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D. P. Report 135
Part I

~~DRAGON PROJECT~~ [redacted]

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Part I

O.E.C.D. HIGH TEMPERATURE REACTOR PROJECT

DRAGON

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Dragon Project Report

THE CIVIL H.T.R. REFERENCE DESIGN STUDY

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by

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DISCLAIMER

DRAGON PROJECT REPORT NO. 135 - PART I

REFERENCE DESIGN STUDY OF A HIGH TEMPERATURE REACTOR SYSTEM

1. INTRODUCTION

The Dragon Project has purchased a reference design study of a high temperature gas cooled graphite moderated reactor, which was prepared by the United Kingdom Atomic Energy Authority in the course of their own work.

2. SCOPE

The reference design study was based upon the information on the Dragon Reactor Experiment which was available up to January, 1962. The design study represents the development of that design into a large land-based power reactor, and thus does not in any sense represent an optimization of design features of high temperature reactors, but simply the scaling up of one particular reactor design concept.

3. CONTENTS

The documents which comprise the design study, are being circulated as Dragon Project Report No. 135. This report is being issued in three parts and the list of contents of each part is given below:-

Reg

REPORT NO.		APPENDIX NO.
135 - Part III	The Physics of the H.T.R. Reference Design	Appendix 8
	Effect of Uncertainty in Protactinium 233 Cross-Section on Core Performance of 1000 MW(H) High Temperature Reactor	Appendix 9
	Fuel Costs for Complete Core Replacement and Continuous Charge-Discharge Fuel Cycles of a High Temperature Reactor	Appendix 10
	The Approach to an Equilibrium Fuel Cycle and the Fuel Costs of a High Temperature Reactor with and without Removal of Fission Products	Appendix 11
	Spatial Burn-up in the H.T.R. - a Preliminary Assessment	Appendix 12

These documents were originally prepared as UKAEA internal reports, and were not written for wide distribution. The UKAEA agreed when the Design Study was purchased that it could be distributed to the Signatories but wished to point out that the original supporting papers were written in the form of memoranda rather than of finished reports. No attempt has been made to edit these documents.

4. SUMMARY

The work reported in D.P. Report 135 - Parts 1, 2 and 3, was based on an early design concept using a 'fission product emitting fuel' with purged fuel elements and an elaborate fission product trapping system. The report points out that if a 'fission product retaining fuel' could be developed and the fission product trapping system simplified, then appreciable economies would result. The work reported here therefore does not attempt to be an optimization of the many design parameters and hence does not exploit the full potentiality of the high temperature reactor system.

Despite these limitations most of the work reported here will be of considerable value in the reactor assessment studies to be done by the Project either directly or under contract. The aim of these studies is eventually to optimize the various design parameters and make a detailed design study of the system or systems likely to lead to an economic land based high temperature power reactor.

THE CIVIL H.T.R.

REFERENCE DESIGN STUDY

by

H.T.R. DESIGN OFFICE

Edited by J. D. Thorn
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SUMMARY

In a programme of work intended to establish the feasibility and costs of building an H.T.R. power station, investigation has been made of a design based closely on 'Dragon'.

The principle objectives in preparing the Reference Design were:-

(a) to determine the probable cost of power generated in a large station requiring the minimum development information beyond what was expected to become available from DRAGON,

(b) to examine the problems involved in scaling up this reactor type,

and (c) to act as a basis against which the cost and performance of other H.T.R. designs could be judged.

The principles of design are identical with those of Dragon although the embodiment of those principles has necessarily differed for many items due to the larger size of the plant.

This report presents the results of the design work. The conclusions are as follows:-

The capital cost to be expected for a large power reactor based as closely as possible on the Dragon design is not sufficiently low for the system to offer a great attraction.

Further study of safety requirements, leading to a revised containment design could lead to a saving of up to £6/KW. The considerations involved are likely to be largely applicable to any High Temperature Reactor using carbon based fuel elements.

The fission product purge system, associated building work and equipment is roughly assessed to cost £10/KW. The possibility of developing a satisfactory fission product retaining fuel is therefore of importance.

With a U/Th/U fuel cycle, there will be an optimum power density. Whilst the optimum value depends on the balance of costs in a particular concept, no justification is seen at present for striving for a design capable of power densities above 10 KW/litre; nor should a simple and cheap form of fuel element be rejected on account of limited power density, at least above say 5 KW/litre.

In applying the Dragon concept to a power reactor problems would arise which will not necessarily be solved by the successful development of Dragon. These include particularly -

(i) Assessment of the leakage of steam and water to be expected from steam generators, and of any special steps necessary in consequence.

(ii) Control elements and mechanisms suitable for the operation of control members within the core proper.

(iii) Design and manufacture of fuel elements, particularly the graphite components, suitable for a core of the required length.

(iv) Active handling of large components.

There is no reason to expect any of these to prove insuperable and some possible solutions have been incorporated in the Reference Design, but development work would be required.

INDEX

INDEX (Cont.)

LIST OF ILLUSTRATIONS

Figure

SECTION 1: BASIS OF DESIGN

Terms of Reference

1. In a programme of work intended to establish the feasibility and costs of building an H.T.R. power station, investigation has been made of a design based closely on 'Dragon'.

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(a) to determine the probable cost of power generated in a large station requiring the minimum development information beyond what was expected to become available from DRAGON,

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and (c) to act as a basis against which the cost and performance of other H.T.R. designs could be judged.

3. The principles of design are identical with those of Dragon although the embodiment of those principles has necessarily differed for many items due to the larger size of the plant.

4. This report presents the results of the design work. Studies of theoretical physics and fuel costs in support of, or closely associated with, this design are reported elsewhere [1-5].

Size and Site

5. In order that a cost comparison may be made with other reactors, it was decided to use the same assumptions with regard to size of plant and siting as were made in some earlier work. As a result the design has been based on a reactor of 1000 MW heat output driving a single 450 MW cross compound turbo-alternator at a site at Oldbury-on-Severn.

Fuel Element

6. As on Dragon, the reactor core consists of vertically disposed fuel elements abutting one against another. Each fuel element is a cluster of vertical rods spaced laterally to allow axial flow passages for the helium coolant. The rods contain fuel in the form of Thorium and Uranium Carbides enclosed in cylindrical graphite boxes which in turn are housed in long graphite sleeves of hexagonal outer profile. Both the sleeves and the boxes are made of "impermeable" graphite to minimise the leakage of fission products into the coolant. The weight of each fuel element is carried through a female spherical seat which makes a seal with its corresponding male spherical spike unit. These units are fixed to the permanent baseplates and are connected to a pipework system through which the purge gas is removed.

Purge Systems

7. A small amount of coolant is induced to flow through each fuel rod in order to sweep fission products out of the fuel as they form, and lead them away to a chemical separation plant, the purified helium then being returned to the circuit.

8. The purge flow enters the fuel sleeves at the bottom of the cluster, flows upwards between the sleeves and the boxes, enters the boxes at the top, flows downwards between the boxes and the fuel, passes over the charcoal traps (which also form part of the bottom reflector) and finally emerges through the conical/spherical joints at the base of the cluster, into the permanent piping system which leads to the chemical plant.

Pressure Circuit

9. The coolant flow path is arranged to sweep the pressure containing parts of the circuit with gas at inlet temperature. Concentric inlet and outlet ducts between heat exchangers and pressure vessel are used and differential thermal movements are accommodated by providing axial flexibility in the internal ducts only. The differential pressure across the sections carrying hot gas is small, and a failure of this barrier leads only to internal by-passing.

10. The heat exchangers are at a high level compared with the pressure vessel in order to provide as much natural thermal convection head as possible and thus the horizontal ducts connect the top of the pressure vessel with the bottom of the heat exchangers.

11. The pressure vessel, ducts and heat exchangers are supported at the same horizontal level, allowing the vertical thermal expansions of these components to occur independently. The heat exchangers are hung from vertical tie bars allowing them to swing outwards to accept the horizontal thermal movements. The hot box occupies most of the top of the vessel because of the need to move long vertical fuel elements in a lateral direction inside it (see Section 9).

Refuelling

12. Refuelling is carried out from above the core with the reactor shut down. The charge/discharge machine accepts the six clusters of fuel elements which are accessible from any one stand-pipe by means of a horizontal parallel motion transfer chute. The fuel is passed via a transfer facility to a gas-cooled storage carousel where it decays before being transported in a flask away from the building. A charge-chute machine, and a special purpose machine for handling broken fuel elements, television, and control equipment, are separate pieces of equipment.

Control

13. The control rods are of boron-carbide sheathed in graphite. The operating mechanisms are situated in the relatively cool region below the core, and actuate the control rods through push rods, the "out" position being above the core. The design provides for removal of control rods and mechanisms through the refuelling stand-pipes in the top of the reactor vessel.

Safety and Containment

14. The reactor is inherently safe as far as power excursions are concerned because of its negative temperature coefficient which is maintained through all phases of operation, the large heat capacity of the core, and the fact that the whole of the core is made from high temperature materials.

15. The two main hazards are associated with (a) the activity of the circuit and especially that of the purge chemical plant and (b) the chemical reactivity of the core materials at operating temperatures.

16. Both these hazards are containment problems. The first is concerned with preventing active materials escaping from the reactor building and the second is concerned with preventing air or water entering the primary circuit.

17. Two containments are used, the inner being of all-welded steel construction designed to withstand an internal pressure of 10 psi and containing the primary circuit, purge chemical plant and refuelling equipment. The outer containment is of concrete construction and the leakage of activity from the system after a serious accident can be further limited by recirculating the contents of this containment space through filters and a carbon bed. The concrete walls of the building also provide a biological shield for the remainder of the site against fission products which may condense on the inside face of the inner containment building. The annular space between the two containments is used for auxiliary plant rooms.

18. During the periods when the reactor is pressurised and the core hot the steel inner containment is filled with nitrogen to prevent the air/graphite reaction and the occurrence of a water gas explosion should a breach of the pressure circuit occur. During refuelling periods the nitrogen is replaced with air. The ventilation system for the inner containment cools and recirculates the nitrogen during reactor operation to maintain reasonable temperatures in the plant rooms and provides a nitrogen cooling circuit for the main concrete biological shield. During refuelling periods the same system cools and recirculates air inside the inner containment enabling access to be made to the refuelling floor and equipment.

SECTION 2: DESIGN CONDITIONS

Selection of Core Parameters

19. In the early stages of the study it was decided to make a survey of the possible parameters for the reference design applying limiting values where these were known. This was not intended to be an optimisation but merely the establishment of the range of parameter values within which a practicable specification of the reference design would lie. This work has been reported by Pexton and Staunton [6] and is summarised below.

20. The design limitations assumed were as follows:-

(a) The maximum allowable tensile stress in the graphite sleeves and boxes (without allowance for hot channel factors) is 1500 psi

(b) Minimum sleeve thickness 0.22 in

(c) Minimum box thickness 0.09 in

(d) Minimum fuel rod pitch 2.2 in

(e) The excess of core weight over upward gas force must be great enough to prevent any danger of the elements being lifted from their seatings

(f) Peak fuel temperature 2000°C
Nominal maximum fuel temperature 1800°C

(g) Maximum surface temperature 1050°C

(h) Maximum bulk coolant outlet temperature 750°C.

21. The following parameters were fixed for all calculations:-

Thermal Output	1000 MW
Radial reflector	3 ft
Axial reflector	2 ft
Radial clearance reflector/vessel	5% of reflector O.D.
Volume of core occupied by control rods	1/7
Pressure Vessel shape	Cylindrical
Pressure Vessel nominal stress	13,500 psi
Width of fuel rod spacing rib	$\frac{1}{8}$ in
Radial gap sleeve/box	0.005 in
Fuel pellet radial thickness	0.26 in
Graphite conductivity (⊥ to extrusion)	$(0.05 - 2 \times 10^{-5}T)$ cal/cm . sec . °C ($T < 1600°C$)

22. It was felt that the range of interest would be covered by considering cores which had voidage between 0.15 and 0.35 and average power density up to 20 KW/litre and the effects of variation in these quantities for the systems specified in the table overleaf, are illustrated in Figs. 22 to 31.

TABLE 1

Maximum Graphite Surface Temperature °C	Coolant Inlet Temperature °C	Coolant Outlet Temperature °C	Core L/D Ratio	Form Factors	Pressure Vessel Thickness In	Figure No. Giving Results
1000	350	750	0.6	As Fig. 36	4	22
950	350	750	0.6	" 36	4	23
1050	350	750	0.6	" 36	4	24
1000	300	700	0.6	" 36	4	25
1000	350	700	0.6	" 36	4	26
1000	350	750	0.6	" 37	4	27
1000	350	750	0.6	" 36	3	28
1000	350	750	0.4	" 38	4	29
1030	350	750	0.4	" 38	4	30
1050	350	750	0.4	" 38	4	31

23. The capital cost of a reactor system is expected to be reduced by an increase in mean power density. This provides an incentive to choose a reference design which has a power density close to the maximum attainable.

24. An increase in core voidage at a given power density results in a decrease of core pressure drop at the expense of a larger number of elements within the core. The minimum voidage on the other hand is set by the chosen limits for graphite stresses and minimum graphite thicknesses and by the desire to avoid fuel levitation without resorting to latches. The best compromise may be estimated from this approach but can be determined precisely only by careful optimisation. Optimisation studies showed that (see Fig. 22) at the temperature limits chosen (T_{sm} 1000°C, T in -350°C, T out -750°C) the maximum power density is limited to about 12 KW/litre by considerations of sleeve thickness, pitch, and spine temperature. Variation of the assumed temperature conditions produced no marked improvement (see Figs. 23 to 31).

25. The above considerations result in a determination of power density but still leave the fissile/fertile ratio and the mutually dependent variables fissile rating and fissile/graphite ratio free to be selected on nuclear performance alone. These are discussed further in Section 14.

26. Section 3 gives the thermal performance parameters for the selected core design which gives a power density of 11 KW/litre with L/D ratio 0.6 and voidage of 0.24. Approximate values are also given of the modified temperatures and stresses when allowance has been made for the hot channel factors given in Ref. [6].

27. The steam conditions to be used in the first instance were selected to be as advanced as possible without incurring uncertainty in capital costs as a result of a novel design. In later optimisations, the effects of other steam conditions could then be investigated. Thus 2300 psia and 1050°F were adapted with single reheat to 1050°F at 575 psia.

SECTION 3: MAIN PARAMETERS

CORE

Geometry

Length of core	11.34 ft
Diameter of core	18.7 ft
Axial Reflector depth	2 ft
Radial reflector thickness	3 ft
Uranium load	0.735 Tonnes
Number of clusters	420
Number of fuel rods	8400
Number of rods per cluster	20
Total Voidage	24%

Performance

Power Density	11.34 KW/litre
Total Heat output	1000 MW
Centre cluster heat output	3.31
Total Gas flow	1060 lb/sec
Mean fuel rating	1.0 MW/Kg U-235
Max. fuel rating	1.72 MW/Kg U-235
Axial form factor	1.235
Radial form factor	1.39
Core inlet gas temperature	350°C
Bulk gas temperature at core outlet	750°C
Maximum nominal fuel surface temperature	915°C
Maximum nominal fuel centre temperature	1820°C
In hand for local peaking	220°C
Peak centre temperature	2040°C
Mean surface heat flux	19.4 watts/cm^2
Max. (Nom.) heat flux	33.3 watts/cm^2
Max. (peak) heat flux	37.35 watts/cm^2
Thorium: U-235 ratio	8.2
Graphite: U-235 ratio	2360
System Pressure	330 psig
Coolant temperature at centre channel outlet	906°C
Channel equivalent diameter	0.505 in
Width of arms of trefoil channel	0.304 in
Total wetted perimeter of core cross section	5292 ft
Mean Reynolds Number	2.89×10^4
Coolant velocity at channel exit at 750°C	275 ft/sec
Mach Number at exit	.0446
Friction pressure drop across core	2.9 psi
Friction + acceleration pressure drop	3.34 psi
Total pressure drop	4.4 psi
Upward force acting on core due to pressure drop	71.32 Tonnes
Weight of core + axial reflectors	152 Tonnes

Fuel Element

Spine diameter	.828 in
Fuel inside diameter	.834 in
Fuel outside diameter	1.354 in
Box inside diameter	1.362 in
Box outside diameter	1.542 in
Sleeve inside diameter	1.552 in
Sleeve width across flats	1.88 in
Width of spacing rib	0.1 in
Pitch	2.2 in
Pitch length of box	11.34 in
Length of fuel per box	10.84 in
No. of boxes per rod	12
No. of rods per cluster	20
Max. nominal stress in box	1200 psi
Max. nominal stress in sleeve	1460 psi

Peak temperatures at position of max. spine temperature (hot channel)

Coolant	750°C
Surface of sleeve	1030°C
Inside sleeve	1260°C
Surface of box	1440°C
Inside box	1590°C
Outside fuel	1730°C
Spine	2040°C

PRESSURE CIRCUIT

Vessel

Shape of barrel	Cylindrical
Shape of ends	Hemispherical
Internal diameter	26 ft 8 in
Internal height (dome to dome)	56 ft $2\frac{1}{2}$ in
Max. plate thickness (top dome)	M.S. 5 in or DUCOL W 30 5 in
Barrel plate thickness	M.S. $4\frac{1}{4}$ in or DUCOL W 30 $3\frac{3}{8}$ in
Nominal max. stress	M.S. 15700 or DUCOL W 30 20000 psi
Design temperature	350°C
Design pressure	363 psig
Working pressure	330 psig
Number of refuelling standpipes (in top dome)	85
Pitch	25.4 in (triangular)
Bore	15 in
Number of control standpipes (in bottom dome)	85
Pitch	25.4 in (triangular)
Bore	5 in

Ducts

Outside diameter of outer duct	6 ft 4 in
Thickness	1 inch
Outside diameter of inner duct	4 ft 8 in
Thickness	$\frac{1}{2}$ inch

Heat Exchangers

Internal diameter	16 ft 10 in
Overall height	75 ft 10 in
Thickness	$2\frac{1}{4}$ in
Design Pressure	380 psig
Design Temperature	370°C
Plate Material	B.W.87
Design Stress	19,040 psi

CONTROL

System

Position of mechanisms	Below vessel
Number of control rods	85
Proportion of total fuel rod positions	1 in 15.2
Control rod diameter	5 in
Wall thickness	0.5 in
Material	Boron Carbide
Cladding	Graphite

Characteristics

Excess reactivity at start up		18%
Reactivity control by burnable poison	10%	
" " " control rods	17%	
	27%	
Shut down reactivity		-9%
"Hot to Cold" requirement		6%
Reactivity held by rods with reactor hot		2%

POWER PLANT

Heat Exchangers

Reactor power	1000 MW(h)
Number of heat exchangers	4
Reactor coolant outlet temperature	1382°F (750°C)
Reactor coolant inlet temperature	662°F (350°C)
Coolant working pressure	345 psia
Total gas mass flow	3.82×10^6 lb/hr
Gas temperature heat exchanger inlet	1382°F (750°C)
Gas temperature heat exchanger outlet	648°F (342°C)
Total heat exchanger duty	1020 MW
Feed water temperature	485°F
Feed water pressure at economiser inlet	2590 psia
Superheater outlet pressure	2420 psia
Superheater outlet temperature	1055°F
Total steam flow	2.94×10^6 lb/hr
Steam pressure at turbine stop valve	2300 psia
Superheated steam temperature at TSV	1050°F

Total reheater steam flow	2.58×10^6 lb/hr
Steam pressure at H.P. turbine exhaust	640 psia
Steam temperature at H.P. turbine exhaust	730°F
Steam pressure at reheater inlet	625 psia
Steam temperature at reheater inlet	728°F
Steam pressure at reheater outlet	590 psia
Steam temperature at reheater outlet	1057°F
Steam pressure at turbine reheat steam valve	575 psia
Steam temperature at turbine reheat steam valve	1050°F
Gas side pressure drop	3 psi

Turbine

Arrangement of cylinders	1 HP.1LP.2LP.
Length of last stage blades	36 in
Overall length of turbine	80 ft

Feed heating

Number of extraction points	7

Feed Pumps

Condensate extraction pumps	4 x 25% electric
Motor power (each)	150 H.P.
Booster feed pumps	3 x 50% electric
Motor power (each)	2000 H.P.
Main feed pumps	1 steam turbine 100%
Motor power for electric pumps (each)	7000 H.P.
Emergency feed pumps	2 steam turbine
Pumping capacity (each)	6×10^5 lb/hr 2600 psi
Shaft power (each)	1500 H.P.
Guaranteed feed pumps	2 electric
Pumping capacity (each)	3×10^4 lb/hr 2600 psi
Motor power (each)	50 H.P.
Condenser circulating water flow	30000 gpm

Alternator

Power at alternator terminals	443 MW
Overall length of alternator	40 ft
Length of stator core	19 ft
Weight of stator core and winding	160 tons
Approximate weight of stator complete	200 tons

Feed Water Make-up and Reserve

Capacity of make-up purification plant	4500 gal/hr
Emergency reserve feed water tank capacity	65000 gal

Power Plant Performance

Steam cycle efficiency	43.1%
Power consumption of electrical auxiliaries	28.8 MW
Nett electrical output of station	414 MW
Overall efficiency of station	41.3%

Auxiliary Power Supplies

Rating of unit auxiliary transformer	50	MVA
Rating of station service transformer	50	MVA

Emergency Power Supplies

Number of diesel generator sets	3	
Maximum continuous rating (each)	1000	KW
Number of motor generator sets	3	
Maximum continuous rating (each)	1000	KW
Battery	700	KWh over 20 min

SECTION 4: FUEL ELEMENT

Form

28. The form of the fuel element is generally similar to the "Dragon" fuel. Fuel bearing compacts in the form of annular cylinders are mounted on graphite spines about eleven inches long. Each assembly is housed in a thin-walled cylinder known as a box made from impregnated graphite. The boxes are capable of spigotting into one another to form a long composite unit of 12 boxes. The spigotted joints would be brazed together or cemented and baked, depending on which process proves to be most successful in the work which is being done in support of the "Dragon" fuel element. By this method a free passage for purge gas flow is provided from box to box but not through the joints from the inside of the box to the outside.

29. Each assembly of boxes is mounted into a long sleeve of impregnated graphite. The bore of the sleeve is circular and the outer profile is generally hexagonal, but with a narrow axial spacing rib along the centre line of each flat face. These fuel rods are supported in a cluster to form a fuel element.

30. Calculations were done to determine dimensions for the various components of the fuel rod which would result in the best compromise between:-

(a) Thermal stress in box and sleeve.

(b) Spine temperature and sleeve surface temperature.

(c) Mechanical strength and ease of manufacture.

31. The resulting cross sectional dimensions were similar to those used on "Dragon".

32. The "Dragon" fuel elements consist of 7-rod clusters. Because the length of the H.T.R. core is 15.3 ft (compared with 8 ft for "Dragon"), it was found necessary to increase the number of rods/elements to obtain adequate handling strength. This arrangement also reduces the off-load refuelling time.

33. Because the diameter of the H.T.R. core is so much larger than "Dragon" (18.7 ft compared with 3.5 ft), it is not possible to provide sufficient reactivity control by means of control rods operating in the radial reflectors only. Control rod positions within the core must be provided. A control rod and its sleeve takes up the same space as 7 fuel rods and the replacement of a complete 19 rod element by a control rod was uneconomic in space. A 7-rod cluster space was allocated to each control rod position and the remaining 12 cluster positions are occupied by fuel rods, two rods being attached to each of the six surrounding clusters.

34. Each cluster now occupies 21 fuel rod positions (Fig. 13), and weighs about 1,000 lb. Handling of the element at the top end during refuelling by means of a grab was not considered acceptable because the whole assembly would then be in tension. Although there is sufficient area available to give an acceptable mean tensile stress, a satisfactory design would have to ensure that (i) the load is equally shared amongst the fuel rods and (ii) the cemented or brazed joints are as strong as the sleeves both before and after irradiation. Possible weakening, distortion or damage to the graphite

as a result of handling and irradiation aggravate these difficulties.

35. Because of these doubts it has been considered prudent to lift each fuel element by passing a grab through it from above to make engagement with a steel coupling at the bottom. The centre fuel rod of each cluster is thus omitted. The rod which takes its place may be of graphite or could be used to contain burnable poison or irradiation samples or if axial temperature measurements were an essential requirement it may be possible to carry the necessary thermocouples in this rod, with trailing leads to the charge face. Its top end carries a stainless steel coupling identical with that at the bottom of each fuel element cluster. The same refuelling machine grab can now be used to remove the rod from the fuel element and by passing down the hole left by the rod remove the fuel element from the core. The centre rod would be of such a shape that the correct coolant flow is obtained for the surrounding fuel rods.

36. The top block of the fuel element is made of graphite and acts as part of the top reflector. The bottom block is made of steel. Both blocks give radial support to the fuel rods and are pierced by axial coolant passages. A system of radial passages in the bottom block collects the separate fuel rod purge streams together in the spike (Fig. 13).

37. The sleeves are designed in two lengths because of the difficulty of boring a blind hole 14 ft long by 1.54 in diameter while maintaining a tolerance of $\pm$ .001 in. The centre block of graphite, (Fig. 13), which is also required to give radial support provides the coupling means between the two halves of the sleeves which are brazed or cemented into it. If subsequent development shows that the sleeves may be manufactured in one length, there still remains therrequirement for radial support. The need to provide coolant passages limits the space available for this component to such an extent that manufacture in graphite would not be practicable.

38. Because all the sleeves are rigidly held together at the centre block, it is convenient to consider this level as the datum from which differential expansions and growths of sleeves occur. In order to accommodate these movements, three widely spaced fuel rods are attached rigidly to the top and bottom blocks; the remaining seventeen are attached flexibly. The flexible connections at the top consist of graphite spigots on the sleeves fitting in holes in the top block. These give radial support but allow axial movement. The flexible connections at the bottom have to provide leak-tight passages for the gaseous fission products, and for this reason stainless steel bellows are specified.

Re-entrant Purge System

39. A re-entrant purge gas system has been specified for HTR in which gas enters the sleeve at the bottom, passes upwards between the sleeve and the boxes, enters the top box, passes downwards between the fuel and the boxes and is finally taken off through the spike. Leakage through the sleeve into the main coolant is from the lower activity inlet purge stream only, consequently a much lower degree of main coolant contamination is achieved using graphite of the same permeability.

40. The top 18 in of each sleeve is left solid to form the main portion of the top reflector (the remainder is made up of the fuel element top blocks) so that the purge stream turns over to enter the boxes just below the top

reflector level. It must pass through the bottom reflector, however, to reach the spike. The boxes are therefore continued through the bottom reflector and connect with the stainless steel bellows. Through the bottom 2 ft the fuel inserts are replaced by charcoal, which in addition to acting as a reflector, forms the first item of the chemical plant by trapping the metal and halogen components in the fission product stream. After passing through these reflector traps the separate streams in a cluster are collected together in the bottom block and pass through the spike. This too is packed with charcoal and, because of its lower temperature, increases the trapping efficiency of the arrangements significantly.

41. The presence of these traps reduces considerably the duty required of the main chemical plant. Heat loads and activity levels are reduced. In the case of the HTR design, the spike has been made a part of the fuel element instead of being a part of the vessel internals to allow the charcoal traps to be renewed each time a fuel element is changed. Thus their efficiency is maintained, and the total quantity of fission products which has to be stored inside the containment over the lifetime of the reactor is reduced.

Fission Product Retention

42. Three main factors determine the effectiveness of the fuel assembly as a fission product retainer:-

(a) The permeability of the graphite can and sleeve.

(b) The soundness of the cemented or brazed joints.

(c) The ability of the ball and cone joint at the base of the fuel element to maintain a seal.

43. Taking these in order:-

(a) The design and estimate have been based on "impermeable" components manufactured in a fine grained graphite with multiple resin impregnation. This is similar to the "Dragon" material, although furfuryl alcohol, the resin used for impregnating "Dragon" components is not suitable for the boxes on HTR because of their thick bases. These would almost certainly cause spalling during normal vacuum impregnation. However, new resins are already being used which are suitable for thick specimens and which will make little difference to the cost. A permeability of 10^{-7} cm^2/sec has been the design figure for this type of graphite, although it is possible that graphite produced by new processes such as the G.E.C. Cellulose Process or the HXT process might achieve a permeability of 10^{-10} cm^2/sec by 1970.

(b) A development programme is at present proceeding in support of the "Dragon" fuel element aimed at producing a jointing method which will exhibit adequate strength at operating temperatures and be as impermeable as the bulk graphite. The first results using a zirconium braze have been disappointing. This type of braze fails under the temperature conditions expected in the reactor. Further work is in progress on molybdenum disilicide as a brazing material and the results so far have shown considerable promise.

(c) The joint between the fuel element and the permanent purge lines in the pressure vessel is in the form of a ball and cone seat which has an effective load of about 400 lb to maintain it. The pressure in the purge system will be about 20 psi less than the surrounding gas pressure at this point, so that a leaking seat will result in reactor coolant entering the purge system here and by-passing the fuel element. The normal purge flow is at very low velocity and restrictors in the purge lines will prevent this occurrence from interfering with the purge flow through other lines (which are connected in parallel). Depending on the size of the leak, the fuel element which has been by-passed may begin to discharge its fission products through the graphite box and sleeve into the main coolant. They will be carried round the circuit, eventually plating out on the cooler surfaces of the heat exchanger, and elsewhere.

44. The effect of one fuel element being by-passed is to increase the coolant activity very considerably and it is thus important to ensure that this method of jointing is satisfactory.

45. The cone feature has been provided at the bottom of the fuel element and the ball on the core baseplates to avoid the possibility of small pieces of debris collecting in the cone seats. The cone is thus renewed with each refuelling, and provision would be made to re-grind any spherical ball seat through the refuelling standpipes if this became necessary.

46. One factor which requires to be checked is the effect of thermal cycling on this large number of fuel elements which have no individual side restraint but which are bound together as an assembly by radial forces. It is not difficult to imagine that under such conditions some of the elements may ratchet themselves upwards off their seats by the small amount required to cause a leak.

Fuel Cycle

47. It had been proposed that reactor shut-down periods should occur once every 12 to 15 months. Planned maintenance and refuelling would occur during these periods. Fuel irradiation life was expected to be about three times the operational period between two consecutive shut-downs and so it was decided that one-third of the core would be replaced each time.

48. The three refuelling batches are not grouped separately but are evenly distributed throughout the core. The first shut-down period would normally occur after 932 days when k_{eff} has decreased to unity but preliminary work has shown that the system settles to an equilibrium cycle sooner with a shorter first fuel cycle life so that the first shut-down is proposed after 600 days. Subsequent irradiation periods would end when k_{eff} has decreased to unity.

49. During reprocessing, the fission products would be removed, Pa_{233} would decay to U-233 the thorium content would be restored to that of the original batch, and the rating in MW/kgm of fissionable isotope restored by adding 93% enriched uranium.

50. It is convenient to imagine four batches of fuel each equivalent to one third of the core. At any time after the first irradiation period three batches would be in the core and one in the reprocessing plant. If the batches are called A, B, C and D the cycle may be depicted as shown overleaf:-

TABLE 2

Irradiation Period No.		1	2	3	4	5	6	etc.
Fuel Batches	In Core	A B C	D B C	D A C	D A B	C A B	C D B	
	In Processing Plant	-	A	B	C	D	A	
Irradiation Period (Days)		600	570	498	483	517	474	
Initial Reactivity		18.0	12.0	12.5	12.0	11.8	11.8	

SECTION 5: GRAPHITE

Reflectors

51. The axial reflectors are formed as part of the fuel element assembly as described in Section 4. The radial reflector is a cylindrical annulus of Reactor Grade A graphite (see Fig. 13). The outermost 12" of the radial reflector are in the form of vertical columns of graphite which are generally square in cross section but with slightly tapered side faces enabling them to lock together as a solid circular arch. Restraint members similar to those used at Calder and A.G.R. are let into the outer face of this arch and when they are subjected to circumferential tightening they apply radial loads to support it. The remaining part of the reflector is built of hexagonal section bricks each one of which, in area, would enclose a hexagonal array of 19 fuel rod positions. The reflector bricks which abut the fuel elements have machined features along their faces which reproduce the correct coolant passages adjacent to the fuel. It is arranged that the inner reflector bricks can if necessary be replaced using the standard charge chute. The remainder of the hexagonal reflector bricks could be replaced by a more lengthy process involving the use of special equipment to give a longer reach than afforded by the charge chute. The graphite forming the outer arch operates wholly at gas inlet temperature (350°C) and this is considered as permanent graphite unlikely to need replacing.

52. A gap of 4" exists between the outer arch and the hexagonal brick section of reflector. Inlet gas passes down through this gap as well as through the gap between the arch and the vessel before returning up through the fuel elements. Axial graphite keys between adjacent hexagonal bricks prevent inlet gas leaking into the core while it passes down the 4" gap. Use is made of the difference in gas pressure which exists between the inlet gas at the top of the reflector and the outlet gas at the top of the core (about 4.6 psi) to provide a uniform radial restraining load to the core which would be automatically relaxed on shut-down when coolant circulation is reduced. However it was considered unwise to rely solely on this method of restraint for accident conditions. If a major breach of the main circuit occurred it would be possible for the inlet gas pressure outside the core to fall abruptly and at a greater rate than the fall in gas pressure inside the core. This could result in a transient reversal of the core restraining load with a consequent radial disruption of the core. The separately restrained arch provides support for the core under these conditions.

Dimensional Changes

53. It is estimated with present information that the effect of shrinkage on the radial reflector is small enough to present no problems. Should subsequent development work indicate that this is not so, bowing problems could be alleviated by use of shorter bricks.

54. Thermal expansion changes will also occur. Vertical expansions are unimportant, but horizontal expansions have been designed for. At the top of the core the fuel element temperature is much higher than the outer reflector temperature. This will result in widening of the gaps between the outer hexagonal bricks (the axial keys will be designed to accommodate this) and a reduction in width of the 4" annular gap by about 0.4".

55. At the bottom of the core the fuel elements being constrained to move with the ball joints, will move outwards at the expansion rate of the steel

diagrid. The outer reflector will expand only at the rate of graphite. This will also result in a reduction in width of the annular gap by about 0.35 in.

56. Dimensional changes in fuel element graphite become important if they occur as differential changes of length between rods within a particular fuel element. The method of flexible assembly of the rods into clusters as described in Section 4 ensures that this type of change in length cannot cause bowing of the assembly. The top and bottom blocks may tilt slightly due to length changes in the three solidly attached rods but sufficient clearance can be allowed between blocks at top and bottom to allow this to occur.

Stored Energy

57. During operation no graphite will experience less than inlet gas temperature (350°C) and present information shows that at this temperature the energy storage problem is small.

Graphite Costs

58. The graphite components in the core are expensive because of their intricate shapes, thin sections and close tolerances and also because of the expensive manufacturing operations associated with the production of highly impermeable graphite.

59. The weights and estimated costs of the various graphites are shown in the following table:-

TABLE 3

	WEIGHT tonne	COST £/tonne	COST £1000
Impermeable fuel element graphite (boxes and sleeves)	75	8,800	660
Other fuel element graphite (spines, top blocks, centre supports, etc).	45	1,100	49
Control mechanism graphite	12	1,000	12
Radial reflector graphite	150	530	79
Total	282	2,830	800

SECTION 6: REACTOR VESSEL

General Approach

60. Because the core consists entirely of the fuel elements placed side by side it is impossible to provide a separate standpipe in the pressure vessel for each fuel element. The design is thus limited to a multi-channel refuelling arrangement which requires means for lateral displacement of the fuel in the space between the top reflector and the vessel dome. The height required in this space for handling is that of the complete fuel stringer which includes the axial reflectors. As a result the shape required to be enclosed by the vessel is a cylinder in which the length is greater than the diameter (about 40 ft long by 25 ft diameter).

Internal Layout and Support

61. The ducts are attached to the top dome rather than to the barrel section to give (i) maximum natural circulation head, (ii) minimum neutron and gamma streaming through the duct passages. Because of the rigid outer duct system the vessel and heat exchangers are supported at the plane of the duct centre line. The vessel must therefore be supported from the top dome and it was impracticable to support the weight of the diagrid and core through the vertical walls of the vessel because they are no longer directly in line with the points of support and severe bending stresses occurred. The complete vessel is suspended from eight hangers attached to brackets on the outer surface of the top dome. Internal brackets directly below these carry hangers which support the internals, thus avoiding stresses in the vessel due to support of the internals.

62. The vessel hangers are suspended from a massive ring beam which is supported on legs, the feet of which rest on a corbel formed in the main biological shield. The level of the corbel in relation to the ducts is such that the principal of co-planar support is maintained.

Internals

63. The suspended diagrid carries 21 six-inch thick mild steel baseplates which are machined on their upper surface and have a coolant channel bored through them at the position of each of the 420 fuel element clusters. Each baseplate is supported on vertical jacking screws from the top surface of the diagrid and can thus be separately levelled. Horizontal jacking screws are provided at the gaps between baseplates to facilitate accurate positioning and to provide a means of locking the baseplate assembly together in the horizontal plane.

64. At each coolant channel a fuel element support stool is spigotted into the baseplates. At the lower end of the stool a central boss, carried on three radial arms, is fitted with the spherical seat which supports the fuel element cluster. The upper end of the stool encircles and gives radial support to the fuel element so that a single cluster is stable on its own account without relying on the presence of adjacent clusters.

65. Attached to the underside of the spherical seats are the fission product purge lines in $\frac{1}{4}$" N.B. austenitic tubing. These 420 pipes are collected together beneath the diagrid and suspended from it, run in vertical banks to the outer edge of the diagrid, and feed into a common

3 in bore ring main. A single 3 in bore line then passes through the pressure vessel wall connecting the purge system with the chemical plant. Because this pipework system is inacessible after the reactor has operated and because a burst pipe could prevent further reactor operation, very special precautions in manufacture and testing would be necessary.

66. A raised platform around the edge of the diagrid carries the graphite blocks forming the radial reflector and the diagrid hangers pass between the reflector and the vessel.

67. At each of the 85 control rod positions a control rod guide tube is attached to the diagrid and hangs vertically from it. Each guide tube connects with an extension to the appropriate nozzle in the bottom dome of the vessel by means of a simple sleeved joint which slides to allow differential thermal expansion to occur. Continuous channels are thus formed which give lateral support to the control rods and their associated push rods and by providing ports in the guide tubes the correct coolant flow along each control rod channel can be achieved.

Material Selection for Vessel

68. There would appear to be a clear advantage in using a low alloy steel rather than mild steel for the vessel. Because of the high integrity demanded, however, and because the possibility of inspection is restricted it is necessary that, before a material can be used for a reactor vessel, extensive experience should have been gained with its use in other less critical applications, and that extensive tests should have been performed on its properties.

69. Ducol W30, a low alloy steel, is to be used in heat exchangers for the Sizewell power station in 3 in thickness to gain experience in the hope that reactor vessels could later be made of this material. Low alloy steels of this type are also being used extensively in other pressure vessels, and boiler drums are being made from these steels (using machine welding) in thicknesses exceeding 5 in.

70. It appears then that adequate experience will have been gained with the use of Ducol W30 within the next eight years to decide positively on i s suitability.

71. For this reason, two alternative vessels are specified in this study: one in Ducol W30 and the other in mild steel. Each is designed for the same working pressure (330 psig) and the maximum plate thickness used in each is 5 in. This maximum thickness occurs in the top dome in each case. For the mild steel case it is necessary for the refuelling tube nozzles to contribute to the strength of the top dome and for this reason thick nozzles and deep, full penetration welds at the nozzles are used. Due to the higher allowable working stress for Ducol W30 (20,000 psi as compared with 15,700 psi) the ligament strength of a 5 in thick dome is sufficient without compensation so that thin nozzles can be used and much lighter welds.

72. Until Ducol has been thoroughly proved in other applications it has been thought prudent to limit the vessel design to the conditions quoted above. Should experience show, however, that full depth penetration welds around nozzles in 5 in Ducol are feasible and reliable then some compensation from thick nozzles would be possible and higher working pressures would be possible.

SECTION 7: HEAT EXCHANGERS AND STEAM PLANT

Heat Exchangers

Introduction

73. It was decided to use four heat exchangers on the basis of safety, duct size and cost. A specification was drawn up to which Babcock & Wilcox, Ltd. have prepared designs, performance data, and costs estimates under contract. [7]

74. The choice of steam conditions is not restricted by the reactor primary coolant temperatures, and it was therefore decided to specify the most advanced steam conditions practicable at the present time. Supercritical pressure was rejected because of unknown capital cost and the uncertainty about water leakage from the heat exchangers into the primary coolant circuit, and the choice was made of the most advanced subcritical steam conditions at present being specified for large steam turbine plant in this country. These steam conditions would lead to a steam cycle efficiency some 3% below that of a supercritical pressure cycle.

Water Leakage

75. It is known that small amounts of water leak from the water and steam side of the reactor heat exchanger into the primary coolant. Because of the high graphite temperature in this reactor it is important that this leakage should be very small, since most of the water entering the primary coolant will react with the core graphite resulting in its removal from the core.

76. Calculations indicated that the maximum tolerable leakage was probably of the order of 1 lb/day; it was specified that the heat exchangers should leak not more than $\frac{1}{2}$ lb/day. It has not been possible to obtain dependable information on the probable leakage rate. The leakage specification on existing heat exchangers is an order higher than the above figures but little difficulty is encountered in achieving the specification figure and it is believed that actual leakage rates obtained are of the order required for this design.

77. The Babcock and Wilcox design study threw no further light on the problem, and it is clear that experiments will be required to obtain further information on this problem.

Heat Exchanger Arrangements

78. Coolant flows to the heat exchangers from the reactor through the internally lagged inner duct, passing through an isolating valve in the base of the heat exchanger shell, and entering the tube bank at the bottom end. The gas flows upwards through the tube banks and returns downward in an annular space inside the shell around the tube banks, passing to the circulator in the base of the exchanger, through a second isolating valve, and so to the reactor through the annulus between inner and outer ducts. The shell of the heat exchanger and the ducts are thus maintained at reactor inlet temperature by the gas flow over them.

79. The steam side uses a "once-through" design - the water passing straight through the economizer, evaporator and superheater with no separating drum. A forced circulation boiler design, with drums and circulating pumps for

the evaporator section could be employed, the difference in cost being small.

80. In arranging the tube banks in the shell, upward flow of the steam-water mixture in the evaporator bank was chosen to assist steam separation. The arrangement of the superheater and reheater banks was considered in some detail, because of the need for economising in the use of high temperature materials.

81. It was found necessary to make part of the superheater, and probably all the reheater of stainless steel tubing. The lowest tube temperatures were obtained by arranging the stainless steel tubed part of the superheater - the secondary superheater - with parallel flow of steam and gas, and the remainder of the banks with counterflow.

Constructional Details

82. The tube banks in the heat exchangers are all in cross flow arrangement. The economizer, evaporator and primary superheater are of Babcock and Wilcox stud tube, in carbon steel; the secondary superheater and reheater are of plain austenitic stainless steel tube.

83. All the banks are made up from multiloop elements, joined to headers external to the heat exchanger shell. Penetrations through the shell are by thermal sleeves, and the penetrations through the baffle surrounding the tube banks, which must accommodate thermal expansion, are fitted with bellows.

84. It is considered that the heat exchanger shells would be erected complete, and tubed in situ. The tube elements are inserted from the bottom of the shell, through the circulator opening, and the banks are supported by hangers from the top of the unit.

85. A hot gas by-pass valve is fitted to assist in controlling gas temperatures at part load. It is expected that it would be fully closed during normal full-load operation.

Steam Cycle

86. The steam cycle is a conventional reheat cycle with seven stage regenerative feed water heating. The steam temperatures and pressures, and the feed temperature are similar to those in large conventional power stations in Britain.

Turbo-Alternator and Auxiliaries

87. The single turbo-alternator is larger in size than any single machine known to have been ordered in Britain, but not larger than is at present considered practicable. It is similar in all respects to other machines operating with these steam conditions.

88. The feed train is conventional, with two stage feed pumping, two low pressure heaters, one deaerating heater, four high pressure heaters. The booster feed pumps, which precede the high pressure feed heaters, are electrically driven; the main feed pump, after the high pressure heaters, is steam turbine driven, with standby electric pumps of 100% capacity.

89. It has been assumed that adequate cooling water supplies are available, and that no cooling towers are required. Since the design of the cooling water arrangements is dependent on the site chosen, no details other than

the water quantity required, and the approximate pump capacity have been calculated. An allowance has been made in the cost estimates, which would cover the cost of the installation on a typical site.

90. The feed water and make-up water plants are of the conventional ion exchange type, but are required to maintain a standard of water purity higher than in most conventional stations, because of the once-through design of boiler, which tends to concentrate impurities on the tube walls rather than in the drums as does a circulation type boiler. The water purification requirements were considered by Babcock and Wilcox as part of the heat exchanger study.

91. The two special features of the steam plant, required because of its application in a reactor rather than a conventional power station, are the guaranteed feed water supply arrangements, and the arrangements for dissipating excess heat from the reactor. These are dealt with below.

Guaranteed Feed Water Supply

92. To remove shut-down heat from the reactor, and cool the core fairly rapidly. 10% of the normal full load feed water flow is required. This is provided by two steam driven emergency feed pumps, designed to operate over the wide range of steam conditions obtaining in the period following reactor shut-down.

93. In addition to these pumps, which require steam from the heat exchangers (steam being always available while any substantial quantity of heat is being dissipated), there are two electrically driven pumps, of 50 horse power each, with guaranteed electrical supply.

Reactor Heat Dissipation

94. In the event of failure of the main turbo-alternator, the power transmission system, or the condenser cooling water supplies, the reactor shut-down is dissipated by blowing off the steam generated in the heat exchangers to atmosphere. A reserve supply of feed water (65,000 gallons) available at all times, is adequate to meet this demand for 24 hours following shut-down. After this, the make-up water treatment plant could be expected to continue supplying the small amount of feed required for an indefinite period.

SECTION 8: CIRCULATORS

Introduction

95. A circulator is mounted in the base of each heat exchanger, and supported off the heat exchanger pressure vessel. This allows a compact and convenient arrangement of ductwork, ensures that the circulators and motors are not in an area of high neutron or gamma irradiation, and gives better access for maintenance during a reactor shut-down. The only other convenient position for the circulators, on top of the heat exchangers, was rejected because it appeared to increase the size of the containment vessel necessary, required the internal baffle in the heat exchangers surrounding the tube banks to withstand external pressure and increased the likelihood of oil from the circulator bearings accidentally entering the gas circuit.

Description of Machine

96. The circulators are required to generate a pressure rise of approximately 9 lb/in^2 which is within the capacity of a single stage centrifugal machine, but would require more than one axial stage. A centrifugal circulator was therefore chosen to minimise the degree of overhang, an overhung design being considered desirable to keep the bearings outside the hot active circuit.

97. Several designs of centrifugal circulator were examined, the most efficient aerodynamically being one of slow speed and diameter too large to be accommodated.

98. A 3000 rpm circulator was chosen because it was of convenient size, and could be driven directly by a 2 pole 50 cycle motor. The blades are backwardly curved to ensure stability with the other circulators running in parallel.

99. A flow control scheme using variable speed drive was examined, but rejected because the first critical speed of the rotor is likely to be lower than the full load running speed. A gas bypass system would provide sufficient control of flow, but this was considered inferior to a scheme using adjustable inlet guide vanes. This has high efficiency at full power and the lower efficiency at low reactor power was not considered important.

100. The circulator motor is completely enclosed in a casing containing coolant at full pressure, to avoid the need for a rotating shaft seal.

101. A pony motor is fitted to the same shaft as the main motor and is capable of running the circulator at approximately 600 rpm. It is supplied with 50 cycle alternating current from the guaranteed bus-bars.

Special Features for Active Circuit

102. It is expected that fission products from the primary circuit might be plated out on any cool parts of the circuit. It will not be possible to prevent activity of this sort on the impeller or other circulator parts in the main gas stream but, to ease maintenance, special provisions are made to minimise the deposition of fission products in the motor casing.

103. A shaft seal is fitted on the impeller side of the top shaft bearing, and this separates the motor casing from the main heat exchanger shell. This

seal is of the mechanical type, in which there is a small controlled gas leakage. The centre of the seal is fed with clean helium returning to the circuit from the purge chemical plant, part of this feed flowing to the primary circuit past the seal and part back in the opposite direction to the motor casing. The motor casing is connected to the primary circuit through a filter which removes any oil which might be picked up in the motor due to defective oil seals on the bearings. The motor casing therefore runs at a slightly higher pressure than the primary circuit, and so long as the shaft seal is fed with helium, there is no contamination of the motor with fission products from the primary circuit.

104. A standstill seal, not normally operative, is fitted on the shaft. The motor can be lowered on to the seal before removing the motor casing or motor shaft.

105. A trolley, running on tracks in the floor under the circulator motors, is provided for handling the circulators and motors. The units are lifted and lowered by jacks attached to the trolley.

106. Removal of the circulator impeller and guide vanes assembly is more hazardous than removing the motor, and requires a gas tight flask to be attached to the motor flange. The impeller, with the seal assembly and guide vane assembly complete would be lowered into this vessel. A blank would be fitted in place of the seal assembly before removing the vessel containing the impeller.

SECTION 9: FUEL HANDLING AND STORAGE

Introduction

107. Fuel handling is arranged on the basis of annual off load refuelling. Up to one third of the core inventory (140 clusters) might be replaced at each refuelling, for which the reactor will be held at an internal pressure of 6 in w.g. with a coolant outlet temperature of 200°C.

Fuel Handling Sequence

108. Fuel clusters are dealt with in groups of six, each group being loaded from an individual standpipe.

109. New elements are loaded into a refuelling machine via a fuel intake lock located at charge floor level. The machine is transferred to an operational skirt located over a standpipe, and the fuel loaded into the core, replacing six irradiated clusters which are removed. A charge chute, previously positioned in the vessel by a chute machine, locates each cluster in turn above the required position.

110. The irradiated fuel is transferred from the charge machine to a transfer machine, via a discharge skirt connecting the machines. From the transfer machine the elements are discharged through a lock into a nitrogen cooled magazine, horizontally mounted to facilitate passage of fuel through the two containment barriers. After cooling, the fuel is discharged into coffins for transportation.

111. Design Features of Handling Equipment

(a) Chute Machine

In addition to locating the charge chute in the reactor, this machine also handles an assembly comprising the standpipe plug, standpipe biological shield and control rod guide tube.

No cooling circuit or relief valves are provided, the machine being designed for a pressure of 6 in w.g. at 200°C. Manually operated valve gear is provided at the nose. Shielding is limited to the bottom 9 ft of the machine.

(b) Charge Chute

The charge chute is basically a guide tube extending from the charge face to core surface. The bottom 18 ft of the chute can be moved radially by parallel motion and rotated to locate the fuel clusters above any of the six required positions. Drive for the chute actuation is through a manually operated gear box, in which the charge chute head is located.

(c) Refuelling Machine

This machine has shielding against gamma radiation only, giving a total weight of 200 tons. Closed cycle cooling of 130 KW capacity is installed, rejecting heat to the containment atmosphere. A delay of 12 hours is assumed before refuelling commences. No cooling is provided during transit from standpipe to discharge skirt, but even if the transit time were 1 hour, compared with an estimated 5 minutes, the mean fuel element surface temperature would be only 300°C.

Positions for seven fuel clusters are provided, allowing withdrawal of one element into the machine, thus giving a space in the core for insertion of the first element. Provision is also made for seven dummy fuel rods and a short gamma plug.

The pneumatically operated fuel element grab lifts the fuel element from the bottom end after removal of the dummy rod, thus eliminating tensile forces in the element during handling.

(d) Skirt and Gear Box

An 18 inch thick cast iron skirt locates over a standpipe and contains the charge chute gear box which is sealed at one end to the skirt and at the other to the standpipe. A rubber gasket contained in the top of the skirt makes the seal onto the chute and refuelling machines respectively, when these are located on the skirt, the seal being maintained by the dead weight of the machine.

(e) Special Purpose Machine

A special purpose machine is provided for handling broken fuel elements, control rods and television equipment. It incorporates a closed circuit cooling system of 20 KW capacity.

(f) Transfer Machine

This machine is used for transferring irradiated elements from the refuelling machine to the cooling magazine. It is basically a barrel, trunnion mounted on a trolley. Irradiated elements are loaded from the refuelling machine into the transfer machine, which rotates from a vertical to horizontal position for transfer of elements to the cooling magazine by means of a built in ram. Suitable sealing arrangements are made at each stage. No cooling is provided, fuel element thermal capacity being sufficient to prevent excessive temperature rise.

(g) Cooling Magazine

The cooling magazine has a capacity of $1^1/3$ times one core inventory i.e. 560 elements. It is mounted horizontally on rollers, which transmit the load from two ring beams, forming the main structure of the magazine, to the containment structure.

Nitrogen cooling is provided, arrangements being made to prevent short circuiting of the gas through unloaded fuel positions. Heat loading of the magazine for the case of a normal 1/3 core discharge, full core discharge, and full discharge in addition to the normal 1/3 discharge is shown in Fig. 15. This shows that the maximum heat load for 1/3 core discharge is 1 MW, and 2.1 MW for a full core. The cooling is arranged as two units each of 1 MW capacity. In emergency, both can run in parallel. If total coolant failure occurs the magazine vault can be water flooded without a criticality problem.

To maintain double containment an air lock is provided on the outlet side of the cooling magazine in which the fuel is transferred into transport coffins, by means of a ramming machine installed in the magazine.

Refuelling Times

112. A breakdown of refuelling times is summarised below:-

Fuel element location, grabbing and hoisting	195 minutes
Purging	225 minutes
Crane travelling, grabbing and hoisting	132 minutes
Standpipe operational times (plug removal, gear box and chute location)	90 minutes
	642 minutes
	= 10.7 hours

113. This is the time for the complete series of operations at one standpipe. Thus the minimum possible time for refuelling 1/3 core (28 standpipes) is $12\frac{1}{2}$ days. If two shifts were allowed for refuelling one standpipe the time required would be 18.7 days on three shift working or 28 days for two shift working.

SECTION 10: CHEMICAL PLANT

Purge Purification Plant

114. The purpose of the plant is to remove the fission products and the chemical impurities from the purge gas stream. It has been designed on the following assumptions.

(a) The reflector and spike traps remove all the metals and cause a 5 days delay of iodine. (This delay reduces the halogen heat load on the external plant from 1 MW to 80 KW, and eliminates 5.7 MW due to metals.)

(b) Water leakage from the heat exchangers into the circuit will be limited to 1 lb per day.

115. Figure 20 shows the flow diagram for the plant. The gases leaving the reactor (910 lb per hour) are cooled from 350°C to 60°C in a precooler to reduce the heat load on the delay beds. The precooler is in the form of a plate-fin gas/gas heat exchanger in stainless steel.

116. The gas then passes to the charcoal filled tubes of the main delay beds in which the krypton and xenon isotopes are delayed for 15 and 200 hours respectively. These tubes are water cooled, cooling water entering at 25°C and removing 38 KW from each bed. The shell is 11 ft diameter by 20 ft long and twenty such delay beds are connected in parallel.

117. On leaving the delay beds the purge stream is heated in a regenerative helium/helium plate fin heat exchanger in stainless steel and by an electrical resistance heater to 450°C. This temperature is required for reaction in the copper oxide beds. Sufficient resistance heating is installed to heat the gas at start up when the heat exchanger is not operative. The heating elements are arranged in banks with only a fraction of them in use during steady operating conditions giving spare capacity to cover heater failure.

118. The carbon monoxide and hydrogen impurities are oxidised to carbon dioxide and water vapour in the copper oxide beds, each of which contains about two tons of copper oxide inside a cylindrical vessel 3 ft diameter by 13 ft long. Electrical trace heating elements are wound around the vessel which is finally lagged with 4 in thick high grade asbestos. Four such units are connected in series.

119. The gas is then cooled in the secondary side of the regenerative heat exchanger described above, leaving at 50°C to pass into the molecular sieve beds. These remove the carbon dioxide and water vapour formed in the copper oxide beds. Each bed is 2 ft 3 in diameter by 10 ft long containing 1000 lb of Linde molecular sieve type 5A. A 25 KW resistance heater is embedded throughout the sieve material. Two of these units are provided and connected in parallel. They are operated alternately for 7 day periods and while one is in use, the CO_2 and water vapour are being driven off the other by means of the built in heaters. These gaseous impurities are stored under pressure in the waste gas receivers.

120. The final section of the plant scavenges the last traces of CO_2 and water vapour and removes the remaining xenon and krypton fission product gases at low temperature. Two regenerative heat exchangers in parallel are provided in which the main gas stream is cooled from 50°C to −186°C by

means of the returning flow of helium entering at $-196^{\circ}C$. These are plate fin type exchangers in aluminium alloy. The two heat exchangers are used alternately and while one is in use the other is allowed to warm up and vaporise the small amounts of solid CO_2 and ice which may have escaped through the molecular sieves.

121. The gas stream then passes to liquid nitrogen cooled traps. These are in the form of 50 tubes 2" N.B. x 10 ft long, bent into the shape of "hair pin" heat exchanger elements, mounted in a tube plate and enclosed in a shell 3 ft 6 in diameter by 6 ft long. The tubes are packed with activated charcoal and the tube/shell interspace cooled with liquid nitrogen. The main gas stream passes through the charcoal packed tubes, where the noble fission product gases are adsorbed. Three traps are provided connected in such a way that any two can be used in series. Each trap runs for thirty days before being brought out of circuit and regenerated. The xenon and krypton are drawn off during regeneration and stored under pressure in the waste gas receivers.

122. The purge gas circulators then return the purified helium to the reactor.

123. The gas line from the reactor to the delay beds is in the form of two concentric tubes. Purge gas from the reactor passes along the inner tube and return gas from the circulators passes in the opposite direction along the annular passage. There are two reasons for this arrangement: firstly, the purge stream is extremely active between the vessel and the delay beds before the xenon and krypton have decayed, and by jacketing this section of the line with clean helium the chances of leakage to atmosphere are very much reduced. Secondly, the arrangement helps to cool the purge line before reaching the delay beds and also heats the return gas before it enters the reactor. A sampling system draws off gas from the annular passage at points along its length to detect leaks in the internal pipe.

124. The chemical impurity removal section of the plant has been placed before the liquid nitrogen cooled traps in order that these traps shall be protected from interference by chemical impurities which might inhibit the adsorption of the noble gases. There are strong advantages in removing the noble gases before the chemical impurity removal plant so that the latter equipment, parts of which are regularly regenerated, would then be operating on an essentially inactive gas stream. However, without more detailed information of the adsorption of mixed gases on charcoal at low temperature it appears at the moment desirable to place the chemical plant up stream of the noble gas traps.

Exhaust Chemical Plant

125. Any gaseous effluent from the reactor building can be directed straight to the stack or through the exhaust chemical plant to the stack depending on its activity. The heaviest duty required of the plant would be in discharging the contents of the inner containment after an accident involving a breach of the main circuit. Based on these conditions a throughput of 100,000 cfm has been specified. The plant is designed as two identical units connected in parallel and each capable of processing 50,000 cfm. One unit would be in general use dealing with normal ventilation effluent, small volume purging operations, etc. and the second unit would be brought in as required for the accident condition.

126. Each unit consists of two lines connected in parellel. Each line

(for 25,000 cfm) contains a coarse and absolute filter, two carbon beds, a further filter and a fan.

127. The carbon bed takes the form of an annular array of 8-12 mesh Sutcliffe Speakman 208/C activated carbon supported between two cylinders of perforated M.S. plate and enclosed in a vessel 4 ft 6 in diameter by 22 ft long. The gas flows radially through the thickness of the annulus from the outside to the inside.

128. The absolute filters will take out fission products which occur in particulate form. The carbon beds will take up the iodine and any other vaporised fission products. The final filters are intended to stop any carry over of carbon particles.

129. If required after an accident, one of these plants may be used to recirculate the contents of the outer containment space to reduce the effects of leakage from this space.

SECTION 11: CONTROL AND INSTRUMENTATION

Control Requirements

130. The optimum thorium:uranium ratio is shown in Section 3 to be 8.2. This gives an initial reactivity of 17.9%. The fuel cycle reactivity variation necessitates a control range of about 18%, estimated hot-to-cold reactivity change is in the neighbourhood of 6% and, allowing a 3% safety margin, the total requirement would be 27%.

131. Throughout the reactor life the moderator temperature coefficient over the operating temperature range is expected to lie between 0 and $-2 \times 10^{-5}/^{o}C$. The fuel temperature coefficient is $-2.2 \times 10^{-5}/^{o}C$ initially and varies only slightly through reactor life.

132. Figure 39 shows the reactivity held for various numbers of control rods of different diameters. Because of the fuel geometry, the present core design provides 85 positions for rods 4 inches diameter. The use of all 85 positions holds 31% reactivity. Alternatively, burnable poison could conveniently be used to hold 10%, leaving only 17% to be taken by a reduced number of control rods.

Control Rods and Mechanisms

133. The design shows the control rod mechanisms in standpipes at the bottom of the pressure vessel. The reasons for this choice were (i) the mechanisms and mechanical connections are at the temperature of the coolant inlet and (ii) the refuelling operations are not complicated by having to disconnect remotely and remove control rod mechanisms from the refuelling standpipes at the top of the reactor vessel before each refuelling. At the same time it was desirable for the rods to be above the core in the "out" position so that gravity would be available to operate the rods when rapid shut down was required. No penalty has to be paid in containment building height which is already controlled by other features.

134. For these reasons the present design is proposed. The mechanism which is installed on bottom standpipes imparts an axial motion to a steel push rod inside the standpipe. This carries the control rod consisting of a boron carbide tube sheathed inside and out with graphite sleeves via an intermediate push rod made of graphite to withstand the high gas outlet temperature at the top of the core. The insertion of a moderator as the control rod is withdrawn increases the efficiency of the control rods.

135. When a control rod is in the "out" position it is supported laterally by a control rod guide tube. The control rod positions in the core occur directly beneath the refuelling standpipes enabling the guide tube to be suspended from the shield plug in the standpipe. When the charge chute machine removes the plug the guide tube is removed with it. The handling of the plug and guide tube does not increase the height of the charge chute machine. A second advantage resulting from the relative positions of control rods and standpipes is the ability to withdraw control rods and both push rods up through the refuelling standpipes, no lateral displacement being necessary. Withdrawal of these components through the control rod standpipes at the bottom of the reactor vessel would have required additional height.

137. Three methods of operation for the control rod mechanisms have been considered: mechanical, pneumatic and electrical. No final detailed design is proposed in this report as it has not proved possible to verify the preferred design physically in the time available. This is of the form of an electrical linear "stepping" motor (see Fig. 16) with the operating coils outside a thimble tube and therefore accessible for maintenance. The only moving parts inside the circuit are the control rods and associated push rods, the lower steel push rod being made up of alternate slabs of magnetic and non-magnetic material. The push rod is lifted by sequential energisation of the coils. Removal of supply results in the rod falling into the core.

138. A pneumatic or electro-mechanical system could be designed in the space available and the choice of system would have little effect on the overall cost of the plant.

Secondary Shut Down Equipment

139. No secondary shut down equipment is proposed at the moment. However, if it were to be decided that such equipment is necessary it could be provided in a form similar to that used on Calder and A.G.R. A supply of boron containing shot could be held at the top of each control rod guide tube and when required, poured through the control rod into the hollow graphite push rod. It would also be possible to provide flexible rods in a similar position.

Faulty Fuel Element Detection

140. If a fuel rod became damaged or a fuel element bottom seal was leaking the purge stream through that element would be partially or wholly by-passed. Fission products would permeate at a greater rate than usual into the main coolant through the unpurged section of fuel element. If the leakage was very small it might be possible to continue operating until the next refuelling period, a check being kept on the main coolant activity. If this activity rose too sharply, however, it would be necessary to replace the offending element.

141. In order to determine which purge line is leaking it is the intention on "Dragon" to monitor the flow rates in all the purge lines. A leak into the system will cause an increase in flow. The adoption of this system on the HTR Design would result in a much greater degree of complexity because of the number of pipes to be brought out through the vessel. On the present design the purge flows are collected together inside the vessel and emerge through the vessel wall as a single 3 inch bore pipe. If all the purge flows had to be monitored independently they would have to emerge through the vessel as 420 separate pipes in order that the flow measuring devices may be outside the vessel. Otherwise the flow measuring devices would have to be in inaccessible positions inside the pressure vessel with at least 420 (and perhaps 840) instrument lines piercing the vessel wall.

142. A further disadvantage of this system of leak detection is that if the source of leakage is a cracked fuel rod, the flow in the purge line may not be significantly affected because of the effect of the purge flows from the other 19 rods. It is therefore proposed to use the control rod support tubes to form the basis for a system of sample lines. Coolant would be drawn up a tube as a representative sample of the gas passing through the six fuel elements associated with that particular refuelling group. The sample would be carried

up through the biological shield plug in the standpipe and emerge through a radial passage in the plug between two circumferential seals (see Fig. 15). Permanent piping would connect the standpipes at this level with detection equipment. The monitoring could operate continuously and give an indication of the severity of a fault and the location of the group of six elements containing it. If it were decided to replace the damaged element, the reactor would be shut down and depressurised and the charge chute inserted. The chute could carry a small bore sample line to detect which of the six fuel elements is faulty.

Temperature Measurement

143. Because of the complexity of the fuel and high temperatures involved, no measurement of fuel surface temperature has been provided for. It may be possible to utilise the dummy fuel rods with trailing leads to measure local gas temperatures within the core. Gas at inlet temperature could be allowed to flow up through a central hole to keep the thermocouples cool.

144. The control rod guide tubes are utilised to carry thermocouples to measure the temperature of the outlet gas at the top of the core. These thermocouples may be renewed any time a guide tube is removed for refuelling.

SECTION 12: CONTAINMENT

Philosophy

145. As on Dragon two containment shells are provided. The inner steel shell is cylindrical with torospherical ends 127 feet in diameter by 194 feet overall height manufactured in $1\frac{1}{8}$ inch thick mild steel plate. The outer shell is reinforced concrete 187 feet diameter by 160 feet height above ground level and 24 inches thick.

146. In the event of an accident involving a breach of the main circuit, the gaseous emissions would be contained within the inner shell. The volume inside the outer containment provides a holdup for gaseous fission products which may leak from the inner containment. A figure of 0.1% per day has been assumed for this leakage and in this case the delay afforded by the double containment reduces the iodine escape to a safe figure. Strontium, which is the other dangerous isotope under these conditions, is not significantly affected by the delay and provision is made for the contents of the outer containment to be continuously recirculated through the exhaust chemical plant at 100,000 cfm.

147. The concrete walls of the outer containment also serve to shield personnel outside the plant from fission products which may have condensed on the steel walls of the inner shell after a circuit breach.

Nitrogen Atmosphere

148. If air or water were allowed to enter the circuit after an accident, extensive chemical reactions with the core material would result releasing large amounts of heat, and giving the possibility of explosive mixtures of carbon monoxide or hydrogen. For this reason the whole of the inner containment will be filled with nitrogen during reactor operation but purged and refilled with air during refuelling periods. Access to the inner containment would normally be possible only during refuelling. An extensive ducted ventilation system is provided which collects the air or nitrogen from all parts of the inner containment, cools it, and recirculates it at a rate of 144,000 cfm. During operation 200 cfm of clean nitrogen (which has boiled off in the chemical plant nitrogen cooled traps) is bled into the system and the same amount discharged up the stack. This serves to keep the activity level down inside the shell. During refuelling periods conditioned air is similarly introduced into the circuit.

149. A similar but entirely separate ventilation system serves the outer containment. This is always air filled and access at all times is possible into the outer containment.

Design of Inner Containment Shell

150. The steel shell has been designed to withstand an external pressure of 0.3 psi which is considered to be the maximum duty required of the shell.

151. The internal test pressure is 14 psi which is adequate for the pressure conditions caused by the escape of the primary circuit and the contents of one heat exchanger.

SECTION 13: SAFETY

Reactor Accident Conditions

152. The safety of the H.T.R. reference design is discussed under two headings (i) Accidents at normal operating pressure (345 psia) and (ii) Primary circuit depressurisation to containment equilibrium pressure (30 psia).

(i) Accidents at Normal Pressure

(a) Failure of Forced Circulation

Natural convection cooling would limit the coolant temperature to a maximum of 1140°C and the maximum fuel temperature would be less than 100°C higher. Hence, apart from slight increase in release of fission products from the fuel, no damage to the reactor core or components would result. Operation of one main circulator at full power or three circulators at 20% power each by use of their pony motors would prevent coolant and fuel temperatures rising above normal except for a very short transient period in which the temperature rise would be less than 100°C.

Rapid control of superheat and reheat steam temperatures would prevent excessive rise in heat exchanger tubing temperature following complete failure of forced circulation. Increase in tubing temperature to 1000°C would not produce failure but increased steam leakage might result.

(b) Reactivity Excursion

The control rods are calculated to control 31% reactivity which would be sufficient to control safely the cold clean core without use of burnable poisons, as proposed. The reactor is intrinsically stable due to the negative fuel and moderator coefficients of reactivity, which would be maintained throughout the proposed fuel life. Calculations have shown that for a 2% accidental addition of reactivity at 0.05% per second, either at full power or at start-up, which is a very pessimistic assumption, the coolant temperature would not exceed 950°C providing full forced convection cooling was available. Hence the control and shutdown systems, as proposed, would keep all feasible reactivity excursions within safe limits. Slight increase in release of fission products from the fuel would occur but would not constitute a hazard condition.

(c) Failure of Heat Exchanger Tubing

This accident would produce the worst overpressure condition in the primary circuit, and the relief valve and dump tank system has been designed to prevent the circuit pressure rising above the design pressure during this condition. Core damage could occur due to thermal shock and chemical interaction between the water/steam mixture and the hot core, and the main circulators must be capable of handling a steam/helium mixture to ensure the quick reduction of core temperature below the minimum value for the steam/graphite reaction so limiting core damage. The magnitude of fission product release from the fuel would depend on the degree of core damage. A district hazard would not occur because of the containment of any fission products released within the primary circuit and dump tanks. Any leakage from this circuit would be adequately controlled by the containment system.

(ii) Depressurisation

Failure of the primary circuit resulting in depressurisation would produce excess temperature conditions in the reactor, the severity of which would depend on the degree of forced convection cooling available. Calculations, assuming cooling by pure helium, at 15 psia have shown that the coolant temperature at outlet from the core would be as follows:-

3 circulator pony motors operating	1160°C
2 " " " "	1290°C
1 " " motor "	2000°C
Natural convection	2000°C in 100 minutes continuing to rise until limited by radiant heat loss and convection at very high temperatures.

Thus with natural convection cooling, core melting would occur with pure helium coolant but, with a depressurisation rate fast enough to cause failure of the forced circulation equipment, the breach would be so large that the vessel would contain a mixture of nitrogen (from the containment) and helium within 15 minutes. Natural convection cooling by pure nitrogen at 15 psia would limit the coolant temperature at core outlet to 1840°C. A mixture of 80% nitrogen 20% helium at the containment equilibrium pressure (30 psia) would limit the maximum temperature of the coolant to below 1840°C due to the higher coolant density (pressure effect) and improved heat transfer properties of the mixture (due to the pressure of helium).

The magnitude of the hazard condition associated with depressurisation will depend on the quantity of fission products released from the system, i.e. on the degree of cooling available following depressurisation. The forced circulation system is designed to give a high probability that some forced cooling (either main motor or pony motor drive) would be available after depressurisation but the design of the containment system has been based on a core meltdown, i.e. release of all fission products from the core. The fission products contained in the fuel purge chemical would not be released providing sufficient water cooling is always available. Alternative independent cooling systems would be provided with facility for operation from a point remote from the reactor.

The containment system, described in Section 12 of this report, will limit release of all activity to below the permitted levels providing a combined plate-out and decontamination factor of 100,000 is provided in the inner containment and primary circuit. If a plate-out factor of 90% is assumed for solid isotopes which are not volatile at low temperatures (below 200°C) this would require a chemical clean-up plant with an ultimate decontamination factor of 10,000. With a recirculation flow of 100,000 cfm through a clean-up plant, with D.F. of 100 from the inner containment and leakage rates of 0.1% and 1% per day from the inner and outer containments respectively, calculations have shown that release of all isotopes would be kept below the maximum permissible levels.

The delay provided by the double containment is sufficient to reduce the release of short level isotopes (e.g. I-131) to low levels compared with the long lived Sr-90. Hence, for this system Sr-90 would be the critical isotope. A calculated total release of 11.6 curies would occur for the period of one year during which the release rate is above the maximum permissible continuous level. The permitted emergency release would be 8 curies at ground level and the integrated permitted continuous release, 9 curies, for the site specified. Hence the containment system, as specified, would appear to be quite adequate especially in the light of the pessimistic assumptions used in the calculations.

The nitrogen atmosphere in the inner containment would eliminate any risk of a water gas explosion following a combined failure of the steam circuit and depressurisation. The reactor shielding (minimum thickness 2' 6") would act as an effective blast shield for the inner containment vessel during depressurisation. The shielding provided by the outer containment (2' 6" concrete) would reduce direct radiation levels outside the containment to well below the specified safe levels.

Reactor Normal Conditions

153. The ventilation plant in the inner containment has been designed to keep concentrations of all isotopes (I-131 being the critical isotope) to below breathing tolerance during normal operation. Radiation and contamination levels in the inner containment due to leakage of fission products, would be kept to low levels allowing personnel access at all times, self-air being required during operation at power. Any of the accidents at pressure, including a disruption of the fuel purge systems, e.g. by an "unseated" fuel element cluster, would lead to increase in fission product concentrations in the inner containment and thus would necessitate control of access.

SECTION 14: COST ANALYSIS

154. A fully detailed estimate has been prepared and is summarised below.

TABLE 4

Item	HTR	
	£ 10^3	£/KW
Containment	957	2.31
Reactor Building and Civil Work	2,500	6.05
Pressure Vessel	496	1.2
Internals	183	.44
Graphite (Side reflector) and core restraint	150	.36
Circulators	800	1.94
Gas Ducts and valves	176	.43
Heat Exchangers	3,000	7.25
Refuelling equipment	350	.85
Fuel element equipment	150	.36
Control rods and mechanisms	880	2.13
Instrumentation	900	2.18
Faulty fuel detection	100	.24
Electrical	2,900	7.09
Exhaust chemical plant	126	.31
Purge chemical plant	2,305	5.59
Ventilation	300	.73
Reactor Crane	80	.19
Turbo-Alternators and auxiliaries	5,080	12.30
Turbine Hall	1,100	2.66
River Works	1,400	3.38
Unspecified items	1,350	3.27
Dump tanks	307	.74
Heat sinks and safety valves	112	.27
	25,702	62.08
Consultants and Services / Design and Engineering Contingencies		13.5
Tender Price		75.58

155. The "Containment" item covers the inner steel containment vessel only. The relatively high cost of the complete containment system is reflected in the "Reactor Building and Civil Work." Economic development of this type of reactor would demand attention to this item and the associated safety aspects. If a single unpressurised "confinement" building of minimum dimensions could be shown to be acceptable, savings of up to £6/KW could result.

156. A further major item which militates against the realisation of a low capital cost, is the purge system. The satisfactory development of fission product retaining fuel, and the consequent elimination of the purge system could lead to savings roughly assessed at £10/KW. This figure would include, apart from the cost of the purge plant itself, the costs of building, shielding and containment necessary to accommodate the plant, associated instrumentation and electrical plant, and other items such as pipework inside the reactor. The design of the fuel element would be freed from an important restriction and this might lead to savings in the fuel handling equipment as well as in fuel costs.

SECTION 15: PRELIMINARY APPRECIATION OF CONSTRUCTION PROGRAMME

157. This design study considers only the construction aspects of the programme. (Fig. 40).

158. The programme as set out is based on the following assumptions:-

(a) that the Biological Shield (i.e. all concrete work within the steel containment) can be completed, apart from positioning the precast units which seal the shielding, before the parts of the Pressure Vessel, Heat Exchangers and Dump Tanks are lowered into position;

(b) that the Pressure Vessel, Heat Exchangers and Dump Tanks will be prefabricated on site into sections which can be lowered into position by the use of a Goliath type crane;

(c) that the Heat Exchangers and Dump Tanks will be prefabricated on site into complete shells and then stress relieved and tested prior to being lifted into position;

(d) that the Pressure Vessel will be pneumatically tested;

(e) that it will not be necessary to pressure test the steel containment prior to constructing the concrete outer containment, that is, that it will be possible to construct these two items simultaneously and test the welds in the steel containment by the use of vacuum boxes;

(f) that the Reactor internals will be relatively simple so reducing both the time needed for their installation and that necessary for the fabrication of the top dome of the Pressure Vessel. (The number of charge and discharge channels is reduced.)

159. Should it be found necessary to test hydraulically the Pressure Vessel then a further month would be added to the programme and should it be found necessary to pressure test the steel containment prior to constructing the concrete containment then a further seven months would be added to the programme.

160. The programme as set out is based on the following construction sequence:-

(i) Foundations.

(ii) Bottom bowl of steel containment.

(iii) Biological Shield (all concrete work).

(iv) Sides of steel containment up to second floor level.

(v) Outer concrete containment building up to second floor level.

<u>Note</u>: Items (iii), (iv) and (v) could proceed simultaneously but with a time-lag between each item so that in effect each would proceed a stage ahead of the other. Item (iii) would be completed apart from the precast removable sections closing the containment of the Pressure Vessel, Heat Exchangers and Dump Tanks.

(vi) Pressure Vessel.

(vii) Heat Exchangers.

(viii) Dump Tanks.

<u>Note</u>: Items (vi), (vii) and (viii) would proceed simultaneously.

(ix) Complete items (iv) and (v).

(x) Pressure Vessel internals, mechanical and electrical installations and ancillary plant.

<u>Note</u>: It would be possible to make a start on some of the mechanical and electrical installations prior to the completion of Item (ix).

161. The programme does not show the turbo-alternators and other ancillary plant, as, undoubtedly, the erection of the necessary buildings and the installation of the plant can be easily phased within the period required for the construction of the Reactor.

2

SECTION 16: PARAMETRIC SURVEYS

162. Studies have been made of the effects on cost per u.s.o., of variation of a number of the design parameters on each side of the reference point. This survey has been conducted by analysing the cost of the individual items given in the preceding section as functions of the variables on which they depend. For example the pressure vessel cost of £679,000, which also includes the cost of such internal items as the diagrid and hot gas manifold, is broken down as

$$D_v\ (2L_c + D_v)\ (71.9t + 5.53t^2) + 856\ W_c$$

Where D_v is vessel diam. (ft), L_c is core length (ft), t is vessel thickness (in), and W_c is core weight (tonnes). The t term takes account of the volume of metal in the vessel, the t^2 term the welding cost, and the W_c term the cost of internals and supports. The ground rules used in costing are given in the Table below.

The prices given in the Table are tentative estimates compiled for the specific purpose of this study and should not be taken out of context. The price for U-235, for example postulates a possible fall from the current level but this is purely an assumption.

TABLE 5

Costing Ground Rules

Interest rate	5.5%
Load factor	0.75
Reactor life	20 years
Initial costs: 235-U	4,387 £/kg
232-Th	15 £/kg
Graphite	5.8 £/kg
Fabrication	55 £/kg of heavy atom
Reprocessing	56 £/kg of heavy atom
Credit: 235-U, 233-U	4,387 £/kg
232-Th	15 £/kg
Fuel Cycle	3 batch with U recycle

163. Variations have been made in the eleven parameters as listed in the table below without any marked improvement in costs being obtained. The importance of the pumping power component in the capital cost figure is shown by the reductions obtained with higher voidage and lower L/D ratio; but these reductions, which are the largest obtained, are not sufficient to indicate that substantial reductions in cost could be achieved by further investigation in these directions.

TABLE 6

The Effect of Variation of Parameters about Reference Design

Altered parameter and new value (standard case and units in brackets)		Capital cost excluding core graphite (d/uso)	Fuel cost no graphite (d/uso)	Graphite cost (d/uso)	Unit cost (d/uso)
Standard case		.285	.111	.023	.487
Power Density (11.34 kW/litre)	14.34 8.34	.285 .288	.113 .110	.026 .021	.492 .486
Carbon/Uranium ratio (2358/1)	3065 2004	.285 .285	.115 .110	.030 .023	.498 .488
Thorium/Uranium ratio (8.2/1)	6.2 10.2	.285 .285	.121 .117	.022 .040	.496 .510
Core L/D ratio (0.606)	.406 .806	.283 .287	.111 .112	.024 .023	.484 .491
Coolant Voidage (0.2075)	.2875 .1475	.281 .290	.109 .112	.023 .024	.481 .494
Radial Reflector Thickness (80 cm)	50 110	.282 .290	.111 .112	.024 .024	.486 .493
Coolant Inlet Temp. (350°C)	300 400	.289 .291	.109 .112	.024 .024	.480 .494
Coolant Outlet Temp. (750°C)	850 650	.282 .297	.110 .112	.024 .024	.484 .501
Heat Exchanger Press. Drop (0.762 psi)	0.262 1.262	.291 .285	.110 .111	.024 .024	.493 .488
T.S.V. Pressure (2300 psia)	1500 3500	.284 .289	.114 .107	.024 .024	.490 .488
Feed Temp. (485°F)	540 430	.289 .283	.111 .111	.024 .024	.492 .486

164. Perhaps the most surprising result is the small effect of power density on capital cost. The variations shown in power density are brought about by change in rating. The rating at the 14.34 KW/litre power density is 1.26 MW/Kg U-235. At this level the build up of protactinium shortens the fuel life and increases the fuel cost. Increased subdivision of the fuel also increases fabrication and component costs. The expected saving in capital costs does not materialise because the increased pumping required just compensates for the decrease in vessel and civil costs.

165. An investigation into the pressure vessel costs in going from 11.34 to 14.34 kW/litre shows a reduction in weight of 10.5% and a reduction in vessel cost of 13% (which makes the correct allowance for welding and lower core weight). When expressed as a percentage of the total capital cost, the 13% is reduced to 0.3%. A similar reduction is obtained from the change in the civil costs where the change in vessel size does not give a proportionately large change in building size because of the effect of the other plant which is housed within the building and which is not reduced with the change to a higher power density. The change in net electrical output due to the higher pumping power requirement is about 0.5% and this taken with an increase in blower costs of about 0.3% nullifies the cost reductions due to vessel, civil work, reflector, etc.

166. The variation in carbon/uranium ratio shown in the table involves also a change in rating because the power density was taken to be constant. This treatment has the merit that capital costs are unaffected and interpretation of the results is therefore simplified. At the higher moderator ratio, the fuel life is shortened due to the higher rating and additional graphite is required. These effects lead to increased fuel costs. At the lower moderator ratio, the decrease in fuel cost due to lower rating is just offset by the effect on the initial reactivity. With a further decrease in moderator ratio, the latter effect would predominate and the fuel cost would increase.

167. Comparison of two cases with about the same rating (i.e. the 14.34 kW/litre and the 3065 C/U cases) shows a slightly higher fuel cost for the more highly moderated reactor. The reason for this is that, due to the softer neutron spectrum, the conversion factor is reduced with a consequent higher fuel make-up cost.

168. The variation of thorium/uranium ratio shows a fairly sharp optimum about the value chosen for the reference design. Again the balance is between initial reactivity and conversion factor.

169. The conclusion drawn from this survey is that no large reductions in cost are likely to be achieved by change from the conditions chosen for the reference design. It is of particular interest that a power density optimum occurs at about 9.5 kW/litre and that a 20% variation about this point produces very little change in costs.

SECTION 17: DISCUSSION

Power Density Limitations

170. The maximum power density achievable with this system is limited primarily by thermal stresses in the graphite fuel sleeves, in conjunction with a minimum thickness of these sleeves required for robustness in manufacture and handling. This statement is made on the understanding that a nominal maximum fuel centre temperature of around 1850°C is acceptable. Greater subdivision of the fuel permits a higher power density within given limits of sleeve stress and centre temperature, but levitation of the core then becomes a problem.

171. The design put forward makes the best compromise on the basis of the assumed graphite properties. Economic optimisation shows however that the power density required is below that considered feasible (not more than 10 KW/litre, compared with about 15 KW/litre) and the degree of fuel subdivision found necessary is then determined by thermal stresses alone.

172. Not enough is known about the behaviour of graphite under temperature gradients, particularly with irradiation, for the problem to be treated in an absolute sense. The data adopted for this design, taken together, are believed to be fairly optimistic. The effect of assuming an improvement of 66% (e.g. increasing permissible stress from 1500 psi to 2500 psi) is to reduce unit costs by 0.1d/u.s.o. The number of (larger) fuel elements required is reduced in similar ratio to the stress increase, and the saving comes from lower manufacturing costs. At the same time, the centre temperature will rise 250-300°C, unless the improvement has arisen from better conductivity.

Fuel Boxes

173. It was not found possible to utilise a sealed fuel box due to the pressure stresses which arise during depressurisation of the reactor unless this is carried out slowly. Considering possible variations of graphite permeability, free internal volume, permissible depressurisation rate, and fuel box end designs, led to the conclusion that the feasible fission product delay in the free spaces of the box would be only an hour or so. Since there is a diffusion delay within the fuel body itself, this extra delay is not of sufficient importance to warrant the tight specification, and risk of failure for the boxes.

174. It was therefore preferred to leave the boxes open ended, and arrange the purge flow to make two passes, outside and inside the box. This reduces the primary circuit activity due to back diffusion of volatiles out of the fuel sleeves, to a very marked extent. Even if it is assumed that this activity remains gas borne, and is therefore eventually collected in the traps, the circuit activity with a single-pass purge is estimated to be of the order of 20 kilocuries, and the two-pass system reduces this to under 1 curie.

Chemical Plant

175. There is considerable uncertainty in selecting a suitable purge flow rate, and as mentioned above, the very existence of a scavenge mechanism for some of the long-life metal fission products is itself an assumption. The

only approach to this problem available for the design study, was to adopt a flow within the range likely to be realistically demonstrated in Dragon. It was considered reasonable to reduce the flow per KW by a factor of about 3.5 which results in a similar flow per element as in Dragon, the elements being 1.75 times as long and operating at twice the power per unit length. If this flow were halved, a saving in the cost of the delay beds of about £1/KW would result, but some extra cost would be involved in the plant for the removal of chemical impurities probably by making this a separate circuit.

176. The heat loads on the chemical plant due to fission products have been assessed at near the lowest figure which will be possible for design purposes since of the total of 750 KW, 670 KW arise from noble gases whose diffusion rates from "normal" fuels are fairly well known. This results from the use of internal traps in the fuel elements, for metals and halogens, and the assumption that sufficient temperature control can be achieved to give high decontamination efficiencies (100% on metals and 92% on halogen activity).

Active Handling

177. It is doubtful whether all released fission products can be scavenged by the purge system so that they enter the primary circuit only by means of gaseous back-diffusion through the fuel sleeve. Other mechanisms certainly arise with some of the long-lived metals. Thus, even with the two-pass purge system deposition of activity in say, the cooler parts of the circuit could cause a serious handling problem during maintenance operations. Likely solutions to several aspects can be visualised, e.g. circulators (flask), and heat exchangers (in situ chemical decontamination), but lacking quantitative data on the activity to be expected some uncertainty must remain. The lay-out and general concept of the present Reference Design is based on the assumption that activity of the order of 10 Ke is distributed over the cooler parts of the circuit.

Retentive and Fully Releasing Fuels

178. The most direct method of reducing the cost of fission product trapping plant is to use retentive fuel. Encouraging results have been reported with various types including those based on the pyrolytic carbon coating of carbide fuel particles. The retention of mobile metals is less developed than the retention of volatiles, but there is similar uncertainty about the behaviour of metals in purged systems. With the purge eliminated, the fuel element design would be freed from a fundamental limitation and quite a different arrangement of core could result.

179. An argument against retentive fuels is that reactivity poisons are retained in the reactor core. The purged fuel as at present conceived, does obtain some advantage from the release of Xe for example, but the gain is a small fraction of the maximum possible which might save .03d/u.s.o. To achieve this maximum, however, high diffusion rates are required for volatiles, and metals such as strontium and caesium must also be extracted. Since the fuel atoms themselves cannot be allowed to become mobile to any significant extent, it is considered that this development is much more uncertain, and of longer term, than that of retentive fuels.

Optimisation

180. The conclusion that the optimum power density is below 10 KW/litre, and that the unit cost is only .005d/u.s.o. higher at about 7 KW/litre could

be valid for systems other than this Reference Design. The effect of variations in cost assumptions has been examined and the optimum power density is not markedly altered. The trend of fuel costs is overriding and gives a minimum fuel cost as 9.3 KW/litre. The effect of capital cost is to raise the optimum, but only a small amount.

181. The situation would be quite different with a different fuel cycle, or in say a transportable reactor where the balance of costs could be markedly changed. It is evident, however, that for large power stations operating on the Th/U-233 cycle, with graphite moderated homogeneous cores, there is no great incentive to strive for high power densities, and that reductions in unit cost depend more on simplification of the fuel element leading to a reduction in fabrication and component cost, and to developments leading to capital cost savings by the elimination or simplification of plant.

SECTION 18: GENERAL CONCLUSIONS

182. The capital cost to be expected for a large power reactor based as closely as possible on the Dragon design is not sufficiently low for the system to offer a great attraction.

183. Further study of safety requirements, leading to a revised containment design could lead to a saving of up to £6/KW. The considerations involved are likely to be largely applicable to any High Temperature Reactor using carbon based fuel elements.

184. The fission product purge system, associated building work and equipment is roughly assessed to cost £10/KW. The possibility of developing a satisfactory fission product retaining fuel is therefore of importance.

185. With a U/Th/U fuel cycle, there will be an optimum power density. Whilst the optimum value depends on the balance of costs in a particular concept, no justification is seen at present for striving for a design capable of power densities above 10 KW/litre; nor should a simple and cheap form of fuel element be rejected on account of limited power density, at least above say 5 KW/litre.

186. In applying the Dragon concept to a power reactor problems would arise which will not necessarily be solved by the successful development of Dragon. These include particularly,

(a) Assessment of the leakage of steam and water to be expected from steam generators, and of any special steps necessary in consequence.

(b) Control elements and mechanisms suitable for the operation of control members within the core proper.

(c) Design and manufacture of fuel elements, particularly the graphite components, suitable for a core of the required length.

(d) Active handling of large components.

187. There is no reason to expect any of these to prove insuperable and some possible solutions have been incorporated in the Reference Design, but development work would be required.

REFERENCES

[1] J. B. Slater — Calculations on various methods of control of an H.T.R. Appendix No.7.

[2] J. B. Slater — Variation of Parameters for a 1000 MW(H) High Temperature Reactor - Part I Appendix No.2.

[3] J. B. Slater — Variation of Parameters for a 1000 MW(H) High Temperature Reactor - Part II Appendix No.3.

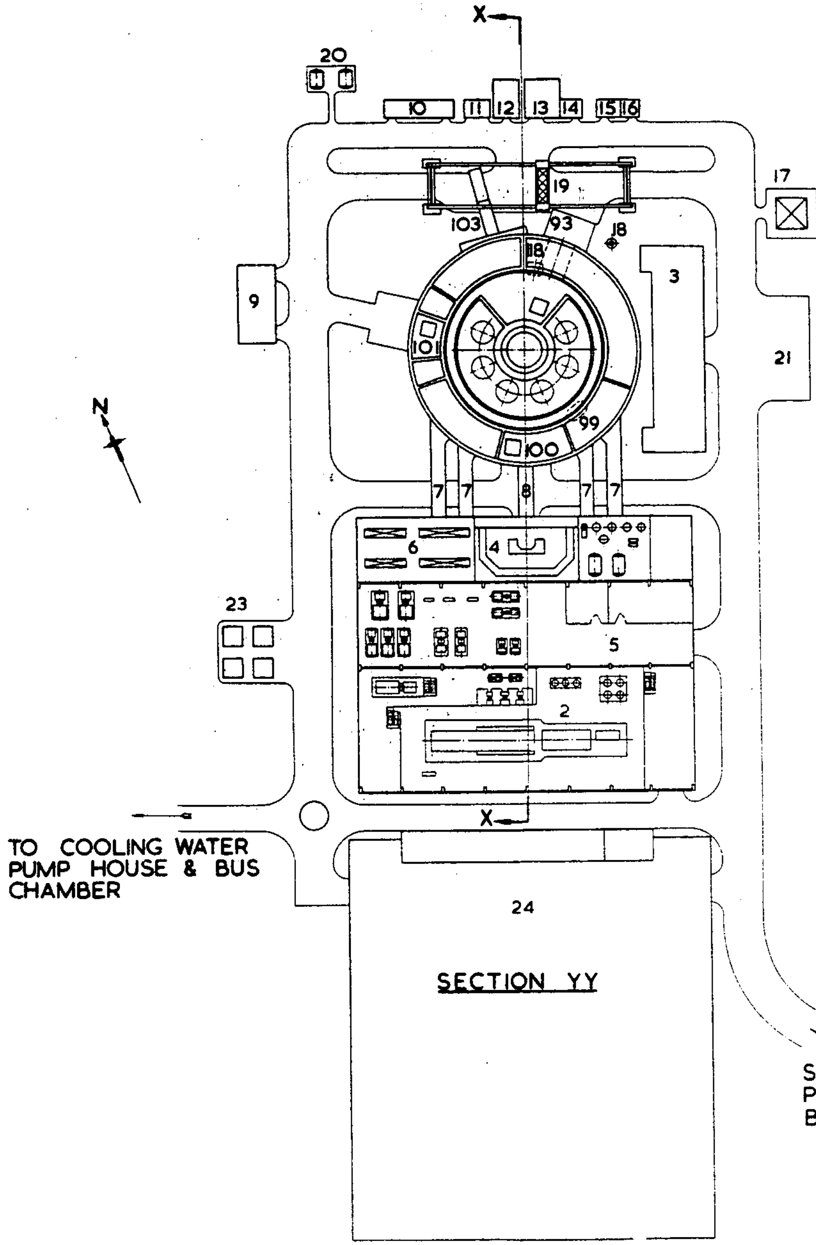
[4] J. B. Slater — Effect of uncertainty in Protactinium 233 Cross-Section on Core Performance of a 1000 MW(H) High Temperature Reactor Appendix No.9.

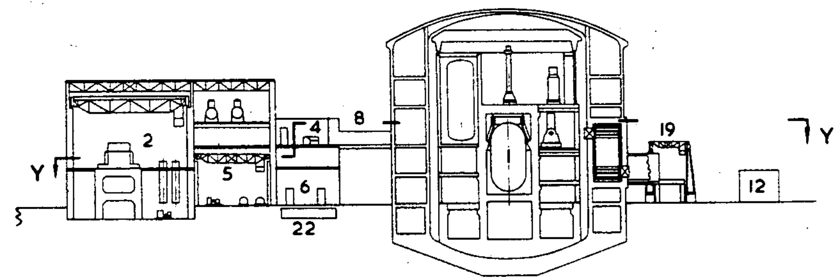
[5] R. F. Burstall, J. B. Slater and D. E. J. Thornton — The Physics of the H.T.R. Reference Design Appendix No.8.

[6] A. F. Pexton and T. J. Staunton — Assessment of Thermal Performance Parameters for a 1000 MW(H) H.T.R. Appendix No.4.

[7] Babcock & Wilcox Ltd. — High Temperature Gas-Cooled Reactor Heat Exchanger Study Appendix No.6.

[8] H. Hatton — Stresses in Graphite Fuel Boxes Appendix No.5.





SECTION XX

LEGEND

1	REACTOR	13	CLEAN WORKSHOP
2	TURBINE HALL	14	STORES
3	ADMIN BLOCK	15	LAUNDRY
4	CONTROL ROOM	16	BOILER HOUSE
5	MECHANICAL ANNEX	17	MECHANICAL COOLING TOWER
6	ELECTRICAL ANNEX	18	STACK
7	PIPE BRIDGES	19	75 TON SERVICE CRANE
8	ACCESS BRIDGE [OUTER CONT]	20	DIESEL FUEL TANKS
9	ACTIVE WORKSHOP	21	CAR PARK
10	LABORATORIES	22	TREATED WATER STORAGE
11	FIRE STATION	23	EMERGENCY FEED WATER TANKS
12	FUEL ELEMENT STORE	24	275 KV COMPOUND

93	GOODS AIRLOCK
99	EMERGENCY AIRLOCK
100	LIFTING WELL CLEAN EQUIPMENT
101	LIFTING WELL DIRTY EQUIPMENT
103	FUEL ELEMENT DISCHARGE LOCK
118	PERSONNEL AIRLOCK

FIG. I. SITE LAYOUT OF 1000 $MW_{(H)}$ H.T.R.

(ASSUMING OLDBURY-ON-SEVERN SITE)

SCALE IN FEET 50 25 0 50 100 150 200

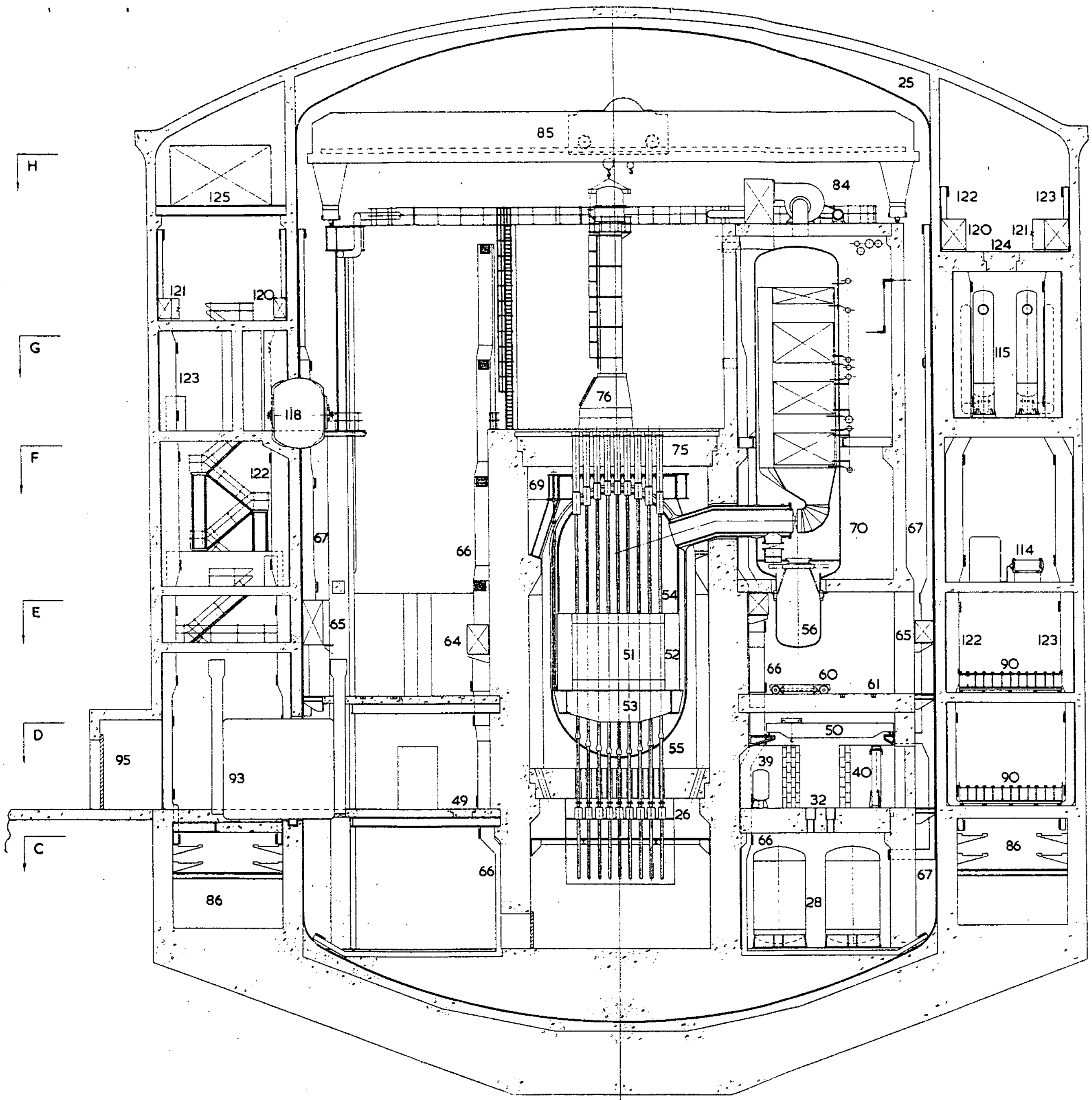


FIG 2

GENERAL ARRANGEMENT
SECTION A-A

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION B-B SEE FIG 3
FOR SECTION C-C SEE FIG 4
FOR SECTION D-D SEE FIG 5
FOR SECTION E-E SEE FIG 6
FOR SECTION F-F SEE FIG 7
FOR SECTION G-G SEE FIG 8
FOR SECTION H-H SEE FIG 9

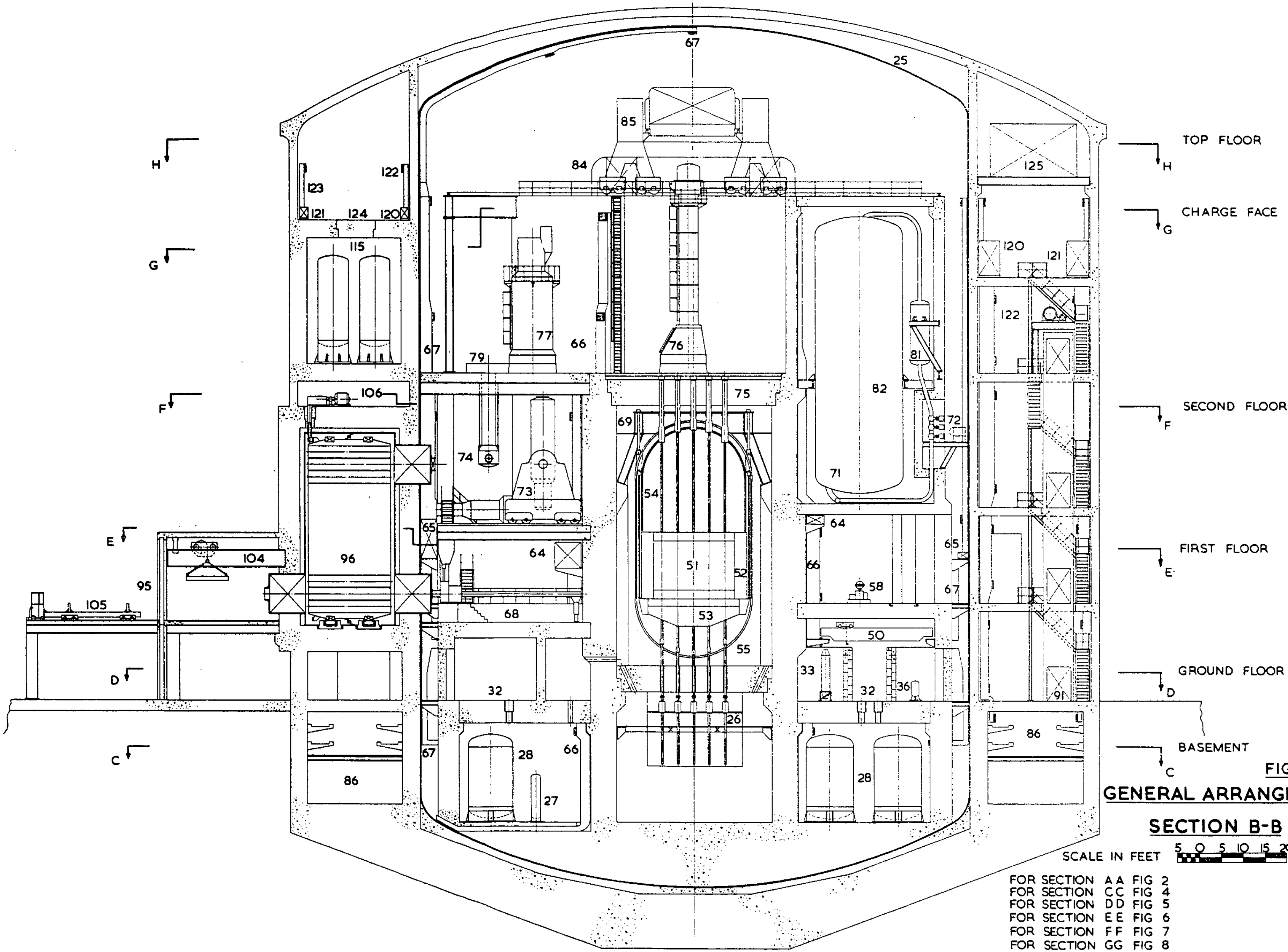


FIG. 3

GENERAL ARRANGEMENT

SECTION B-B

FOR SECTION AA FIG 2
FOR SECTION CC FIG 4
FOR SECTION DD FIG 5
FOR SECTION EE FIG 6
FOR SECTION FF FIG 7
FOR SECTION GG FIG 8
FOR SECTION HH FIG 9

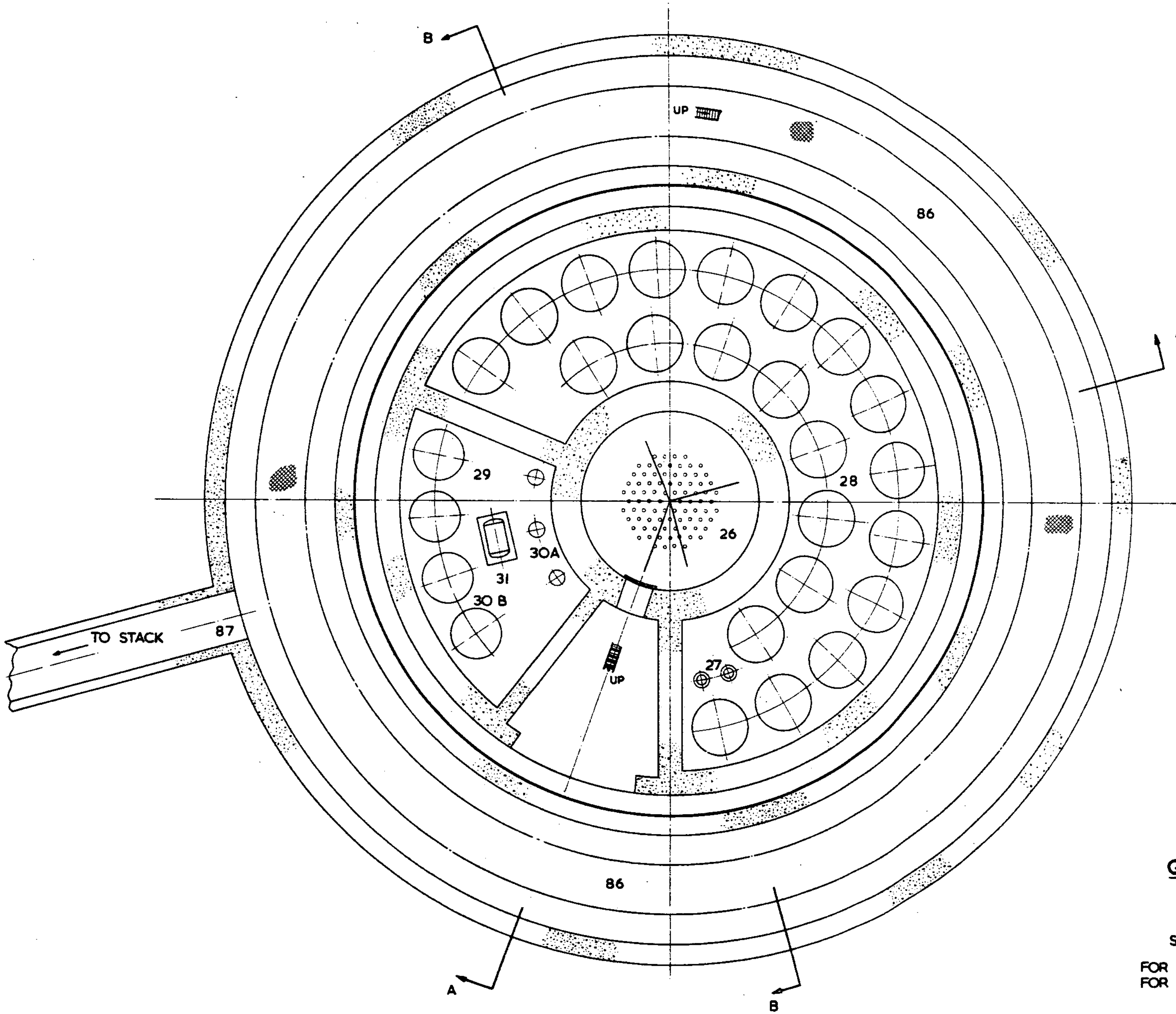


FIG. 4

GENERAL ARRANGEMENT

SECTION C-C

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION A-A SEE FIG. 2
FOR SECTION B-B SEE FIG. 3

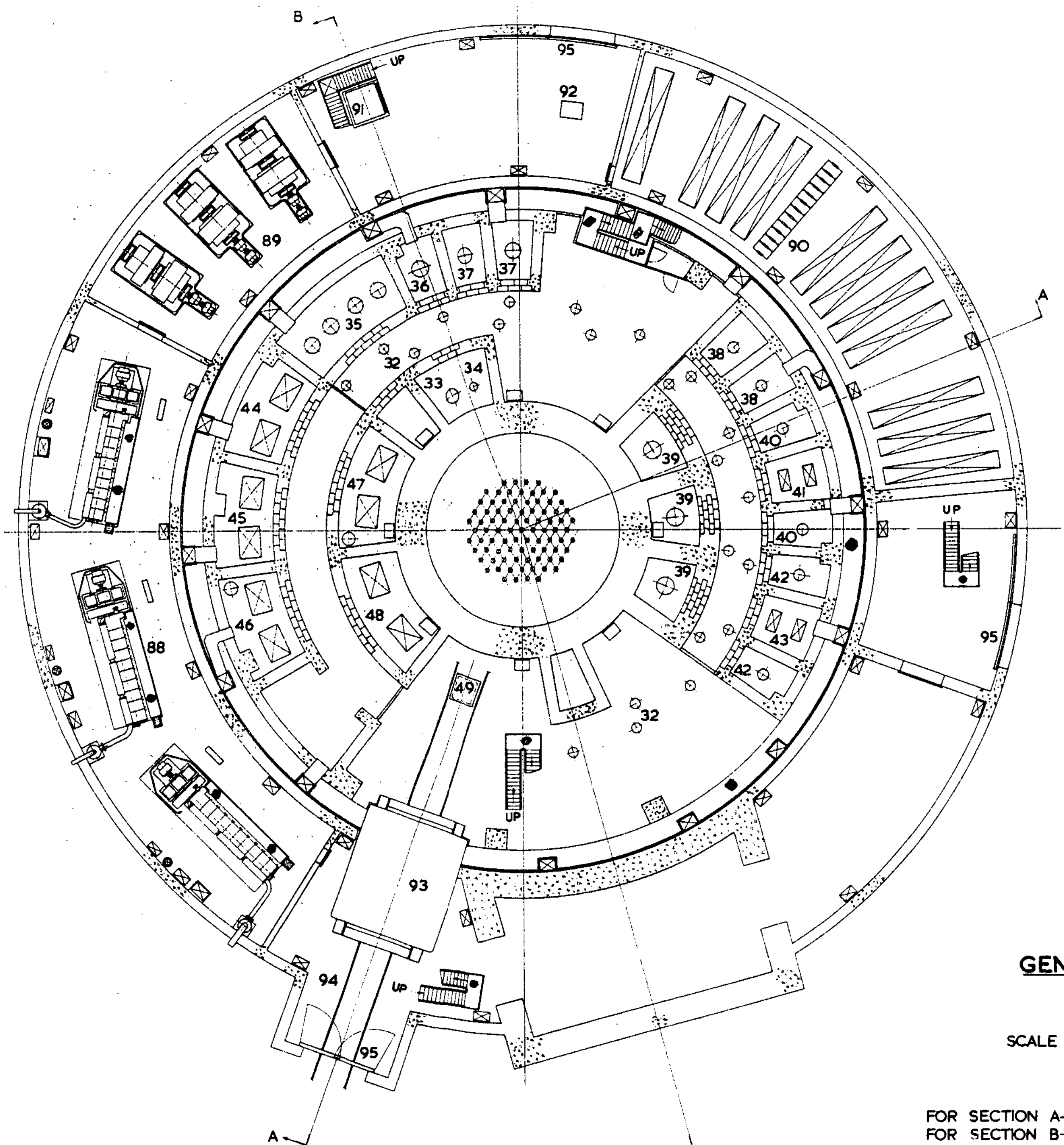


FIG. 5

GENERAL ARRANGEMENT

SECTION D-D

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION A-A SEE FIG. 2
FOR SECTION B-B SEE FIG 3

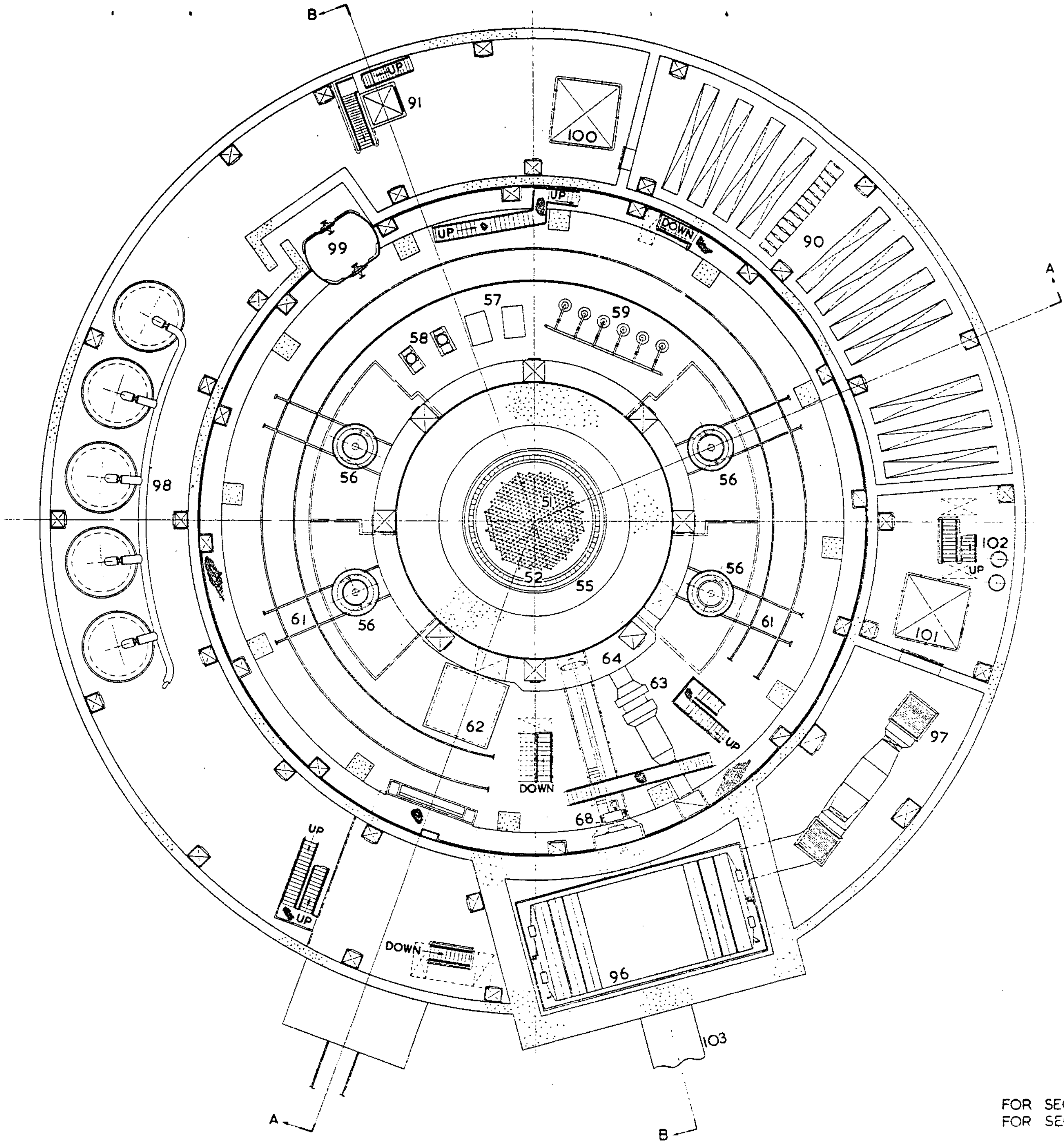


FIG. 6

GENERAL ARRANGEMENT

SECTION E-E

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION AA SEE FIG 2
FOR SECTION BB SEE FIG 3

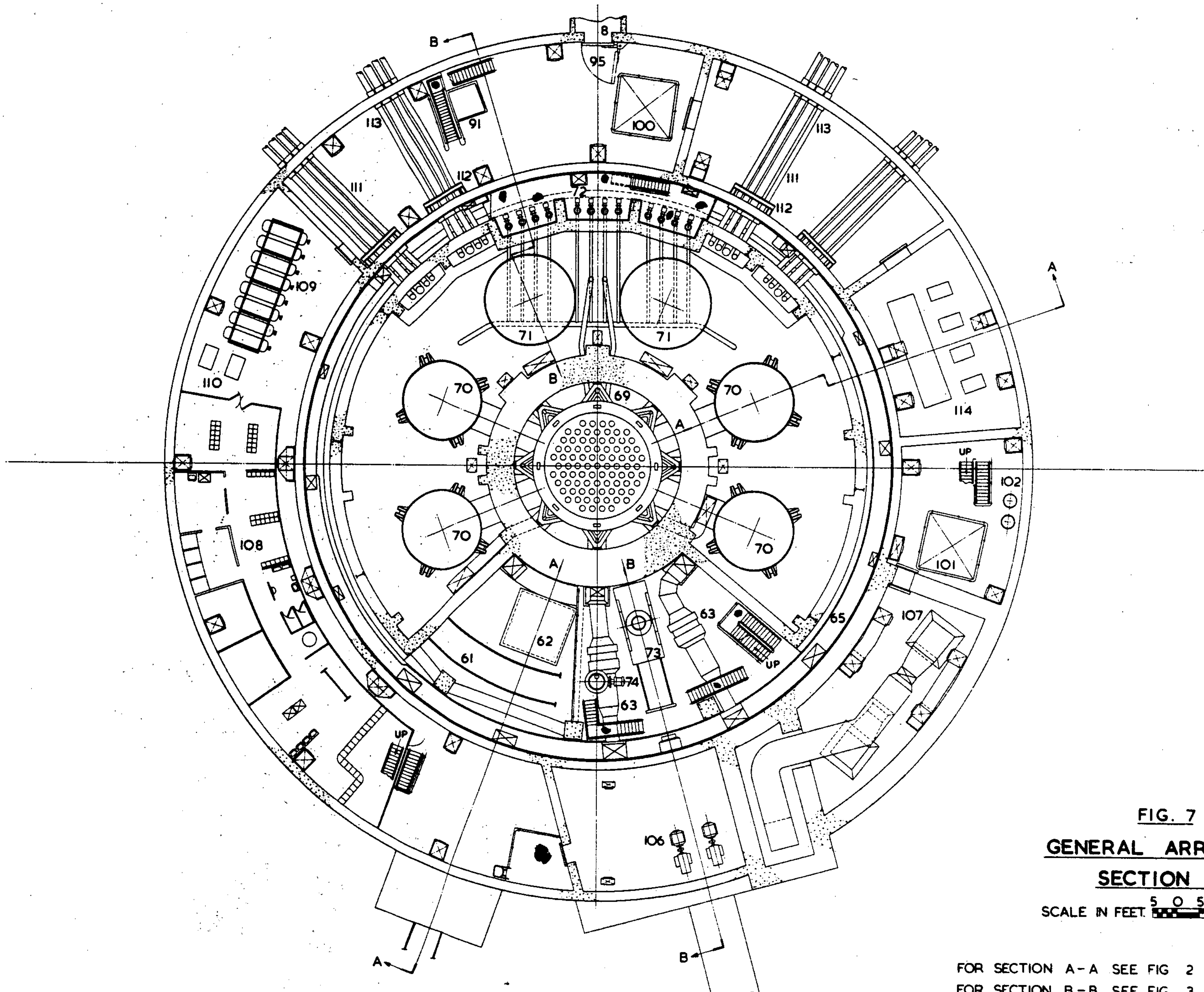


FIG. 7

GENERAL ARRANGEMENT.

SECTION F-F

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION A-A SEE FIG 2

FOR SECTION B-B SEE FIG 3

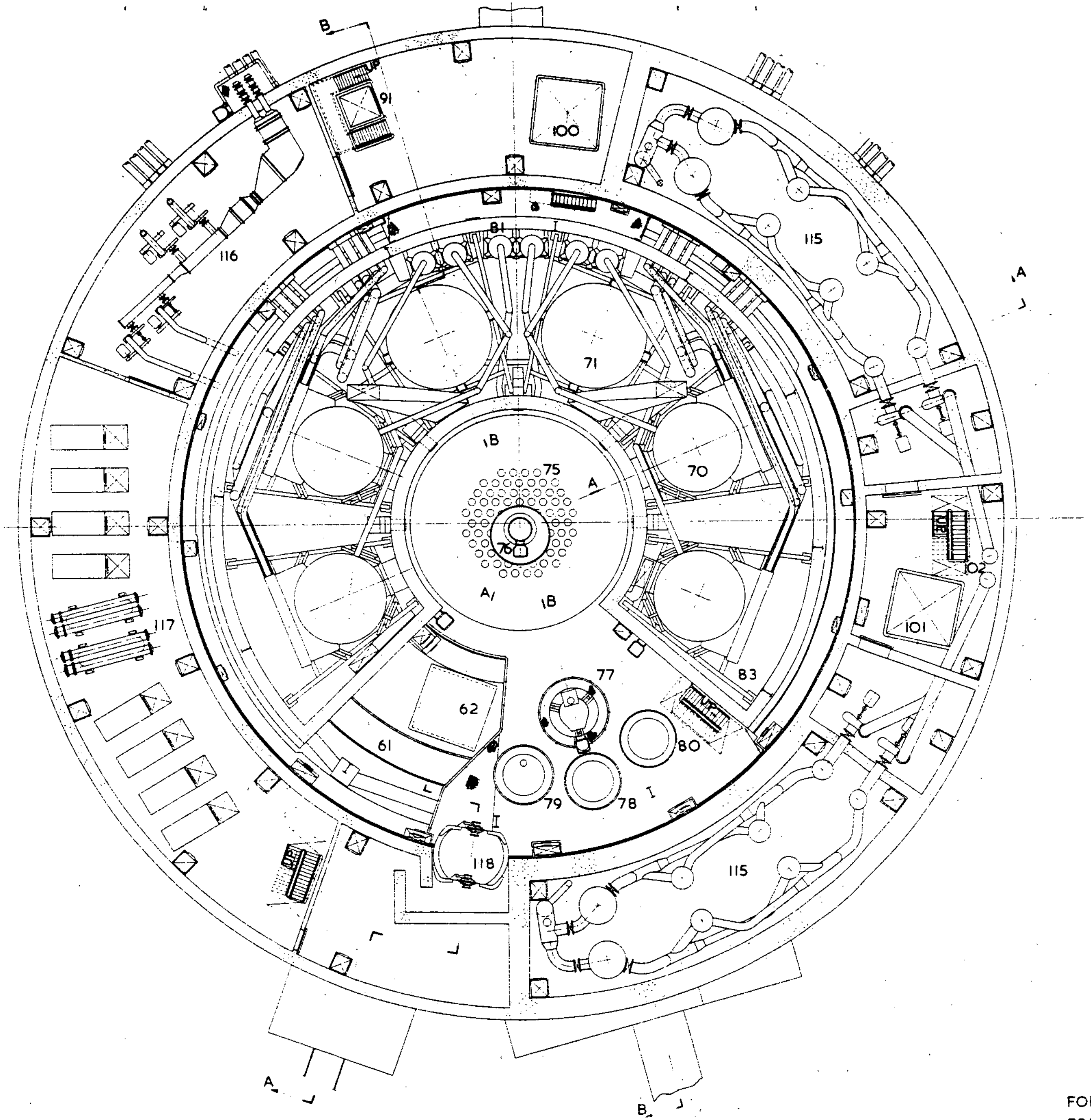


FIG. 8

GENERAL ARRANGEMENT
SECTION G—G

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION A-A SEE FIG 2
FOR SECTION B-B SEE FIG 3

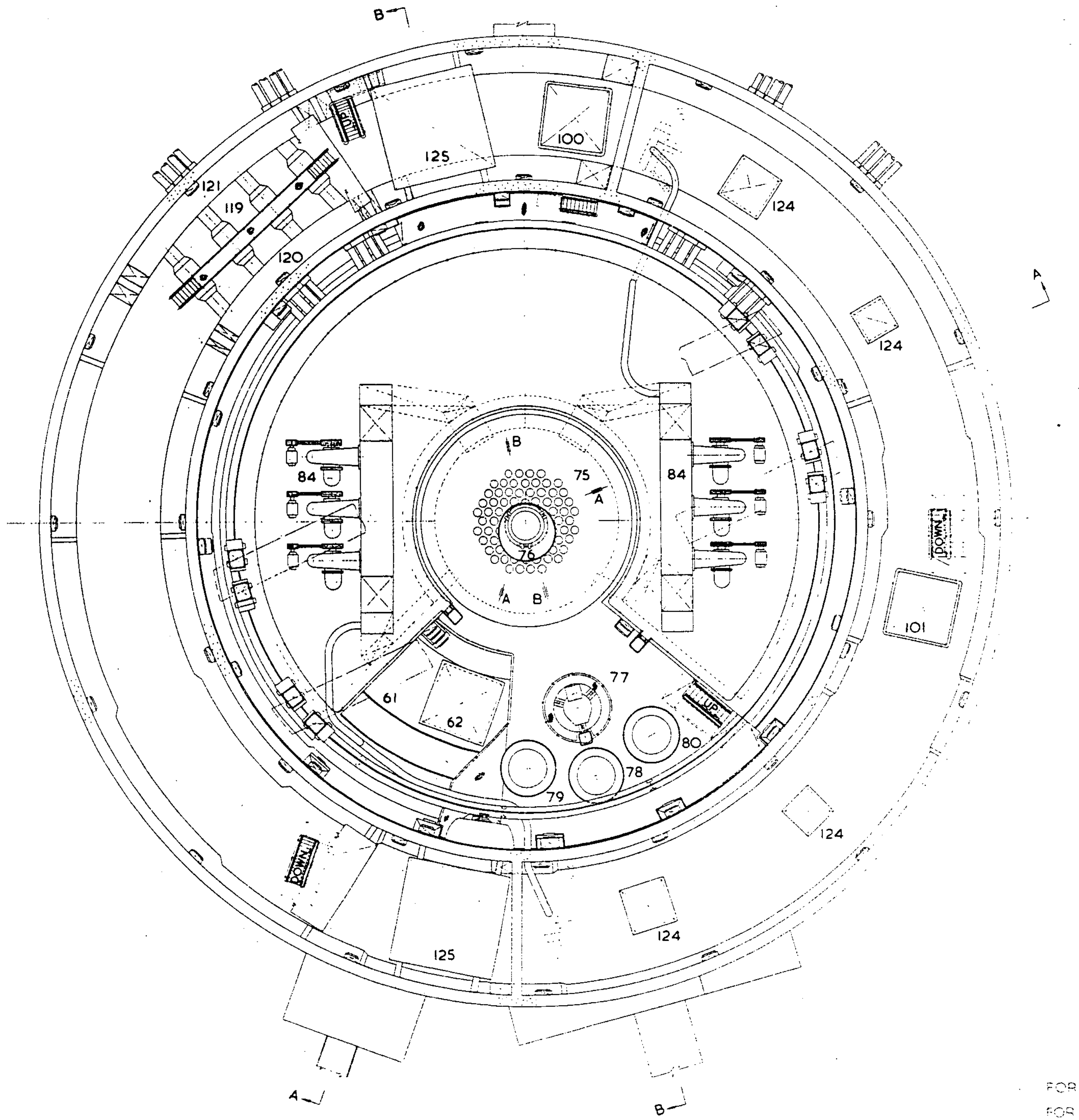


FIG 9

GENERAL ARRANGEMENT

SECTION H – H

SCALE IN FEET 5 0 5 10 15 20

FOR SECTION A – A SEE FIG 2

FOR SECTION B – B SEE FIG 3

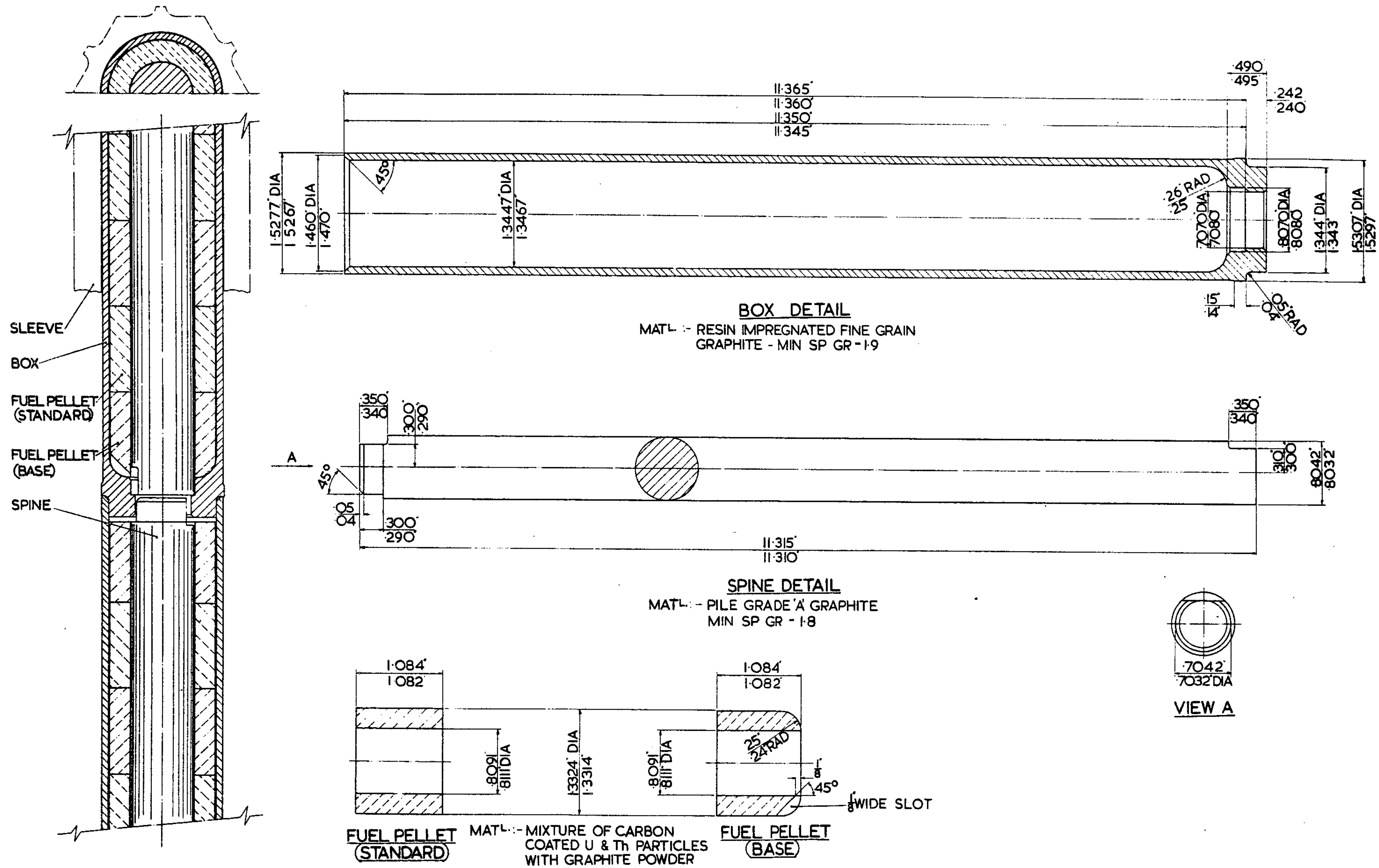


FIG. 10 FUEL BOX ASSEMBLY

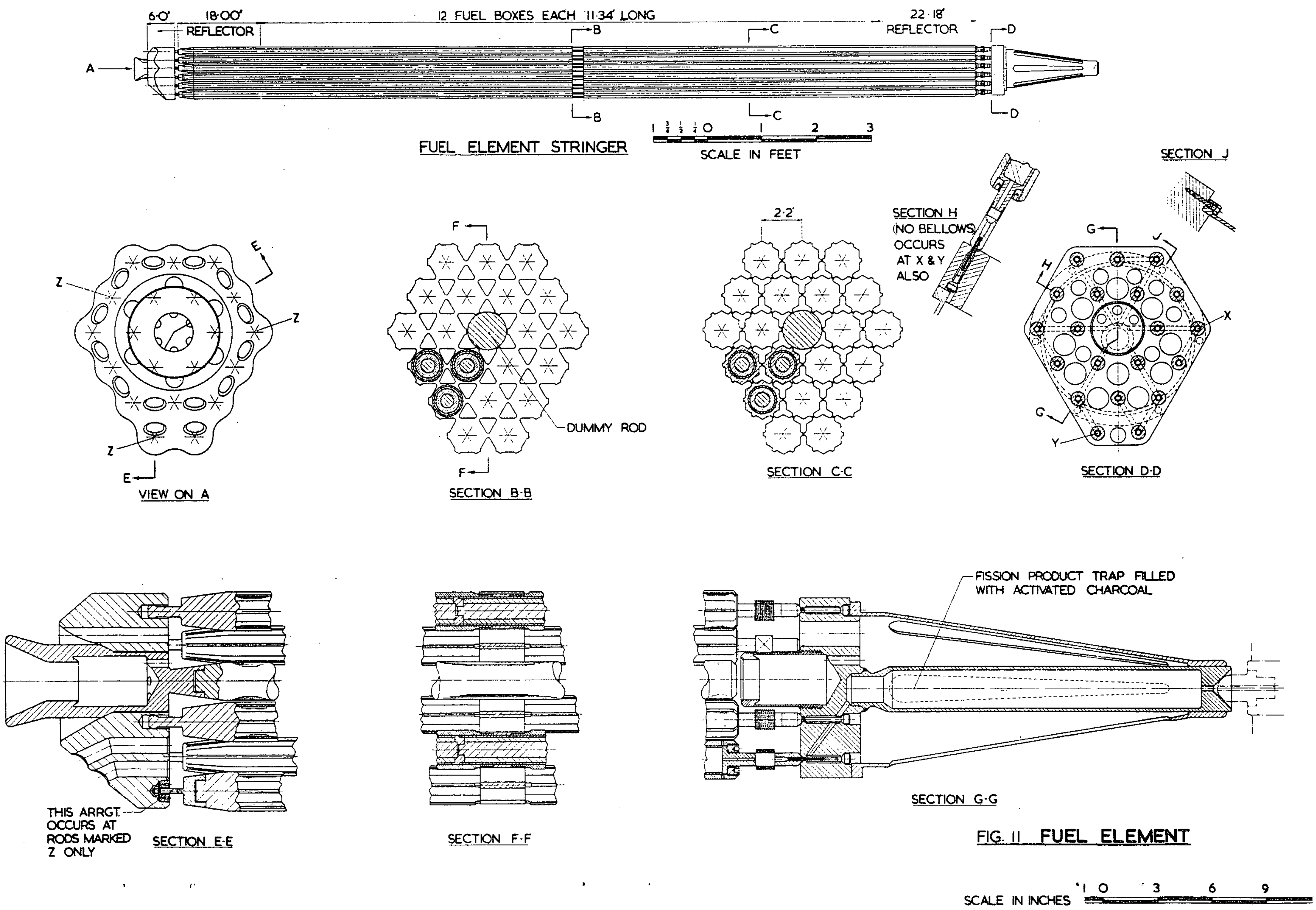


FIG. 11 FUEL ELEMENT

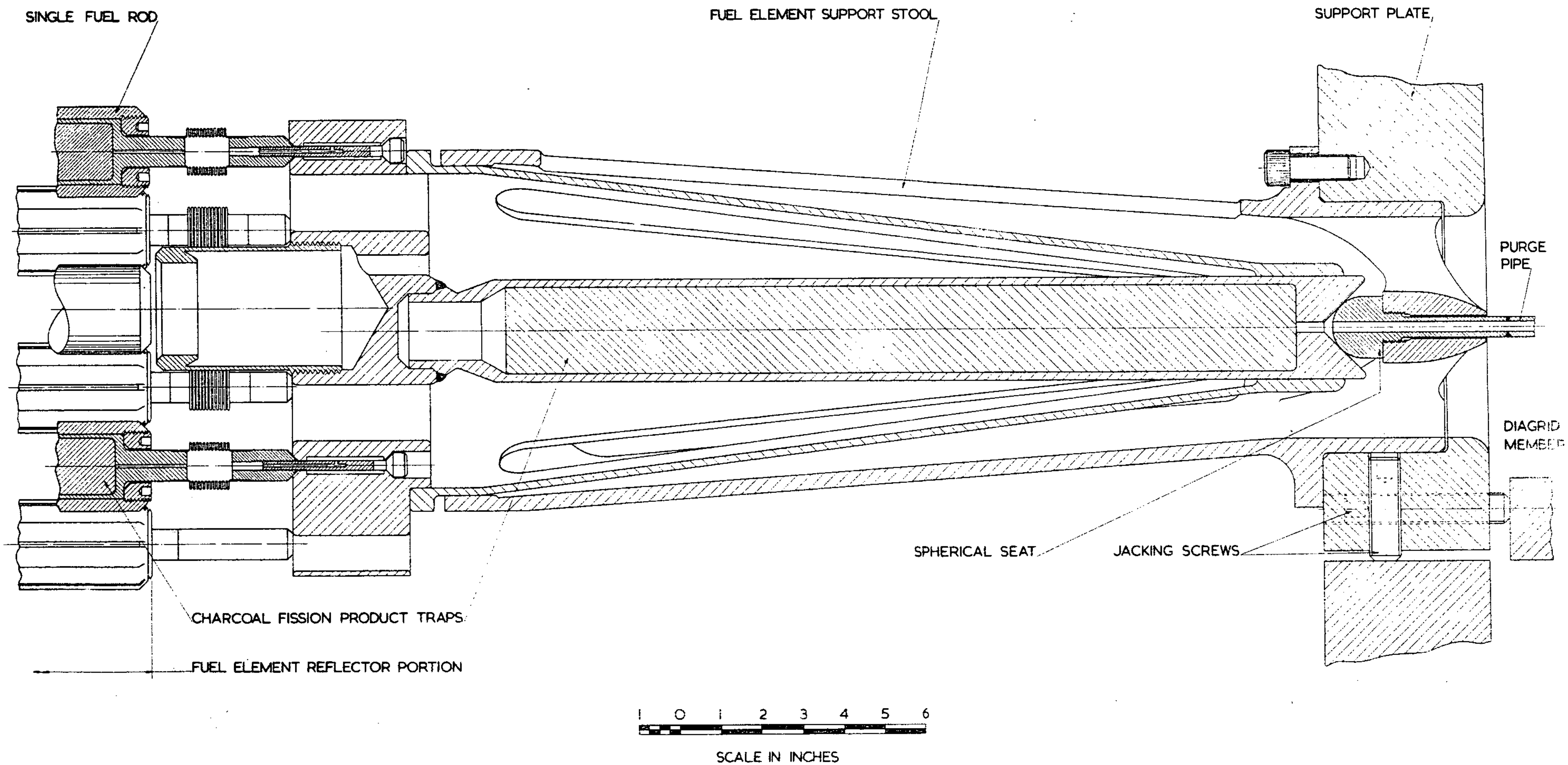


FIG. 12 FUEL ELEMENT SUPPORT

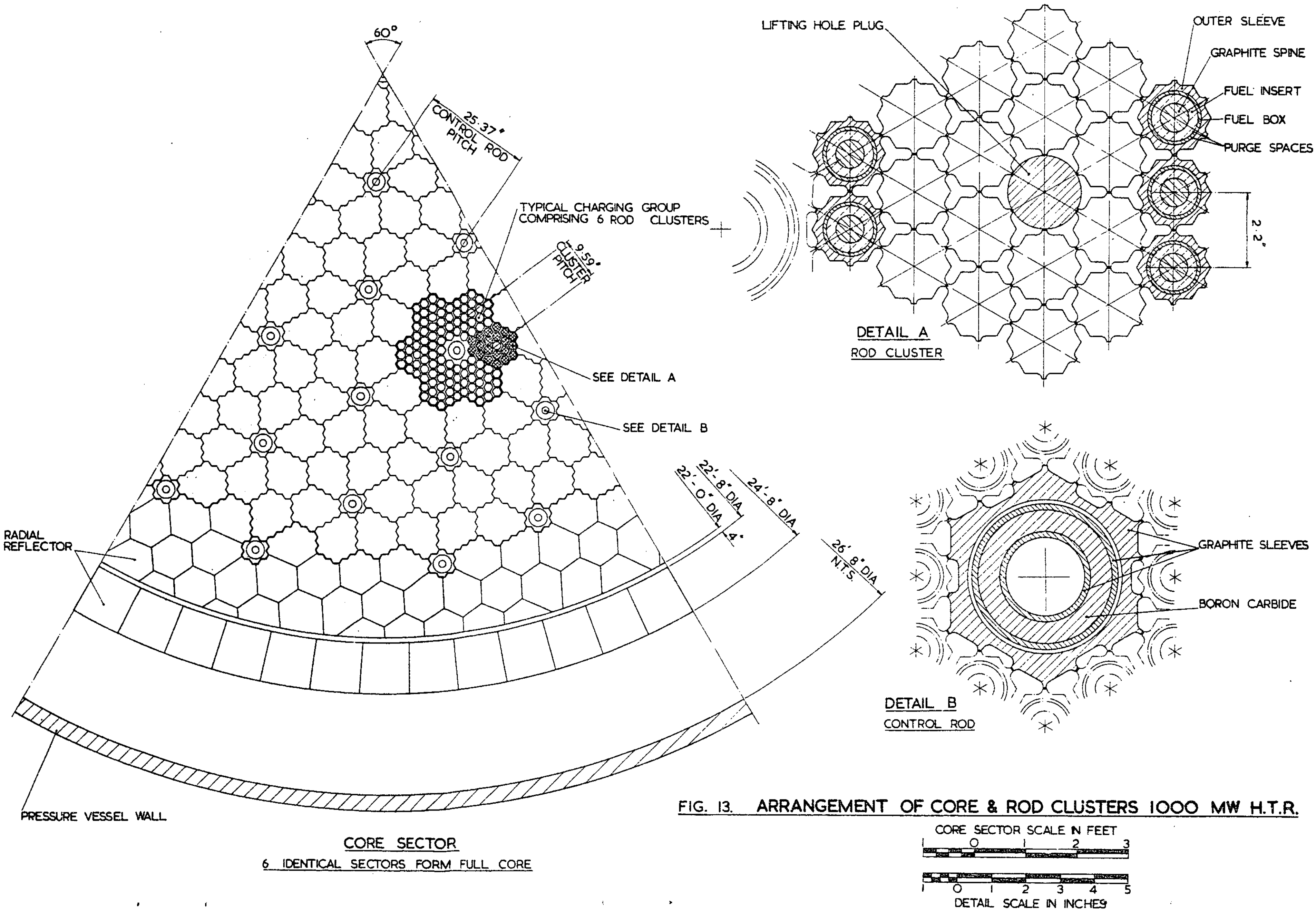


FIG. 13. ARRANGEMENT OF CORE & ROD CLUSTERS 1000 MW H.T.R.

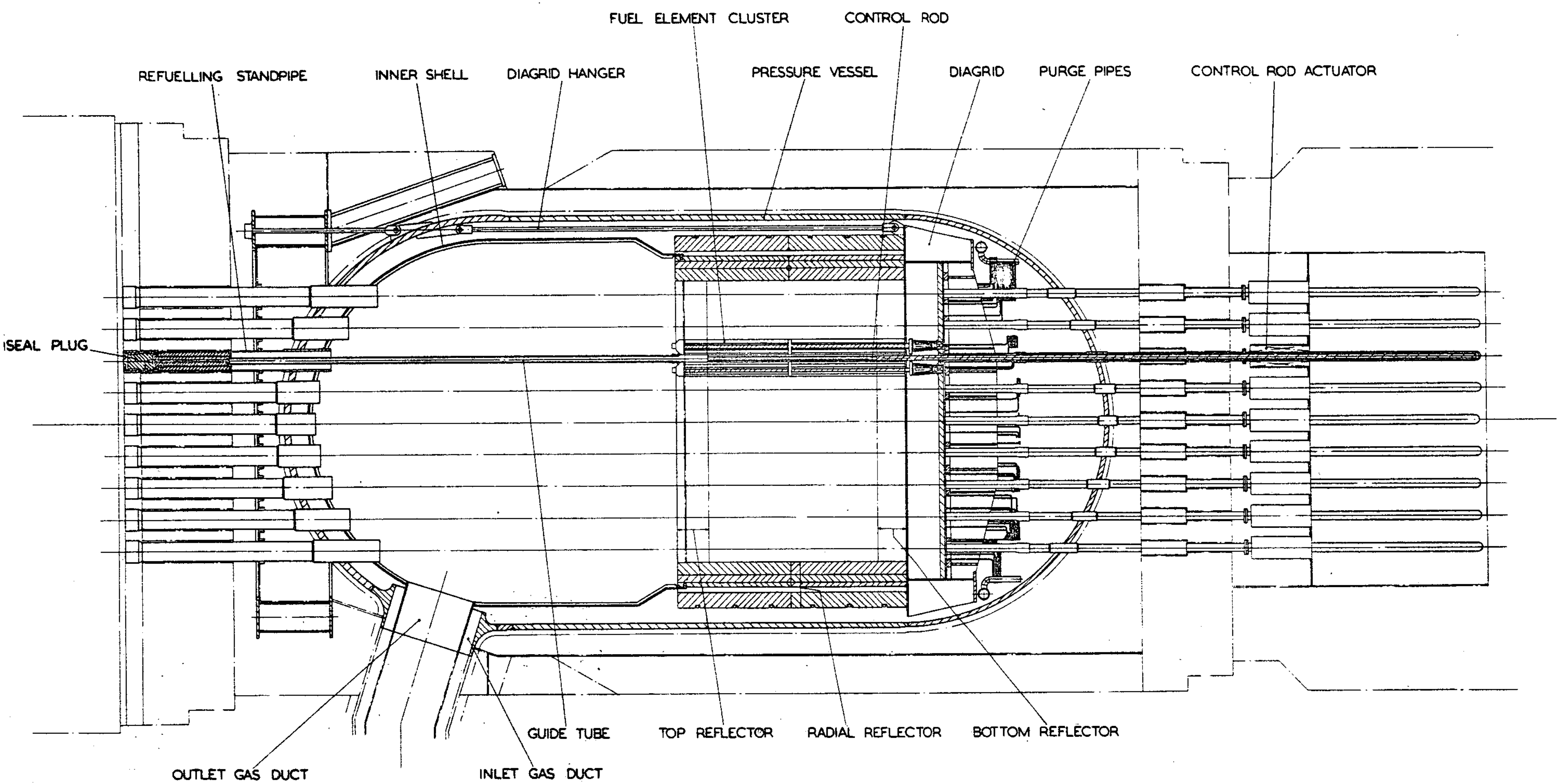
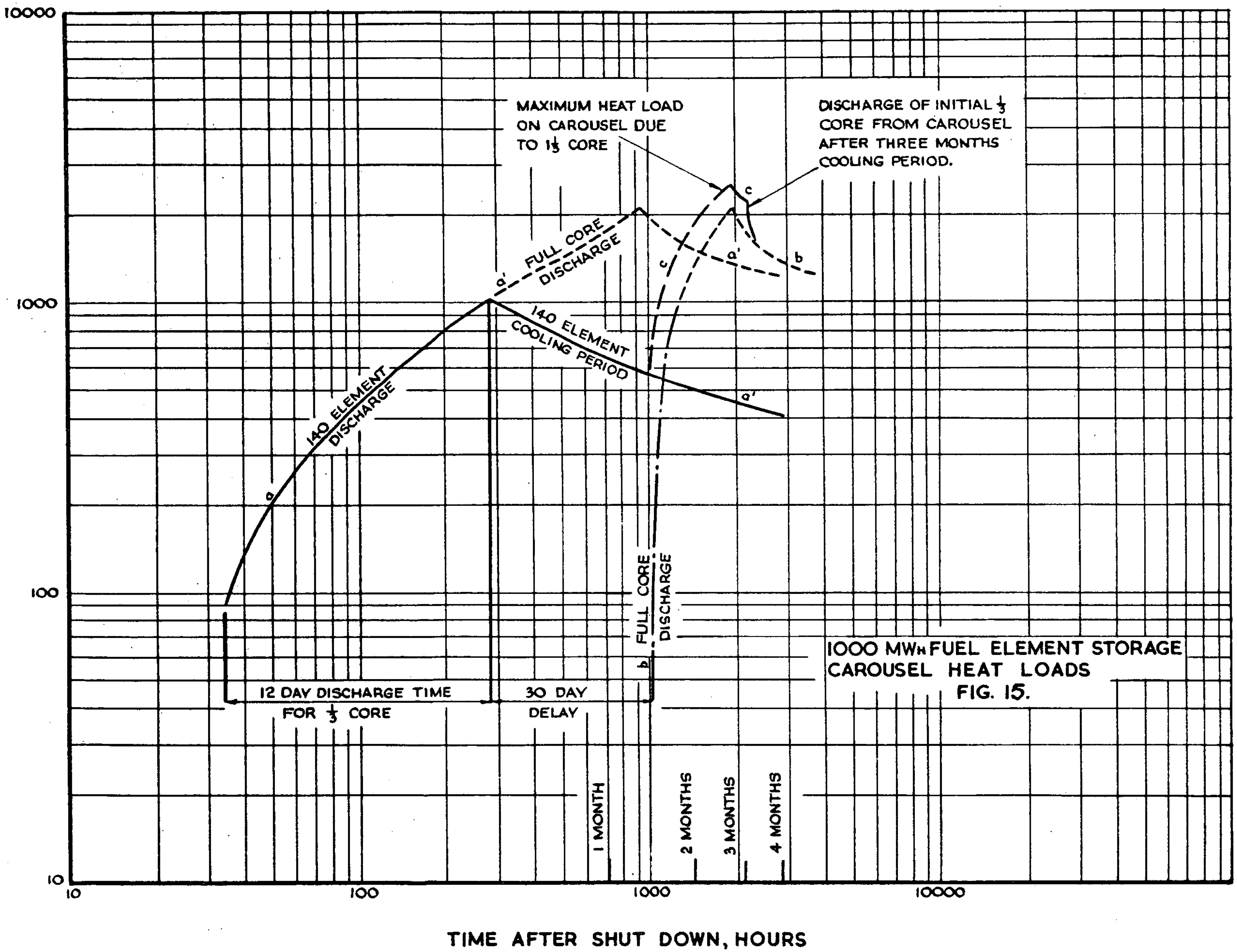
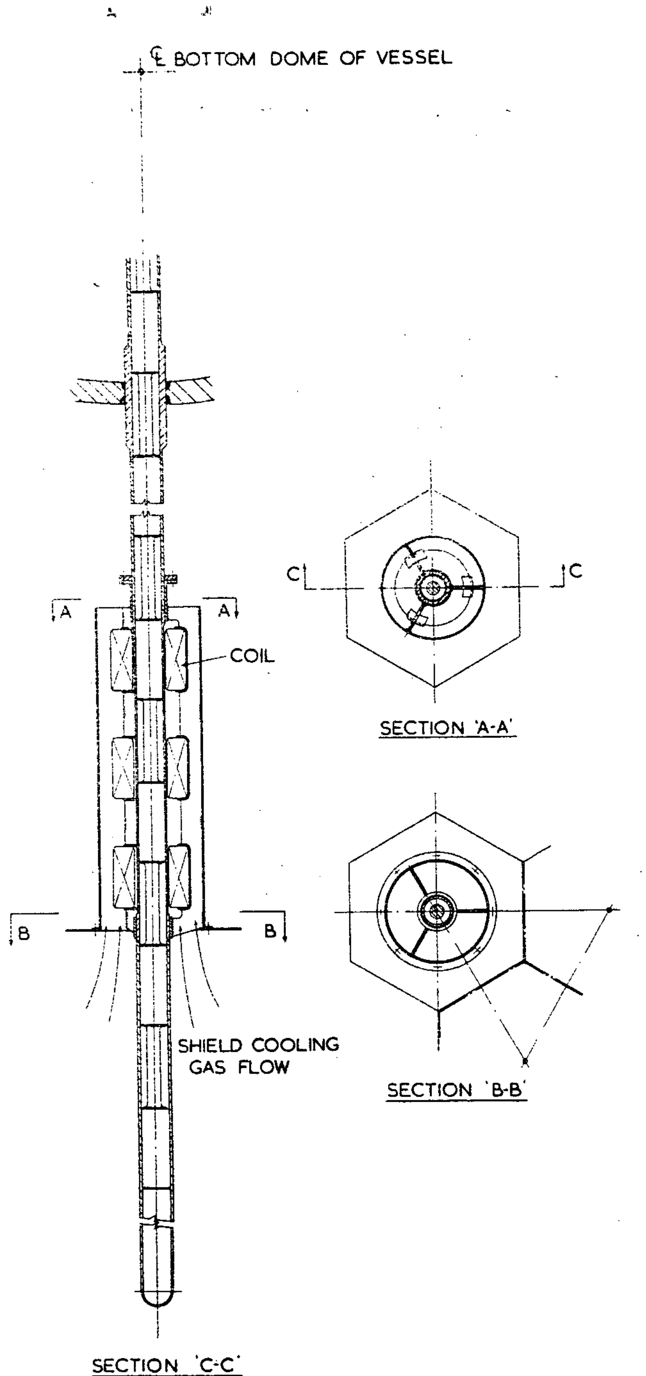


FIG. 14 ARRANGEMENT OF PRESSURE VESSEL

1 0 1 2 3 4 5 6 7 8 9 10
SCALE IN FEET



1000 MWH FUEL ELEMENT STORAGE
CAROUSEL HEAT LOADS
FIG. 15.



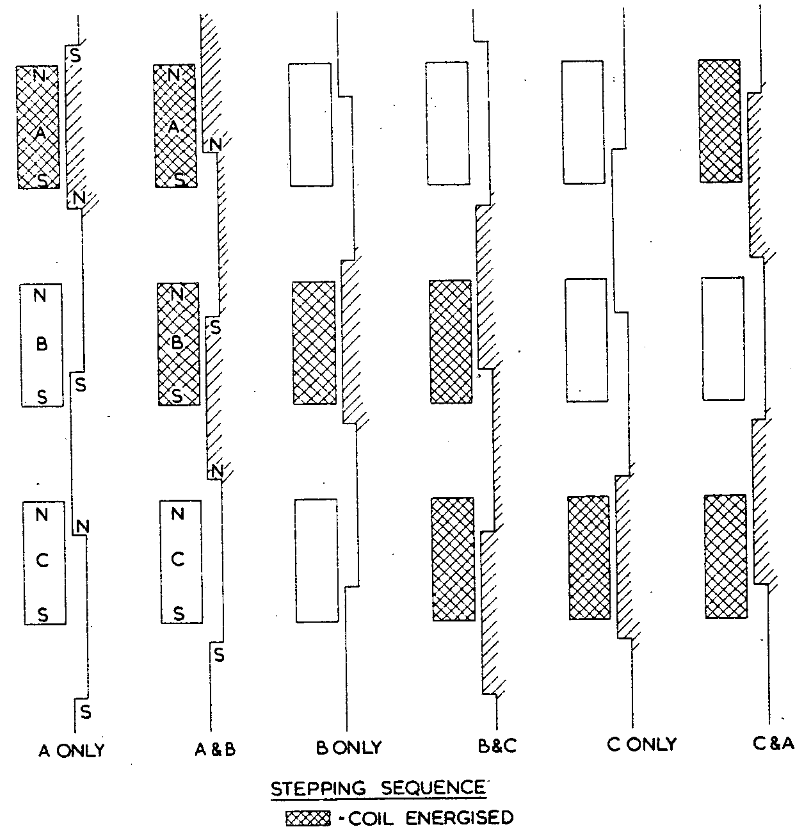


FIG. 16 CONTROL ROD MECHANISM

SCALE IN FEET 1 0 1 2 3

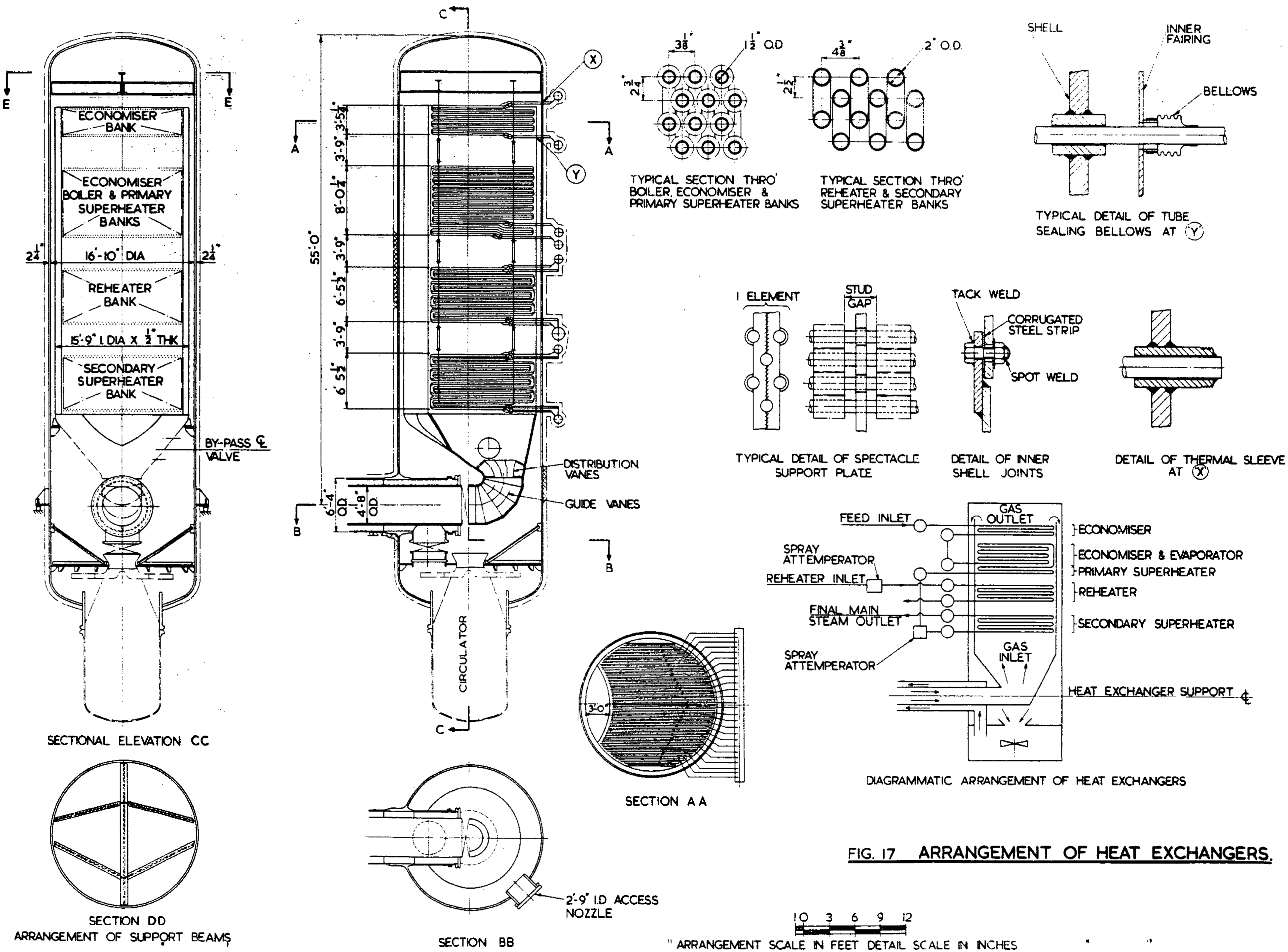
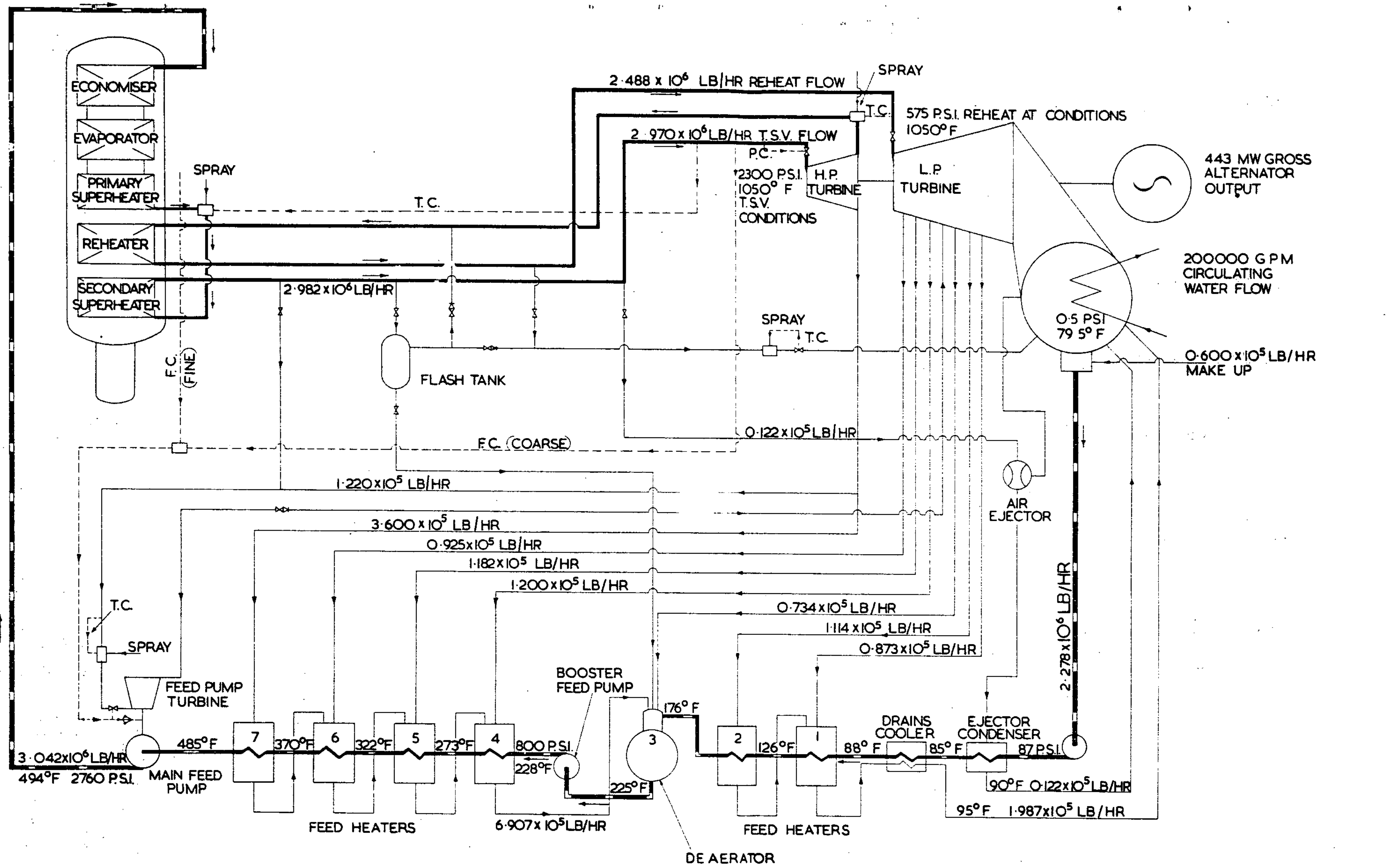


FIG. 17 **ARRANGEMENT OF HEAT EXCHANGERS.**



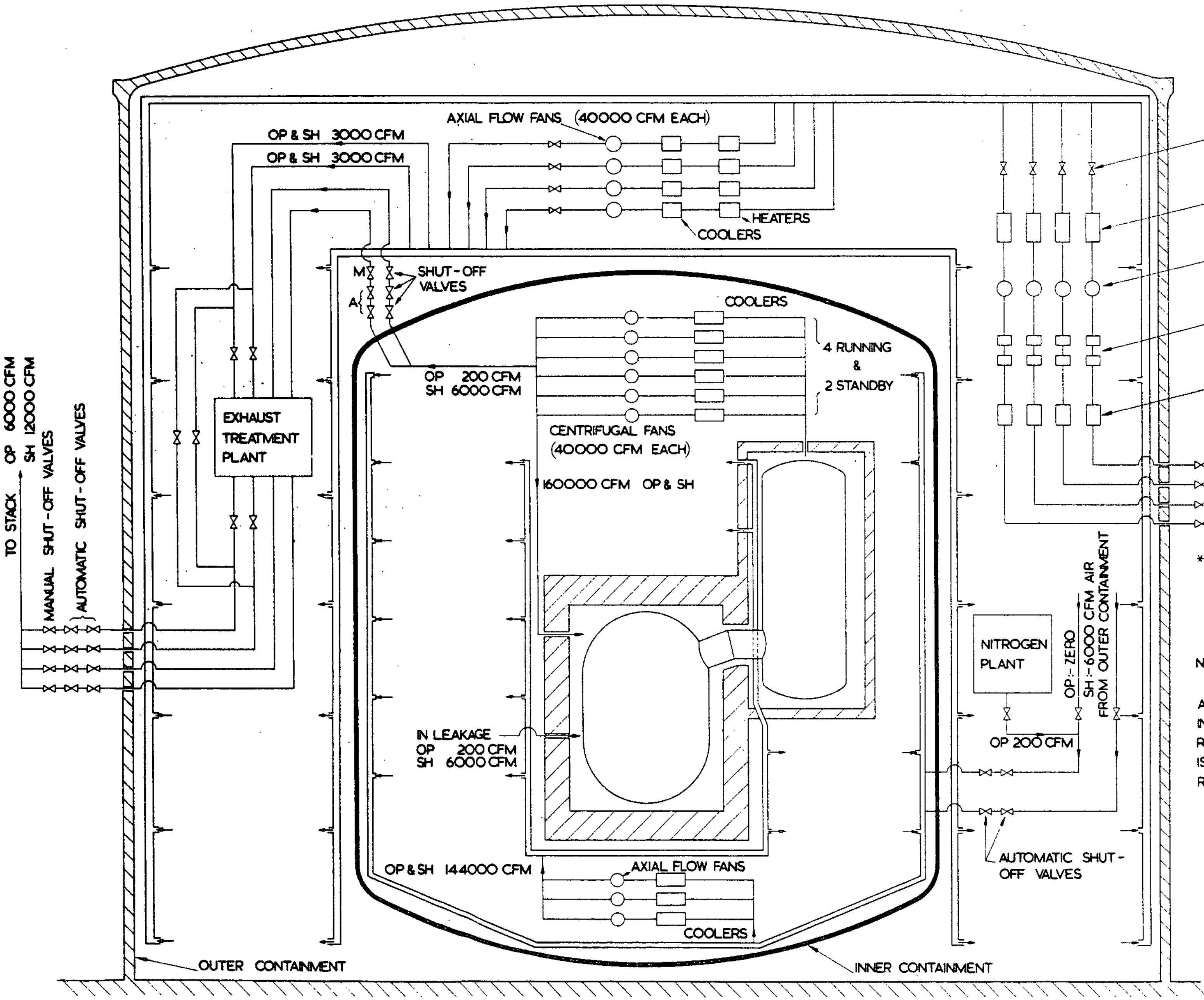
ALL PRESSURES ARE ABSOLUTE

STEAM FLOW
FEED WATER FLOW
BLEED STEAM FLOW
CONTROL SIGNALS

T. C. TEMPERATURE CONTROL
P. C. PRESSURE CONTROL
F. C. FLOW CONTROL

FIG. 18

STEAM FLOW DIAGRAM
FULL LOAD CONDITIONS



* TWO UNITS USED DURING REACTOR OPERATION WITH TWO 50% STANDBY UNITS. ALL FOUR UNITS USED DURING REFUELLING.

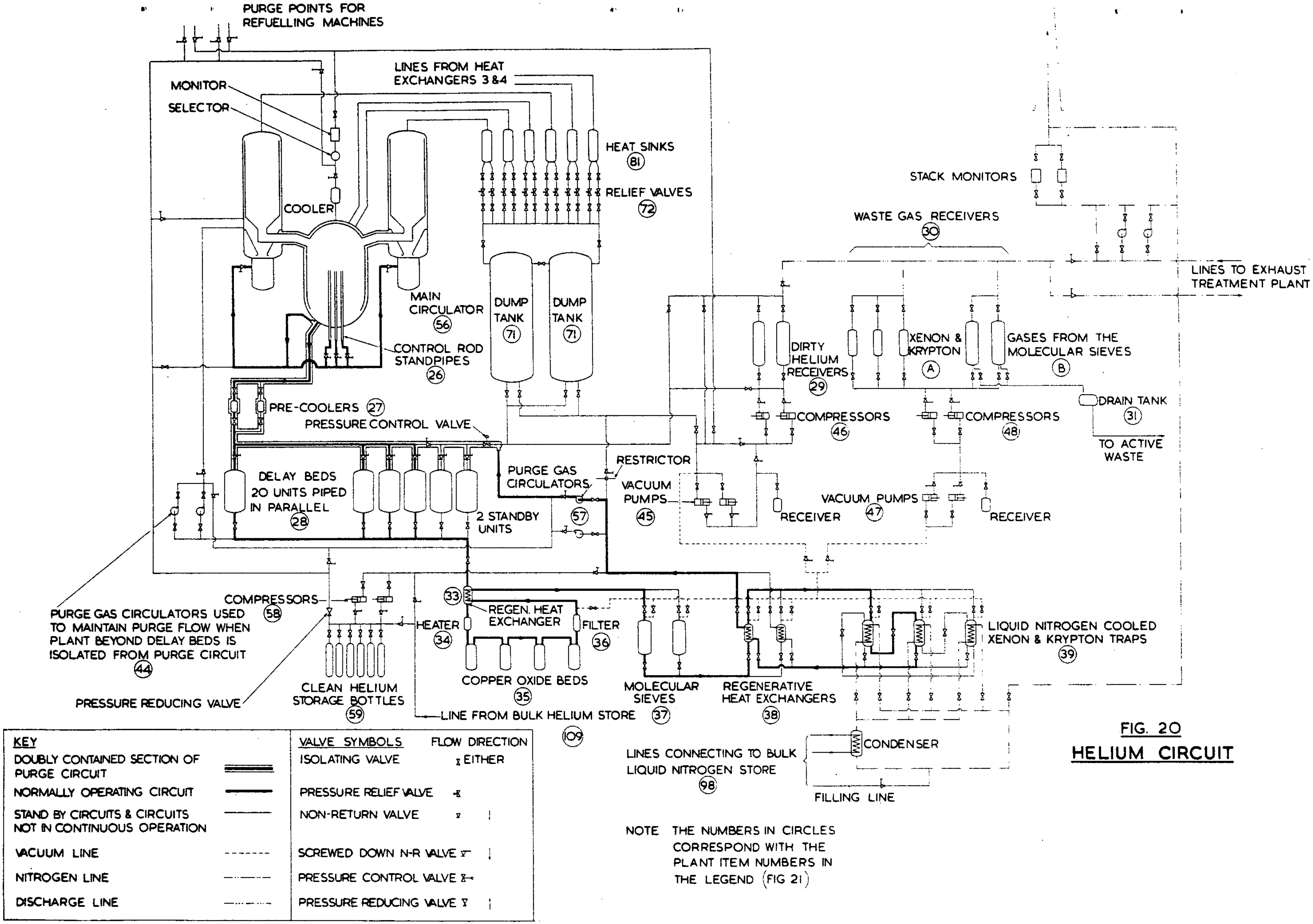
NOTE

THE OUTER CONTAINMENT ATMOSPHERE IS ALWAYS AIR. THE INNER CONTAINMENT IS AIR DURING REFUELLING & MAINTENANCE BUT IS NITROGEN DURING NORMAL REACTOR OPERATION.

OP DENOTES OPERATING
SH " SHUT DOWN

VENTILATION FLOW SHEET

FIG. 19.



NOTE THE NUMBERS IN CIRCLES CORRESPOND WITH THE PLANT ITEM NUMBERS IN THE LEGEND (FIG 21)

FIG. 20

HELIUM CIRCUIT

25	STEEL CONTAINMENT VESSEL			
	INNER CONTAINMENT			**OUTER CONTAINMENT**
26	CONTROL ROD DRIVES & STANDPIPES		86	SERVICES GALLERY
27	PRE-COOLERS	BASEMENT	87	SERVICE TUNNEL TO STACK
28	DELAY BEDS			
29	DIRTY HELIUM RECEIVERS			
30	WASTE GAS RECEIVERS (A) XENON & KRYPTON (B) DIRTY GASES FROM MOLECULAR SIEVES			
31	DRAIN TANK			
32	DELAY BED VALVE ACCESS		88	DIESEL-ALTERNATOR SETS
33	REGENERATIVE HEAT EXCHANGER (HOT)	GROUND FLOOR	89	MOTOR-ALTERNATOR SETS
34	HEATER		90	BATTERIES
35	COPPER OXIDE BEDS		91	LIFT
36	FILTER		92	ACCESS HATCH TO BASEMENT
37	MOLECULAR SIEVES		93	GOODS AIRLOCK
38	REGENERATIVE HEAT EXCHANGER (COLD)		94	GOODS TROLLEY RAILS
39	LIQUID NITROGEN Xe & Kr TRAPS		95	GAS TIGHT DOORS
40	DELAY BED WATER HEAT EXCHANGER			
41	CIRCULATORS (WATER)			
42	PRE-COOLER He / H_2O HEAT EXCHANGER			
43	CIRCULATORS (SECONDARY HELIUM)			
44	PURGE SHUNT CIRCULATORS			
45	VACUUM PUMPS (DIRTY HELIUM)			
46	COMPRESSORS (DIRTY HELIUM)			
47	VACUUM PUMPS (WASTE GASES)			
48	COMPRESSORS (WASTE GASES)			
49	ACCESS HATCH TO BASEMENT			
50	2 TON GANTRY CRANE			
51	CORE		96	SPENT FUEL COOLING MAGAZINE
52	FIXED REFLECTOR	FIRST FLOOR	97	MAGAZINE COOLANT BLOWERS & COOLERS
53	DIAGRID		98	LIQUID NITROGEN BULK STORAGE TANKS
54	CONTROL ROD GUIDE TUBES		99	EMERGENCY AIRLOCK
55	PRESSURE VESSEL		100	LIFTING WELL CLEAN EQUIPMENT
56	MAIN COOLANT CIRCULATORS		101	LIFTING WELL DIRTY EQUIPMENT
57	PURGE GAS CIRCULATORS		102	RETURN MAIN FROM EXHAUST TREATMENT PLANT TO INNER CONTAINMENT
58	CLEAN HELIUM COMPRESSORS			
59	CLEAN HELIUM STORAGE BOTTLES		103	FUEL ELEMENT DISCHARGE LOCK
60	CIRCULATOR TROLLEY		104	COFFIN HOIST
61	CIRCULATOR TROLLEY RAILS		105	COFFIN TROLLEY
62	LIFTING WELL			
63	VENTILATION SYSTEM BLOWER / COOLERS			
64	VENTILATION DELIVERY RING MAIN			
65	VENTILATION SUCTION RING MAIN			
66	VENTILATION DELIVERY DUCT			
67	VENTILATION SUCTION DUCT		106	COOLING MAGAZINE DRIVE
68	SPENT FUEL EJECTION RAM		107	MAGAZINE COOLANT FILTERS
			108	CHANGE ROOM
69	RING BEAM & 'A' FRAMES		109	CLEAN HELIUM BULK STORAGE BOTTLES
70	HEAT EXCHANGERS	SECOND FLOOR	110	COMPRESSORS
71	DUMP TANKS		111	STEAM & FEED WATER PIPES
72	RELIEF VALVES		112	PIPE ANCHOR FRAMES
73	TILTING MACHINE		113	THERMAL SLEEVES
74	NEW FUEL HOLDING FACILITY		114	MONITORING EQUIPMENT
75	STANDPIPES		115	EXHAUST TREATMENT PLANT
76	CHUTE MACHINE	CHARGE FACE	116	INLET AIR TREATMENT PLANT
77	CHARGE MACHINE		117	REFRIGERATION PLANT
78	SPECIAL PURPOSE MACHINE		118	PERSONNEL AIR LOCK
79	NEW FUEL INTAKE SKIRT			
80	CHUTE MACHINE PARKING SKIRT		119	VENTILATION BLOWER / COOLERS
81	HEAT SINKS		120	VENTILATION DELIVERY RING MAIN
82	DUMP TANK SUPPORT FRAME		121	VENTILATION SUCTION RING MAIN
83	HEAT EXCHANGER SUPPORT FRAME		122	VENTILATION DELIVERY DUCT
			123	VENTILATION SUCTION DUCT
84	SHIELD COOLING EQUIPMENT		124	ACCESS HATCHES TO EXHAUST TREATMENT PLANT
85	200 TON CRANE	TOP FLOOR	125	EMERGENCY WATER STORAGE TANKS (AUXILIARIES)

FIG. 21 LEGEND

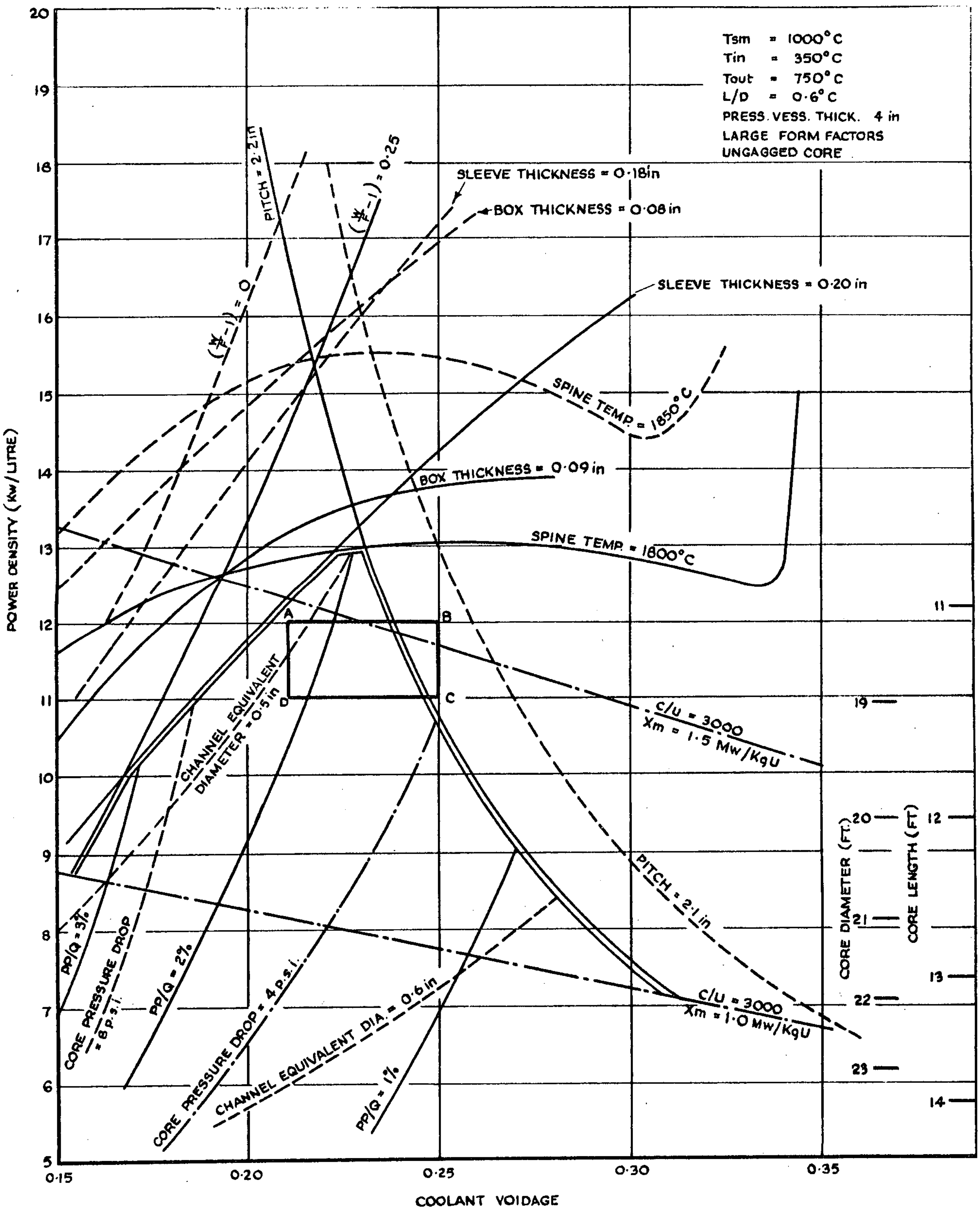


FIG.22. THE POSSIBLE DESIGN RANGE FOR CHOSEN STANDARD INPUT PARAMETERS. THE FOLLOWING CURVES REPRESENT THE EFFECT OF CHANGING ONE OR MORE OF THESE PARAMETERS.

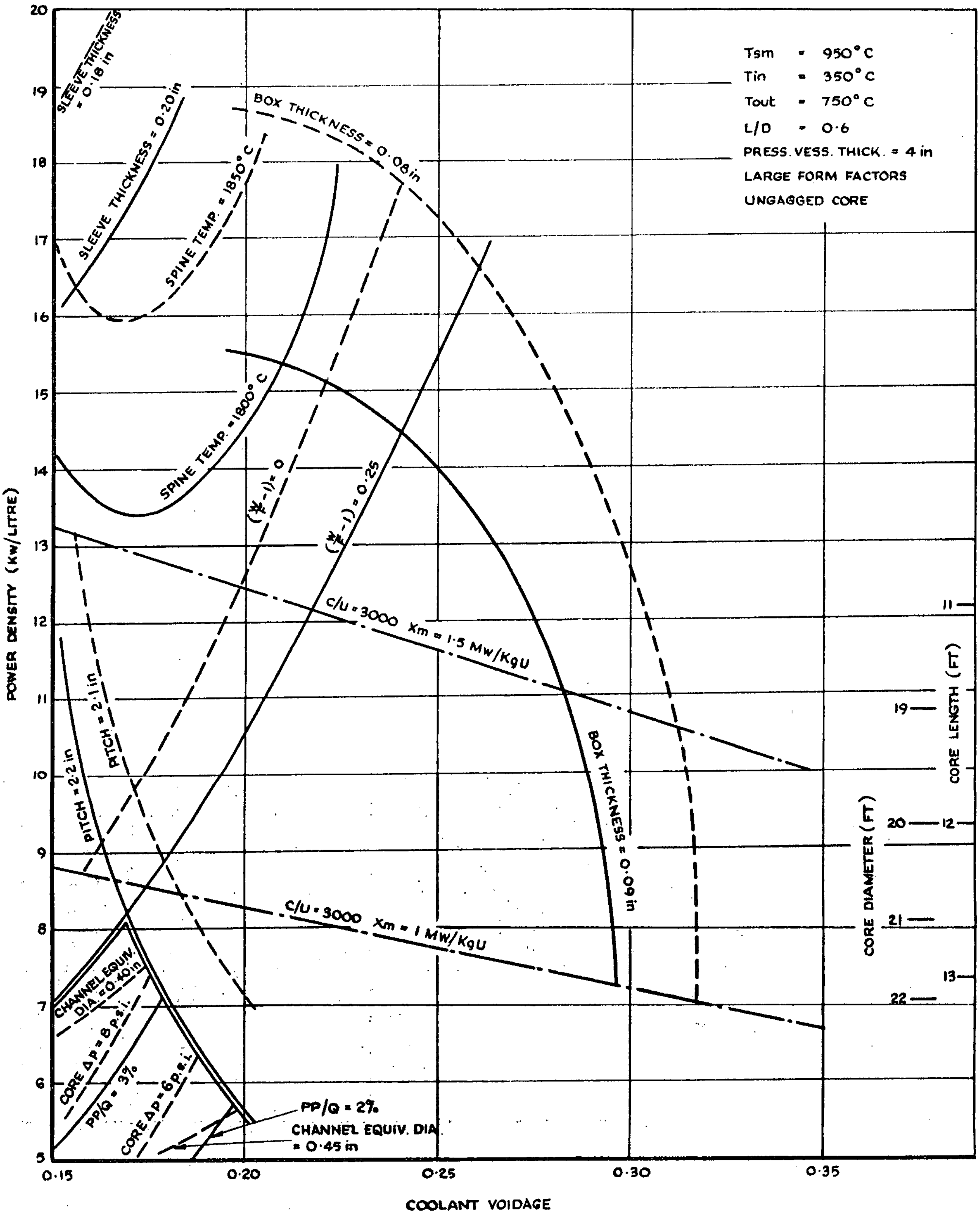


FIG.23. THE EFFECT OF REDUCING MAXIMUM GRAPHITE SURFACE TEMPERATURE TO 950° C.

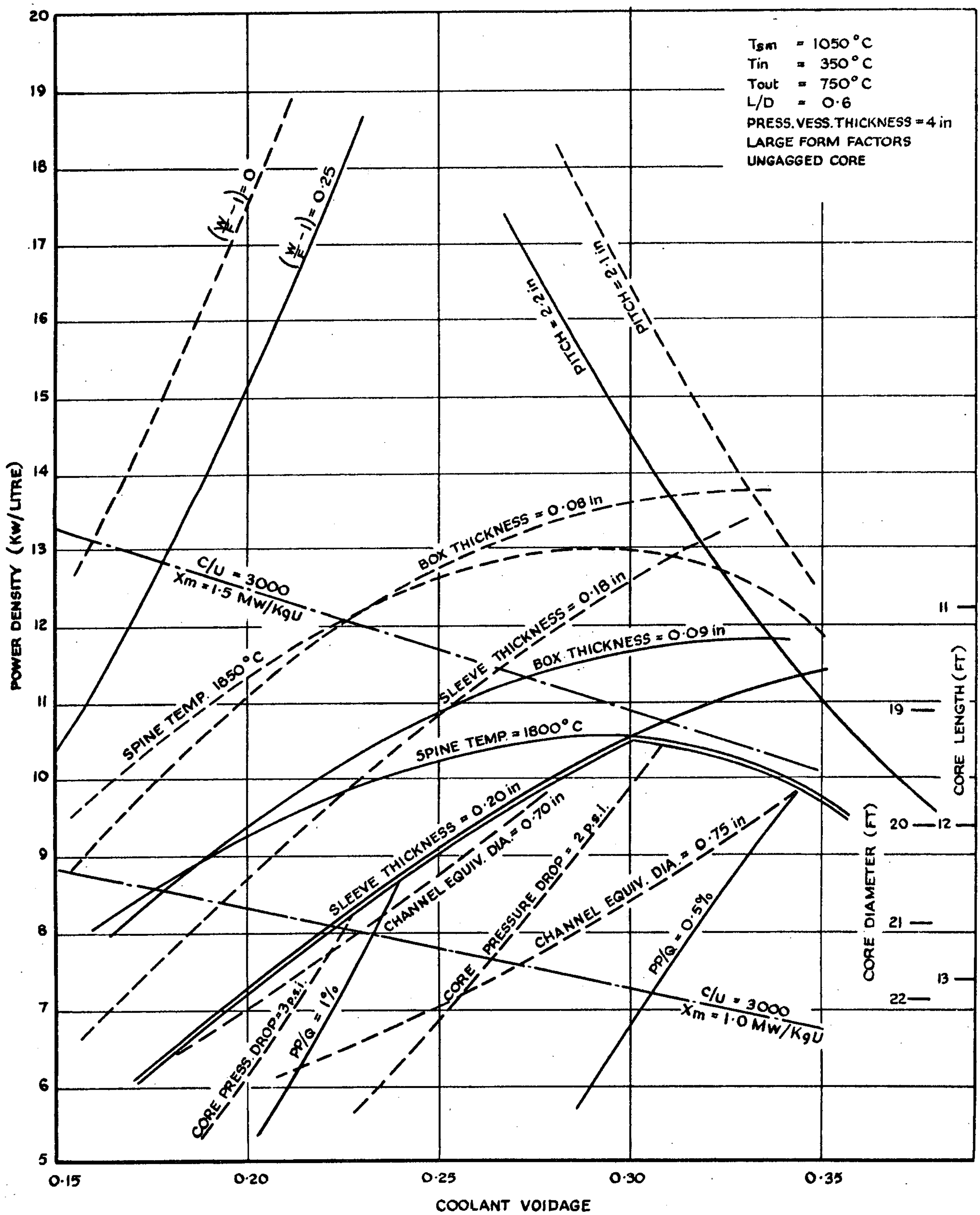


FIG. 24. THE EFFECT OF INCREASING MAXIMUM GRAPHITE SURFACE TEMPERATURE TO 1050° C.

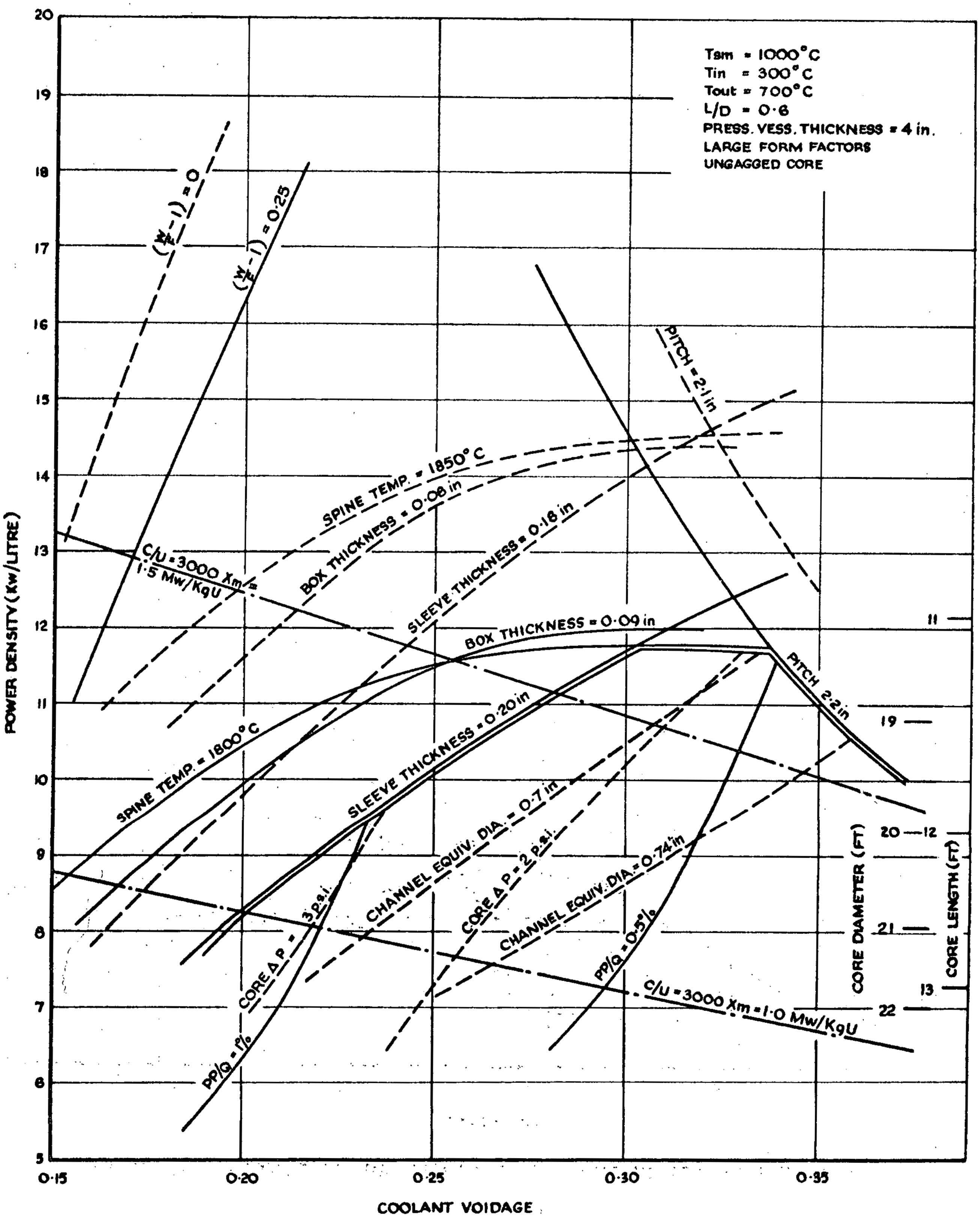


FIG. 25. THE EFFECT OF REDUCING COOLANT INLET AND OUTLET TEMPERATURES.
(DIFFERENCE FROM FIG. 6 DUE TO PHYSICAL PROPERTY CHANGES)

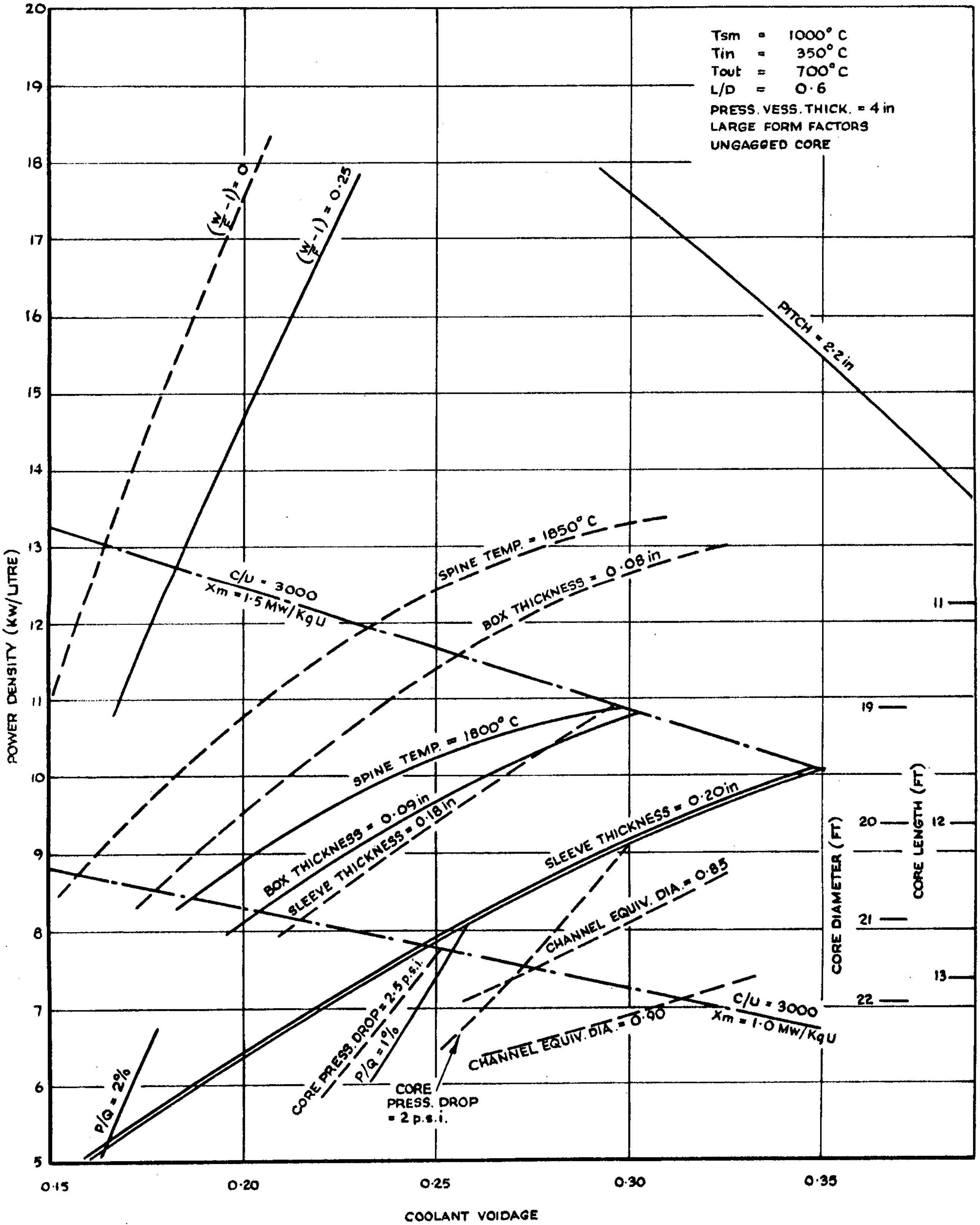


FIG. 26. THE EFFECT OF REDUCING COOLANT OUTLET TEMPERATURE TO 700° C.

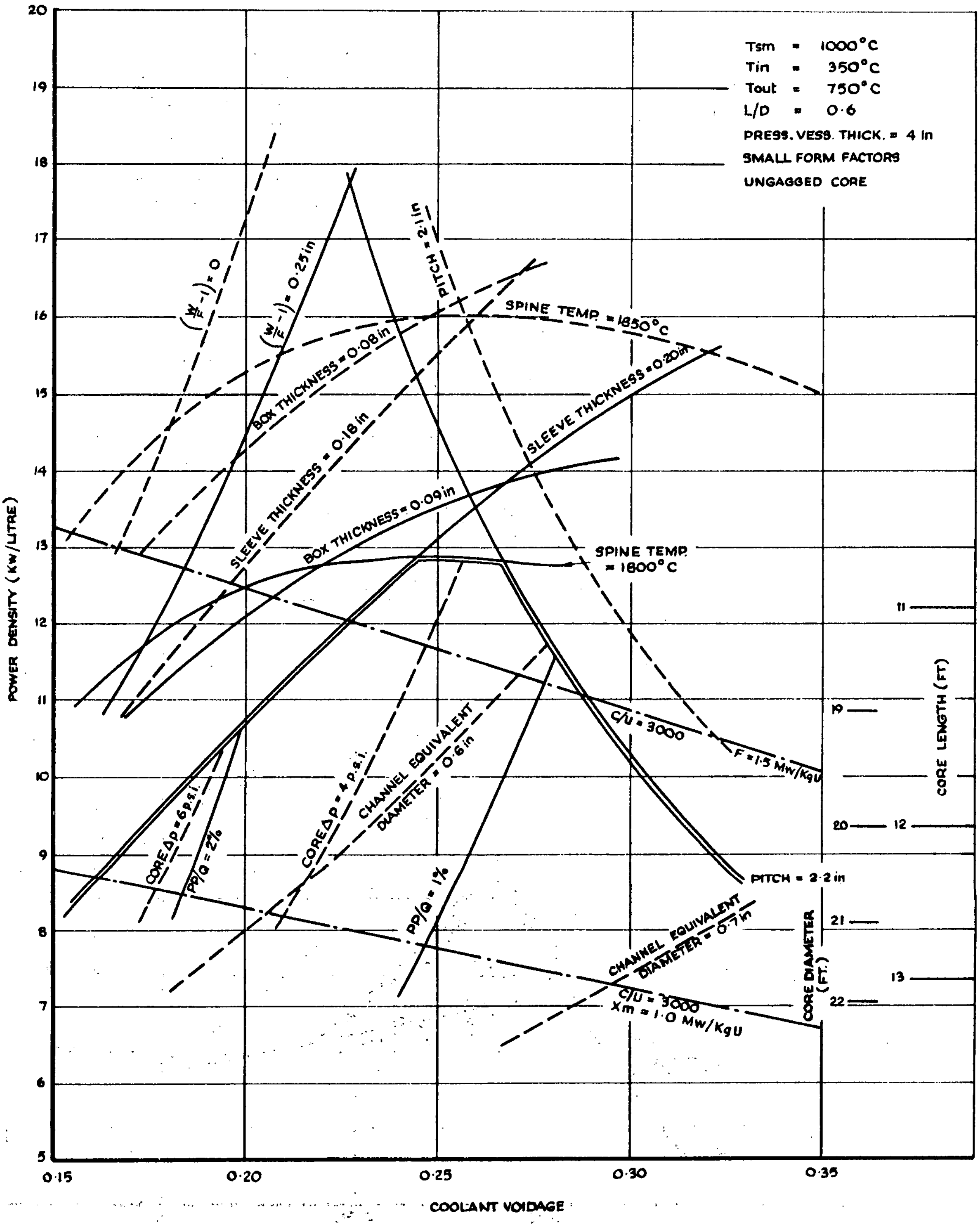


FIG. 27. THE EFFECT OF A REDUCTION IN FORM FACTORS.

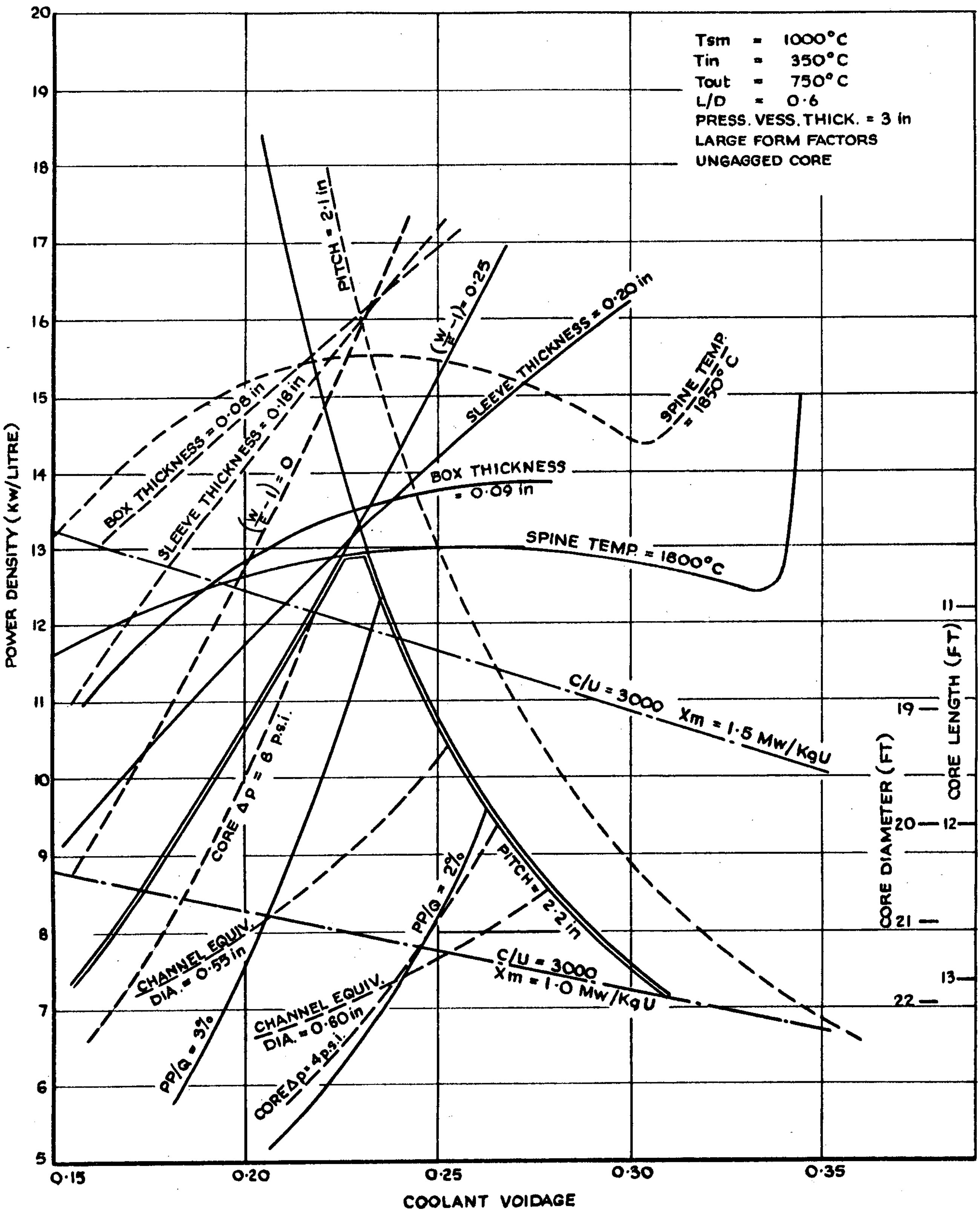


FIG. 28. THE EFFECT OF A REDUCTION IN PRESSURE VESSEL THICKNESS.

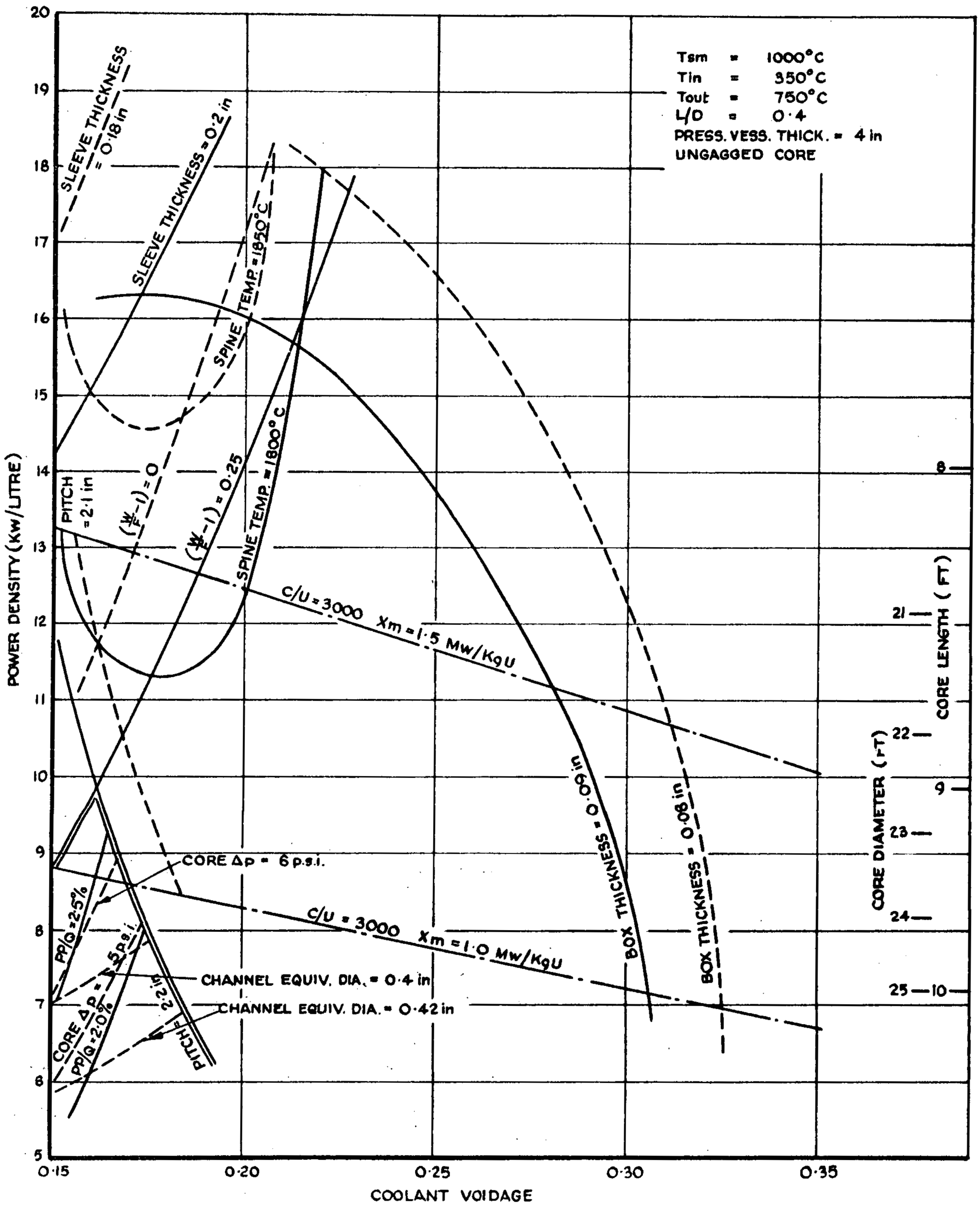


FIG. 29. THE EFFECT OF REDUCING L/D TO A VALUE OF 0·4.

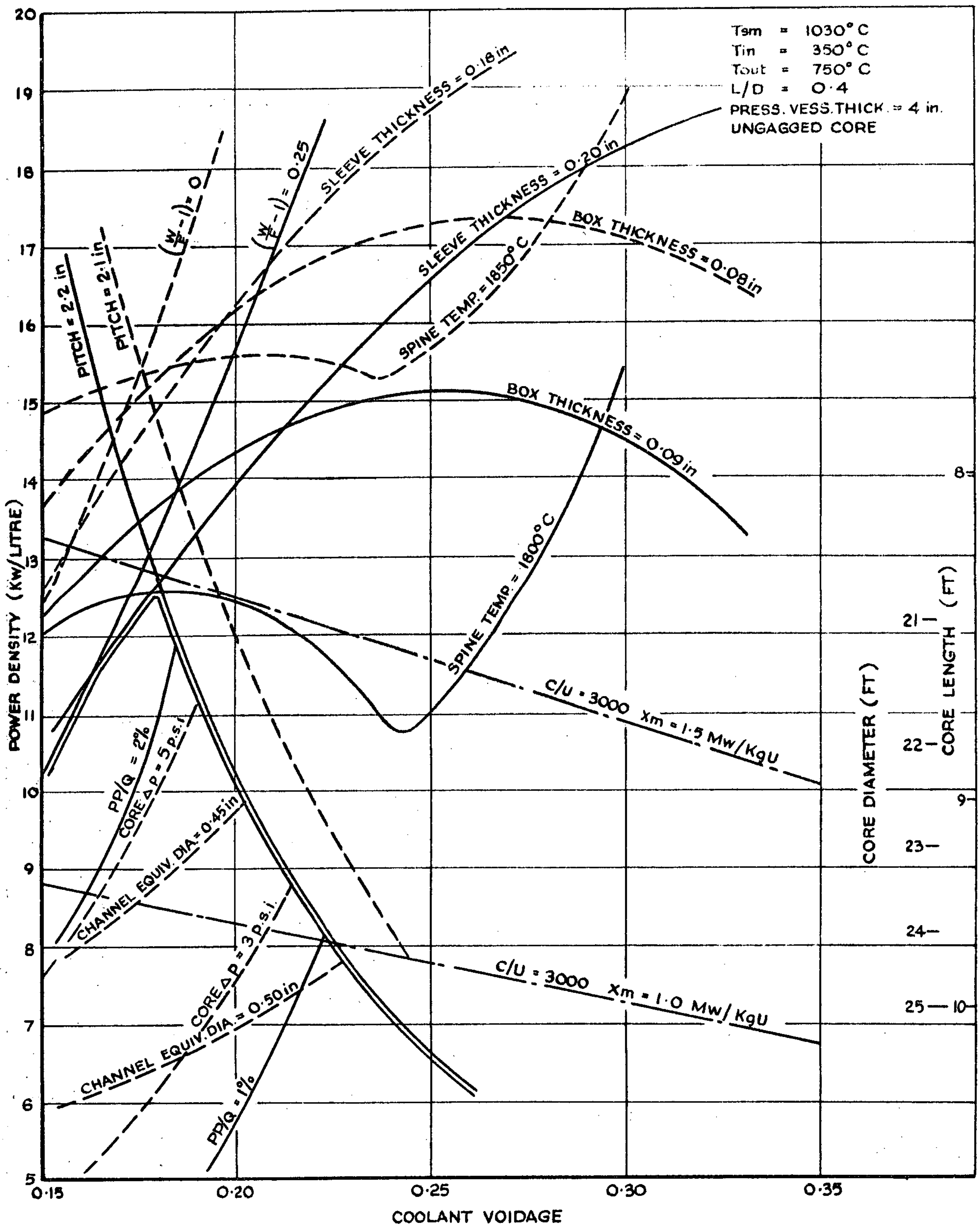


FIG.30. THE EFFECT OF INCREASING THE MAXIMUM GRAPHITE SURFACE TEMPERATURE TO 1030°C (L/D 0·4)

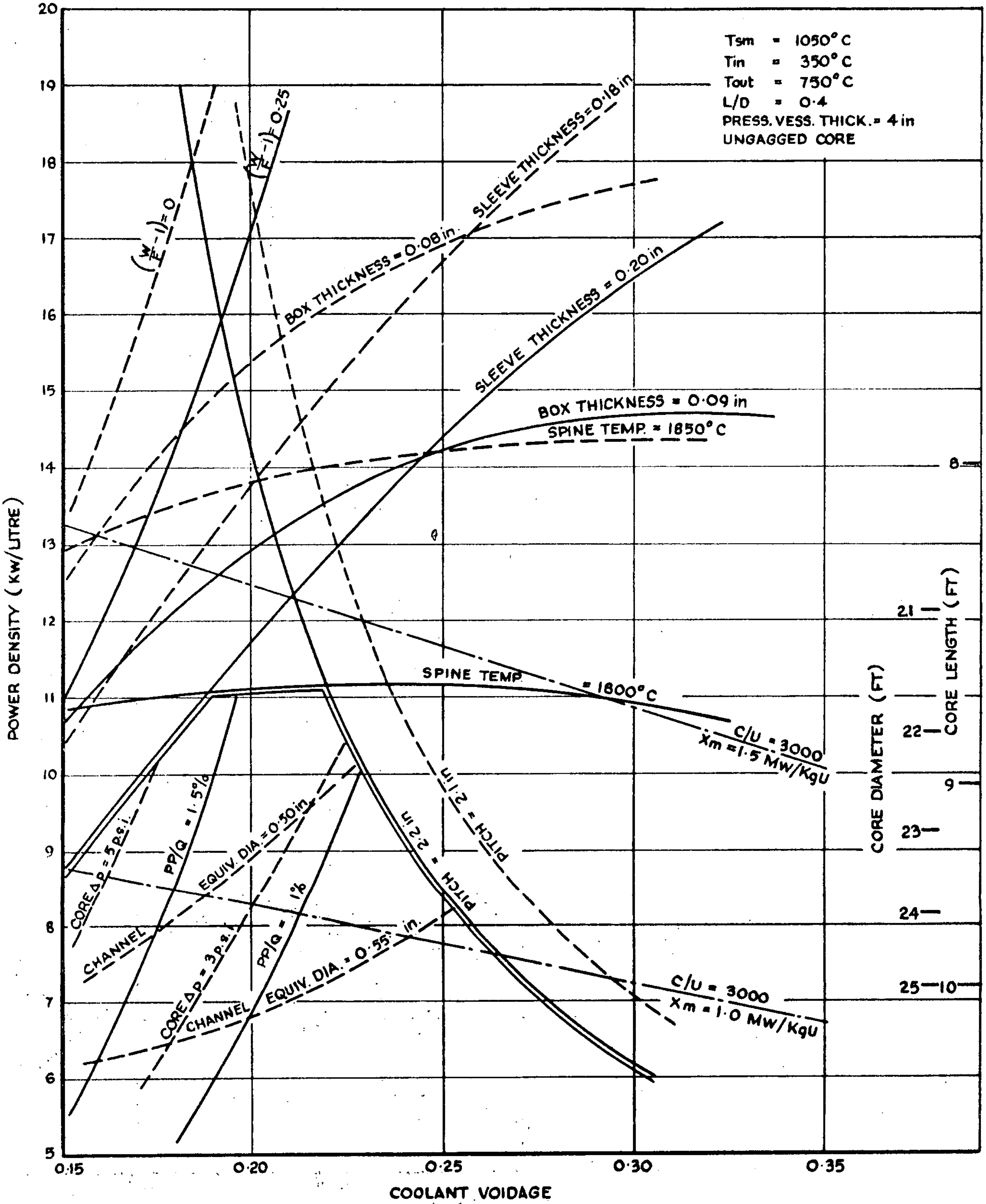


FIG. 31. THE EFFECT OF INCREASING THE MAXIMUM GRAPHITE SURFACE TEMPERATURE TO 1050°C (L/D 0·4)

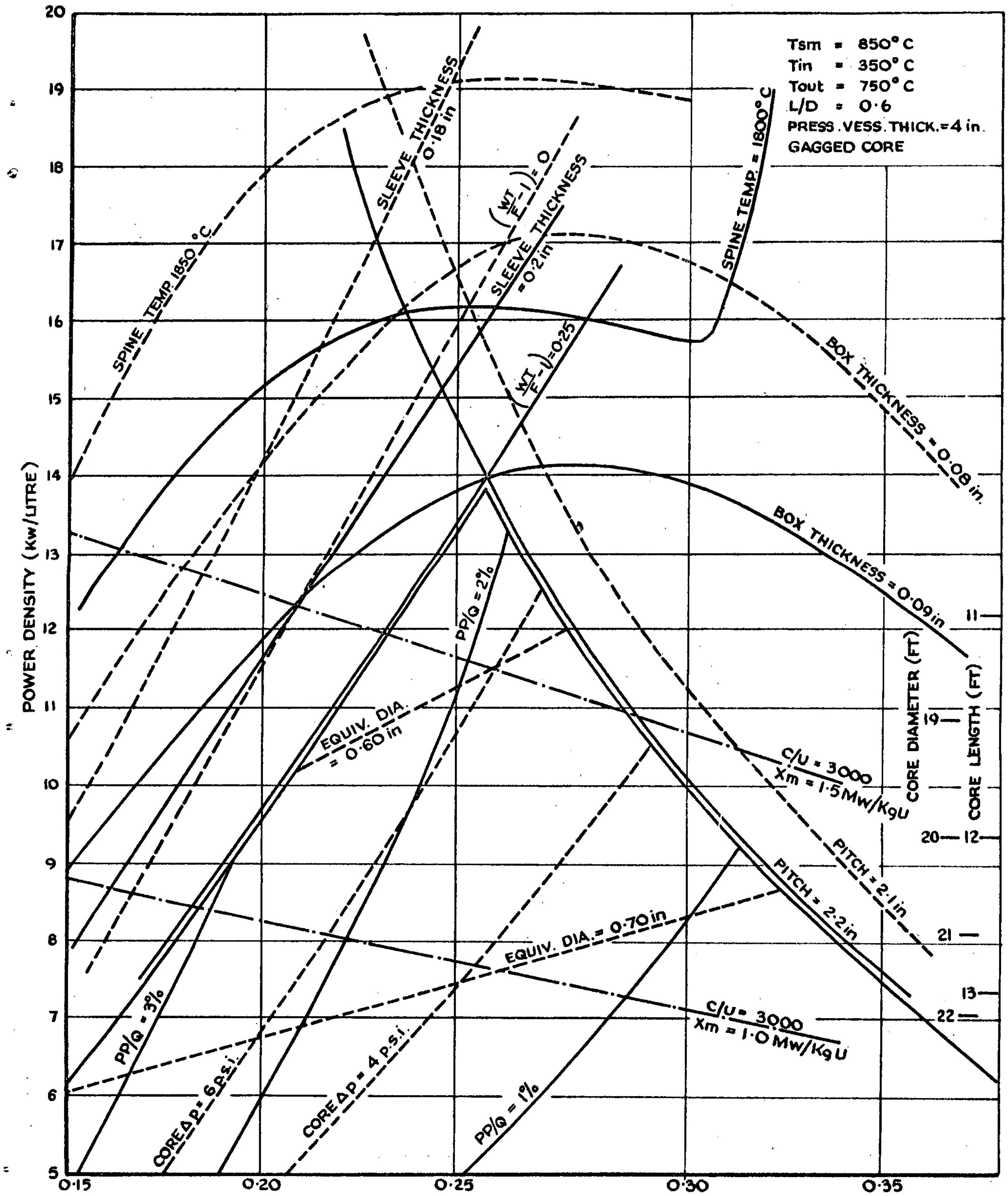


FIG.32. APPROXIMATE CONDITIONS FOR MAXIMUM POWER DENSITY FROM A GAGGED CORE WITH A L/D RATIO OF 0·6.

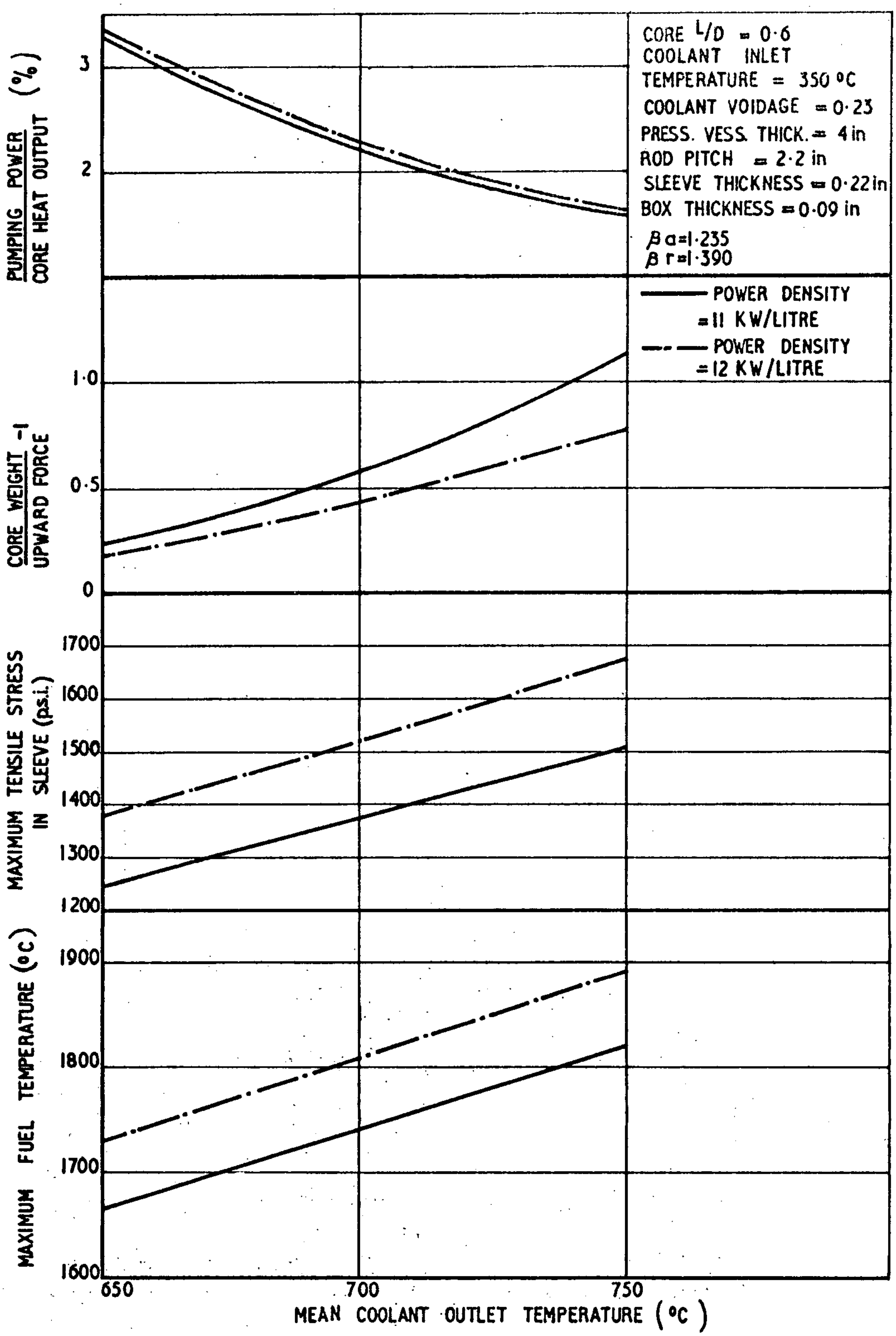


FIG. 33. THE EFFECT OF VARYING COOLANT OUTLET TEMPERATURE ON CORE PERFORMANCE

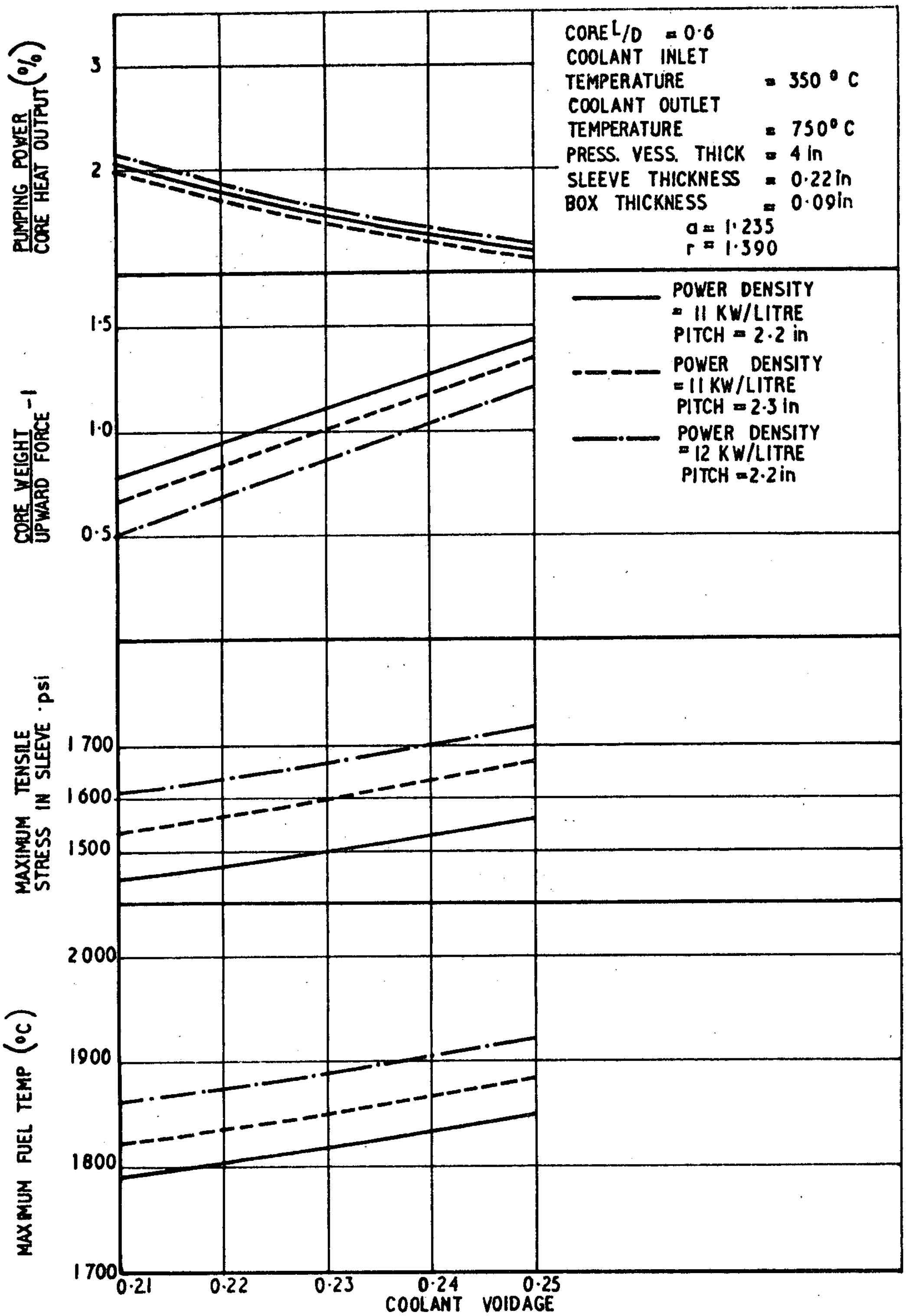


FIG.34. THE EFFECT OF SMALL VARIATION IN VOIDAGE, ROD PITCH AND POWER DENSITY ON CORE PERFORMANCE.

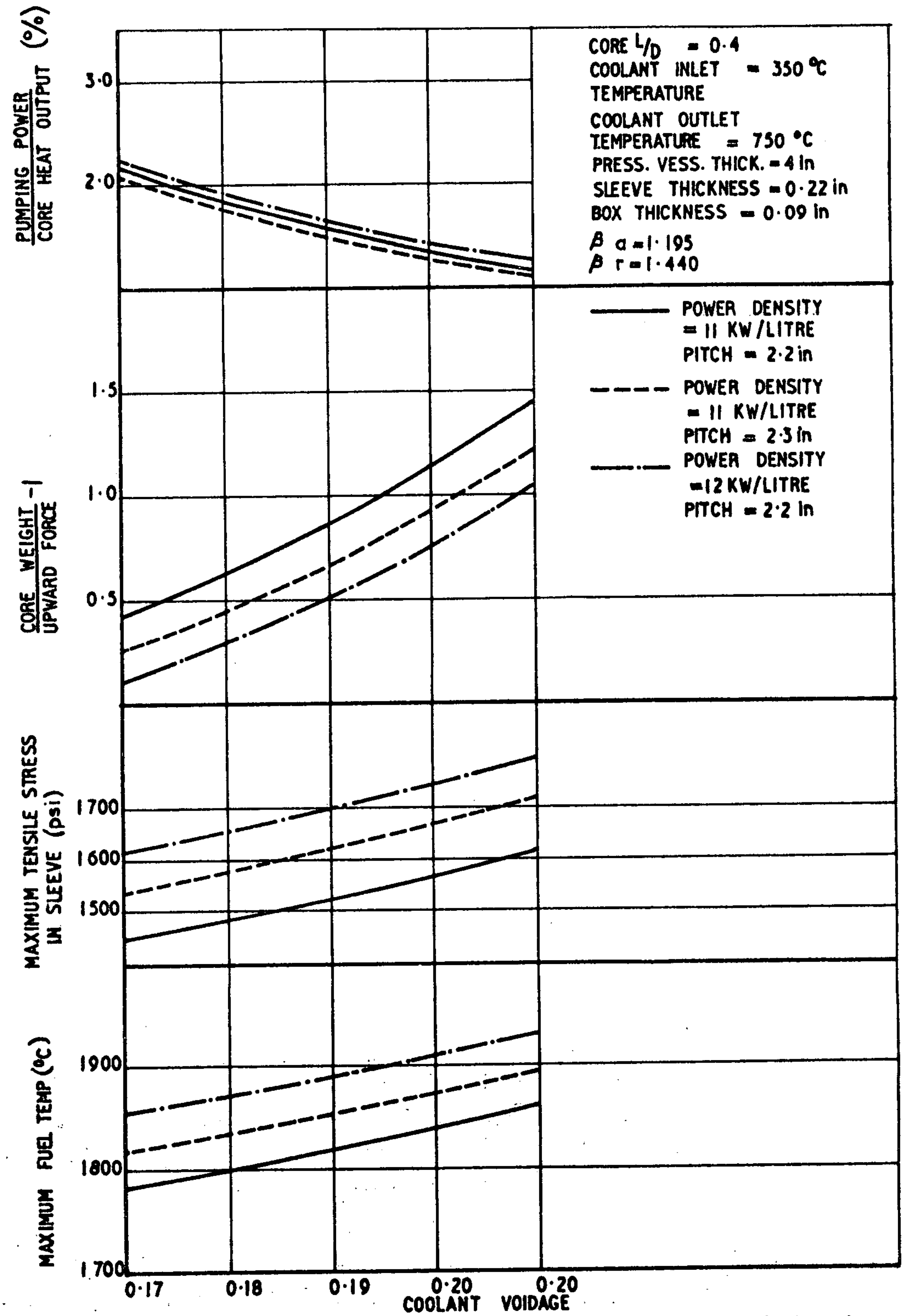


FIG. 35. THE EFFECT OF SMALL VARIATIONS IN VOIDAGE, ROD PITCH, AND POWER DENSITY ON CORE PERFORMANCE.

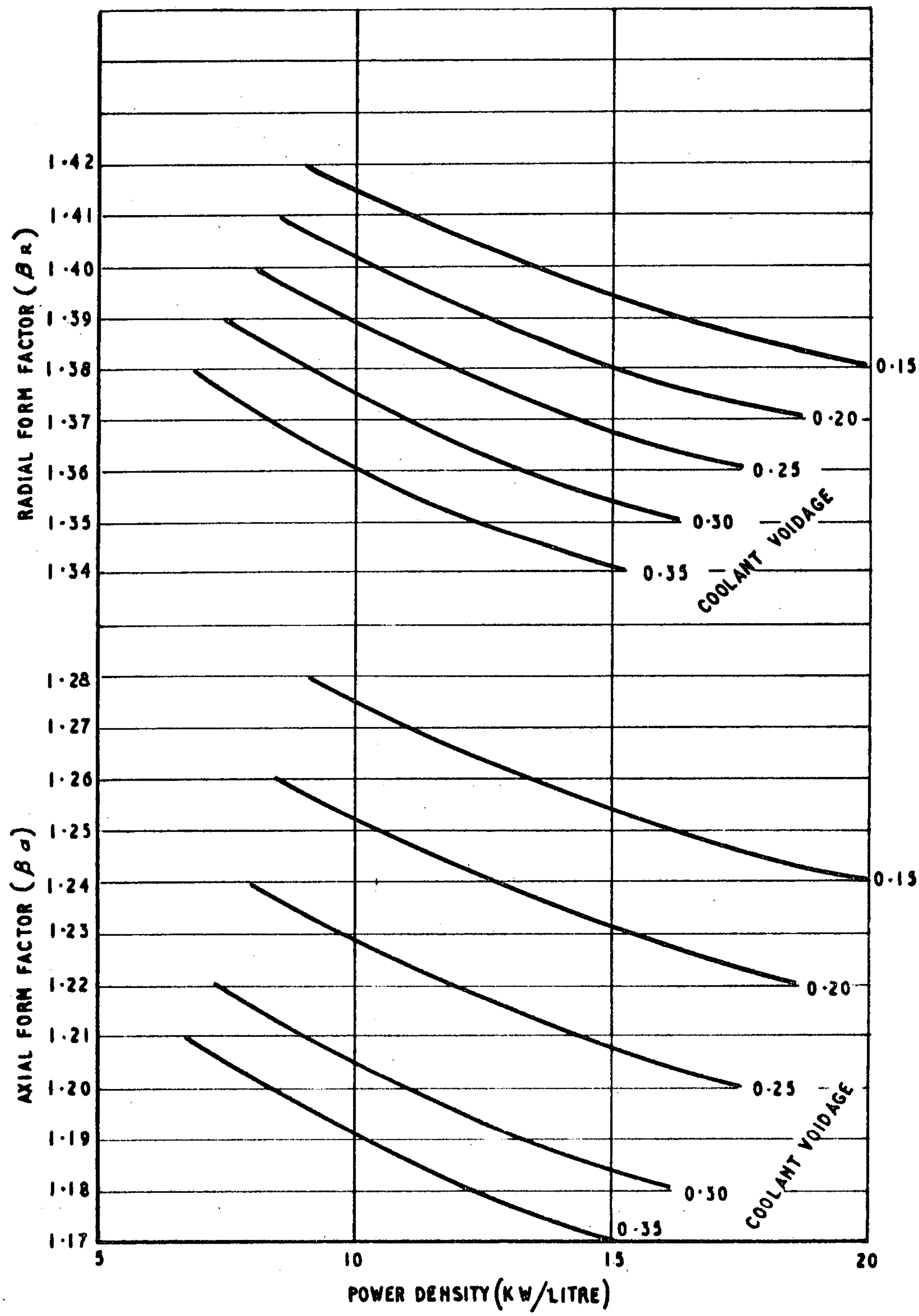


FIG. 36. "LARGE FORM FACTORS" (CORE L/D = 0·6)

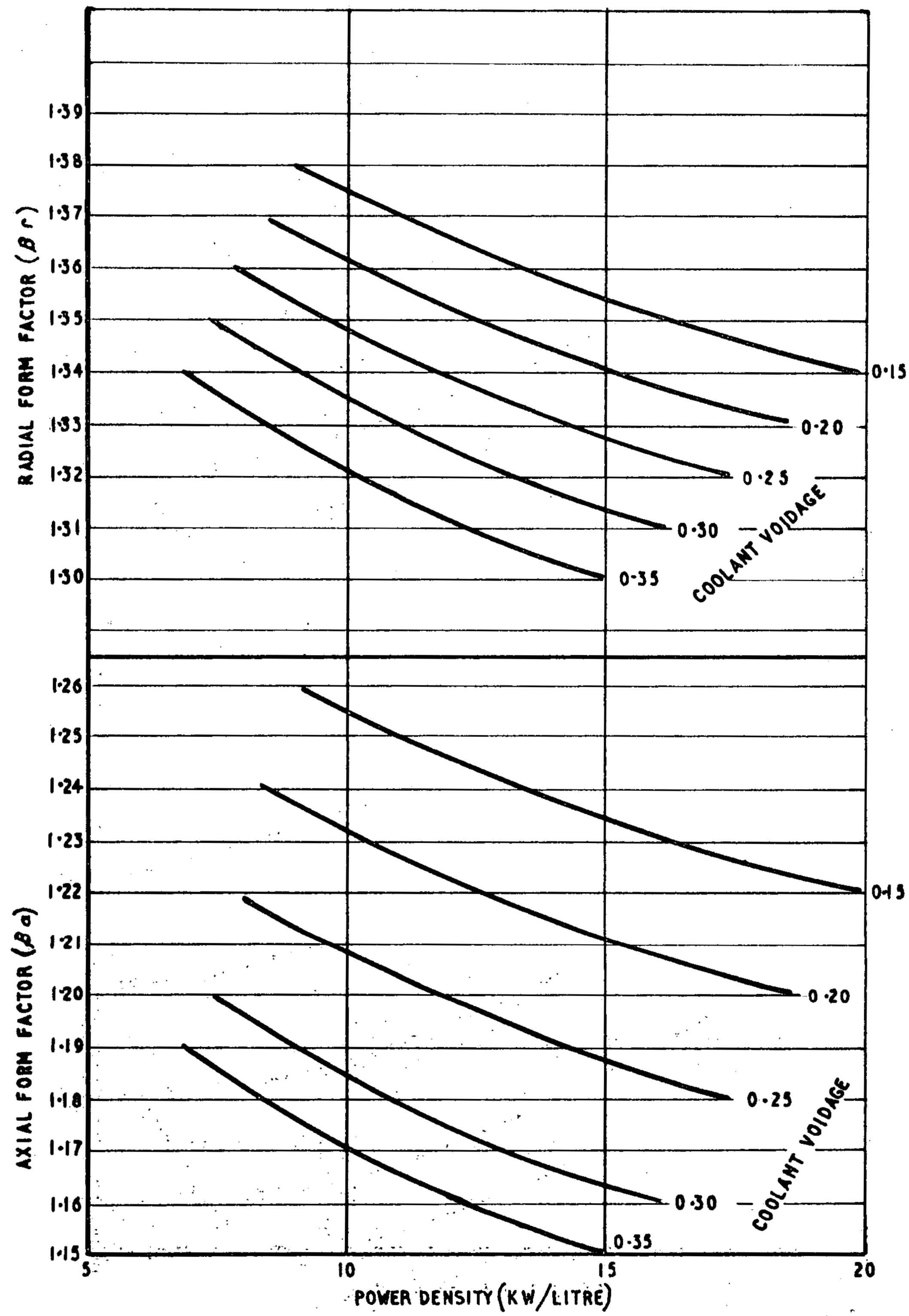


FIG.37."SMALL FORM FACTORS" (CORE L/D = 0·6)

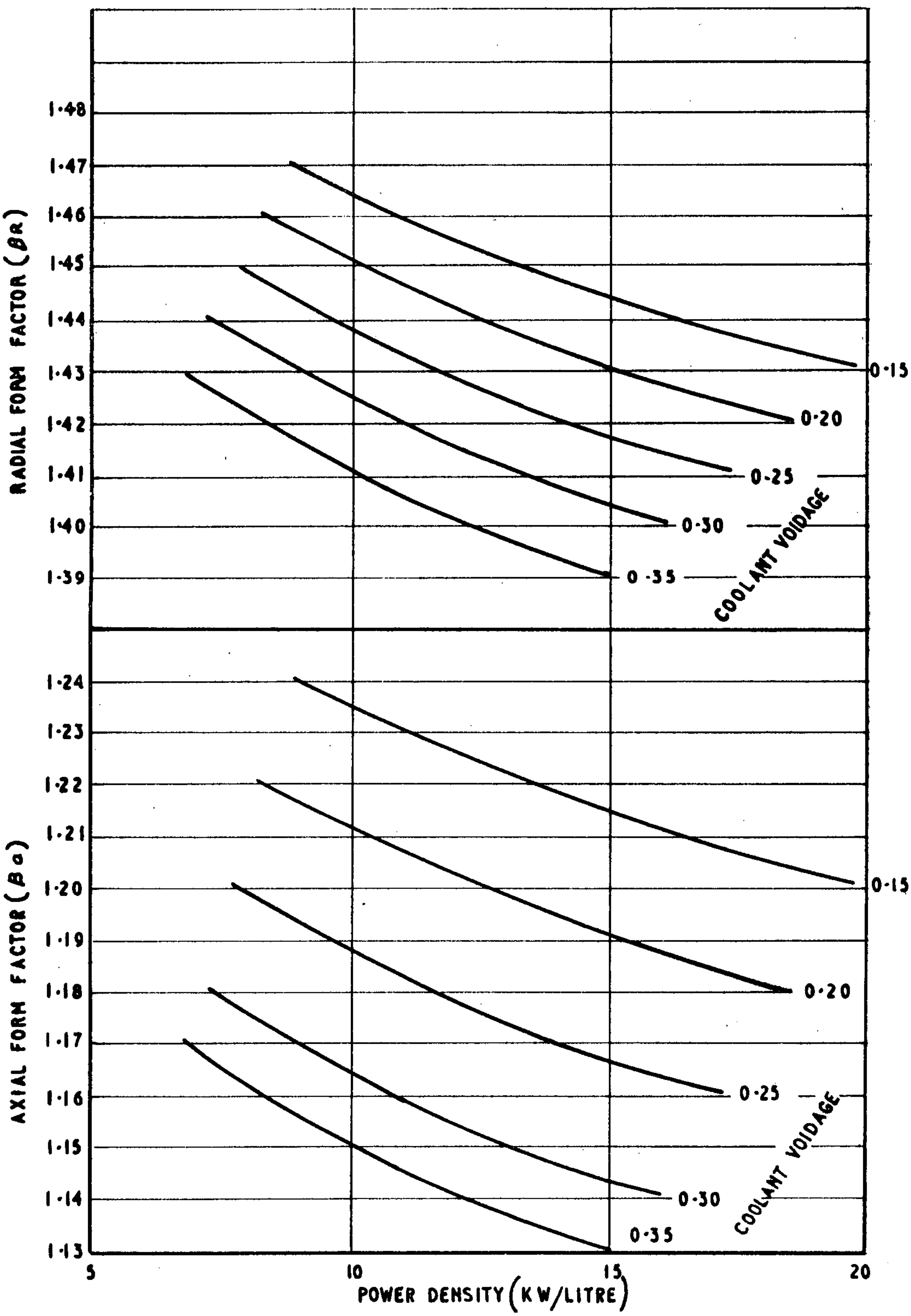


FIG.38. FORM FACTORS (CORE L/D=0·4)

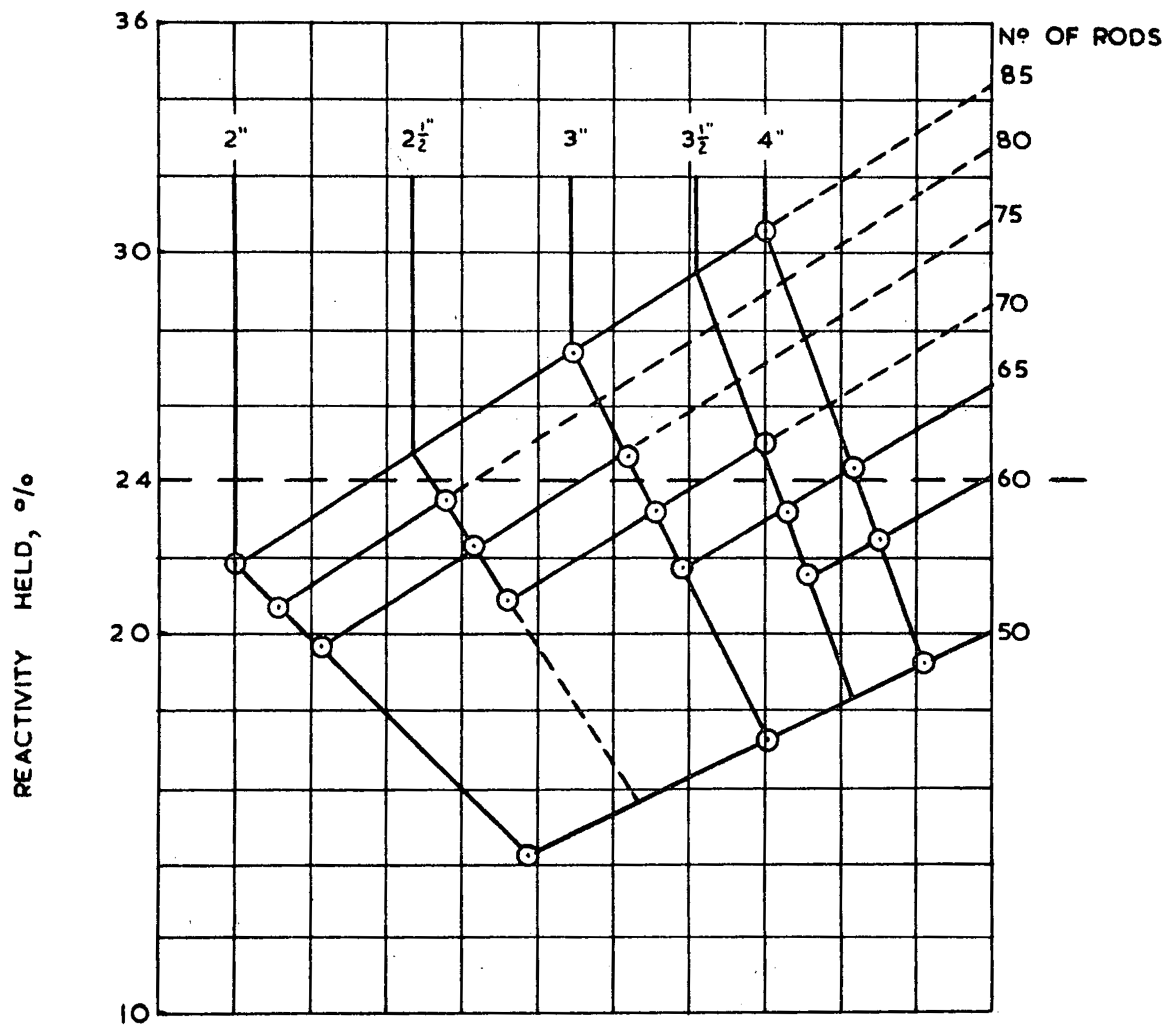


FIG. 39 REACTIVITY HELD BY VARIOUS ROD NUMBERS AND DIAMETERS

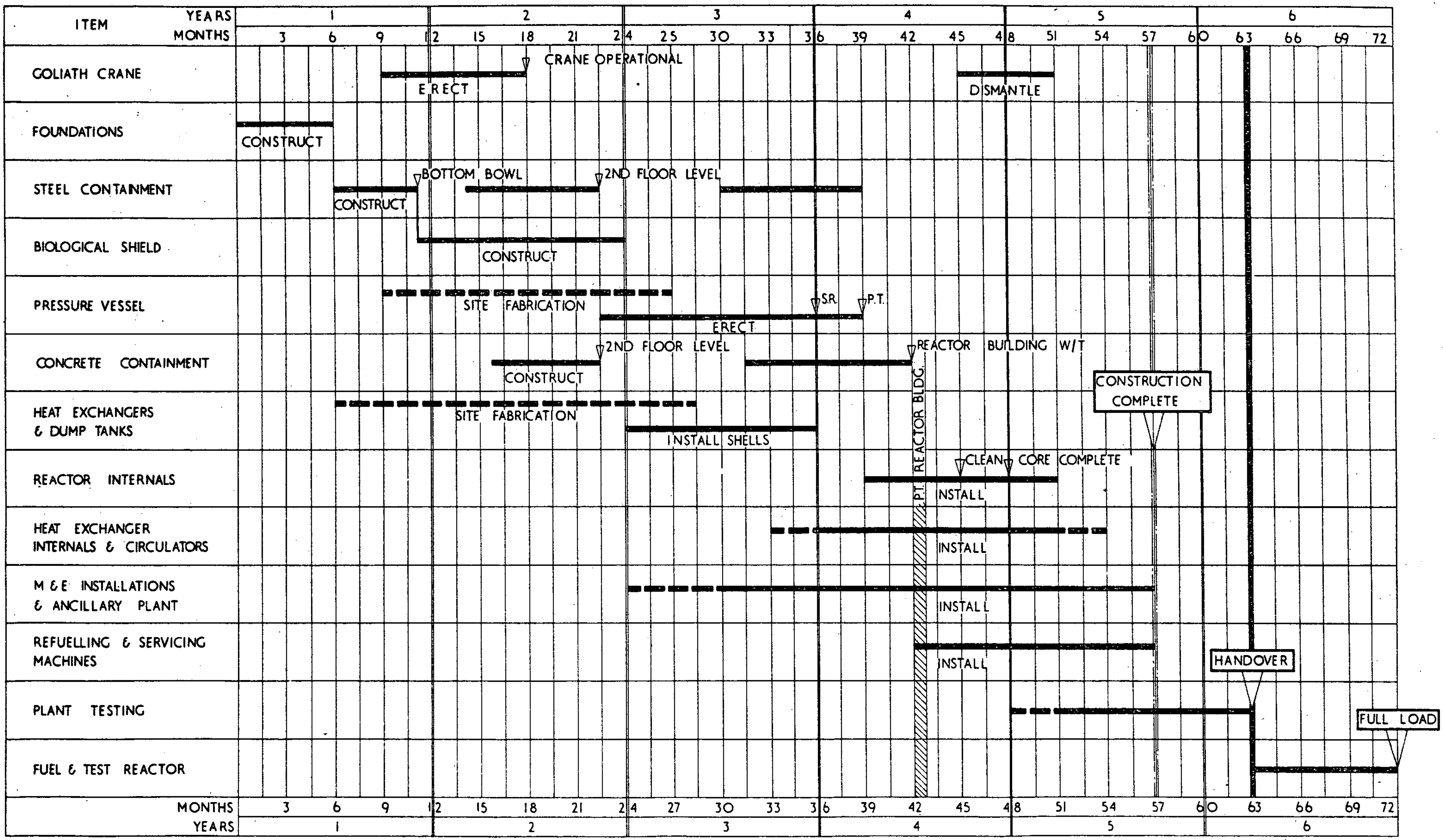


FIG. 40 CONSTRUCTION PROGRAMME