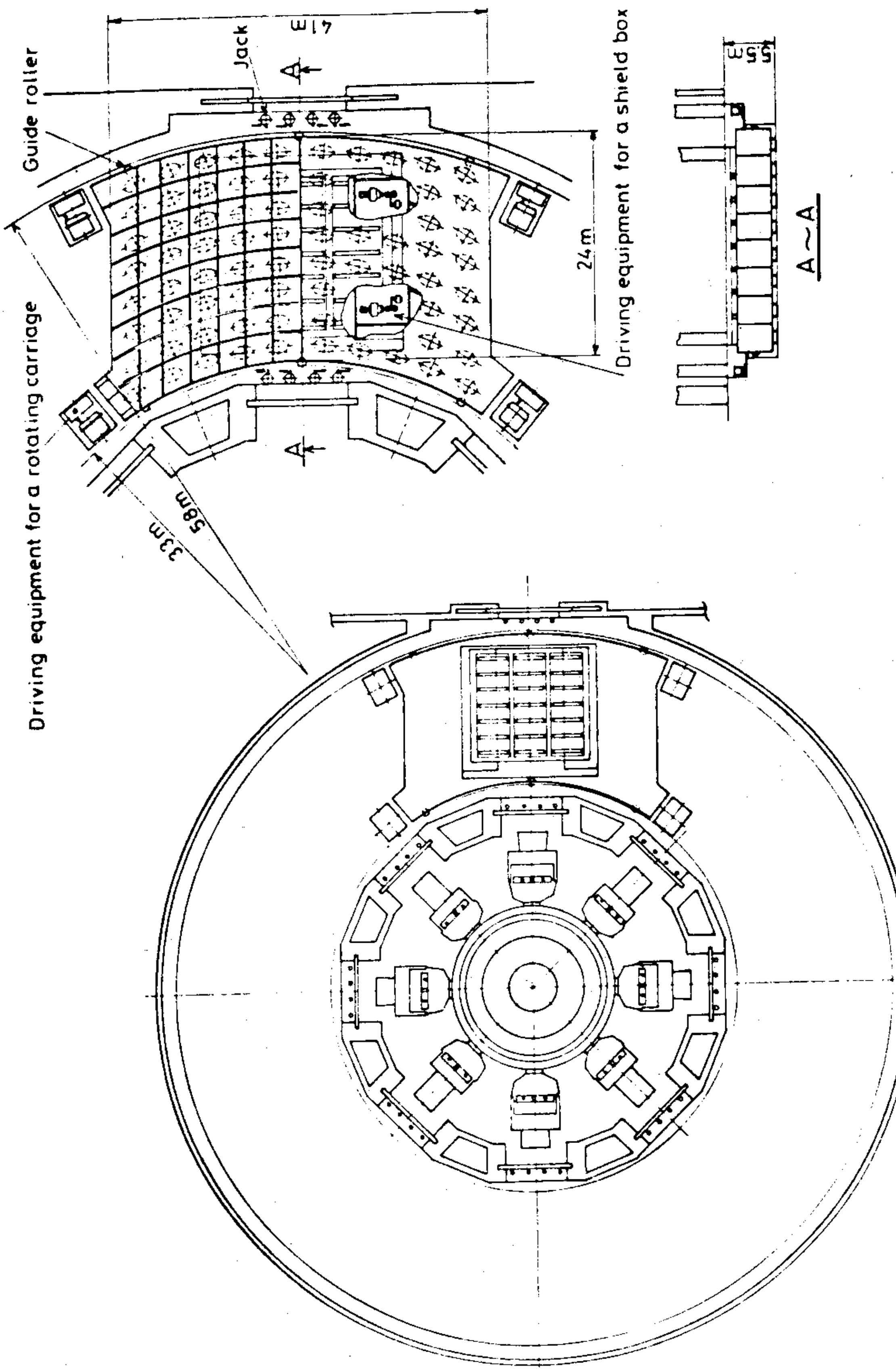


Fig. A.5 ROTATING CARRIAGE (AIR BEARING TYPE)

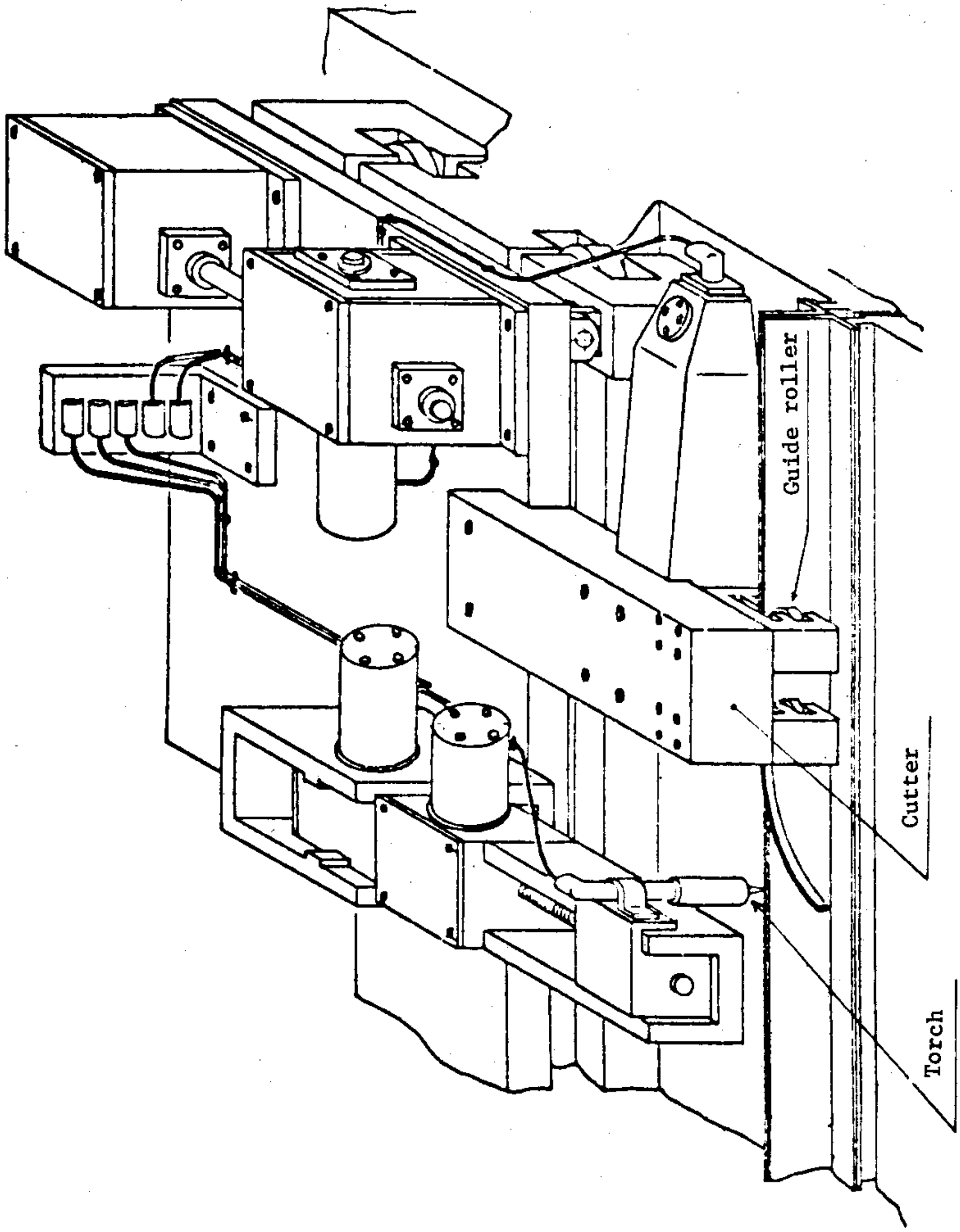


Fig. A.6 Auto welder/cutter

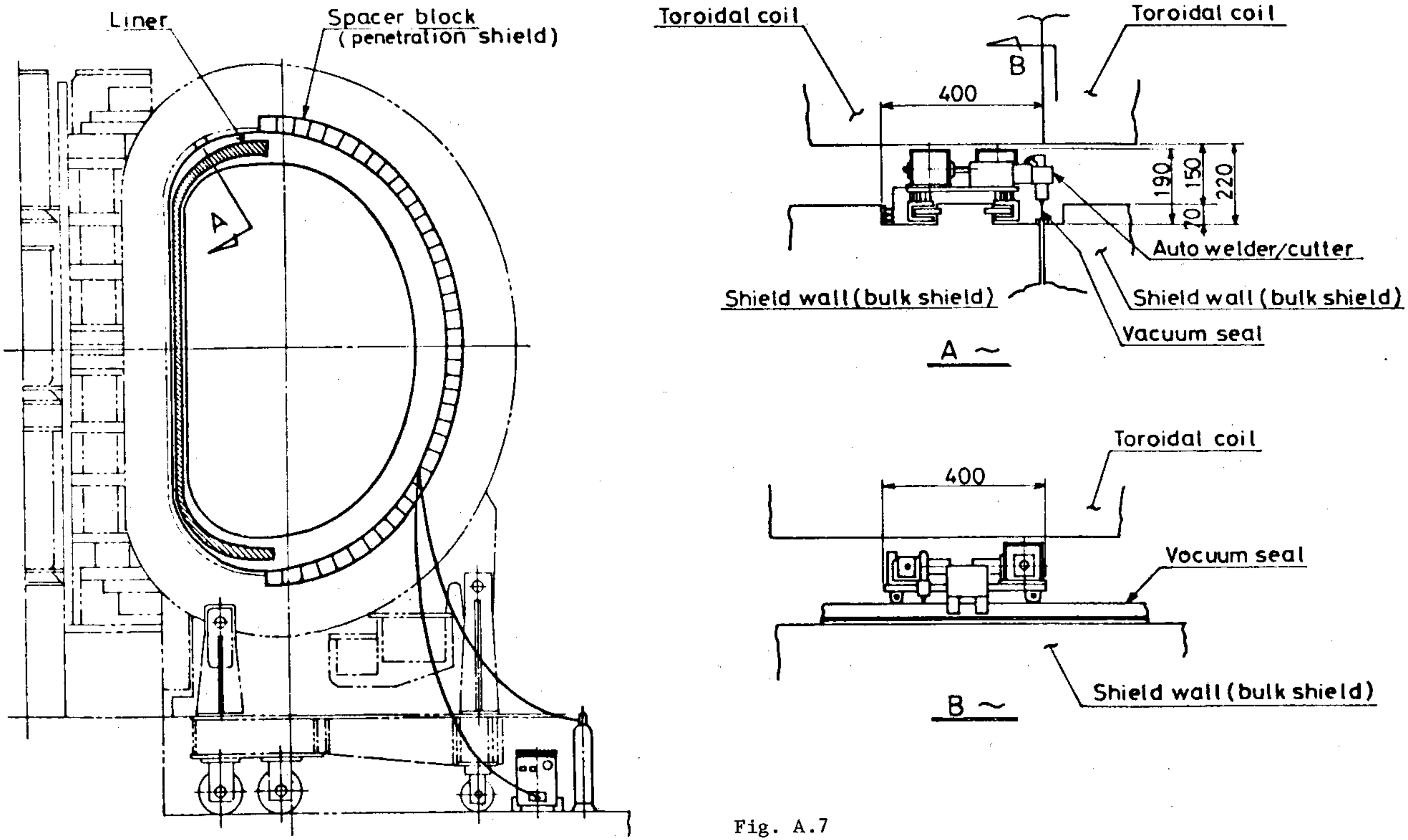


Fig. A.7

INSTALLING FOR AN AUTO WELDER/CUTTER

6. Group 10 Radiation Shielding and Personnel Accessibility*

(Edited by Y. Seki)

Introduction

Radiation shielding and personnel accessibility are closely related. In order to allow personnel access, radiation shielding must be provided.

In most tokamak reactor designs, shielding consists of a primary shield placed between the plasma and toroidal field coils, and a secondary biological shield which also serves as a reactor building. When the reactor is in operation, personnel access is only permitted outside the secondary shield. Whether to permit personnel access inside the secondary shield after sufficient cooling period depends on the design philosophy. Personnel access to the reactor will not be required if all the necessary repair and maintenance works on the reactor can be done by remote handling. In view of the present status of remote handling technology, direct personnel access to the reactor seems to be indispensable.

Admitting the necessity of personnel access, the requirements for the physical locations and time duration for the access must be evaluated in detail. Radiation shielding must be provided to sufficiently reduce the induced activation of reactor components to enable personnel access where necessary. Requirements for the primary shield are as follows:

1. To protect superconducting magnets and other radiation sensitive components from the damage caused by neutron and gamma ray radiation.
2. To prevent the activation of components outside the primary shield so as to reduce biological dose rate low enough, thus allowing sufficient personnel access during reactor shut-down.
3. To serve as a part of the biological shield during reactor operation.

Generally, the second requirement is most severe and the thickness and composition of the primary shield on the outside of the torus is determined by the second requirement in some designs. On the other hand, the primary shield on the inside of the torus is generally designed to satisfy only the first requirement, resulting in a thinner shield.

Some studies have indicated that minimizing the shield thickness on the inside of the torus is very effective in reducing the total reactor cost. In carrying out such a study, the thickness and composi-

* Report for Session 2 of IAEA INTOR Workshop.

tion of the shield must be changed within the thermal and structural limitations of the shield.

Tokamak reactors have complex geometry with many penetrations. The accurate evaluation of induced activity dose around various components of the reactor in various location in the reactor building is a difficult task for the present day fusion reactor neutronics. One of the main problems in the shielding design calculation is the lack of a method to solve deep penetration problems under complex geometry. Automated induced activity dose calculation code system which can treat 3 D model is also missing. As regards the nuclear data, more accurate data for the anisotropy of secondary neutrons, activation reaction cross sections and decay schemes are required.

In the following, the answer to the Group 10 Home Task is given. A result of a one-dimensionel shielding calculation for the INTOR proposal and the R & D schedule for fusion reactor neutronics in Japan Atomic Energy Research Institute is appended.

1. Direct personnel access in the reactor building after shutdown

a. Importance of direct personnel access

In view of the present status of remote handling technology, direct personnel access into the reactor building is necessary.

b. Feasibility of direct personnel access

The access is feasible for most of the reactor components outside the primary shield but access to those components placed near the plasma will be nearly impossible and to those connected to plasma via penetrations will be limited.

c. Shielding requirements for direct personnel access

Shielding should be provided to enable direct personnel access where necessary.

d. Expected limitations on the physical locations and time duration for the presence of a maintenance person

Physical locations	Time duration (after one week cooling period)
1st wall, blanket	Not allowed
outside bulk shield	
far from penetrations	~300 hours / 3 month, ~500 hours/year
near penetrations*	~ 10 hours / 3 month
NBI, instrumentations	~ 3 hours / 3 month

* with supplemental lead shield added.

2. Routine maintenance or replacement

a. Related components

Neutral beam injector (ion source; electrodes)

Instrumentation (lens and mirror)

Vacuum pumps, cryogenic system

b. Maintenance procedure for the above components, including time schedule

Difficult to estimate without detailed design. Maintenance procedure takes hours. Required activity cooling down period will be about a week.

3. Unscheduled failure

a. Related components

Blanket cooling system (pipings, first wall)

Superconducting magnets (SCM)

Vacuum vessel wall

b. Maintenance procedures

Requires more than a month if disassembly of the sectors (modules) are required.

4. Radiation exposure and sources for personnel

a. Maximum admissible radiation exposures for workers and the public

No reason to change ICRP recommendation, namely

1. dose rate (rem/hr, person)

3 rem / 3 month for workers

500 mrem / year for the public

2. cumulative doses (rem/person)

D = 5(N-18)(N; age) for a worker

3. cumulative dose to the workforce (man-rem/y)

D = 5 × M (M; number of workers older than 18)

b. Sources of occupational exposure during operation (components, air, coolant, corrosion products, etc.)

plasma ---------- $\frac{316\ \text{MW}}{14.1 \times 1.6 \times 10^{-19}} \sim 1.4 \times 10^{20}\ \text{n}\cdot\text{s}^{-1}$ (14 MeV)

air ------------- ^{41}Ar, 2×10^{-7} μC/cc

coolant --------- helium will not be activated

corrosion products ---------- not accounted at present

c. Sources of radiation exposure in the maintenance work after shutdown

At one week after shutdown (estimates based on JXFR(1))

1st wall and blanket structure ----- $\sim 10^{8}$ rem/hr

outside bulk shield ---------------- ~10 mrem/hr

NBI, instrumentations -------------- ~ 1 rem/hr

air -------------------------------- negligible

corrosion products ----------------- not accounted

(~ means "in the order of")

5. Radiation protection criteria for each component

It is unpracticable to decide radiation protection criteria for each component without their detailed designs. However, as a first step guiding parameter, the following criteria are proposed for INTOR design. They are subject to change as the design progresses.

a. Radiation protection criteria for

SC magnets	
maximum nuclear heating	0.001 Watt/cc
maximum neutron fluence	2×10^{18} n/cm^2
maximum copper DPA/yr	10^{-4} DPA/yr
maximum epoxy dose	2.4×10^{9} rad
NBI components	
maximum inorganic insulator neutron fluence	5×10^{19} n/cm^2 (En > 0.1 MeV)
maximum induced radiation dose	~1 rem/hr
vacuum pump	
total nuclear heating	~5 kW
diagnostic instrumentations	
maximum induced radiation dose	~1 rem/hr
radiation damage	n.a.
nuclear heating	n.a.

b. Radiation dose for the above components

SC magnets (based on preliminary 1D shielding calculation, Appendix A.)	
maximum nuclear heating	1.2×10^{-4} W/cc
maximum neutron fluence	1.1×10^{18} n/cm^2
maximum copper DPA/yr	4.3×10^{-5} DPA/yr
maximum epoxy dose	1.2×10^{9} rad
NBI components	
maximum inorganic insulator neutron fluence	not accounted
maximum induced activity of the copper charge collector at one week after shutdown	~10^{-2} Ci/cm^3 (2)
Stainless steel piping near the NBI port	~10^{-1} Ci/cm^3 (2)

(estimates for a demonstration reactor from Ref.(2))

Vacuum pump

total nuclear heating in

main vacuum pump	640 W
NBI cryopump	120 W

(estimates for a demonstration reactor from Ref.(3))

Diagnostic instrumentation

Induced activity has not been accounted but the problem seems to be very severe judging from the results in Ref.(4)

6. Radiation Streaming

a. Location and size of penetrations for JXFR[(1)]

		cross section	length	number
NBI	thru blanket	120 cm$^{\phi}$	× 85 cm	4
	drift tube	16.5 × 60 cm	× 200 cm	12 × 8
	neutralizing cell	14 × 60 cm	× 450 cm	12 × 8
vacuum pump	thru blanket	120 cm$^{\phi}$	× 85 cm	----- 8*
	vacuum duct	130 × 350	× 360	
		250 × 160	× 70	
		500 × 250	× 200	

* 4 of the 8 ports serve dual purposes for NBI and vacuum pumping.

diagnostic instrumentation		
laser scattering	6~24 cm$^{\phi}$ × 750 cm	8
micro wave interferrometer	~5 cm$^{\phi}$ × 750 cm	8
coolant piping thru shield	~100 cm$^{\phi}$	8
gap between blanket vessel double wall	~1 cm	
gap in shield between adjacent sectors (modules)	~1 cm	

Location and size of penetrations for INTOR

	location	cross section × length	number
NBI	thru blanket		
	drift tube		
	neutralizing cell		

vacuum pump duct — thru blanket		6*

* 3 of the 6 ports serve dual purposes for NBI and vacuum pumping.

diagnostic instrumentation		
laser scattering	(6~24) cm^{ϕ} ×	8
micro wave interferrometer	~5 cm^{ϕ} ×	8
coolant pipings thru shield	~100 cm^{ϕ}	6
gap between the double wall of blanket vessels	~1 cm	
gap between adjacent sectors (modules)	~1 cm	

b. Protection methods

1. Increase the thickness of shield materials near the penetrations.
2. Add supplemental fixed shield where necessary.
3. Add supplemental removable shields when personnel access is required.
4. Develop remote handling methods.
5. Devise maintenance scheme minimizing personnel access.

7. Radioactive inventory

a. Radioactive inventory of all components

See Ref.(5) for the case of JXFR

b. Required radwaste system

If tritium can be removed from activated structural materials, the storage of solid radwaste should be relatively easy.

REFERENCES

(1) K. Sako, T. Tone, Y. Seki, H. Iida, H. Yamato et al., "First Preliminary Design of an Experimental Fusion Reactor," JAERI-M-7300 (1977) (in Japanese)

(2) T. Ide, Y. Seki and H. Iida, "Effects of Neutron Streaming Through Injection Ports on Neutronic Characteristics of a Fusion Reactor," Proc. Topl. Mtg. Technology of Controlled Nuclear Fusion, CONF-760935-P2, (1976) 395

(3) H. Iida, T. Ide and Y. Seki, "Nuclear Heat Deposition in Cryosorption Pumps of a Fusion Reactor," Proc. 7th Symposium on Engineering Problems of Fusion Research, Vol.II, 1658 Knoxville, October 25-28, 1977 (1977)

(4) L.C. Pwu, V.C. Baker, "Neutron Streaming Study for the Diagnostic Penetrations of the TFTR," Trans. Am. Nucl. Soc., Vol27, 798 (1977)

(5) H. Iida and Y. Seki, "Induced Activities of Fusion Reactor Structural Materials," JAERI-M (7582) Japan Atomic Energy Research Institute (1978) (in Japanese)

Appendix A. One-Dimensional Bulk Shielding Calculation for the INTOR Proposal

As the first step of the shielding design for the INTOR proposal, one-dimensional bulk shielding calculations are carried out. An optimization study of the shield material on the inside of the torus was performed to minimize the displacement damage of the copper stabilizer of the superconducting toroidal field magnet. On the other hand, the thickness and the constituent materials of the blanket on the outside of the torus were determined mostly from the requirements for tritium breeding and from the results of a thermal and structural analysis. The outer shield configuration was determined by the criteria of allowing personnel access to the outside of the shield. The result of the optimization study of the inner shield is described in Section 1 and overall result of the shielding calculations in Section 2. A summary is given in Section 3.

A.1 Minimum Shield Thickness

In the design of JAERI Experimental Fusion Reactor (JXFR)(1), an attempt was made to minimize the thickness of the inner shield(2) under the following three limitations; (1) Inner blanket will be cooled by helium gas and hence about 10% of the volume must be left for the coolant path. (2) Stainless steel structure will occupy at least 17% of the volume of the inner blanket. (3) Heavy concrete will be the main structural material for the inner shield.

As a result, the configuration shown in Table A.1 was selected for the inner blanket and shield of JXFR. The total thickness of the bulk blanket and shield at mid-plane was 84.5 cm. It was found that stainless steel was effective in attenuating 14 MeV neutrons and that tungsten was excellent as the shield material for the fast neutrons with energies greater than 1 MeV.

Based on the result of the JXFR shielding design, an optimization of the inner shield material for INTOR was carried out. The limitations are as follows.

(1) The thickness of the blanket is 31.5 cm.

(2) The thickness of the shield is 40 cm.

(3) The coolant for the blanket was supposed to be helium in the beginning of the design but was changed to water at the later stage.

(4) Stainless steel is to be used as the structural material occupying more than 30% of the volume of both the blanket and shield.

(5) The displacement damage in copper stabilizer of the superconducting toroidal field magnet is the quantity to be minimized.

The choice of copper displacement damage as the minimizing parameter arises from the fact that it was the critical quantity in the JXFR shielding design, and that the resistivity increase of copper by irradiation is better known experimentally than the critical dose for organic insulator, which is another important quantity. The maximum copper DPA-rate in the toroidal coil for five configurations are shown in Table A.2. Among the first four cases with helium-cooled blanket, Case 3 with tungsten (W) volume ratio of 50% yielded the lowest DPA-rate. The volume ratio of 40% for tungsten was considered better because it uses less tungsten (which is quite expensive) with little increase in the DPA-rate (Case 4). By changing the blanket coolant from helium to water, further decrease of the DPA-rate was obtained as shown for Case 5, resulting in the value of 4.3×10^{-5} DPA per year under 25% availability condition.

This means the toroidal field coil can be operated for more than two years without room temperature annealing under the assumed design condition of the magnet.

The composition of the inner blanket and shield adopted for further investigation is shown in Table A.3.

A.2 Shielding Calculations and Their Results

The $P_5 - S_8$ approximation was used in the calculation of neutron and gamma-ray fluxes using the one-dimensional discrete ordinates transport code ANISN[3]. A 42 group-neutron, 21 group-gamma coupled cross-section library for 40 nuclides GICX40[4] was used. The nuclear data base of the GICX40 Library is ENDF/B-III and IV[5] for neutron data and POPOP4 Library[6] for gamma-ray production data. Cylindrical calculational models for the inner and outer shield are given in Tables A.4 and A.5, respectively.

Table A.6 summarizes the tentative criteria and the results obtained from one-dimensional shielding calculations for the inner and outer superconducting magnets (SCM) of the INTOR proposal. Cumulative fluences are calculated assuming 25% availability and 10 years lifetime. Copper is supposed to be annealed to room temperature once every year.

Figures A.1 and A.2 show the neutron and gamma-ray fluxes for the inner and outer models, respectively. Nuclear heating rates in the inner and outer models are depicted in Figures A.3 and A.4.

A.3 Summary

As shown in Table A.6, the inner model of Table A.4 with the total bulk thickness of 71.5 cm satisfies all of the tentative criteria adopted for the SCM irradiation by a factor of two. This means the total shield thickness can be reduced a few centimeters if the present criteria proves to be reliable. But with the estimated error of the calculated result being ~100%, there is little sense in the reduction.

The minimum thickness of 71.5 cm is obtained under several limitations based on a thermal and structural analysis. By choosing the shield material considering only the shielding effectiveness, the shield thickness may be reduced another ten centimeters under the same tentative shielding requirement criteria. However 71.5 cm seems to be a practical limit for a realistic design.

(1) Sako, K. et al., "First Preliminary Design of an Experimental Fusion Reactor," JAERI-M 7300 (1977)(in Japanese)

(2) Seki, Y., Iida, H. and Ide, T., "Evaluation of Shielding Design for Superconducting Magnets (III)," JAERI-M 6783 (1976)(in Japanese)

(3) Engle, W.W., Jr., "A User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering," K-1693, Oak Ridge Gaseous Diffusion Plant (1967)

(4) Seki, Y. et al., to be published.

(5) Drake, M.K., "Data Formats and Procedures for the ENDF Neutron Cross Section Library," BNL-50274 (T-601, TID-4500)(1970) Rev. 1974

(6) Ford, W.E., III., "The POPOP4 Library of Neutron-Induced Secondary Gamma-Ray Yield and Cross Section Data, CTC-42 (1970)

Table A.1 Composition of inner blanket and shield for JXFR[1]

Region No.	Region composition	Thickness(cm)
1.	Carbon coating	0.5
2.	Stainless steel (SS)	1.0
3.	Helium (He)	5.0
4.	SS	1.0
5.	SS(90%) + He(10%)	9.0
6.	SS(17%) + He(10%) + W(73%)	24.0
7.	SS	4.0
	Subtotal for the blanket	44.5
8.	Heavy concrete(90%) + Borated water(10%)	30.0
9.	SS(40%) + B_4C(40%) + H_2O(20%)	10.0
	Subtotal for the shield	40.0
	Total of the blanket and shield	84.5

Table A.2 Optimization of inner blanket and shield

	Limitations	Case 1	Case 2	Case 3	Case 4	Case 5
Blanket	SS $\geq$ 30% Coolant $\geq$ 10% thickness 31.5cm	He-cooled	He-cooled	He-cooled	He-cooled	Water-cooled
Shield	SS $\geq$ 30% $H_2O \geq$ 10% thickness 40cm	SS 90% H_2O 10%	SS 30% H_2O 10% W 60%	SS 30% H_2O 20% W 50%	SS 30% H_2O 30% W 40%	SS 30% H_2O 30% W 40%
Cu-DPA [DPA/y]	10^{-4}*	3×10^{-4}	9×10^{-5}	5.5×10^{-5}	6.0×10^{-5}	4.3×10^{-5}

* If once a year annealing of the copper is allowed, the limit for the DPA-rate becomes $\sim10^{-4}$ DPA/year, corresponding to the resistivity increase of 2.5×10^{-8} Ω·cm.

Table A.3 Composition of inner blanket and shield for the INTOR proposal

Region No.	Region composition	Thickness(cm)
1.	TZM protection wall	1.0
2.	SS tube wall	0.5
3.	SS(80%) + H_2O(20%)	10.0
4.	SS(90%) + H_2O(10%)	20.0
	Subtotal for the blanket	31.5
5.	SS(30%) + W(40%) + Borated water(30%)	40.0
	Subtotal for the shield	40.0
	Total of the blanket and shield	71.5

Table A.4 Inner model for the INTOR proposal

Region No., i	Region	Region Radius, r_i(cm)	Mesh No., j_i	$\sum_i j_i$	Mesh Width Δr_i(cm)
1.	Plasma	120.0	1	1	120.0
	Vacuum	133.5	1	2	13.5
2.	TZM protection wall	134.5	2	4	0.5
3.	SS tube wall	135.0	2	6	0.25
4.	SS(80%) + H_2O(20%)	145.0	5	11	2.0
5.	SS(90%) + H_2O(10%)	165.0	10	21	2.0
6.	Vacuum	170.0	1	22	5.0
7.	SS(30%) + W(40%) + Borated water	210.0	40	62	1.0
8.	Air	220.0	1	63	10.0
9.	SS cryostat vessel	225.0	2	65	2.5
10.	SCM, SS(60%) + Cu(28%) + Nb(7%)	310.0	40	105	2.125

Table A.5 Outer model for the INTOR proposal

Region No., i	Region	Region Radius, r_i(cm)	Mesh No., j_i	$\sum_i j_i$	Mesh Width Δr_i(cm)
1.	Plasma	120.0	1	1	120
	Vacuum	133.5	1	2	13.5
2.	TZM protection wall	134.5	2	4	0.5
3.	SS tube + cell wall	137.0	2	6	1.25
4.	Li_2O(21.3%) + SS(4.74%) + He(21.3%)	155.0	6	12	3.0
5.	Li_2O(53.05%) + SS(20.15%) + He(10.5%)	181.6	16	28	1.66
6.	SS end wall	185.0	2	30	1.7
7.	Vacuum	270.0	1	31	85.0
8.	SS(90%) + Borated water(10%)	370.0	50	81	2.0
9.	Air	380.0	1	82	10.0
10.	SS cryostat vessel	385.0	2	84	2.5
11.	SCM, SS(60%) + Cu(28%) + Nb(7%)	480.0	40	124	2.375

Table A.6 Results of bulk shielding design calculations of superconducting magnets for the INTOR proposal

Items*)	Tentative design criteria	Values at inner SCM	Values at outer SCM**)
Maximum DPA in copper (DPA/year)	9.8×10^{-5}	4.3×10^{-5}	4.7×10^{-8}
Maximum neutron fluence (n/cm^2)	2.0×10^{18}	1.1×10^{18}	2.4×10^{15}
Maximum nuclear heating ($Watt/cm^3$)	10^{-3}	1.2×10^{-4}	2.4×10^{-7}
Maximum epoxy dose (rad)	2.4×10^{9}	1.2×10^{9}	1.5×10^{6}

*) Availability and the reactor lifetime are assumed to be 25% and 10 years, respectively.

**) The values at outer SCM surface should be multiplied by ~30 if the effect of neutron streaming through the injection ports are taken into account.

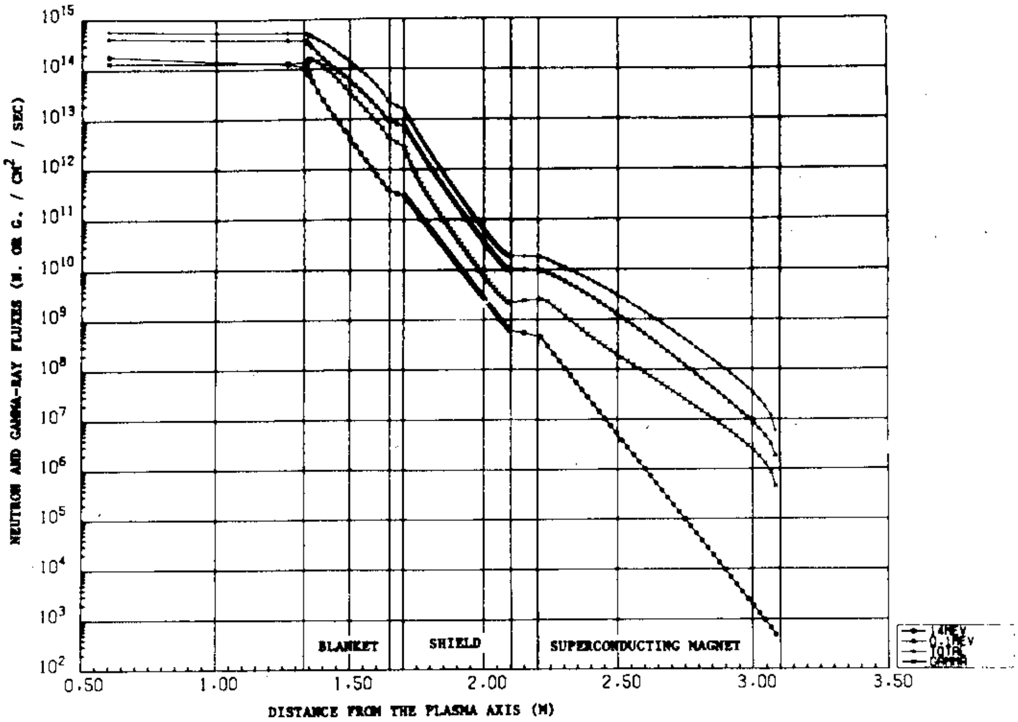


Fig. A.1 Neutron and gamma-ray fluxes in the inner blanket, shield and toroidal field coil

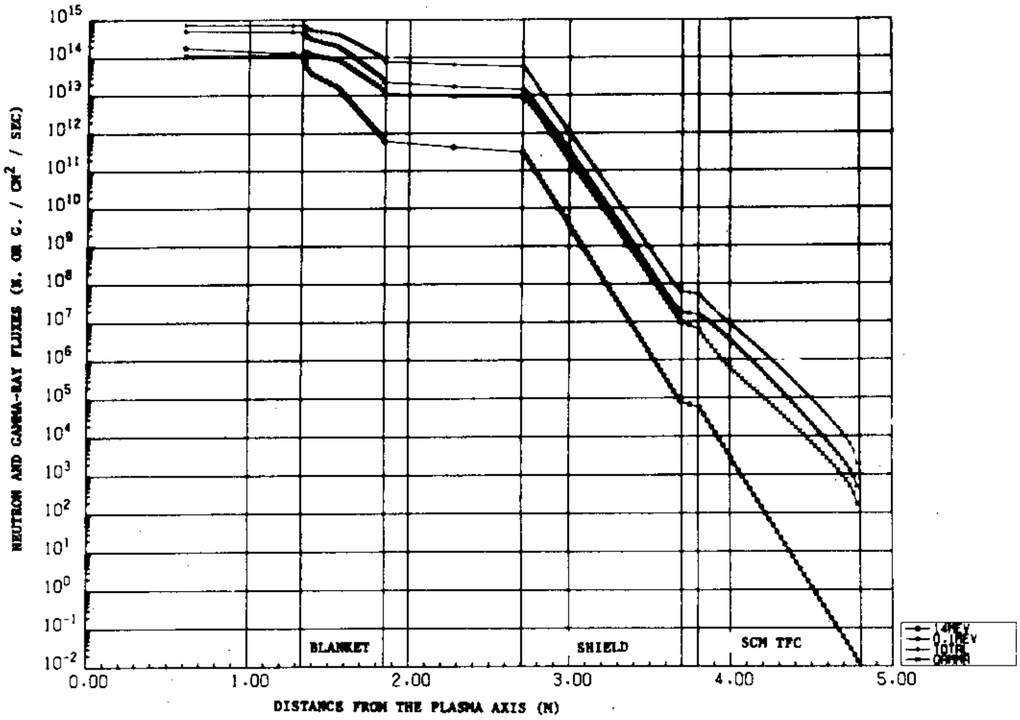


Fig. A.2 Neutron and gamma-ray fluxes in the outer blanket, shield and toroidal field coil

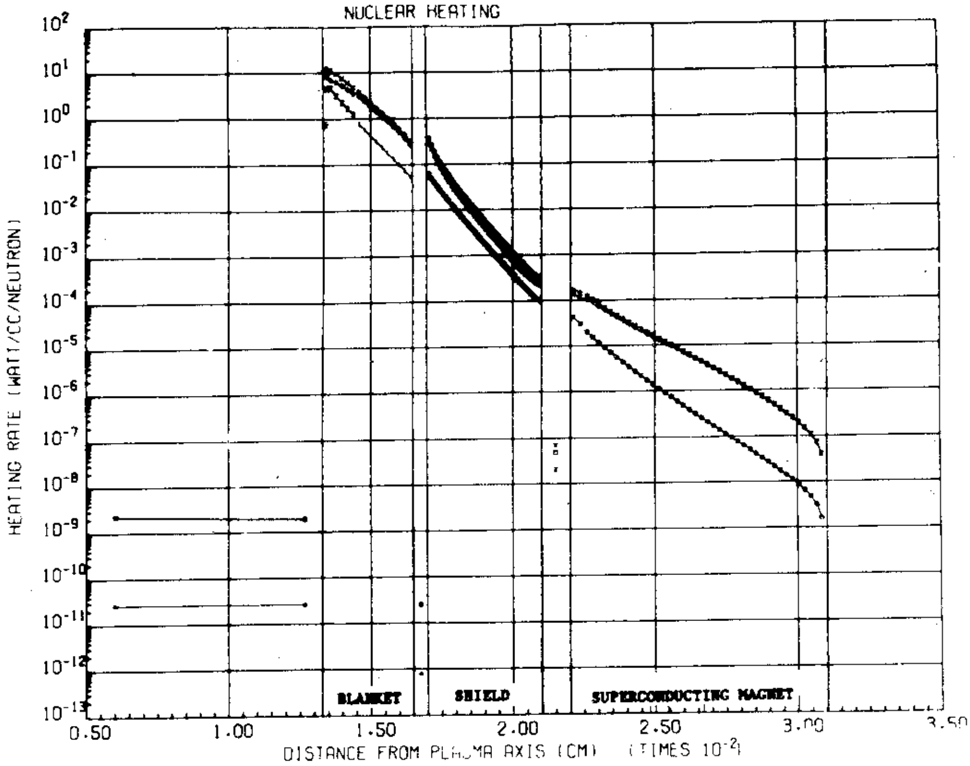


Fig. A.3 Nuclear heating rate in the inner blanket, shield and toroidal field coil

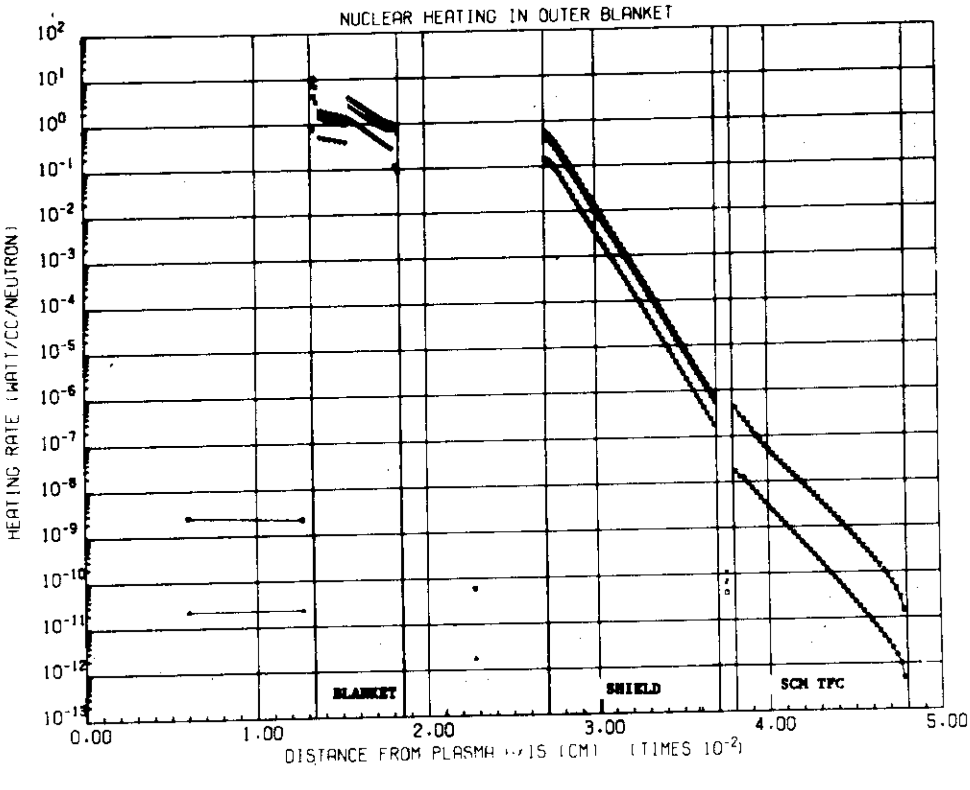


Fig. A.4 Nuclear heating rate in the outer blanket, shield and toroidal field coil

Appendix B R & D Schedule for Fusion Reactor Neutronics in Japan Atomic Energy Research Institute

Experimental research in the field of fusion reactor neutronics has been carried out in Japan Atomic Energy Research Institute (JAERI) ever since 1973. The importance of fusion reactor neutronics after the achievement of reactor grade D-T plasma has been fully recognized and it is reflected in the Fusion Neutronics Source (FNS) program in JAERI. The accelerator and associated equipments will be completed by the end of 1979. The FNS has the neutron source intensity of up to 5×10^{12} neutrons/sec and is capable of generating pulsed neutron source.

The proposal for the R & D schedule for fusion reactor neutronics in JAERI for the next five years is given in Table B.1.

Similar emphasis on the fusion neutronics research in several universities in Japan may also be seen symbolically in the construction of another D-T neutron source of about the same capacity as the FNS, in Osaka University.

Table B.1 Proposed R & D Schedule for Fusion Reactor Neutronics in JAERI

FY1980 = April, 1980 ~ March, 1981

Items, descriptions	FY 1980	FY 1981	FY 1982	FY 1983	FY 1984
(1) Blanket neutronics experiments					
o Blanket benchmark experiments	Lithium benchmark	Li_2O benchmark			
o Neutronically optimized blanket			Optimized blanket assembly		
o Blanket mockup experiments				EPR blanket mockup	
o Measurement technique development	Tritium breeding rate,	Nuclear heating rate,		New methods	
(2) Induced acivity experiments	Short-life induced nuclides decay heat	Medium-lifetime induced nuclides relevant to personnel dose		Long-lifetime induced nuclides for rad-waste evaluation	
(3) Shielding neutronics experiments					
o Streaming experiments	Around blanket	NBI port	Penetration thru primary shield	Mockup exp.	
o SCM neutron irradiation			Neutron damage evaluation exp.		
o Skyshine measurements	Near the FNS	Far from the FNS			
(4) Analysis, data and methods development					
o Integral transport calculation method for deep penetration problem (rigorous treatment of inelastic neutron anisotropy)	1D code and input data	2D code and input data	3D code development		
o Reactor design codes	$2DS_N$-Monte Carlo coupling, Advanced Monte Carlo Code 3d-induced activity calculation				

Group 10 Radiation Shielding and Personnel Access* (Edited by Y. Seki)

1) Minimum thickness inboard shield is discussed in Appendix A.7 of the design report.[(1)] Results of the radiation shielding calculations and detailed structural design of a shield are given in Section 2.3 and Chapter 4 of the report, respectively.

2) Radiation dose rate distribution is given in Section 2.2 of the design report.

3) The shielding design philosophy of INTOR-J is to do without the additional shield as much as possible. Thin lead slabs will be used as additional shield if required.

4), 5) Works in progress.

6) R. & D Schedule for radiation shielding in JAERI has been given in the Appendix B of the Group 10 Report for Session 2.

REFERENCE

(1) Engineering Aspects of the JAERI Proposal for INTOR (II), JAERI-M 8518 (1979)

* Report for Session 3 of IAEA INTOR Workshop

7. Group 15 Safety and Environment*

A. What are the regulations for:

1. tritium release

a. maximum permissible concentrations

	Concentrations in air ($\mu Ci/cm^3$)	Concentrations in water ($\mu Ci/cm^3$)
Controlled area (48 hours/week)	5×10^{-6}	7.5×10^{-2}
Off-site* (3 month average)	2×10^{-7}	3×10^{-3}

* These values are applied at release points on the average as a first step for the control.

b. maximum permissible annual release

The release is to be limited to make off-site dose less than 500 mrem/year.

An estimate of permissible amount of tritium under the assumptions of 1) less than 5 mrem/year off-site, 2) 0.5 km distance from the facility to the site boundary, 3) release height of 50 m, and 4) exposure pathway; inhalation, skin absorption and food chain becomes about 1×10^4 Ci/year.

2. release of activated gases, liquids, etc.

a. maximum permissible concentration

o ^{41}Ar from air activation

o activated impurities in helium coolant

o activated impurities and corrosion products in water coolant

The MPC for these nuclides should be less than 1/10 of the MPC for individual nuclides in the ICRP Recommendations.

* Report for Session 2 of IAEA INTOR Workshop

b. maximum permissible daily/annual release should be limited to achieve less than 500 mrem/year at off-site.

N.B. The actual design should try to achieve the goal of less than 5 mrem/year off-site.

3. radiation exposure of operating personnel

3 rem / 3 month for operating personnel

1.3 rem / 3 month for woman

5(N-18) rem cumulative dose at age N.

radiation exposure of the public

500 mrem / year for the public

4. release of toxic materials

Boron, chromium, copper, lead, lithium, mercury and nickel are some of the non-radiological toxic materials present in a fusion reactor.[1] Their release should be restricted in a manner similar to any chemical plant or mining facility.

5. magnetic field limits for operating personnel

No regulation exsists at present. Some safety standards can be found in Ref.2.

Safety standards for magnetic fields recommended by the Stanford Linear Accelerator Center[3]

	Extended periods (hr)	Short periods (min)
Whole body or head	200 G	2,000 G
Arms and hands	2,000 G	20,000 G

Safety standards for magnetic fields recommended by A.M. Vyalov[4]

	Field (Oe)	Gradient (Oe/cm)
Whole body	300	5 - 20
Hands	700	10 - 20

6. designing for natural disasters?

- earthquake
- flood
-

B. How will personnel safety and environmental considerations affect design codes, standards, quality assurance procedures, redundancy, etc.?

design codes, standards

- o the thickness of the primary shield and the wall thickness of the reactor building
- o impurity control of materials to minimize induced activity

quality assuarance procedures

- o non-destructive inspection of welded parts
- o double walled pipings
- o the higher quality assurance for the more difficult to replace or to repair components (linked poloidal coil, transformer coil, etc.)

redundancy

- o enough redundancy must be provided for those components with high failure frequency e.g. NBI electrodes, ion sources

C. Do we have the necessary physical property data base and computational tools to analyze accidents that could endanger operating personnel or the public or that could produce an adverse environmental impact?

1. tritium release from the fuel processing system and blanket test modules.

   No detailed accident analysis has been done.

REFERENCES

(1) Okrent, D. et al., "On the Safety of Tokamak-Type Central Station Fusion Power Reactors," Nucl. Eng. Design, 39, 215-238 (1976).

(2) Easterly, C.E. et al., "Health Physics Aspects of Fusion Power," ORNL/TM-5461 (1976).

(3) Kaufman, G.E. and Michaelson, S.M., "Critical Review of the Radiological Effects of Electric and Magnetic Fields," in Biological and Crinical Effects of Low-Frequency Magnetic and Electric Fields, ed. by J.G. Llaurado, A.Sances, Jr., and J.H. Battacletti, 1976.

(4) Novitskiy, Y.I. et al., Radio Frequencies and Microwaves, Magnetic and Electric Fields, NASA Tech. Trans. NASA TTF-14.021 (1971).

Group 15 Safety and Environment* (Edited by T. Tone)

Introduction

Tritium release problem is considered to be most important in the safety of a fusion reactor. The amount of tritium which has been handled in Japan is very small compared with that required for a reactor, and regulations being practically applicable to INTOR do not exist.

Accident phenomena resulting from tritium release depend on design details and at the present state of a reactor design it would be difficult to make accurately a quantitative safety assessment. A preliminary analysis for maximum hypothetical accident was made for JXFR(1). The results indicate that the tritium release to environment would be within the regulations. The reason for that even such an accident brings no significant influence on environment is due to the JXFR design philosophy where the total tritium inventory is as low as 0.5 kg and there is no fear of fire like Li-H_2O reaction by adopting a Li_2O-fertile and He-coolant system.

Safety studies for the major components of JXFR were made. The information obtained and the analytic techniques developed will be helpful to the counter-measures for accidents considered for INTOR. The components and items made an object of the safety study are as follows.

∘ Superconducting magnets(2)(3)

Works made are accident classification, FMEA (Failure Mode Effect Analysis) and FTA (Fault Tree Analysis) analyses, coil stability and quench behavior calculations, failure detection and coil protection system designs, structure analysis, fracture and fatigue studies, and earthquake response analysis. Accident analysis of cryostat and refrigeration system was also performed. For poloidal magnets in particular the deterioration of thermal insulation in the vacuum jacket was studied and safety valves were designed to cope with it.

∘ Cooling system(4)

The objectives are to evaluate the tritium releases in normal operation and accidents of the main cooling system, and to choose a tritium treatment and containment system for reducing the releases.

Performed were (1) evaluation of pressure and temperature behaviors of the vacuum vessel during a piping failure of the primary cooling system and analysis of the behavior of tritium adsorbed on the cryosorption-panel and (2) evaluation of the tritium releases from the main cooling system and study on a method to reduce the tritium releases.

* Report for Session 3 of IAEA INTOR Workshop

∘ Fuel circulating system[5]

A three-stage containment system of tritium fuel is settled to control tritium release to the environment as low as possible. The system safety is examined by fault trees, failure effect and related analyses. The components and piping are designed to minimize leakage into the secondary containment which encloses most of the primary containment. The secondary containment is of glovebox type with an inert gas atmosphere, provided Glovebox Atmosphere Purification System of flow rate 70 m^3/min. The tertiary containment system cosists of the rooms enclosing secondary system and tritium-bearing gas processing systems, a 3 m^3/min Atmosphere Cleanup System and a 150 m^3/min Energency Containment System.

∘ Neutral beam injector[6]

Failure and accident classifications were made and counter-measures considering protection and exchange of equipments were studied. A particular attention was paid to failures which have a possibility of bringing about serious accidents such as rapture of coolant channels or pipes used for ion source electrodes, recovery electrodes and neutrizer cell. These accidents are due to a locally concentrated heat deposition of high-energy beam and repeated thermal stress. The material used and cooling conditions (coolant, temperature) were investigated.

References

(1) Sako, K., et al. : JAERI-M 7300 (in Japanese), (1977).
Sako, K., et al. : JAERI-M 8286 (1979).
Sako, K., et al. : Proc. of the 6th Conf. on Plasma Physics and Controlled Nuclear Fusion Research, Berchtesgaden, 1976, Vol.III, 239 (IAEA, Vienna, 1977).

(2) Reactor System Laboratory : JAERI-M 7963 (in Japanese), (1979).

(3) idem. : JAERI-M 8140 (in Japanese), (1979).

(4) idem. : JAERI-M 7772 (in Japanese), (1978).

(5) idem. : JAERI-M 7964 (in Japanese), (1978).

(6) idem. : to be published.

Safety study for JT-60

Safety analysis for JT-60 was made and the measures for safety were examined. The major items considered are as follows.

(1) Noise : Flywheel generators and a transformer substation are considered to be major sources making a loud noise. The noise level at the site boundary was evaluated.

(2) Vibration : The value of vibration at the site boundary due to air-compressor was evaluated.

(3) Waste water : There is no radioactive product in the waste water of JT-60.

(4) Exhaust air : There is no radioactive product and toxic gas in the air exhausted from JT-60. A particular attention is paid to sulpheroxide exhausted from the boiler with a chimney of 27m height.

(5) Exhaust heat : The secondary cooling tower and the boiler were examined.

(6) Mist : The amount of mist from the secondary cooling tower is estimated. An eliminator suppressing the mist is considered.

(7) Waste material : There is no radioactive product in waste materials.

(8) Magnetic field : Diagnostic instruments are protected by magnetic shields.

(9) Electromagnetic field

The design criteria considered for the items (1) - (9) meet the regulations for personel and environment.

(10) Electric power system : Half the power required for toroidal magnets are directly supplied from the major line of Tokyo Electric Power Company. The influence on a local electric line due to pulse operation is considered to be small.

(11) Earth system : The function is equivalent to or superior to that at a transformer substation of electric power companies.

(12) Power failure : Two kinds of electric generator systems are prepared for emergency.

(13) Failure of water supply : Electric generator for emergency is prepared for power failure. Water leak from cooling systems is monitored and when the leak is detected, interlocking works to protect equipments.

(14) Magnetic energy : A protection circuit is designed to cope with the release of magnetic energy by accident.

(15) Gas : Leak of hydrogen gas is monitored. Explosion due to hydrogen-oxygen reaction in the vacuum vessel is small enough against the strength of the vacuum vessel. Cyanide concentration produced in a cyan laser is low enough.

(16) Fire : Appropriate measures are taken. There is no fear to release radioactive products.

(17) Radiation : X-ray generated by run-away electrons and resulting neutrons from (γ,n) reaction were evaluated. The concrete ceiling of 50cm thickness and the concrete side wall of 200cm thickness reduce the radiation level to a value which has no influence on environment.

(18) Earthquake : The results of investigation assuming an earthquake of class B reveal that the JT-60 building is earthquake-resistant enough.

These safety analyses were examined by Consulting Committee for Safety of JAERI and the results were summarized in internal memoranda.

Regulations for transportation

The Ministry of Transport established the regulations for tritium transportation by road, seagoing vessels and air December 1978, according to the Regulations for the Safe Transport of Radioactive Materials by IAEA and WHO in 1973.

The amount of tritium activities permitted to transport is divided into different classes in accordance with specifications of packaging and the chemical form of tritium. The upper activity limit for Type A package is 1000 Ci. The transportation of activities beyond these values requires Type B package.

At present JAERI is constructing FNS (Fusion Neutron Source) which utilizes two kinds of rotating tritium targets. The respective tritium activities are 330 and 1800 Ci. The former targets were imported. Three targets which total to 990 Ci were already transported to JAERI with Type A package. For the latter, Three metal targets (1800 Ci X 3) are to be imported. The tritium activities of 1800 Ci X 3 is the largest case to which the new regulations are applied.

Home task presented at Session 2

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