DR. R. CAIRNS.

UNCLASSIFIED

AAEC/E 122

AAEC/E 122

AUSTRALIAN ATOMIC ENERGY COMMISSION

RESEARCH ESTABLISHMENT

LUCAS HEIGHTS

THE AUSTRALIAN HIGH TEMPERATURE GAS-COOLED REACTOR RESEARCH PROJECT

by

K. F. ALDER

W. H. ROBERTS

Issued Sydney, July 1964

UNCLASSIFIED

AUSTRALIAN ATOMIC ENERGY COMMISSION

RESEARCH ESTABLISHMENT

LUCAS HEIGHTS

THE AUSTRALIAN HIGH TEMPERATURE GAS-COOLED REACTOR RESEARCH PROJECT

by

K. F. ALDER

W. H. ROBERTS

ABSTRACT

A brief history is given of activities at the Lucas Heights Research Establishment, and the reasons that led up to the selection of beryllium oxide for investigation as a promising moderator material in high temperature gas-cooled reactor systems.

The choice of a pebble-bed core for an initial design study of an actual reactor is discussed; the results to date and an outline of the future programme are presented.

CONTENTS

1. INTRODUCTION

The Australian Atomic Energy Commission Research Establishment at Lucas Heights near Sydney, Australia, has as its main research programme a study of the technical and economic feasibility of a High Temperature Gas-Cooled Reactor (H.T.G.C.R.).

The concept which is under detailed study features a semi-homogeneous core of the "pebble bed" type, moderated by beryllium oxide and using a dispersion type fuel. This was selected as the most promising core design for beryllium oxide moderation, after a study over the past few years of a wide range of possible systems and configurations. A large part of this research has been concerned with the fabrication and properties of materials.

2. BRIEF HISTORY OF THE A.A.E.C. PROJECT

The study of H.T.G.C.R. systems moderated by beryllium or beryllia was originated by the initial scientific staff for the Lucas Heights establishment, during the years 1954 - 57 when a large Australian contingent was attached to the Atomic Energy Research Establishment, Harwell, U.K. The task of selecting a future research programme for Australia in the field of power reactors was approached on the principles that:

(i) Australia could not mount a large effort compared with countries with larger population and resources.

(ii) There was little point in selecting a system already under detailed study in an advanced country.

(iii) Australia needed a promising project in order to train nuclear scientists and engineers and retain their interest and enthusiasm.

(iv) The choice of a system not studied elsewhere would enable Australia to contribute original material to world knowledge in atomic energy.

(v) This in turn would encourage improved overseas relations and interchange of knowledge and personnel.

A prime object of the Australian programme was to recruit, train, and maintain in Australia a body of personnel experienced in atomic energy, in order to introduce the benefits arising from developments in nuclear science and technology, to keep abreast of overseas developments, and to advise and assist in the eventual planning for the introduction of nuclear power.

This object is being fully realised, and in addition the results of the research to date give promise of the development of an advanced reactor system of intrinsic technical and economic merit.

Initially two types of reactor system were chosen for study. One was a sodium slurry system (Alder 1958), the other was a gas-cooled system, and some work was done on both by Australian scientists at A.E.R.E. Harwell, and in the first buildings erected at Lucas Heights in 1956 - 57. However, by mid-1958 it was realised that the level of effort available in Australia could support only one project, since the Research Establishment had many other responsibilities such as for radioisotopes, radiation biology, instrumentation, and other allied fields. Australian experience can provide useful background for other countries entering the field.

2.1 Preliminary Research Phase - Sorting Tests on Materials

In 1958, when research on H.T.G.C.R. systems began on a small scale at Lucas Heights, it was decided that no attempt would be made to do conceptual or parametric design studies in any detail on a reactor until the most promising fuel and moderator materials had been selected (Dalton 1958).

There is no pressing need for nuclear power on a large scale in Australia, so it is possible to contemplate a long-term programme towards a highly advanced system. The general criteria which were set for this system were the use of modern steam conditions, with some thought to future development towards gas turbines, low capital and fuel costs, simplicity, inherent stability, and safety (Roberts 1958).

Moderation by the beryllium atom was concentrated on because:

(i) The nuclear properties of beryllium were considered to offer promise of compact cores and good neutron economy, and hence a trend towards low capital cost if the advantages of the metal or its oxide could be shown to outweigh their own high costs.

(ii) Beryllium and beryllia appeared to be the only suitable materials, other than graphite, for use at high temperatures in a dispersed fuel system. They both showed promise of suitability for all or nearly all components of reactor cores, that is, moderator, fuel matrix, fuel cladding, and construction materials, the object being to eliminate parasitic neutron absorbers as far as possible.

(iii) Both were thought capable of development as fission product retentive matrixes for dispersion fuels, and both had adequate thermal conductivity for high rating of such fuels. Beryllia in particular had outstanding thermal conductivity as a ceramic although the effects of irradiation on this and other properties were unknown.

(iv) At that time, no other country had elected to study this field in detail for civil power purposes.

Accordingly, research was begun on beryllium and beryllia as possible moderators, and on possible fuels with high integrity, capable of operating at high temperatures, ratings, and burnups.

The sorting tests on reactor materials, which led to the evolution of the concept to the present beryllia moderated pebble-bed type, are described in detail in publications by Commission staff. Recent research on the most promising materials is discussed by Smith (1964) and many aspects of the detailed work will be published in 1964 in the Proceedings of the 1963 International Conference on Beryllium Oxide held at Newport, Australia (North-Holland Publishing Co.). A summary only of the overall programme follows:

2.2 Moderator

Fabrication routes involving cold pressing and sintering, hot pressing, and high speed extrusion for beryllium metal, using imported metal powders, were developed at Lucas Heights (Wright and Silver 1961). Studies were made of the effects on the mechanical and physical properties of powder analysis and characteristics, and fabrication route, fabrication variables, and neutron irradiation. Particular attention was paid to the corrosion of beryllium by carbon dioxide containing traces of water vapour, including tests under irradiation. All this work covered the possible use of beryllium metal as matrix and cladding for dispersion fuels and as moderator.

The general conclusions were:

(i) The corrosion of beryllium by carbon dioxide of purity to be reasonably expected in a reactor environment, particularly in relation to water content, may be made acceptably low at gas temperatures up to about 700 °C, by suitable choice of composition and fabrication route (Smith et al. 1961, Smith 1960). In particular, raising of the oxide content of the metal under carefully controlled conditions improves this resistance. However, there is little prospect of adequate performance above 700 °C.

(ii) The production of helium within beryllium metal as a product of the n,2n reaction, leads to formation of helium bubbles at high temperatures following high fast neutron doses. The extent and distribution of this bubble formation depends upon the composition and fabrication history of the metal. It does not seem possible to eliminate deleterious changes in mechanical properties due to helium bubbles, and, at temperatures above 600 °C, swelling due to these bubbles could reach unacceptable levels at fast neutron doses in excess of 10^{21} nvt (Alder 1956, Hickman 1961, Hickman and Chute 1963, Chute 1963, Hickman and Stevens 1963).

(iii) The high cost of fabricated beryllium metal is a deterrent. The above limitations on its performance in a reactor make it unlikely that the advantages of using beryllium as a moderator in a power reactor system will outweigh its high cost. This is not to say that beryllium has no prospects in the much smaller quantities which are involved in its use as a fuel cladding material.

For these reasons, research on beryllium metal as a possible moderator for H.T.G.C. systems ceased at Lucas Heights in 1960, and attention was concentrated on beryllium oxide, on which studies had been proceeding in parallel.

Research on beryllium oxide followed a similar pattern of studies on the fabrication, properties, corrosion resistance, and irradiation stability, with prime emphasis on obtaining an understanding of basic mechanisms in order to be able to produce better materials. A detailed summary of progress in this field is given in Section 3.3 below. Considerable success has been obtained in elucidating these mechanisms, and hence in explaining not only the results of work at Lucas Heights, but also the unpromising results of other workers, particularly on irradiation damage. Generally, experience with beryllia at Lucas Heights to date indicates that its properties are sufficiently promising to warrant proceeding with the conceptual design of a beryllia moderated system.

2.3 Fuels

Initially four dispersion fuels were selected for study. These were (Dalton 1958):

(i) $UO_2 - ThO_2$ dispersed in beryllium

(ii) $UBe_{13} - ThBe_{13}$ in beryllium

(iii) $UC_2 - ThC_2$ in impermeable graphite

(iv) $UO_2 - ThO_2$ in beryllia.

In each case the fissile material could be Pu239, U235, or U233, and the fertile material either Th232 or U238. For materials research purposes the U-Th systems were studied for simplicity.

The dispersions of $UO_2 - ThO_2$ in beryllium metal were eliminated as a result of studies by Hanna (1961) who demonstrated the results of the thermodynamic instability of the dispersions above 600 °C.

Early research on $UBe_{13} - ThBe_{13}$ dispersions in beryllium gave promising results (Hanna et al. 1961, Hanna and Turner 1963). However irradiation testing at temperatures above 650 °C demonstrated that the burnup of this fuel would be inadequate, because of the onset of swelling and disintegration caused by fission product release (Hanna et al. 1962, 1963). The irradiation experiments were done in a predominantly thermal flux, without the complication of fast neutron damage and the effects of the n,2n reaction in the beryllium, which were under separate study. These results, with those described above from the work on beryllium moderator, were sufficiently discouraging to stop further work on this class of fuel.

The dispersions of the mixed carbides of fuel/fertile materials in impermeable graphite gave promising results (Warner and May 1961), but the work was discontinued because of the even more promising results being obtained with mixed oxide dispersions in beryllia, which was emerging as the preferred moderator. In any case the Australian effort on graphite matrix dispersion fuels was small in relation to the amount of work being done in Europe and the United States, and was undertaken only to provide a background in this technology in case dispersion fuels based on beryllium or beryllia proved not feasible.

2.4 The Choice of the Present Concept

By early 1963, sufficient evidence had been accumulated to warrant more detailed consideration of a reactor based on beryllia. More attention was therefore given to design concepts and to the choice of fuel and fertile materials and the overall fuel cycle.

One possible approach was the use of a fixed beryllia moderator block with coolant channels containing fuel compacts encapsulated in a high temperature–resistant alloy. However, the parasitic neutron absorption and other limitations imposed by the use of metals or alloys in the core would partly offset many of the potential advantages of beryllia. Therefore it was decided to concentrate on all–ceramic cores of the greatest possible simplicity, in which an absolute minimum of parasitic construction materials would be used. Design studies were commenced on a wide range of possible core configurations, covering both "parallel–flow" and "pebble–bed" type cores.

Studies of possible parallel–flow cores led to the conclusion that it would be very difficult, if not impossible, to design a core which did not contain a considerable amount of metallic construction material. One criterion which had to be met was the need to remove all of the beryllia from the core at some stage of its life, because even the most promising irradiation results on beryllia do not predict a 20 year lifetime for fixed moderator components. This meant that at some stage the removal of heavy assemblies of inherently fragile material would be required. Another difficulty was the possible distortion of ceramic components of large assemblies, where again the fragility of the components would cause difficulty in their removal. Moreover ceramics can be machined only by expensive grinding operations, and it did not seem reasonable to consider shapes requiring dimensional precision beyond that attainable by conventional ceramic fabrication processes.

On the other hand, the pebble–bed concept has many advantages for an all–ceramic system. Removal and replacement of "fuel elements" is relatively easy in principle, their possible distortion and dimensions do not appear to be critical, and balls lend themselves to mass production by the remote handling techniques that may be required for fuel element fabrication. A further advantage is the possibility of attaining core flexibility and maximum burnup by the recirculation of pebbles.

Investigation of fuel cycles centres around the use of U233, U235, or Pu (commercial) as the fissile element and Th232 as the fertile element. Pu and U233 are of more interest than U235 for two reasons:

(i) U235 is produced in diffusion plants whereas U233 and Pu can be manufactured in power reactors.

(ii) U233 offers the possibility of thermal breeding and commercial plutonium has interesting potential with regard to fuel lifetime because of the presence of Pu240 and Pu241 (Puechl 1962).

We chose to study Pu initially, largely because of the paucity of U233 nuclear data and the world scarcity of U233.

3. PRESENT STATUS OF THE PEBBLE–BED FEASIBILITY STUDY

3.1 Parameters of the System

The initial investigation has been confined to a reactor of a given output and the following parameter ranges.

Heat output (MWT)	500
Fissile rating (av. MWT/kg original fissile material)	1 – 3
Core power density (kW per litre)	8 – 20
Coolant gas	CO_2
Coolant gas outlet temperature (°C)	700 – 800
Coolant gas inlet temperature (°C)	350 approx.
Coolant gas pressure (atmospheres)	35
Fuel – PuO_2 and ThO_2 dispersed as 150 micron composite particles in a BeO sphere	See Figure 1
Max. fuel surface temperature (°C)	up to 1100

Carbon dioxide has been selected as the coolant largely because it is compatible with BeO up to high temperatures, is cheap and readily available, and requires blowers of only moderate size.

3.2 Main Problems

At the time of writing this paper the decision to study the pebble-bed scheme is little more than 9 months old. Although materials experimental work is continuing, the engineering and physics experimentalists have barely had time to determine their programmes and commence the gathering together of equipment. Results to date in the physics and engineering fields, therefore, follow mostly from preliminary theoretical considerations.

Since there is little published data relating to the main problems to be solved, a large amount of work must be done before solidly based conclusions can be reached. Main factors to be considered are:

(i) The balls must not weld together in the core.

(ii) There should be insignificant wear of the ball surfaces during their passage through the reactor.

(iii) Thermal stress (including cycling and thermal shock effects) must not cause ball failure.

(iv) The extent of the release of fission products and tritium into the gas stream.

(v) The lifetime of the fuel elements from both the materials and reactivity viewpoints.

(vi) The heat transfer, gas pressure drop, and gas flow characteristics through beds of BeO balls.

(vii) The design of the bottom support grid. With an upflow core the power density would be limited by levitation to about 5 watts/cm^3 unless stress resistance of the BeO, primary circuit gas pressure, and/or gas temperature rise through the bed can be raised. Since, for good economics, a power density of 10 watts/cm^3 or greater is required, a downflow bed is necessary with the present assumed parameters. This requires a grate which will hold the balls, allow the gas to pass through, act as a reflector, withstand the pressure drop forces, operate at temperatures greater than 900 °C, and also last the life of the reactor unless some satisfactory method can be devised for removing it.

(viii) The nature of the packing of beds of moving BeO balls subjected to a forced circulation of gas. Theoretically there is an infinite number of possible packings between about 26 and 48 per cent. voidage, and it is essential to determine the equilibrium average value of this packing in the operating reactor, and have an accurate idea of probable local variations in voidage. The latter in particular can affect temperature/stress considerations in the pebbles.

(ix) The setting up of sufficiently accurate methods of calculation for the solution of the numerous physics problems of the core, and the assessment of nuclear data from which physics calculations will be made. There are some gaps and experimental work is being done.

(x) Control and transient analysis.

(xi) The economics of the system.

3.3 Materials Considerations

Materials considerations are dealt with in detail by Smith (1964).

Briefly:

(i) Beryllia, of grain size around 5 microns appears capable of withstanding irradiation doses corresponding to high burnup (2-3 fissions per initial fissile atom) of the dispersed fuel, without significant deterioration of its physical properties.

(ii) The modulus of rupture, which is a critical property in determining maximum ball size seems large enough (25000 - 32000 p.s.i. at room temperature) to permit the use of about 1 inch diameter balls, which are acceptable from core gas pressure drop and mechanical handling considerations.

(iii) Release of fission products through to the outer surface of the fuel element seems to be low enough to indicate that shielding of the circuit should not be a problem of great significance, provided that the integrity of the balls is maintained. Tritium release, however, will probably be large enough to necessitate the inclusion of some means of cleanup in the reactor design.

In connection with the question of the possible welding together of the balls in the reactor core, theoretical considerations indicate that this is unlikely, and a preliminary experiment, in which a group of 5/8 inch (1.6 cm) diameter balls at 1000 °C was subjected to a loading of 60 lb (27 kg approx.) for 15 weeks, has not shown the slightest indication of adhesion of the surfaces.

Some initial work has also been done on the wear of BeO surfaces in contact and the results suggest that this will be acceptably small under reactor conditions.

These experiments are encouraging and more elaborate tests on adhesion and wear are in progress.

3.4 Core Nuclear Studies

To date these studies have been preliminary and have concentrated on a fissile/fertile/moderator atomic ratio of 1:10:2000. The object has been to delineate the theoretical problems to be solved and the gaps in the data to be filled by the experimental programme. Even with the simple techniques used to date it appears that:

(i) Burnups approaching 250 per cent. appear feasible. In order to achieve reasonable overall unit power charges the fuel contribution should preferably be less than 0.2 pence/kWh, and on the basis of present day costs this would require a burnup of at least 200 per cent. (Wright 1963) if the irradiated fuel is assumed to have no value.

(ii) There is little plutonium left at the end of the fuel lifetime and in the equilibrium core of the pebble-bed system there are indications that U233 is the predominating fissile isotope. It is thus necessary to undertake experimental work on U233 systems.

(iii) The large changes in isotopic composition with irradiation pose problems associated with the neutron spectrum, but these may be mitigated by fuel handling schemes peculiar to the pebble-bed design.

(iv) With the promise of long irradiation lifetime and therefore the accumulation of relatively large quantities of fission products has come the necessity for a more accurate theoretical and experimental determination of fission product data.

(v) In order to achieve economic burnups for the mean core temperatures envisaged and taking into account BeO irradiation dose limits, the moderator/fuel atom ratio should not be less than about 1200 (as shown in Figure 2).

The approximate results so far obtained have been sufficiently promising to warrant further and more thorough studies of the plutonium fuelled pebble-bed system.

3.5 Control

On the basis of a calculation using an initial atom ratio of 1:10:2000 the temperature coefficient of reactivity was found to be strongly negative, being about 16×10^{-5} $\Delta k/k$ per degC. However, for the corresponding equilibrium core composition it is only about one fifth of this. The approach to equilibrium therefore must be via a fuel management route, (for example, different initial composition) that does not have large negative temperature coefficients since otherwise the provision of suitable control would be difficult.

The requirement of shutting down to cold sub-critical means that control absorbers must be provided to cope with temperature coefficient, possible accidents (such as ingress of steam from boiler tubes into the core) and xenon effects. This in turn leads to the location of control rods in the core which incidentally introduces problems of voidage in the bed of pebbles.

3.6 Overall Economics

The overall economic picture takes into account the burnup studies referred to above and includes an estimate of the capital cost. Again the calculations are a preliminary assessment but the results are encouraging. The most important tentative conclusions are:

(i) A capital cost of £A120 - £A130 per kW, and possibly lower, in a 200 MWE (500 MWT) installation should be attainable. A major contribution results from the high gas outlet temperature (about 750°C) which produces favourable steam conditions and a high cycle efficiency.

(ii) If burnups of about 250 per cent. can be attained, and assuming some modest improvements in BeO costs predicted recently at the International Conference on BeO, a unit cost of less than 0.6 pence/kWh at 80 per cent. load factor looks possible at prevailing Australian interest rates. This compares favourably with predicted energy cost for pressurised or boiling water installations in Australia on the same basis; it must be emphasised that this is a very preliminary estimate for a 200 MWE system, and that there appears to be considerable promise of lowering these costs by further research and development.

(iii) The cost of energy per kWh seems surprisingly insensitive to core power density over a wide range of power densities for a given initial composition. This result is still under close scrutiny but if it is subsequently confirmed an extra degree of freedom is added to the determination of the optimum system.

3.7 Ball Size and Pumping Power

All the above has tacitly assumed that a reasonable ball size can be calculated accurately for a given ball power density. The success of the pebble-bed design is largely affected by the ball size attainable. Small pebble sizes result in high pumping power and may introduce problems of blockage in the grate. Assuming that elastic theory is applicable, the ball diameter d is given by:

$$d^2 = \frac{R(T)}{q} \; \frac{60}{F} ,$$

where $R(T)$ = thermal stress resistance of material

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

σ = maximum allowable stress,

k = thermal conductivity,

μ = Poisson's ratio,

α = coefficient of linear expansion,

E = Young's modulus,

q = ball power density at design point, and

F = a factor to allow for local variations of temperature, voidage, etc. in the core.

The value of F is not yet available. It can be determined approximately, however, from the results of pebble bed studies carried out elsewhere (Fraas 1961, General Atomic 1963) and is initially assessed at 3.5. Based on this and sinusoidal power distribution we find that a 1 inch diameter ball for example should sustain about 10 watts/cm^3 core power density at 40 per cent. core voidage. It is likely that stress relaxation effects in BeO will enable the ball size to be increased, but sufficient information on this is not yet available.

Pressure drop through the core has been determined using friction factor correlations from overseas work. Pumping power versus ball size is plotted in Figure 3 for gas pressures of 40 to 60 atmospheres. Clearly the pumping powers are acceptable in the power density range of interest.

3.8 Design of the Reactor

Currently four fuelling schemes, all with downflow of the gases are being considered (Figure 4). Scheme A assumes that after a certain operating time the whole bed is replaced, requiring the reactor to be shut down for a period. In Scheme B the balls are rejected and replaced at a rate determined by a maximum estimated burnup. Schemes C and D require burnup detectors since they reject only those balls which have reached the maximum allowable burnup and recirculate those which have not.

On preliminary nuclear data, arrangement D seems to offer the best fuel economy by holding up the circulated fuel for a period to permit decay of protactinium and xenon. It is not yet clear, however, whether the extra complication of apparatus required by C and D will largely offset the advantages of these over B. Obviously B, C, and D will permit continuous reactor operation.

Actual engineering designs have been reduced from quite a large number originally considered to the four illustrated by Figure 5. These are being dealt with in greater detail to determine their technical difficulties and economic worth.

Because of the difficulty of grate support, variation in pebble residence times, probable flux flattening requirements, and the problems of in-core control rods, a seven-outlet (rather than a single outlet) arrangement has been adopted. This also facilitates core dumping for safety.

The grate design with downflow of the gases has proved to be difficult, mainly owing to the heavy loading and high temperature. Two concepts have been examined. The first, all ceramic, will withstand the temperature but is doubtful structurally under temperature cycling and would require thorough experimental investigation. In the second it is proposed to overcome the structural problem by the use of insulated steel members cooled by the inlet gases. This is thought to be a feasible arrangement, subject to experimental checking of insulation integrity.

The reflectors have also posed a problem since beryllia is expensive and suitable only for use at high temperatures because of irradiation damage. A compromise design has been produced using beryllia beneath the core and graphite cooled by the inlet gases for the top and side reflectors. It is proposed to line the lower (hotter) part of the radial graphite reflector with ceramic bricks.

Some work has been done on designs employing upflow of coolant through the core. The practicability of these is not yet certain, but if lower power densities are permissible consistent with economy, it may be possible to raise both gas pressure and core outlet temperature, and bypass some gas around the core to cope with the levitation limit, mixing it with the core exit gas in order to preserve the same gas temperature at the heat exchanger inlet. A concrete pressure vessel would almost certainly be required, to cope with high gas pressures.

The thermal stress in a ball is not quite clear since superimposed stresses can be generated by irradiation growth. The growth is lower at higher temperatures so that the ball expands less in the centre than at the outer regions. If creep effects are small the initial tensile stress on the ball exterior could change (through zero) to a compressive stress as burnup proceeds, giving a tensile stress at the centre. Ultimate failure could then occur at the centre of the ball with a consequent rise of centre temperature and increased release of tritium and fission products. However, high temperature stress relaxation experiments are presently showing very high creep rates in fine grained beryllia, and this could change the whole situation.

3.9 Theoretical Physics

Most of the theoretical effort during the years 1958 to 1961 was devoted to the numerical and analytical calculation of spectra in a variety of moderators. Analytical studies of the moderation of neutrons in the vicinity of source energies led to a generalisation of the Fermi age equation (Pollard 1960, Keane 1961) and provided a simple technique for the inclusion of the effect of anisotropic scattering at high energies (Keane and Pollard 1962). The numerical evaluation of the neutron spectrum in beryllia included a detailed treatment of the (n,2n) and (n, α) reactions (Axford et al. 1964). At a later date an interesting application of renewal theory to the study of neutron moderation (Wilkins 1962) provided both a deeper understanding of the process and many novel results, which are of particular value in the planning of Monte Carlo calculations.

The resonance absorption of neutrons has been the subject of continuous investigation. Studies have included the exact evaluation of absorption in an isolated resonance using the code PEAS (Pollard 1964a), the investigation of simple models (McKay and Pollard 1963), and the provision of an approximate formula for the determination of the J function (Doherty 1963). Recently the Doppler broadened contour functions have been extended to the complex domain (Keane and Clancy 1964), allowing more powerful mathematical techniques to be applied to the treatment of resonance absorption.

The methods developed in the study of neutron moderation and resonance absorption have been included in the two codes MULGA (Clancy et al. 1963) and GIBBA (Pollard 1964b). MULGA gives multigroup averaged cross sections, while GIBBA treats burnup without the complication of spatial dependence. Continued development of these codes provides a stimulus and focal point for research into the many theoretical problems arising from the present pebble-bed feasibility study.

3.10 Experimental Work

Physics and engineering experimental work relevant to this reactor system is just beginning, and is summarised below:

3.10.1 Engineering

The main problems are briefly the investigation of bed packings, gas pressure drops, ball flow and heat transfer characteristics, together with thermal cycling and thermal shock effects.

Experiments designed on a statistical basis are being carried out to investigate the effect of the relevant variables on ball flow and packing in a pebble bed. Zirconium silicate spheres are used, and the various ball configurations will be fixed in wax and cut into sections for examination of local packings through the use of radiation absorption and other methods.

The same sectioned parts of the bed (after removal of the wax by melting) can be employed to determine pressure drops, and to build up models for heat transfer experiments on groups of BeO balls. Glass and steel balls are being used for preliminary tests on velocity patterns and ball handling principles and devices.

Heat transfer experiments are also being made on a copper sphere surrounded by wooden balls to find the interference with the heat transfer pattern, and on a heated ball of similar conductivity to BeO in a bed of similar balls. This work, combined with a mathematical analysis, could lead to a better understanding of thermal stresses in the bed.

3.10.2 Physics

The physics experimental programme is divided broadly into three parts, utilising the Commission's 3 MeV Van de Graaff accelerator and the 10 kW reactor MOATA. These are:

(i) Exponential experiments to give information on critical sizes.

(ii) Effects on reactivity of such items as fission products, structural materials, the Pu240 and 241 contents of the fuel, Li6 and U233 which are produced during ball lifetime, and Th and Pu particle size.

(iii) Determination of cross sections.

Exponential experiments have been made on simple lattices of U235/BeO as a check on the cross section data and the calculation method (Duerden et al. 1964). Measurements are being extended to BeO/U238/U235 and BeO/Th/U235 lattices, and later to plutonium-fuelled lattices.

In these experiments the fuel is heterogeneous in form whereas in the actual reactor it is dispersed as fine particles, for which due allowance must be made.

In connection with (ii) an internal lattice will be built into MOATA with a neutron spectrum similar to that in a critical assembly but with differences which can be allowed for by calculation. Doppler coefficient measurements will also be carried out by inserting heated samples into the reactor centre. These integral experimental results will be used to check theoretical estimates of reactivity changes for any particular material.

4. CONCLUSION

The Australian Atomic Energy Commission has built up a Research Establishment at Lucas Heights with most of the facilities, including two research reactors, necessary to pursue a feasibility study on an advanced reactor concept.

A prime aim has been to establish nuclear technology in Australia, and the H.T.G.C.R. programme is the main theme adopted to meet this aim.

The feasibility study consists of two phases:

(i) Sorting tests on reactor materials leading to selection and development of the most promising fuel and moderator materials based on the element beryllium.

(ii) Selection and study of the most promising reactor system based on these materials, and comparison of this system, both technically and economically, with present reactors and other advanced concepts.

The study has recently entered the second phase. It is expected that the initial feasibility study will be completed in another one to two years. If it remains promising, it will require several years more of research and development, particularly in engineering, before a reactor can be designed in detail.

Present indications are that the beryllia moderated gas-cooled pebble-bed reactor has a number of outstanding scientific and engineering attractions, particularly long burnup fuel, its departure from the precision engineered types of core in current use, stability of fuel at high temperatures, the absence of parasitic neutron absorbers in the core, and high gas temperatures (and hence high cycle efficiency).

It remains to be seen how much these advantages offset the high cost of beryllia, but there are good prospects that the system will be more attractive both technically and economically than present generation power reactors.

5. ACKNOWLEDGMENTS

The authors' thanks are due to those members of the Lucas Heights staff who provided information and critical comments during the writing of this paper.

6. REFERENCES

Alder, K.F. (1956).- Growth of Beryllium under Irradiation. AAEC/E6.

Alder, K.F. (1958).- Liquid Metal Fuel Reactors. Symposium on the Peaceful Uses of Atomic Energy in Australia, June 1958. A.A.E.C.

Axford, T., Hines, K.C., Pollard, J.P. (1964). - Slowing Down of Neutrons in Beryllia. Reac. Sc. & Tech. 18 (3) : 131.

Chute, J.H. (1963). - Electron Microscope Studies of Irradiated Beryllium Metal. AAEC/E108.

Clancy, B.E., Doherty, G., Keane, A., Kletzmayr, E., Pollard, J.P. (1963). - MULGA, A Complex of Codes for the Determination of Multigroup Averaged Neutron Cross Section Data. AAEC/E114.

Dalton, G.C.J. (1958). - Gas-cooled Reactors of a Mixed Fuel-Moderator Type. Symposium on the Peaceful Uses of Atomic Energy in Australia, June 1958. A.A.E.C.

Doherty, G. (1963). - An Approximation to the Doppler Broadened Function $J(\theta,\beta)$ for Resonance Absorption Calculations. Reac. Sc. & Tech. 17.

Duerden, P., McCulloch, D.B., Britliff, E. (1964). - Buckling and Integral Spectrum Measurements on U235/BeO Subcritical Assemblies. AAEC/E123.

Fraas, A.P. (1961). - Preliminary Design of a 10 MW(t) Pebble-Bed Reactor Experiment. ORNL CF 60-10-63.

General Atomic (1963). - Advanced BeO Concepts Report - GA3783.

Hanna, G.L. (1961). - The Compatibility of Beryllium with Oxides of Uranium and Thorium. Conference on the Metallurgy of Beryllium, Institute of Metals, London, 1961.

Hanna, G.L., Hickman, B.S., Hilditch, R.J. (1962). - The Irradiation Behaviour of Beryllium Based Dispersion Fuels. AAEC/E96.

Hanna, G.L., Hickman, B.S., Hilditch, R.J. (1963). - The Irradiation Behaviour of Beryllium Based Dispersion Fuels. AAEC/E112.

Hanna, G.L., Turner, D.N. (1963). - Preparation of the Beryllides of Uranium and Thorium. AAEC/E102.

Hanna, G.L., Turner, D.N., Smith, P.D. (1961). - Preparation of Beryllium Based Dispersion Fuel Specimens for Irradiation Testing. AAEC/TM113.

Hickman, B.S. (1961). - The Effects of Neutron Irradiation on Beryllium Metal. Conference on the Metallurgy of Beryllium, Institute of Metals, London.

Hickman, B.S., Chute, J. (1963). - The Behaviour of Helium in Irradiated Beryllium. J. Aust. Inst. Metals, 8 (2).

Hickman, B.S., Stevens, G.T. (1963). - The Effect of Neutron Irradiation on Beryllium Metal. AAEC/E109.

Keane, A. (1961). - Slowing Down from an Energy Distributed Neutron Source. Nuc. Sc. & Eng. <u>10</u>.

Keane, A., Clancy, B.E. (1964). - Doppler Broadened Contour Functions in the Complex Domain. AAEC/TM229 and J. Aust. Math. Soc. (In press).

Keane, A., Pollard, J.P. (1962). - Slowing Down with Anisotropic Scattering. Nuc. Sc. & Eng. <u>12</u>.

McKay, M.H., Pollard, J.P. (1963). - Effect of Temperature Variation on "Intermediate-Resonance" Formulas. Nuc. Sc. & Eng. <u>16</u>.

Pollard, J.P. (1960). - Spectrum Calculations for Neutrons Slowing Down by Elastic Collisions. AAEC/E54.

Pollard, J.P. (1964a). - PEAS, A Resonance Absorption Programme. AAEC/E126.

Pollard, J.P. (1964b). - GIBBA, A Burnup Code. Unpublished A.A.E.C. report.

Puechl, K.M. (1962). - The Potential of Plutonium as a Fuel in Near Thermal Converter Reactors. Nucl. Sc. & Eng. <u>12</u>.

Roberts, W.H. (1958). - Some Major Design Parameters for the H.T.G.C. Reactor Project. AAEC/E10.

Smith, R. (1960). - Some Metallographic Observations on the Oxidation of Beryllium in Wet Carbon Dioxide. I.A.E.A. Conference on Fuel Element Fabrication, Vienna, May 1960.

Smith, R. (1964). - The Development of Beryllia-based Fuels for H.T.G.C. Reactor Systems. Submitted to the Third International Conference on the Peaceful Uses of Atomic Energy, Geneva 1964.

Smith, R., Stuart, W.I., Van Peer, W., Price, G.H. (1961). - Inhibition of Breakaway Oxidation of Beryllium. Conference on the Metallurgy of Beryllium, Institute of Metals, London.

Warner, R.K., May, J.R. (1959). - Graphite Fuel Studies. AAEC/E41.

Warner, R.K., May, J.R. (1961). - Graphite Fuel Studies AAEC/E61.

Wilkins, C.A. (1962). - Renewal Theory and the Slowing Down of Neutrons. Nuc. Sc. & Eng. 14.

Wright, W.J. (1963). - The Contribution of Ceramic Processing Costs to H.T.G.C.R. Fuel Cycle Costs. AAEC/TM141.

Wright, W.J., Silver, J.M. (1961). - The Extrusion of Beryllium. Conference on the Metallurgy of Beryllium, Institute of Metals, London.

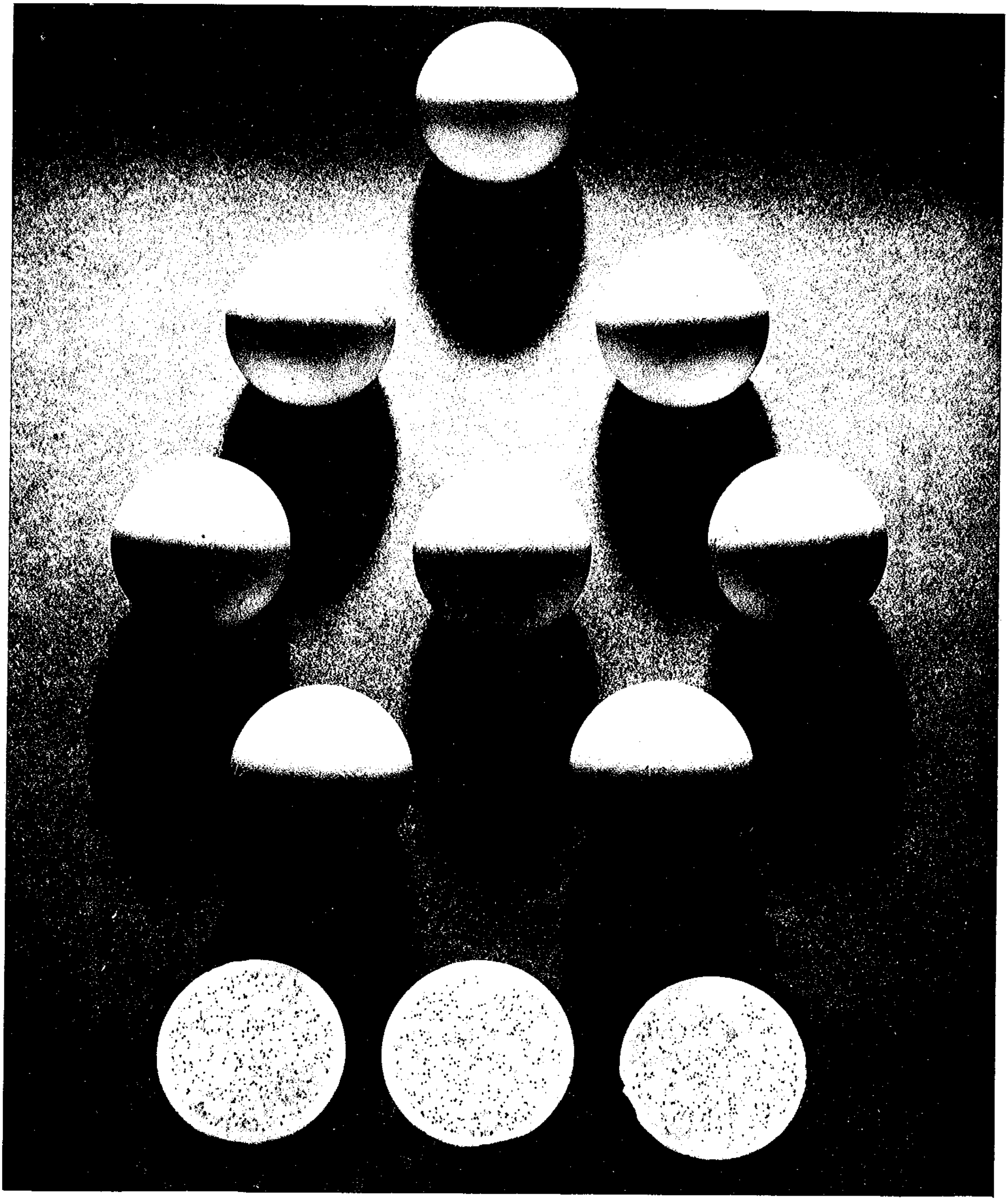


FIGURE 1. SPHERICAL FUEL ELEMENTS FOR PEBBLE BED REACTOR
Dispersion of Fertile and Fissile Materials in BeO.
(Actual size 1 inch diameter).

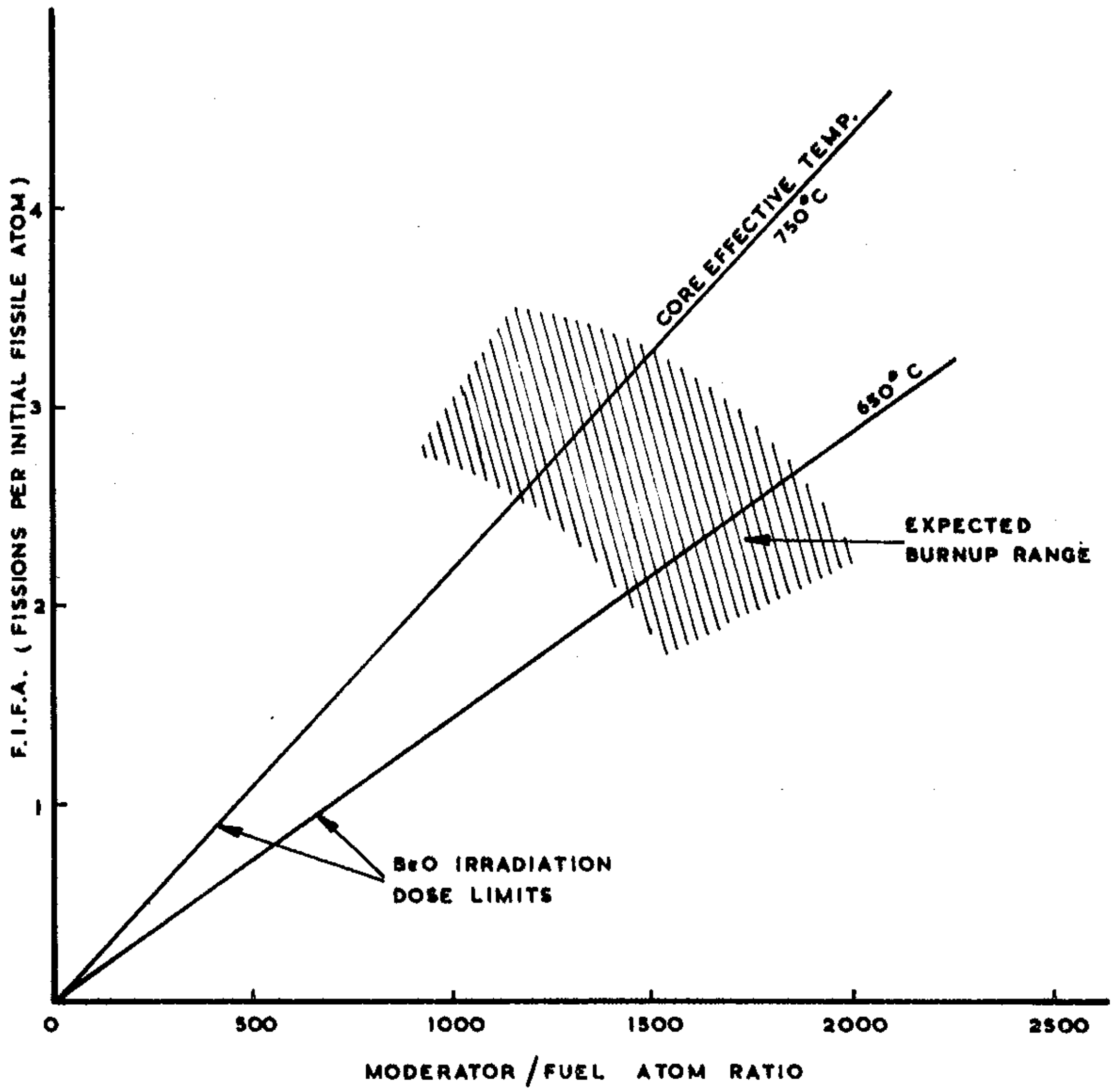


FIGURE. 2. BURNUP LIMIT BY FAST FLUX IN BeO MODERATOR.

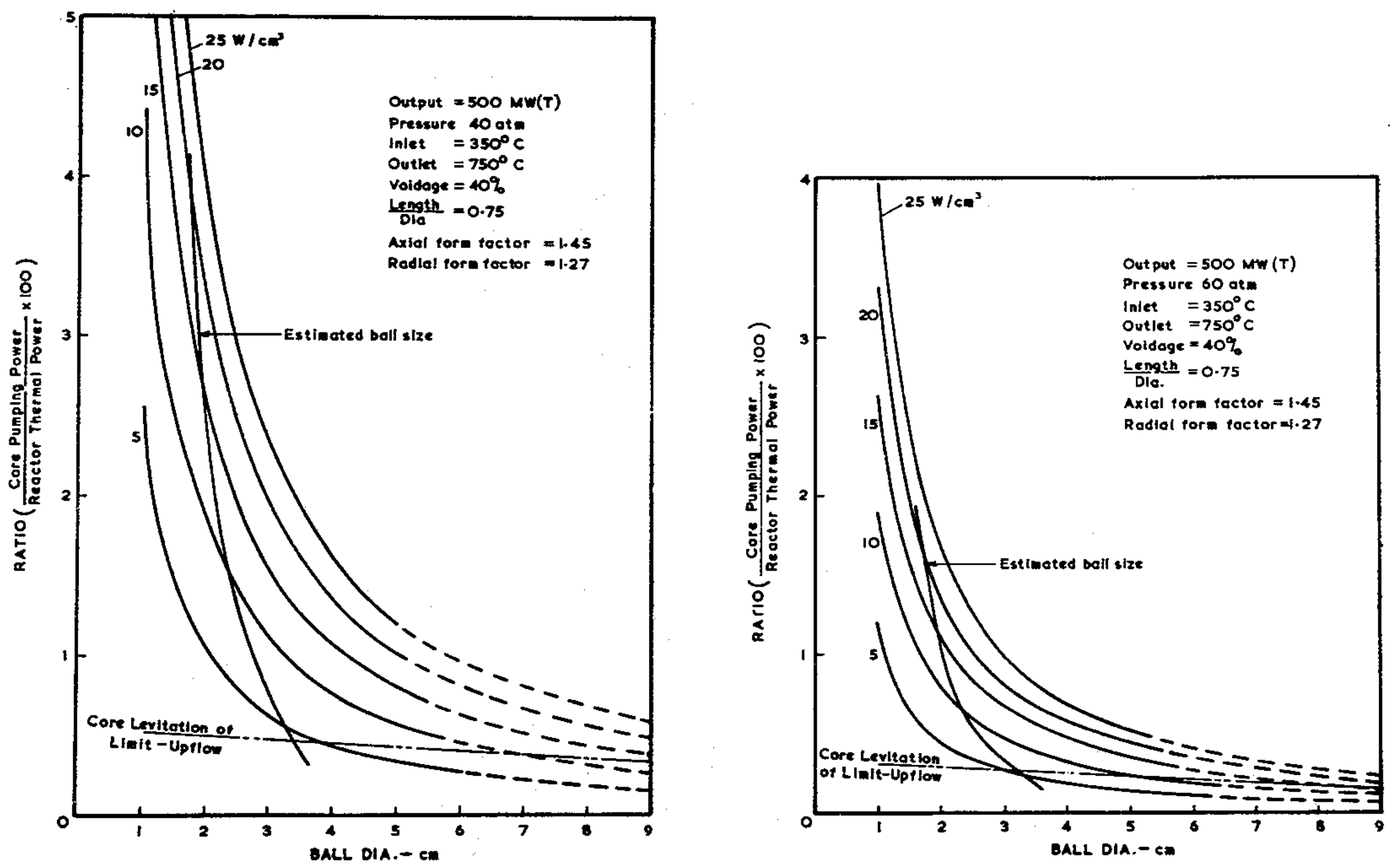


FIGURE.3. RELATIONSHIP BETWEEN BALL SIZE & CORE PUMPING POWER.

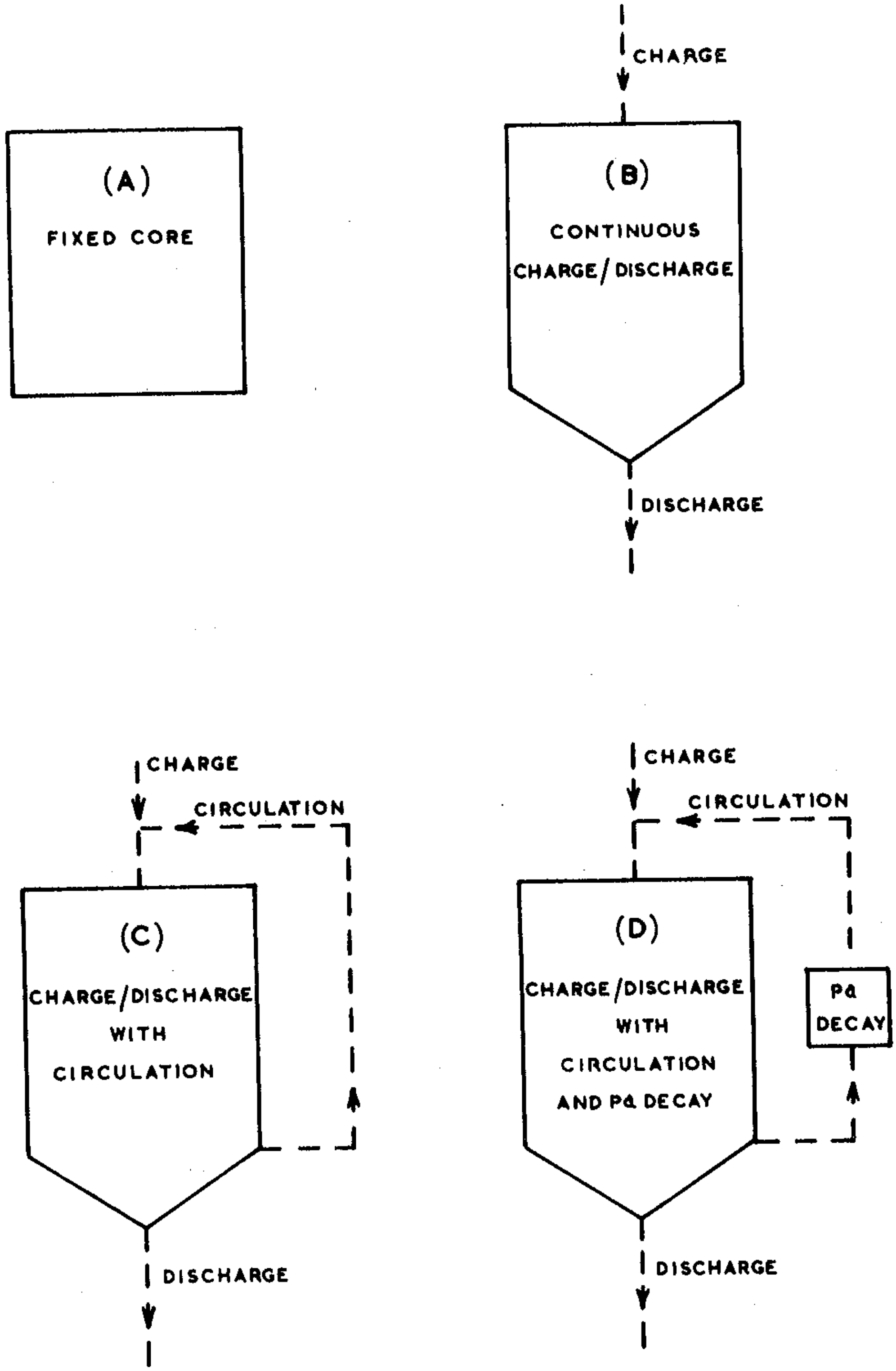
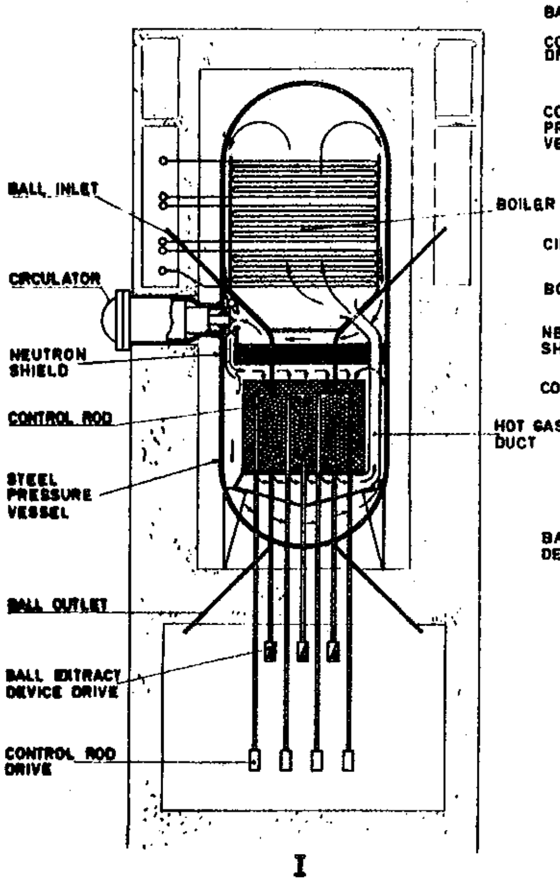
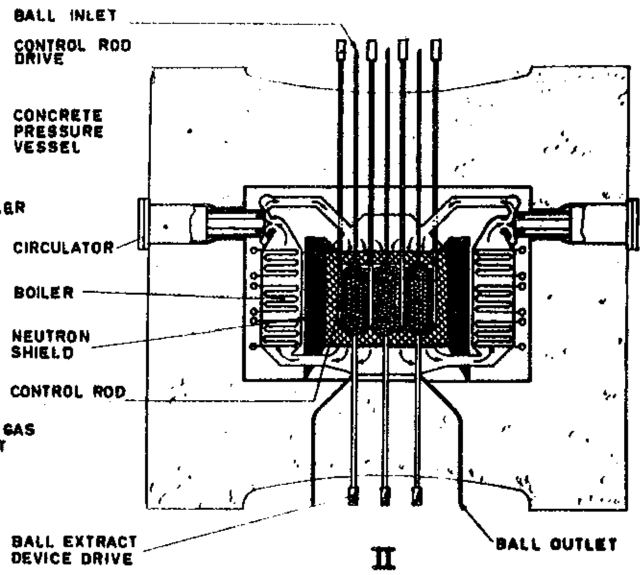
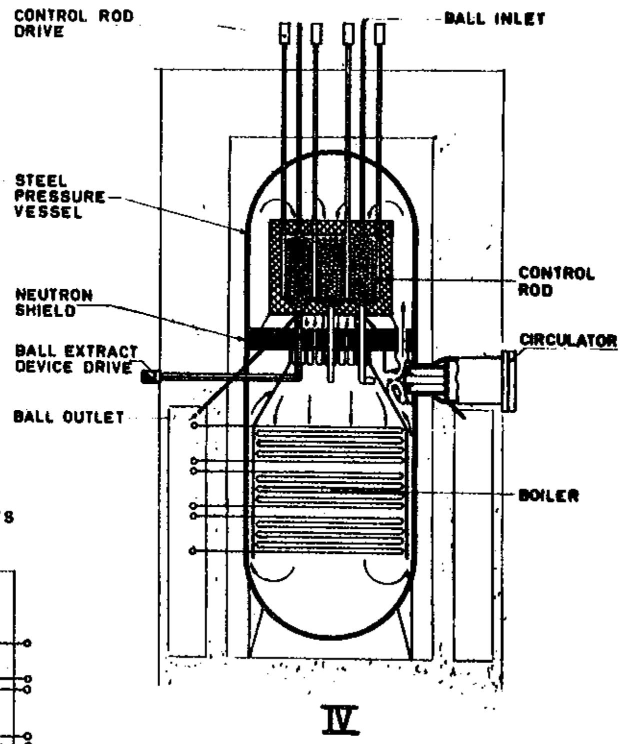


FIGURE .4.   PEBBLE BED SCHEMES.







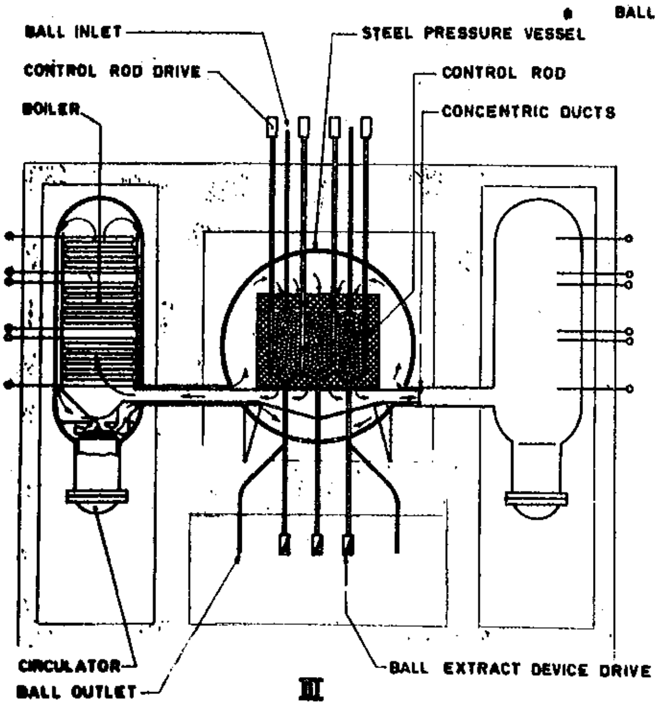


FIGURE 5.

SUGGESTED DOWNFLOW PEBBLE BED REACTOR LAYOUTS.