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Our Ref: FOI2019/08613

[redacted]

22 November 2019

Dear [redacted]

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We attach the following document:

- File AB15/3457 – Nuclear Power Conference 1953 – PIPPA – The First British Nuclear Power Project

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Paper N.P.C. 53/11

Copy No. 54

NUCLEAR POWER CONFERENCE 1953

PIPPA

THE FIRST BRITISH NUCLEAR POWER PROJECT

by

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A.E.R.E. Harwell
August, 1953
/DH

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I. BRIEF HISTORY OF THE PROJECT

In September 1950 a conference was held at A.E.R.E. Harwell to discuss the possibility of generating electric power from the heat of a natural uranium pile. It was decided that plants using natural uranium fuel for the simultaneous production of power and plutonium operating with fuel irradiations of at least 2000 Megawatt Days per tonne, might well constitute the best way of beginning a national scheme for electricity production from nuclear fuels and a feasibility study was recommended.

Accordingly a Harwell design committee was formed (of which the authors were respectively Secretary and Chairman) and the problem was examined in a general way. After considering different types of reactor with various combinations of moderator and coolant, it was decided that a graphite moderated reactor cooled by gas under pressure was the best proposition for evolving into a practical design within a reasonable time.

When this conclusion was reached it was thought that a small reactor using about 35 tonnes of natural uranium for a heat rate of 40 MW would be feasible from the stand point of critical size and it was decided to design a pilot plant of this rating with the specific purpose of supplying A.E.R.E. with electric power and steam for heating as an economic proposition. Selected engineering firms together with the Ministry of Works and the British Electricity Authority were invited to assist and the B.E.A. were asked to accept responsibility for co-ordinating the design of the steam plant. A second committee was formed in March 1952 to co-ordinate these outside activities.

In June 1952 it became clear that the small reactor would need enrichment to achieve critical size and, since during the preceding 3 months some of the more awkward design problems had begun to look easier, it was decided to alter the target to a full size plant using natural uranium and generating a significant amount of electric power. By the end of 1952 the feasibility study and general design of a 150 MW reactor and steam plant to generate 35 MW of electric power had been completed; the Atomic Energy Board decided that the Production Establishment at Risley should undertake the detail design and construction of this plant and work on the site commenced in April 1953. This paper summarises the work done at Harwell prior to the transfer of the project to Risley.

II. GENERAL FEATURES OF THE PLANT

The main features of Pippa were settled by three basic decisions viz:-

(a) To use natural uranium as fuel.
(b) To use graphite as moderator.
(c) To use a gas under pressure as coolant.

The decision to use natural uranium was taken on grounds of availability, cost and the general usefulness of a convertor type of reactor in a comprehensive power programme. Heavy water or Beryllia as alternative moderators were not considered beyond the early survey stages since the design of a heavy water power reactor was being actively pursued elsewhere in the Commonwealth. Beryllia did not seem to offer any outstanding advantages over Graphite and neither material was likely to be available in the U.K. within a reasonable space of time. The delicate neutron balance of a graphite/natural uranium reactor will not support the kind of through-tubes required for a liquid or liquid metal coolant at the desirable working temperature; a gas, which does not require though tubes, is therefore the only possible coolant.

The gas used for heat transfer has to be circulated through the reactor core channels, the heat exchangers and the connecting pipework. It is essential for the blowing power to be only a small fraction of the heat power released in the pile. Only about 25% of the pile heat can be turned into electrical energy. If the blowing power were 10% of the heat power it would be 40% of the electrical power and the net electrical power output would be greatly reduced.

It can be shown that the blowing power for a given rate of heat removal varies roughly inversely as the square of the absolute pressure of the gas, so that the blowing power at 7 atmospheres is about 1/49th of what it would be at atmospheric pressure under similar conditions. It follows therefore that the gas must be pressurised which means that the whole reactor must be enclosed within a very large steel pressure vessel. The diameter of the pressure vessel is fixed by the critical size of the reactor and the gas pressure made as high as is commensurate with the difficulties of fabricating pressure vessels of this size. The pressure vessel for Pippa will be 37 ft. internal diameter and about 70 ft. long, working at a gauge pressure of 100 pounds per square inch.

In all the graphite reactors built to date the axis of core and fuel rods are horizontal. After much thought it was decided that for a reactor inside a pressure vessel, which is bound to change its dimensions slightly under pressure and temperature, a vertical axis design is superior.

The main features of the design evolved will be apparent from Fig. I. The graphite core has to be built up out of individual machined blocks to a diameter of 36 ft. and a height of some 27 ft. The mass of about 1500 tons is carried by a steel honeycomb grating about 4 ft. in depth. The grating openings, each about 4 ft. square, are covered individually by thick flat steel plates each supported from the grating by levelling screws which can be adjusted to compensate for deflection during assembly.

Since heat expands steel more than graphite, the graphite has to be supported from the steel structure on a large number of rockers.

The cooling gas enters at the bottom and flows upwards through channels in the graphite core. Flow of coolant through the chinks of the core and in the annulus surrounding it must be prevented. Each core channel contains 5 uranium fuel rods about 4 ft. long supported by collapsible "crows foot" stands resting on circumferential ledges in the vertical channel. These fuel rods must be inserted and removed through stand pipes in the upper domed end of the pressure vessel; each stand pipe serves 16 core channels.

With this vertical design the reactor has only one operating face - the top - which can be locally roofed. A complete enclosing building is not needed and a substantial saving in cost is thus achieved.

The top face of the reactor is extended into the main building to provide a handling space for the uranium fuel elements before loading into the reactor. The insertion and removal of the fuel rods will be done by semi-automatic charge/discharge machines, the discharge machine having the requisite amount of shielding. Irradiated fuel elements are recovered from the reactor and assembled within the discharge machine into "baskets" of 20 elements. The machine is then traversed on rails to vertical shafts within the main radiation shield communicating with the storage pond at the base of the reactor, in which the spent elements are stored for cooling off prior to despatch to the chemical separation plant.

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ILLUSTRATIONS

The main control room for both reactor and turbo alternator is situated between the reactor and the turbine hall. The control room communicates directly with the turbine hall while access to the top face of the reactor is by a short staircase. The heat exchangers are designed as outdoor plant while the 4 main gas circulators are housed in pairs in below-ground spaces on either side of the main reactor. The whole design is compact and compares favourably with conventional coal fired power stations.

The excess reactivity of the reactor will be taken up by a number of 2" boron steel control rods suspended from flexible stainless steel wires from the drive mechanism. The latter is housed in the vertical stand pipes from the pressure vessel, these stand pipes being cooled sufficiently to provide a satisfactory ambient temperature. When the reactor is being charged or discharged the drive mechanisms will be disconnected from the control rods and removed, the rods themselves remain within the reactor. The flexible suspension will allow for limited relative movement between pressure vessel and graphite while the vertical stroke system enables gravity to be used as an operating force in the event of an emergency.

III. FEASIBILITY PROBLEMS OF THE REACTOR

The technical design of the reactor was governed by three over-riding requirements:-

(a) Minimum neutron wastage in reactor materials, in order to keep the diameter of the reactor within the limit of possible pressure vessels.
(b) Chemical compatibility of all materials within the reactor under radiation.
(c) Optimum heat transfer.

3.1 Factors governing the size of the Reactor

The elementary theory of thin cylindrical shells under internal pressure leads to the relation: pressure x diameter = 2 x hoop stress x effective thickness. Taking 11,000 lbs/sq.inch as a reasonable working stress, 2" as the greatest thickness convenient to weld and a corrosion allowance of 1/16" gives: pressure (p.s.i.g.) x diameter (ft.) = 3540. To limit blowing power we wished the pressure to be at least 7 atmospheres or 90 p.s.i.g. which sets 39 ft. as the maximum reasonable diameter of the shell. The lower limit of diameter for a natural uranium reactor generating negligible power is about 26 ft. but the critical diameter increases with temperature and output. Our problem was to obtain the greatest possible output within the permitted limit of diameter, for which it was essential to minimize neutron wastage.

In a graphite reactor of the kind considered wasteful neutron capture occurs in the graphite, in the coolant, and in the metal cans enclosing the fuel elements.

The importance of developing a reactor graphite of low capture cross section has long been recognised and development has been continuous over the past 5 years. Using a particularly pure coke and observing strict control over plant cleanliness during manufacture it is now possible to produce a reactor graphite with about 80% of the capture cross section of the graphite used in the Harwell Pile. The use of this graphite results in greatly improved criticality conditions.

The second large source of neutron capture is the material used to can the uranium. The only four possibilities are Beryllium (0.00104 cm^{-1}); Magnesium (0.0026 cm^{-1}); Aluminium (0.013 cm^{-1}) and Zirconium (0.017 cm^{-1}).

Beryllium is hard to obtain and difficult to fabricate into a complicated shape like a can with cooling fins. It could be used for the fuel element supports. Zirconium, in the hafnium free state essential for use in a reactor is not at present available to us in quantity and is incompatible with our chosen coolant. The choice of canning material therefore lay between aluminium and magnesium. Magnesium had not so far been used partly because its low capture cross section only became known in 1948 and partly because reactors have been air cooled and its rapid oxidation was feared. In the case of Pippa however the use of a Magnesium alloy offered considerable advantages and, after very careful metalurgical investigations has been adopted.

Once it had been decided that a graphite moderated gas cooled reactor was the best proposition for immediate development a thorough survey was made of all probable and improbable cooling gases. While this survey shewed that Helium was best technically, the supply of Helium seemed precarious since it would have to be imported from the U.S.A. under the restrictions imposed by the MacMahon Act. We were therefore compelled to choose the only worth while alternative, carbon dioxide. Although as a coolant CO_2 is somewhat inferior to Helium it has the virtues of being abundant, cheap and easy to handle. The fact that it is not chemically inert has necessitated intensive experimental work to prove the practicability of using it.

The three features of purer graphite, Magnesium alloy cans and CO_2 cooling have enabled the critical size of the reactor to be reduced to an extent that would have seemed impossible a few years ago without enriching the fuel.

3.2 Problems of Chemical Compatibility

In Pippa the following substances are in physical contact inside the reactor: uranium and magnesium alloy; magnesium and CO_2; CO_2 and graphite; CO_2 and mild steel and, in the event of a can failure, CO_2 and uranium. It is necessary to be certain that no deleterious chemical reactions will occur in spite of the high temperature and radiation conditions. Since in the heat exchangers the steam is at a higher pressure than the CO_2 there is a possibility of steam leaking into the CO_2. The reactions with wet CO_2 are therefore of interest.

3.2.1 The Uranium-Magnesium Contact

One of the outstanding advantages of using magnesium for canning is that (unlike aluminium) no chemical reaction occurs between it and uranium even under pile radiation.

3.2.2 The CO_2 - Magnesium Contact

Magnesium is not oxidised appreciably by air below about 350°C; above this temperature it is attacked and at 550°C the heat of the reaction is sufficient to cause ignition.

In CO_2 phenomena are very different. With dry gas no significant action occurs below 550°C, this figure being lowered by some 50°C if the gas is wet. Above this temperature the protective film breaks down completely and rapid corrosion occurs.

Consideration of these results led R. A. U. Huddle to suggest that the addition of beryllium to magnesium might improve corrosion resistance; an extra mural contract was accordingly placed with Magnesium Elektron Ltd., to make a series of alloys containing Beryllium and certain other elements. The results of tests on all these alloys at Harwell was most encouraging and one (containing approximately 0.05% Be 0.1% Ca and 1% Al) shews really remarkable resistance to corrosion. With this alloy no

measurable attack occurs below 600°C in wet air, which is the most generally corrosive environment. Tests on one of these alloys have shewn that on melting in a bunsen flame or electric furnace a protective film contains the molten metal in the same way as for aluminium and no ignition occurs. A welded specimen shews a similar resistance to the unwelded metal. Tests on the corrosion resistance of these alloys with CO_2 at 400°C - 450°C in the presence of pile radiation are in progress.

3.2.3 The CO_2 - Graphite Contact

At the start of the feasibility study it was predicted that the CO_2 -graphite-reaction would make it impossible to use CO_2 as a coolant, but estimates of the rate at which the reaction would proceed were very different. It was therefore decided to make an intensive experimental study of the matter.

The reaction of CO_2 with graphite is indicated by the equation

$$CO_2 + C \rightleftharpoons 2CO$$

The rate of this reaction at the reactor operating temperature is negligible in the absence of radiation and it was the influence of pile radiation that had to be ascertained.

In a closed circuit cooling system as used in Pippa the action is roughly as follows. Graphite is initially removed into the gas phase until an equilibrium state is reached when no further removal occurs. This does not mean that the reaction ceases, because different parts of the reactor at different temperatures and under different intensities of irradiation cannot all be in equilibrium with the particular steady state composition reached by the gas. Parts of the core will therefore lose graphite to the gas stream and other parts will take up graphite from the gas stream the result being a mass transfer of graphite from one part of the core to another. The rate of loss or gain from any one part of the core will be greatest if the gas is either pure CO_2 or pure CO. Experimental work has sought to establish two facts:

(a) The initial rate of reaction with pure CO_2 which gives the most pessimistic estimate of the graphite loss.
(b) The equilibrium concentration.

At the time of writing this work is not yet complete but preliminary results (with a margin of uncertainty) are as follows

(a) Initial rate of weight loss of graphite lies between 0.1% and 0.2% average per year, with a possible maximum of 0.6% per year in the high flux parts of the reactor.
(b) The equilibrium CO concentration is less than 1%.
(c) A water vapour content of 0.7% by volume makes no detectable difference to the reaction rate.

It therefore seems that the CO_2 graphite reaction will manifest itself in Pippa by a very slow mass transfer of graphite from one part of the core to another altering, over a long period of time, the density of the graphite by small amounts. The foreseeable effects of this are small and certainly do not preclude the use of CO_2 as a coolant. If our conclusions turn out to be optimistic the life of the reactor may be somewhat shorter than the 20 years which we hope to achieve.

3.2.4 The CO_2 - Steel Contact

At first sight it might seem that dry CO_2 being less oxidising than oxygen or air would not attack iron and steel at any temperature below 400°C. This however is not so: above 250°C a dark brown velvety scale forms at a rate comparable to that in air. Normal oxidation theory explains this phenomenon while preliminary tests have established rough scaling rates. These are

Temperature	Inches per year penetration
250°C	nil
350°C	0.003
450°C	0.010
550°C	0.030

Since 350°C is the maximum temperature at which steel will be worked and a corrosion allowance of 1/16" has been provided this scaling effect will not impair the life or safety of the pressure vessel.

An awkward problem can arise however if the scale flakes off or becomes entrained in the gas stream. Although the scale is not thick the large areas of steel especially in the heat exchangers could produce considerable quantities. Surface areas of steel in contact with CO_2 are as follows.

	sq. feet
Pressure vessel shell	12,550
Inside of Pipework	12,800
Inside of Heat Exchangers	19,480
Heat transfer surface of Heat Exchangers	360,000
Total	404,830

About half this area is at a temperature higher than 250°C and the amount of metal removed at a rate of even 0.001" p.a. would be some 34 cubic ft. Various scale preventive measures are being studied and provision is also being made for filtering a small bleed off from the main gas stream, so that dust and scale of any origin can be continuously removed.

3.2.5 The CO_2 - Uranium Reaction

This is important because it determines the way in which a can failure will develop. In general the effect of CO_2 on uranium at the working temperatures of Pippa is less than the effect of air on uranium in the existing air cooled piles. A burst fuel element will not therefore develop at a rate greater than can occur in the plants where detection methods are efficient enough to prevent this occurrence from becoming serious.

3.3 Problems of Heat Transfer

The heat rate or thermal power of a reactor is limited mainly by the difficulty of getting the heat away. The temperature range over which heat is removed is limited from above by properties of material, from below by properties of thermodynamic power cycles. The problem is to get as much heat power as possible out of the reactor within this limited temperature range, without resorting to methods that greatly increase capital costs or reduce overall plant efficiency.

The possibilities of heat removal in a graphite/natural uranium reactor are rather circumscribed by the nuclear physical requirements governing reactor design.

The best shape for the core of such a reactor is a right cylinder (length = diameter); usually the diameter is a little greater. The optimum lattice pitch or centre distance between parallel uranium rods is about 8"; the optimum fuel rod diameter is a little over 1"; the diameter of the channel in the graphite containing the rod cannot be much over 4" and is preferably less. The problem therefore is to find the maximum heat transfer coefficient which can be achieved for a fuel rod about 1 inch in diameter in a 3" to 4" diameter channel, observing the restrictions imposed on the use of cooling fins (a) by neutron capture (b) by Blowing power which, by Reynolds Hypothesis, unfortunately increases with heat transfer.

A considerable amount of heat transfer data for reactor channels was available at the start and this suggested that circumferential cooling fins on the fuel element cans would have a performance superior to longitudinal fins. The preliminary design was therefore based on circumferential finning and an extensive experimental progress started to optimize the fin dimensions.

Two experimental rigs are being used. One works with air at normal pressure, the desired range of Reynolds Number being achieved by using high air speeds. The results can be applied to pressurised CO_2 through the standard similarity laws of heat transfer theory.

The other is an exact full scale reproduction of the proposed reactor centre channel in a pressurised CO_2 circuit which will duplicate the exact final conditions chosen for the reactor. This work is still in progress but it is clear that a channel heat output of about 160 KW is possible, with the gas emerging some 150°F cooler than the hottest part of the cans.

The 8" lattice pitch gives 2½ channels per square foot of core end surface or 1700 channels for a 31 ft. dia. core fitting (with a 3 ft. reflector) inside a 37 ft. pressure shell. The ratio of average to maximum channel output in the radial direction for an unflattened reactor will be about 0.53 so that the attainable heat rate is at least

$$Q = 0.53 \times 160 \times 1700$$
$$= 144 \text{ MW}.$$

At say 21% net thermal efficiency this corresponds to some 30 MW of electrical power.

IV. THE DESIGN FOR BEST OVERALL PERFORMANCE

This can be reached only by a series of cut and try calculations since the number of interacting variables is rather large.

The dominant factor is the maximum surface temperature of 400°C (752°F) imposed by metallurgical limitations on the magnesium alloy cans. Since heat must flow from the cans into the coolant the temperature of the latter at the reactor main outlet will be lower than 750°F. This relatively low maximum temperature makes a gas turbine cycle unattractive and forces us to use a steam cycle, shewn in its simplest form in Fig. II.

Let the temperature of the coolant entering the heat exchanger be $(750 - x)$°F and let y be the approach or temperature difference across the hot end of the exchanger, then the maximum temperature of the steam

will be $(750 - x - y)^{\circ}F$. Note that x and y will increase with the rate of heat transfer, i.e. with the load on the plant.

The heat exchanger falls into three sections - the super heater, the evaporator and the feed heater - and conditions must be such that the coolant is always hotter than the steam/water substance in the other circuit. This means in particular that the coolant leaving the evaporator must be hotter than the entering feed water which is at saturation temperature ts.

Let z be the approach at this point then the coolant temperature is $(t_s + z)$. The fall in temperature of the coolant from entry to this point is $[(750 - x) - (t_s + z)]$ and if M is the coolant mass flow (lbs/hr) and c its specific heat, the heat transferred is $Mc[(750 - x) - (t_s + z)]$ Bt_u/hr. If n is the evaporation rate (lb/hr), H the enthalpy of the steam at temperature $(750 - x - y)^{\circ}F$ and h_s that of water at saturation temperature then

$$Mc[(750 - x) - (t_s + z)] = n(H - h_s)$$

$$\text{whence} \quad \frac{M}{n} = \frac{H - h_s}{c[(750 - x) - (t_s + z)]}$$

Table I below gives this ratio worked out for 3 different steam pressures and a steam temperature of $650^{\circ}F$ ($x = 70^{\circ}F$; $y = z = 30^{\circ}F$; $c = 0.25$)

TABLE I

STEAM CYCLE DATA

Steam pressure psia	200	300	400
Saturation temperature t_s	381.8	417.3	444.6
H at $650^{\circ}F$	1348.0	1341.8	1335.2
h_s at temp. t_s	355.4	393.8	424.0
$H - h_s$	992.6	948.0	911.2
c $680 - (t_s + 30)$	67	58.2	51.4
M/n	14.8	16.3	17.8
Lowest coolant temperature $^{\circ}F$.	335	368	395

The coolant leaving the evaporator at temperature $(t_s + 30^{\circ})$ has still to supply the feed heat $n(h_s - h_o)$; (where h_o = enthalpy at hot well temperature, taken as $100^{\circ}F$). The last line gives the temperature of the coolant leaving the feed heating section of the heat exchanger and returning to the reactor.

The conclusions to be drawn from this table are:

(a) That the mass flow of CO_2 will be of the order of 15 times the evaporation rate so that blowing power is likely to be an important factor.

(b) That increase of steam pressure, which improves steam plant efficiency, means a greater mass flow of CO_2 and a higher pile inlet temperature. The blowers at pile inlet have to handle more and hotter gas and blowing power rises to an extent which may offset any gains on the steam side.

(c) That feed heating by bled steam is of doubtful benefit because it will increase the temperature of the gas handled by the blowers.

(d) That everything possible must be done to transfer heat with small temperature drops, i.e. to minimize x, y and z.

The general nature of the optimization problem will be clear from Fig. III. Here the abscissae represent the heat rate of the reactor, the ordinates electrical output from the plant. With steady steam conditions and a constant plant efficiency the gross electrical output is proportional to the reactor output (line OA)* To get the net electrical output we must subtract the blowing power which, for a given reactor, varies roughly as the cube of the reactor heat rate [since blowing power varies roughly as $(\text{mass flow})^3$ and mass flow must vary as output to keep the coolant temperature range steady]. The curve of net electrical output so derived is OC and the important point to note is that it has a maximum. The best position for the working point on this curve is largely a matter of economics. The reactor is costly and its high annual capital charges must be spread over as large an electrical power output as possible. When the annual capital charges greatly exceed the annual fuel bill minor changes of efficiency are unimportant compared with increased output. It therefore pays to push for output, the limits being set by technical feasibility and increasing costs more than by loss of efficiency.

V. THE CHOICE OF STEAM CYCLE

Calculations made for a number of different steam pressures and pile conditions shewed that the maximum net efficiency obtainable from the simple Rankine cycle plant of Fig. II would be rather low. It has been shewn above that in such a plant the steam pressure pretty well fixes the temperature of the coolant at the blower. A low temperature at the blower means less blowing power, because the mass flow rate of gas is smaller and the density is greater. However a low blowing temperature also means a low steam pressure and an inefficient steam plant. Increase of steam pressure improved steam plant efficiency but also increases the blowing power.

These considerations led Messrs. Babcock & Wilcox to suggest a cycle in which steam is generated at two different pressures; a proportion is generated at a much higher pressure than could be considered with a single pressure cycle while the remainder is generated at a low pressure and injected into an appropriate stage of the turbine.

The optimization of this cycle is rather involved but it is clear that a plant in which some 65% of steam is generated at 400 p.s.i.g. and the rest at 130 p.s.i.g. gives a worth while improvement in overall net efficiency in comparison with the best single pressure cycle. The proposed plant will have an overall thermal efficiency of about 24% whereas a single pressure plant would have given about 21%. (See Appendix).

The performance of the dual pressure cycle has been considered over a wide range of reactor operating conditions and principles of operation and control have been evolved which provide a very substantial measure of protection to the reactor during both normal and abnormal conditions of operation.

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* In actual fact, keeping the can temperature constant, the hot gas temperature will decrease with increasing heat output and steam conditions and plant efficiency will therefore get worse. In reality therefore line OA bends downwards a little.

VI. FUEL ELEMENT DEVELOPMENT

The development of a satisfactory fuel element is of course vital to the success of the whole plant. The fuel element is a uranium rod about 1" in dia. and 4 ft. long suitably canned. The principal problems that have to be solved are:

(a) Fabrication of uranium bar so that bending of the fuel element and irradiation damage are minimised.
(b) Manufacture of Magnesium alloy can.
(c) Method of sealing the bars in the cans.
(d) The design and manufacture of the fuel element support system.

Although development work (particularly on (a)) continues, research work by the Metallurgy Division A.E.R.E. indicated that these problems were soluble and actual prototypes successfully withstanding the pressure and temperature conditions prevailing in the reactor, were made and tested.

To carry out these tests four furnaces were built in which completed fuel elements could be tested in CO_2 at 100 p.s.i. either at steady temperatures up to 500°C (932°F) or by thermal cycling between 50°C and 500°C.

When subject to thermal cycling the can tends to creep along the bar and a large amounts of effort has gone in to minimizing this effect. The fuel element development work is described in more detail in a companion paper by Mr. L. M. Wyatt.

VII. PROBLEMS OF GENERAL DESIGN

7.1 Charging and Discharging Fuel Elements

The uranium fuel elements - 5 to a channel - are each supported at their base by a collapsible spider which in turn rests on a rebate in the graphite at the periphery of the channel. The fuel elements are inserted and subsequently recovered through the top end of the reactor.

In a reactor the neutron flux varies with radius and if the irradiation received by different rings of fuel elements varies accordingly. For this reason it is desirable to charge and discharge fuel elements in batches rather than all simultaneously. The reactor has to be blown down to atmospheric pressure during charging/discharging operations.

The following principles have been observed in the design of the charge/discharge gear.

(a) In order to keep down the number of holes through the pressure shell 16 graphite channels are served from one stand pipe.
(b) No equipment associated with the operation is part of the permanent structure of the reactor. All mechanisms can therefore be maintained and tested under normal conditions and are not exposed to radiation and heat for long periods.
(c) The system provides a continuous path between each channel in turn and the operating face of the reactor for the passage of the fuel elements. The charge/discharge operation involves some 8500 fuel elements per complete charge each of which must be handled without mishap. For this reason a full scale test rig has been built and is being tested for reliability.

7.2 The Graphite Structure

The details of construction and probable life of the graphite structure are influenced very considerably by the phenomena produced by radiation in graphite e.g. changes of dimensions, and change of physical properties such as elasticity modulus and thermal conductivity in an undesired direction. In comparison with existing graphite piles the higher temperature at which the graphite of the Pippa type reactor will operate tends to reduce materially these undesirable effects.

The graphite stack has been designed as a 24 sided prism about 36 ft. across and 27 ft. high. It is built up from some 32,000 graphite blocks each weighing about 100 lbs; 2 ft. 1" long and about 8" square; this size is convenient for handling and erection. The main requirements to be met are:

(a) The stack should remain stable and not lose alignment.
(b) The shape and linearity of the channels must be preserved.
(c) Spalling within the channels must not occur.
(d) The Wigner growth, (due to irradiation) both within blocks and differentially across the stack, should produce no serious effects.
(e) Temperature stresses arising from heat flow within the graphite block should be within the strength of the block.
(f) Temperature gradients across the stack in both steady state and transient condition should not cause instability.

The detail design is based on the principle of maintaining a closely packed reflector, while allowing increasing gaps for Wigner growth towards the centre of the core where the neutron flux is highest.

The stack has to be supported by small steel rollers which allow the steel base to expand without disturbing the graphite. Spherical roller guides locate the stack within the pressure vessel and it is also strapped round with ties compensated for temperature expansion.

The annulus between stack and shell is sealed by flaps held against the graphite by gravity assisted by gas pressure.

7.3 The Reactor Pressure Vessel

The design of the pressure vessel was evolved with the help of the Structural Engineers of H.M. Ministry of Works, The National Boiler and General Insurance Co. Ltd., and last but not least Messrs. Whessoe Ltd. We were influenced very considerably by the design of the 47 ft. Vertical Spinning Tunnel at the National Aircraft Establishment, Bedford, built by Whessoes.

Considerable importance was attached to the material of construction so as to ensure best conditions for fabrication and welding. All welds will be subject to radiographic examination and it is also proposed that the vessel shall be stress relieved. The number of openings in the shell and their diameter has been kept as small as possible. The only openings with high stress compensation are the four gas outlet openings.

The vessel will contain some 1200 tonnes of graphite and the support of this weight through the wall of the pressure vessel is a difficult problem. The method proposed is a plane grid framework the ends of which rest on a circular ring guider internal to the vessel. The grid has been designed for a maximum centre deflection of $\frac{1}{4}$" which is considered acceptable for the graphite stack.

The ring girder in turn is supported at 20 points on its periphery by brackets welded to the vessel which transmit their loads through the

vessel to an equal number of external brackets. The latter rest on column tops having a curvature which transmit the entire weight of the vessel and its contents to 10 grillages equally spaced in a circle. This ensures minimum constraint for thermal expansion.

Hydraulic testing would impose load conditions far more severe than exist in service and since the shell is enclosed by a heavy concrete shield only a pneumatic test is contemplated.

The pressure shell is of course traversed by the leakage neutrons from the reactor. This irradiation, on present knowledge, will not affect the mechanical strength of the shell, although it is not impossible that a little hardening and loss of ductility may occur in the course of years at some points. The steel will acquire residual radio-activity by Neutron capture (a) in Fe 58, turning it into Fe 59 with a half life of 46 days and 1.1 + 1.3 MeV gamma rays. (b) in Co 59, turning it into Co 60 with 5.3 yr. half life and 1.13 + 1.17 MeV gamma rays.

The Fe 59 is not important because very little is formed and its half life is short but the residual activity from Co 60 seemed likely to make maintenance operations on the pressure vessel hazardous for perhaps 6 months after shut down and unloading of the fuel elements, and even then difficult and restricted.

Typical steels were therefore analysed for their cobalt content and every attempt was made to see whether a steel could be produced with an exceptionally low cobalt content. The conclusion of this investigation was that there is little hope of obtaining steel with an average cobalt content below 0.005%. The following table gives an estimate of the radiation dosage in terms of health tolerances at the pressure vessel wall assuming 0.07 p.p.m. cobalt in the graphite and 50 p.p.m. in the steel and no fuel elements in the reactor.

TABLE II

RESIDUAL ACTIVITY OF PRESSURE VESSEL

Period of irradiation	Time from Shut Down			
	1 day	1 month	6 months	1 year
1 year	66	46	10	6
2 years	71	51	16	11
3 "	77	57	21	16
4 "	81	61	25	19
5 "	85	65	28	23
30 "	114	94	56	48

7.4 Radiation Shielding

The function of a Biological Radiation Shield is to reduce radiation to a level that ensures safe working conditions around the plant. Protection is needed against (a) neutrons (b) gamma rays.

In addition sufficient shielding should be given to equipment requiring maintenance (e.g. control rod mechanisms, reactor isolating valves, expansion joints etc.) to limit to a low level the residual neutron induced radio-activity.

Other conditions which must be met are:

(a) No leakage of active gas or dust from within the shield.
(b) No serious distortion which would obstruct the free movement of the pressure vessel and its attachments.
(c) No spalling or break up of the shield under thermal stresses.
(d) A shield cooling system that is adequate and does not choke up.

A very careful study was made of the temperature gradients to be expected in solid concrete subjected to radiation and direct heat from the lagged pressure vessel. It was found that a thermal shield of 4" steel would be needed at roof and floor and at the sides a 1' 6" thermal shield of concrete blocks encased in $\frac{1}{4}$" mild steel. The thermal shield absorbs 90% of the incident radiation which is removed as heat by air circulated by exhaust fans discharging up a stack. The main biological shield is reinforced Portland cement gravel aggregate concrete about 6' thick.

Some shielding of the external CO_2 circuit will also probably be necessary due to a curious short lived activity induced in the oxygen atoms of the CO_2.

Auxiliary cooling is also required to remove heat from the main pressure vessel branches passing through the shield provided by conduction from the pressure vessel. This is necessary to prevent overheating of the concrete and also in the case of the top branches to provide suitable ambient conditions for the main control and shut-off equipment.

7.5 The CO_2 Gas System

A major advantage of CO_2 as a coolant is that there is an established industry to supply it. The purity of commercial solid CO_2 is satisfactory and the 23 tons needed for a filling can be passed into the system at the rate of 3 tons per 8 hour shift, using 4 standard convertors. The initial displacement of air and moisture from the system may require several fillings during the first year but CO_2 is not expensive.

Moisture will appear in the circuit during the early life of the plant, coming from the graphite from other materials of construction and a little from the CO_2 itself. A tolerance on the water vapour in CO_2 of 1 part per 1000 is below dew point at room temperature and will reduce risk of aqueous corrosion during shut down.

It is therefore proposed, when filling, to pass the gas through a 40 c.f.m. (compressed measure) silica gel drier, while continuous drying is provided during the operation of the plant, by passing a 0.5% bleed of gas through duplicate 300 c.f.m. silica gel drying plants.

Condensers will cool the gas to 70°F before entering the absorbers. Elements will be regenerated by heating with steam and the absorbed moisture recovered by a small reverse flow of hot CO_2 which would be passed through a cooler to condense the moisture. Once moisture has been reduced to 0.01% this plant would be capable of maintaining this condition against an in leakage of a few pounds per hour.

7.6 The CO_2 Ducts, Circulators and Valves

Each heat exchanger is connected top and bottom to the pressure vessel by a 4' 6" diameter circular duct of mild steel. The main blowers are located in the lower, cool gas, duct one to each of the 4 circuits. Isolating valves of the wedge gate type are included in

each hot and cool duct. They will have divided packed glands, sealed with CO_2 in excess of system pressure. These main isolating valves are backed by butterfly valves and with valves closed complete isolation of the heat exchangers from the reactor is assured by exhausting the space between each pair of valves.

The large relative movements of thermal origin between pressure vessel and heat exchangers are difficult to accommodate and the solution of this problem was greatly assisted by the experience of the Gas Turbine Dept. of C. A. Parsons. Hinged bellows within the duct work run are extensively used, disposed so that servicing during reactor shut down is possible.

The main gas circulators, 4 in number, are of the single stage overhung centrifugal type rigidly coupled to a D.C. variable speed motor. End thrust is carried by a thrust block on the outboard end of the motor.

The design of a shaft seal to prevent leakage of pressurised CO_2 out of the circulation necessitated a good deal of development work which is still in progress. The main D.C. drive motors are supplied on the Ward Leonard system permitting a speed range of 10 to 1.

VIII. THE STEAM PLANT

8.1 The Main Boilers

Each of the four heat exchangers is of shell and tube construction, the gas flowing over the outside of the tubes in cross flow. The shells are 17 ft. 6" in diameter and 69 ft. 6" long mounted vertically. The construction and erection of these vessels will closely follow established practice for oil refinery plant.

The economizer, evaporator and superheater all consist of multiloop elements of mild steel tubing 2" O.D., with elliptical studs welded on externally to increase the heating surface. The tube sections will be installed on site after the erection of the shell is completed. The pressure drop from end to end of a complete heat exchanger at full flow will be below 1 p.s.i.

As described in V steam is raised at two different pressures in each heat exchanger. Both H.P. and L.P. steam is superheated to the same temperature of 635°F (335°C). The evaporator sections are under forced circulation from submerged electric pumps which maintain a circulation of four times the evaporation rate of the section. Each evaporator has a single steam drum 48" in diameter and 15 ft. long containing conventional cyclones and scrubbers.

Particular attention has had to be given in the design to preventing leakage of water or steam into the CO_2 circuit which is at the lower pressure. No tube joints which have to be welded on site are situated inside the pressure vessels; the ends of each tube element pass separately through the shell so that if leakage at a joint does occur it will be to atmosphere. Steam and water headers are also outside the pressure shell for the same reason. The tube element will be rigourously inspected and tested and it is believed that the probability of leaks developing in these continuous tube sections is very small indeed. Careful control of feed water quality and deaeration will minimize corrosion.

Thermal sleeves at the points where the tubes pass through the shell and the multiloop construction of the element will minimize thermal stresses.

Finally means have been proposed for detecting and tracing very small leakages of water into the CO_2 should such develop and for subsequent repair to be accomplished. The heat exchangers are designed, protected and lagged for outdoor operation and much attention has been given to providing adequate maintenance and repair facilities.

The foregoing design was evolved by Messrs. Babcock & Wilcox Ltd., to meet A.E.R.E. requirements.

8.2 The Main Generating Plant

The turbine is a two cylinder reaction machine of well established C.A. Parsons Ltd., design. The H.P. cylinder has two steam inlets the first for taking steam at 390 p.s.i.g. 625°F, the second for steam at 120 p.s.i.g. 625°F. The L.P. cylinder has twin exhausts which discharge into a surface condenser operating at a vacuum dependent upon the cooling water conditions at the selected site. For the purpose of the feasibility study a vacuum of 28.9" with cooling water at 60°F (15.5°C) was assumed. During normal operation the condensate is heated and deaerated by bled steam and returned to the heat exchangers by an electrically driven feed pump.

The turbine is directly coupled to a 3000 R.P.M. air cooled alternator of conventional design. Except during start up and shut down this machine supplies the main plant auxiliaries and exports some 35 MW of electrical power.

A separate electric supply will be available for starting up and shutting down and to cover the shut down requirements of the plant.

A separate "dumping" condenser is provided in which the full load steam production of the boilers can be condensed without passing through the turbine. The condensate from this dumping condenser is deaerated and fed back into the cycle.

The dumping condenser will be used during normal starting up and shutting down and during the commissioning of the plant. It will also permit the reactor to run on full load in the event of turbine outage.

Steam for process and heating could be made available (say 50,000 lb/hr. at 80 p.s.i.g. saturated) at some sacrifice of electrical output. L.P. steam from the main boilers would be passed to evaporators where it would condense and be returned. Condensate from the process side which might contain impurities would not be mixed with the water in the power plant cycle. A study of the possibilities for process steam from such a plant has been made by Mr. Berridge of the B.E.A.

The source of condenser cooling water will depend upon the site and natural draught or induced draught cooling towers may have to be employed.

The foregoing design for the steam plant owes a great deal to the experience of Mr. F. Shakeshaft of B.E.A.

IX. CONTROL AND SAFETY OF THE PLANT

9.1 The Control System

The basic problem of reactor control arises from the fact that the neutron balance changes according to operating conditions.

When the reactor is hot and fully poisoned the rate of wasteful neutron absorption in uranium and graphite is greater than when the

reactor is cold and unpoisoned. The amount of fissile material put in must be enough to cover operation in the hot and poisoned state and is therefore excessive in the cold unpoisoned condition. The neutron surplus in the cold unpoisoned state has therefore to be absorbed in special absorbing elements termed control rods, which are gradually withdrawn during start up. If these rods are withdrawn too rapidly the neutron surplus in each neutron generation will be too large and the rate of increase of power very rapid, leading to partial or complete loss of control and dangerous conditions. The amount of rod withdrawal corresponding to a given rate of power build up is different in different states of the reactor.

The general principles of control of Thermal Neutron Reactors are discussed in a recent I.E.E. Paper by one of the present authors and will therefore not be discussed further here.

The following table gives the total excess K* of Pippa for various average central channel rating (M.W./tonne), with average graphite and uranium temperature of 300°C and 450°C respectively over the core.

TABLE III

BREAKDOWN OF REACTIVITY

Central Channel Rating (mean)	2.50	2.75	3.0
Xenon poison, steady state	1540	1620	1680
Xenon poison, build up	160	200	240
Uranium temperature effect	600	600	600
Graphite temperature effect	1590	1590	1590
Total Built in excess K ($\times 10^{-5}$)	3890	4010	4110

In a cold unpoisoned state the excess K is thus around 4000×10^{-5}. After a few days at full power almost all this is taken up by the reactor itself. It is proposed to invest all this excess K in a main control rod system with a speed of withdrawal so slow that even the severe fault condition of all the rods travelling out together would have to be maintained for 15 minutes before the temperature of the fuel element rose by 100°C.

The slow release of K ensures safe start up but is inadequate to deal with normal plant control arising from load changes. For this purpose it is intended to provide a quick acting system controlling a very limited amount of K - an amount so small that even if it were all suddenly released the reactor would stabilize under the uranium temperature coefficient within a safe temperature rise of the fuel element.

The rate of heat production in the uranium fuel elements is controlled by adjustment of the neutron level. The heat so produced is removed by the coolant and its measure given by the product of mass flow and rise in temperature as it passes through the reactor. Therefore to transfer a varying amount of heat from the reactor to the boiler either involves variation of mass flow or variation of temperatures.

---

* K = multiplication constant of the chain reaction
$= \frac{\text{neutrons in given generation}}{\text{neutrons in preceding generation}} = 1 + e$

e = excess of K above that required for criticality.
If e exceeds about 760×10^{-5} the reactor becomes supercritical under prompt neutrons above, which is most dangerous.

Mass flow in a pressurised gas system could be varied by means of speed regulation of the main blowers, the gas pressure remaining constant, or by gas pressure regulation with the blowers operating at full speed. The latter method was investigated and found to be expensive and inefficient requiring multistage compressers of large power and high pressure ratio together with very large storage vessels. Variation of mass flow can best be achieved by speed regulation of the main blowers.

The choice of control system therefore lies between varying the temperature of the coolant or by speed regulation of the blowers. Both systems have their attendant disadvantages but speed regulation was preferred as the resulting constant temperature operation limits thermal stressing throughout the pressure circuit, and avoids the unpredictable thermal time lags involved in variable temperature operation.

A wide choice of speed regulation methods exist but after review, in spite of small cost and efficiency disadvantages, Ward Leonard control was adopted, as this system has a finer reliability and gives rise to continuous regulation over a wide range and so to a fully flexible system.

With this system, the control of the plant involvs 4 main controllers:-

(a) The pile power controller. This enables any heat output of the reactor to be selected and that value automatically maintained.
(b) The circulator speed controller. When pile power is altered the circulator speed is automatically adjusted to maintain the temperature of the outlet gas from the pile constant.
(c) The L.P. boiler pressure controller. The input to this controller is the return-coolant temperature to the pile. Any tendency for this temperature to increase is opposed by a decrease in steam pressure in the L.P. evaporating sections.
(d) The Turbine steam supply controller. A change in H.P. boiler pressure causes the valves controlling the steam supply to the Turbine to operate to restore constant steam pressure.

The following example illustrates how the system control would operate for a reduction of power initiated in the pile power controller.

As the heat output from the reactor falls the speed of the gas circulators automatically decreases and the pressure in the L.P. evaporating section of the boilers automatically decreases. The turbo-alternator sheds load to oppose the tendency of the H.P. boiler pressure to fall. Equilibrium conditions are reacted when the output of the turbo-alternator equals the output of the pile, neglecting losses. During the transition from one power level to another the inlet and outlet gas temperatures to the pile remain sensibly constant.

It has already been mentioned that the control rod system is so designed that a very large release of reactivity cannot occur suddenly. Another important safety feature of the plant follows from the physical fact that the CO_2 coolant has negligible absorption for thermal neutrons. Loss of system gas pressure due, for example to a leak in the pressure circuit is not accompanied by a release of reactivity so that normal action of the shut-off rod system will deal adequately with this otherwise serious fault condition.

9.2 Auxiliary Cooling Circuit

When the reactor is shut down cooling is still needed to remove the heat produced by delayed neutrons and fission product decay.

The auxiliary circuit designed for this purpose comprises a filter, heat exchanger and gas circulator connected directly across the reactor pressure vessel, the circulator being fed from a battery supported by a diesel generator. This auxiliary circuit is also capable of maintaining temperatures within the shut down reactor at an acceptably low level after the system pressure has been reduced to atmospheric for unloading the fuel elements.

The volume of fission products issuing from a hole in the fuel element can of a size well within the detecting range of the main burst slug detection gear is small and insufficient to cause gross contamination of the plant. Uranium oxide issuing from this hole may cause trouble in this respect and it is for this purpose that a filter is included in the auxiliary circuit so that such products can be trapped.

9.3 Burst Slug Detection

Detection of faulty fuel elements by the gaseous fission products released will be done by previously developed machines altered to suit the pressurised gas circuit. Samples of gas from a group of channels will be piped to the outside of the reactor where group selection is by means of an automatic sequence timer which synchronizes the operation of solenoid valves with the action of the detecting machine. Sensitivity will depend upon the background due to the uranium dust on the outside of the cans. Assuming this to be 5 micrograms per foot run estimated detection capability is 0.1 to 0.2 cm^2 of exposed uranium in maximum and minimum flux regions respectively. The size of the actual puncture through the wall of the can may be less than this area as the gaseous fission products tend to escape from an area of uranium greater than that of the puncture itself. With 4 of these machines it should be possible to scan the complete reactor in less than 30 minutes. In order that a bad burst occurring suddenly within this period will not go undetected burst slug detectors working on the same principle (but necessarily less sensitive due to the larger volume of coolant samples) will be installed in each of the 4 main outlet ducts. After shut down and blow down to atmospheric pressure the load hole associated with the burst will be opened and a flexible sampling tube introduced via the loading chute. The reactor is then run at low power and the chute indexed to each hole in succession until the faulty channel is located.

9.4 Discharge of CO_2 to Atmosphere

The only significant activities in the cooling gas will be due to A 41 and C 14 arising from argon and nitrogen present in the CO_2 as impurities. Assuming an argon contact of 10 p.p.m. and a nitrogen contact of not more than 1.0% the A 41 will reach an equilibrium value of 58 curies in some 6 hrs while the C 14 (half life 5720 yrs) will build up linearly reaching a value of 320 curies at the end of 5 years.

Assuming that discharge takes place 4 hrs after shut down and that the rate of discharge is uniform over 1 hour the output rate of A 41 will be less than from Bepo and will constitute no hazard. The output rate of C 14 (after 5 yrs irradiation!) will be 90 millicurie per second if all is discharged, resulting in an estimated peak ground level concentration of 8 microcuries per $metre^3$ with a 50 metre stack and 1 m/sec. wind speed. This is about 8 times the permissible concentration for prolonged exposure but since the exposure will last only 1 hour it could be tolerated.

X. LEADING DIMENSIONS, PERFORMANCE & COSTS

Heat Rate of Reactor	143.7 MW (= 490 x 10^6 Btu/hr)
Volume of Active Core	15100 ft^3
Specific Heat Rate	9.5 KW/ft^3 (= 34,400 Btu/hr/ft^3
Maximum Can Temperature	752°F (400°C)
Coolant Upper Temperature	662°F (350°C)
Coolant Lower Temperature	356°F (180°C)
Coolant Mass Flow (max.)	6.5 x 10^6 lb/hr.
Steam Conditions	400 p.s.i.g., 625°F, 285,000 lb/hr.
(Dual Pressure Cycle)	130 p.s.i.g., 625°F, 153,500 lb/hr.
	28.9" vac.
Gross Electrical Output	42.5 MW
Blowing Power	4.84"
Other Auxiliaries	2.3 "
Net Electrical Output	35.37 "
Overall Efficiency	24.6%
Steel, Structural	1,350 tons
" in Plant	4,050 "
Concrete	18,000 $yds.^3$
Graphite, Rough Blocks	2,350 tons
Uranium	100 " (approx.)
Costs (guessed)	
Reactor	3½ million pounds
Heat Exchangers	1¼ " "
Power Plant	2¼ " "
Site	1 " "
Total	8 million pounds

Total cost per KW = £227.

XI. ECONOMICS

The salient economic fact is the high capital cost per kilowatt installed, which makes it essential to achieve a long life and a high load factor. The life of the reactor will almost certainly be that of the graphite which it is impossible to foretell. For the purposes of discussion we shall assume a 20 yr. life and 80% load factor.

Assuming redemption over 20 yrs. by sinking fund accumulating at 4% and interest at the same rate the annual capital charges are 7.336% or £16. 13. 0. per KW of plant capacity. At a load factor of 80% this will be spread over 7000 units making the annual capital charges 0.571 pence per KW hr. sent out. The 100 tons of uranium locked up in the pile will cost perhaps £1,500,000 and 4% interest on this sum is £60,000 p.a. or £1 -14. per KW. Spread over 7000 units this amounts to 0.0584 pence/KW.hr.

The amount of heat energy that can be extracted per ton will probably exceed 2000 MW days or 48,000,000 KW hrs. The electrical energy (at 25% efficiency) will therefore be around 12,000,000 KW.hrs/ton. The corresponding unit cost is therefore (15000 x 240) ÷ 12,000,000 = 0.3 pence.

There will finally be an operating charge, which for steam stations is around 0.08 pence per unit S.O. The total unit charge at 80% L.F.

therefore builds up as follows.

Interest and redemption on plant	0.571
Interest on uranium invested	0.0584
Cost of uranium burned	0.3
Operating charge	0.08
Total, pence per unit	1.0094

A figure of one penny per KW hr. sent out is therefore realistic, but will be optimistic if graphite trouble develops to shorten the estimated life. This is considerably higher than the average generation cost of B.E.A. Stations but takes no account of the value of the plutonium formed in the fuel elements.

ACKNOWLEDGEMENTS

In January 1952 the British Electricity Authority were invited to participate in the project and to look after the general design of the steam and electrical plant. On their advice (which in each case confirmed Harwell opinion) the following engineering concerns were invited to assist.

Babcock & Wilcox Ltd., on the choice of the steam cycle and the design of the boiler plant.

C.A. Parsons Ltd., on the design of the steam plant.

In the same month the Ministry of Works was asked to assist with the Structural Design of the Reactor. On their advice the following firms were consulted.

Whessoe Ltd., Darlington, on the design of the reactor pressure vessel.

The Consett Iron Co. Ltd., on allied metallurgical problems.

Invaluable help has also been given by the following concerns under direct contract with A.E.R.E.

C.A. Parsons Ltd., and the Parolle Electrical Plant Co. Ltd., in connection with the CO_2 duct work, main circulators, auxiliary cooling circuit and heat transfer rig.

The National Boiler & General Insurance Co. Ltd., in connection with the preliminary design of the Reactor pressure vessel.

Metropolitan-Vickers Elec. Co. Ltd., in connection with the reactor control mechanism.

Magnesium-Elektron Ltd., in connection with the development of the magnesium beryllium alloy for the uranium can material.

Finally acknowledgement is due to the Ministry of Works who prepared architectural and structural drawing of the plant to A.E.R.E. requirements, dealt with all ventilating problems, made cost estimates and evolved, together with B.E.A. and A.E.R.E. a skeleton construction programme. With so many concerns involved it would be invidious to mention individuals.

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APPENDIX

THE DUAL PRESSURE STEAM CYCLE

The approximate flow diagram for the cycle proposed is given in Fig. IV. The computations are as follows.

1. Work Done in H.P. Turbine

The steam enters at 415 p.s.i.a. and 635°F with enthalpy of 1325.5 Btu/lb. Adiabatic expansion through the turbine to 145 p.s.i.a. would reduce the enthalpy to 1218, the adiabatic heat drop being 107.5. Assume a turbine internal efficiency of 80% then the actual heat drop will be 86 and the exhaust condition 1239.5 Btu/lb. at 145 p.s.i.a. and 438°F. The work done on the turbine is

$$\frac{86 \times 299{,}000}{3413} = 7520 \text{ KW.}$$

2. Mixing of H.P. exhaust and L.P. steam

The L.P. steam is at 145 p.s.i.a. and 635°F with enthalpy of 1343.7 Btu/lb. the flow being 161,000 lb/hr. Hence

Heat of H.P. steam = $1239.5 \times 299{,}000 = 370 \times 10^6$

Heat of L.P. steam = $1343.7 \times 161{,}000 = \underline{216.5 \times 10^6}$

506.5×10^6

Combined Mass = 460,000 lb/hr.

∴ Combined Enthalpy at 145 p.s.i.a. = $\frac{586{,}500{,}000}{460{,}000} = 1276$ Btu/lb.

the temperature being about 503°F.

3. Work done in L.P. Turbine

Expansion occurs first to 23 p.s.i.a. where 52,700 lb/hr. is bled off for feed heating, and then down to 1 p.s.i.a. and 12% wetness.

Enthalpy at inlet	1276
Enthalpy at bleeding point (1.5% wet) say	1145
Heat Drop	131

Work done on rotor = $\frac{131 \times 460{,}000}{3413} = 17.6$ MW.

The mass flow after bleeding is reduced to 407,300 lb/hr.

Enthalpy at inlet	1145
Enthalpy at exhaust	982
Heat Drop	163

Work done on rotor = $\frac{163 \times 407{,}300}{3413} = 19.4$ MW.

Total work in L.P. Turbine = 17.6 + 19.4 =	37 MW.
Total work in H.P. Turbine	7.5 MW
Total work done on rotor	44.5 MW

Turbine loss 1%, Alternator losses 2.5%, hence reduce by 3½%.

∴ Alternator Gross output = 43.0 MW.

The temperature relations in the heat exchanger are indicated in Fig. 5.

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APP.

FIG. III.

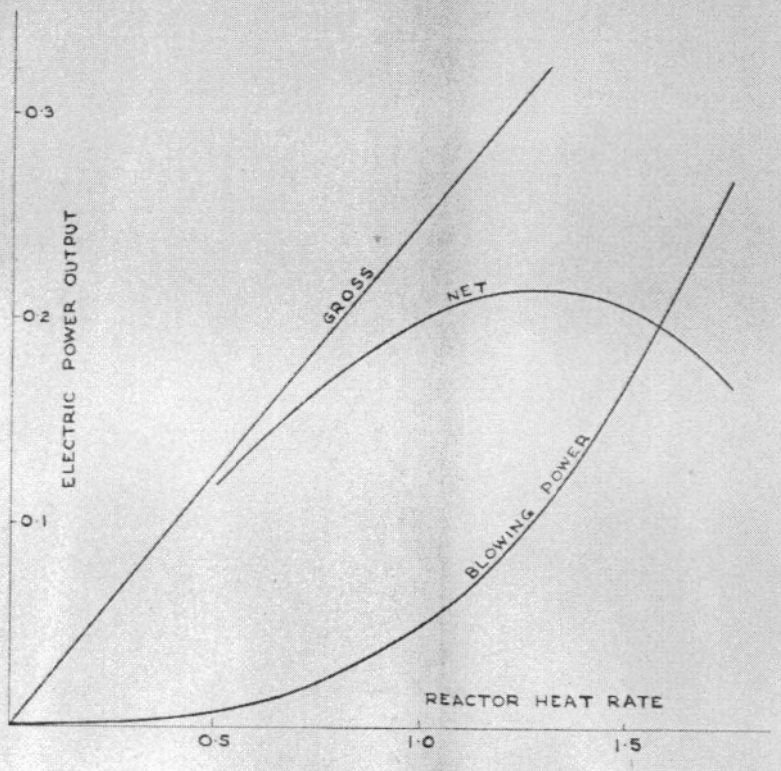

FIG. III.

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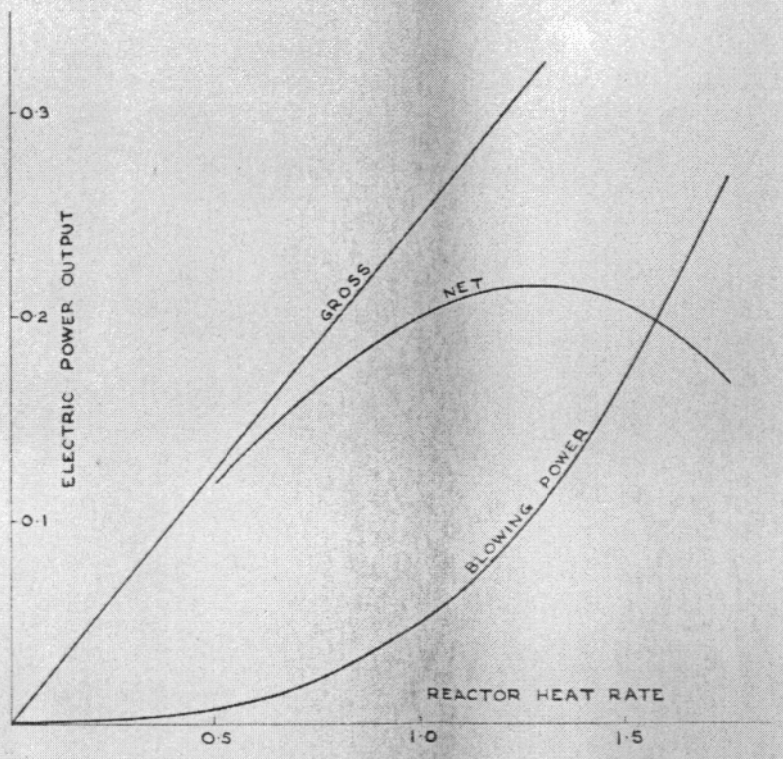

FIG. III.

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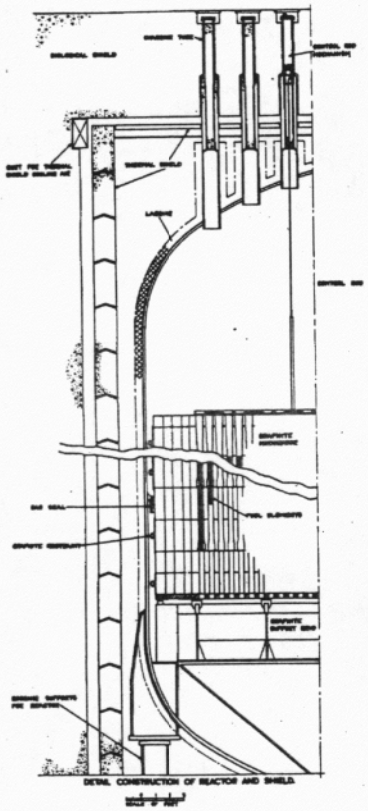
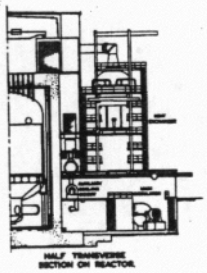
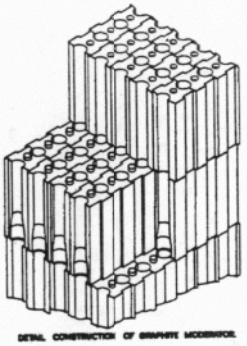


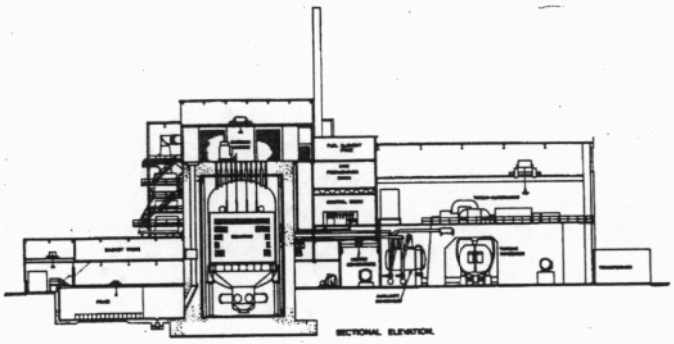
DETAIL CONSTRUCTION OF REACTOR AND SHIELD.



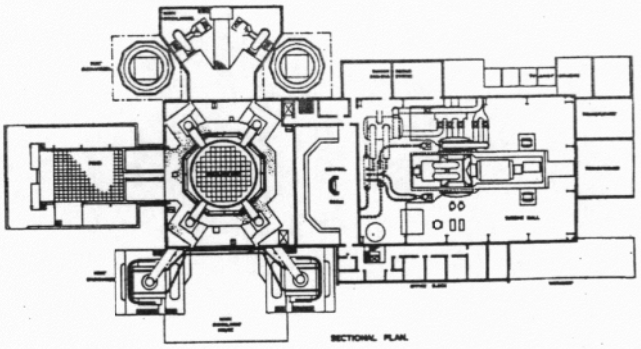
HALF TRANSVERSE SECTION ON REACTOR.



DETAIL CONSTRUCTION OF GRAPHITE MODERATOR.



SECTIONAL ELEVATION.



SECTIONAL PLAN.

NATURAL URANIUM POWER REACTOR. PROJECT "PIPPA."

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FIG. II.

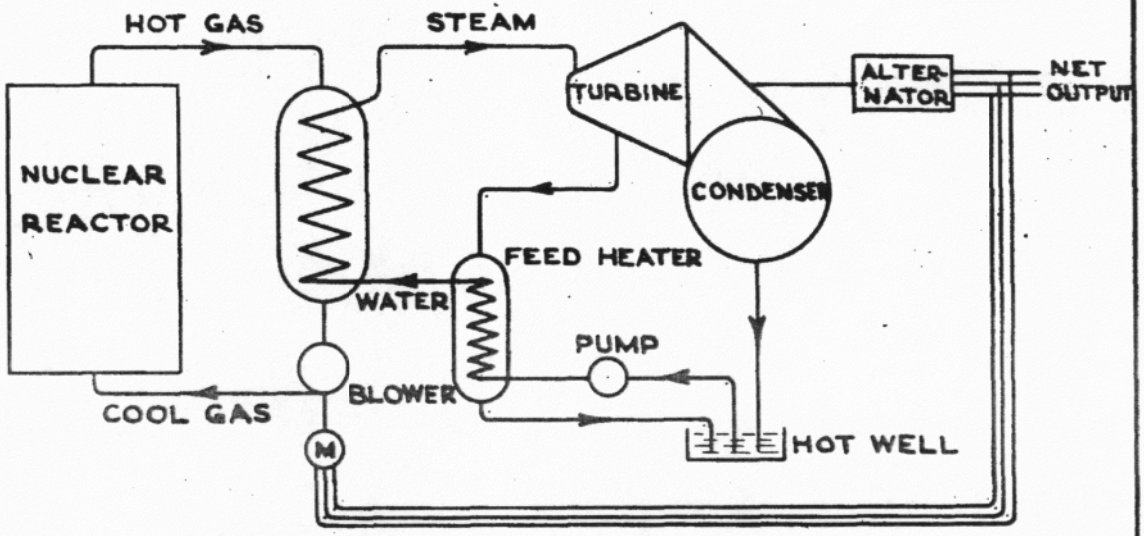

FIG. II.

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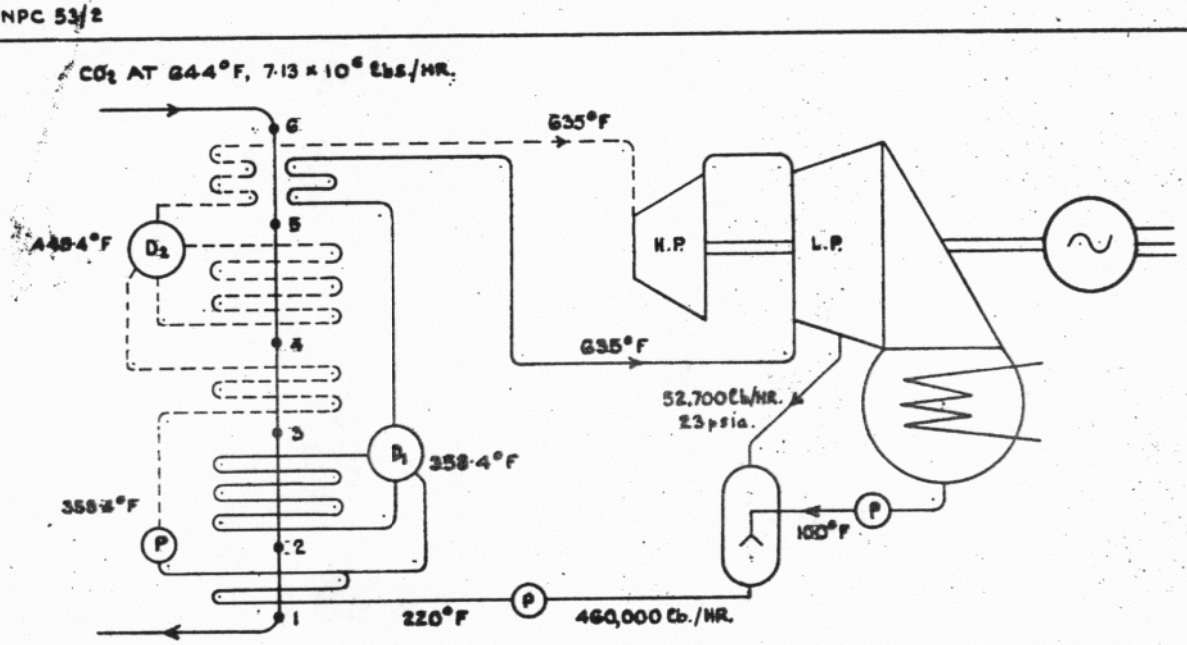
NPC 53/2

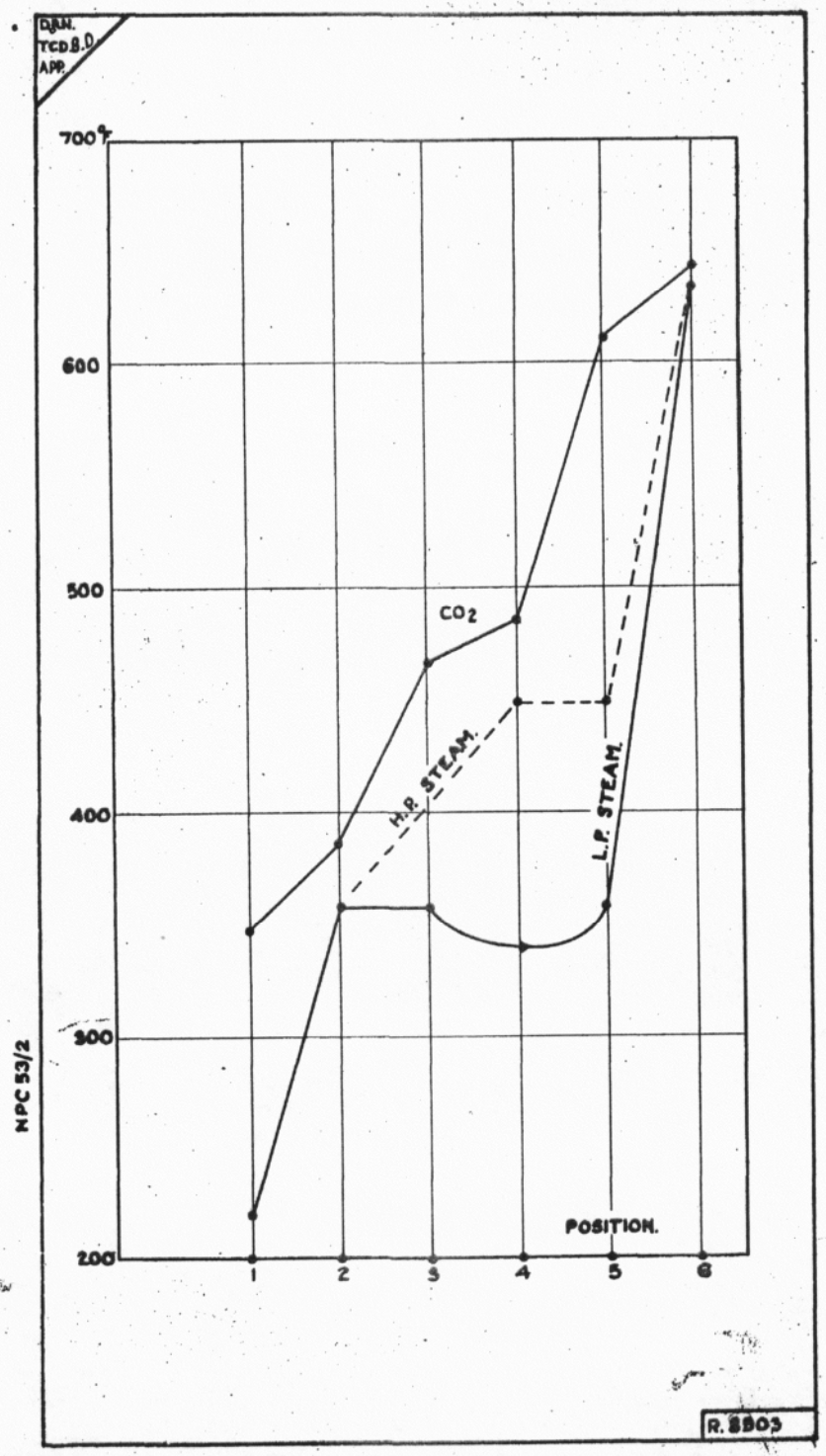


POSITION:	1	2	3	4	5	6
CO_2 TEMPERATURE:	347	382	407	484	611	644 °F

FLOW DIAGRAM FOR DUAL PRESSURE STEAM CYCLE.

D_1 = L.P. STEAM DRUM, 150 psia, 358·4°F, 164,1000 lbs/HR.
D_2 = H.P. STEAM DRUM, 420 psia, 448·4°F, 293,900 lbs./HR.

R. 8504


D.&W.
TCD.S.D.
APP.
700°F
600
500
400
300
200
CO_2
H.P. STEAM.
L.P. STEAM.
POSITION.
1
2
3
4
5
6
R. 8503

NPC 53/2