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Some ITER Shield Problems Examination

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A study on shielding problems of the International Thermonuclear Experimental Reactor (ITER) is presented. Considerable effort was spent on the design of the blanket and shield of ITER during Conceptual Design Activities (CDA) Phase of work. However, many problems of shielding are to be solved during Engineering Design Activities (EDA) Phase of work. The study has been performed to solve some ITER shielding problems and to rise up others which are to be solved.

The present study is composed of four parts, each of which dedicated to a group of common problems which are a greater concern in terms of shielding performance. These groups are the inboard ITER shield problems, outboard ITER shield problems, top/bottom ITER shield problems and shielding recommendations in connection with the ITER maintenance.

Most of problems were solved analytically by extrapolations of previous results of neutron transport calculations on the design elaborated in Japan after the CDA. The problem of neutron streaming through the ITER divertor piping was solved by carrying out neutron transport calculations for the two-dimensional DOT3.5 code.

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Keywords: ITER, Shielding, Neutronics, Biological Shield, Blanket, Concrete Cryostat, Neutron Transport Calculations

ITERの遮蔽構造に対する検討

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国際熱核融合実験炉（ITER）におけるブランケット／遮蔽構造については，ITERの概念設計（CDA）段階において種々の検討がなされた。しかしながら，次の工学設計（EDA）段階において解決されるべき問題も多いと考えられる。本検討は，CDAで残された問題に対する解決案を示すと共に，さらに詳細な検討を要する事項を明らかにすることを目的として行った。

ここでは，ITERのインボード領域およびアウトボード領域，上部／下部領域の設計の進展に伴い，それぞれの領域における遮蔽設計評価を行い，各部に対して最適な遮蔽構造の提案を行った。また，炉内構造物の分解修理シナリオの進展に対応し，分解修理時に際して考慮すべき遮蔽対策・構造についての提案を行った。

本検討の多くは，CDA終了後さらに日本において進められた設計検討に対して，従来の核計算を内・外挿することにより行った。ただし，ダイバータ用冷却水配管周囲のギャップにおける中性子ストリーミングについては2次元輸送計算コードDOT 3.5を用いた計算結果に基づいて評価した。

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1. INTRODUCTION.

Conceptual design activities (CDA) of the International Thermonuclear Experimental Reactor (ITER) has been carried out in the three years from 1988 to 1990. During the CDA, considerable effort was spent on the nuclear design of the blanket and shield of the reactor along with the shielding calculations. Two main activities should be done now, at the transitional period between the CDA and EDA (next ITER Engineering Design Activity Phase of work): 1) the comparative analysis of CDA results and 2) the analysis of new design decisions and recommendations.

As it was mentioned above, the CDA phase of work was finished in December 1990. However, the significant design work under the ITER project was carried out in 1991 and 1992 in Japan. Many details of the ITER design were clarified by this work. On the other hand, many problems were also risen up during this work, for instance, how to design the blanket, shield and vacuum vessel of the ITER. Thus, the neutronic analyses (both the analytical and calculational ones) were carried out to support above-mentioned design work. Such a type of work was done for some problems of both the toroidal field (TF) coils and biological shield of ITER.

Some zones in ITER were marked during the ITER CDA, where the shield thickness will not be sufficient to decrease the radiation level down to the limit. The TF coils can be damaged or the access in a reactor room can not be allowed in these zones. Three main zones of a greater concern in terms of shielding performance are shown in Fig. 1.1. The relatively small thickness of the shield in the inboard central zone is dangerous from the view point of the TF coil shield (zone 1). The neutron streaming through the NBI channel can be dangerous from the view point of both the biological shield and TF coil shield (zone 2). The neutron streaming through water channels and void gaps around channels in the Top/Bottom zones is dangerous from the view point of biological shield (zone 3).

Therefore, all neutronic analyses presented here are related mainly to these three zones of the ITER blanket/shield. It should be underlined that a large attention was paid to the biological shield problems in this examination. It is explained by the importance of this subject for the ITER EDA phase of work. Besides, many details of the maintenance have been elaborated in JAERI in 1991-1992. Thus, it was possible to study this subject on the biological shield effectiveness with a mechanical configuration more detail than the CDA design.

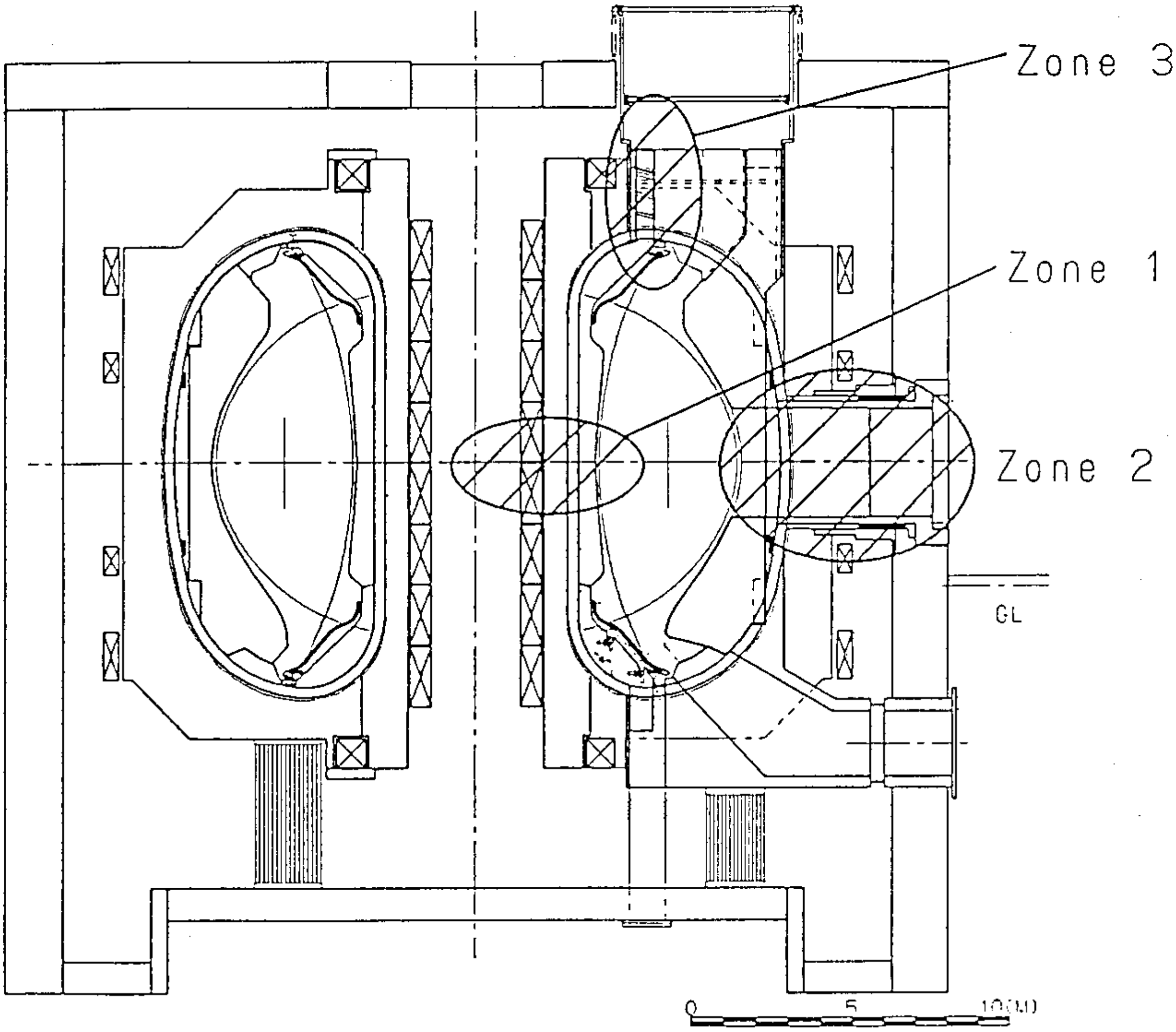

Fig. 1.1 Zones marked from the viewpoint of shielding

2. INBOARD ITER SHIELD PROBLEMS.

2.1 The comparison of Japanese ITER inboard blanket shield effectiveness with Russia , EC, and U.S. reference options.

The shielding effectivenesses of different blanket options presented as reference blankets by all Parties were compared during ITER July/August 1990 Biological shield workshop [1]. Later the same analysis was done in Ref. 2. The results of both above mentioned works are summarized in Table 2.1.

Table 2.1 The comparison of shielding effectiveness of different blanket options in arbitrary units

	Russia	U.S.	EC	Japan
Shielding effectiveness according to [1]	1	1	0.65	0.7
Shielding effectiveness according to [2]	1	1.2	0.6	0.9-0.75

Major design parameters of each blanket and shielding criteria for the superconducting magnets are indicated in Tables 2.2-2.3 [3] and Table 2.4 [3], respectively. Concept of each blanket is shown in Figs. 2.1-2.4 [3].

According to Ref. 2 the 5 cm tungsten layer at the back vacuum vessel surface can reduce the dose to magnet electrical insulator below the limit for the U.S. blanket design only.

According to Ref. 1 the dose to electrical insulator of inboard TF coils will be marginal for the Russia blanket option that is close to results for the Russia blanket option presented in Ref. 2 (where the recommendation was made to increase the inboard shield thickness of ITER build-up with the Russia blanket option by about 3 cm to decrease the dose to electrical insulator below the limit).

All above-mentioned results [1,2] indicate that the Japanese option of a ceramic breeder pebble bed blanket will be not effective enough to protect the TF coils from irradiation during ITER lifetime (3.8 FPY). It means that an increase of shielding effectiveness for the inboard build-up is required.

The simplest way is to increase the total inboard shield thickness by about 3 cm (the evaluation based on Ref. 1) or 5 cm (the evaluation based on Ref. 2). But such a possibility should be coordinated to be consistent with the design of other reactor components (see Appendix 2), and even with plasma physics.

The more realistic way is to increase the shielding effectiveness of the vacuum vessel. For example, if borated stainless steel is used in the thin-wall vacuum vessel proposed by Japan, it will increase the shielding effectiveness by about 25-30%. This corresponds to about 2.5 cm increase of a total shield thickness.

Table 2.2 Design parameters of the lead-lithium breeder design option

Parameter	Value	Parameter	Value	Parameter	Value
Li(17)-Pb(83) eutectic mass, T	1000	^{6}Li enrichment, %	90	Pressure losses	
- inboard blanket (IB)	125			Physics/Technology,MPa	
- outboard blanket (OB)	875	Tritium breeding ratio	0.76-0.8	IB segment	0.159/0.10
				IB channel	0.15/0.095
IB segment mass, T	40	Number of tubes, from		Ob segment	0.25/0.073
		IB/OB segments	7/9	First channel of OB	0.14/0.06
OB segment mass, T:				Second channel of OB	0.10/0.099
- lateral	80	Maximum tube diameter,mm		Third channel of OB	0.10/0.099
- central upper/lower	64/35	- IB segments	80		
		- OB segments	120	Eutectic maximum temperature	
Breeding channels number in				in breeding channel	
IB/OB segments	3/11	Thermal power from the		Physics/Technology, °C	
		breeding		IB	241/211
Breeding channels diameter, mm		Physics/Technology, MW	546/440	first channel of OB	222/193
- IB/OB	180/190	IB	95/76	second channel of OB	181/169
		OB	451/364	third channel of OB	131/124
Breeding zone thickness at		Channel thermal power			
the midplane		physics/technology, kW		Maximum temperature of channel	
IB/OB, mm	180/510	IB	990/792	wall physics/technology, °C	
		OB first row	1500/1208	Ib	115/109
Blanket thickness		OB second row	688/556	first channel of OB	93/90
at the midplane		OB third row	333/271	second channel of OB	117/114
IB/OB, mm	490/1100			third channel of OB	105/104
		Coolant flow rate, kg/s:			
Eutectic mass, t		- blanket	3266/2581	Total helium flow rate through	
- IB channel	1.6	- IB segment	18.2/14.4	blanket, kg/s	1-20
- OB channel, 1-st row/2-nd		- OB segment	57.4/45.8		
and 3-d rows	2.0/1.65	- IB channel	6.08/4.81	Range of helium temperatures	
		- first channel of OB	8.9/7.1	at the blanket inlet, °C	100/450
Water content in breeding		- second channel of OB	4.5/3.6		
channels, %		- third channel of OB	2.0/1.7		
- 1-st row/2-nd and 3-d rows	15.9/35.6				

Table 2.3 Design parameters of the ceramic breeder design options

PARAMETER	LAYERD DESIGN INBOARD	LAYERD DESIGN OUTBOARD	LAYERD PEBBLE BED DESIGN INBOARD	LAYERD PEBBLE BED DESIGN OUTBOARD	BIT DESIGN INBOARD	BIT DESIGN OUTBOARD
BREEDER						
MATERIAL:	Li_2O	Li_2O	Li_2O		$LiAlO_2$	$LiAlO_2$
^6Li ENRICHMENT (%)	95		50		50	
FORM:	sintered	sintered	sintered Pebbles		sintered Pellets	sintered Pellets
GEOMETRY:	blocks	blocks	Pebble (60%PF)		Cylindrical Pellets	Cylindrical Pellets
MASS(MT):	2.07	10.53	~3	~90	5.5	62.5
TEMPERATURE (C) WINDOW:	400-1000	400-1000	400-1000	400-1000	450-900	450-900
PHYSICS PHASE						
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	509-603	502-609	—	—	—	—
120% NORMAL OPERATION:	611-737	610-744	—	—	—	—
150% NORMAL OPERATION:	778-961	764-964	—	—	—	—
TRITIUM INVENTORY (g):	0.42	2.18	—	—	—	—
TECHNOLOGY PHASE						
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	452-526	451-537	450-700	450-800	500-750	500-750
120% NORMAL OPERATION:	539-638	541-649	~500-800	~500-900	570-900	570-900
150% NORMAL OPERATION:	677-821	673-833				
TRITIUM INVENTORYa (g):	1.94	11.6	~9.2	~117	—	~5
MULTIPLIER						
MATERIAL:	Be	Be	Be		Be	Be
FORM:	sintered	sintered	Fused Pebbles		Sintered Pellets	Sintered Pellets
GEOMETRY:	blocks	blocks	Pebble (60%PF)		Annular Pellets	Annular Pellets
MASS(MT):	19	187	~8	~185	15	174.5
TEMPERATURE (C) WINDOW:	<600	<600	<600	<600	<600	<600
PHYSICS PHASE						
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	172-465	130-471	—	—	—	—
120% NORMAL OPERATION:	190-558	143-576	—	—	—	—
150% NORMAL OPERATION:	216-712	164-752	—	—	—	—
TRITIUM INVENTORY (g):	4.6	22.4	—	—	—	—
TECHNOLOGY PHASE						
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	162-415	123-422	~100-500[d]	~100-700[d]	60-100	60-100
120% NORMAL OPERATION:	178-491	135-512	~100-600[d]	~100-800[d]	60-100	60-100
150% NORMAL OPERATION:	201-620	154-662	—	—	—	—
TRITIUM INVENTORY[a] (g):	230	1130	—	—	—	—
BREEDER He PURGE GAS						
IN-OUT PRESSURE (MPa):	0.2 - 0.1	0.2 - 0.1	0.11-0.10	0.11-0.10	0.2	0.2
FLOW RATES (MOL/S):	0.53	2.71	0.31	2.79	—	2.6
INLET H2 (%):	0.2	0.2	0.817(1vpm)	0.817 (1vpm)	1	1
MAX.-AVE. OUTLET HT(HTO) PRESSURE(Pa):	13.3-12.8	18.0-12.3	15.5(0.82)	15.5(0.82)	10-1	10-1
H/T RATIO:	29.7	30.6	100	100	100	100
PUMPING POWER (KW):	2.66	13.6	—	—	—	10

Table 2.3 Design parameters of the ceramic breeder design options (cont.)

PARAMETER	LAYERED DESIGN INBOARD	LAYERED DESIGN OUTBOARD	LAYERED PEBBLE BED DESIGN INBOARD	LAYERED PEBBLE BED DESIGN OUTBOARD	BIT DESIGN INBOARD	BIT DESIGN OUTBOARD
CLAD MATERIAL FOR BREEDER AND MULTIPLIER	SA 316SS	316SS	SA 316SS	SA 316SS	CW316SS	CW316SS
MIN.-MAX. THICKNESS (mm):	1-1	1-1	1	1	0.3-1.0	0.3-1.0
MASS(MT):	2.2	13.8	~4	~45		
TEMPERATURE (C) WINDOW:	<550	<550	<500	<500	420-550	420-550
PHYSICS PHASE						
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	483-496	475-497	—	—	—	—
120% NORMAL OPERATION:	584-595	571-607	—	—	—	—
150% NORMAL OPERATION:	744-755	716-791	—	—	—	—
TECHNOLOGY PHASE			—	—		
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	431-440	424-446	400-600[d]	400-680[d]	480	480
120% NORMAL OPERATION:	514-523	506-541	450-700[d]	400-780[d]	550	550
150% NORMAL OPERATION:	647-658	631-693	—	—	—	—
STRUCTURE MATERIAL (COOLANT CHANNELS, SIDE WALLS, AND FIRST WALL)	SA 316SS	SA 316SS	SA 316SS	SA 316SS	SA316SS	SA316SS
TEMPERATURE (C) WINDOW:	<400	<400	<400	<400	<400	<400
MASS(MT):	19	187	~97	~500	—	—
PHYSICS PHASE						
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	95-239	85-278	60-200[e]	60-300[f]	—	—
120% NORMAL OPERATION:	98-250	90-293	—	—	—	—
150% NORMAL OPERATION:	103-266	95-326	—	—	—	—
TECHNOLOGY PHASE					—	—
MIN.-MAX. TEMPERATURE (C) DURING						
NORMAL OPERATION:	92-187	77-225	60-170	60-170	60-100	60-100
120% NORMAL OPERATION:	93-207	78-255	—	—	60-100	60-100
150% NORMAL OPERATION:	93-237	79-299	—	—	—	—
BLANKET/SHIELD COOLANT						
MATERIAL:	H20	H20	H_2O	H_2O	H_2O	H_2O
DIRECTION:	POLOIDAL	TOROIDAL	Poloidal	Poloidal	Poloidal	Poloidal
IN-OUT TEMPERATURE (C):	60-100	60-100	60/100	60/100	60/92	60/92
IN-OUT PRESSURE (MPa):	1.50-1.32	1.50-1.45	1.5-1.3	1.5-1.2	1.2/1	1.2/1
MIN.-MAX. VELOCITY (M/S):	0.70-4.35	1.20-2.20	<3.5	<3.5	3-4	3-4
TRITIUM PERMEATION RATE (Ci/D):	—	—	—	—	<0.1	<0.1
PUMPING POWER (MW):	—	—	—	—	0.2	0.2

Table 2.3　Design parameters of the ceramic breeder design options (cont.)

PARAMETERS	LAYERED DESIGN INBOARD	LAYERED DESIGN OUTBOARD	LAYERED PEBBLE BED DESIGN INBOARD	LAYERED PEBBLE BED DESIGN OUTBOARD	BIT DESIGN INBOARD	BIT DESIGN OUTBOARD
GEOMETRICAL DETAILS						
FW-BLANKET RADIAL THICKNESS (mm)						
MIDPLANE:	15-95	14-261	15-150[g]	15-562[g]	170	627
TOP/BOTTOM:	15-171	14-571	15-	15-863[g]	170	—
LAYER DIMENSION IN THE RADIAL DIRECTION OR TUBE DIMENSIONS AND ARRANGEMENT						
MIDPLANE:	Table VIII.1-2	Table VIII.1-1	Fig.2.2-1	Fig.2.2-2	46	46/64
TOP/BOTTOM:	Table VIII.1-2	Table VIII.1-1	—	Fig.2.2-3	46	46/64
PENETRATION ACCOMMODATION FOR						
NBI:	—	Fig.	—		—	Fig.
PORTS:	—	Fig.	—		—	Fig.
TRITIUM BREEDING PERFORMANCE						
NUCLEAR DATA BASE:	ENDF/B-V		JENDL3		FFFI	
TRANSPORT METHOD:	ONEDANT-MCNP CODES		1-DSN		MCNO-3B	
MIDPLANE-EXTREMITY LOCAL POLOIDAL TBR:	0.755-0.895	1.375-1.310[b]	0.54-	1.35-1.46	—	—
MIDPLANE-EXTREMITY LOCAL TOROIDAL TBR:	0.206-0.217	0.948-0.966[b]			—	—
NET ESTIMATE TBR:	0.14	0.70[b]	0.08	0.72		
NET TBR BASED ON 3D CALCULATION:	0.12	0.69[b]	—		0.09	0.71
NUCLEAR HEATINGc (MW)						
TOTAL NUCLEAR POWER						
PHYSICS PHASE:	240	860	—	—		
TECHNOLOGY PHASE:	179	640	—	—	-94	-750
BREEDER POWER DENSITY (MW/M^3)						
MIN.-MAX. AT MIDPLANE:	-30.9	- 58.7	1.5-18	1-24	—	-25
MIN.-MAX. AT TOP/BOTTOM:	14.8	39.8	0.7-8	0.06-19	—	- 15
MULTIPLIER POWER DENSITY (MW/M^3)						
MIN.-MAX. AT MIDPLANE:	- 3.9	- 6.8	1.4-3	0.17-4	—	-4.5
MIN.-MAX. AT TOP/BOTTOM:	- 1.5	- 3.3	0.62-1.3	0.02-1.8	—	-2.7
DECAY HEAT AT SHUT DOWN (MW):	14.8		—		—	

a) Burn time of $2x10^6$ and $1.2x10^8$ s for physics and technology phases without any tritium release.
b) Copper stabilizer included.
c) Only nuclear heating
d) Will be lowered by optimization of layer thickness
e) With radiative-coiled tile
f) At local high flux region of 0.6 mW/m^2 with radiative-cooled tile
g) Including first wall and back wall

Table 2.4 Shielding performance parameters (safety factors are included)

Response	Design Limit	Calculated Value
Total nuclear heating in toroidal field coils, KW	55	57
Peak nuclear heating in winding pack, mW/cm^3	5	1.8
Peak dose to electrical insulator, rads	$5x10^9$	$5x10^9$
Peak fast (E>0.1MeV) neutron fluence to superconductor, n/cm^2	10^{19}	$6x10^{18}$
Peak displacement in copper stabilizer, dpa	$6x10^{-3}$	$2.7x10^{-3}$
Peak helium production in type 316 stainless steel of the vacuum vessel to permit rewelding, appm	1	11
Biological dose outside cryostat one day after shutdown, mrem/h	< 0.5	0.5^a

a) A total thickness of 175 cm is assumed for the outboard first wall, blanket, shield, and cryostat.

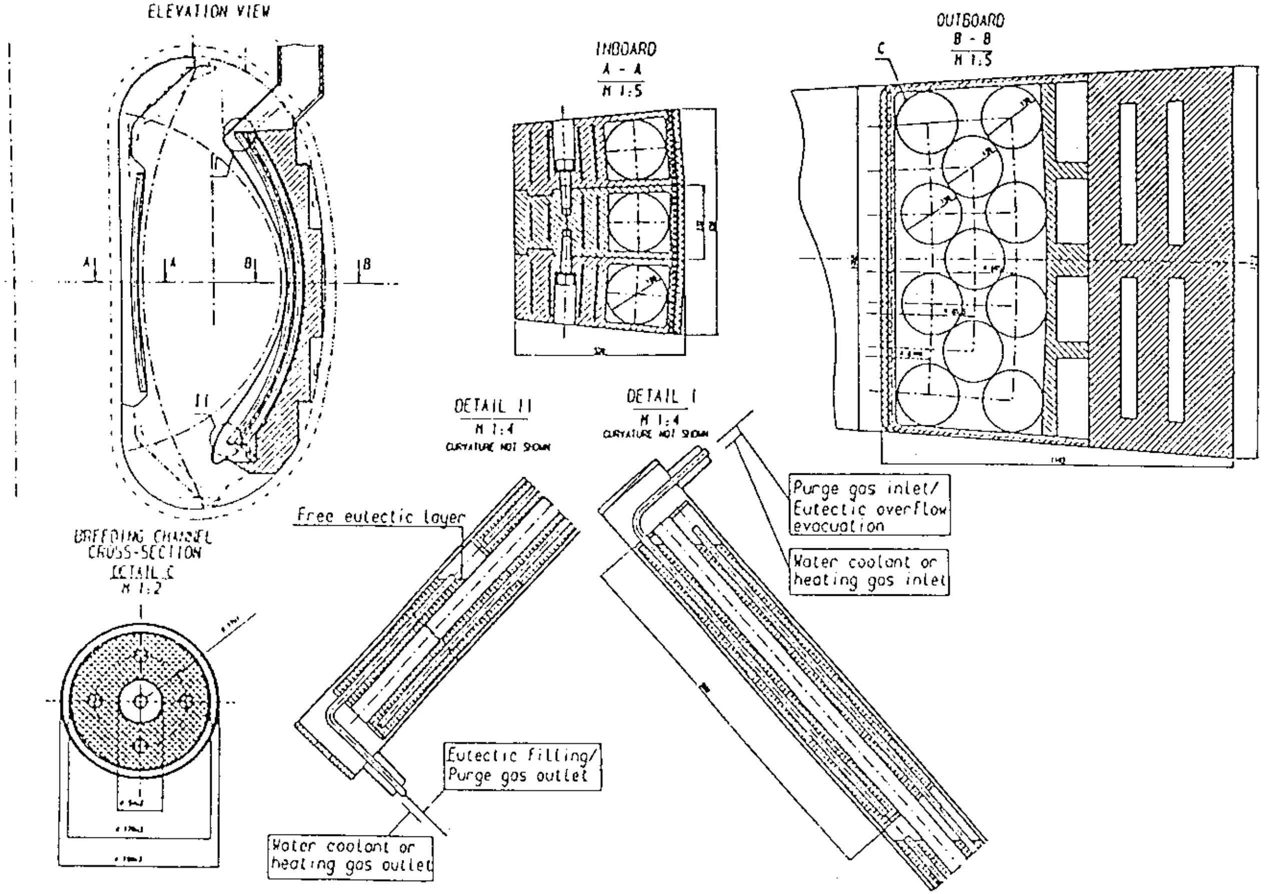

Fig. 2.1 Cross sectional view of 83Pb-17 Li breeder blanket design with breeder-in-tube configuration and poloidal cooling

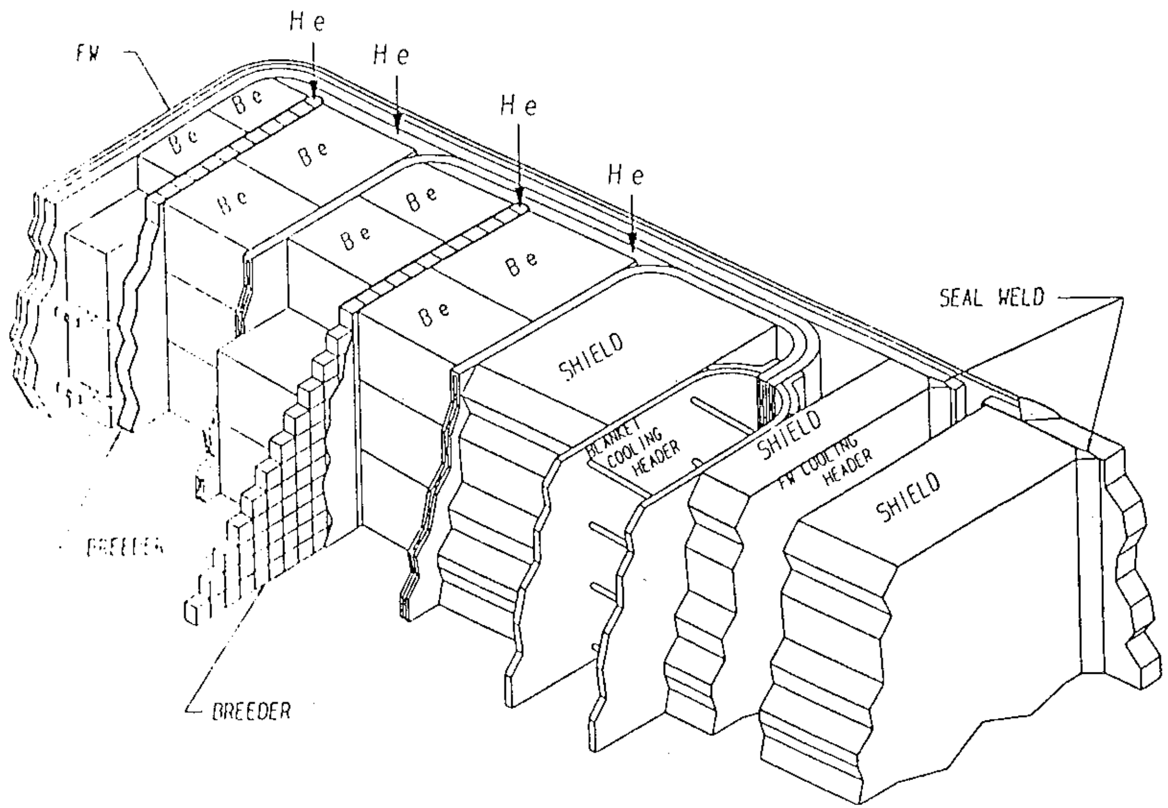

Fig. 2.2 Isometric view of multilayered ceramic breeder blanket design with toroidal cooling and both Li_2O breeder and Be neutron multiplier in the form of sintered blocks

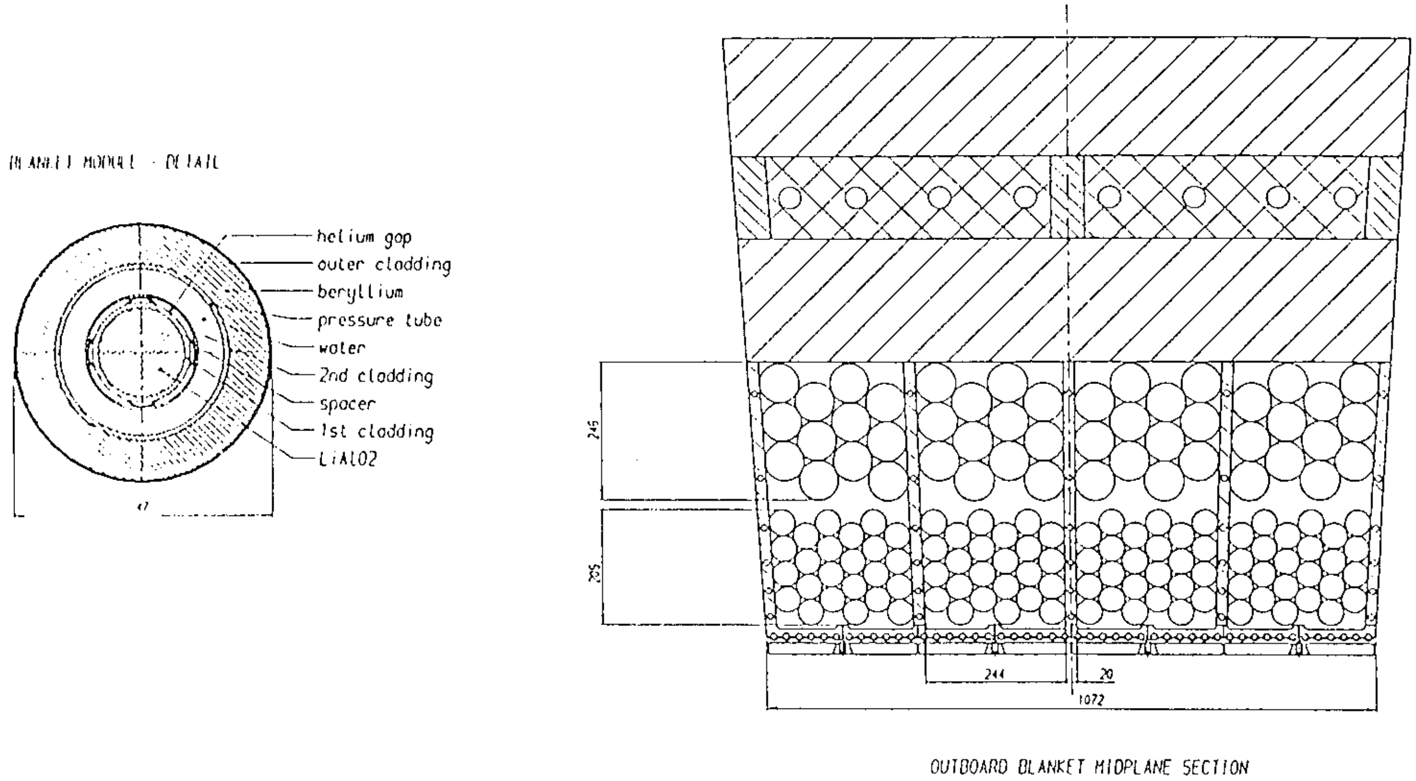

Fig. 2.3 Cross sectional view of BIT ceramic breeder blanket design with poloidal cooling and both $LiAlO_2$ breeder and beryllium in the form of sintered pellets

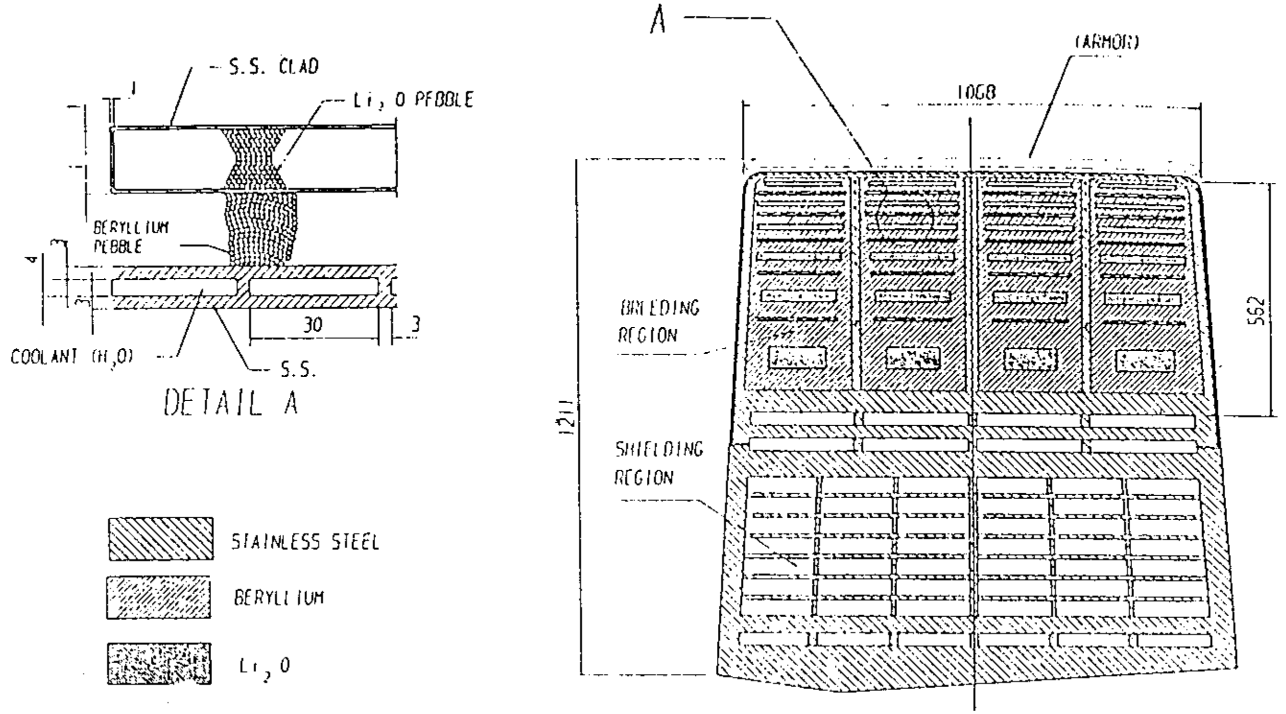

Fig. 2.4 Cross sectional view of multilayered ceramic breeder blanket design with Li_2O breeder and Be neutron multiplier in the form of small sintered pebbles

2.2 The biological dose in the central reactor zone behind an inboard bulk shield.

The personal access is not planned in the ITER inboard zone (behind the inboard TF coils). However, the relatively high level of the biological dose in the inboard zone (by some orders of magnitude higher than the limit) could influence on the biological dose level at the top zone where the personal access is planned 24 hours after reactor shut down.

The 84 cm ITER inboard build-up in the reference design is relatively thin to decrease the biological dose level to the limit. However, the thickness of inboard TF coils should be taken into consideration in the biological dose evaluation because the TF coils have no gaps in the inboard zone (in comparison with the outboard zone where there are relatively large cavities between adjacent TF coils). Therefore, the total thickness of "the inboard biological shield" will be ~ 151 cm. Taking into account of the porosity of inboard blanket and the 2 cm gap in the inboard build-up, the effective total "inboard biological shield thickness" will be about 134 cm as shown in Fig. 2.5.

For the outboard bulk shield the effective thickness is only 124 cm because the thickness of TF coils is not included. Therefore, taking into consideration of above-mentioned speculations on the shield thickness for biological shielding and the 1.5 times difference in the neutron wall load at the inboard and outboard first walls, about 8 times difference in the biological dose rate will be obtained in the mid-plane between the inboard and outboard zones behind FT coils. The neutron streaming through the NBI duct wall will rise the biological dose rate in the outboard zone (inside cryostat) by about two orders of magnitude. Thus, finally the difference between the biological dose rates in the inboard and outboard zones can reach about three orders of magnitude. This means that the problem of a biological shield in the inboard reactor zone is not so important as for the outboard zone and seem to be solved relatively easily in coming ITER EDA Phase of work.

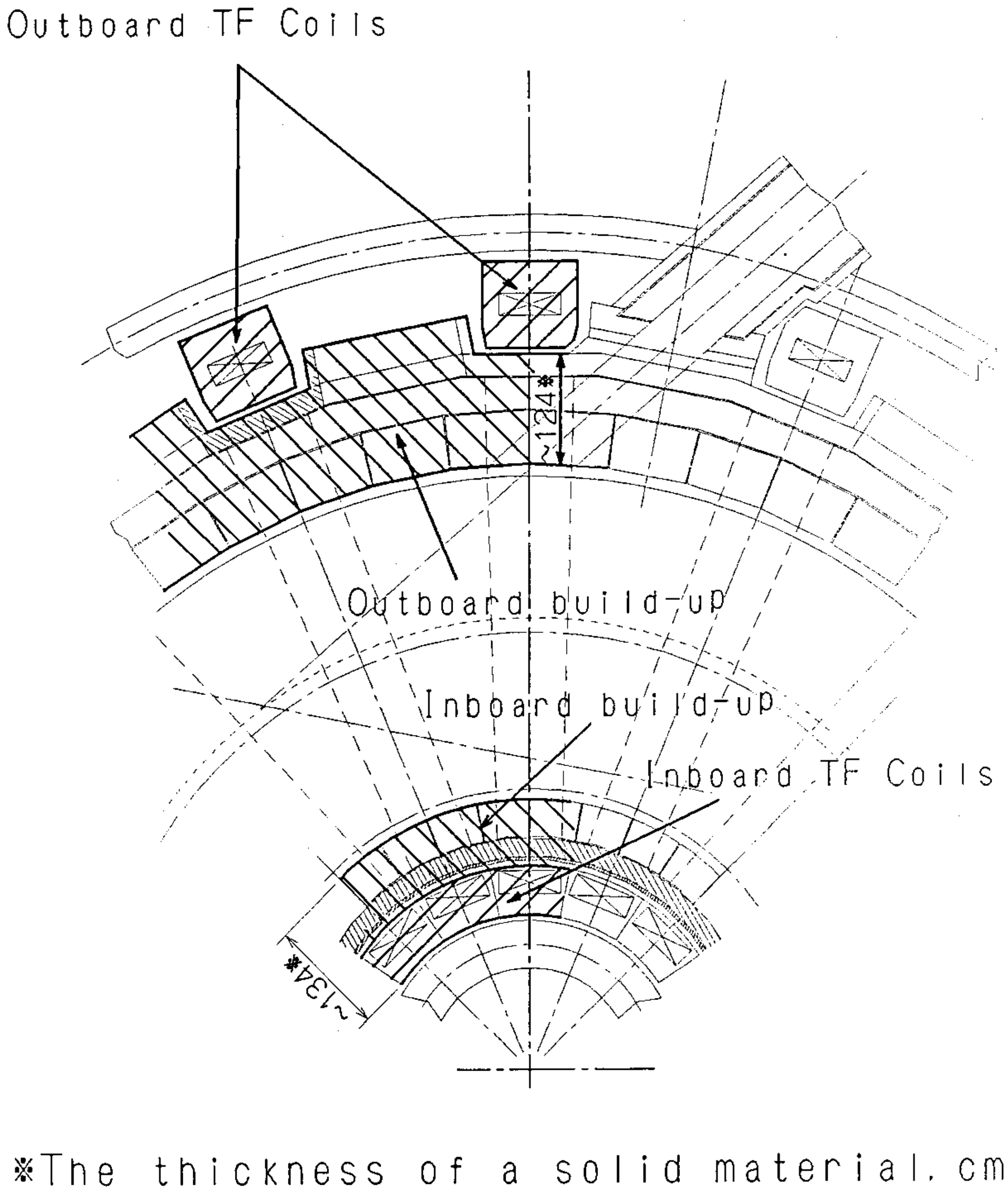

Fig. 2.5 Reference ITER equatorial plane cross-section

3. OUTBOARD ITER SHIELD PROBLEMS.

3.1 The main ITER CDA results of outboard bulk biological shield analyses and the difference in Japanese approach to the biological shield concept.

3.1.1 The main ITER CDA results of a bulk biological shield analysis.

The main results were presented in Refs. 3-6.

In Ref. 4 the U.S. team presented the results of calculations of the 155 cm outboard bulk shield with a ceramic breeder/beryllium blanket (with ~ 70% dense beryllium). The safety factor of 10 was included to account mainly for the 120 cm deep assembly gap between blanket/shield modules in the outboard region. The (2-3)$\cdot 10^3$ mrem/h biological dose rate was calculated just behind the 155 cm outboard bulk shield 24 hours after shutdown. This is in a good accordance with the results of the estimation of the biological dose rate (after the Technology Phase) behind the 151 cm bulk shield extrapolated from Japanese results [7] for the outboard bulk shield calculations. This dose was evaluated to be (2-5)$\cdot 10^3$ mrem/h.

The main result of calculation for the dose after shutdown with the U.S. outboard bulk shield configuration (with a ceramic blanket shown in Fig. 2.2) is that the 2.5 mrem/h limit could be achieved behind a double wall steel cryostat (the effective solid thickness of the steel is 7 cm) in a reasonable time (~1 day) only after the Physics Phase with the 0.05 full power years (FPY) of operation. Thus, it can be derived from these results that the biological dose rate after the Technology Phase with 3.7 FPY will exceed the limit by more than 50 times (if above mentioned double wall steel cryostat will not be filled by water). Therefore, it takes about one month to reach the 2.5 mrem/h limit after the Technology Phase [4].

In Ref. 5 the Russian team presented results of calculations of 151 cm outboard bulk shield. The main result for the Russia configuration of outboard bulk shield described in [5] is that the 2.5 mrem/h limit could be achieved behind a double wall steel cryostat (the effective solid thickness of the steel is 7 cm) 1 day after shutdown only after the Physics Phase and only if the outer surface of the cryostat is additionally covered by 2-3 cm of lead.

In the ITER summary report [3] it was mentioned that "**a total thickness of 175 cm for the first wall, blanket, shield, vacuum vessel, and cryostat is required to achieve a dose equivalent value of 0.5 mrem/h one day after shutdown outside the cryostat**". However, according to the reference ITER concept [3], there is only 151 cm of total outboard bulk shield thickness in the mid-plane and the double wall steel cryostat with the effective 7 cm steel thickness, thus 158 cm in total shield thickness. All results [4-7] were obtained in consistence with the reference 151 cm thickness of ITER outboard build-up. Thus, a need of a biological shield outside a cryostat goes (as it was shown above in this paragraph) from all results [4-7] presented during ITER CDA Phase of work.

Possible approaches to a biological shield concept could be summarized as follows :

1) to increase the thickness of outboard build-up in the mid-plane from 151 cm to 168 cm in consistence with the above-mentioned recommendation [3] (168 cm of outboard build-up + 7 cm effective thickness of double wall steel cryostat = 175 cm of a total thickness "for the first wall, blanket, shield, vacuum vessel, and cryostat");

2) to design, in the case of reference 151 cm ITER outboard build-up, an additional biological shield outside or inside the double wall steel cryostat (the thickness of biological shield in this case should be about 20 cm of steel/water structure);

3) to fill water in the double wall steel cryostat;

4) to design a concrete cryostat in place of the double wall steel cryostat as mentioned below.

It should be mentioned that all above approaches are based on the results of calculations for solid outboard shield. According to Ref. 7, the neutron streaming through the 35 cm-thick NBI-duct side wall near TF coil will increase the biological dose rate outside the cryostat by about three orders of magnitude. Thus, more effective biological shield has to be design for ITER. This factor was taken into account in Japan approach of concrete cryostat to the ITER biological shield.

3.1.2 The difference between Japan and other Parties' approaches to the biological shield concept.

The main difference in Japanese approach to the ITER biological shield is the idea to joint functions of a cryostat and biological shield. In this case, the relatively thick concrete wall could serve both as a cryostat and biological shield.

On the contrary, the double wall steel cryostat with 70 cm thickness (the effective steel thickness of 7 cm) was used in Russia and U.S. options. For this type of cryostat some vague indications about a necessity of an additional biological shield outside cryostat can be found in Refs. 4-6 as mentioned above.

However, the level of a biological dose behind ~151 cm ITER outboard bulk shield was practically the same in all presented works [4-7]. These results indicate that an additional biological shield should be designed outside an outboard bulk shield. A thick concrete cryostat could be used as a biological shield in this case. A double wall steel cryostat (U.S. and Russia reference options) could be used as an additional shield as well if it is filled by water.

3.1.3 Three possible concepts of ITER biological shield.

The possible approaches to the biological shield concept were discussed at JAERI and are summarized as follows :

1). *The 180-200 cm thick concrete cryostat.*

The thicknesses of the NBI duct wall inside and outside a cryostat proposed for this thick concrete cryostat are ~ 50 cm and ~ 120 cm, respectively. However, it was recognized that in this case the duct wall thickness in a small zone near TF coils shown as "C" in Fig. 3.1 can be only 35 cm. Because of a neutron streaming through this thin wall region, a dose rate 24 hours after reactor shutdown in zone 4 in Fig. 3.2 will be about 10^4 mrem/h [7]. Therefore, the thicknesses of the top and bottom cryostat lids should be ≥ 130-150 cm and the thickness of a steel-rich shield in the zone 1 (see Fig. 3.2) should be ≥ 70-75 cm to allow the personal access outside the cryostat and in zone 2. The design of the zone 3 will be very complicated and include many cavities and slots. Therefore, in spite of the 70-75 cm steel-rich shield in the zone 1, the photon leakage from the zone 4 to the zone 2 is possible. Thus, this problem should be solved for above concept.

2). *The 110-120 cm relatively thick concrete cryostat.*

Proposed thicknesses of the NBI duct wall inside and outside of the relatively thick concrete cryostat are ~75-85 cm and ~120 cm, respectively. The thickness of the top and bottom cryostat lids in this case should be ≥ 60 cm and the thickness of a steel-rich shield in the zone 1 (see Fig. 3.2) should be ≥ 30 cm. The problem of a personal access in the zone 2 in this case would be solved more easily by reducing the required shield thickness in zone 1. The disadvantage of this approach is the difficulty to design a 75-85 cm shield everywhere around the NBI duct (inside the cryostat) because of a restriction on a space for the shield, especially near TF coils.

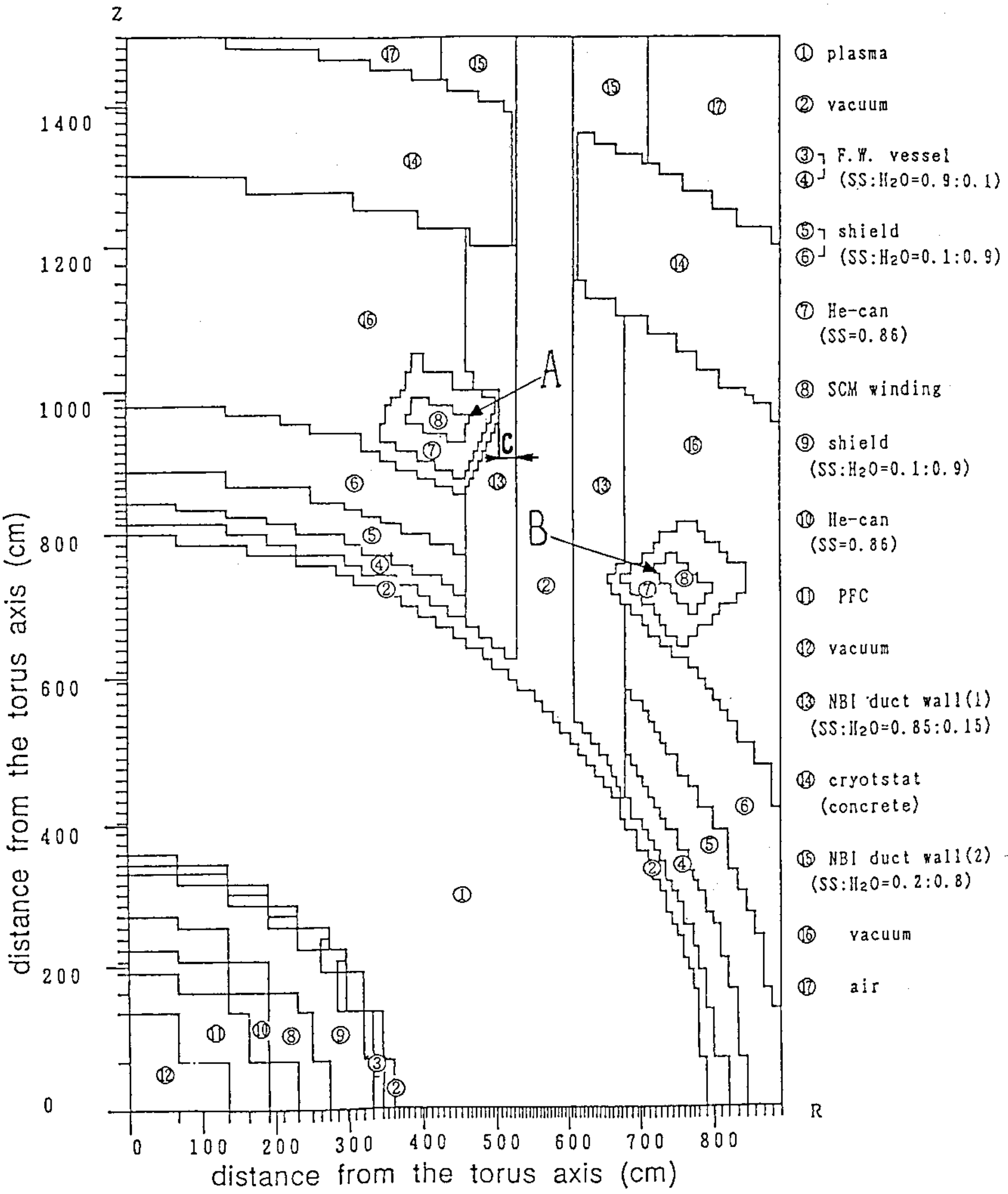

Fig. 3.1 Two dimensional R-Z cylinder model of NBI duct from the reactor torus center to just outside the cryostat

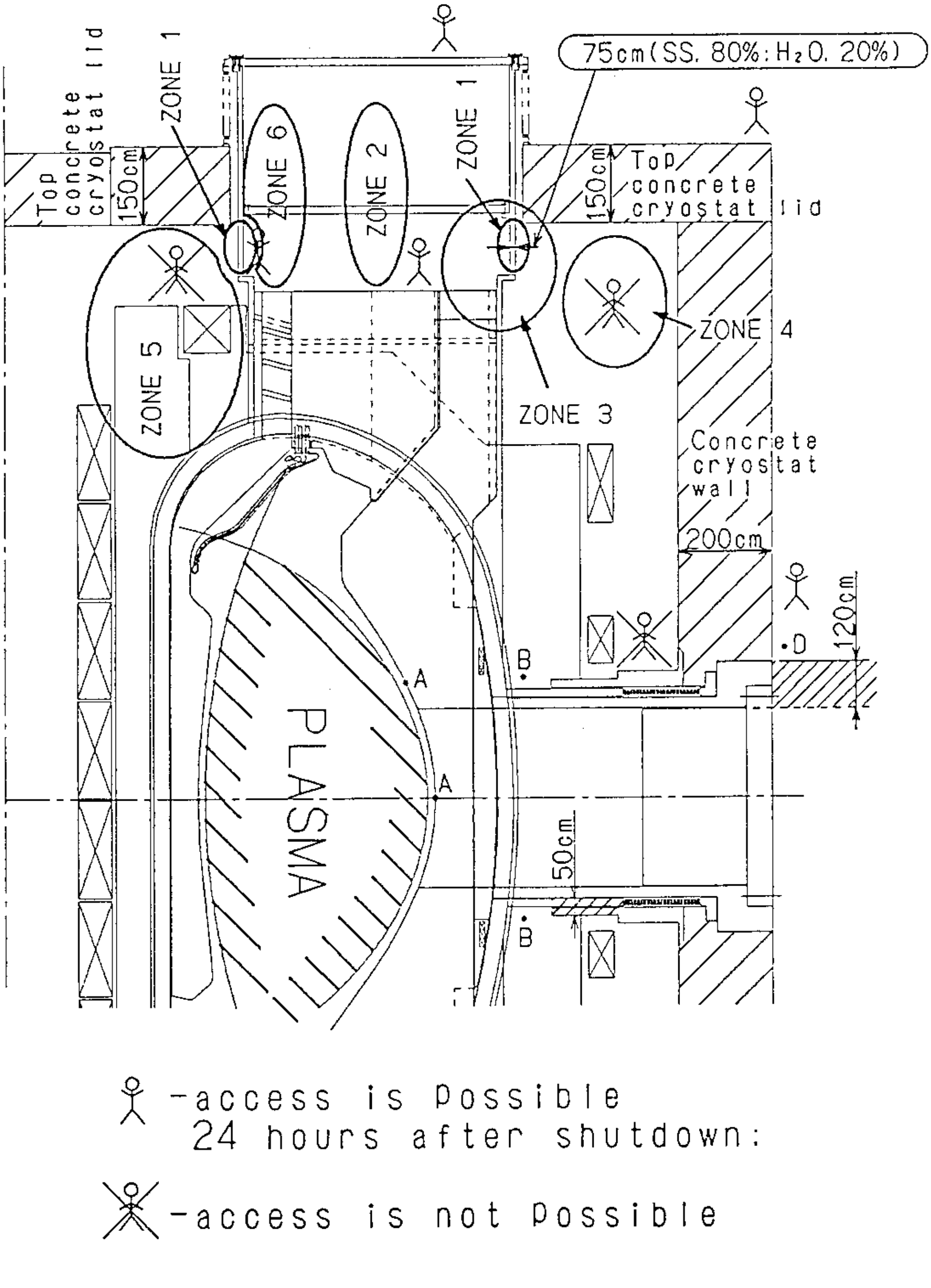

Fig. 3.2 Elevation view of ITER with designs of upper port and cryostat after the CDA

3). *The outboard shield thickness increased by 30-40 cm.*

In case of outboard shield thickness increased up to 180-190 cm, thicknesses of ~100-110 cm and ~120 cm are proposed for the NBI duct wall inside and outside of a cryostat, respectively. The thicknesses of the wall in the zone 1 (see Fig. 3.2), cryostat lids and cryostat side walls could be decreased down to some cm in this case. The wall of upper port (zone 1) is not required a shielding performance, either. This option has similar difficulty for increasing NBI duct wall thickness as above approach, and needs consensus for increasing the outboard shield thickness in the design of whole reactor structures.

Above-mentioned evaluations of thicknesses of top/bottom and side cryostat walls were obtained on the basis of two results from Ref. 7:

1) the biological dose rate in a zone of first wall 24 hours after reactor shutdown is about 10^{10} mrem/h ;
2) every 15 cm of shield which consists of 75% steel and 25% water (steel-rich shield) decreases the biological dose rate by 10 times ;
3) every 30 cm of concrete shield or shield which consists of more than 75% water and, correspondingly, smaller than 25% steel (water-rich shield) decreases the biological dose rate by 10 times.

On the other hand, according to Ref. 8, the concrete cryostat can be a part of the building structure and the minimum thickness of the cryostat is requested to be 1.7 m from the view point of a structural design. It was underlined in Ref. 8 that "the mechanical strength is dominated by the bonding strength between a steel and a concrete in the cryostat structure. Therefore, the strength of a concrete is not affected by the steel fraction in it and, correspondingly, the thickness of the concrete cryostat can not be decreased with increasing the steel fraction."

Therefore, the second and third options of a cryostat mentioned above were excluded from the next consideration. And the only problem to be discussed is the possibility to decrease the total thickness of cryostat from 200 cm (the present size which was recommended on the basis of neutron calculations [7] carried out in Japan during the CDA) to about 170-180 cm (the above-mentioned recommendation from the view point of the mechanical design).

This was confirmed in Ref. 9 where the concrete cryostat option was recommended as a reference one on the basis of two main arguments : 1) The concrete cryostat is preferable from the view point of a biological shield simplicity; 2) An additional building structure is required around the cryostat in the case of a steel double wall cryostat. This corresponds to about 10-15% of a total building area increasing. Therefore, the longer horizontal port is needed. In addition, the layout of devices such as the heating/current drive and vacuum pump is extremely restricted.

3.2 Examination of the TF coil shield effectiveness around the NBI duct.

3.2.1 Introduction.

The analysis of the TF coils shielding effectiveness during operation around the NBI duct was made and presented during the ITER CDA [10]. The particularly high radiation load on the TF coils where the duct passes the rear part of the coil was found. The analysis shows that the difference in a dose between two most dangerous points A and B (see Fig. 3.1) of the TF coils is 31 times. The calculated dose for the point A exceeds the limit by 2.23 times, and for the point B the calculated value is about 14 times smaller than the limit. Therefore, the recommendation was made in Ref. 10 to increase the duct wall thickness next to point A from 20.4 cm to at least 30 cm.

The decision was made after the discussion that the required duct wall thickness would be $\geq$ 35 cm. The corresponding design decision leads to a redistribution of shield thicknesses for two neighboring coils (350 mm and 420 mm next to the point A and B, respectively) which assured acceptable local nuclear responses in the TF coils.

However, some gaps both in a toroidal and radial directions will be included in the design of NBI duct wall as shown in Fig. 3.3 which is under consideration in Japan after the CDA. The additional analysis was done for this part of a shield.

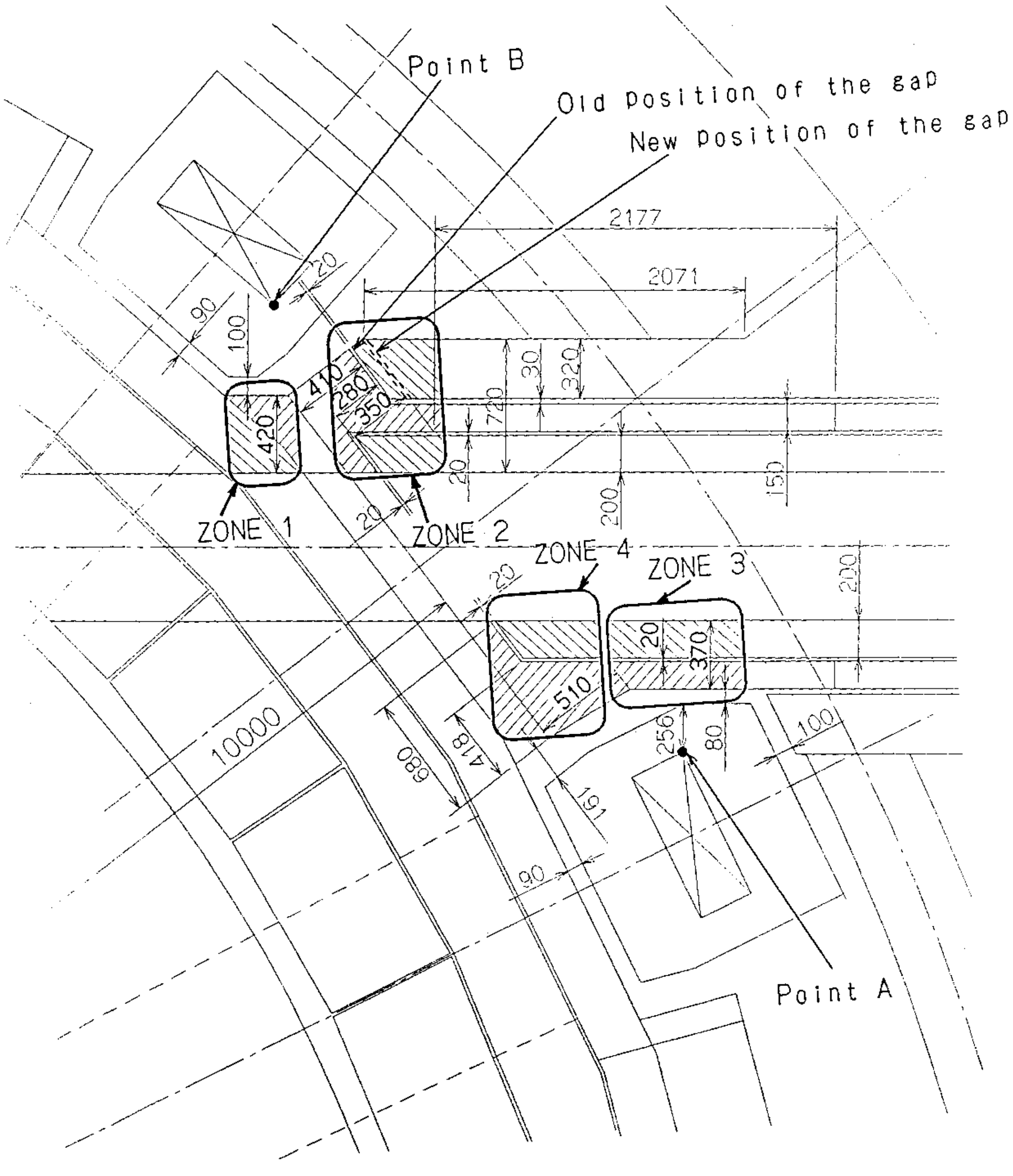

Fig. 3.3 The NBI duct design in the zone of TF Coils
(all sizes are shown in mm)

3.2.2 Analysis for the point "B".

The smallest thickness (42 cm, see zone 1 in Fig. 3.3) of the shield for the point B was not changed in the proposed design in comparison with the reference CDA one [3] , but two 20 mm gaps in the toroidal direction and two (20 and 30 mm) gaps in the radial direction are designed in the zone 2 where the thickness of the duct wall is 720 mm including the gaps.

The effective thickness of the duct wall in the zone 2 is about 681 mm. The gaps in the toroidal direction are not perpendicular to the wall surface. Besides the 350 mm distance between the edges of the gaps is planned. Thus, the relatively thick (but with the gaps) zone 2 seems to be safer from the view point of a magnet shield than the relatively thin (but without gaps) zone 1. Therefore, according to ITER CDA recommendations [11] and above mentioned speculations about the influence of the gaps in the NBI duct wall at the shield effectiveness, the TF coils in the point B will be overshielded.

3.2.3 Analysis for the point "A".

The 20 mm gap in the radial direction is planned in the zone 3 where the thickness of the duct wall is minimal and previously reached 35 cm only. This gap decreases the solid shield thickness in this zone by 20 mm. This deterioration of the shield will rise the dose, heating and copper damage in TF coils by about 25%. To avoid this effect, the total thickness of a solid duct wall in this zone was designed as in a previous design without a gap (35 cm). It was possible because of decreasing the gap between the NBI duct wall and the magnet case from 100 mm to 80 mm. Therefore, the total thickness of the shield of proposed design in the point A (zone 3 in Fig. 3.3) is 37 cm including the 2 cm gap.

The position of the 20 mm gap in the toroidal direction in the zone 4 was considered not dangerous from the view point of a neutron streaming. The thickness of the solid shield in the zone is ~ 680 mm with the 20 mm wide gap penetrated at ~ 262 mm only. Thus, from the view point of the TF coil shield, this zone 4 must be safer than the zone 3.

3.2.4 Conclusions.

The TF coil shield in the proposed NBI duct wall design decreases all relevant nuclear responses below the limit in spite of some gaps in the NBI duct wall in the toroidal and radial directions.

All gaps penetrated the NBI duct wall in the toroidal direction are designed in relatively thick NBI duct wall zones. In zones with a minimal thickness of a duct wall, the shielding capability for TF coils was not deteriorated by keeping total solid thickness equal to the CDA design. However, it should be underlined that the 3D calculations of these zones will be mandatory at a later stage of the ITER EDA Phase of work.

3.3 The examination of a biological shield effectiveness around the NBI duct.

The latest NBI duct wall design proposed by Japan based on the work after the ITER CDA is shown in Fig. 3.4. According to this design, gap widths during ITER operation are relatively small (see Fig. 3.4) and will not exceed 20-30 mm. Besides, there are no straight through gaps in the proposed design, which are most dangerous for the neutron streaming.

The gaps A and B penetrate in the radial direction (parallel to the NBI duct wall) up to the middle of a cryostat which consists of 200 cm thick concrete. These gaps decrease the effective thickness of the duct wall by 20 and 30 mm, respectively, and are not dangerous for the neutron streaming because of 1) very long length of gaps (about four meters); 2) very small width of gaps (20 mm and 30 mm, respectively); 3) the large distance between the gap exits and a reactor room (about 110-160 cm).

The gaps C1 and C2 are more dangerous for the neutron streaming through the NBI duct zone. However, 1) they are not straight through gaps; 2) they are thin gaps (20 mm); 3) the neutron source on the inside wall of NBI duct next to the gaps entrance is very slanting (it was confirmed in [12]) meaning that most of fast neutrons go in the direction perpendicular to the gap axes. The maximum value of a neutron streaming seems to be found for the cryostat zone with the gap C1. It is explained by the smaller distance between the exit from the gap C1 and the external surface of the cryostat than in the case of the gap C2.

The calculational results and analysis of peaking factors of fast neutron flux, total flux and 14 MeV-neutron flux were presented in Ref. 13 for an iron-bearing shield with straight-through gaps of 1-10 cm widths and 100 cm length. These results could be used for evaluation of peaking factors for neutron streaming through gaps C1 and C2. With the isotropic neutron source and the gap of 2 cm width and 100 cm length peaking factors of $\sim 10^5$ and ~ 10 were found in Ref. 13 for the 14 MeV-neutron flux and fast neutron flux, respectively. However, as mentioned above, Ref. 12 indicates that the angular distribution on the internal surface of the NBI duct is nearly slanting in the distance range over 10 m from the entrance of the NBI duct wall (that corresponds the zone of the NBI duct with the gaps C1 and C2). Therefore, the peaking factors of fast neutron flux for gaps C1 and C2 have to be much smaller (< 2) than the ones (~10) in the Ref. 13. This rough extrapolation was based on assumption that, because of a slanting neutron source, the number of fast neutrons penetrated into the entrance of the gap C1 or C2 seem to be in about ten times smaller than in the case of isotropic neutron source.

However, it should be underlined that according to the proposal shown in Fig. 3.4, the total thickness of the NBI duct wall outside the cryostat is 1500 mm. According to the ITER CDA report [3], the total thickness of the NBI duct wall to decrease the dose after reactor shutdown to the limit in the case of the 2-meter concrete cryostat should be ~1200 mm for SS-rich duct wall (80%SS + 20%water) or ~1800 mm for water-rich duct wall (80%water+ 20%SS). Therefore, for the proposed design the biological dose in the reactor room will be below a limit if SS/water composition in the NBI duct wall is appropriately chosen between above two compositions.

Other 5 mm wide gaps in the cryostat structure could not be taken seriously into consideration because of a small thickness and length.

3.4 Proposals for the ITER NBI duct zone design.

Recommendations about the possibility to decrease the NBI duct wall thickness from the design proposed above are shown in Fig. 3.5. These proposals are based on the ITER CDA recommendations [3] and above consideration. According to Ref. 3, the SS-rich duct wall thicknesses of 50 cm and 120 cm for inside and outside of a cryostat, respectively, are enough to keep the biological dose outside the 2 m thick concrete cryostat in the limit. In the case of a water-rich (or concrete) NBI duct wall, the total NBI duct wall thickness should be ≥180 cm outside the cryostat.

Above-mentioned total thickness of the NBI duct wall outside the 2 m concrete cryostat was evaluated on the basis of 2D calculations (using the DOT3.5 code). The uncertainty of this evaluation can be compensated by the safety factor of 10. This corresponds to ~15 cm uncertainty in a solid shield thickness. Thus, the 3D Monte Carlo calculation will be needed during the ITER EDA.

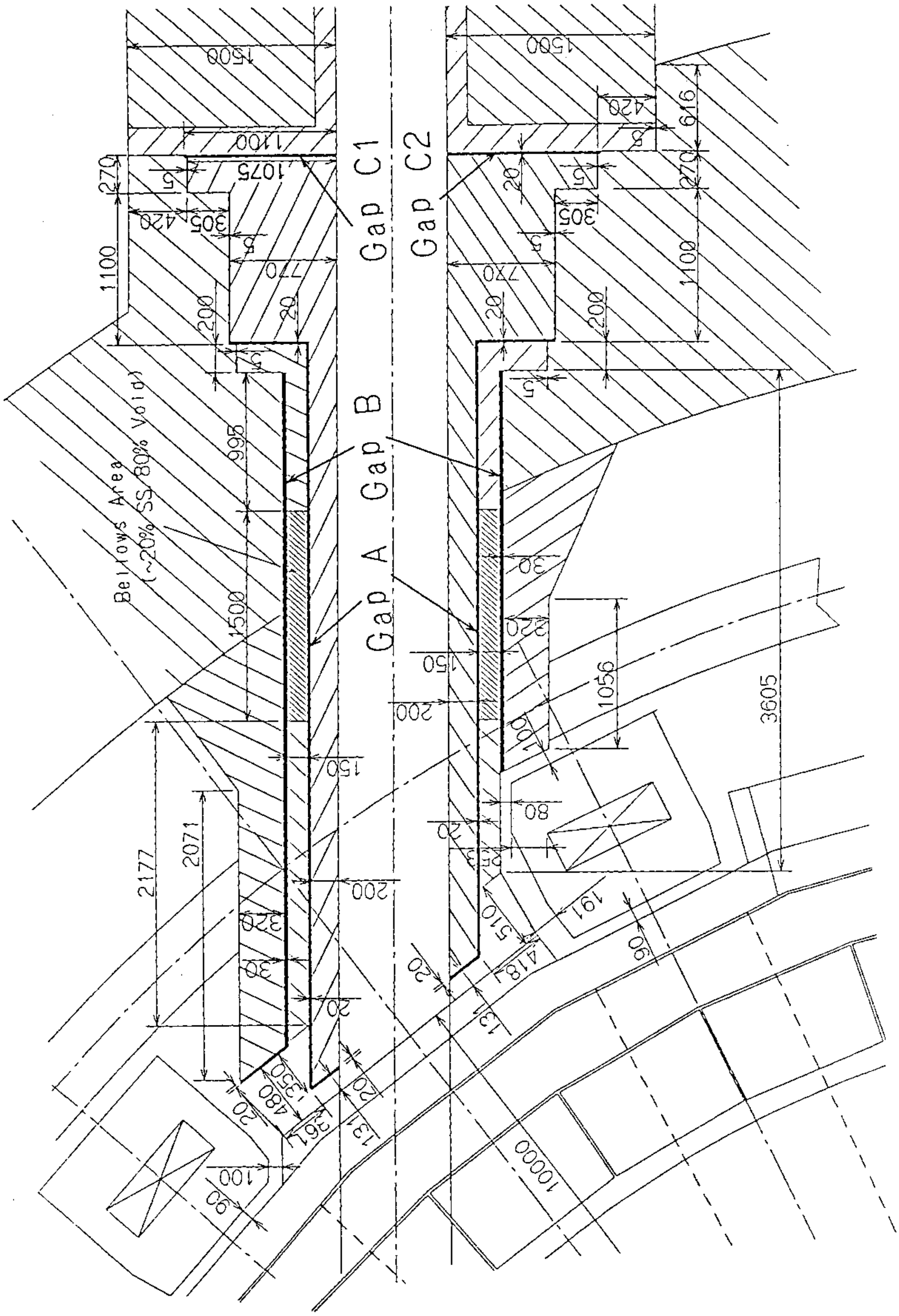

Fig. 3.4 Proposed design of the NBI duct (all sizes are shown in mm)

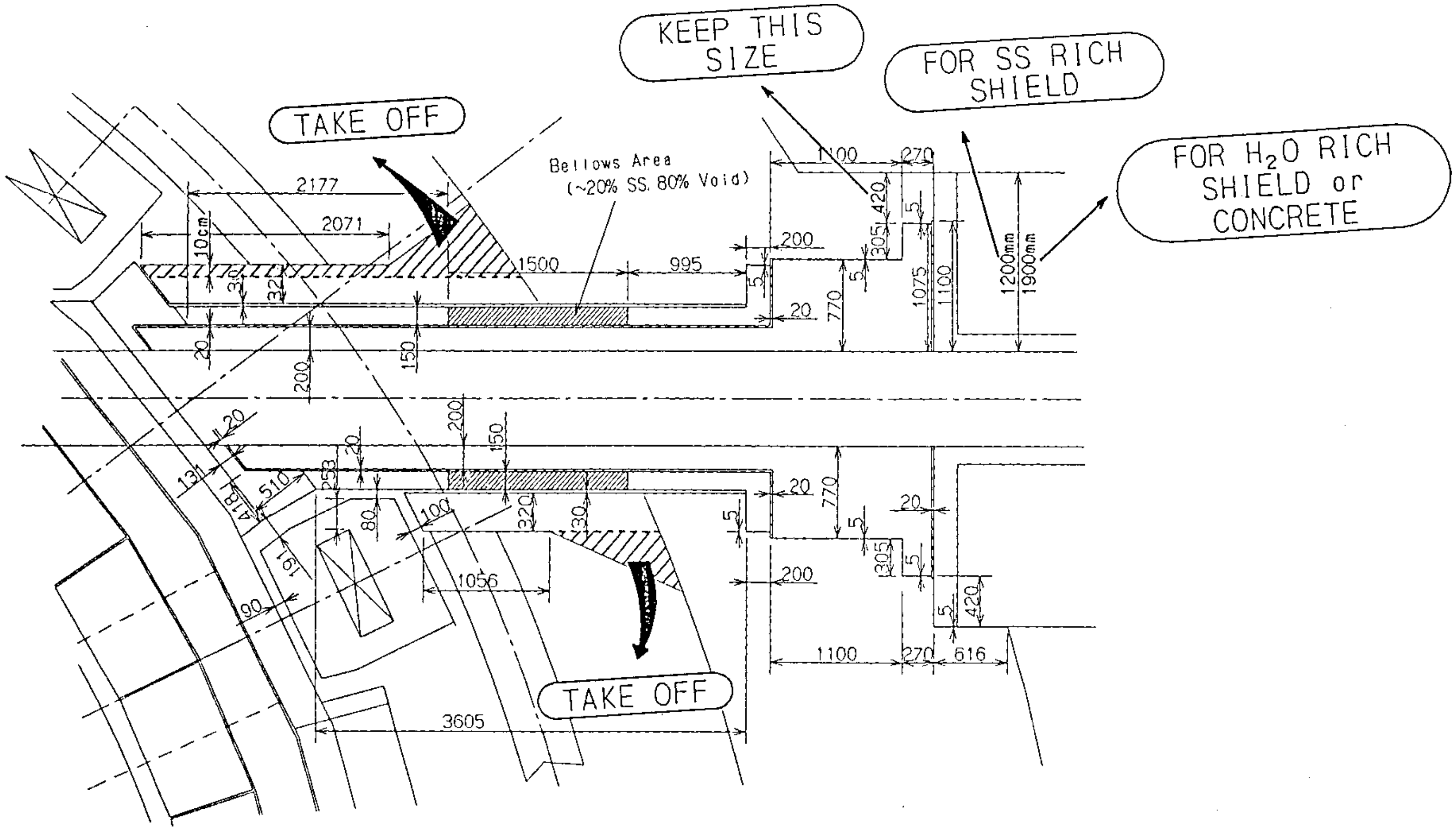

Fig. 3.5 Proposed design of the NBI duct and suggestion for its improvement (hatched zones could be taken off without deterioration of a shield effectiveness; all sizes are shown in mm)

4. TOP/BOTTOM ITER SHIELD PROBLEMS.

4.1 The neutron streaming through the divertor channel (filled with water during operation) and adjacent gap between a channel and shield structure.

4.1.1 Introduction.

A design of the upper port region has been developed in Japan after the CDA. In this design, water coolant pipes and in-vessel coils for separatrix sweeping have been incorporated. Around the pipes for divertor coolant which penetrate through the upper shield plug and around a lead of separatrix sweeping coil provided behind the inboard blanket in the port region, there should be some gaps for remote maintenance.

The neutron streaming through a separate water channel and adjacent gap and the neutron streaming through the gap made for separatrix sweeping coils will be considered here. There are other problems like a neutron streaming through the 20 mm assembly gap in the zone of top/bottom shield plug, influence of water channels etc. Attention to these problems must be paid at a later stage of the ITER design.

4.1.2 The methodic of neutron transport calculations.

Code : DOT3.5

Angular quadrature set : 138

(According to Refs. 12 and 13, the DOT3.5 results with symmetry angular quadrature sets are generally smaller than those of line-of-sight, so the 138 biased up angular quadrature set was used in these calculations)

Geometry : RZ cylinder

Nuclear group constant set : FUSION-40 (42 N , 21 γ)

Number of intervals : 11446 (194 x 59)

4.1.3 The calculational model and the main results.

The position of divertor coolant pipes and gap around them are shown in Fig. 4.1. The calculational scheme is shown in Fig. 4.2. The width and the length of the gap are 1 cm and about 3 m, respectively. A stepwise gap was used to reduce the neutron streaming through the gap. The calculational model was adjusted to a possibility to change the position of a step for gaps. After the optimal step position for the gap was found, the final calculation was carried out. More detail description of these calculations are shown in Ref. 14.

According to these calculations the position of the offset halfway along the gap is the most effective to reduce the streaming effect. The peaking factor of a fast neutron flux can reach about $5{\cdot}10^3$ in this case. In spite of it, the access in the zone 1 (Fig. 4.1) will be possible because of : 1) the sharp decreasing of a fast neutron flux in a radial direction; 2) the large decreasing of a fast neutron flux (by about 13 orders of magnitude) in a ~300 cm top shield plug which consists of 80% of water and 20% of steel. Therefore, the position of the offset halfway along the gap of 1 cm width sufficiently decreases the fast neutron flux and dose for personal access in the zone 1 (see Fig. 4.1).

4.2 The neutron streaming through the empty divertor channel (after reactor shutdown).

In terms of maintenance, it is desirable that the cooling channel of the divertor could be empty after shutdown and the personal access could be provided in the zone 1 (Fig. 4.1) to cut the bellows around the cooling channel. However, the photon streaming from highly activated reactor core through the empty divertor channel will increase the biological dose rate in zone 1 to about 10^6 -10^7 mrem/h (as it was mentioned above the biological dose rate in the reactor core is about 10^{10} mrem/h [7]). Thus, the lead plug of about 20-25 cm thickness has to be provided in the top zone of the channel, as shown in Fig. 4.1, to allow the personal access in zone 1 24 hours after reactor shutdown. This plug would decrease the biological dose rate by about 10^6 -10^7 times. The estimation of the plug thickness was based on the assumption that 3 cm of lead could decrease the photon flux by about 10 times [15]. The stainless steel plug is not effective in this case. The thickness of the steel plug should be increased in comparison with the lead one at least by 50-60 cm.

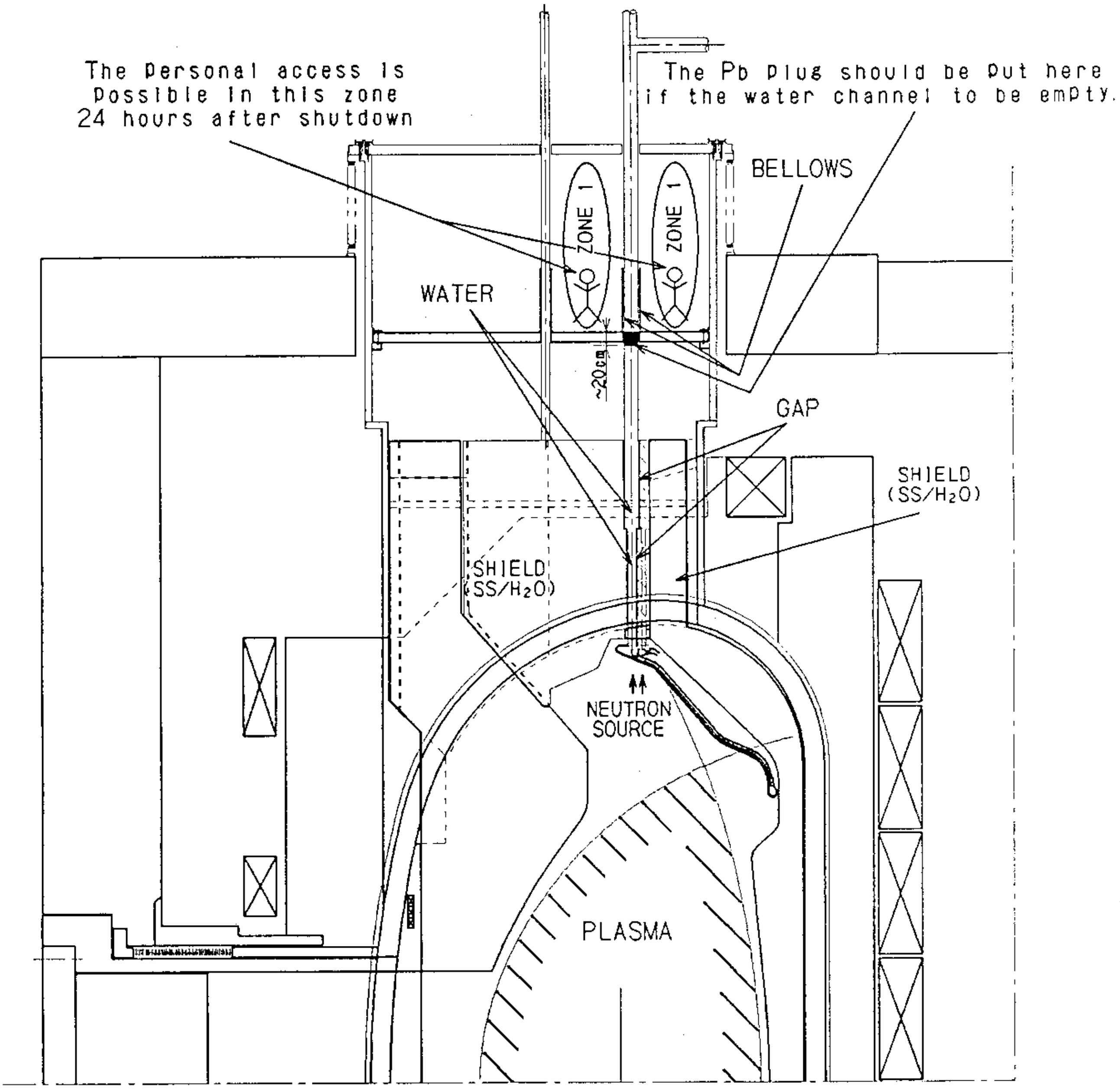

Fig. 4.1 ITER top zone design

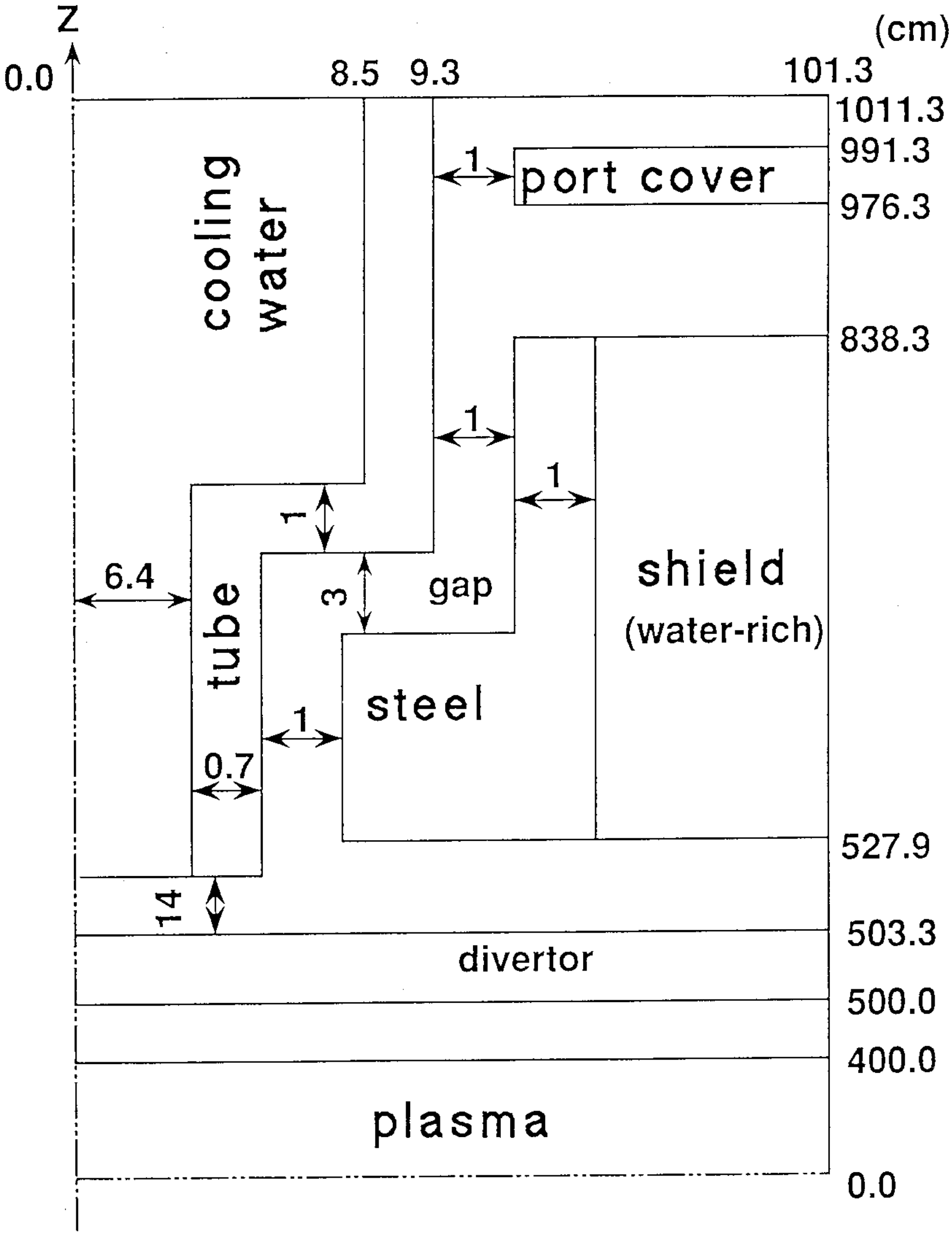

Fig. 4.2 Two-dimensional R-Z model for the analysis of neutron streaming through a gap around divertor coolant pipe

4.3 Gap around a lead of separatrix sweeping coil.

Figure 4.3 shows the top inboard zone designed in Japan after the CDA. A lead for a separatrix sweeping coil is provided along the inner surface of the port wall. There is a 2 cm wide gap around the coil lead for remote maintenance. At least 50 cm of a steel/water shield and a divertor structure corresponding to about 5.5 cm solid material (the mixture of graphite, water and steel) are designed between the gap entrance and neutron source. On the basis of this design, the extrapolation of a biological dose rate at the point A (see Fig. 4.3) could be made. The most conservative case could be marginal. The biological dose rate from the reactor core (10^{10} mrem/h) will be decreased by about 10^6 -10^7 (this extrapolation was based on assumption that every 15 cm of steel/water shield decreases the biological dose rate in 10 times [7]) and 10^4 -10^3 (this extrapolation was based on [12,13] results) times in the gap around the coil lead and in the steel/water shield of 55.5 cm thickness, respectively. Therefore, the biological dose rate in the point A seems to be <1.0 mrem/h.

But it should be underlined that the extrapolation is extremely difficult and thus this result is extremely conservative. The extrapolation of the fast neutron flux was done for the neutron spectrum of a plasma source. In the real situation the neutron spectrum is much more softer and a streaming effect seems not to be pronounced. Another factor is the fact that the gap is not infinitely deep. The real gap's depth reaches only 30 cm as shown in Fig. 4.4.

On the basis of this speculation the biological dose rate at the point A seems to be much smaller than the limit (1.0 mrem/h). Thus, there is no need to design the step in the gap.

However, there is still a small problem which is related to the 2 cm gap between inboard blanket modules (they were marked in Fig. 4.3 as "steel/water shield"). Consequently there will be a small zone in the ITER design were the plasma can be seen from the point A (see Fig. 4.4). The size of this zone is 2cm×2cm. This square straight through channel is formed at the intersection of two 2 cm wide perpendicular gaps. Therefore, from the view point of a shield, this problem could be considered as a neutron streaming through the very long, about 4-6 m, channel of about 2 cm diameter.

The analytical formula (for the cylindrical channel) for line-of-sight neutrons can be used in this case for a neutron streaming evaluation:

$$F = F_o \times (R^2/ 2L^2) ,$$

where R= duct radius, L= duct length, F_o= 14 MeV neutron flux at the entrance of the duct, F= 14 MeV neutron flux at the exit of the duct. According to this formula the 14 MeV neutron flux will be decreased at the exit of the duct only by about 6 orders of magnitude. So the level of the biological dose will be 100-1000 times higher than the limit at point A in Figs. 4.3, 4.4.

However, this high dose is related only to the small zone around the exit from the duct. Thus, a restriction of the personal access within about 60-30 cm distance from the channel exit or a lead plug of about 7-10 cm diameter and about 10 cm of height placed at the top of the gap could reduce the dose down to the limit for personal access at point A.

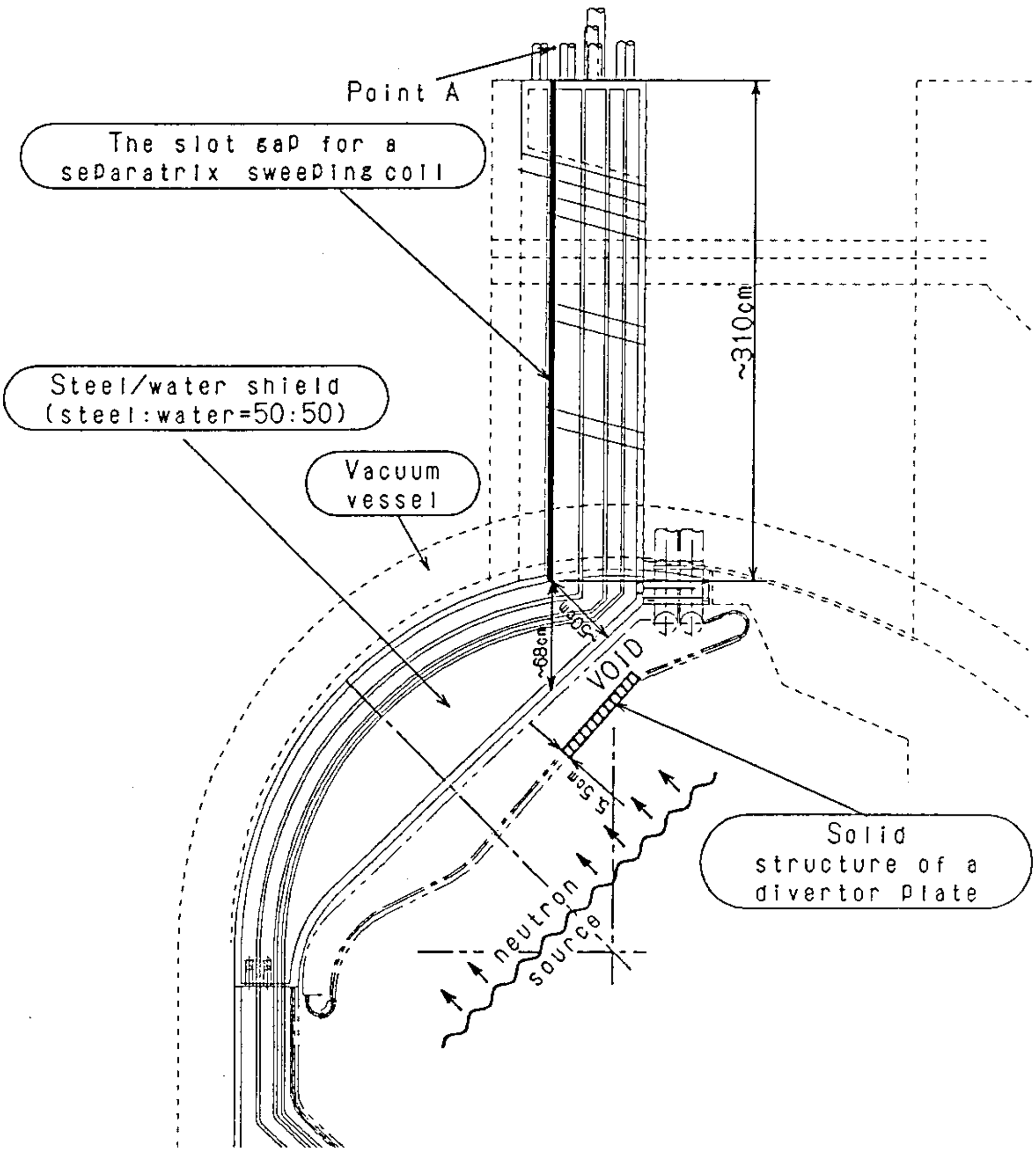

Fig. 4.3 ITER top inboard zone design

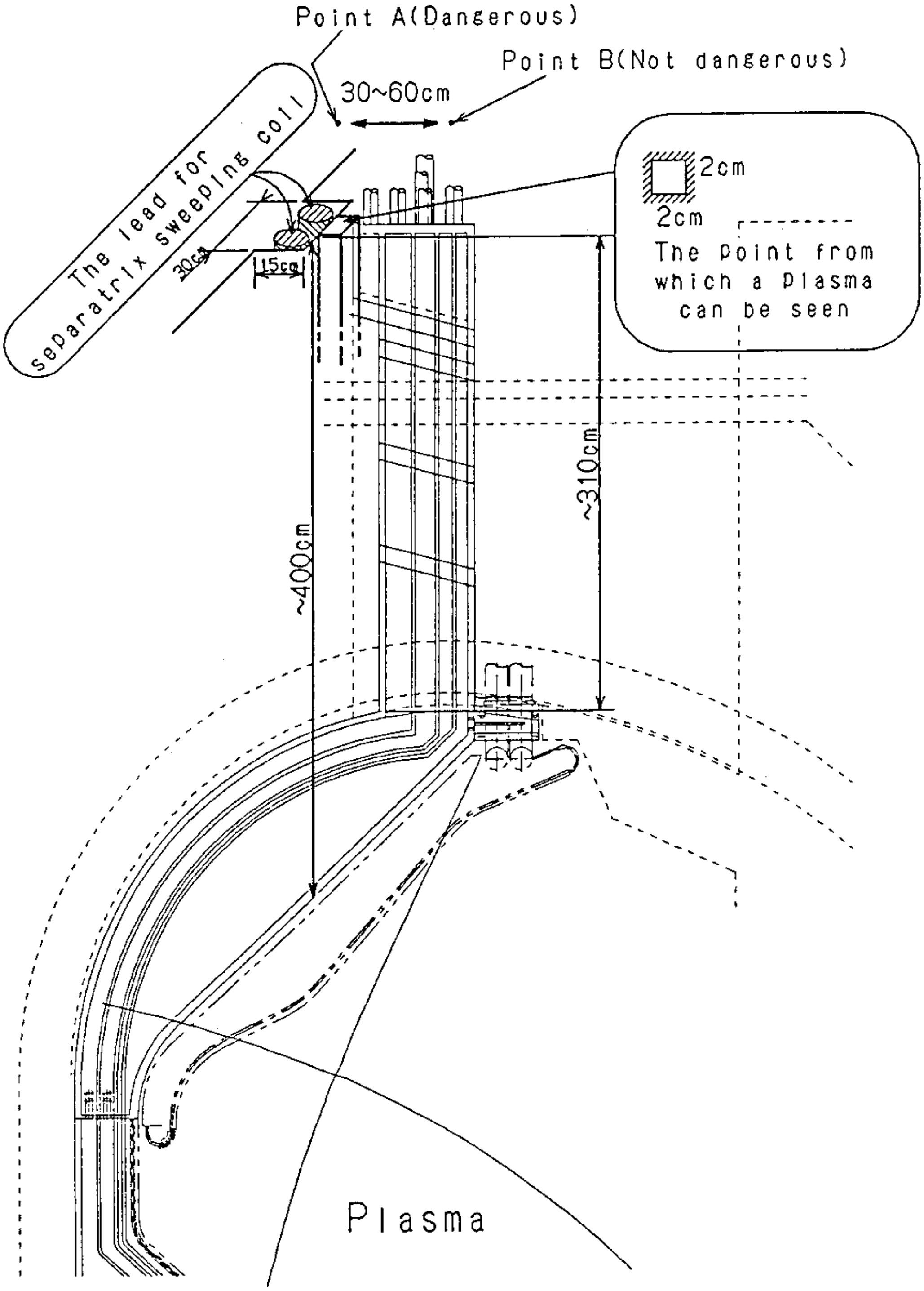

Fig. 4.4 ITER top zone design

5. RECOMMENDATIONS TO BE TAKEN INTO CONSIDERATION DURING THE MAINTENANCE OF ITER.

5.1 Thickness of the bottom shield plug.

Two options of a desirable decreasing (from the view point of a divertor remote maintenance) of a bottom shield plug thickness are shown in Figs. 5.1 and 5.2. Options 1 and 2 decrease an average total shield thickness in the zone 1 from about 66 cm in the CDA design to about 55 cm and 58 cm, respectively. It could increase a total heating in the TF coils during the Physics Phase from 1.4 kW [12] to about 7 and 4 kW (safety factors [3] are included), respectively, which correspond to about 15% and 8% of a total heating increasing, respectively. It seems difficult for these modification to be allowed because the present total heating ~55 kW is already closed to the limit [3,12].

The third option is shown in Fig. 5.3. An average total shield thickness in the zone 1 in this case is about 62 cm. This could increase a total heating in TF coils in this zone during the Physics Phase to about 2.5 kW (the safety factor is included). Thus, an increase of the total heating in TF coils is about 1 kW that corresponds to about 2% of a total heating in all 16 TF coils. The third option can be recommended for the discussion in the ITER EDA Phase of work.

5.2 Wall thickness of the maintenance cask.

During remote maintenance of in-vessel components, a cask as shown in Fig. 5.4 is used to prevent the tritium and radioactive dust from being released to the reactor room.

If the personal access around the cask is planned, it must be designed also from the view point of the biological shield because, in this case, it must decrease high energy photon fluxes from the highly activated in-vessel components, e.g., by about 10 orders of magnitude for blanket module to decrease the level of the biological dose outside the cask to about 1 mrem/h.

The desirable wall thickness of the cask from the view point of biological shield strongly depends on the wall material. The heavy elements like Pb, Fe etc. are most effective for decreasing a photon flux. The lead is the most effective material because of the largest μ value in all energy range (μ is the coefficient of photon flux decreasing in a material). A 25-30 cm lead wall seems to be enough for the maintenance cask. This recommendation is shown in Fig. 5.4.

In the case of a stainless steel wall, the thickness of the maintenance cask must be increased up to the 50-60 cm.

Therefore, to design the cask from the view point of biological shield is very difficult because of its large size and, correspondingly, large weight. More detail discussion and neutronic calculations during ITER EDA are to be carried out if the decision will be made to design the cask from the view point of biological shield.

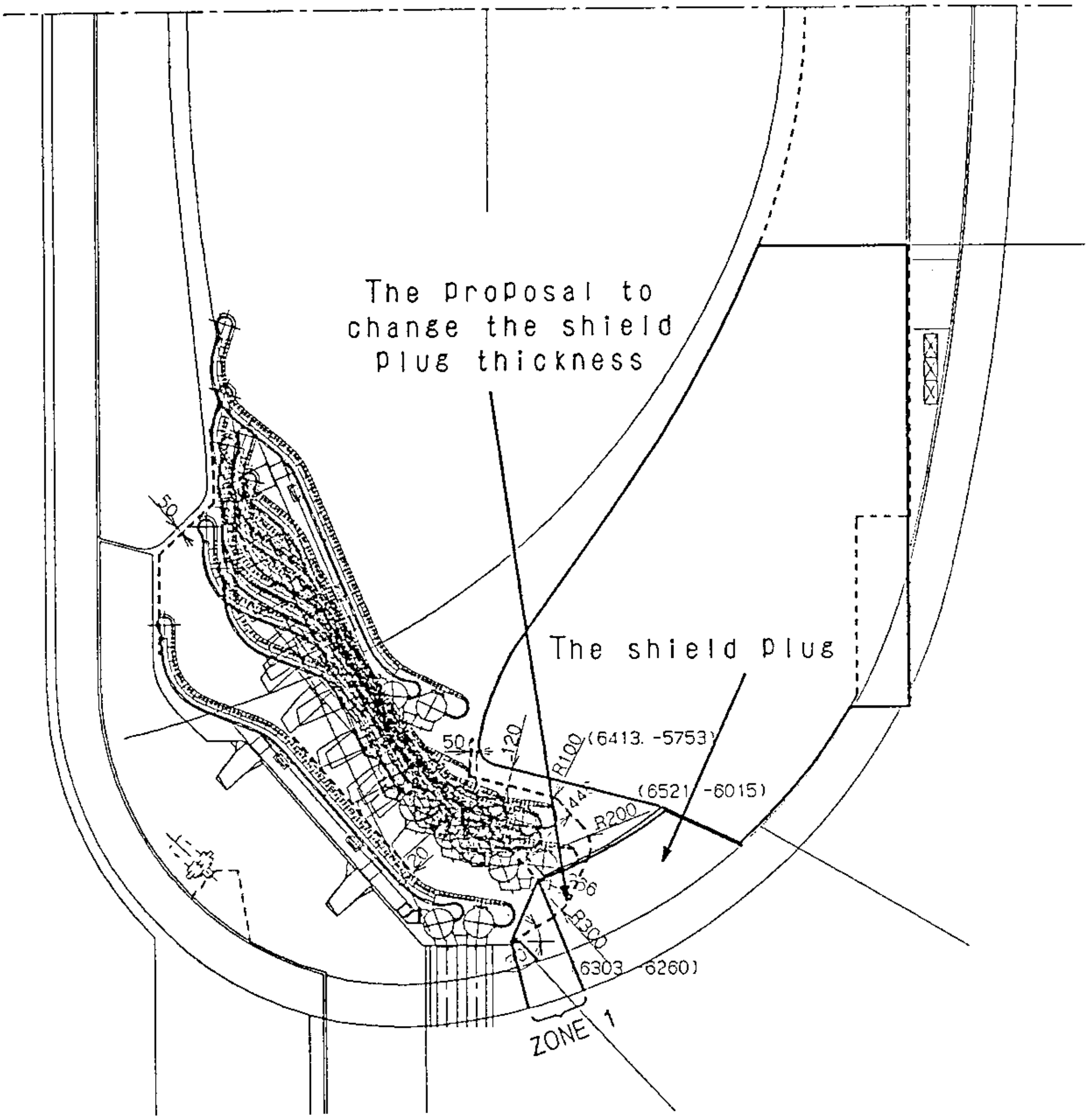

Fig. 5.1 The desirable decreasing of the bottom shield plug thickness from the view point of a divertor remote maintenance (option 1)

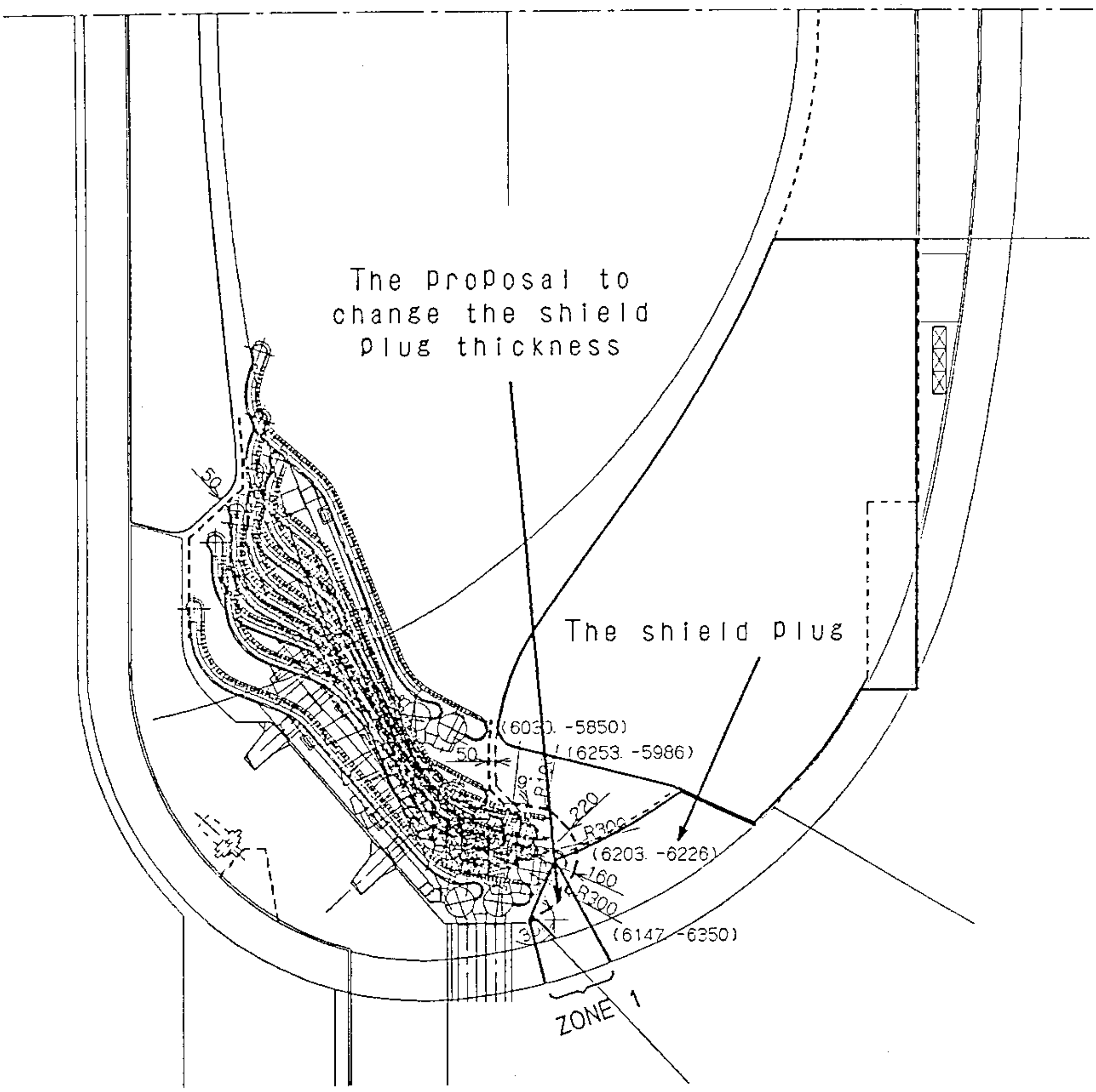

Fig. 5.2 The desirable decreasing of the bottom shield plug thickness from the view point of a divertor remote maintenance (option 2)

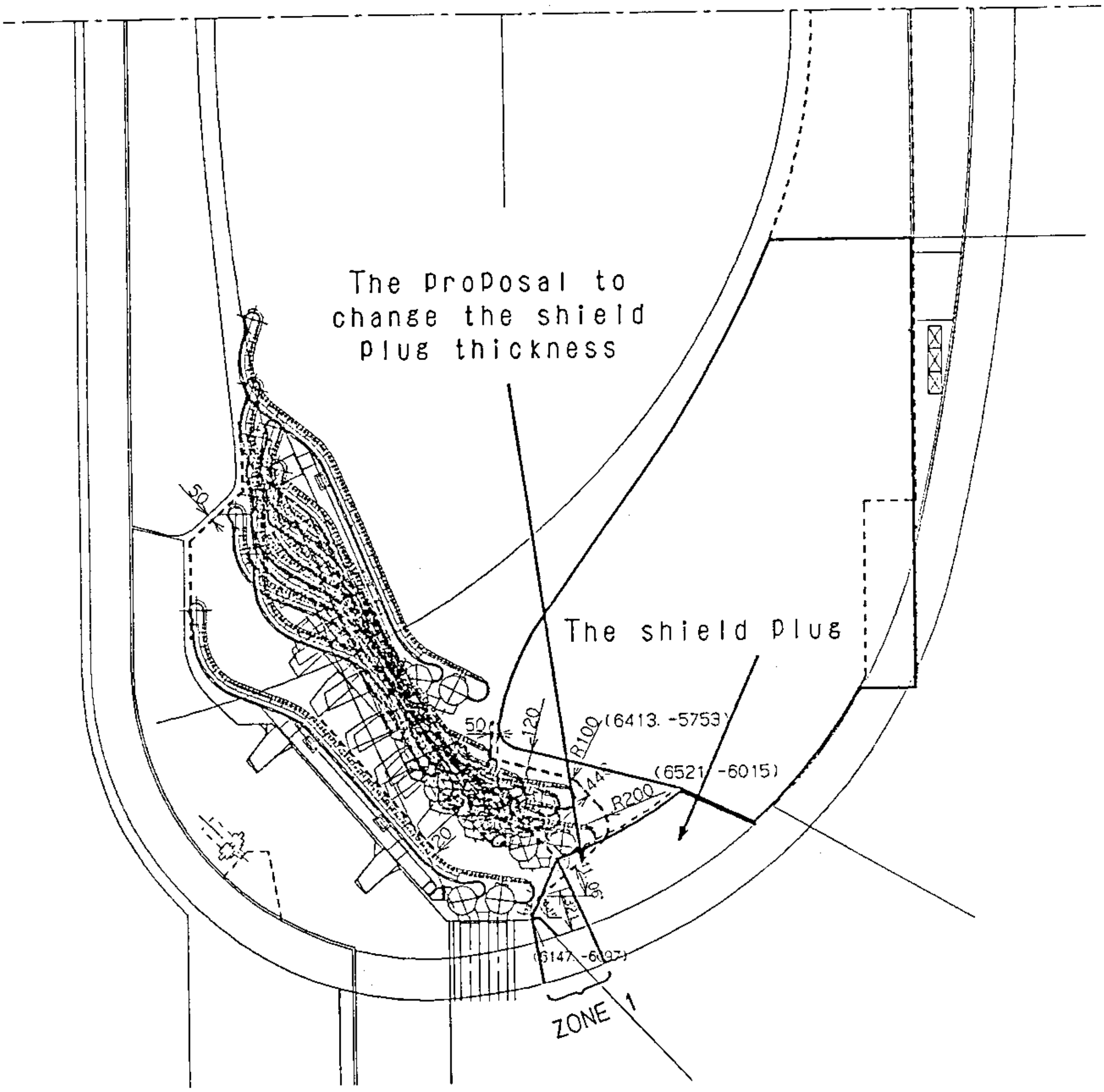

Fig. 5.3 The desirable decreasing of the bottom shield plug thickness from the view point of a divertor remote maintenance (option 3)

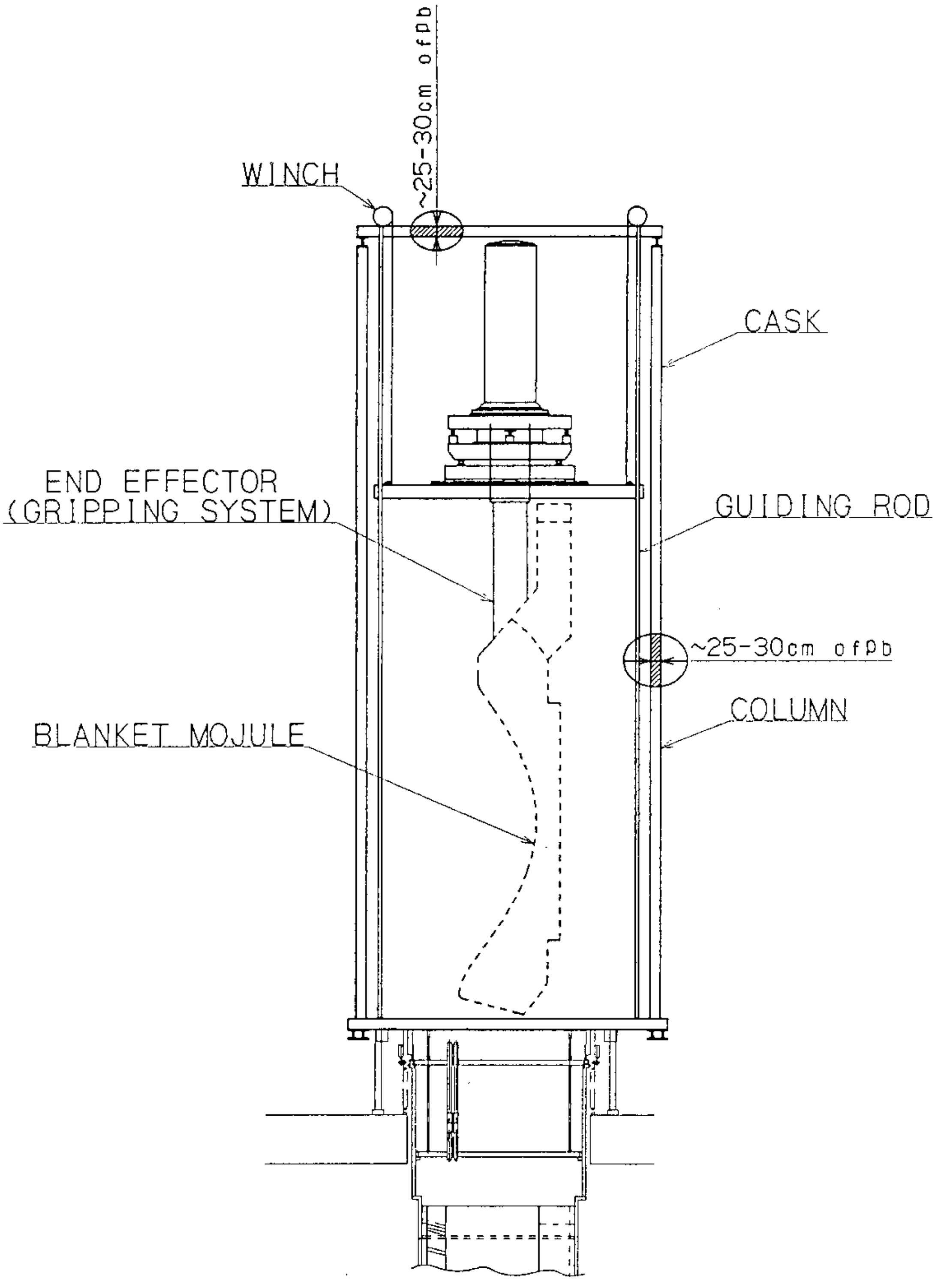

Fig. 5.4　Blanket module handling system

6. CONCLUSIONS.

1) Japanese option of an inboard pebble bed ceramic breeder blanket consists of about 40% of void. But because of a small thickness (about 10-12 cm) of this blanket, the decreasing of a shield effectiveness is not so large and reach about 20% and 40% in comparison with Russia ($Li_{17}Pb_{83}$ blanket) and U.S. blanket (block ceramic breeder blanket) options, respectively.

2) The design of TF coils in the inboard zone of ITER was performed without gaps between coils. This means that the problem of a biological shield in the inboard reactor zone is not so severe as for the outboard zone and seems to be solved relatively easy in coming ITER EDA Phase of work.

3) Results of a biological shield analyses were not clearly indicated in the ITER CDA summary report. The consensus should be reached on this item including a cryostat type and a thickness of a cryostat to be used in ITER.

4) The main difference in Japanese approach to the ITER biological shield in comparison with U.S. and Russia ones is the idea to joint functions of a cryostat and a biological shield by using concrete. A concrete cryostat is preferable from the view point of a biological shield simplicity and building design.

5) The reference Japanese option is the 180-200 cm thick concrete cryostat. This cryostat is to be used as a part of a building structure. The minimal thickness of the cryostat based on required mechanical strength is 1.7 m and can not be decreased by increasing a steel fraction in the cryostat because the mechanical strength of the cryostat is dominated by the bonding strength between steel and concrete.

6) In spite of some gaps in NBI duct walls both in toroidal and radial directions, the dose to TF coils insulator will be in a limit after 3 $MW \cdot a/m^2$ operation.

7) The 180 cm water-rich or 120 cm steel rich shield around the NBI duct outside a concrete cryostat is effective enough to keep a biological dose below the limit in spite of some gaps in the cryostat and the shield.

8) The neutron streaming through a water channel and adjacent 1 cm gap between the channel and shield plug structure could be an obstacle for personal access in the top zone of ITER. The position of the offset halfway along the gap is the most effective to reduce the streaming effect. It could decrease a fast neutron flux and biological dose below the limit. The neutron streaming through the water channel is not pronounced because of the small channel diameter (~15 cm) and a rather large height (~300 cm) of a water-rich shield plug.

9) The neutron streaming through the 2cm×2cm square channel (connected with an installation of separatrix sweeping coils) of about 4 m length will be the obstacle for the personal access in the top zone. However, a 10 cm height lead plug of about 7-10 cm diameter placed at the top of the gap could sufficiently reduce the photon streaming effect.

10) The total heating in TF coils is very sensitive to the thickness of the bottom shield plug. Thus, the proposal of decreasing the average thickness of some part of this plug from about 66 cm to about 62 cm in terms of a divertor plate maintenance could increase the total heating in all 16 TF coils by about 1 kW. More decreasing of the shield thickness seems to be impossible because of a sharp increasing of a total heating in TF coils.

11) The 25-30 cm lead wall of the maintenance cask sufficiently decreases the biological dose rate below the limit for personal access outside the maintenance cask.

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APPENDIX 1. A biological shield criterion.

The 0.5 mrem/h biological dose 24 hours after reactor shutdown was indicated in ITER CDA summary report [3] as a reference criterion. Previously this value was 5 times larger (2.5 mrem/h). Such a decreasing in the criterion value will require more concrete biological shield thickness by ~ 10 cm. This corresponds to ~750 - 1000 tons of cryostat weight increasing.

Based on International Commission on Radiological Protection (ICRP) recommendations in 1991, in which the occupational dose limit was revised from the previous recommendation (50 mSv/y) to 20 mSv/y, the biological dose limit has been decreased from 2.5 mrem/h to 1.0 mrem/h as a temporary guideline in this investigation.

APPENDIX 2. Comments about the possibility of ITER parameters reconsideration.

It should be underlined that new ITER parameters reconsideration was proposed by Japan [16]. The proposal includes an increase of major radius by 25 cm to solve insufficiency of thickness in 15 cm for central solenoid coils, 5 cm for toroidal coils in terms of mechanical strength, and 5 cm for inboard shield.

In addition, the neutron fluence of 1 $MW \cdot a/m^2$ was recommended as a criterion for the ITER EDA Phase of work.

The U.S., Russia and the EC have shown close views on these items [16]. The implementation of above-mentioned proposal would increase the effectiveness of the ITER shield and would help to solve some problems raised up in this report.