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PROGRESS IN SHIELDING DESIGN IN THE DRAGON PROJECT

by

D. B. CRAWLEY

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## 1. INTRODUCTION

The Dragon Reactor Experiment is a 20 MW High Temperature gas cooled reactor being built as a joint O.E.C.D. project at the United Kingdom Atomic Energy Establishment at Winfrith, Dorset.

The primary objective of the Dragon reactor is to provide design and operating experience of a reactor of this type before building a full scale power generating plant. Major objectives include obtaining general physics information, testing fuel elements of advanced design, and controlling the release of fission products in an active primary coolant circuit.

The calculations for the Dragon reactor shield design were initiated in 1957, based largely on simple semi-empirical methods of comparison with existing reactor shields at Harwell. Later computations are based on the method of calculation developed by the Shielding group of Reactor Division, Harwell.

This paper sets out to summarise the work that has been carried out to solve the shield design problems encountered, and also the preliminary investigations into the shielding requirements for a 1250 MW High Temperature power reactor design study.

### 1.1 General Description of the Dragon Reactor Experiment

The Dragon reactor experiment is a graphite moderated helium cooled reactor. The enriched uranium fuel is in carbide form, dispersed in a graphite matrix. The moderator is used both as cladding and for structural support of the core.

The power output is 20 MW thermal, which is not used to generate electricity but is dissipated via secondary and tertiary heat exchangers, through an air cooler.

Provision has been made in the reactor design for the release of fission products from the fuel, the rate being controlled by diffusion through the graphite. It is probable that fuel inserts of a fission product retaining character will be employed. This has been achieved by coating the fuel particles with an impermeable layer of pyrolytic carbon. Any release of fission products is swept into a purge stream, which passes through traps into a delay and clean-up system, consisting basically of cooled charcoal beds where impurities are adsorbed and active fission products are allowed to decay.

The reactor, fission product plant, and the secondary coolant circuit are all contained within a leak tight steel containment shell, designed to contain any release of activity from the primary circuit. The shell is enclosed by a concrete building which also provides shielding in the event of an emergency release of activity into the containment.

The shield description is given in Section 2.2. It was originally envisaged that it might be necessary to utilise shield materials capable of withstanding temperatures of 300-400°C, but subsequent design has made it possible to avoid the use of materials other than graphite or steel in any hot regions.

General data relating to the plant, and the shielding is given in Appendix I.

## 2. ASSESSMENT OF THE DESIGN OF SHIELDING FOR THE DRAGON REACTOR EXPERIMENT

The main fundamental departure from normal shielding design, arises from the release (or potential release) of fission products into the primary coolant circuit. Furthermore under accident conditions the activity could be released to the inner containment.

In addition, severe limitations are imposed on the impurity concentrations in the helium coolant, because of corrosion of the core graphite and fuel at high temperatures. It is therefore not permissible to employ materials with an appreciable vapour pressure inside the pressure circuit. For example, concrete cannot be used inside the pressure vessel.

### 2.1 Historical Background

Initial shielding calculations were carried out by Harrison and Absalom at Harwell in 1957 [1] for the HTGC Reactor experiment proposals. These proposals were subsequently modified, and the shield design was considerably changed although certain features such as the 15" thick steel-water thermal shield were retained. In the early conception, the fission product delay and clean-up plant was situated in a separate building, so its inclusion within the reactor containment shell introduced the idea of providing primary and secondary shielding for the reactor core.

The calculations for the design of the Dragon Reactor Experiment shield were performed in 1960 by the Author of this report, and are reported in [2]. There have, not surprisingly, been a few minor modifications since then, necessitated by engineering and other reasons, and a final shielding report will be published in a few months time. Figs. 1, 2 and 3 show the layout of the Dragon Reactor Experiment and its shielding.

### 2.2 Basis of Shield Design

The shield design has as its basis the fulfilment of the following three requirements.

- (a) During operation, uncontrolled access is required to the plant rooms outside the containment shell.
- (b) After shutdown, access is required to plant within the shell, except within designated areas. In certain areas, e.g. in the fuel element transfer chamber, limited access is required for a few planned maintenance operations.

- (c) After an accident involving the release of activity to the containment, there shall be no radiation hazard to the general public, or restrictions imposed on the remainder of the site. (Fulfilment of this last condition also renders it unnecessary to evacuate the control building.) It must also be possible to evacuate personnel from the reactor building without exceeding the maximum allowable dose (25 Rem).

To satisfy the requirements the Radial shielding is as follows:-

(i) Primary Reactor Shield

(a) Graphite reflector	90 cm	(3 ft 0 in)
(b) Steel (shield and P.V.)	10 cm	(4 in)
(c) Thermal shield (steel-water)	38 cm	(15 in)
(d) Concrete (primary shield)	175 cm	(5 ft 9 in)
(e) Steel (containment vessel)	2.5 cm	(1 in)
(f) Concrete (Secondary shield)	61 cm (minimum)	(2 ft 0 in)

(ii) Containment Shield

(a) Steel	2.5 cm	(1 in)
(b) Concrete	61 cm (minimum)	(2 ft 0 in)
(c) Concrete	61 cm	(2 ft 0 in).

It should be noted that items (i)e and (i)f are also (ii)a and (ii)b.

Top Axial Shield

The top shield plug thickness had been fixed in an earlier design, assuming that normal or heavy concrete would be used. Subsequently when this was precluded by the stipulation that there should be no hydrogenous material inside the primary circuit, the graphite-steel shield was designed to reduce the radiation level in the transfer chamber to a minimum. The steel graphite proportions were chosen as a compromise for shielding from shutdown gammas (core and activation gammas from inside the shield) and for shielding neutrons during operation which would lead to activation outside the shield.

A summary of the thickness of other parts of the reactor shield, and ancillary shielding is given in Appendix I. The assumptions made in assessing the shielding requirement may appear mutually inconsistent. Because there was considerable doubt at the design stage on the release of fission products from the fuel, it was necessary to make

assumptions that would always be pessimistic. For example, in the discharge of fuel elements 100% retention is assumed, in the fission product delay and clean-up plant the release is assumed to be the maximum for which the plant has been designed.

In the design of the containment shield, it was assumed, that the fission product release is 100% of noble gases and halogens, and 25% of all other fission products. A treatment of a similar case is given in [3].

## 2.3 Methods of Calculation

In the approach to the design of the Dragon reactor shield, the following principles were adopted:-

- (i) The design method should be as simple as possible to limit the cost of design.
- (ii) Standard methods should be used to avoid the need for development of new computer programmes.
- (iii) No attempt should be made in refining the standard methods of calculation, because the complex geometry of the Dragon shield layout renders the accuracy of the calculation uncertain.
- (iv) The use of expensive materials should be avoided if possible.

The uncertainties of calculation lie principally in the attenuation of neutrons in the reactor shield, in the activity source strength in the fission product plant, and deposited activity in the coolant circuit. Fortunately under operating conditions the first of these is unimportant as the dose rate at the shield surface results principally from secondary gamma radiation arising from neutron absorption in the steel of the core support structure and the pressure vessel. The region in which there is greatest uncertainty is in the transfer chamber, (see Fig. 1) where the dose rate after shutdown determines the permissible access time. This is not important however as access should only rarely be necessary, and the entry may be delayed to allow for induced and fission product activity to decay.

In general, calculations have conformed to the recommendations made by the Shielding Group, A.E.R.E. Harwell, in [4].

### 2.3.1 Reactor Shielding

The primary radiation source is fission neutrons generated in the core, and the assumed core distribution, which was evaluated for a hot unpoisoned core using a 2 group diffusion code, is given in Figures 4 and 5.

#### A. Neutron Attenuation

Prediction of the neutron attenuation is made using an energy dependent "first flight" kernel to calculate the removal flux of fission neutrons, and multigroup age - diffusion equations to describe their subsequent slowing down, diffusion and capture.

The fission neutron source is calculated from the known fission density in the core. The boundary flux condition for the slowing down neutrons at the core-reflector interface (used as the starting point for the calculations) is calculated assuming a  $1/v$  energy spectrum (epithermal energy to 2 MeV) for the "fast" neutrons evaluated by the 2 group method in the core and reflector.

The removal flux at a distance  $r$  from a point source is given by:-

$$\phi(r) = \int \frac{F(E) \exp(-\Sigma_{rem}(E) r) dE}{4\pi r^2}$$

and the source strength of neutrons leaving the removal flux is:-

$$A(r) = \int \frac{F(E) \exp(-\Sigma_{rem}(E) r) \Sigma_{rem}(E) dE}{4\pi r^2}$$

where  $\phi(r)$  and  $A(r)$  are the flux and removal source strength of "penetrating" neutrons.

$F(E)$  = spectrum of neutron source.

$\Sigma_{rem}(E)$  = Energy dependent removal cross section.

For details of the computer programme used to evaluate the flux and source strength, see [5].

The removal source strength  $A(r)$  is considered as the source neutrons for the 1st group of the multigroup age diffusion equations, and in the method of [4] is assumed to be isotropic in the laboratory system. Whilst this is true for scattering by heavy elements some inaccuracy is introduced for a shield material such as water, due to a preponderance of forward collisions. In [6] it is suggested that the removal collision density can be multiplied by a factor of 1.35 to give the source density. In the context of the Dragon shield design this was considered an unnecessary refinement, as the steel water shield secondary gamma source makes an insignificant contribution to the external dose rate.

In the initial calculations, simplification was made by assuming the steel water shield was homogeneous. Subsequent calculations indicate that whilst the local flux variation is considerable the overall effect on neutron flux leaving the concrete shield is unimportant, and on the total dose rate is negligible. (See Fig. 6)

The general form of the multi-group equations used in the

second part of the neutron attenuation calculations is:-

$$\frac{d^2 \phi_i}{dx^2} + \frac{p}{x} \frac{d \phi_i}{dx} - (K_i^2 + \alpha_i^2) \phi_i + \psi_i = 0$$

A set of equations of this type for each energy group, applies to each shield region, with appropriate correlation of boundary conditions at the interfaces

$\phi_i$  and  $\psi_i$  are the neutron flux and source strength for the  $i$ :th energy group.

$K_i$  is the inverse slowing down length (or thermal diffusion length).

$\alpha_i$  is the parameter defining epithermal absorption and/or buckling.

$p$  is 0, 1, 2 for plane cylindrical or spherical geometry respectively.

$$\text{for } i = 1, \psi_i = \frac{A(x)}{D_1}$$

$$i > 1, \psi_i = \frac{D_i - 1}{D_i} K_i^2 \phi_i - 1$$

These sets of equations are solved using the RASH programme in Autocode for the Ferranti Mercury computer, [4], the source term for the 1st energy group being fed in from the output of a previous run of the SHARN programme [5], as above.

Parameters for the components of the Dragon reactor shield are given in [2], Appendix II.

## B. Gamma Attenuation

Sources of gamma radiation during operation are principally:-

- (i) From the reactor core - mainly fission gammas and from the decay of fission products. These latter constitute the major source after shutdown.
- (ii) From absorption of thermal neutrons in the reactor shield. These secondary gammas form the largest proportion of those penetrating the shield.
- (iii) Inelastic scattering of fast neutrons. As these are less than about 1% of those from absorption they have been neglected.

In general gamma radiation penetration can be evaluated by straightforward numerical methods from the relationship for

the flux from an isotropic point source

$$\phi = \frac{S \cdot B(E, \mu r) \exp - \mu \cdot r}{4\pi r^2}$$

where S photons/sec is the source strength

$\mu$  is narrow beam absorption coefficient for intervening medium

r is distance from source

B(E,  $\mu r$ ) is the build-up factor related to the gamma energy, absorbing medium, and shield thickness.

In the case, however, of secondary gammas generated throughout the shield, the amount of numerical calculation tends to become excessive, so the computation is done by machine - using the GASH programme on the Ferranti Mercury computer, see [7]. This programme is used in conjunction with the neutron shielding programme RASH.

For manual computation, it is convenient (if possible) to use a build-up factor in the form

$$B = \alpha \cdot \mu \cdot r$$

By this means the expression for the flux at the shield surface, or the heating at a point in the shield is considerably simplified.

It is generally assumed that the source is an infinite plane, and a cylindrical or spherical correction is applied to correct for the geometry. The result for a uniform slab, containing an exponential distribution of isotropic plane sources, shielded by a second uniform slab is given by:-

$$\phi = \frac{S_0 \alpha}{2} \left[ \frac{\exp(-\mu_1 t_1) - \exp(-K t_1)}{K - \mu_1} \right] \exp(-\mu_2 t_2)$$

where source in slab 1 is  $S_0 \exp(-K \cdot Z)$   $0 < Z < t_1$ .

Thickness of slabs 1 and 2 are  $t_1, t_2$ .

Narrow beam absorption coefficient 1 and 2 are  $\mu_1, \mu_2$ .

Build-up coefficient  $\alpha$  is appropriate to the equivalent material, e.g. for  $\mu_2 t_2 \gg \mu_1 t_1$ , this would be material 2.

Corresponding values can be obtained for other source distributions, e.g. uniform, when  $K = 0$ .

For the heating close to a source, it was not permissible to use a linear build-up factor as above in which case Taylor

coefficients were used.

$$B = A \exp(-a_1 \mu r) + (1 - A) \exp(-a_2 \mu r).$$

For the expressions used see [4] and for numerical data used see [2], Appendix II.

In the shield calculations, Aluminium build-up factors were used for surface dose rates, Iron build-up factors for heating.

The same method was used for calculating the surface dose rates, in the reactor pit and the transfer chamber after shutdown, due to fission products in the core and activation of shield components.

### 2.3.2 Containment Shielding

The method of calculation of the dose rates outside the containment building following the release of fission products in an emergency, is described in [3] and is given in some detail in Appendix II.

Briefly, radiation reaches the dose point either directly or via scattering in the air (skyshine). The values of these components for a uniform fission product distribution, at different times after shutdown have been derived and the results are plotted on Figures 7 and 8.

### 2.3.3 Ancillary Shielding

It is not intended to go into detail to describe either the method used or the calculations carried out to assess the shielding of the many components containing active sources, although in total this has constituted quite a large proportion of the work which had to be done.

These sources are almost exclusively beta and gamma as the fuel elements will not be discharged for at least 18 hours after reactor shutdown.

Suitable assumptions as to the geometry of the source are made and numerical methods as above were employed using tabulated functions as appropriate from [8].

## 2.4 Calculations for the Reactor Shield

Because of the cylindrical symmetry of the Dragon Reactor core, reflector and shields it was only necessary to calculate the flux distributions in the shielding in 3 directions.

- (i) On the reactor midplane - in a radial direction.
- (ii) On the vertical reactor axis below the core.
- (iii) On the vertical reactor axis above the core.

The layout of the shielding in these three general directions is given in Figs. 1, 2, 3 and in Appendix I.

In addition to the evaluation of these distributions, an examination was made of all possible streaming paths where these led to areas of access. This work will not be discussed here except in one or two important instances, notably in the top shield plug where streaming down the control rod holes could not be avoided. The manner in which this problem was tackled is outlined in 2.4.3.

#### 2.4.1 Radial Shielding

Pessimistically it was assumed that the "source" was an infinite cylinder, and the appropriate  $1/R$  corrections were made in the calculation.

The calculated neutron flux distribution through the shield is given in Fig. 6 and Fig. 9.

It should be noted that at the main airlock there is no concrete secondary shield. A cast-iron shield has therefore been let into the main concrete biological shield. In Table 2.1, the neutron and gamma dose rates are given at points near the outside of the shield. It will be seen that the contribution from neutrons is unimportant, and thus a 25 cm (10 in) cast-iron shield ( $\rho \approx 7.1$  g/cc) provides the necessary gamma shielding in place of 2 ft 10 in concrete.

Table 2.1

Location	Main Biological Shield		Secondary Shield
	25 cm inside	surface	Outer surface 61 cm thick
Neutron			
1. Removal flux	$n/cm^2$	sec	$0.5 \times 10^{-4}$
2. Epithermal flux	"	"	$1.5 \times 10^{-4}$
3. Thermal flux	"	"	$5 \times 10^{-4}$
Total neutron dose rate	m	rem/hr	$\sim 10^{-5}$
Total gamma	"	"	"
	not evaluated	33	0.32
TOTAL DOSE RATE	"	"	33
			0.32

There are three main penetrations through the radial shield (see Fig. 1).

- (i) For the precooler - streaming only into area of high activity where delay beds are located.

- (ii) For drives to the instruments in the thermal column below the reactor. Stepped plugs are provided.
- (iii) Viewing system - provided with a shield plug to protect the viewing equipment during operation and secondary concrete shielding outside the main shield.

The dose rate at the surface of the main shield is negligible after shutdown.

#### 2.4.2 Bottom Shielding

When calculations were carried out for the reactor shield below the core, the two subjects of primary interest were:-

- (i) The neutron flux levels
  - (a) in the thermal column, and
  - (b) near the surface of the main shield which determines the activation gamma level after shutdown in the pit.
- (ii) Shutdown gamma radiation levels in the reactor pit, and in the thermal column.

The neutron flux distribution which was calculated for the bottom shield, neglecting the perturbation effect of streaming along penetrations, is given in Fig. 10. For the shielding calculations, a cylindrical geometry was assumed, with a cylinder diameter equal to the core height. It might be argued that a sphere of the same diameter more closely approximates to the actual core, but it was felt that for the shielding this might lead to an optimistic answer.

The calculated surface dose rates during operation and after shutdown are given below in Table 2.2.

Table 2.2

Type of Radiation	During Operation	18 hours After Shutdown
1. Removal neutrons	} .03 m rem/hr	-
2. Epithermal "		-
3. Thermal "		.18 " "
Total neutron dose rate	.21 " "	-
Core gamma " "	} 1000 " "	0.8 mR/hr
Secondary gamma dose"		(activation) 0.2 mR/hr
TOTAL DOSE RATE	1 rem/hr	1.0 mR/hr

Access is not required to the pit during operation at power - sufficient shielding is also provided to reduce the gamma radiation

from other sources, e.g. fuel element roundabout and delay beds, to 1.5 mR/hr.

There are a number of experimental tubes (see Fig. 1) penetrating the bottom shield, these are stepped and in most cases plugged (steel-graphite plugs where inside primary circuit). Where this is not possible, e.g. for the neutron source drive external shielding is provided.

### 2.4.3 Top Shield

The top shield consists of three parts

- (i) Inner shield - top shield plug within the pressure vessel, and a thick block shield outside.
- (ii) Main shield - concrete, which additionally provides shielding for the transfer of fuel elements during discharge, and for fission product deposition in the primary heat exchangers.
- (iii) Containment shield, which is the roof of the building.

Since there is normally no access to any area during operation, the dose rates are not of prime importance, they have been calculated for the outside of the main shield and above the roof (see Table 2.3).

After shutdown unrestricted access (except during fuel element discharge) is required to the floor which the main shield provides, and limited access on occasions to the region above the inner shield, i.e. the transfer chamber.

The outer concrete block shield provides (for other reasons) substantially more shielding than is necessary. As has been indicated above it was essential to optimise the steel-graphite plug to reduce the total gamma dose rate after shutdown in the transfer chamber. It is considered that access will be delayed for at least 7 days after shutdown and the optimisation was made on this basis.

The calculations were performed with the same assumptions as for the bottom shield, but because of the geometry the over-estimate is likely to be greater. This is comforting, however, because the total neutron leakage through the control rod tubes seems likely to increase the overall activation level by at least a factor of 3<sup>x</sup>. Because of the design of the control rod winding head, it was impossible to eliminate the direct streaming path. Its effect has been minimised by superimposing a boron containing, Hiduminium alloy sheet over the hole, to absorb thermal neutrons (and avoid production of activation gammas).

Fig. 11 gives a plot of neutron distribution and Table 2.3 a summary of dose rates.

The other streaming paths are all stepped to minimise radiation leakage.

Table 2.3

Location	Top of Shield Plug (see " - page 15)	Outside Main Shield	Outside Building
1. Operation			
(a) Fast Neutron	200 Rem/hr	~10 mRem/hr	0.1 mRem/hr
(b) Thermal "	1 Rem/hr	~10 " "	0.035 " "
(c) Gamma	100 Rem/hr	~60 " "	3.0 " "
TOTAL	300 Rem/hr	80 " "	3.2 " "
2. Shutdown (after 1 day)			
(a) Core Gammas	~10 mR/hr	negligible	negligible
(b) Activation Gammas	~50 mR/hr	3 mR/hr	"
TOTAL	60 mR/hr	3 mR/hr	"

N.B. These figures are preliminary only, and subject to final revision.

## 2.5 Miscellaneous Shielding Problems

### 2.5.1 Charge, Discharge and Storage of Fuel Elements

The irradiated fuel elements are raised by the charge machine into the transfer chamber, after the reactor is shutdown and cooled. They are removed from the pressure vessel and transferred to the canning cell by the transit flask. Up to this stage the element is bare and any fission product activity released during depressurisation or transit is as far as possible removed by a purge system and trapped in a filter. Access is not normally permitted within the containment during discharge, but in emergency local operation can be carried out, if remote control fails.

Shielding is provided at this stage by the main 3' 0" thick concrete shield wall (and in transit by the 6" thick lead flask). Maximum dose rate at the wall surface is 450 mR/hr and in the outer annulus 0.75 mR/hr. The canned element is lowered down the lead shielded discharge chute to the fuel storage roundabout, which has a capacity large enough to contain a full core load of elements. Fuel elements are normally stored for 1 month to cool. The main shield provides a minimum of 175 cm (5 ft 9 in) normal concrete or equivalent, and the maximum dose rate at the surface (when 37 elements have been discharged) is 0.3 mR/hr.

From the roundabout storage, the elements are removed in the transit flask (Fig. 13) to the storage building, or to the active handling facility where they are processed.

Fig. 12 shows the general arrangement for the discharge operation.

## 2.5.2 Fission Product Plant

Subsequent to the initial design conception, which led to the current shield design, a number of improvements have been incorporated in the system, which should lead to a considerable reduction in the fission product source strength.

The plant and shield design were based on the use of a fission product releasing fuel, which consisted of uncoated fuel particles in a graphite matrix, contained in an impermeable graphite fuel box. Fission products diffusing through the walls of the box were swept by a purge flow through a flow restrictor and a dust cyclone to the pre cooler unit of the fission product plant (see Fig. 1). An equivalent delay of 1 hour was assumed between the time of production to escape into the purge for all fission products. Source strengths were calculated on this basis, for the present design of shield.

Investigations were carried out to study the effect of the permeability of the fuel box on the release of fission products from the fuel, and these indicated that the initial assumption of a 1 hour delay for all fission products was not accurate over the range of half lives, but that it could easily be achieved by the use of the impermeable graphites that had been developed. The incorporation of a charcoal trap at the base of the fuel element further modifies the escape of fission products to the pre cooler, etc. These investigations are reported in [9].

At the present time, it is envisaged that the fission product load will be further reduced considerably, by the use of coated fuel particles. Because of the much smaller escape of fission products, the fraction of short lived products reaching the plant will be considerably reduced. The longer lived products for which the liquid nitrogen cooled charcoal beds, "Cold Plant," are designed, however, may not be substantially affected.

Fig. 14 shows a layout of the fission product plant, which consists principally of

- (i) Pre cooler (partly inside reactor shield)
- (ii) Delay beds - water cooled charcoal beds
- (iii) Hot plant (not very active)
- (iv) Cold plant - liquid nitrogen cooled charcoal beds
- (v) Compressor room.

The "hot" area also contains dump tanks and dirty gas receivers, which may also contain active gas.

It is not anticipated that entry will ever be necessary after active operation has begun to the area containing 1 and 2, but it may be needed to 3, 4, and 5 which are therefore partially shielded from 1 and 2. Provision is also made for the removal and replacement of

the valves located in and below the roof of the shielded area. This will be done remotely using automatic cutting and welding tools behind shielding. These tools have been or are being developed by the Project.

Shielding is principally in normal concrete 5' 9" thick (total) for the ring wall, but the roof slabs are of the equivalent thickness in barytes concrete (above 5 and 6) or steel shot concrete (above 1, 2, 3 and 4). This latter was necessitated by an increased plant space requirement, and added considerably to the shield cost. The shield plugs for the valves let into the roof slabs are generally of steel shot concrete.

In the original design, surface dose rates in the outer plant room were kept below 0.75 mR/hr whereas up to 100 mR/hr (based on equilibrium source strengths) was allowed in the compressor room and the hot plant area. (In the latter case only after regeneration of the cold plant beds.)

### 2.5.3 Activity Deposition in the Coolant Circuit

The level of induced activity in the helium of the primary circuit is comparatively low (there will be some tritium from  $\text{He}^3$ , which will be removed in the cold plant freezer unit in oxide form).

There are however several kilocuries of fission products which have diffused through the outer graphite sleeves of the fuel elements. An early estimate was made by H. Schmidt [10, 11] of their distribution and deposition, but research since carried out at Harwell by Bromley and Large [12] shows that the flow mechanism through the graphite is more complex and that there may be a higher release of Caesium, Strontium and Barium.

Deposition in the ducts is shielded by the enclosing thermal shield (steel-water for cooling) and concrete blocks (see Fig. 1), and in the heat exchangers by 15 cm (6 in) steel or cast iron in the transfer chamber, and the 90 cm (3 ft 0 in) concrete walls. Detailed calculations are given in [2].

## 2.6 Irradiation Damage

In general it has been possible by the choice of a suitable radiation resistant material to avoid the need for protection of components from radiation damage. A short resumé follows of the consideration that was given to damage of components outside the reactor core. An extensive programme of irradiation testing has been and is being carried out on core materials and components [13] and [14].

### 2.6.1 Damage of the Pressure Vessel and Support Structure

The maximum integrated dose of fast neutrons (Energy > 1 MeV) to which the pressure vessel will be subjected has been estimated to be  $8 \times 10^{18}$  h/cm<sup>2</sup> over 20 years. During operation the temperature of the pressure vessel varies between 300°C and 350°C according to location. At these temperatures the damage tends to be self

annealing and the effect on the transition temperature (ductile - notch brittle) is less marked than at lower temperatures. Accelerated damage tests to an integrated dose of  $3 \times 10^{19}$  n/cm<sup>2</sup> show a maximum temperature rise of 80°C, for an irradiation temperature of 290°C [15].

In addition to these tests which have been carried out on specimens of steel and weld material used in the vessel, further tests will be carried out on specimens irradiated in the reactor under the same conditions as the vessel. Provision is also made for annealing out the damage at 375-400°C should the transition temperature rise above a temperature which it is impracticable to maintain after shutdown. Possibility of a brittle fracture is therefore not significant.

The core support grid will be subjected to a fast neutron dose of approximately  $2.5 \times 10^{19}$  n/cm<sup>2</sup> maximum but its operating temperature is 450°C, and at this temperature damage is insignificant.

Fig. 15 shows the fast neutron flux variation on the pressure vessel.

#### 2.6.2 Gasket Materials, Seals and Lubricants

Damage to these materials is considered together as the radiation resistance of plastics, elastomers and lubricants covers much the same range.

Except in one or two isolated instances, the materials chosen are capable of withstanding, without the need for shielding, the maximum radiation dose over the reactor lifetime. This has been achieved as follows:-

- (i) Engineering design - avoidance of the need for gaskets and lubricants in areas of high radiation intensity, e.g. within the main reactor shield all P.V. joints are welded, and mechanical seals are used in the plenum, etc. (see Fig. 1)
- (ii) Choice of high resistance materials, e.g. Molybdenum disulphide as a lubricant, where it has not been possible to avoid above features in the design.

A complete list of the radiation resistant materials used, the estimated dose rates and temperature conditions to which they will be subject will be given in the Dragon Reactor Experiment final shielding report.

#### 2.6.3 Activation and Radiolysis of Thermal Shield Water

##### (a) Activation of Demineralised Water - no additives

Because of the very slow flow rate of water through the thermal shield, the products of the following reactions decay too

rapidly to be significant.

- (i) Thermal neutron reaction  $O^{18} (n; \gamma) O^{19}$   $T_{1/2} = 29 \text{ sec}$
- (ii) Fast neutron reactions  $O^{16} (n; p) N^{16}$   $T_{1/2} = 7.4 \text{ sec}$   
 $O^{17} (n; p) N^{17}$   $T_{1/2} = 4.1 \text{ sec.}$

The additional reactions  $H^2 (n; \gamma) H^3$  and  $O^{17} (n, \gamma) O^{18}$  are not important from the shielding point of view as the decay radiation is only  $\beta$  which is shielded against very easily. With the very slow rates of flow in the thermal shield it is improbable that any "crud" will be removed.

(b) Activation of 1000 ppm Sodium Chromate Corrosion Inhibitor in the Water

Both the  $Na^{23} (n; \gamma) Na^{24}$  and the  $Cr^{54} (n; \gamma) Cr^{55}$  reactions result in  $\gamma$  emitting products, and the shielding for the parts of the thermal shield water circuit outside the main shield is designed accordingly [2]. A maximum of (45 cm) 18 in of normal Portland concrete is required for the water storage tank and the heat exchanger.

(c) Radiolysis of Water

The pessimistic assumption was made that the effect of  $Na^+$  and  $Cr O_4^{2-}$  ions have a serious effect on recombination of the active H and  $OH^+$  radicals (and others formed by further reactions) as ferrous ions.

The maximum production rate of gaseous hydrogen was estimated to be 140 litres/hour and provision of a recombination unit was made on this basis.

2.6.4 Liquid Nitrogen - Explosion Hazard

The maximum gamma dose rate in the liquid nitrogen coolant in the charcoal filled cold traps for clean-up of the He purge circuit, was estimated as  $< 10^6$  Rads/hour. The mean level was considerably lower.

Experimental work carried out at A.E.R.E. Harwell and the Wantage Research Laboratory indicates that at this level there is no hazard for commercially pure liquid nitrogen.

2.7 Heating Due to Nuclear Radiation

The method of assessment of nuclear heating in the shield regions has been discussed in Section 2.3.1.

The baffles and the graphite support plate were increased in thickness to 5 cm (2 in) to reduce the fast neutron flux at the vessel, and also to reduce the heating.

The thermal shield (steel-water) is present primarily to absorb thermal radiation from the pressure vessel, but it also reduces the nuclear heating in the concrete biological shield to an insignificant amount ( $< 1 \text{ watt/ft}^2$ ).

No reinforcement or cooling coils are therefore required in the concrete to control thermal stresses. There is of course nominal reinforcement for structural reasons.

Values of estimated nuclear heating at various points on the pressure vessel and other parts of the shield are given in Fig. 15.

A 2 dimensional gamma absorption programme written by the author for estimating the gamma heating distribution throughout the reflector and neighbouring regions, gives values that indicate the axial variation of heating is about a factor of 2.

### 3. ENGINEERING ASPECTS OF THE SHIELD DESIGN

#### 3.1 Structural Materials - Factors Governing Their Choice

In deciding upon the most suitable material for use as a shield, a number of factors must be considered and the choice will not necessarily be determined by the properties of the material as an attenuator of radiation. The situation is further complicated by the obvious fact that the shield often has to fulfil other and sometimes more important functions than shielding. It will often be preferable to utilise an apparently less suitable shielding material on the grounds that it also provides say, structural support or a thermal radiation baffle. An example of a dual or rather triple function is in the steel-water thermal shield of the Dragon Reactor. In addition to acting as a radiation shield, it acts as a heat sink to absorb heat radiated from the pressure vessel, and as a support for the vessel and the reactor core.

Some of the factors influencing the choice, several of which are interrelated, are discussed below:-

##### (i) Absorption (By Removal or Slowing Down) of Fast and Thermal Neutrons

The most efficient slowing down material is hydrogen, and therefore it is desirable to have as high a hydrogen density as possible in a reactor shield. [Reference [16] gives a list of such materials.] Concrete, which has a 0.5-1.0% water content in normal conditions is not so efficient, i.e. a greater thickness is required. Graphite is less efficient, but can be used under the arduous conditions inside the reactor pressure vessel for fast neutron slowing down. Steel is a very versatile material from the shielding engineers point of view, as it slows down penetrating fission neutrons by inelastic scattering into the energy region where hydrogenous materials are efficient. It also absorbs thermal neutrons strongly, though unfortunately with the emission of penetrating gamma radiation, and it has a long lived radioactive product ( $\text{Fe}^{58} (n; \gamma) \text{Fe}^{59} T_{1/2} = 46 \text{ days}$ ). In addition the hole in the scattering cross section in the 20-30 KeV region makes it unsuitable on its own, and it is therefore used in conjunction with another material such as water or graphite. In the Dragon Reactor, except in one instance (see Section 3.2) boron containing materials have not been used for shielding. The use

of a boron cast-iron is being considered for the 1250 MW H.T.R. for thermal neutron absorption to reduce the activation level inside a concrete vessel (see Section 4). This will permit longer access times. Boron inclusion also reduces the high energy secondary gammas during operation, as only 0.5 MeV gammas result from thermal neutron absorption in Boron.

(ii) Absorption of Gamma Radiation

The attenuation of gamma radiation is determined almost entirely by the total weight of shield/unit cross section. The thickness required is therefore inversely proportional to the material density. This is a slight over simplification as high energy gamma radiation is weight for weight, attenuated slightly more rapidly by materials composed of elements of high atomic weight.

Minimum thickness gamma shields use high density materials such as steel shot concrete, steel or lead. It is unfortunate that in general the specific cost (cost/unit weight) increases with the density - this will be discussed later.

In the Dragon Reactor, high density materials have only been used where essential, to minimise the shield cost, but they have been necessary in certain areas to leave the maximum space for plant.

(iii) Structural Integrity

It is of course essential that under operating conditions the shield performance will not be impaired through loss of integrity. The principle requirements are:-

- (a) Ability to withstand the high (or low) temperature to which it may be subjected (normal concrete below about 80°C for a long period).
- (b) Ability to withstand the appropriate level of neutron and gamma radiation - this may be very important close to the core, and such materials as polythene cannot be used where the integrated dose will exceed  $10^{17}$  n.v.t.
- (c) Suitable mechanical properties - e.g. thermal conductivity, compressive and/or tensile strength. In some cases, e.g. water and lead, provision may be made to support the shield, at some cost penalty. Admiral Rickover in his foreward to [8] comments that "Water has no cracks", but fails to add that it must be contained.

(iv) Vapour Pressure at High Temperature and Resistance to Corrosion (applicable to Materials used Inside the Dragon Pressure Vessel)

The very stringent requirements relating to impurities in the coolant, precluded the use of hydrogenous and many other materials. Steel and graphite were therefore used almost

exclusively.

(v) Shield Weight and Size

Except in the few instances mentioned previously it has not been necessary to use high density, high cost materials for Dragon shielding. Lead shielding has been used for the fuel element discharge chute because of space limitations and in the transit flask to keep the weight below the 25 ton limit set by the crane capacity.

(vi) Construction and Fabrication

In deciding on the choice of materials for the shield, the programme of construction and erection had to be considered, and also the ease with which the shield could be placed at the planned time of erection.

Instances of this are in the filling of the bottom shield prior to installation, the use of brick size concrete blocks in the top shield, and precast units for the infilling of temporary construction access holes. Removable precast beam units are used in the floor over the fission product clean-up plant, and the compressor room to permit future access.

(vii) Cost

Whilst this is not the most important factor in a research reactor, every effort has been made to choose materials, and a form of construction that would result in a minimum cost. Overriding factors have prevented this in some instances. It must be remembered, that a direct comparison based on the actual cost of construction, is not the sole criterion of economy. For instance, the saving in space, and the shortening of mechanisms penetrating the shield may result in an overall economy. An instance where this is very significant (see Section (iv)) is in the saving of space within the pressure vessel of a power reactor system; where the cost per cubic ft of contained volume is likely to £5-£10.

### 3.2 Shield Penetrations - Clearances

As far as possible the specification of unduly restrictive tolerances has been avoided, to minimise the cost of construction. Where a very small clearance would otherwise have been necessary the penetration has been stepped.

The following was used as a guide:-

Construction	Tolerance
Concrete	$\pm \frac{1}{4}$ in
Cast-iron (unmachined)	$\pm \frac{1}{8}$ in
Steel plate "	$\pm 1/16$ in
Machined steel	$\pm 0.01$ in.

In a few instances, where it was not possible to step the penetration, e.g. the source and instrument drive mechanisms penetrating the bottom shield, a closer tolerance had to be specified.

To allow erection of the vessel, etc., large clearances were made, e.g. in the bottom shield penetrations, and the space between the liner and the shield was grouted in after erection.

### 3.3 Operational Considerations in the Assessment of the Permissible Dose Rate

In order to minimise the shielding cost, certain areas within the reactor building which do not normally need to be accessible, are only partially shielded.

Operational considerations call for the following, broadly speaking:-

Table 3.1

Access requirement	Maximum Dose Rate	Exposure Time	Total Dose
Unlimited access	2.5 m rem/hr	40 hours/week	5 rem/year
Unlimited after shutdown	"	"	"
Active equipment	100-200 " surface 2.5 " at 1 metre	Not specified Continuous	"
Normal maintenance access (operating staff)	25-50 "	4-2 hours/week	100 m rem
Planned maintenance access	up to 10 rem/hr	Variable	1 rem per occasion

### 4. A PRELIMINARY CONSIDERATION OF THE SHIELD DESIGN FOR A 1250 MW (Th) H.T.R.

At the present time, proposals for the power reactor design study are only tentative, and it is possible that the preliminary consideration outlined in this section will bear little or no resemblance to the final design.

The most important decision affecting the shielding is the choice between a steel and a prestressed concrete pressure vessel. In a design study made by the U.K.A.E.A., H.T.R. Design Office, Reactor group, Risley, which is reported [17], a 4½ in (108 mm) mild steel pressure vessel is proposed with a main biological shield 8 ft 0 in thick (244 cm) and 2 secondary shields 2 ft 0 in thick (61 cm) outside the containment shell as in the Dragon Reactor Experiment.

In the current investigation by the Dragon Technical Assessment Section, and under contract, a prestressed concrete pressure vessel is proposed. In this design the heat exchangers and circulators would be situated within the pressure vessel, and hence the shielding problems to be solved are somewhat

different to those in the Dragon Reactor Experiment shield.

The thickness of the prestressed concrete shield is likely to be a minimum of 12 ft (365 cm) for structural reasons and hence the external dose rate would be well below the permissible level. The present containment philosophy, requiring a secondary containment shell to retain activity released from the pressure circuit, would require a 2 ft thick concrete shield externally (possibly thicker). Most of the reactor shielding problems arise from the inclusion of the heat exchangers and circulators inside the pressure vessel, and some internal shielding will be necessary to reduce

- (a) Activation and radiolysis of the secondary coolant
- (b) " of the circulators
- (c) Heating of the concrete pressure vessel.

Fuel element charging and discharging will not be carried out whilst the reactor is operating at power, so the problems will be similar to those in the Dragon Reactor except that the heat dissipation from the fuel element carousel could reach a maximum of over 1 MW (2 megawatts in the event of it being necessary to discharge a full core).

The fission product clean-up plant is likely to be simpler than for the Reactor Experiment, although this has not yet been closely examined. It is almost certain that as in the experiment, the plant will be located outside the main concrete shield, in its own shielded cell, with access provision for maintenance.

#### 4.1 Results of Preliminary Shielding Calculations

Representative core source distributions and boundary fluxes were obtained from a typical Fever print out, from a burn-up calculation performed by the Assessments group, (see [18]). This was done for a 1000 MW reactor, so a correction factor of 1.5 was applied, (being 1.25 for power ratio, 1.2 axial flux max/mean ratio).

The core boundary fluxes were thus

$$\phi_{th} = 9.4 \times 10^{13} \text{ n/cm}^2 \text{ sec}, \quad \phi_f = 11.5 \times 10^{13} \text{ n/cm}^2 \text{ sec.}$$

Shielding assumed was as follows

Reflector graphite	90 cm
Shielding steel	15 cm
P.V. membrane steel	2 cm
P.V. concrete	365 cm.

The effect of plant within the vessel, and voids in the concrete for prestressing steel was ignored. In the initial calculation the steel was assumed distributed throughout the concrete.

Fig. 16 shows the calculated neutron flux variation in the shield, obtained by the method of Section 2.1.

The following Table 4.1 gives values of the neutron flux at the points of most interest, i.e. those determining activation and radiolysis and also the gamma heating in the steel and concrete.

Table 4.1

Location	R cm	Neutron Flux		Radiation Heating		
		$\phi_f$ (> 1 MeV) n/cm <sup>2</sup> sec	$\phi_{th}$ (equ) n/cm <sup>2</sup> sec	Neutron MeV/cm <sup>2</sup> sec	Gamma MeV/cm <sup>2</sup> sec	Total watts/ft <sup>2</sup> adsorbed
1. Outer edge reflector	376	$\sim 7 \times 10^8$	$3.8 \times 10^{12}$	Current out $2.2 \times 10^{13}$	-	-
2. Outer edge steel shield	411	$\sim 1.2 \times 10^8$	$2.9 \times 10^8$	out $1.3 \times 10^{10}$	out $7 \times 10^{10}$	$3.28 \times 10^3$
3. P.V. membrane	761	$\sim 10^8$	$2.9 \times 10^8$	out $1.0 \times 10^{10}$	out $4 \times 10^{10}$	4.92
4. Inner face concrete vessel	763	$9 \times 10^7$	$3.1 \times 10^8$	$1.0 \times 10^{10}$	$4 \times 10^{10}$	7.45
5. Outer face concrete	1128	$5 \times 10^{-7}$	$3 \times 10^{-7}$	-	Dose rate $10^{-6}$ mR/h	-

#### 4.2 Permissible Radiation Levels Inside the Main Shield

It appears that the following considerations impose limitations on the neutron flux or gamma heating levels in the shield.

- (i) It is desirable to avoid the need for shielding the steam turbine or condensers, and hence activation of the steam in the heat exchangers is limited. A fast neutron flux of  $< 10^8$  n/cm<sup>2</sup> sec and a thermal neutron flux of  $< 5 \times 10^{10}$  n/cm<sup>2</sup> sec is suggested as a guide.
- (ii) Radiolysis of the steam should not result in a sufficient generation rate of hydrogen to produce a flammable mixture in the air ejected from the steam condenser - i.e. the hydrogen generation should not exceed 5% of the minimum air ejection rate. The radiolysis rate of the steam is determined by the energy absorption, but back reaction rates leading to recombination, are critically dependent on impurity concentrations in the water (or steam).

(iii) Cooling of the Shield Concrete

Restrictions are placed though not necessarily on the nuclear heating directly by the temperature limitations in the concrete. These are determined by

- (a) Thermal stresses in the concrete and the steel membrane (due to temperature variation)
- (b) Creep of the concrete at elevated temperatures which may lead to excessive loss of prestress in the steel.

The temperature in the concrete may be controlled by

- (a) Limiting the incident radiation energy
- (b) Providing cooling coils inside the shield.

This limitation has not yet been studied in detail but it would appear that to avoid need for coils inside the concrete, the maximum incident energy current should not exceed about  $30 \text{ MW/cm}^2$ . Since cooling coils have to be provided at the surface, to cool the membrane, it may be cheaper to have internal cooling rather than add to the steel or cast-iron thermal shield.

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GENERAL DATA - DRAGON REACTOR, AND SHIELDINGA. Principal Reactor Data

Type	Thermal homogeneous
Fuel	U-235 (fissile) Th-232 (fertile) in the carbide form
Moderator	Graphite
Power rating	20 MW Thermal
Uranium loading	20 kg U-235 (93% U-235)
Power density	14 MW/m <sup>3</sup>
Primary coolant	Helium
	Pressure - 20 atm
	Inlet temp. - 350°C
	Outlet " - 750°C
	Mass flow - 21 lb/sec
Circulators	6, Gas bearing, single stage centrifugal
Heat removal	Forced circulation boiling water Secondary circuit
	" " pressurised water Tertiary "
	" draught air cooler
Pressure vessel	Material MARWE 426 MA Steel
	Plate thickness 2 in
	Overall height 58 ft 1 in
	Maximum diameter 11 ft 6 in
	Weight 48 tons
	Operating temp. 320°C (mean)
Containment	Steel shell diameter 66 ft
	height 106 ft
	design pressure 10 lb/in <sup>2</sup> internal
	1/4 lb/in <sup>2</sup> external

Concrete building diameter 110 ft  
 height 125 ft (overall)  
 design pressure  $\frac{1}{2}$  lb/in<sup>2</sup> internal  
 $\frac{1}{4}$  lb/in<sup>2</sup> external.

B. Reactor Shielding Data

Core size	diameter	106 cm	(42 in)
	height	160 cm	(63 in)
Reflector thickness	radial	90 cm	(35 in)
	axial	42 cm	(16.5 in)
Steel Support skirt (radial neutron shield)		5 cm	(2 in) thick
Pressure vessel		5 cm	( " ) "
Bottom shield	graphite	15 cm	(6 in) "
	steel	5 cm	(2 in) "
Thermal shield (water filled tanks)			
	steel	18 cm	(7 in)
	water	20 cm	(8 in)
Concrete shield			
1. Radial	main	175 cm	(5 ft 9 in)
	secondary	61 cm	(2 ft 0 in)
2. Axial	(bottom)	91 cm	(3 ft 0 in)
Top shield			
1. Top shield plug (total thickness of layers)			
	steel	35 cm	(14 in)
	graphite	80 cm	(32 in)
2. Concrete shield		61 cm	(2 ft 0 in)
3. " roof		38 cm	(1 ft 3 in)

CONTAINMENT SHIELDINGBASIC ASSUMPTIONS

- (a) 100% of all volatile fission products (noble gases, iodine and bromine) and 25% of all other fission products are uniformly distributed over the containment building surface. For convenience of calculation they are divided into 3-energy groups 0.8, 1.7 and 2.5 MeV.
- (b) The radiation proceeds either directly to the dose point, (direct dose) or via scattering in the air (skyshine). It is further assumed that the skyshine source is only radiation attenuated in the thin roof slabs. (1 ft 3 in thick compared with the 4 ft thick concrete wall.) The angular distribution of  $\gamma$  radiation emerging from the roof is assumed to be cosine.
- (c) Except in determining the dose rate in the building, and at the outer wall surface, the effect of the shield structure within the containment is ignored.
- (d) In view of the approximate nature of the calculations, and the possibility of non-uniformity of fission product distribution, the basic assumptions have been augmented by a factor of 4 to provide a margin of safety

1. DIRECT RADIATION

- (a) Outer annulus of building, and outer surface of building.

Skyshine will be insignificant inside the building and can be ignored. The dose rate is calculated on the pessimistic assumption of a uniform plane source, with a radial correction to allow for the cylindrical geometry.

$$\text{Dose rate} = \Sigma D_E \cdot \frac{1}{2} S(E)_A \cdot B(\mu_E t) E_1(\mu_E t) \cdot 4 \cdot \frac{r_1}{r_2}$$

- where  $D_E$  = dose rate conversion factor (mR/hr - per MeV/cm<sup>2</sup> sec)
- $S(E)_A$  = source strength in MeV/cm<sup>2</sup> sec of energy E at time t = 0
- $B(\mu_E t)$  = build-up factor in concrete
- $\mu_E \text{ cm}^{-1}$  = narrow beam absorption coefficient
- t = shield thickness
- r = radius of dose point
- $r_1$  = radius of containment shell.

To obtain the average dose rate from time t = 0 to time t = t, the values of dose rate are multiplied by a factor  $\bar{F}(E, t)$  [3] or for the instantaneous dose rate by a factor F(E, t).

- (b) Outside the building r > 100 ft.

Skyshine can no longer be ignored and will be accounted for separately.

A pessimistic assumption for estimating the direct dose, is to assume a uniform line source with no self-shielding within the containment shell. The effect of the radiation source below ground level can also be ignored, because of earth shielding.

If  $h$  be the height of the shell above ground level,

$$\text{Dose rate (t = 0)} = 4 \Sigma D_E \cdot S(E)_L \cdot \frac{F(\theta, b)}{4\pi r} \cdot B(b)$$

where  $S(E)_L$  = Line source strength in MeV/cm sec of energy  $E$  at time  $t = 0$

$B(b)$  = Build up factor in concrete for each energy

$b$  =  $\Sigma \mu_E t$  for concrete and air

$$\theta = \tan^{-1} \frac{h}{r}$$

## 2. SKYSHINE

$\gamma$  radiation emerging from the reactor building roof is assumed to have a cosine distribution and the flux, assuming that the fission products are distributed on the top surface of the containment only is then pessimistically

$$\Sigma \frac{1}{2} \cdot \frac{S(E)}{\pi r_1^2} B(\mu t) E_1(\mu t)$$

This is an overestimate by at least 30%, and depending on the fission product distribution in the containment shell could be considerably more.

From [3], we then have, allowing a safety factor of 4,

$$\text{Dose rate (t = 0)} = 4 \cdot \Sigma D(E) \pi r_1^2 \cdot \frac{S(E)}{2\pi r_1^2} \cdot B(\mu t) E_1(\mu t) \frac{I_1'(E, \mu r, a')}{2\pi r}$$

where  $I_1'(E, \mu r, a')$  is the  $I$  function defined in [3] for an exclusion radius of  $a'$ . For these calculations  $a'$  has been assumed to be 55 ft (radius of outer wall).

$$I_1'(E, \mu r, a') = I_1'(E, \mu r \frac{a}{a'}, a) \exp \mu r \left( \frac{a}{a'} - 1 \right)$$

where  $a = 95$  ft, exclusion radius for which  $I$  functions are plotted in [3].

Dose rates, or average dose rates are obtained as in previous sections by use of multiplication factors  $F(E, t) \bar{F} \cdot (E, t)$ .

## 3. NUMERICAL VALUES USED IN CALCULATIONS

(a) Source strengths, and narrow beam absorption coefficients.

Table 1

Mean Energy (MeV)	$\mu$ (air) ( $\text{cm}^{-1}$ )	$\mu$ (conc.) ( $\text{cm}^{-1}$ )	$\mu$ steel ( $\text{cm}^{-1}$ )	S (MeV/sec)	D(E)
0.8	$0.86 \times 10^{-4}$	0.164	0.535	$3 \times 10^{17}$	$1.9 \times 10^{-3}$
1.7	$0.58 \times 10^{-4}$	0.113	0.362	$2 \times 10^{17}$	$1.7 \times 10^{-3}$
2.5	$0.47 \times 10^{-4}$	0.090	0.301	$1.5 \times 10^{17}$	$1.55 \times 10^{-3}$

Units of D(E) are mR/hr per  $\text{MeV}/\text{cm}^2$  sec.

(b) Build up factors used for aluminium (See [2]).

(c) Shielding consists of:-

Walls - 1 in steel + 4 ft concrete

Roof - 1 in steel + 1 ft 3 in concrete.

(d) Numerical values of functions  $I_1'$ ,  $F(E, t)$ ,  $\bar{F}(E, t)$  are taken from [3], and  $F(\theta, b)$  from [8].

(e) Dimensions of containment shell and building:

(1) Radius of shell = 33 ft (10 m)

(2) Radius of building = 55 ft (17 m)

(3) Height of shell = 81 ft (25 m)

total height (above and below ground level)

= 105 ft (32 m)

(4) Total surface area =  $2 \times 10^7 \text{ cm}^2$ .

(f) Outer annulus - Inner radius = 38 ft (11.5 m)

shielding = 2 ft concrete (61 cm).

#### 4. RESULTS OF CALCULATION

The instantaneous dose rates at time  $t = 0$  were calculated to be as follows:

(a) Outside Building

Origin of co-ordinates is centre of building.

Table 2

Radius (metre)	17	117	217	417	617	817
Total direct mR/hr	112	18	4	0.45	0.1	0.02
Total skyshine mR/hr	210	99	32	5.4	1.2	0.3
Total 0.8 MeV	72	23	6	0.65	0.1	0.01
Total 1.7 MeV	90	38	12	1.9	0.4	0.09
Total 2.5 MeV	160	56	18	3.3	0.8	0.22
Total dose rate (Direct & skyshine)	322 mR/hr	117 mR/hr	36 mR/hr	5.85 mR/hr	1.3 mR/hr	0.32 mR/hr

Instantaneous dose rates at times of 1 hour, 10 hours, 10 days and 100 days derived from these values are plotted on Fig. 7.

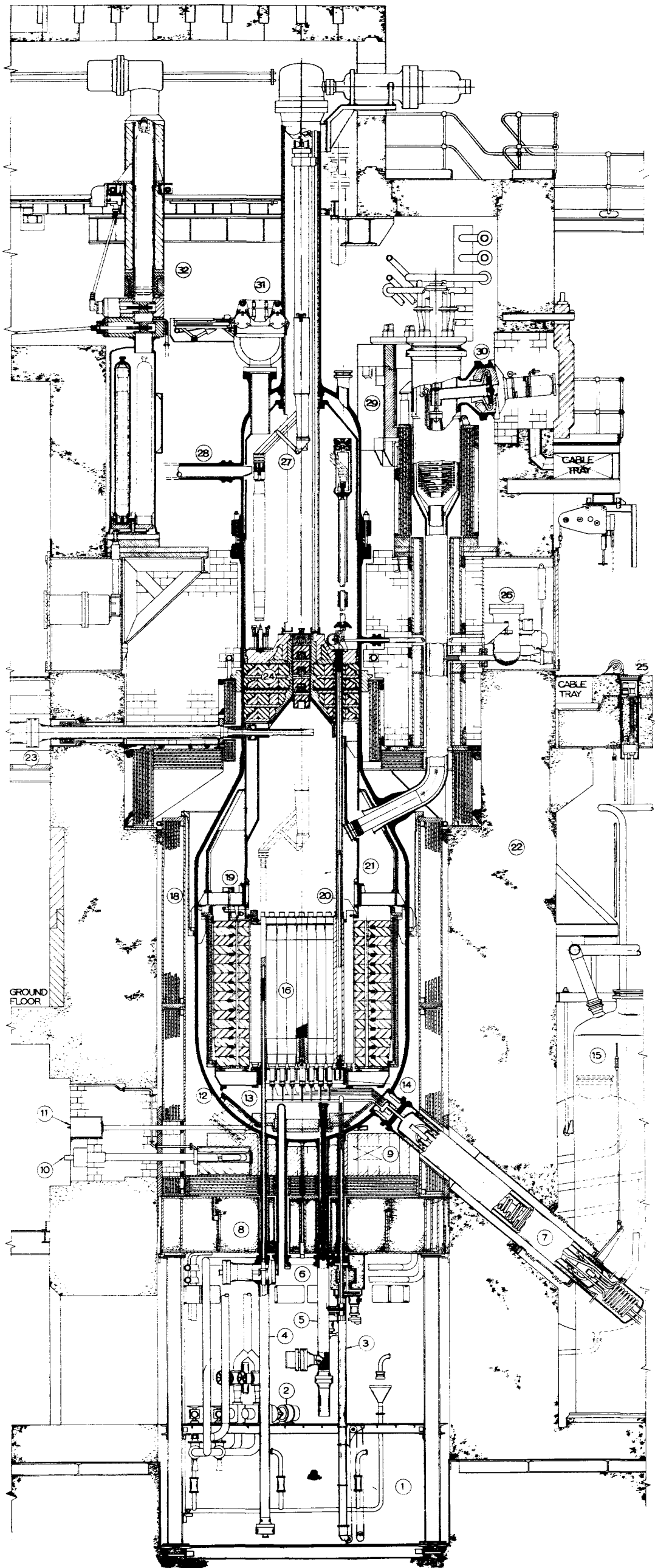
Integrated doses derived similarly are plotted on Fig. 8.

(b) Outer Annulus

The instantaneous dose rates on the inner wall surface after the accident are tabulated in Table 3.

Table 3

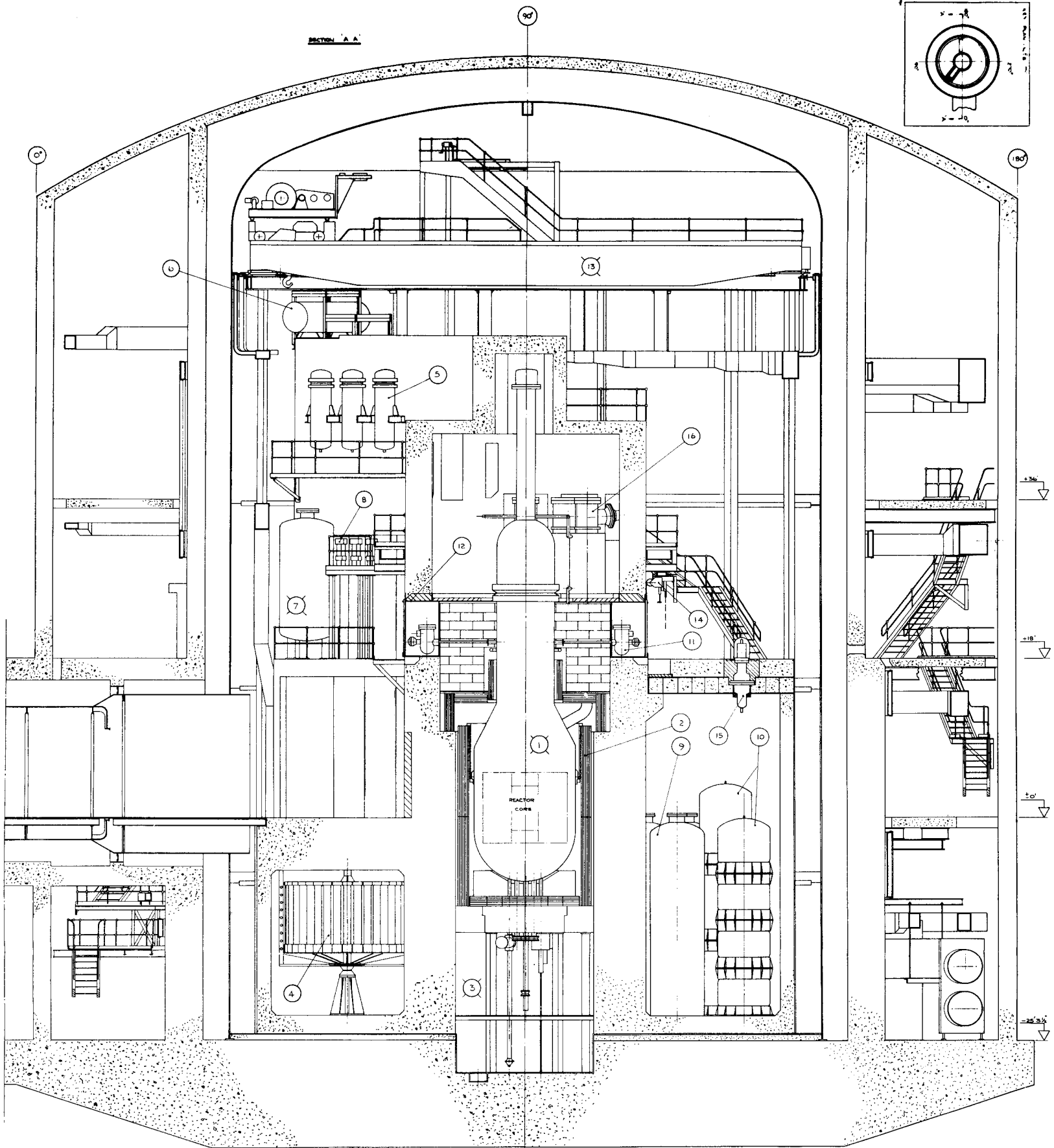
Time after accident	0	1 hr	10 hrs	10 days	100 days
Dose rate R/hr	49	24	6.7	2.5	0.12



REF. NO.	DESCRIPTION
1	ARRANGEMENT OF REACTOR VESSEL
2	SPECIMAN FACILITY COOLANT CIRCUIT
3	FSSION CHAMBER ASSY
4	NEUTRON SOURCE DRIVE
5	THERMOCOUPLE BRANCH
6	RE-ENTRANT TUBE
7	PURGE GAS PRE-COOLER
8	BOTTOM BIOLOGICAL SHIELD
9	THERMAL COLUMN
10	ION CHAMBER
11	SPECIMAN FACILITY
12	REACTOR PRESSURE VESSEL
13	SUPPORT STRUCTURE INSIDE PRESSURE VESSEL
14	CORE & STATIC GRAPHITE BEDPL
15	HEAT SINK & DUMP TANK INSTALLATION
16	FUEL ELEMENTS
17	REFLECTOR ASSEMBLIES

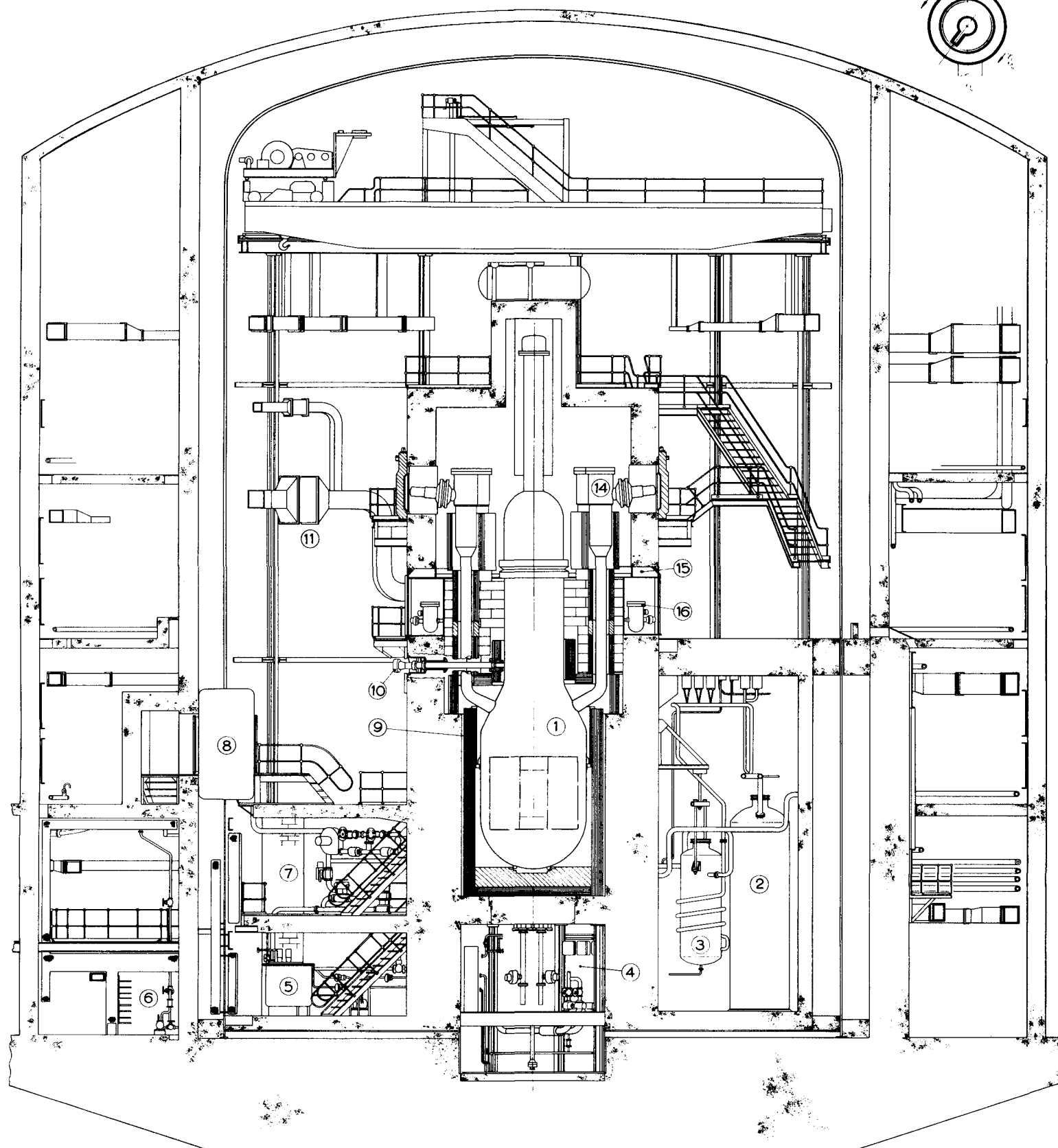
REF. NO.	DESCRIPTION
18	THERMAL SHIELD
19	CORE UNCLAMP MECHANISM
20	CORE SEAL
21	SUPPORT STRUCTURE INSIDE PRESSURE VESSEL
22	CONCRETE BIOLOGICAL SHIELD
23	RETRACTIBLE TV VIEWING FACILITY
24	MAIN SHIELD PLUG
25	VALVE FLOOR
26	CONTROL ROD DMECHANISM
27	CHARGE / DISCHARGE MACHINE INSTALLATION
28	NON-RETRACTING TV VIEWING FACILITY
29	HEAT EXCHANGER SHIELDING
30	PRIMARY HEAT EXCHANGER & CIRCULATOR
31	MAIN ENTRY VALVE
32	LOAD FACILITY
33	25 TON CRANE

FIG. 1 GENERAL ARRANGEMENT OF REACTOR



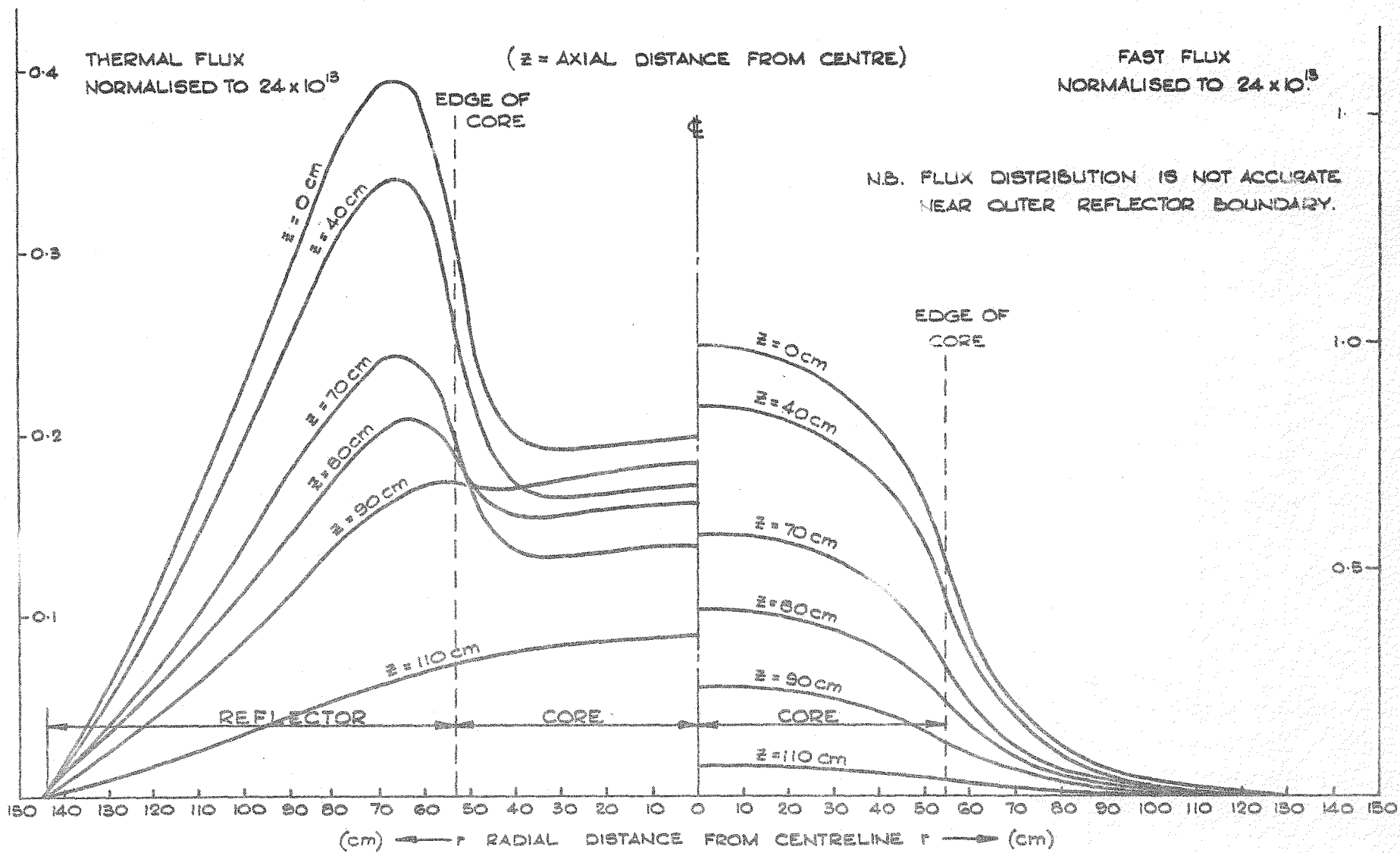
REF. No.	DESCRIPTION	REF. No.	DESCRIPTION
1	PRESSURE VESSEL.	9	DUMP TANKS.
2	THERMAL SHIELD.	10	DIRTY GAS RECEIVERS.
3	ARRANGEMENT OF EQUIPMENT ON PIT WALL.	11	P.V. ABSORBER ROD DRIVES.
4	G.A. OF ROUNDABOUT.	12	STEEL SUPPORT RING FOR UPPER BIOLOGICAL SHIELD.
5	G.A. OF SECONDARY HEAT EXCHANGERS.	13	25 TON CIRCULAR CRANE
6	STEAM DRUM.	14	MAIN HOIST.
7	QUENCH TANKS.	15	G.A. OF 37 WAY VALVE.
8	SECONDARY CIRCUIT PUMPS.	16	PRIMARY HEAT EXCHANGERS.

FIG. 2 REACTOR-SECTION AA  
(0°-180°)



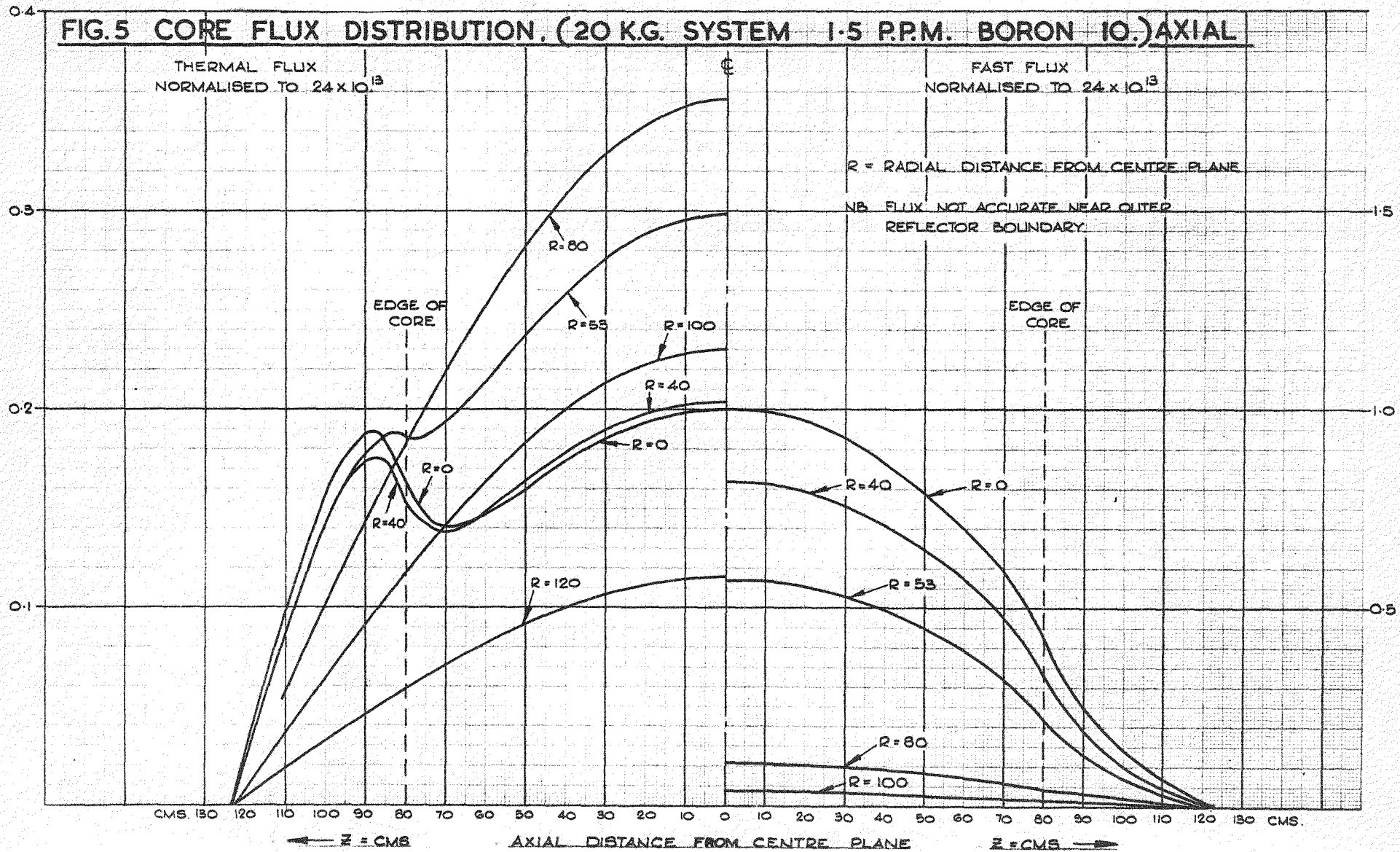
REF. No.	DESCRIPTION	REF. No.	DESCRIPTION
1	PRESSURE VESSEL	9	THERMAL SHIELD
2	DUMP TANKS (HELIUM)	10	VIEWING TUBE
3	HEAT SINK	11	NITROGEN RECIRCULATION SYSTEM
4	ARRANGEMENT OF EQUIPMENT ON PIT WALL	12	STEAM DRUM
5	SUMP TANK	13	25 TON CIRCULAR CRANE.
6	PUMP, VALVES ETC.	14	PRIMARY HEAT EXCHANGERS
7	COOLING FANS ETC.	15	STEEL SUPPORT RING FOR UPPER BIOLOGICAL SHIELD.
8	EMERGENCY AIR LOCK	16	P.V. ABSORBER ROD DRIVES

FIG. 3 REACTOR-SECTION BB  
(126°-306°)



**FIG. 4 CORE FLUX DISTRIBUTION, (20 KG. - SYSTEM  
 1.5 P.P.M. BORON 10.) RADIAL.**

FIG.5 CORE FLUX DISTRIBUTION. (20 K.G. SYSTEM 1.5 P.P.M. BORON 10.) AXIAL



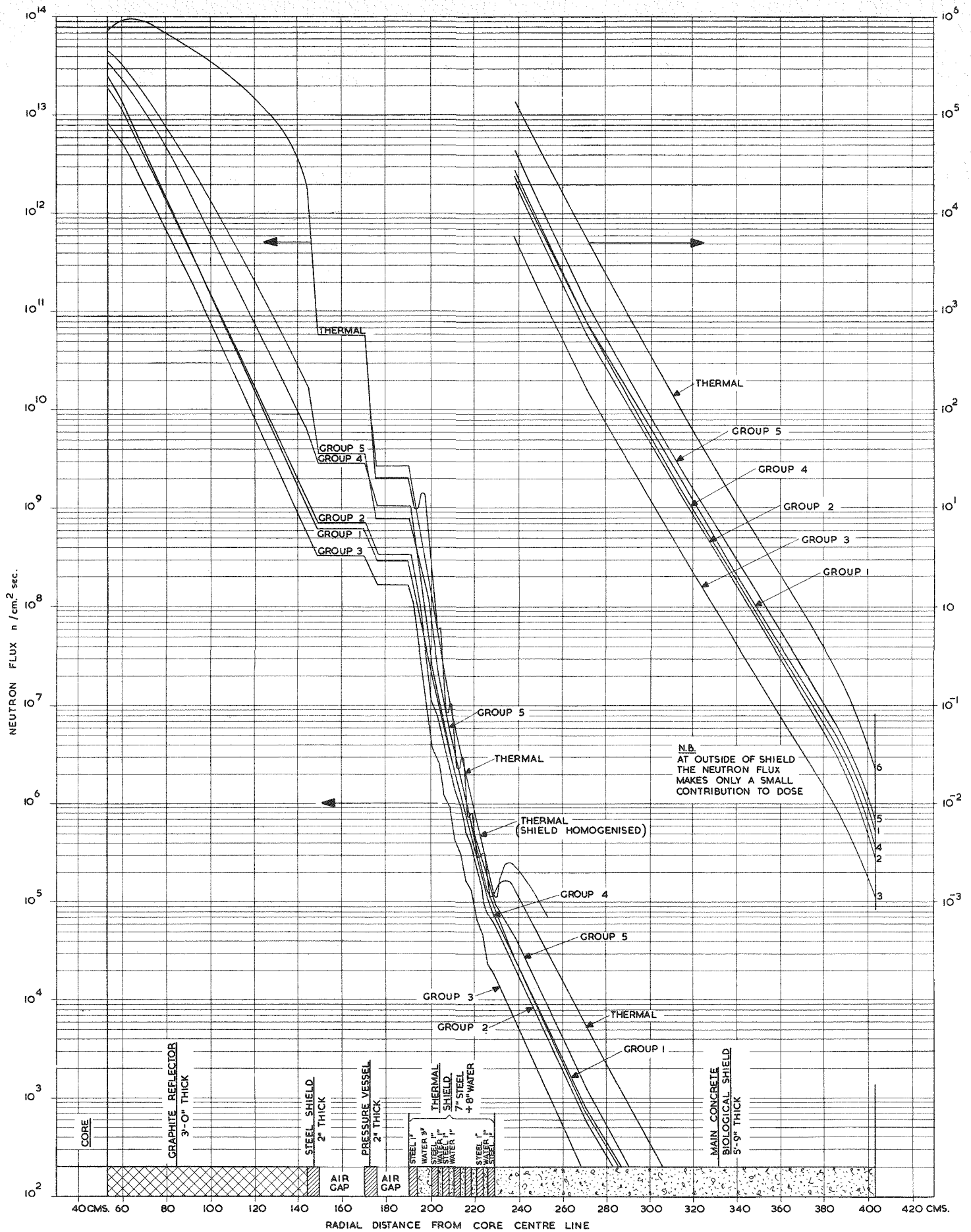


FIG. 6 NEUTRON FLUX INNER RADIAL SHIELD.

FIG. 7

DOSE RATE FROM BUILDING FOR DIFFERENT TIMES AFTER ACCIDENT

(DIRECT AND SKYSHINE)

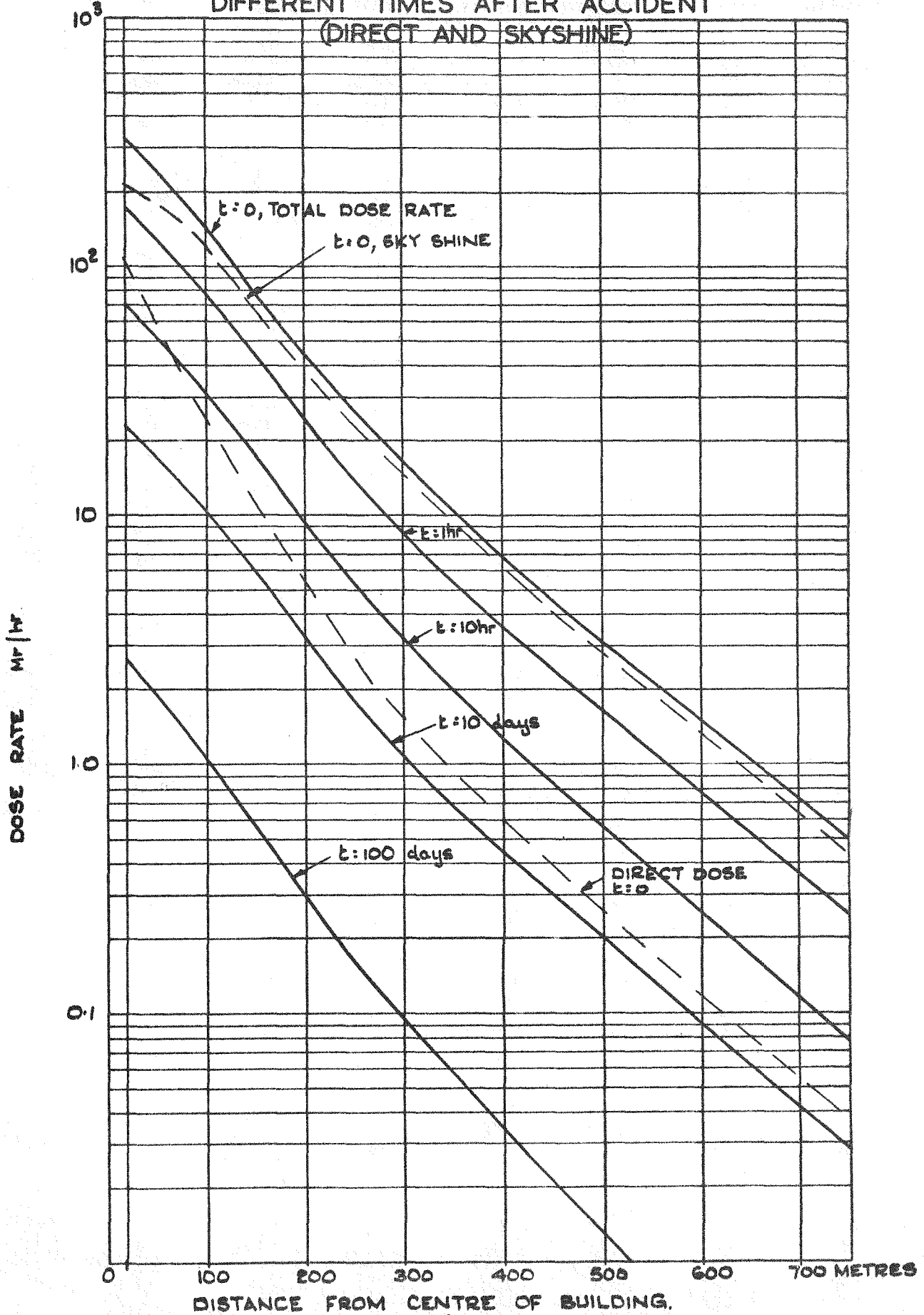
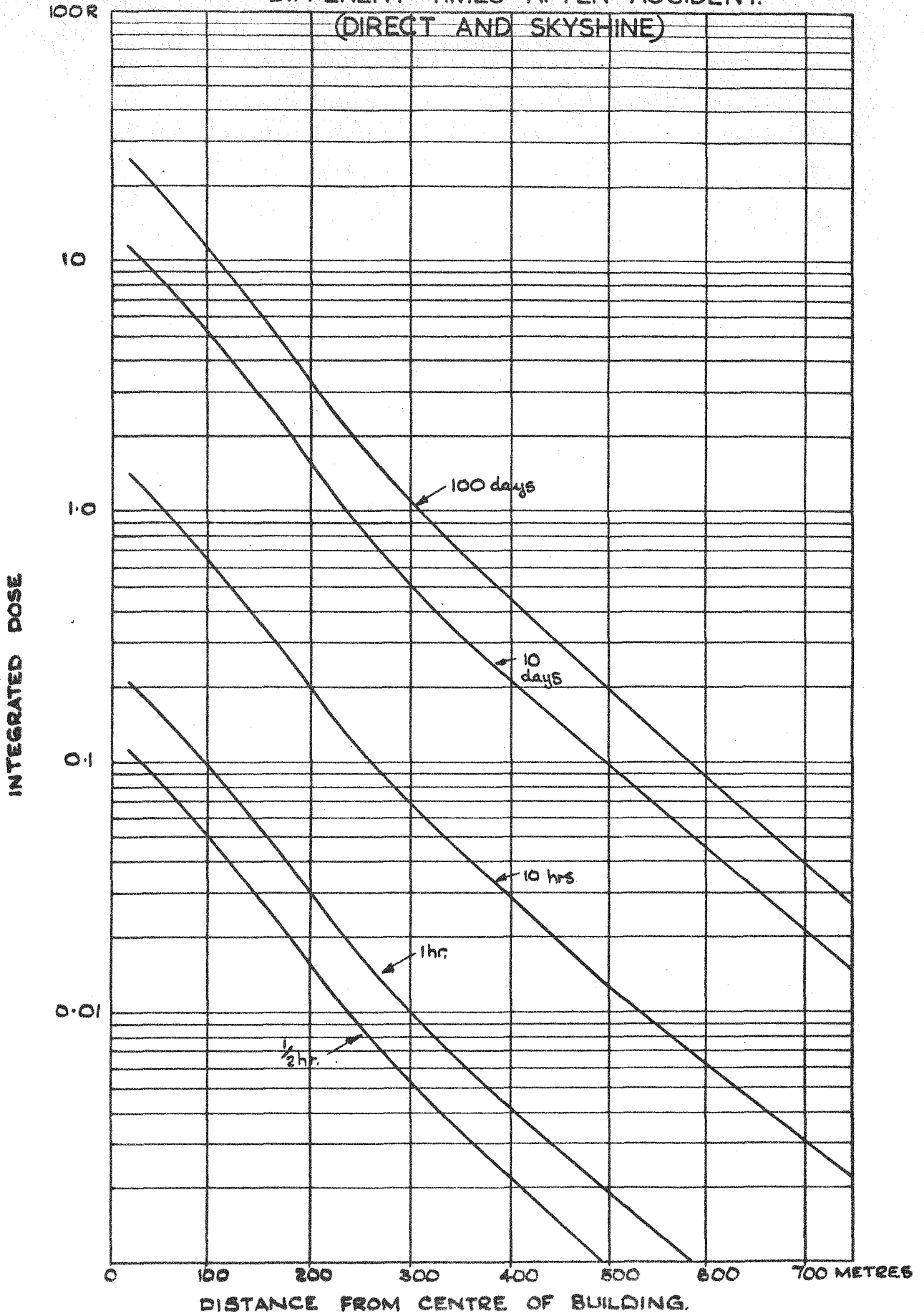
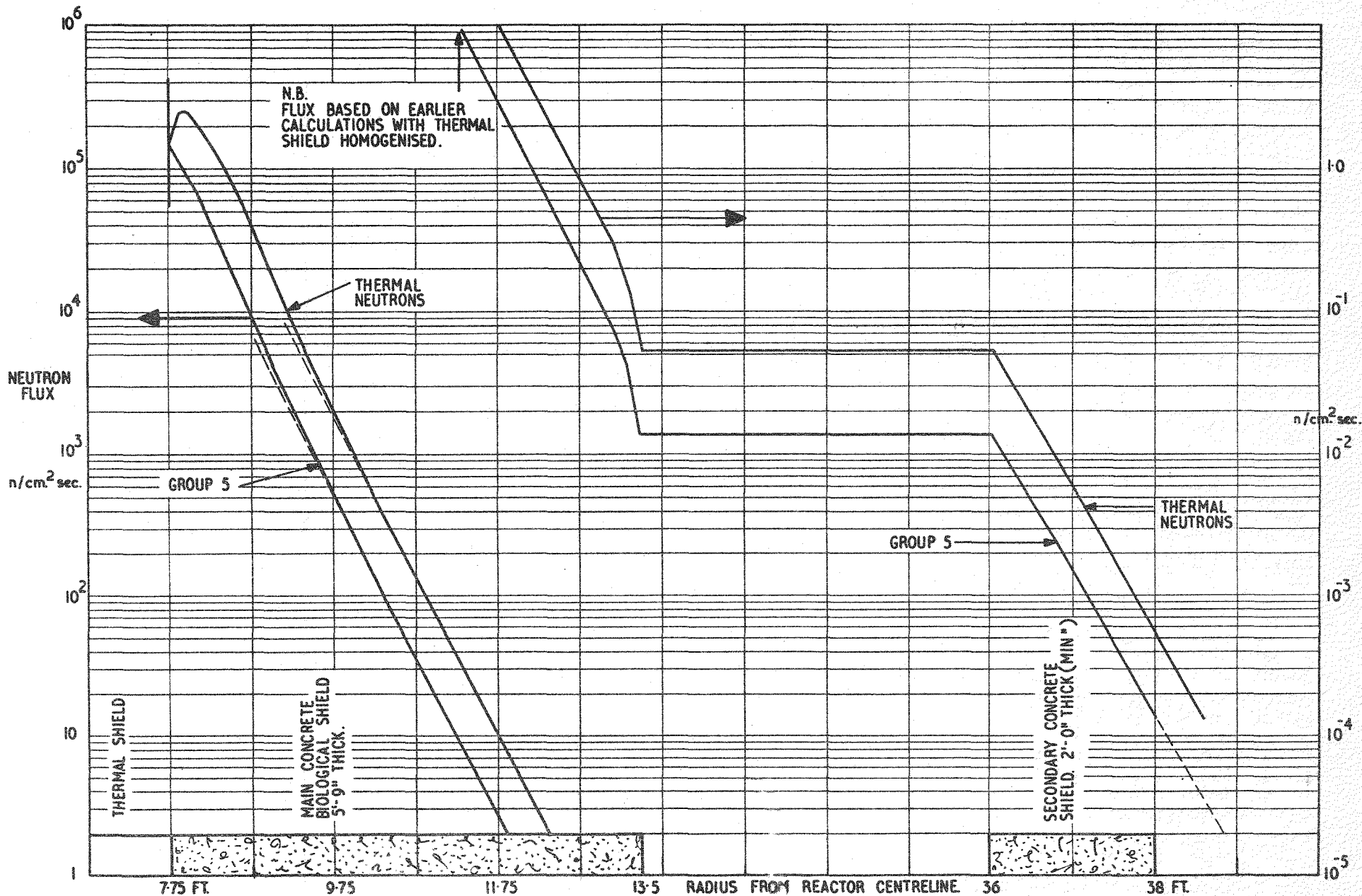


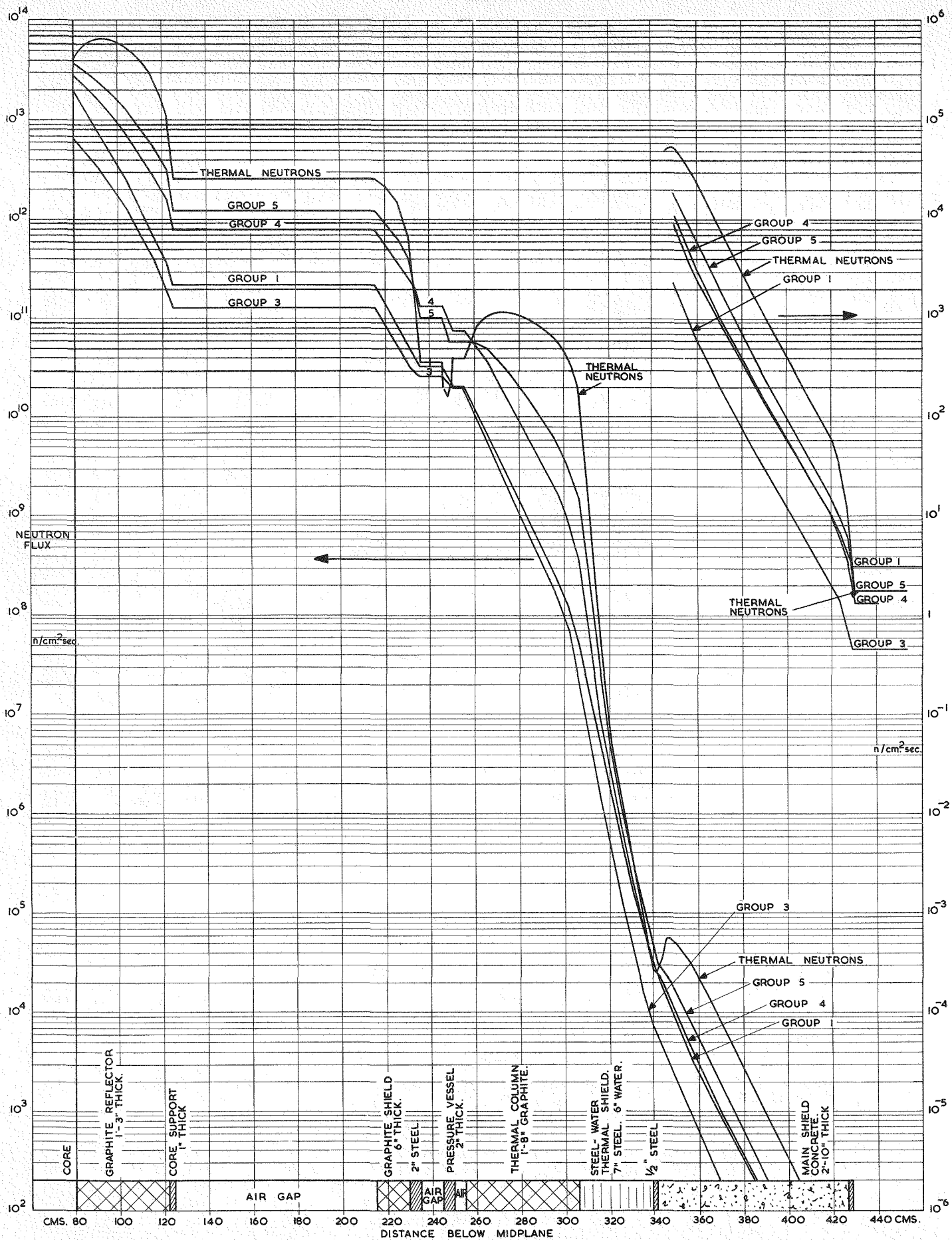
FIG 8

INTEGRATED DOSE FROM BUILDING FOR DIFFERENT TIMES AFTER ACCIDENT.





**FIG.9 NEUTRON FLUX OUTER RADIAL SHIELD**



**FIG. 10 NEUTRON FLUX DISTRIBUTION IN THE BOTTOM SHIELD.**  
 (ON AXIAL CENTRELINE)

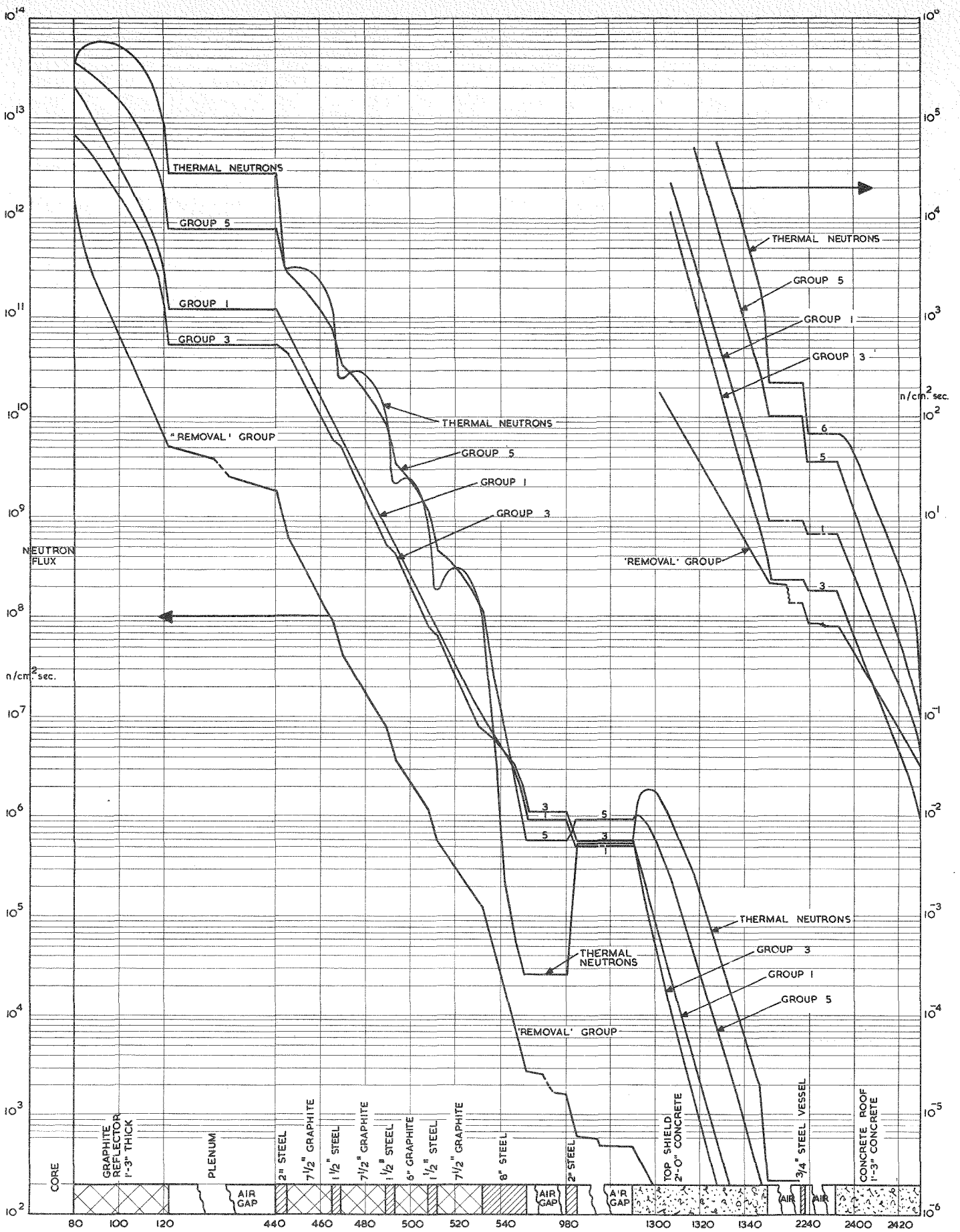
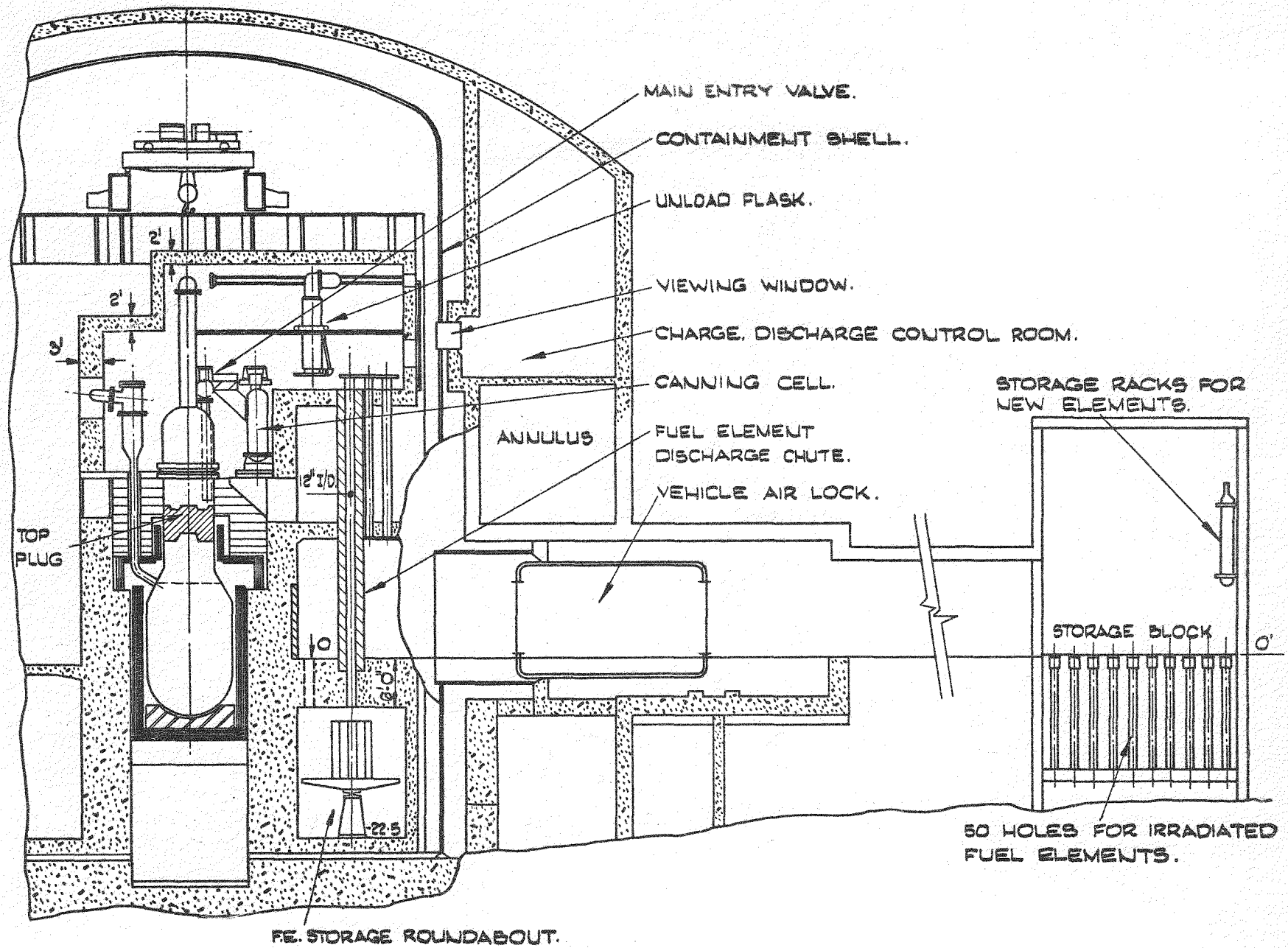


FIG. II NEUTRON FLUX DISTRIBUTION IN THE TOP SHIELD.  
(NEAR AXIAL CENTRELINE.)



**FIG. 12** CHARGE, DISCHARGE AND STORAGE OF FUEL ELEMENTS.

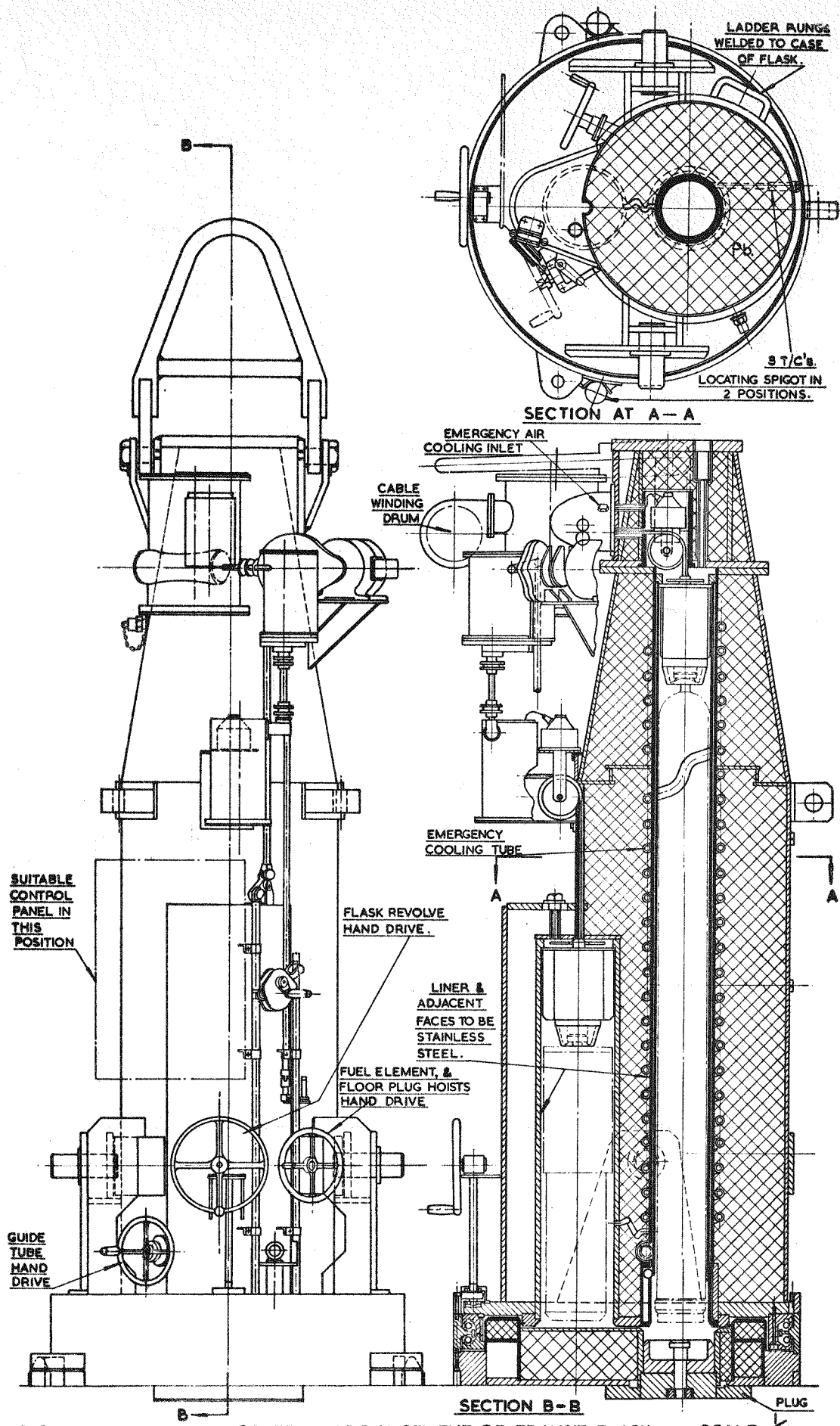


FIG- 13

GENERAL ARRANGEMENT OF TRANSIT FLASK.

SCALE: 1/16

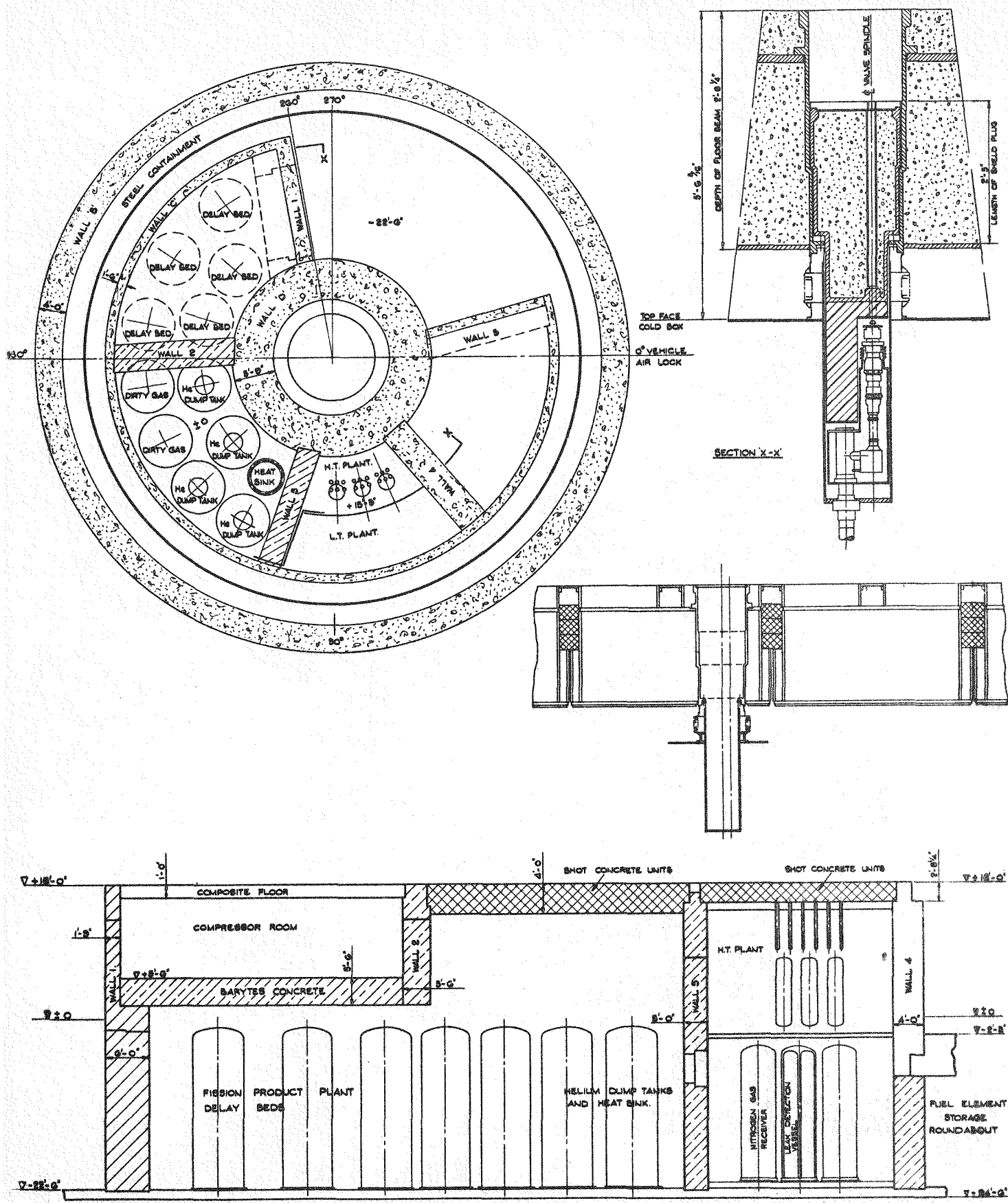
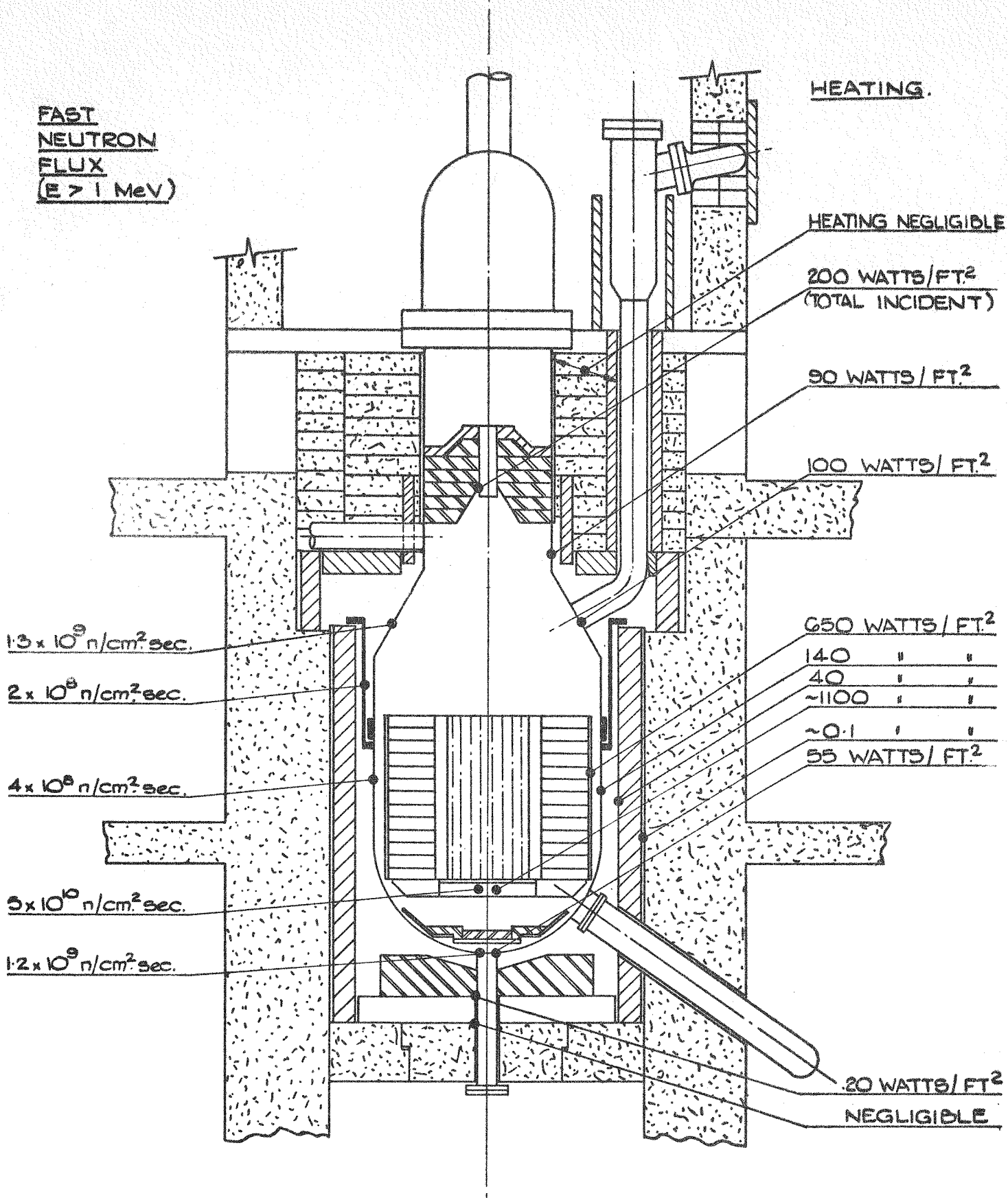


FIG. 14 FISSION PRODUCT PLANT SHIELDING.



**FIG. 15 NUCLEAR HEATING AND FAST NEUTRON FLUX  
- PRESSURE VESSEL.**

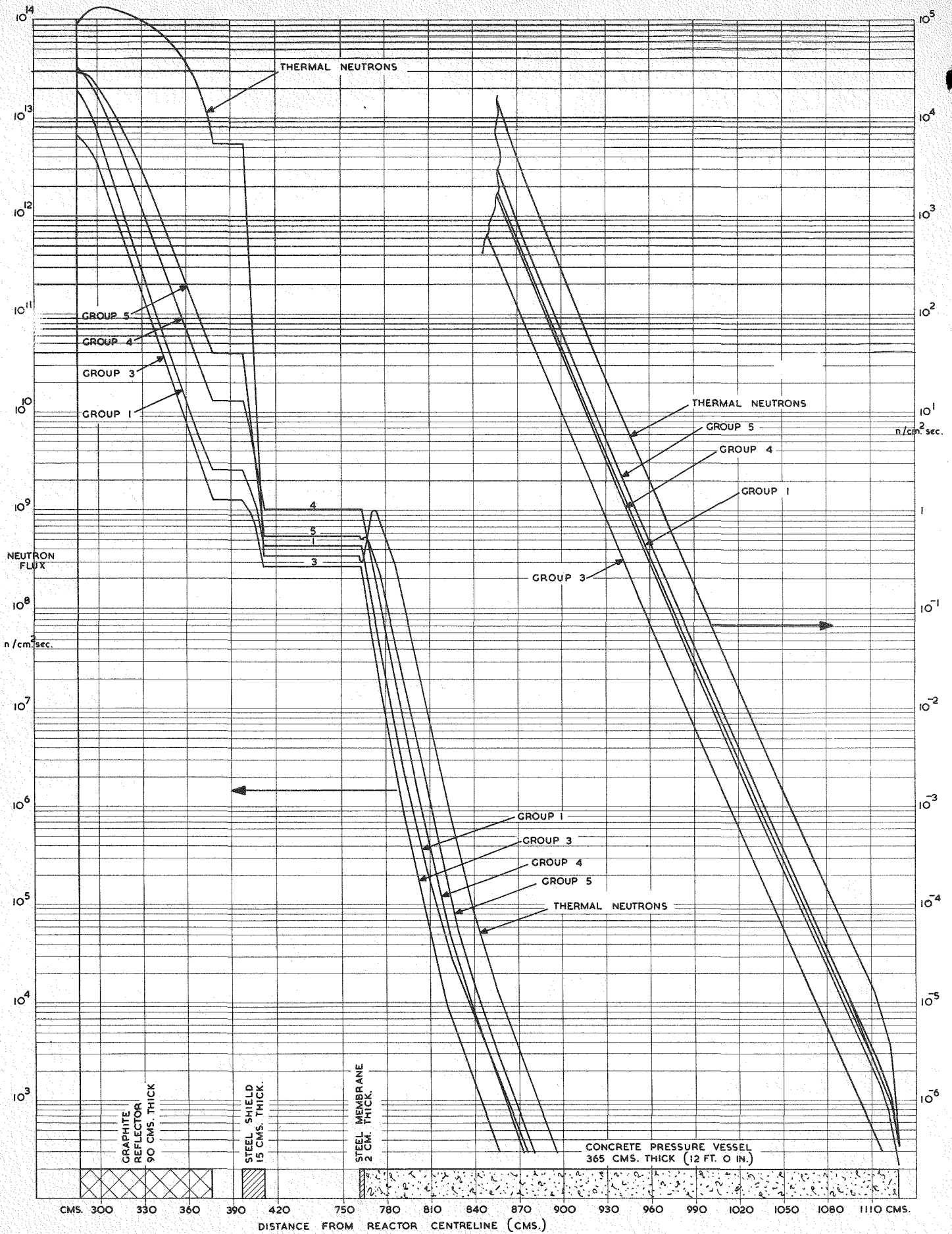


FIG.16 1250 MW. H.T.R. RADIAL SHIELD NEUTRON FLUX DISTRIBUTION.