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LUCAS HEIGHTS

A SURVEY OF THERMAL AND MECHANICAL ASPECTS OF SOME H.T.G.C. REACTOR CORE DESIGNS UTILISING BERYLLIUM OXIDE MODERATION

by

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FOREWORD

This report, originally distributed within the A.A.E.C. Research Establishment, contains much of the basic engineering information from which in September 1963 the decision was made to concentrate the Commission's research programme on high temperature gas-cooled reactors into an assessment in depth of beryllium oxide moderated reactors of the pebble bed type.

The report is published now since it presents the findings from a definite phase of the research programme when effort was directed towards assessment of a range of reactor types which might lead to economic application of beryllium oxide moderation in nuclear reactors.

As publication has been made without alteration to the results or conclusions, there are some statements that would be modified or changed in emphasis in the light of subsequent experience. In general, however, the comments and predictions remain valid and have been well supported by two further years of detailed development and analysis of designs utilising the pebble bed reactor concept.

K. F. ALDER.
Director

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ABSTRACT

This report presents results of a study, from July 1962, to June 1963, by the Core Development Group (Engineering Research Division), of various reactor core configurations for a high temperature gas cooled reactor system, using fuels dispersed in unclad BeO with BeO moderation and CO_2 gas coolant. The configurations are naturally classified into two groups: one, Parallel Channel Reactors (cores with fixed internal geometry and specific coolant channels), and the other, Pebble Bed Reactors.

The major advantages and design problems of each type are discussed. Methods for solution of the problems are suggested. Although the reactor types are compared, it is impossible to recommend any one type in preference to the others, owing to the lack of reliable nuclear and cost data.

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1. INTRODUCTION AND HISTORICAL BACKGROUND

1.1 Beryllium Oxide Moderated H.T.G.C. Reactor Concept

Beryllium oxide as a neutron moderator and fuel diluent has a number of desirable characteristics which might be put to good use in a high temperature gas-cooled (H.T.G.C.) nuclear reactor system. For some years the A.A.E.C. Research Establishment has conducted a research programme to evaluate the potential of beryllium oxide in nuclear power reactors and engineering effort has been put into a survey of a wide range of reactor core types to select those in which beryllium oxide might be profitably utilised. This report discusses this work and suggests the most promising systems to be used as a basis for further, more detailed, studies.

As a starting point the desirable characteristics of beryllium oxide, and dispersion fuels using beryllium oxide as the matrix were enumerated as follows:

(i) Beryllium oxide is a good neutron moderator. Some neutron enhancement may be obtained from the (n, 2n) neutron reaction. Fissile and fertile material combined with beryllium oxide has potential as fuel for a thermal or near thermal converter or breeder system.

(ii) Beryllium oxide has good high temperature strength and excellent corrosion resistance in dry oxidising atmospheres. Hence it may be used in direct contact with suitable relatively cheap coolants such as carbon dioxide. This means that the use of high temperature metals as fuel cladding may be avoided, thus side stepping their attendant problems of coolant compatibility, poor high temperature strength, and substantial neutron absorption. Some steel structural components may be necessary but they can be preferentially cooled and kept to a minimum within the core.

(iii) Beryllium oxide used as a matrix for a dispersion of fissile and fertile material (also in oxide form) provides a fuel element material with potential for good fission product retention to temperatures of at least 1200 °C.

(iv) Dispersion fuels have further advantages in that owing to the relatively high thermal conductivity of beryllium oxide, the fission heat generated is not highly concentrated and may be more easily transmitted to the coolant without problems of high centre fuel temperatures.

(v) From the point of view of irradiation damage the dispersion material promises greater stability because of the spreading of fission product damage throughout a larger volume of material and the support of the fuel-fertile material particles in a sound matrix of beryllium oxide. Dispersion fuels also have greater thermal inertia in over-heating accident conditions because of the added heat capacity of the beryllium oxide and are therefore inherently safer.

(vi) The very high melting point of beryllium oxide and its stability at high temperatures should permit considerable over-heating of the core without causing major damage to the fuel elements or the core structure.

These characteristics of beryllium oxide moderation form the basis for the reactor core design surveys.

1.2 Historical Background and Design Basis

The study was begun in July 1962 and limited to core types consisting of, as far as possible, only fuel and fertile materials and beryllium oxide.

An important feature of the study was that the fuel was not to be clad in steel or other metallic closed cans so that advantage could be taken of the expected good fission product retention characteristics of the BeO matrix surrounding the fuel particles. Because of the nuclear performance advantages in keeping steel content to a minimum it was decided to consider reactor designs with the least possible volume of structural steel within the core.

Early in the study it became clear that high thermal stresses at the cooling surfaces of the fuel shapes were the major restriction on the design and not the peak surface or centre fuel element temperatures. The reactor core was not significantly fuel temperature limited since the heat exchanger superheater tubes and associated steam plant provided the major restriction to the coolant temperature at outlet from the reactor. Gas turbine circulators and topping power sets could be used to overcome the coolant temperature limitation. They were not considered at this stage.

Because of the brittle nature of beryllium oxide ceramics and consequent lack of confidence in their use as structural pieces and also the uncertainty of their behaviour under thermal stress, the following design principles seemed fundamental to any successful all-beryllium-oxide ceramic core design:

(i) Fuel and moderator shapes should be relatively small in section and dimensions. Temperature gradients through the core, especially in control rod regions, could superimpose high stresses in large section pieces onto the thermal stresses due to local cooling, introducing additional restrictions on the power output. Thus any core design would necessarily consist of a large number of small pieces.

A second reason for choosing small pieces was the difficulty of manufacturing large cold pressed and sintered beryllia blocks (greater than 15 cm x 15 cm x 5 cm) or sufficiently straight and long length extruded and sintered shapes (greater than 30 cm long).

(ii) The ceramic fuel elements and the core and reflector configuration must be such that a small percentage of the fuel and moderator shapes may crack during normal operation of the reactor. Since the material properties, in particular the modulus of rupture strength, of ceramics (Weil 1962) vary from specimen to specimen, often within a wide scatter band, any attempt to avoid thermal stress cracking completely would impose severe penalties on the reactor performance and its economic attractiveness.

Naturally high quality ceramic materials will be necessary with stringent quality control to remove sub-standard material in order to minimise the width of the strength scatter band.

Nevertheless, from the reactor designer's point of view, a component such as a fuel element must be such that a degree of overloading will not cause disastrous failures. This may be likened to the built-in safety against overload which is provided by metal components when yielding occurs, but not complete failure.

Hence thermal stress cracking of a proportion of the fuel elements must not cause disastrous failure, that is, break-up of the fuel elements. The designs selected therefore rest on the assumption which has yet to be adequately proved by experiment that the fuel shapes do not fall apart but maintain their integrity under limited thermal stress cracking (Crandall and Ging 1955). Thus designing to accept a proportion of cracked fuel shapes has two implications:

(a) Higher design stresses may be permitted in the fuel shapes as the allowable stress need not be restricted to the lowest limit of the material strength scatter-band.

(b) The fuel shape and the core configuration must be such that some cracking of core components will not precipitate more serious or extensive failures and will not impair the continued normal operation of the reactor or the reliable removal of the fuel elements.

(iii) The ceramic fuel or moderator shapes should be supported in such a way that structural or handling tensile stresses are minimised or better still avoided. As the strength of a highly irradiated fuel element may be considerably reduced it is undesirable to subject it to high and often ill defined tensile stress loading during removal from the core.

The reactor core configurations studied were first based on the concept of the fuel particles being uniformly dispersed through all of the BeO moderator, such that heat was generated throughout the solid material in the core. This was the logical point from which to develop the assessment since, as indicated in Section 1.1, the fully dispersed fuel concept has the advantage of lower material power densities and larger cooling surface as well as an approximation to nuclear homogeneity within the core. Comparative elastic thermal stress analysis for a number of dispersion fuel shapes (Binns 1962) showed that fuel shapes containing a matrix of cooling holes on a triangular pitch (Figure 1 a) were most favourable. The first core designs considered this fuel shape in the form of stacks of blocks pierced by coolant holes and supported in the core on a metal "cake stand" arrangement. The only metal within the core was a thin walled high temperature alloy steel lifting tube in the centre of each ring of stacks of fuelled blocks.

A second interesting fuel shape conforming to the concept of a fully dispersed fissile, fertile, and moderator mixture was the fuelled sphere randomly stacked into a core to form a pebble bed reactor. The pebble bed concept was considered because of its complete lack of structural material within the core and because of the relative ease of fuel handling.

Because of the wide difference between the structure and cooling passages in the parallel coolant channel reactor cores and the pebble bed reactor cores the study was conveniently separated into an appraisal of promising arrangements for the parallel channel reactor (P.C.R.) and promising arrangements for the pebble bed reactor (P.B.R.).

For reasons of clarity, the history of the development, the alternatives considered, and the description of the final recommendations for the P.B.R. and the P.C.R. core types are considered separately (Sections 2 and 3). This is followed by a discussion of some of the relative merits of each type of core. (Section 4).

1.3 Basis of Comparison of Reactor Cores

Where possible, the study was simplified by setting up a basis for comparison of reactor core types. Firstly, some definition of the application and size of the reactor was necessary. An output of 200 MWe was chosen as being within the expected range of interest for base load power generation. It was considered that ultimately studies of both higher and lower output reactor systems would be assessed. Secondly, it was decided that modern conventional steam power conversion equipment would be used to permit concentration of effort on the reactor core itself. (Alder and Gerrand, Unpublished).

A number of variables, such as changes in primary circuit pressure, were expected to affect all of the core designs in a similar manner. Values for these were chosen, based on the best information currently available, so that the comparison studies could be restricted to those variables of greatest importance in assessing the relative merits of the core designs. In the final evaluation of course, the validity of these simplifications must be considered with the conclusions drawn.

Table 1 gives the basic performance requirements which were used for all core designs in the study. Physical and mechanical property data for fuelled and unfuelled beryllium oxide shapes were taken from the A.A.E.C.'s Beryllium Oxide Data Manual (Unpublished) with adjustments for irradiation effects (Hanna et al. 1963, Hickman and Pryor 1964). For the reactor fuel temperatures under consideration it was assumed that elastic thermal stress analysis would be adequate for the comparison study. Since creep in beryllium oxide is definitely a factor at material temperatures of 1000 °C and above and possibly as low as 800 °C, it was assumed to have a similar effect in all designs, provided fuel temperatures were of the same order, and therefore would not be a major factor in the assessment of the relative merits of the core designs.

2. PEBBLE BED REACTOR CORE STUDY

2.1 Introduction

The pebble bed reactor (P.B.R.) type was considered initially because it appeared to be one of the simplest core configurations which could be constructed without structural or other unfuelled moderator shapes within the core, and because its novel features could result in economies in power cost not possible in the parallel channel reactor (P.C.R.) core designs. These features are primarily the avoidance of a fixed core structure and the freedom gained from straight-forward on-load fuel changing.

Initially a survey was made of published information on P.B.R. designs and thermal and mechanical performance of randomly packed beds of spheres (Oak Ridge National Laboratories, Sanderson and Porter 1960, the Brown Boveri-Krupp organisation in Germany, and some early work by Denton in the United Kingdom). Surveys were made to correlate the information available on heat transfer, fluid friction, and on coolant voidage in randomly packed beds, the types of ball packing that occur in the bed, and on ball flow and ball handling methods.

For P.B.R. reactor designs conforming to the basic performance specification (Section 1.3), three bed arrangements were considered with the coolant flowing axially upwards, axially downwards, and radially outwards (Figure 2). Two types of radial outwards flow cores were studied, one a cylindrical shaped core with a central coolant entry hole, and the second a trapezoidal shaped core where the flow area is maintained closely constant in a radial direction.

2.2 Pebble Bed Reactor Thermal Performance

A survey was made of heat transfer and fluid friction loss correlations and the following were recommended for the purposes of the study (Sanderson and Porter 1960):

$$\mathrm{St} = 0.50\ \mathrm{Re}^{-0.30}\ \mathrm{Pr}^{-0.66} \tag{1}$$

$$f = 61.3\ \mathrm{Re}^{0.27}\ . \tag{2}$$

The heat transfer coefficient derived from Equation 1 has been reduced by 15 per cent. from the average expected value to cover the confidence limits for the correlation.

For the core thermal output of interest the performance of the pebble bed reactor using BeO based fuelled balls was limited by the cooling thermal stresses on the surface of the ball, and in all cases the peak surface temperatures were within the estimated maximum surface temperature limits (1100 - 1200°C) of the material in the reactor environment. The ball centre temperature was not significantly higher than the surface temperature (approx. 40°C maximum difference). Table 2 compares the major thermal performance parameters of the four core designs, including alternative designs incorporating the shell fuel ball concept (Section 2.3.3). However, the shell fuel ball concept was not included in the comparison study since, if feasible, it would provide similar improvements to each of the four P.B.R. arrangements.

The ball diameter necessary to achieve the desired average power density of 15 watts/cm^3 in the core was estimated to be 2.2 cm. This ball size causes relatively high pressure loss across the core and would definitely introduce ball levitation in an upflow core at average power densities much less than half the desired value of 15 watts/cm^3 (see Figure 3 and Table 2). Ball levitation occurs firstly as a spinning and then lifting of the balls at the top of the bed. This is not permissible in a stable core configuration. In fact the pressure drop in the upwards flow core has been limited to 80 per cent. of the drop which would cause levitation.

Thus, to achieve a power density of 15 watts/cm^3, it is necessary to have a bed configuration where the levitation restriction does not apply. The axial downflow cooled bed or the radial flow bed designs fulfil this requirement. In the radial flow designs the core pressure loss and hence the pumping power is less than that in the axial downflow core. This pumping power advantage would be unlikely to result in great changes in capital or running charges and could well be outweighed by the penalty in nuclear performance of the large central coolant entry hole. The fuel surface temperatures in the downflow and radial flow cores are comparable. Thus, from a thermal performance point of view, there is little to commend the radial flow core designs in comparison with the axial downflow design, except the lower pumping power requirements. This may be compensated to a large degree by further flattening the shape of the downflow core.

2.3 Thermal Stress Analysis

As already mentioned the surface thermal stresses in the fuel balls are the major restrictions on the ball size for core average power densities of some 15 watts/cm³. Careful appraisal of their evaluation has therefore been necessary.

2.3.1 Ball stress history and effects of material property variations

The thermal stress history of a fuel ball is complicated. The history of each ball depends on its position and path through the bed, the number of passes through the bed, and hence the irradiation dose and the necessity to anneal the cumulative irradiation damage from previous passes. It is difficult to predict the point in the history of the fuel balls when failure is most likely to occur. For this survey the weakest point, and therefore the design point, has been taken as the substantially irradiated ball where the material properties have been reduced in accordance with predictions of Hickman and Pryor (1964).

Since the thermal stress resistance of the fuel element material,

$$\frac{\sigma(1-\mu)k}{E\alpha},$$

strongly decreases with increasing temperature (primarily due to the reduction in k with increasing temperature), the point where the fuel element is most likely to fail is not at the centre of the core where the highest power density occurs, but at a point nearer to the hotter coolant outlet end of the core (Figure 10). At this point, the required thermal stress resistance of the fuel element approaches closest to the thermal stress resistance of the material. A numerical method has been used to determine this point of failure. However, it was found, for the cases under consideration with a 1.84 overall form factor, that the value for the average power density in the core could be used, provided that the thermal stress resistance was evaluated at the maximum fuel element surface temperature (approximately 1000°C). This approximation is estimated to be within ± 20 per cent of the accurately determined value and well within the accuracy of the data. It has therefore been used for the survey calculations (see Appendix).

The relation used for the calculation of the allowable ball diameter was:

$$d = \sqrt{\frac{\sigma(1-\mu)k}{E\alpha}\,\frac{60}{q_b}\,\frac{1}{F}},$$

for a uniformly cooled elastic sphere, but including the uncertainty factor (F) discussed in the Appendix.

2.3.2 Surface coating on the ball

In a uniform dispersion of fuel particles in a matrix of beryllium oxide a number of particles will lie on the surfaces of the fabricated balls. These particles will be subject directly to abrasion which will add to primary circuit contamination problems caused by fission product release. If this release and the particle abrasion problem is to be avoided a fission product retentive wear resistant coating is necessary. This could be a layer of beryllium oxide or other suitable ceramic material surrounding the ball.

Preliminary calculations have been carried out to determine the increase in elastic thermal stress due to the coating on the assumption that the coating behaves elastically and has the same material properties as the fuel ball material. Coating thicknesses of 3 per cent. of the ball outside radius increase the thermal stress in the coated surface of a fully dispersed fuel ball by 1.09 times the stress in the uncoated ball. On a 2.54 cm diameter ball this amounts to a coating thickness of 0.38 cm which should be sufficient for surface particle fission product retention. Provided coating thicknesses are less than 3 per cent. of ball radius the increase in stress does not substantially affect ball diameter.

2.3.3 Shell fuelled ball concept

Since thermal stress is the prime limitation in this study, any changes to reduce thermal stress and hence increase ball size may have substantial value in improving performance in the upflow core and reducing pressure drops in the downward and radial flow cores, not to mention the réduction in the number of balls required.

One such improvement is the shell fuelled ball concept where, in effect, the fuel particles are moved towards the cooled surface of the ball introducing an unfuelled moderator core in the centre of the ball and a substantially richer fuel dispersion in the outer shell. This permits higher heat output from the same ball diameter for the same surface stress at the cooled surface, or more desirably an increase in ball size for the same power density in the ball. Figure 4 illustrates this improvement (Binns 1964). As an example, for a shell inside to outside radius ratio r_1/r_3 = 0.8 the heat output from the uncoated shell fuel ball would be 2.3 times that from an equal diameter fully dispersed fuelled ball, assuming that the core and shell are integrally bonded.

Manufacture of the shell fuel ball has some unresolved difficulties and it would almost certainly involve greatet cost. It is not known how manufacturing tolerances and variations in the allowable strength of the surface material would affect the performance of the shell fuelled ball except to say that the actual improvement would not be as high as predicted by the curves of Figure 4. Because of these uncertainties the improvements derived from the shell fuel ball design have not been included in the comparison study. Table 2 does, however, give the effect on P.B.R. performance of using a 0.25 cm thick shell in the upflow and downflow designs. The most notable improvement is in the increased power density obtainable from the upflow P.B.R.

2.4 Nuclear Performance

As nuclear survey calculations have been confined to generalised homogeneous reactor systems, only qualitative comments may be made on the effects of the P.B.R. design on the nuclear performance of the reactor. The P.B.R. core with fully dispersed fuel balls comes much closer to the homogeneous core model used in the nuclear survey calculations than most of the P.C.R. core designs considered. Thus the results of the nuclear calculations in the survey may be taken as a good first approximation to the conditions in the P.B.R.

A comparison between the different P.B.R. types (Figure 2) indicates that they are closely similar in nuclear performance except for the deleterious effects of the large central coolant entry hole in the radial coolant flow designs. The importance of this effect is not known. The effect of the hole also depends on the material used to form it. If steel were used then a substantial loss in reactivity would almost certainly occur because of the quantities required. If reflector material such as graphite or beryllium oxide were used, there could well be a peaking of the power distribution at the core inner boundary, adding to the probable maximum heat ratings already occurring in the central core region.

For the survey calculations the radial and axial form factors given in Table 1 have been taken without allowance for the effects of the central hole in the radial flow designs.

2.5 Mechanical Design Features

2.5.1 Reflector materials and design

The choice of reflector materials for the pebble bed reactor has been governed by the following requirements:

(i) Ability to withstand doses of the order of 10^{22} nvt (>1 MeV) so that the reflector need not be replaced during the design life of the station. This implies good dimensional stability and maintenance of strength throughout the life of the station.

(ii) Compatibility with the coolant at both inlet and outlet temperatures.

(iii) Ability to withstand wear and abrasion, in the case of side and bottom reflectors, due to the passage of the balls through the bed.

(iv) Ease and accuracy of fabrication of block shapes suitable for the reflector.

(v) Ability of the material to withstand environmental conditions such as thermal cycling.

(vi) Cost of the material selected.

With these points in mind graphite is suggested for the cooler parts of the reflector (up to temperatures of approximately 450 °C because of the CO_2 – graphite reaction) for reasons of its lower cost, ease of manufacture, and stability under irradiation. The reflectors bounding the hot outlet region of the core must be of beryllium oxide to be compatible with the carbon dioxide coolant in the 700 – 800 °C range.

Extrapolated irradiation damage data for fine grained ($2 - 3\,\mu$) beryllium oxide blocks at doses expected at the core–reflector interface indicate that the outlet reflector material should last the design life of the station without replacement, provided the beryllium oxide is maintained above 700 °C (Hickman and Pryor 1964). If beryllium oxide were used in the cooler regions of the reflector, micro–cracking and attendant loss of strength would almost certainly occur.

Magnesium oxide or aluminium oxide blocks may also be of use as reflector components where high temperature compatibility and good irradiation stability are required. Magnesium oxide is known to have substantially better irradiation damage characteristics than beryllium oxide.

Since it has not been possible to guarantee that the reflectors will not need replacement at some stage during the life of the station, consideration has had to be given to replacement once during this period. The replacement of part or all of the reflectors in a pebble bed reactor poses some difficult engineering problems. One method which appears feasible is to use remote mechanical manipulators inserted through the fuel outlet port. This procedure is practicable but would involve substantial cost and take considerable time. It would appear from the study made that the reflector replacement in a P.B.R., if necessary, is a major weakness of this concept.

The passage of coolant through the reflectors, and in the case of the bottom reflector the movement of the balls in the conical centre of this reflector, appears to offer no major problems provided the ball diameter is of the order of one inch or larger. The coolant passages in the reflector for balls less than one inch in diameter become increasingly small, forcing the manufacture of larger numbers of smaller or more elaborate reflector blocks. Reflector wear, particularly on the bottom reflector, due to the passage of the balls across it, is an unknown which must be checked experimentally.

2.5.2 Core support methods

The problems of core support differ for each of the P.B.R. types (Figure 2). Support under conditions of high temperature (coolant outlet conditions) poses the most difficult problem since uncooled steel grid structures at these temperatures are considered unsuitable due to creep problems and corrosion in carbon dioxide.

In the case of the upflow core the weight of the core may be carried on a low–alloy steel grid structure maintained at coolant inlet temperature by the coolant entering the core. The top reflector must however be supported in the hot outlet gas region. This presents some problems in the use of steel hangers to support the beryllium oxide reflector. The best solution appears to be a steel hanger structure where the structural members are preferentially cooled by inlet temperature gas. The more difficult feature of this arrangement is considered to be the method of attaching the beryllium oxide reflector to the steel hangers so that they could be removed and replaced by remote handling if necessary.

The support of the core in the radial flow designs is simplified since the coolant does not pass through the lower reflector. Thus the reflector itself acts as the thermal barrier between the core and the low–alloy steel grid structure. The support for the side beryllium oxide reflectors in the trapezoidal core design looks feasible in the first instance as the small height of this reflector may be supported by a cooled and insulated cage consisting of hollow vertical tubes carrying the core side loads in bending.

The axial downflow core support structure must cope with the effects of the high gas outlet temperatures, which could reach some 900 – 1000°C prior to mixing in the outlet plenum region, as well as the combined loading from the weight of the core and its pressure drop. In the reference design this load amounts to some 44 p.s.i. over the lower reflector area. This set of circumstances amounts to a difficult problem which has yet to be solved.

2.5.3 Control

Control rod and safety rod requirements in any of these reactor systems will vary widely depending on the fuel cycle and also the composition selected for a given fuel cycle. Without data on where and how many control rods would be required, the study considered first the feasibility of rods positioned in the reflector or near the reflector and then rods well within the core.

The simplest and most practical design in the axial flow cores is achieved if the control rods are placed in channels in the side reflectors where they may be positively and reliably guided. For control rods positioned well within the core boundary it was concluded that a robust steel guide tube was necessary. It would need to be located at top and bottom of the core and thermally insulated from the bed with concentric BeO sleeves. The steel tube and the control rod itself would be cooled by inlet temperature coolant bled off the main stream. This control rod design necessitates some steel remaining permanently within the core. A further complication is the need to remove the control rod guide tubes for inspection and replacement of the BeO insulating sleeves, at intervals of three to five years in the case of the reference design.

Clearly every effort should be made in the design of the axial flow pebble bed reactors to permit all control rods to be located in the reflector region. The more probable alternative of control rods within the core constitutes a further difficulty in the design of axial-flow pebble bed reactors.

In the radial flow core design the control rods are more easily locatable within the core by placing them around the central coolant inlet duct. In this arrangement they may be used to form part of the inlet duct itself. The fact that the central rods are at inlet temperature is a definite advantage in the radial flow designs.

Summarising, the introduction of control rods into the P.B.R. designs is not a significant problem if adequate control is obtainable from reflector rods only. In the case of axial flow cores there are a number of problems associated with insertion of rods into the body of the core. The radial flow designs can cope with rods within the core provided that they form part of the central inlet channel structure.

2.6 Fuel Handling and Management

Probably the greatest single advantage of the P.B.R. core designs is the freedom obtainable from continuous fuel charging and discharging. Thus all fuel may reach an equal burnup, repeated discharging may be carried out with inspection at a number of intervals during the life of the ball, and significant power flattening and reactivity lifetime advantages are possible.

Fuel balls may be loaded at any one of a number of points at the top of the bed and discharged from one or possibly more outlet ports at the bottom of the bed. Ball handling machinery at outlet from the core has been developed overseas and should not offer any major obstacles to the design.

The equipment for handling, inspecting, storing, and returning balls to the reactor does not appear to present major obstacles. The cost of such equipment is not expected to be unreasonable and could well be significantly lower than that required in the parallel flow core designs considered in Section 3.

2.7 Development Potential of Pebble Bed Reactor Concept

2.7.1 Improvements in thermal performance

As the P.B.R. core design data, determined for the conditions specified in the basic reference design, are for comparison purposes only, there is scope for improvement and optimisation of any of the designs. Considerable reductions in core pumping power can be achieved in the axial flow designs by further flattening the core (Holy, Unpublished). Further reductions in the core pressure drop, and hence the pumping power, may be made by decreasing the coolant inlet temperature or increasing the coolant outlet temperature and by increasing the primary circuit pressure (Fraas et al. 1961).

2.7.2 Ball diameter improvement

Since the designs are thermal stress limited, ball diameter is a direct function of the material properties and the design factors such as power form factor. It is expected that when the ball stress history is analysed in detail some increase in ball size should be obtained since more realistic estimates may be made of the hot spot and power distribution uncertainty factors (see Appendix). Improvements in the material properties will of course give corresponding improvements in ball size. However the feature with by far the greatest development potential from the design point of view is the shell fuel ball concept, where substantial increases in ball size could be realised.

2.7.3 Improvements resulting from on-load fuel changing

Substantial scope for development of the P.B.R. comes from the ease of fuel changing while the reactor is on load, and the resultant freedom of fuel management procedures.

Within the core, the ball flow rate can be adjusted to permit many passes of a given ball so that all balls achieve a given burnup. A mixture of new and partly consumed balls may be suitably loaded into the bed after the initial start-up period, so that radial power flattening is achieved. Multiple ball pass arrangements also improve the uniformity of core burnup along the core length. Outside the core, balls which make a number of passes through the bed may be inspected at intervals during their lifetime and even treated by annealing to extend their life or recover mechanical or physical properties which have deteriorated under irradiation.

Another feature of the pebble bed design is that mixtures of fuelled and unfuelled balls may be used. If, as currently seems likely, high reprocessing charges are incurred for reprocessing all of the moderator with the core, savings in fuel cycle cost may be achieved using mixed fuelled and unfuelled balls. However while ball size is of the order of one inch or less in diameter, mixed beds are unattractive because the ball size must be decreased to maintain the same core power density.

2.8 Recommendations Concerning Pebble Bed Reactors

From the study of the four core types, the axial downflow core (Figure 2b) has been selected as the most promising P.B.R. system conforming to the basic specification set down in Section 1 (Figure 5). It was chosen because:

(i) There is no ball levitation limit restricting the power density from the core. This is the prime reason for discarding the upflow design since the core power densities attainable in that design are expected to have an adverse effect on the economics of the system.

(ii) Both coolant flow and ball flow are in the same direction. This permits the bed height to be varied without by-passing coolant as would occur in the radial flow designs and ensures that all balls pass through the hot region of the core; (for beryllium oxide beneficial irradiation damage annealing takes place in the higher temperature regions of the core (Hickman and Pryor 1964).

(iii) The coolant voidage, already high because of the inherent nature of a randomly packed bed, is not further increased by a central coolant entry passage in the core. Power distribution form factors may also be adversely affected by the central coolant passage.

(iv) In the axial flow core designs better control is available over the form of the radial power distribution by suitable ball loading procedures.

(v) The reflector design problems, although major problems, are comparable in each of the designs, with the exception that for control rod location within the core the radial flow designs are more favourable.

The importance of the design problems in the axial flow P.B.R. should not be minimised. The current major problems are considered to be:

(i) The suitable design and choice of materials for the reflector structure. Every effort must be made to design the reflector to last the life of the reactor since the alternative of reflector replacement is considered to be a serious weakness of the concept.

(ii) The installation of control rods and guide tubes within the core if they are found necessary. The design of reliable rods within the core is considered feasible but they will raise a number of problems and probably serious penalties in core performance and ball flow.

(iii) Designs for the core support structure to withstand the coolant at outlet temperature for the life of the station look feasible provided refractory materials can be found to stand the temperature and thermal cycling conditions envisaged. Experimental proof of the structure will be needed before the problem can be considered solved.

(iv) Wear and contact point loads between balls and between balls and the core support structure are important unknowns affecting the life of the balls and the reflectors.

3. PARALLEL CHANNEL REACTOR STUDY

3.1 Introduction

As discussed in Section 1 the parallel channel reactor (P.C.R.) core studies were first based on designs approaching a full dispersion of the fissile and fertile material in all of the beryllium oxide moderator. Difficulties in the structural soundness of the core and in fuel element removal led to designs incorporating steel structural components and then beryllium oxide structural components in an effort to minimise the quantities of steel involved.

Since thermal stress due to cooling of the fuel shape was a major design restriction (Section 1.2) and beryllium oxide was known to behave elastically up to temperatures of at least 800 °C, possible fuel shapes were evaluated using elastic stress theory for survey purposes. Fuel shapes considered in some detail are shown in Figure 1. The most promising shapes were found in each case to depend significantly on the type of core structure and on the proportion of beryllium oxide not containing fuel or fertile materials.

To expedite the survey the P.C.R. concept was further defined in the light of the characteristics of beryllium oxide moderation and the basis for the design discussed in Section 1, as follows:

(i) Off-load fuel charging and discharging was assumed because of the long burnup and fuel life envisaged in the reactor concept. On-load fuel changing for the P.C.R. core types studied was considered impracticable.

(ii) Estimates of beryllium oxide moderator life within the reactor core (Arthur, Unpublished) require that all core components be removable for inspection and replacement.

(iii) The size of the reactor core precludes the possibility of fuel changing by removal and replacement of the whole core in one unit. Thus the fuel shapes must be assembled into units, termed the fuel elements, which are then stacked side by side, constituting the reactor core. Each fuel element must be a separately removable entity such that inserting or removing it from the core is a simple and reliable operation not affecting the fuel element itself or its neighbours.

Throughout the study of P.C.R. cores considerable use has been made of the published work on reactors of a similar type, in particular the EURATOM reactor experiment DRAGON, and the General Atomic reactor experiments, the H.T.G.R. at Peachbottom and the Beryllium Oxide Reactor EBOR.

3.2 Core Configuration and Mechanical Design Features

A summary is now given of the core arrangement envisaged as the most suitable of the designs considered (Scott-Rogers, Unpublished). This is followed by discussion of the thermal performance of the fuel shapes (Section 3.3) .

3.2.1 The fuel element

The most promising fuel shapes proposed for the P.C.R. cores are long and of relatively small cross section, and made by extrusion techniques. A number of ways to assemble these into stacks forming larger and more manageable units termed fuel elements, were considered. The most promising and most reliable was regarded as a box-like structure, hexagonal in section, surrounding and supporting individual fuel shapes stacked side by side and end to end. (Figure 6). Initially study was concentrated on fuel element boxes made of high temperature steel or nickel alloys because of the small volume of core occupied by such a box. Both sheet steel and cage-like steel structures were considered but in all cases studied it was not found possible to make a sufficiently rigid and reliable box with less than some 2 per cent. by volume of steel within the core. The box sections used as a basis for this work were hexagonal in section and 30 cm across flats. Preliminary analysis of the effects that steels dispersed in the core would have on the nuclear performance of the U235 - Th fuel cycle (Lawrence private communication) indicated that steel substantially limited the breeding potential of the system and that every effort should be made to minimise the quantities of steel necessary in the core. Similar effects are expected in the other fuel cycles under consideration with the possible exception of the plutonium fuel cycles.

Effort was then directed towards the formulation of a beryllium oxide ceramic box structure. By restricting the dimensions across the flats it was considered feasible to construct ceramic pieces as shown in Figure 6. The "boomerang" links illustrated in fuel element type (A) would be formed by cold pressing. The shapes dove-tail together to form a rigid box structure which is tied into one unit by three tubular steel tie rods. The rods are adequate in section to lift the fuel element for handling purposes and ensure that, even though some of the "boomerang" shapes might fracture during operation, the fuel element will remain intact for removal.

The alternative of a ceramic box consisting of a full hexagonal extruded section (type (B)) appears to be the most promising box type because of its rigidity of section and ease of assembly. Fabrication of this section by extrusion should be feasible. Three steel tie tubes are however still necessary for lifting purposes and also to help maintain the integrity of the fuel element if some fractures in the ceramic sections do occur.

In each of the above cases the volume percentage of steel in the core has been reduced to 0.25 - 0.30 per cent. This volume is based primarily on the fuel element lifting requirements and therefore appears to be the minimum possible steel content for this type of P.C.R. core. When the reactor is in operation the tie tubes are preferentially cooled by streams of coolant at core inlet temperature.

The box type ceramic fuel elements may contain any one of a number of fuel shapes; the final box dimensions must however be determined in relation to the fuel shape to be used.

The end fittings for the fuel elements are of high temperature steel or nickel alloy (Figure 7) chosen to operate in the reactor environment under conditions of low stress for the life of the fuel element (up to some five years). Sealing to avoid uncontrolled coolant by-pass between fuel elements is effected at the inlet end of the core. The fuel element lower end fittings locate in the main core support grid and the top end fittings are held butted together by a clamping arrangement (see Section 3.2.2). A small gap is maintained between elements for thermal expansion and irradiation damage growth. Both end reflectors are included in the fuel element.

3.2.2 The core configuration

Because of the relatively small dimensions of coolant passages attainable in the proposed fuel shapes, the core pressure drop for the reference design conditions is sufficient to lift the ceramic box type fuel elements if the full frontal area of each element has the core pressure drop across it. Design of a core which does not lift is still quite feasible by using a relatively smaller inlet frontal area to each fuel element and permitting coolant at or close to the outlet pressure to surround the element.

It is, however, difficult to stop the fuel shapes themselves from moving and rattling within the fuel element. (Because of the dimensional tolerances and allowance for dimensional changes under irradiation the fuel shapes cannot be rigidly clamped together). To alleviate the problem some changes in the basic design parameters can be made such as core flattening and an increase in primary circuit pressure. Nevertheless, the problem of rattling of the fuel shapes is likely to introduce an important limit in an upflow design. Probably the surest way to deal with the problem would be to reverse the coolant flow. Schematic arrangements of both upflow and downflow P.C.R. cores are illustrated in Figures 8 and 9; limited consideration of each one has not revealed any insurmountable difficulties in developing a suitable design, but the downflow arrangement presents problems of core support.

Location of the top ends of the fuel elements is provided in both the upflow and the downflow arrangements by a set of pneumatic clamping devices surrounding the upper periphery of the core and positioned in the inlet coolant gas stream. The clamps act radially inwards holding the fuel element end caps tightly packed in an hexagonal configuration. The use of the core pressure drop to maintain the clamping force was considered but difficulties were encountered in centring the core and also in maintaining an adequate clamping force under part load conditions. An external pressure supply was therefore considered necessary to supplement the clamping effect provided by the core pressure drop. If the supply failed the core pressure drop would still maintain some clamping load on the core.

3.2.3 Reflector materials and design

The design and selection of the materials for the side reflectors is not regarded as a difficulty since they would be made in similar overall dimensions to the fuel element and hence be removable as fuel elements. Thus beryllium oxide is suitable for the side reflectors adjacent to the core since it can be replaced at intervals during the life of the reactor plant. The end reflectors are simply made as part of the fuel elements. Beryllium oxide would again serve the purpose but graphite could possibly be used at the inlet end of the core to reduce costs.

3.2.4 Control

As with the P.B.R. study the control rod and safety rod requirements will vary widely, depending on the fuel cycle and the composition selected for that cycle. Without specific information on their location, assessment was made of a rod which could be located within any fuel element in the core or in the reflector region. The arrangement is essentially a high temperature steel or nickel alloy guide tube positioned centrally in a fuel element. The annular space remaining in the fuel box may be filled with fuel shapes, as for the complete fuel elements. The steel control rod guide tube and the control rod inside it would be preferentially cooled with inlet gas.

3.2.5 Dimensional tolerances on fuel and moderator shapes

Because the components of the fuel elements consist of relatively small pieces, each of which has tolerances on its dimensions to accommodate variations in manufacture and to ensure reliable assembly, cumulative tolerances can have an adverse effect on the core performance and fuel element rigidity. Nevertheless, to keep fabrication costs down, the tolerances specified should be no more than really necessary for the design.

For beryllium oxide shapes it was decided that as far as possible the shapes would be used in their sintered condition to tolerances of ± 2 per cent. on cross-section dimensions to avoid costly final grinding operations. The permissible straightness tolerance on extruded shapes was set at 0.006 cm/cm of extruded length. (Clare and Wright private communication).

The direct effect of these tolerances is to produce a tolerance on the possible voidage within the core. The study was limited to considering extreme cases only, and in the worst and most unlikely extreme when all components were on their lowest tolerance limits, the core coolant voidage could vary by as much as ± 2.5 per cent. and the total voidage by ± 6 per cent. for the central tube design. Variations of this magnitude would have an important bearing on both the thermal and the nuclear performance of the core. The figures do, however, give a pessimistic view of the voidage scatter which would be much closer to the datum value, provided the dimensions of the fabricated components were well distributed within their respective dimension tolerance bands.

From this appraisal, for the tolerances given, it was concluded that the dimensional tolerances of the fuel and moderator shapes have an important bearing on both the nuclear and thermal performance of the P.C.R. core. Efforts should be made to improve the dimensional tolerances, particularly the straightness of extruded shapes, beyond those used in this assessment.

3.3 Assessment of Fuel Shape and Arrangement

3.3.1 Restrictions imposed on the choice of fuel shape dimensions

The determination of the fuel shape dimensions for any of the fuel shapes of Figure 1 (b), (c), (d), and (e) depended on a number of factors summarised as follows:

(i) Fabrication restrictions were put on the minimum wall thickness of tubes that could be economically extruded on a production basis. This limited the ratio of external to internal tube diameter to approximately 1.25 for the diameter range of interest. Fabrication restrictions were also applied to the wall thickness of the beryllium oxide shapes forming the fuel element box structure (2.5 cm box wall thickness was considered to be the lower desirable limit). A width across the flats of the boxes of 25 cm was considered a desirable upper limit.

(ii) Restrictions were put on the quantity of dispersed phase present in the beryllium oxide matrix, owing to the fission product recoil damage to the matrix structure under irradiation (Hickman 1962), and also the probable strength reduction and fabrication problems associated with concentrated dispersions of fuel and fertile material in the matrix. A figure of 25 per cent. by volume of the dispersed phase was assumed as the upper limit.

The interrelation of these restrictions with the basic parameters of the reference design for each of the fuel shapes leads to a study to determine the optimum dimensions of the fuel element; the desirable goal being the largest fuel shapes compatible with the thermal stress resistance of the material under the most stringent operating conditions. The most probable point of failure was taken to correspond to the point where the demand thermal stress resistance curve (based on a sinusoidal power distribution) touches the allowable thermal stress curve for fuelled irradiated beryllium oxide, drawn as a function of temperature (Figure 10).

3.3.2 Selection of fuel shape and arrangement

The choice of fuel shape was found to depend significantly on the type of core structure considered; either the metal containment box or the ceramic box type structure of Figure 6. Of the shapes considered none stood out above the others in all respects, and in each case there remained some unresolved problems. Features of the fuel shapes are given in Section 3.3.3.

It was found that peak surface temperatures were well below the upper limits for the fuel (approx. 1200 °C) so that increases in coolant channel and hence fuel shape dimensions, if achievable, could result in improvements in the core design because they would give:

(i) fewer and larger pieces with an expected reduction in fabrication charges,

(ii) reduced core pumping power requirements,

(iii) less difficulty with cumulative tolerance problems within assemblies of fuel shapes.

In the case of the metal fuel element structure the central tube design (Figure 1 d) was found to give the greatest size increase compared with the hexagon tubes (Figure 1 b) firstly because of the separation of some of the moderator from the fuel and secondly because of the improved stress distribution from cooling both sides of the tube.

In the case of the ceramic box structure fuel element (Figure 6) where a substantial portion of the moderator is already separated from the fuel, either of types (b), (d), or (e) were shown to give fuel tubes of similar diameter 1.3 – 1.4 cm bore diameter) and no preference from the point of view of size could be given. Details of the thermal performance of types (d) and (e) are given in Table 3. The performance and dimensions of the hexagon tubes type (b) are similar to those of the tube cluster type (e) and are not given separately.

Because each fuel arrangement discussed here has some advantages over the others it is recommended that some freedom be retained to select the most promising design in the light of new information, particularly in relation to nuclear performance. Further, since the fuel arrangements all use similar extruded and cold pressed shapes, the materials problems would in the main be common to all types and would not be greatly affected while further work was concentrated on engineering and physics problems to resolve the situation.

3.3.3 Features of fuel shapes considered

(i) Multi–hole block arrangement (Figure 1 a)

A survey of the thermoelastic stress of a number of fuel shapes (Binns 1962) concluded that a matrix of cooling holes on a triangular pitch provided the lowest thermal stress for equal power density, channel equivalent diameter, and coolant voidage conditions. This arrangement was physically realised as a multi–hole block fabricated by cold pressing and sintering to overall dimensions of up to some 15 cm across flats and 5 cm deep. The design was discarded because a pitch, between holes, of less than two hole diameters was considered insufficient for economic manufacture of the blocks, and because non–linear flux gradients across the blocks would superimpose stresses in addition to those due to cooling at each hole. Considerable difficulty was also encountered in designing a workable assembly of these blocks so that they could be remote–handled into and out of the reactor.

(ii) Hexagonal tube arrangement (Figure 1 b)

A second way of realising a hexagonal matrix of cooling holes is by using extruded hexagonal tubes stacked side by side in either a metal or ceramic box structure. For the metal box structure and no separated moderator, tube bores of no more than 1 cm were obtained. In the case of the ceramic box structure with an arrangement similar to Figure 6 b, but containing hexagonal instead of circular tubes, tube bores of 1.39 cm were achieved.

(iii) Sleeve fuel tube arrangement (Figure 1 c)

This fuel shape has the advantage of partial separation of fuel and moderator material, permitting increased shape dimensions. However, with the manufacturing tolerances accepted in the study very loose fits were found necessary to ensure assembly of the fuel sleeve into the moderator tube. This in its turn meant that considerable unnecessary waste voidage was incurred. Further problems could occur on disassembly if irradiation growth in the BeO and the fuelled BeO shape caused jamming of the sleeve in the tube.

(iv) Central cooled tube arrangement (Figure 1 d)

This design, although having advantages because of the internally and externally cooled tube, is still subject to assembly problems and also to an unbalanced stress distribution if tube misalignment occurs. Fracture of a fuel tube could well lead to partial coolant channel blockage, further non-uniform cooling, and even failure in neighbouring pieces. A typical set of performance parameters for this arrangement is given in Table 4.

(v) Tube cluster arrangement (Figure 1 e)

This fuel element arrangement attempts to combine the advantages of cooling on both sides of the tube with a simple shape not requiring assembly of small pieces within small pieces. To achieve the largest tube size the design favours a highly separated fuel and moderator structure. A typical set of performance parameters for this arrangement is given in Table 4. As with the clustered hexagonal tube arrangement (which has tubes of similar dimensions) the assembly and disassembly of the fuel element is relatively straightforward while the number of components is of the same order as the central cooled tube arrangement. Thus, provided considerations of nuclear heterogeneity do not greatly affect the issue, from a design point of view the simpler arrangements of hexagon or circular tube clusters (Figures 1 b and e) seem the most favourable for the ceramic box type fuel element.

3.4 Performance Aspects of the P.C.R. Reactor

3.4.1 Nuclear performance

Probably the greatest unknown affecting the performance of the P.C.R. core arrangements has been the effect of the fuel and moderator configurations on the conversion and burnup of the system. The choice of a particular fuel arrangement also depends on the fuel cycle chosen and, especially, whether a plutonium or uranium fuel is to be used to start the plant. In the case of the plutonium fuel cycle it could well be advantageous to have a heterogeneous system, while the opposite is probably true for a U235/Th cycle.

Tentative fuel composition ratios of 1:20:2000 for a U235/Th/Be oxide mixture and 1:10:2000 for a Pu/Th/Be oxide mixture were used to check the dispersion limits given in Section 3.3.1. Mixtures of this order would appear to be confirmed by work reported by General Atomic (1963).

3.4.2 Thermal performance of P.C.R. cores

As for the P.B.R. study the performance of the P.C.R. core for the conditions laid down is limited by high thermal stresses in the fuelled shapes and not by surface or material temperature restrictions. The thermal performance of the P.C.R. cores was considered by Hawker (Unpublished) for sinusoidal and flat power distributions both with and without the outlet coolant gas temperature restriction of 750°C. The study used the relation for heat transfer in a circular coolant channel:

$$St = 0.023\, Re^{-0.2}\, Pr^{-0.6} \quad ,$$

and for friction in the same channel:

$$f = \frac{0.046}{Re^{0.2}} \quad ,$$

based on a survey of correlations by Gerrand (Unpublished). The concept of channel equivalent diameter was used for non-circular coolant passages.

Both the heat transfer coefficient and the friction factor were increased by 20 per cent. to allow for misalignment steps between successive lengths of the fuel and moderator tubes forming the coolant channels. Entry and exit losses from the channels were also included in the total pressure drop and core pumping power estimates. It was assumed that the fuel elements, other than the centre element, would be gagged to give a constant outlet temperature across the core. No allowance was made for heat production in the unfuelled moderator shapes. Thus the heat input to the fuel shapes is overestimated, making the design conservative on this point.

For the reference design conditions (Table 1) a summary of the thermal performance of the fuel shapes of Figure 1 d and e is given in Table 5. The average maximum fuel surface temperature in each of these designs is no higher than 905 °C leaving a considerable margin for temperature peaking factors and also for increase in coolant outlet temperature from this type of core. By flattening the core shape some useful reductions in pumping power may be achieved.

3.4.3 Fuel shape stress analysis and its effect on the core thermal performance

The equations for the stresses generated in the fuel tubes of the configuration of Figure 1 have been given by Binns (1962) for uniform heat generation throughout the fuel bearing material. A summary of the factors of uncertainty taken into account in the analysis is given in the Appendix.

In the case of the central tube cooled on both sides, (Figure 1 d), the largest shape dimensions are achieved when the surface temperatures on both inner and outer surfaces are equal. The channel sizes within the restrictions of Section 3.3 and the structural form of the core have been determined for the case of the reference design so that the condition of equal surface temperatures is approached on the inside and outside surfaces of the fuelled tubes. The condition cannot be achieved simultaneously at all points along the coolant channel without mixing the coolant from the inner and outer channels. A similar analysis was applied in the case of the tube cluster arrangement (Figure 1 e).

3.5 Fuel Element Handling

An important aspect affecting the P.C.R. cores is the feasibility of handling the fuel elements. The handling machinery considered by Manning (Unpublished) is shown schematically in Figure 8. It consists of a radius arm arrangement capable of lifting or placing a fuel element at any of the location points in the core or reflector region. The arrangement has many features in common with the handling arrangements of the DRAGON and Peachbottom H.T.G.R. reactors, so emphasis was laid on problems of special interest to the beryllium oxide type cores considered here. The first problem was the feasibility of reliably lifting large, and therefore heavy, fuel elements into and out of the core. The large size was proposed primarily to minimise the quantity of structural material required in the core. It was concluded that it was quite feasible to lift and handle fuel elements weighing 2000 lb. It was also concluded that the period required for charging and discharging a core would not impose reactor down-time periods in excess of those needed for normal plant maintenance.

The second problem studied was the feasibility of remotely locating the tops of the fuel elements when their actual positions were not known owing to core movement, particularly when the radial clamping of the top of the core was released to remove the elements. An out-of-true position of up to 5 cm for any fuel element lifting head was found to be within the capacity of the handling machine proposed.

An important problem arises in the behaviour of the fuel element itself during handling. Since the element boxes are constructed of a number of sections stacked on top of each other, there must necessarily be some flexibility at each of the joints. The element with the least number of joints and hence the more desirable design is that shown in Figure 6 b).

To ensure that the leading and trailing edges of the ceramic box pieces do not foul neighbouring fuel elements, some rounding off of the edges is necessary. The arrangement should be practicable but will need experimental verification.

Since the fuel element box pieces are unfuelled, consideration was given to their re-use with replacement fuel shapes resulting in a considerable saving in the usage of beryllium oxide and hence important reductions in fuel cost. The effects of irradiation damage to these pieces may be substantially removed by annealing at 1200 °C before re-use. However the net effect of irradiation damage and then annealing is to induce growth in the beryllium oxide. Linear dimension changes of some 3 per cent. could be expected from beryllium oxide irradiated to its limiting dose and then subsequently annealed (Hickman and Pryor 1964). It thus appears that the dimensional changes in the beryllium oxide moderator pieces would be the major restriction on their re-use for further fuel shape lives.

3.6 Development Potential of the Parallel Channel Reactor Core

Some improvements that may be made to the P.C.R. core concept as currently envisaged give an indication of the development potential of this core type. It should be emphasised that the features mentioned fall within the basic definition of the reactor system described. Broader avenues for development involving changes to the basic definition of the system (as discussed in Section 1) are not considered here.

3.6.1 Improvement in thermal performance

In each case studied, the core thermal performance and fuel shape dimensions were determined from the basic reference design parameters (Table 1) to assist in making comparisons between the core types. Hence there is scope for development and optimisation of any of the designs since the reference design parameters are not regarded as an optimised set.

Objectives worth pursuing are:

(i) Reduction of core pumping power and fuel shape rattling problems by flattening the core and possibly increasing the primary circuit coolant pressure.

(ii) Optimisation of the coolant voidage in the core since this affects fuel shape dimensions (higher voidage in general results in larger shape cross–sections).

(iii) Adjustment of coolant passage dimensions within the restrictions of the material to give the largest fuel shapes (this has already been done to some extent for the central tube fuel element design).

3.6.2 Improvements in the fuel shape and core configuration

A re–appraisal of the fabrication restrictions given in Section 3.3.1 for a given core arrangement could well lead to improvements in the size of fuel shapes and in the rigidity of the fuel element. Two important restrictions here are the minimum tube wall thickness and also the restrictions on the manufacture of large box pieces for the fuel element structure.

Closer dimensional tolerances, particularly on straightness, would permit the use of much longer and hence fewer extruded lengths in the fuel elements. This would also result in substantial improvement to the structural rigidity of the fuel element and some simplification of its design and assembly.

3.7 Recommendations Concerning Parallel Channel Reactors

The purpose of this study has been to consider ways of constructing all beryllium oxide moderated parallel channel reactor cores, assessing the problems that might be met in the design of such reactors. Because of the freedom one has in the choice of fuel shape and especially core configuration, and the limited nature of the study, no claim is made that the shapes and arrangements considered here are the best. The study has however highlighted the design problems of these cores. The following conclusions are drawn from this part of the study.

(i) Within the restrictions of this report no practical design of parallel channel reactor core consisting of components containing only fissile and fertile particles dispersed in all of the beryllium oxide moderator appears feasible. This is primarily because some structure, including steel, is needed to remove and replace the fuel elements in a reliable way.

(ii) There is a strong design incentive to introduce a portion of the moderator as unfuelled moderator since this permits increased fuel shape dimensions. The partial separation of fuel and moderator is also expected to introduce savings in reprocessing charges Cairns (Unpublished) particularly if the moderator may be re–used for greater than the fuel life. However a most important remaining question is the effect on the nuclear performance of the reactor when the core contains substantial proportions of separated fuel and moderator.

(iii) Of the fuel shapes considered, relatively thin walled tubular sections should offer greatest promise. Extrusion techniques for their manufacture have advantages since it is feasible to fabricate relatively long pieces.

(iv) Dimensional tolerances are an important problem in relation to assembly and fuel element structural rigidity. However, it does appear that as-fabricated shapes could be used with a minimum of final grinding (for example at the ends of the fuel tubes alone).

(v) Two alternative core structures appear feasible. Firstly a steel box structure for each fuel element is the more reliable, provided that approximately 2.0 per cent. by volume of steel can be tolerated in the core. Secondly, rather large cross-section beryllium oxide block shapes stacked together to form hexagonal fuel element boxes; but still incorporating some 0.25 per cent. by volume of steel, appear to be the best that can be achieved in using beryllium oxide as structural material. The beryllium oxide box, because of the core volume it occupies is certain to constitute a substantial departure from nuclear homogeneity.

Finally it is of interest to summarise what are considered to be the most important problems with this type of design:

(i) Construction of a reliable core structure consisting of removable elements without using substantial quantities of steel in the core.

(ii) Design of the fuel elements to avoid rattling and vibration of the fuel shapes.

(iii) Design to cope with the growth under irradiation of the beryllium oxide components constituting the core.

(iv) The assessment of the importance of fuel shape cracking in relation to channel blockage and the initiation of further core damage.

4. RELATIVE MERITS OF PEBBLE BED AND PARALLEL CHANNEL REACTORS

4.1 Introduction

An adequate assessment of the relative merits of pebble bed and parallel channel reactor cores requires an appraisal not only of the thermal performance and design problems, but also of the detailed nuclear characteristics, the primary circuit and the plant layout, and the relative costs of the systems.

At the time of the study the information in these fields available to the authors was meagre and reliable cost estimates were almost non-existent. Therefore a comparison could be made only on the broadest lines using a number of rather sweeping assumptions. In the final evaluation of course, the validity of these assumptions must be considered in relation to the conclusions drawn. The abovementioned factors are therefore discussed against this background.

4.2 Comparison of Thermal and Mechanical Aspects

4.2.1 Thermal performance features

Accepting the recommendations of Section 2 on P.B.R. cores the thermal performance of the downflow arrangement suggested is somewhat inferior to that of the P.C.R. designs. The major differences lie firstly in the higher core pumping power required for the P.B.R., and secondly in the greater difficulty in controlling peaking of fuel and gas temperatures in the P.B.R. where coolant gagging is not possible. Routine direct fuel and local gas temperature measurement is also probably an impossibility in the random packed P.B.R. type core.

4.2.2 Degree of dispersion of fuel in beryllium oxide

The P.B.R. is an arrangement particularly amenable to a core containing a complete dispersion of fuel and fertile particles in all of the beryllium oxide. It was shown that in many cases more satisfactory parallel channel core designs were possible with substantially separated fuel and moderator material. Hence the P.B.R. concept using beryllium oxide will suffer if heterogeneity turns out to be a desirable feature since the already rather small ball size would need to be further reduced to operate a bed of mixed fuelled and unfuelled balls at the same overall power density.

4.2.3 Mechanical design features and problems

As the mechanical design features and problems for each core type have been discussed in Sections 2 and 3 only a summary is presented here without detailed comment on their relative importance or degree of uncertainty.

P.B.R. Core advantages over P.C.R. Core:

(i) In principle the P.B.R. core design is extremely simple.

(ii) Great freedom of fuel handling and fuel management is possible.

(iii) Little or no metal structure is needed with the core.

(iv) There is the possibility that the reactor might be controlled by temperature coefficient and hence blower power.

(v) Substantially equal burnup in all spent fuel should be attainable.

(vi) Fuel handling equipment is expected to be less costly.

P.C.R. Core advantages over P.B.R. Core:

(i) Reflector design of the P.C.R. is much simpler and reflector components are easily removable without additional equipment or cost.

(ii) The coolant flow through the bed, and coolant outlet temperature distribution, are under closer control and are better known. Channel gagging may be used.

(iii) Control rods may be placed with equal ease at any position in the core.

(iv) Measurement of fuel temperatures within the core is possible.

(v) The design of the support structure for a downflow P.C.R. core offers less problems.

(vi) The problems of wear, abrasion and contact loading of the fuel shapes are expected to be small compared with the P.B.R. where both balls and core side and bottom reflectors are subject to wear and high point contact loading.

(vii) The state of knowledge of thermal performance and mechanical design is better for P.C.R. type cores.

4.3 Relative Cost Comparison

4.3.1 Capital costs

Both types of reactor give compact core sizes and similar plant efficiencies so that outside the core and reflectors the plant and also the capital cost charges may be regarded as comparable. There may however be some gain in the handling equipment costs for the P.B.R. over the P.C.R. The capital cost of the first core is probably also of the same order for either core type. Overall it would appear that from a capital cost point of view there is no strong reason for preference of either type of core.

4.3.2 Fuel cycle costs

The assessment of relative fuel cycle costs has proved difficult because of lack of cost and reactivity lifetime data for the systems of interest. However, based on the work of Wright (1962 and 1963), some relative figures have been produced as a guide. It will be assumed that the first cost of P.B.R. and P.C.R. cores is comparable for equal core power densities (materials costs form about 75 per cent. of the cost so that differences in fabrication costs are unlikely to have a major effect). Thus the basic fuel charge for fully dispersed fuelled cores which are loaded into the reactor and remain there until either reactivity lifetime or irradiation damage restrictions are encountered, is assumed comparable for each design.

Wright (1963) and Bardwell (Private Communication) have determined the effects of making improvements in this basic fuel cycle by fuel management techniques and partial separation of fuel and moderator in the P.C.R. core design.

The figures for fuel cycle costs given in Table 5 are taken from Wright (1963) and therefore their validity relies on the data used and the assumptions made there. It must be pointed out that the cost reductions indicated are only orders of magnitude to show the relative costs of the two types of cores. Inaccuracies in the base fuel cycle cost figure of 0.22d/kWh will not significantly affect the comparison. An important assumption in the fuel cycle costs has been that the reprocessing charges are equivalent to the costs of fresh raw materials.

The figures quoted in Table 5 are a crude indication of what might happen regarding fuel costs. The cost improvement, (Item 4 in the Table), is regarded as optimistic since efficient radial fuel shuffling is likely to be difficult to achieve in practice. A second point which could reduce the P.B.R. fuel cost in comparison with the P.C.R. cost, is that the P.B.R. core requires no initial excess reactivity to be controlled in the early stages of operation of the reactor. The importance of this effect clearly depends on the unknown reactivity lifetime characteristics of the fuel cycle used.

The inference to be drawn from the comparison is that the fuel cycle cost of the fully dispersed fuelled P.B.R. is at least comparable to, and possibly better than that of the substantially separate fuelled P.C.R. concept. It is important to note that the comparison includes the cost saving from the partly separated fuel and moderator in the P.C.R. fuel cost only. If a similar improvement could be made in the case of the P.B.R. concept then fuel costs should most certainly favour the pebble bed reactor. Difficulties with mixed fuelled and unfuelled ball pebble bed reactors are discussed in Section 2.

(It has been assumed in this discussion on fuel costs that the introduction of substantial quantities of separated beryllium oxide does not impair the reactivity lifetime characteristics of the reactor. This may well be untrue (General Atomic 1963) and therefore favour the P.B.R. concept).

5. FINAL CONCLUSIONS AND COMMENTS

Within the framework of this survey some general conclusions may be drawn.

1. It is considered that workable arrangements of both pebble bed and parallel channel beryllium oxide moderated reactor cores should be possible, even though there remain some important unresolved technical problems in both types of core. In the case of the P.B.R. core there appears to be a greater risk of not adequately solving these problems (because of their nature) and hence a less economic design could result. For example to maintain a reasonable ball size (not less than 2.5 cm diameter) one may be forced to limit the core power density to values well below 15 watt/cm^3.

2. Thermal performance in terms of lower pumping power and better control of fuel surface temperatures favours the P.C.R. cores but this has not been found to be an overriding consideration.

3. The P.B.R. offers the most practicable fully dispersed fuel and moderator core arrangement.

4. Capital costs for the two core types are not expected to be significantly different.

5. Ease of fuel management is an important advantage in the P.B.R. core. Other things being equal, this is the prime factor in giving the pebble bed reactor the prospect of the lower fuel cycle costs.

6. Although having good prospects as a fuel dispersion medium, beryllium oxide has been found difficult to use as a nuclear reactor core or reflector structural material. While carbon dioxide is considered as coolant the only feasible alternative structural material is steel which must be preferentially cooled and may introduce penalties in nuclear performance.

6. ACKNOWLEDGEMENT

Thanks are due to Mr. I. P. Arthur, Mr. L. R. Scott–Rogers, and Mr. E.L. Rollinson for their help in the study and to Mr. U. Mardus, Mr. P. Linstead, Mr. N. Ridgeway, and Mr. O. Osmotherly who carried out the design draughting work and many of the calculations.

7. NOTATION

d = ball diameter (cm)

F = factor of uncertainty used in prediction of fuel shape dimensions (dimensionless)

f = friction factor (dimensionless)

E = Young's Modulus (p.s.i.)

k = Thermal conductivity

Pr = Prandtl Number (dimensionless)

q_b = power density in ball (watt/cm^3)

Re = Reynolds Number (dimensionless)

St = Stanton Number (dimensionless)

α = coefficient of linear thermal expansion (deg C^{-1})

σ = Maximum allowable tensile strength (p.s.i.) (based on Modulus of Rupture data)

μ = Poïsson's Ratio

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APPENDIX

FACTORS ALLOWED FOR IN THE PREDICTION OF FUEL SHAPE DIMENSIONS FROM THE ALLOWABLE THERMAL STRESS RESISTANCE IN THE MATERIAL

Material Properties

The thermal stress resistance $\sigma(1-\mu)k/E\alpha$ of irradiated fuelled BeO was determined as a function of the material operating temperature using unpublished data of Whatham, and Hickman and Pryor (1964) for corrections for irradiation. We used average values of all properties except the strength where the mean value of modulus of rupture, less 5000 p.s.i., was taken as the design value, to cope with the strength scatter band of the material.

Design Point

For the P.C.R. core calculations the design point was taken as the point where the demand thermal stress resistance curve (based on a sinusoidal power distribution) touched the allowable thermal stress resistance curve for irradiated fuelled beryllium oxide plotted as a function of material temperature (Figure 10). The P.B.R. core calculations originally assumed a fixed datum temperature of 650 °C for the evaluation of all material properties. However in check calculations to permit comparison of the relative fuel shape sizes, the more accurate method matching the thermal stress resistance curves was employed as discussed in Section 2.3.1.

Thermal Stress Resistance and Uncertainty Factors

In Table 4 the basic materials data and maximum allowable thermal stress resistances used are given.

The factor of uncertainty (F) quoted in each case has been estimated to take account of the combined effects of the following:

(i) Local power variations about the mean radial and axial distributions.

(ii) Non–uniformity of fuel dispersion within the matrix.

(iii) Variations in heat output due to variations in fuel quantity between fuel shapes.

(iv) Variations in heat output due to self–shielding effects in the heterogeneous P.C.R. core structure and fresh and partly-spent balls lying side by side in the P.B.R. cores.

(v) Output variations in over–power and start–up transients etc.

(vi) Output variations due to instrumentation inaccuracy.

(vii) Confidence in the material properties used to permit a sufficiently low proportion of fuel shapes cracking under thermal stress.

(viii) Non–uniformity of cooling of the fuel shape, contact loadings (in P.B.R.), and irradiation growth and creep effects.

(ix) Tolerances on fuel shape dimensions.

(x) Output reductions due to the addition of surface coatings to the fuel shapes.

(xi) End stress effects in tubular shapes.

(xii) Errors in gagging (P.C.R.) and in coolant mass flow prediction.

(xiii) Voidage variations across the bed in the P.B.R.

(continued)

The factors quoted should be of the right order, though many aspects affecting them can be little more than guessed at this stage. If anything the factors used for the P.C.R. studies are a little conservative compared with those used for the P.B.R. study, leading to the conclusion that the P.C.R. fuel shape dimensions could well be a little larger than given here. However, for this survey the sizes of the fuel shapes in the P.B.R. and the P.C.R. studies are comparable.

TABLE 1

BASIC PERFORMANCE DATA USED IN THE STUDY

Reactor core useful heat output = 500 MWt

Station net electrical output = 200 MWe (approx.)

Station overall efficiency = 40% (approx.)

Reactor primary circuit coolant = carbon dioxide

Primary coolant average temperature at reactor inlet = 350 °C

Primary coolant average temperature at reactor outlet = 750 °C

Primary coolant circuit pressure at reactor inlet = 603 p.s.i.a. (40 atm. gauge)

Secondary circuit coolant = steam

Steam pressure at turbine stop valve = 2300 p.s.i.a.

Steam temperature at turbine stop valve = 1050 °F

Reheat steam temperature at turbine stop valve = 1050 °F

Core average power density = 15 MW/m^3

Total voidage in core: P.C.R. = 25%

P.B.R. = 40%

Fissile material = UO_2 or PuO_2

Fertile material = ThO_2

Moderator material = BeO

Axial form factor = 1.45

Radial form factor = 1.27

Overall form factor = 1.84

Load factor = 0.8

Design life = 25 years

TABLE 2

PARAMETER LIST COMPARING PEBBLE BED REACTORS

Parameter	Units	Upflow Core		Downflow Core		Radial Trapezoidal Core	Radial Cylindrical Core
		(a)	(b)	(a)	(b)	(a)	(a)
Core diameter	cm	540	455	384	384	480	390
Diameter of internal hole in core	cm	–	–	–	–	72	72
Core height: at centre	cm	405	341	288	288	384	292
at edges	cm	405	341	288	288	58	292
Core overall voidage	%	40	40	40	40	42.8	42.2
Core average power density	watt/cm^3	5.4	9.0	15.0	15.0	15.6	15.0
Fuel ball diameter	cm	3.7	5.8	2.2	4.0	2.3	2.2
Thickness of shell	cm		0.25		0.25		
Number of balls	x10^{-6}	2.1	0.33	3.5	0.63	3.2	3.5
Thermal pumping power ratio within core	%	0.5	0.4	2.1	1.0	1.0	0.75
Average surface to coolant temperature difference	deg C	38	90	34	70	40	52
Approximate peaked maximum surface temperature	°C	1070	1250	1060	1180	1025	1075
Pressure drop across bed	p.s.i.	8.2	7.1	37	17	17	11

Columns headed (a) refer to fully dispersed fuel balls.

Columns headed (b) refer to the shell ball concept (Section 2.3.4)

TABLE 3

PARALLEL CHANNEL REACTOR CORE PARAMETER LIST

Parameter	Units	Fuel Element Type A (Fig.6)	Fuel Element Type B (Fig.6)
Core effective diameter	cm	348	348
Core effective length	cm	348	348
Coolant voidage ratio in core	%	20	20
Waste voidage in core	%	5	5
Heat bearing volume ratio in core	%	9.1	11.1
Direction of coolant flow		upwards	upwards
Thermal pumping power ratio in primary circuit	%	1.7	1.3
Thermal pumping power ratio in core only	%	1.3	1.0
Size of fuel shape (a) fuelled tube (i) i.d.	cm	1.30	1.35
(ii) o.d.	cm	1.98	1.76
(b) moderator tube (i) i.d.	cm	2.69	–
(ii) dist. across flats	cm	3.22	–
Size of fuel box (mean dimension)			
(i) inside distance across flats	cm	14.9	10.9
(ii) outside distance across flats	cm	19.6	18.5
Length of fuel shape	cm	15	15
Length of fuel element	cm	495	495
Weight of fuel element	kg	373	341
Volume percentage of steel in core	%	0.26	0.28
Maximum surface heat flux on fuel shape surface	watt/cm^2	Inner Surface 45.2	Inner Surface 34.8
Maximum surface temperature in core (average value)	°C	815	804
Maximum material temperature in core (average value)	°C	845	835
Surface to coolant average temperature difference	degC	100	87
Total fuel element pressure loss	p.s.i.	26	19
Number of elements in active core		261	289
Number of shapes in active core: (i) fuelled tubes		101,790	248,540
(ii) moderator tubes		101,790	–

TABLE 4

MATERIAL THERMAL STRESS RESISTANCE AND UNCERTAINTY FACTORS APPLIED IN THE ANALYSIS

Central Tube	Tube Cluster		Downflow Pebble Bed
	Type A	Type B	
Design point temperature °C	905	855	1000
Volume proportion of dispersed phase (v/o)	25	20	3.0
BeO relative density (v/o of theoretical)	96	96	96
Maximum allowable thermal stress resistance for irradiated material (watt/cm)	6.3	7.8	5.7
Factor of uncertainty (F) used to convert from idealised fuel shape and conditions to expected worst conditions	3.6	4.6	3.4
Idealised conditions	tube uniformly cooled on both sides	tube uniformly cooled on both sides	sphere uniformly cooled at its surface

TABLE 5

FUEL CYCLE COST REDUCTIONS FROM IMPROVEMENTS TO THE BASIC FUEL CYCLE FOR P.B.R. AND P.C.R. CORES

Nature of Improvement to Fuel Cycle	P.B.R. (d/kWh)	P.C.R. (d/kWh)
1. Basic fuel cycle cost for FIFA of 150%. (homogeneous fuel dispersion, no fuel changing or movement before end of life)	0.22	0.22
2. Modified fuel cycle cost from separation of 75 per cent. of the BeO from the fuel dispersion:		
(i) Due to savings in fabrication and reprocessing costs,	none allowed	0.20
(ii) due to re-use of unfuelled moderator for a total of two fuel lives.	none allowed	0.17
3. Modified fuel cycle cost due to burnup improvement from continuous fuel charging and management in the P.B.R.	0.12 – 0.16	none allowed
4. Modified fuel cycle cost due to burnup improvement from batch changing and radial shuffling at 6 – 12 monthly intervals	none allowed	0.12 – 0.14
5. Final estimates of fuel cycle costs	0.12 – 0.16	0.12 – 0.14

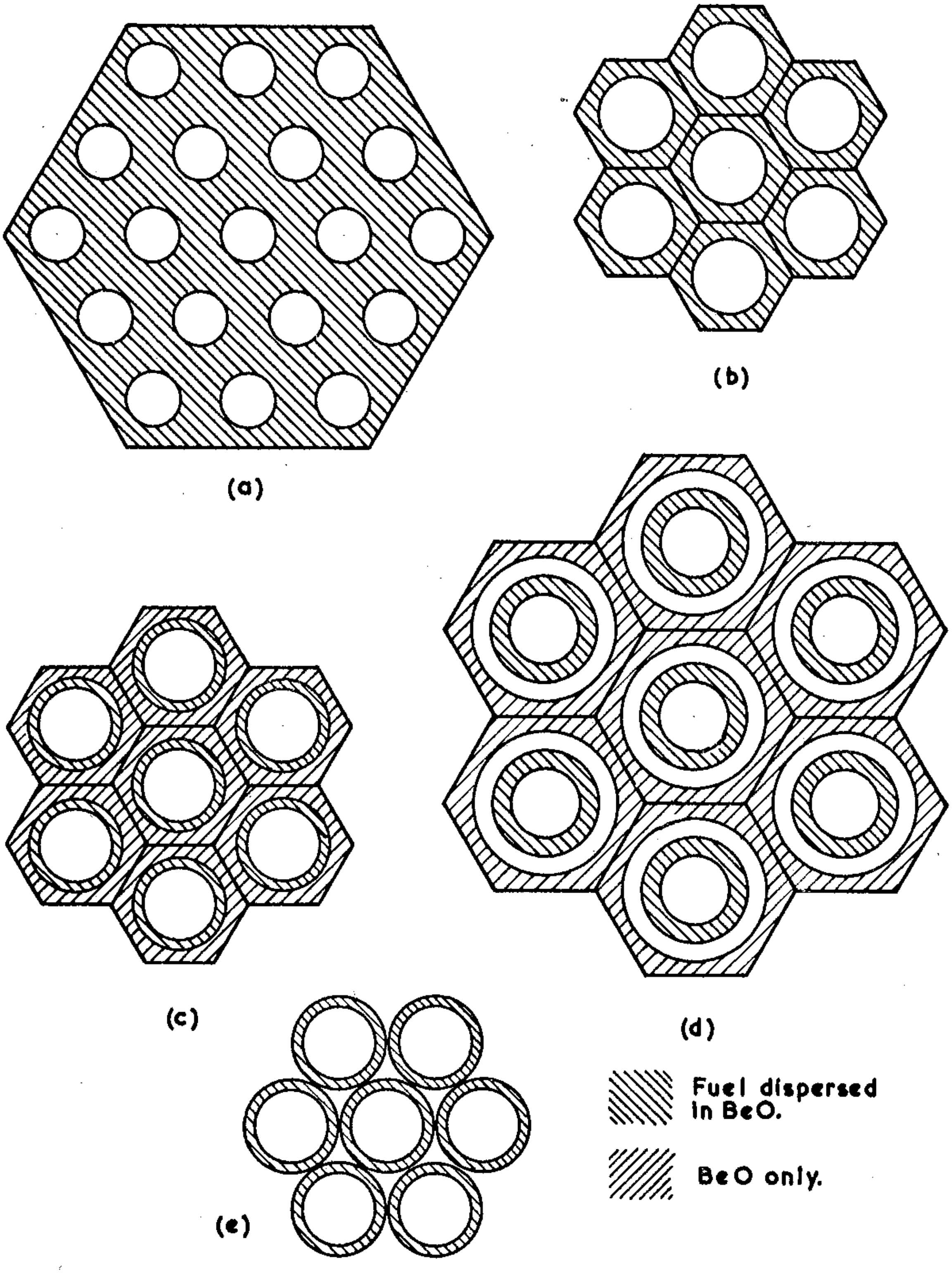

FIGURE 1. PARALLEL CHANNEL REACTOR CORE FUEL SHAPES

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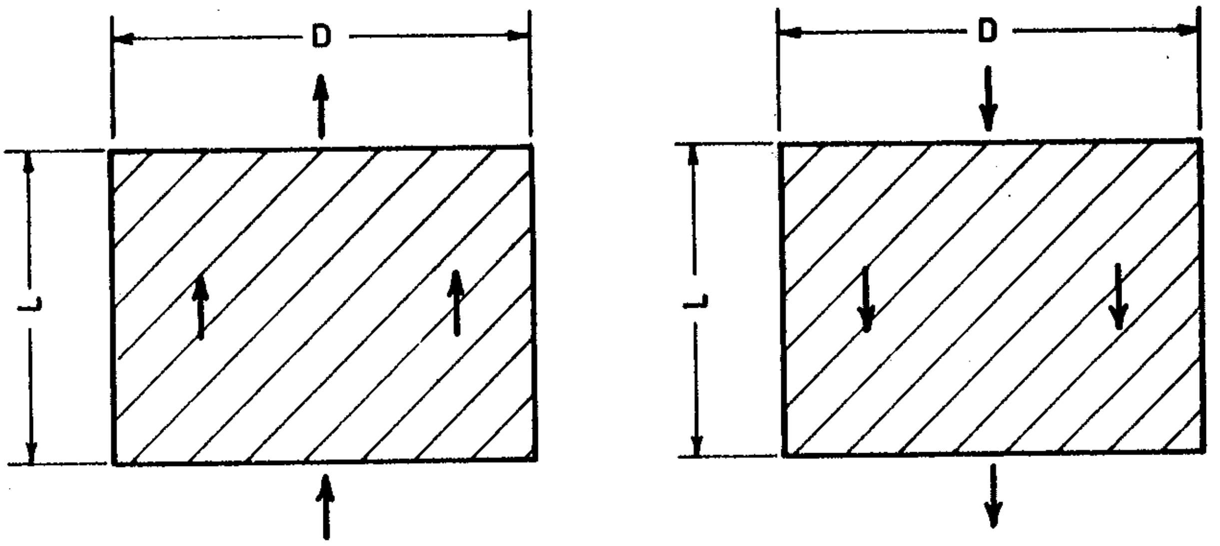
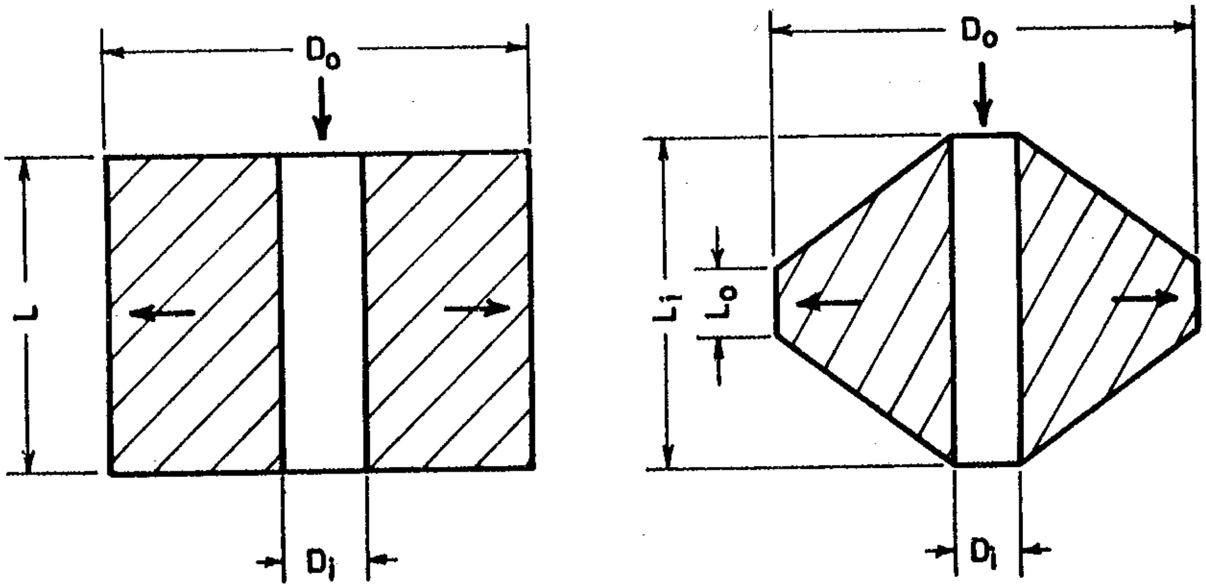


(a) Axial upflow (cylindrical) (b) Axial downflow (cylindrical)



(c) Radial outflow (cylindrical) (d) Radial outflow (trapezoidal)

FIGURE 2. PEBBLE BED REACTOR CORE AND COOLANT FLOW ARRANGEMENTS

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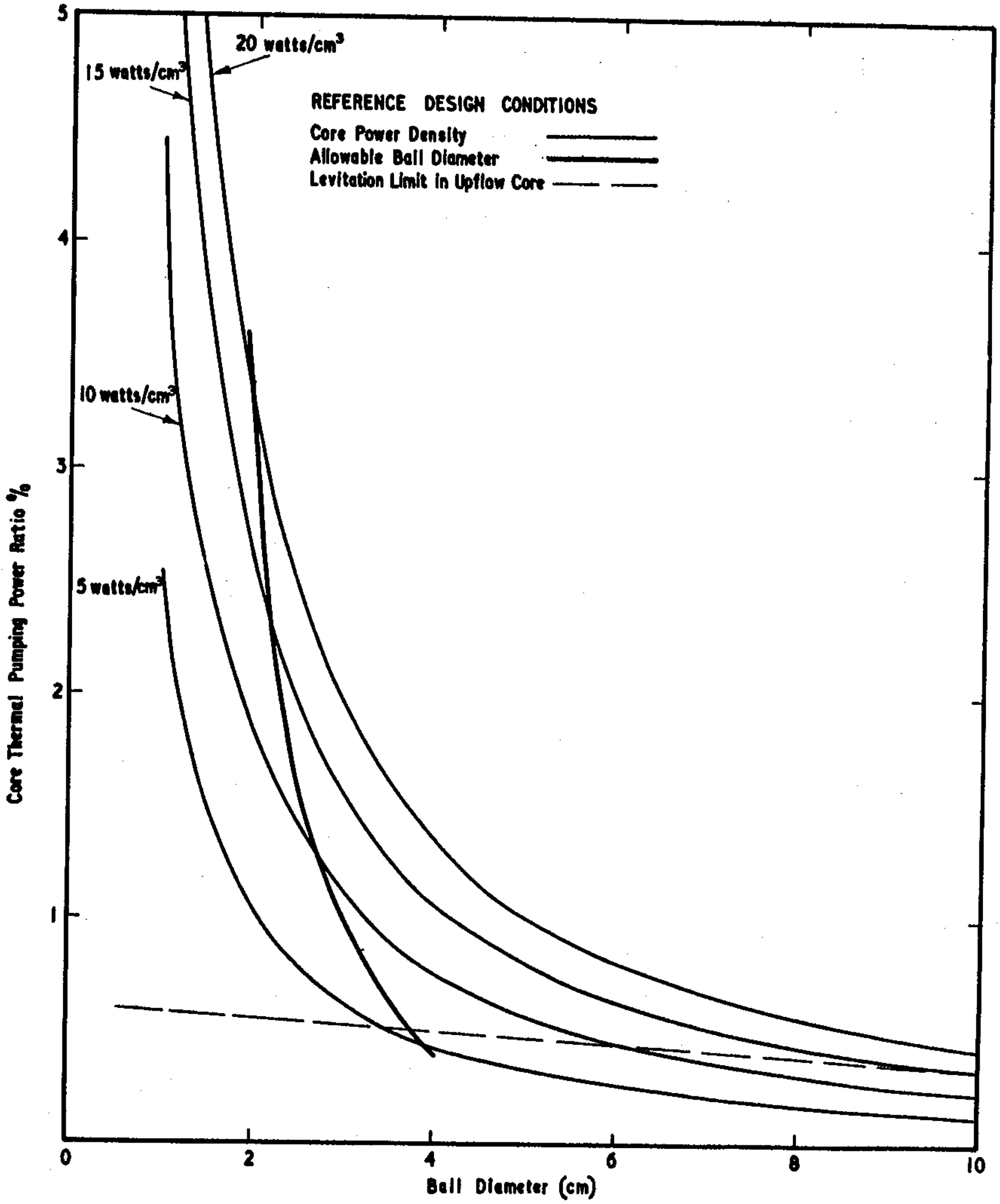
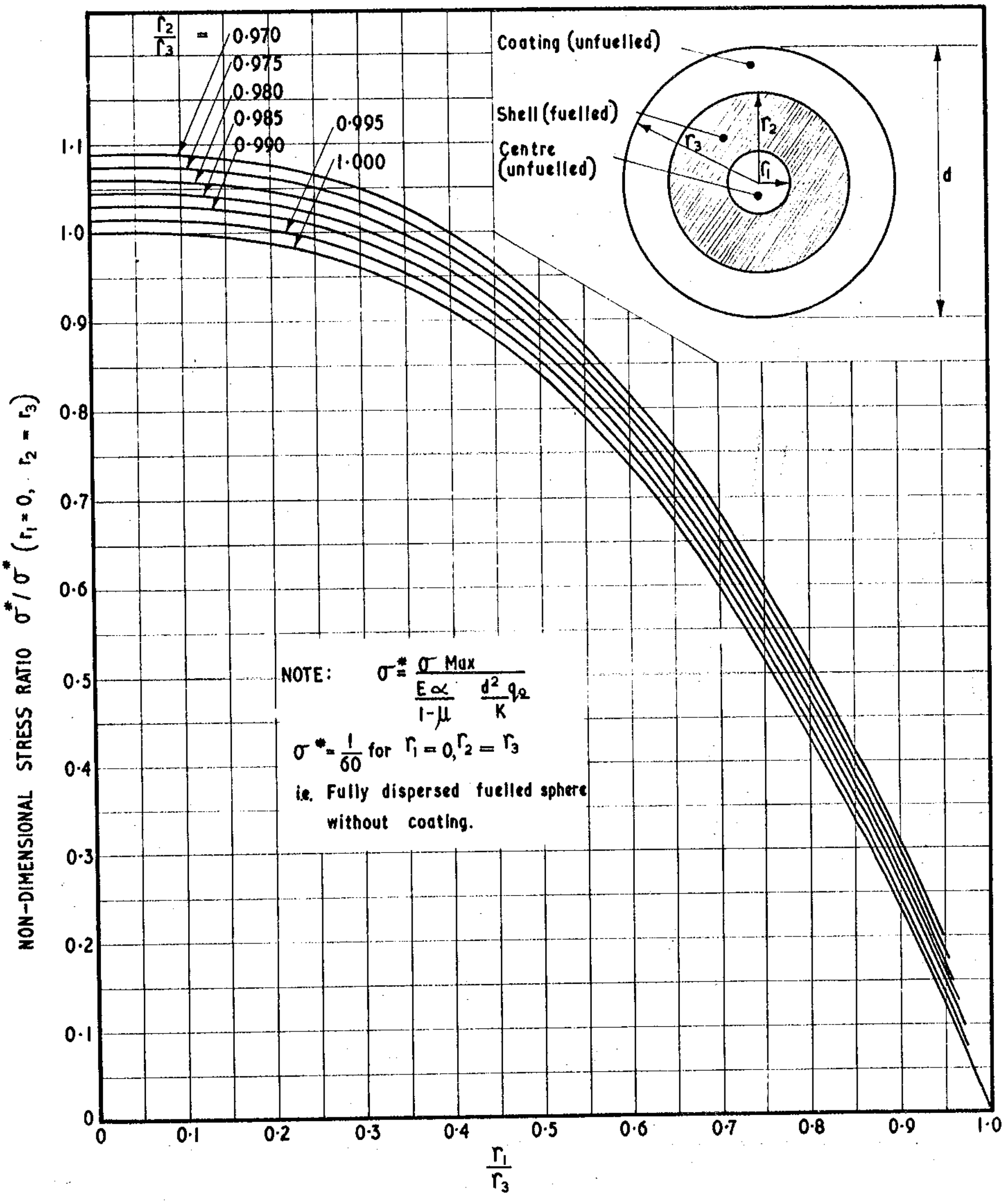


FIGURE 3. ESTIMATED BALL DIAMETER AND PUMPING POWER REQUIREMENTS

P 562



NOTE: $\sigma^* = \dfrac{\sigma\ \text{Max}}{\dfrac{E\alpha}{1-\mu}\ \dfrac{d^2 q_o}{K}}$

$\sigma^* = \frac{1}{60}$ for $r_1 = 0, r_2 = r_3$

i.e. Fully dispersed fuelled sphere without coating.

FIGURE 4. MAXIMUM THERMO-ELASTIC STRESS IN FUELLED SPHERES WITH COATINGS

P562

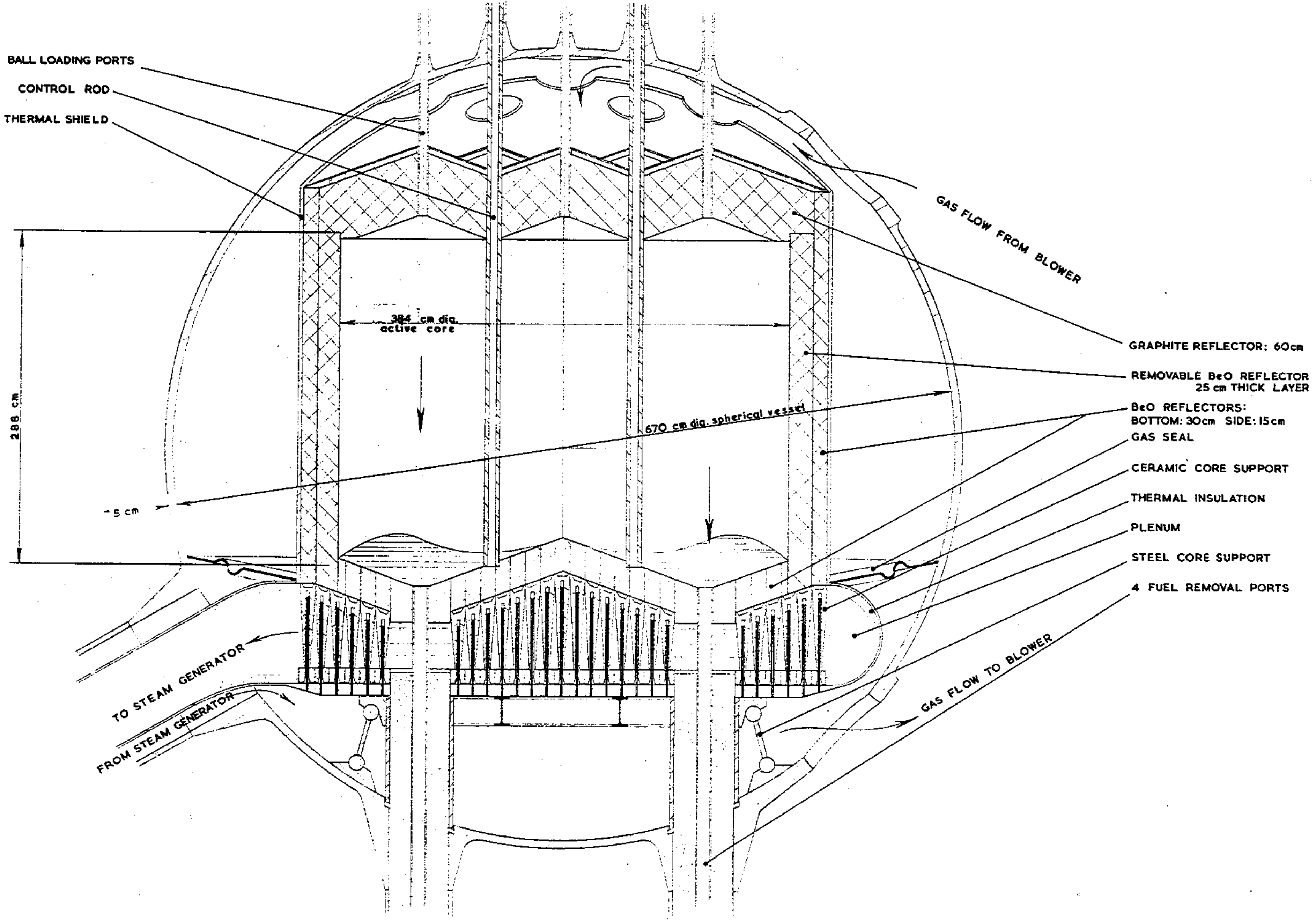
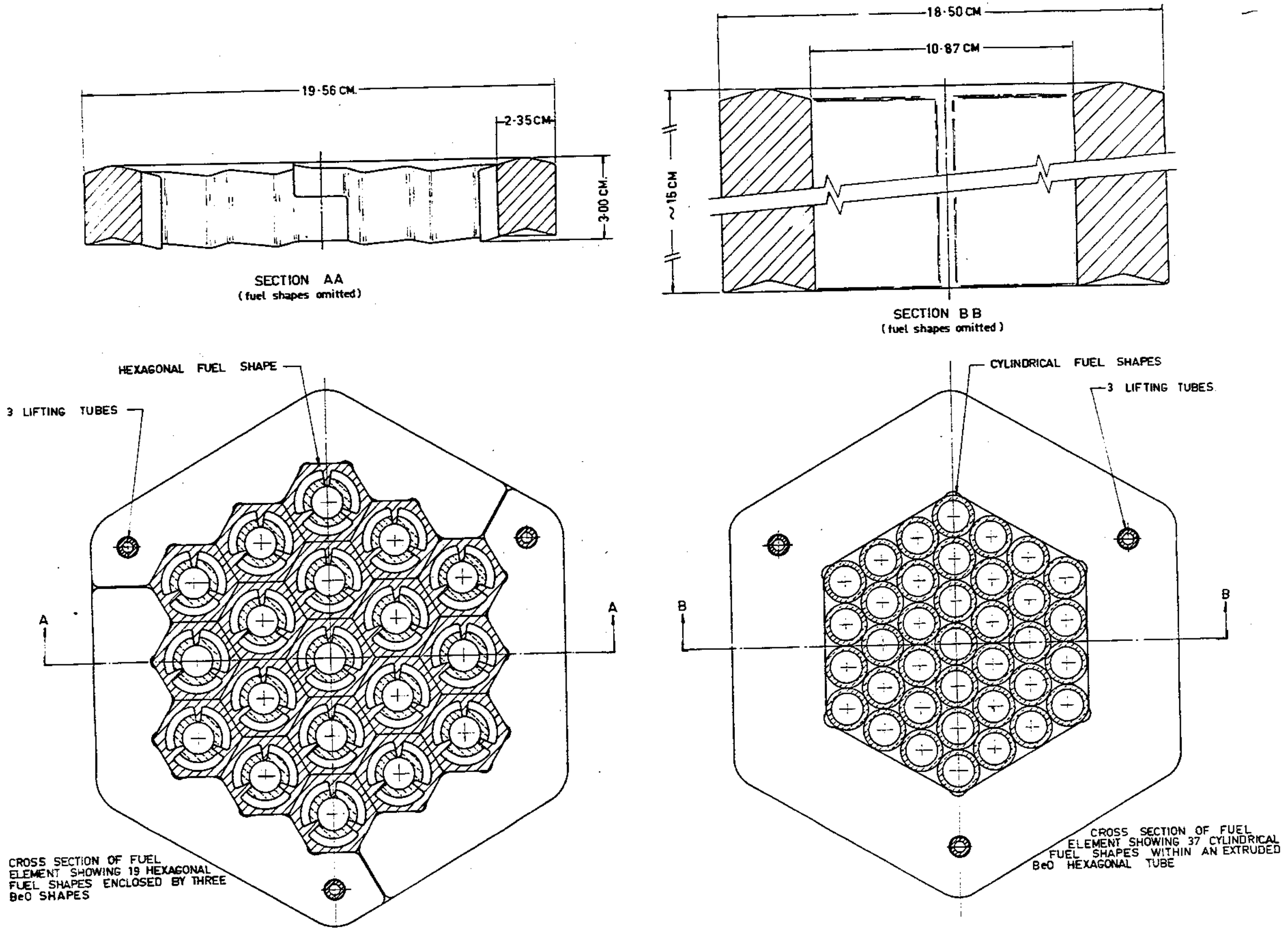


FIGURE 5. SCHEMATIC ARRANGEMENT OF A 500 MWt PEBBLE-BED AXIAL DOWNFLOW CORE



FUEL ELEMENT A **FIGURE 6. CERAMIC FUEL ELEMENT CROSS-SECTIONS** FUEL ELEMENT B

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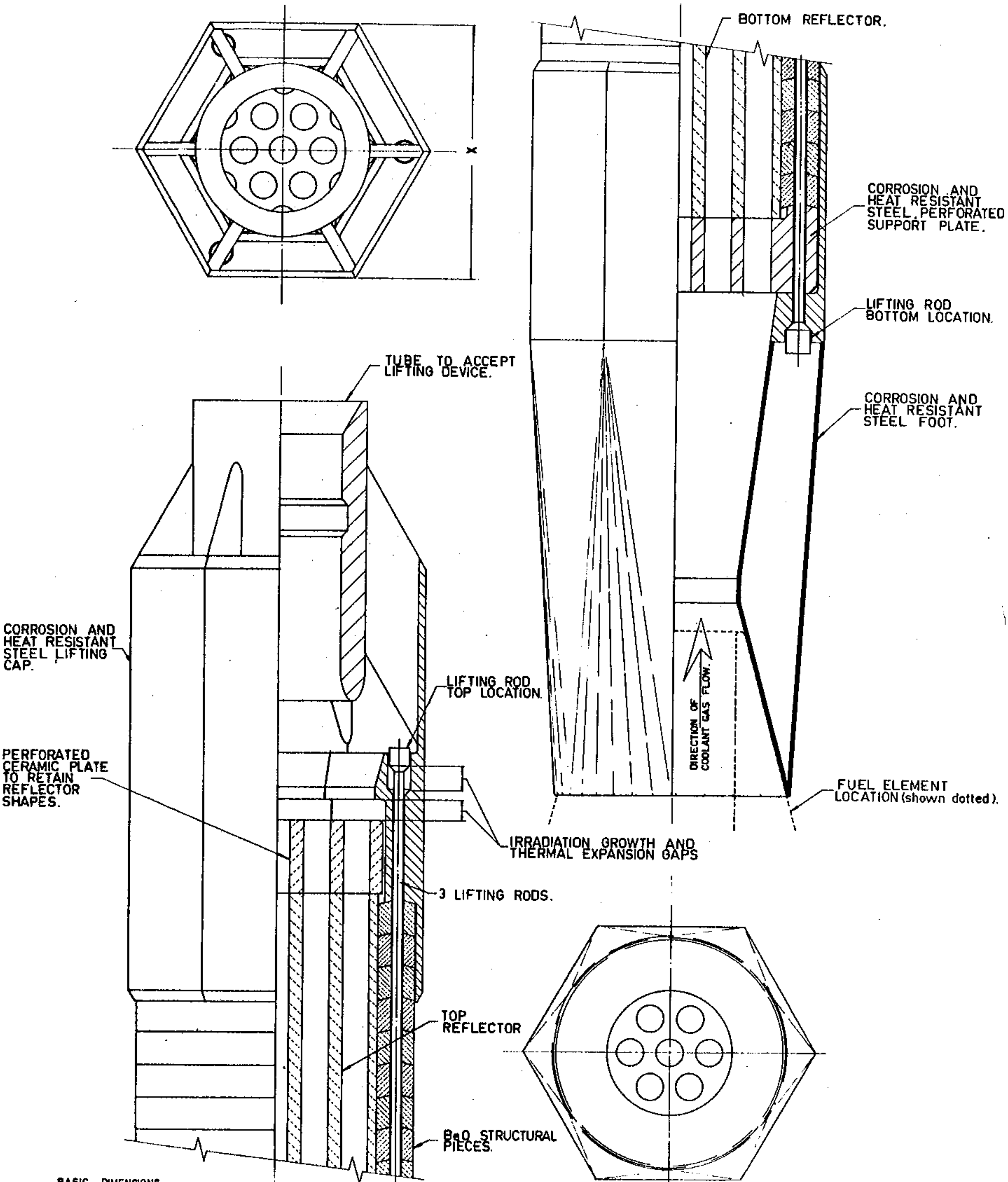

BASIC DIMENSIONS

LENGTH OF FUELLED PORTION OF ELEMENT : 348 cm.
" " REFLECTOR AT EACH END : 30 cm.
" " LIFTING CAP : 40 cm.
" " FOOT : 47 cm.
OVERALL LENGTH OF ELEMENT ASSEMBLY : 495 cm.
ACROSS FLATS DIMENSION 'X' (inter element clearance 0·5cm
FUEL ELEMENT A : 20·06 cm.
FUEL ELEMENT B : 19·00 cm. } SEE FIGURE 6

FIGURE 7. CERAMIC FUEL ELEMENT ASSEMBLY

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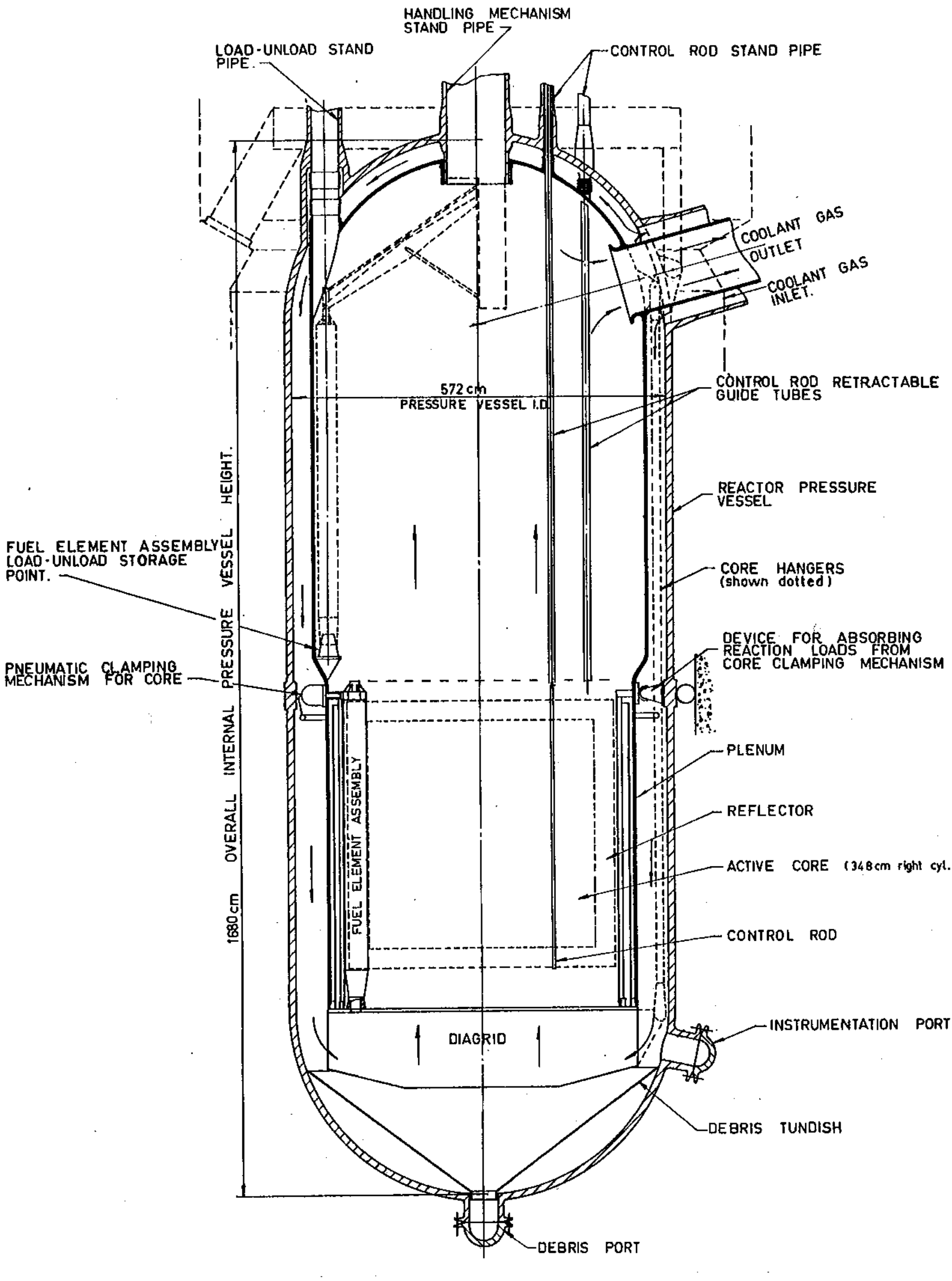

FIGURE 8. SCHEMATIC LAYOUT OF A PARALLEL CHANNEL REACTOR BASED ON A 500 MWt OUTPUT

ERP 91.

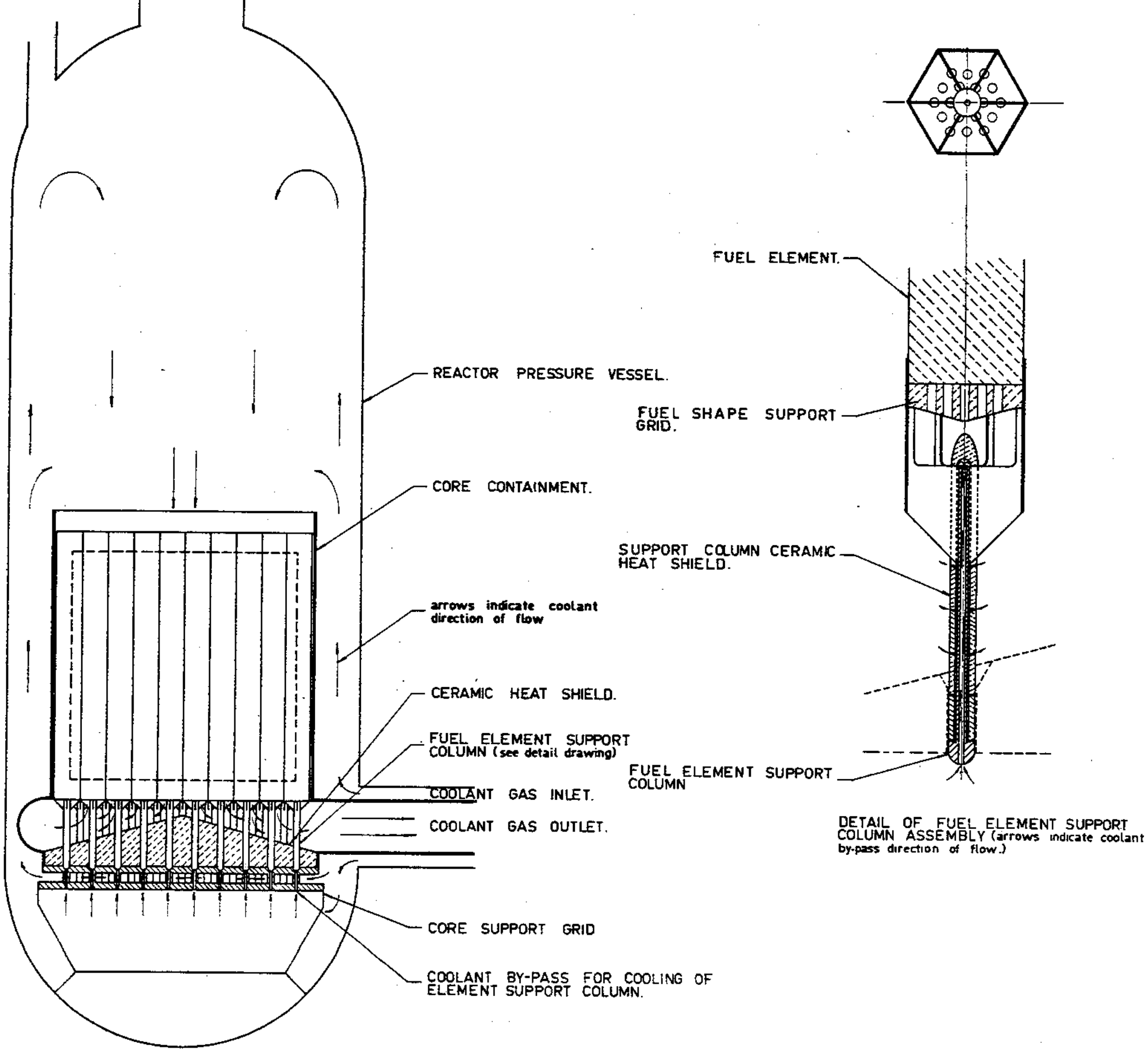

FIGURE 9. SCHEMATIC ARRANGEMENT OF A PARALLEL CHANNEL REACTOR DOWN-FLOW CORE

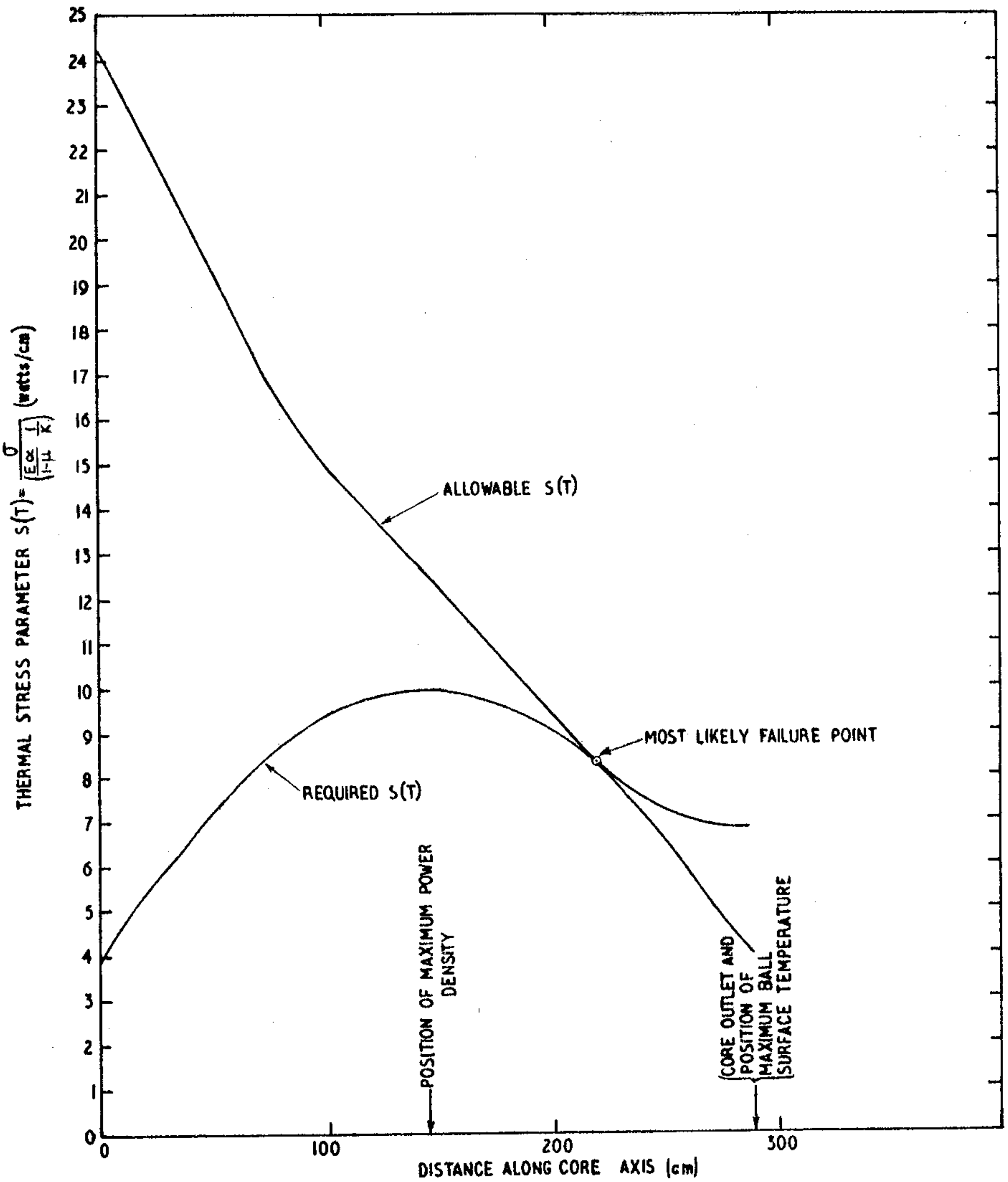

FIGURE 10. TYPICAL THERMAL STRESS RESISTANCE CURVES FOR A DOWN-FLOW PEBBLE BED REACTOR

P650