THE PRESSURIZED WATER REACTOR AS A SOURCE OF HEAT FOR STEAM POWER PLANTS

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INTRODUCTION

In his James Clayton lecture on 'Nuclear Reactors and Power Production' (Hinton 1954)‡ Sir Christopher Hinton spoke of the probable development of a number of different types of nuclear reactor forming a sequence in ascending order of thermal rating and descending order of core size. At one end of the scale there is the graphite-moderated gas-cooled reactor of the Calder Hall type. At the other extreme there is the plutonium-fuelled fast-fission reactor.

The graphite-moderated gas-cooled reactor is already being developed for widespread use during the first stage of the United Kingdom's nuclear power programme. It has many advantages, particularly in its ability to use *natural* uranium and in its inherent safety. When by-product plutonium becomes available in quantity from the early nuclear reactors, however, it will be desirable to have other reactors following on, which will be suitable for burning the by-product plutonium in conjunction with natural or depleted uranium in the most efficient way. Among the more promising possible types of thermal reactor which could be developed for use during this second stage are the sodium-cooled graphite reactor and the pressurized water-moderated reactor.

The purpose of the present paper is to examine the pressurized water reactor from the power plant engineer's point of view, and to consider some of the different ways in which it could be employed as a source of heat for a steam power station. The description 'pressurized water reactor' will be taken to include both light-water- and heavy-water-moderated reactors employing a pressurized water-cooling system. The general engineering features would be similar in either case. A heavy-water-moderated reactor, however, is capable of using natural uranium provided a calandria type of construction is employed, so that most of the moderator can be maintained at high density and, because of the rather open lattice arrangement required, it involves a rather larger diameter of core. A light-water-moderated reactor on the other hand must be fuelled either with enriched uranium or with a mixture of uranium and plutonium, and the ratio of moderator volume to fuel volume is much smaller than with the heavy-water reactor.

The core of a pressurized-water reactor will consist of a cylindrical assembly of fuel elements arranged in lattice formation and housed inside a pressure vessel. Each fuel element will normally be located inside a coolant channel or duct, and water will be circulated at a fairly high velocity to secure the required rate of heat removal. The moderator and coolant water must normally be pressurized to prevent the occurrence of boiling inside the core.

It is a feature of every type of nuclear reactor that the fuel element presents the most difficult problem of any part of the design. The fuel itself may take the form of metallic uranium, an alloy of uranium, or the oxide UO_2. As in the case of other reactors, the fuel must be protected by means of a can or sheath to prevent oxidation (in the case of metallic fuel) and to prevent the escape of fission products. The canning material must meet stringent mechanical, nuclear, and chemical requirements. It is not within the scope of this paper, however, to discuss the particular problems of the fuel element and core design. The approach adopted here is to look at the reactor as part of a steam power plant, to put some engineering limits on rates of heat removal, reactor pressures, coolant temperatures, etc., and, finally, to estimate the practicable range of steam conditions and turbine output. This having been done, it is easier to decide whether the pressurized-water reactor is of sufficient potential merit to justify the development work necessary to overcome the metallurgical and nuclear problems of the fuel element and core design and, above all, the problem of corrosion.

Notation

b	Effective perimeter of the element.
c_p	Specific heat of coolant.
d_1	Diameter of uranium rod.
d_2	Outer diameter of fuel element.
E	Young's modulus.
H	Rate of heat release per unit length at the centre.
H_0	Maximum rate of heat release per unit length at the centre.

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* *Kennedy and Donkin.*
† *Kennedy and Donkin.*
‡ *An alphabetical list of references is given in Appendix II.*

h Heat transfer factor.
k_s Thermal conductivity of the material of the sheath.
k_u Thermal conductivity of uranium fuel.
L Length of core.
L' Extrapolated length of core.
m Mass flow of coolant in the channel.
N_s Specific speed of pump.
p_h Hoop stress.
Q Flow, or heat release.
T Coolant temperature.
T_s Surface temperature of the element.
T_1 Inlet coolant temperature.
T_2 Outlet coolant temperature.
α Coefficient of expansion.
ΔT Radial temperature difference through wall of sheath.
σ Poisson's ratio.

THE NUCLEAR REACTOR AS A SOURCE OF HEAT

There is now sufficient published information on neutron diffusion theory and the properties of uranium to enable engineering calculations to be made regarding the sizes of certain types of nuclear reactor and the distribution of the heat release (Littler and Raffle 1955; Glasstone and Edlund 1952). Consider the case of a thermal reactor with a cylindrical core and reflector. The heat release per unit volume of core will be given by an equation of the form

$$Q = Q_{max} J_0(\alpha r) \cos(\beta x) \quad . \quad . \quad . \quad (1)$$

that is, the heat release follows a Bessel function distribution with radius, and a cosine distribution with axial distance (Fig. 1).

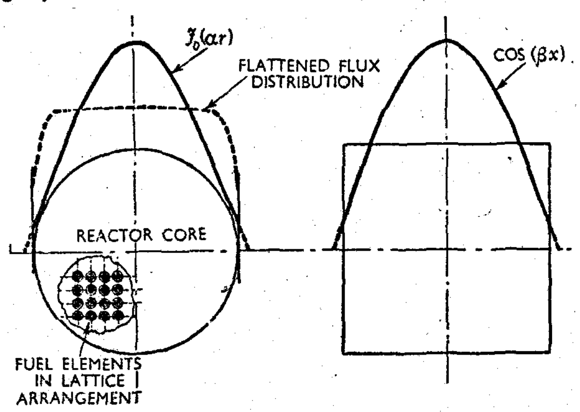


Fig. 1. Heat Release in a Reactor Core

In many reactor designs the radial distribution of the neutron flux and heat release is deliberately altered by 'flattening'. This can be achieved by using two different types of fuel element with different degrees of enrichment, by introducing neutron-absorbing material in the central region of the core, or by other devices. From an engineering point of view it is obviously desirable to avoid a large peak in the heat release at the centre of the core. It will therefore be reasonable to confine the investigation of the heat transfer between fuel element and coolant to a single channel, and to assume a channel rating equal to the radial average for the whole core.

The axial distribution of neutron flux and heat release may also be subject to distortion if control rods are employed which operate by axial movement into and out of the core. However, for the purposes of a preliminary calculation it will be sufficient to take the normal cosine distribution, that is, the heat release per unit length for a single channel will be given by

$$H = H_0 \cos \frac{\pi x}{L'} \quad . \quad . \quad . \quad . \quad (2)$$

where H_0 is the maximum rate of heat release per unit length at the centre of the channel; x is the axial distance measured from the centre; and L' is the extrapolated length of the core.

For heat transfer from the surface of a fuel element to the coolant, the heat flux q will be given by

$$q = \frac{H}{b} = h(T_s - T)$$

and hence

$$T_s - T = \frac{H_0}{bh} \cos \frac{\pi x}{L'} \quad . \quad . \quad . \quad . \quad . \quad (3)$$

where T_s is the surface temperature of the element; T is the coolant temperature; h is the heat-transfer factor; and b is the effective perimeter of the element.

The temperature rise of the coolant in flowing along a channel is given by

$$mc_p dT = H dx$$

or

$$\frac{dT}{dx} = \frac{H_0}{mc_p} \cos \frac{\pi x}{L'} \quad . \quad . \quad . \quad . \quad . \quad (4)$$

and hence

$$T_2 - T_1 = \frac{H_0}{mc_p} \frac{2L'}{\pi} \sin \frac{\pi L}{2L'} \quad . \quad . \quad . \quad (5)$$

where T_1 is the inlet coolant temperature; T_2 is the outlet coolant temperature; m is the mass flow of coolant in the channel; and c_p is the specific heat of the coolant.

The resulting temperature distribution along the surface of the fuel element is shown for a typical case in Fig. 2.

If boiling of the water inside the core is to be prevented, the maximum surface temperature of a fuel element must not be allowed to rise above the saturation temperature at the reactor pressure. This limitation will fix the permissible temperature levels throughout the reactor.

The difference between the maximum surface temperature $T_{s_{max}}$ and the coolant inlet temperature T_1 is of particular interest, and this is given conveniently by Ginns's equation which can be derived from equations (3) and (4).

$$T_{s_{max}} - T_1 = \tfrac{1}{2}(T_2 - T_1) + \sqrt{\Delta T_0^2 + \left\{\frac{\tfrac{1}{2}(T_2 - T_1)}{\sin \frac{\pi L}{2L'}}\right\}^2} \quad (6)$$

where ΔT_0 is the value of $(T_s - T)$ at $x = 0$;

that is, $\Delta T_0 = H_0/bh$

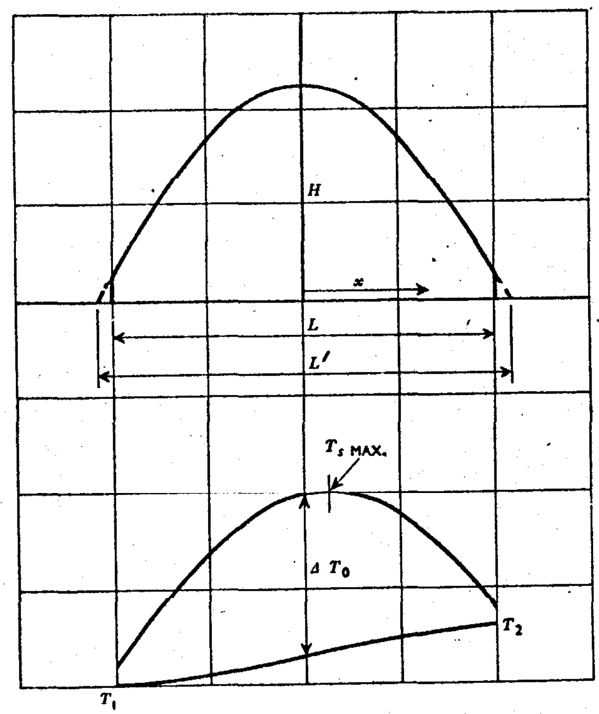


Fig. 2. Heat Release and Temperature Distribution in a Channel

Size and Rating of Fuel Elements

It is convenient to start with the case of a fuel element of circular cross-section surrounded by an annular passage for the coolant, as shown in Fig. 3. The nuclear calculations for the core are usually based on an equivalent lattice element, or unit cell, of this type even though a different shape for the fuel element may ultimately be chosen.

The heat-transfer calculation is most easily carried out by taking a particular value for the coolant velocity, evaluating the heat-transfer factor h from the normal data for forced convection to fluids flowing in pipes and annuli, and then calculating ΔT_0 and (T_2-T_1) for various values of H_0. The Ginns's equation may then be used to give the overall temperature difference $(T_{s_{\max}}-T_1)$. Temperature differences may then be plotted against coolant velocity for various types of fuel element and for different values of the thermal rating. Some typical results are shown in Fig. 4 for a channel 8 feet in length with circular fuel rods of $\frac{1}{2}$, $\frac{3}{4}$, and 1 inch outside diameter and with a $\frac{1}{8}$-inch annulus for the coolant passage. Some allowance has been made in these calculations for deterioration of the heat-transfer surfaces as a result of corrosion or deposition by the coolant. The precise value of the fouling factor to be employed in estimating the overall heat-transfer factor h will naturally depend on the material used for the fuel element cans and also on the treatment of the water.

The upper practical limit to the coolant velocity will be set by the pressure drop in the channel, and by other considerations such as erosion. The calculated loss of head for

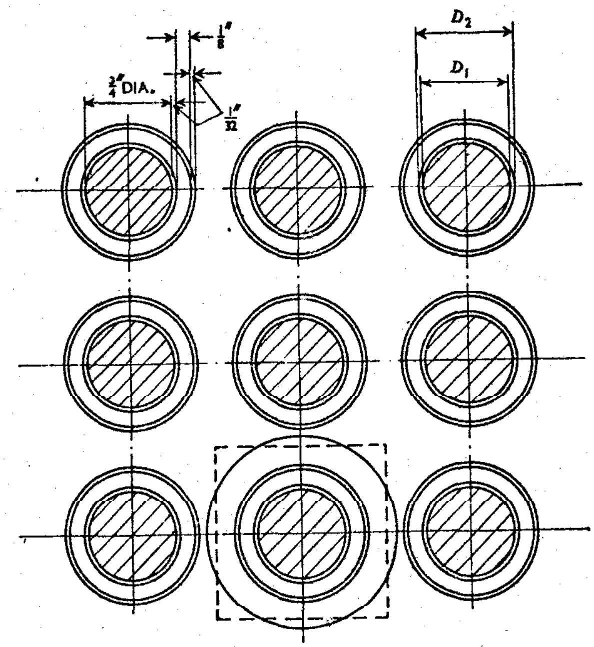


Fig. 3. Typical Lattice Arrangement Showing Fuel Rods and Coolant Channels

water flowing in the channel is shown by the dotted curve in Fig. 4. If the permissible coolant velocity is fixed it will be observed from Fig. 4 that, for a given temperature difference $(T_{s_{\max}}-T_1)$, there will be a maximum permissible fuel rating for every different shape and size of fuel element. To take an example, if the coolant velocity is limited to 25 ft. per sec., and if the reactor design aims at a temperature difference $(T_{s_{\max}}-T_1)$ which is not to exceed 100 deg. F., the permissible heat rating of the fuel for circular rods (from Fig. 4) would be as follows:

Rod diameter, inch	$\frac{1}{2}$	$\frac{3}{4}$	1
Million B.Th.U. per hr. cu. ft.	21·5	13·7	10·0

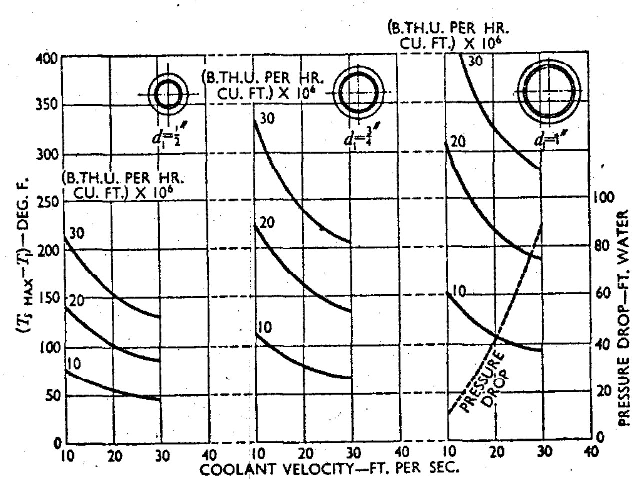


Fig. 4. Heat-transfer Calculations

A light-water reactor of the type under consideration might have between 40 and 50 tons of slightly enriched uranium in the core. If the design aims at a total heat output of some 500 MW., the average fuel rating would have to be about 20 million B.Th.U. per hr. cu. ft. This could be achieved either with circular rods of slightly over ½ inch diameter, or by using fuel elements of a different shape altogether but having an equivalent diameter of approximately ½ inch. The need to employ elements of small diameter, and the large number required, points to the desirability of grouping elements in clusters for convenience in charging and discharging. Some typical non-circular fuel elements are shown diagrammatically in Fig. 5. These may be compared with the particular proposals reported for different design studies (Simpson and others 1955; Iskenderian and others 1955; Harrer and others 1955).

Having reached the provisional conclusion that a heat rating of the order of 20 million B.Th.U. per hr. cu. ft. should be feasible as regards heat transfer, it is necessary to examine the limits imposed by heat flow within the fuel itself and the thermal stresses in the fuel elements. For a plain circular fuel element the difference between the central metal temperature and the surface temperature is given by

$$T_0 - T_s = \frac{H}{2\pi}\left[\frac{1}{2k_U} + \frac{1}{k_s}\log_e \frac{d_2}{d_1}\right] \quad . \quad . \quad (7)$$

where k_U is the thermal conductivity of the uranium fuel; k_s is the thermal conductivity of the material of the sheath; d_2 is the outer diameter of the fuel element; and d_1 is the diameter of the uranium rod.

It is generally desirable for a fuel element to be designed so that the sheath provides the necessary structural and mechanical strength, that is, no reliance should be placed on the uranium. The maximum thermal stress set up in a thin cylindrical sheath is given by

$$p_h = \frac{\alpha E \Delta T}{2(1-\sigma)} \quad . \quad . \quad . \quad . \quad (8)$$

where p_h is the hoop stress (compression at inner surface, tension at outer surface); α is the coefficient of expansion; E is Young's modulus; σ is Poisson's ratio; and ΔT is the radial temperature difference through the wall of the sheath.

An additional hoop stress, however, may be caused by the expansion of the fuel inside the sheath.

Table 1. Heat Flux and Thermal Stress

Fuel-rod diameter, inch	½	¾	1
Average value of H	27,200	61,300	109,000
Average surface heat flux	208,000	312,000	416,000
Maximum heat flux, B.Th.U. per hr. sq. ft.	312,000	468,000	624,000
$(T_0 - T_s)$ max., deg. F.	210	442	759
ΔT max., deg. F.	44·5	67·4	90
Hoop stress, tons per sq. in.	4·82	7·28	9·7

The results in Table 1 refer to an average heat rating of 20 million B.Th.U. per hr. cu. ft. The peak rating (at the centre of the channel) is taken to be 1·5 times the average. A stainless-steel sheath of thickness $\frac{1}{32}$ inch is assumed.

It is evident that a definite limit will be imposed on the diameter of the fuel rod for a given thermal rating, that is, the higher the rating required the smaller must be the equivalent diameter. This rating will depend, however, on the choice of materials and on the detail design of the fuel element. The above figures refer only to the simple case of a plain cylindrical element with stainless-steel sheath, but they give a fair picture of the order of magnitude of the internal temperature gradients and thermal stresses. An upper limit would have to be set in this case at a diameter of about ¾ inch for an average thermal rating of 20 million B.Th.U. per hr. cu. ft.

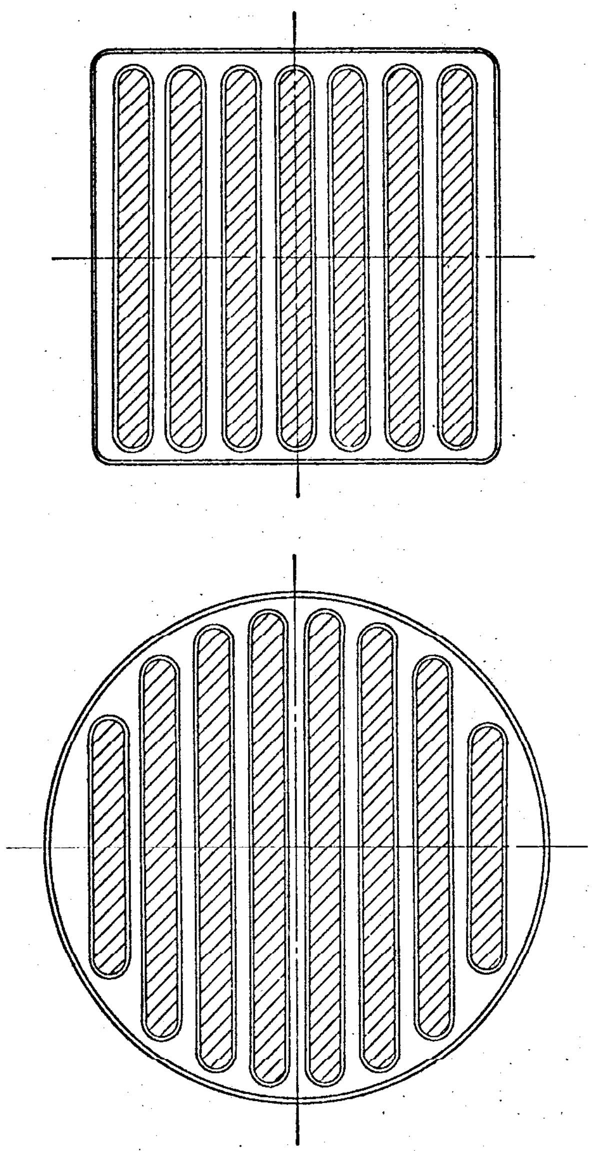


Fig. 5. Flat-strip Fuel Elements

THE PROBLEM OF THE PRESSURE VESSEL

The core diameter of a light-water reactor of the type under consideration might be between 6 and 8 feet, depending on the degree of enrichment and quantity of the fuel to be used. The core diameter of a heavy-water reactor, on the other hand, would be larger than this, owing to the greater moderator/uranium ratio required and the correspondingly more open lattice.

Taking the case of the light-water reactor and allowing for an adequate thickness of reflector surrounding the active core, a pressure vessel of diameter 9 or 10 feet would be required. The absolute temperature level in the reactor will be fixed by the permissible reactor pressure, and it is therefore necessary to determine the upper practical limit. Table 2 gives the wall thickness required for a fusion-welded mild-steel pressure vessel of internal diameter 10 feet.

Table 2. Wall Thickness and Working Pressure

Reactor working pressure, lb. per sq. in.	1,000	1,500	2,000
Wall thickness, inches	5	7½	10

The 1,500 lb. per sq. in. case probably represents a reasonable upper limit, and should not involve going far outside the range of normal welding techniques or existing manufacturing capacity.

The most difficult part of the pressure vessel design will be the method of end closure. It would be desirable to have a removable end or top cover which could be opened up for major refuelling operations, protection from radiation under these conditions being provided by flooding the reactor with water to a level of some 6 or 8 feet above the top of the core. The design of a satisfactory flanged and bolted joint for the diameter and pressure envisaged, however, presents a difficult problem.

The leading proportions of a typical pressure vessel of 9 feet internal diameter, designed for a working pressure of 1,500 lb. per sq. in., are shown in Fig. 6. Taking 1,500 lb. per sq. in. as the limiting reactor pressure, the corresponding saturation temperature will be 596 deg. F. and this will fix the permissible temperature levels throughout the plant. In the limiting case, with local nucleate boiling at the hottest point on the surface of the fuel element, the permissible value of $T_{s_{max}}$ would therefore be equal to the saturation temperature 596 deg. F. Some provision, however, must be made for departure from design conditions, and in the following analysis of the steam cycle an additional 20 deg. F. will be allowed, that is, the maximum surface fuel-element temperature will be taken as 576 deg. F.

There are various possible methods for maintaining and controlling the pressure in the reactor-vessel and coolant circuit. One arrangement described (Simpson and others 1955) involves the use of a special pressurizing tank or steam dome, in which the water is heated electrically to the boiling point corresponding to the reactor pressure.

PRIMARY COOLANT CIRCUIT AND EXTERNAL HEAT EXCHANGERS

It has been noted that with fuel elements of approximately ½ inch equivalent diameter, the temperature difference $(T_{s_{max}} - T_1)$ could be limited to about 100 deg. F. at a heat rating of 20 million B.Th.U. per hr. cu. ft. without employing excessive water velocities. The temperature rise $T_2 - T_1$ of the coolant in a channel will depend on the heat rating for the channel and the mass flow of water. For a given heat rating for the fuel, and for a given water velocity, the

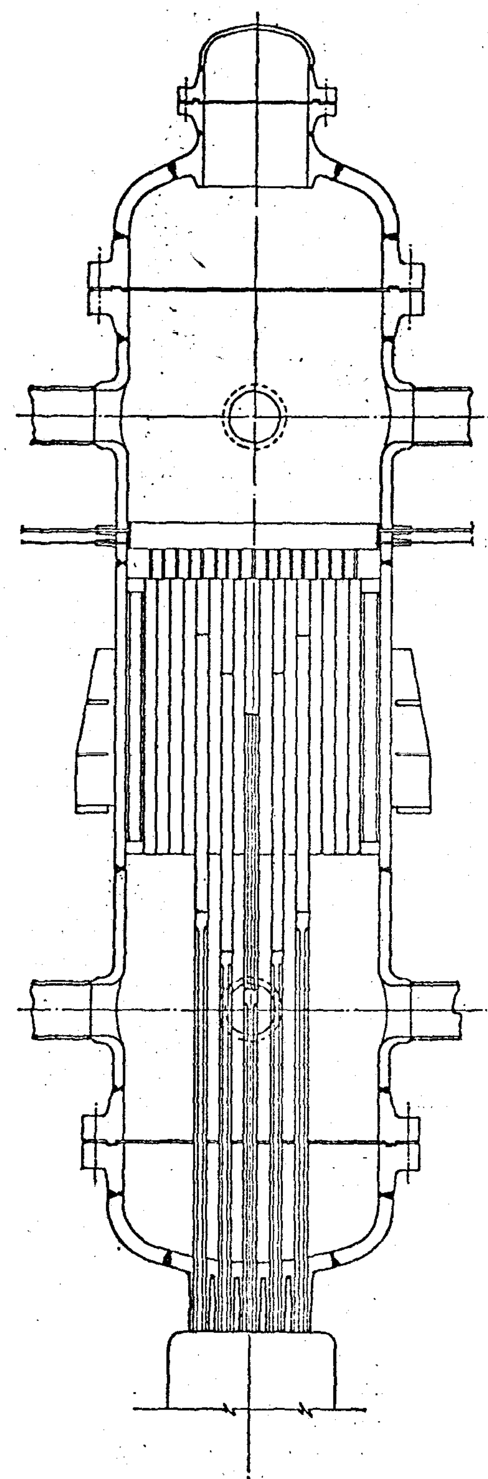


Fig. 6. Reactor Pressure Vessel

coolant temperature rise $T_2 - T_1$ will therefore depend on the geometry of the channel. In the following calculation a value of 36 deg. F. will be taken as a typical figure. For a 500-MW. reactor this would imply a total mass flow of water of approximately 12,000 lb. per sec. or 90,000 gal. per min. at 1,500 lb. per sq. in.

Taking these values for the temperature differences, and taking a maximum fuel element surface temperature of 576 deg. F., the following typical figures are arrived at for the temperatures in the reactor and coolant circuit:

	Temperature, deg. F.
Maximum central uranium . .	786
Maximum fuel element surface . .	576
Coolant outlet	512
Coolant inlet	476

For the external heat exchanger in this case a steam pressure of 450 lb. per sq. in. and saturation temperature of 456 deg. F. may be assumed. The heat-transfer diagram is shown in Fig. 7. It is assumed that the feed water is heated

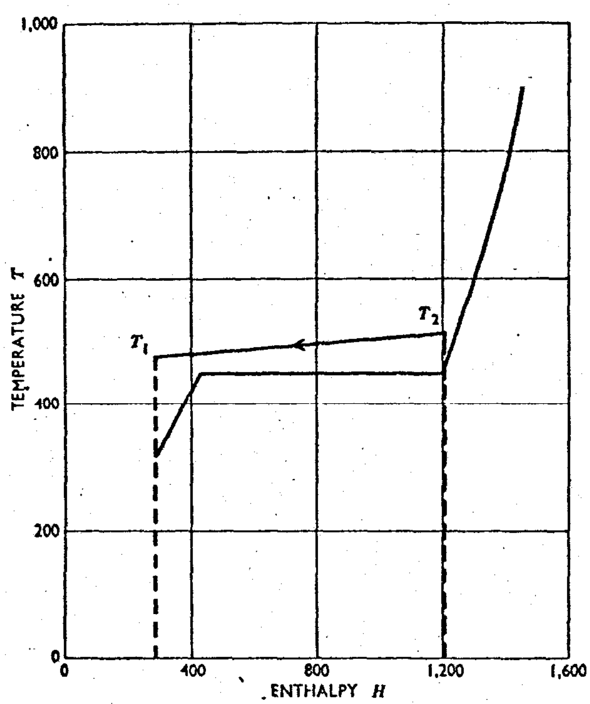


Fig. 7. Heat-transfer Diagram

to approximately 315 deg. F. by bled-steam feed-heating before entry to the heat exchanger. The water is thus heated from 315 to 456 deg. F. in the main heat exchanger and evaporated at the latter temperature to give a supply of saturated steam at 450 lb. per sq. in. Taking an average overall heat-transfer coefficient of about 800 B.Th.U. per sq. ft. hr. deg. F., the total surface area required for a heat load of 500 MW. would be approximately 50,000 sq. ft.

One possible design for an external heat exchanger is shown in Fig. 8. This diagram refers to a shell and tube heat-exchanger arranged as a natural circulation unit. Feed water enters the shell at the bottom and after passing through a feed preheating section it mixes with the boiler circulation water which is returned from the boiler drum. The mixture of steam and water leaves the heat exchanger at the top of the shell and passes to the boiler drum. High-pressure water

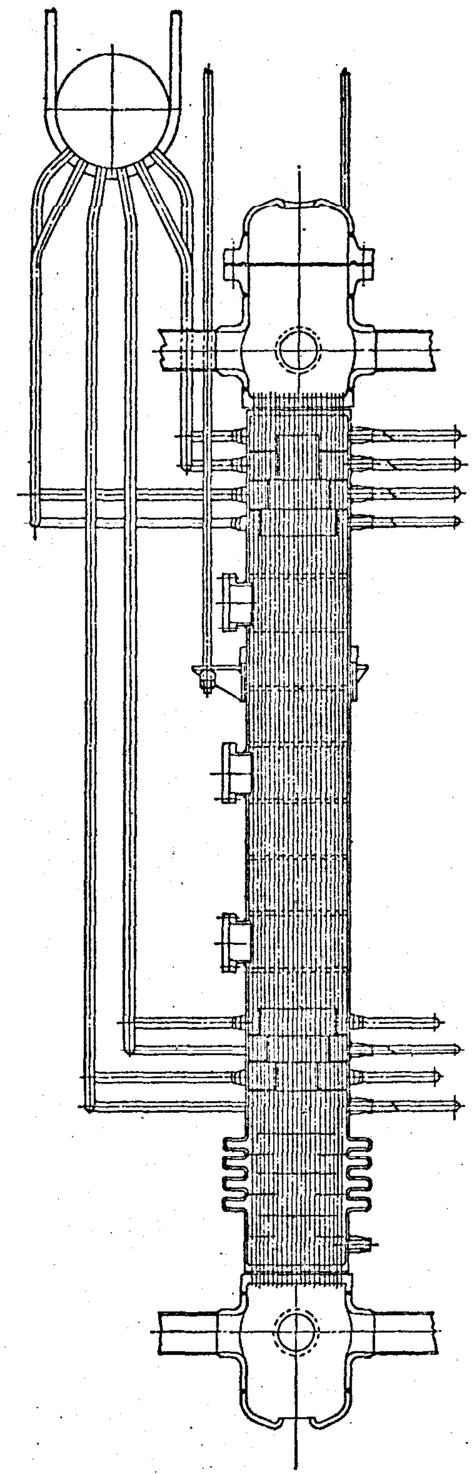


Fig. 8. Vertical Steam-raising Plant

from the reactor enters the header at the top of the unit and passes downward inside the tubes of the heat exchanger, returning to the reactor by way of the header at the bottom. Another alternative would be to use a horizontal heat exchanger of the shell and tube type with a single fixed tube-plate and high-pressure header. The tube bundle could be made up from hair-pin tubes or, alternatively, straight tubes could be used with a floating header at the other end.

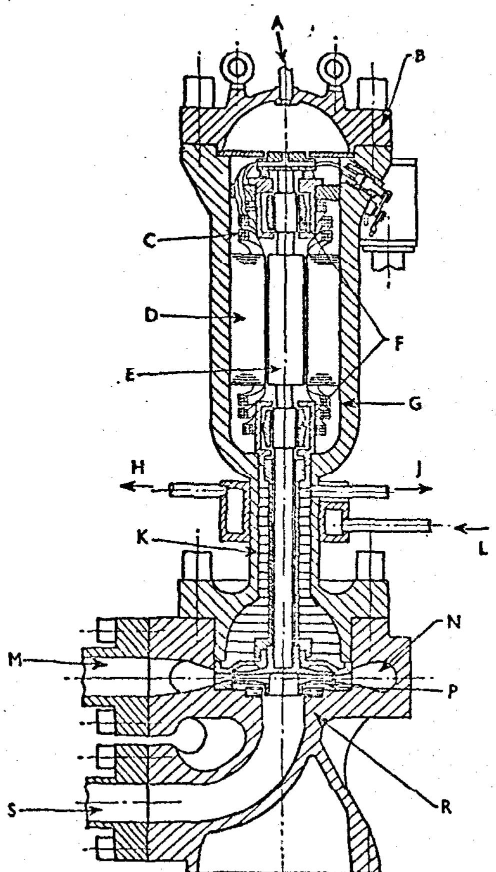


Fig. 9. Glandless Circulating Pump

A High-pressure circulation in.
B Pressure cover.
C Stator windings.
D Stator laminations.
E Rotor.
F Tilting-pad journal bearings.
G Pressure shell.
H Low-pressure cooling-water from jacket.
J High-pressure circulation out.
K Heat baffles.
L Low-pressure cooling-water to jacket.
M Discharge.
N Volute.
P Impeller.
R Pump case.
S Suction.

Other arrangements for the heat exchanger are also possible, and either natural or forced circulation may be employed.

If the coolant circuit is divided into four parallel branches, the flow in each circuit will be 22,500 gal. per min. to provide for a total heat removal of 500 MW., with a coolant temperature range of 36 deg. F. Alternatively, six parallel circuits might be employed, each with a flow of 15,000 gal. per min. Taking the latter case, and assuming a total pumping head of 100 feet, the power input to the water required from each pump would be about 360 h.p. and the horsepower required for each driving motor would be about 500. The specific speed of a pump designed for this duty and driven at a speed of 970 r.p.m. would be $N_s = NQ^{\frac{1}{2}}/H^{\frac{3}{4}} = 3{,}760$, which is a reasonable value for a single-stage mixed-flow pump. In view of the high pressure in the primary coolant circuit and to avoid any danger of leakage, it would be desirable to employ a glandless pump with totally enclosed motor drive. One possible type is shown in Fig. 9.

From the point of view of the boiler designer, it will be desirable to control the pH of the water to an alkaline value between 10 and 11. It is possible, however, that if any of the normal boiler additives are used trouble may be experienced owing to decomposition under irradiation in the reactor. The possibility must therefore be faced of having to use pure demineralized and deaerated water, which could be extremely corrosive at the temperatures and pressures considered. It may therefore be necessary to employ stainless-steel tubes for the heat exchanger and to provide a stainless interior surface to the entire primary coolant circuit. This problem of materials can only finally be settled in the light of operating experience with a complete plant. There will certainly be a strong incentive to find a suitable chemical additive for the water which would enable mild steel to be used throughout the coolant circuit.

THE EXTERNAL STEAM CIRCUIT

Having arrived at the conclusion that, with a reactor pressure of 1,500 lb. per sq. in., saturated steam can be generated under practical conditions at a maximum pressure of about 450 lb. per sq. in., the next question is the choice of steam cycle and the design of the external plant. In the following comparison a number of different arrangements are analysed on the assumption that steam is generated at a normal working pressure of 420 lb. per sq. in. These schemes are shown diagrammatically in Fig. 10.

(a) Saturated Steam with Water Separation

The simplest arrangement is shown in Fig. 10*a*, where it is assumed that saturated steam at a pressure of 420 lb. per sq. in. is available at the turbine stop-valve. Expansion takes place inside the turbine in the wet region, but at an intermediate point the entire flow of steam and moisture is discharged from the turbine to an external separator where moisture removal takes place. Steam from the separator, which is assumed to be only 1 per cent wet, then passes to the low-pressure cylinder of the turbine, where the expansion is continued in the wet region down to a final wetness of approximately 15 per cent at the exhaust to the condenser. The difficulty with this scheme is that unless the separator is very large, the pressure drop associated with the moisture removal process may become excessive. It is extremely difficult to remove finely divided particles of water from a steam mixture without incurring a large drop in pressure.

(b) Saturated Steam with Intermediate Reheat

As an alternative to water separation, reheating of the steam at the intermediate pressure could be employed as indicated in Fig. 10*b*. The difficulty about this scheme, however, would be the complication of the primary coolant circuit and the need to bring high-pressure water from the reactor to another heat exchanger adjacent to the turbine.

(c) Saturated Steam with Live Steam Reheat

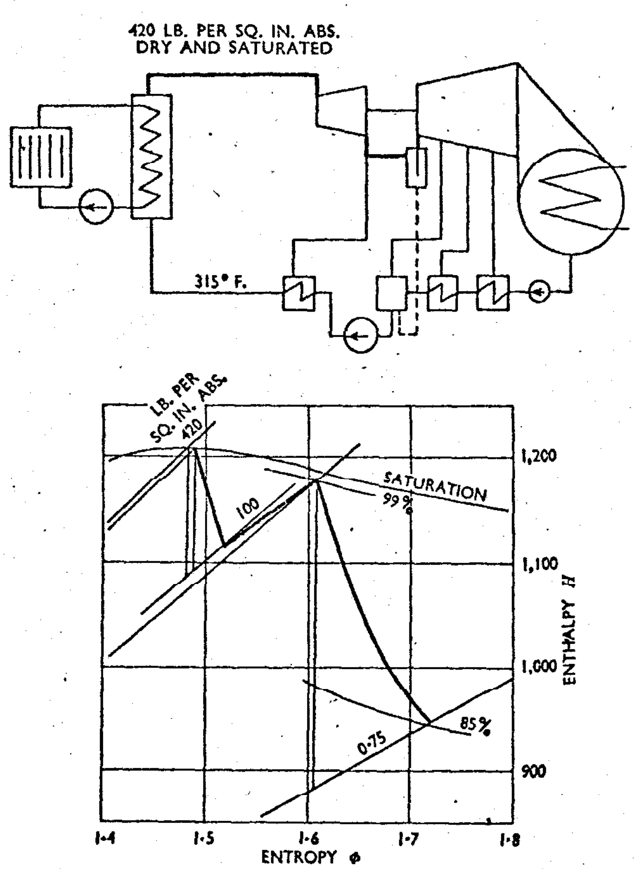
As an alternative to employing high-pressure water from the reactor as the reheating agent, live steam from the main heat exchanger could be employed for this purpose and the arrangement is shown diagrammatically in Fig. 10*c*. This would involve some sacrifice in efficiency, but the plant would be relatively simple.

(d) Superheat from the Reactor with Lowered Boiler Pressure

In arrangement (d), shown in Fig. 10*d*, the boiler pressure is lowered to 210 lb. per sq. in., which enables sufficient superheat to be supplied from the reactor, using the same terminal primary coolant temperatures to enable the steam to be expanded directly through the turbine with no intermediate extraction. This arrangement offers some advantages in simplicity of plant but is basically wrong thermodynamically, and results in a lower overall efficiency.

(e) External Superheating from a Combustion Source

Arrangement (e) is shown in Fig. 10*e*. Saturated steam at a pressure of 420 lb. per sq. in. is taken from the boiler drum of the heat exchanger and is heated in a coal-fired or oil-fired superheater to a temperature of 900 deg. F. The superheated steam can then be expanded through the turbine without complication from excessive wetness at the exhaust end. Thermodynamically the arrangement is good. Heat from the nuclear reactor is used at moderate temperature to evaporate the water, while combustion heat is employed only for the high-temperature end of the cycle where it can be used at high thermal efficiency. The difficulty about this arrangement is that the plant would be much more complex. All the problems of fuel storage, fuel handling, dust removal, and ash disposal associated with a conventional thermal station would be encountered, in addition to the added problems of the nuclear reactor and the special arrangements for handling of fuel elements. This type of scheme could be justified, however, on favourable sites for units of the largest sizes.



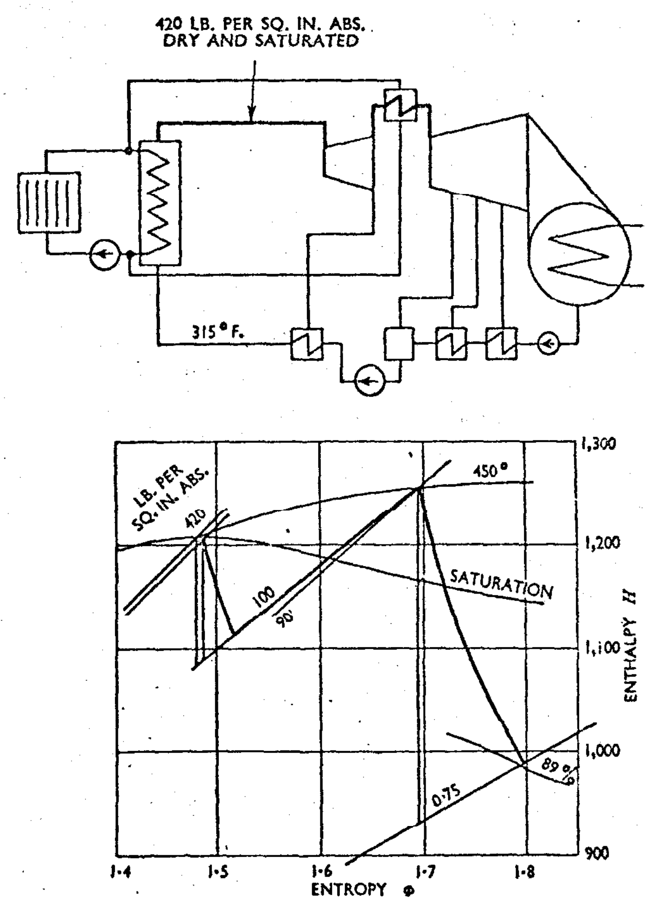


Fig. 10. Steam Cycle Arrangements

Table 3. Comparison of Steam Cycles

	a	*b*	*c*	*d*	*e*
Cycle	Water extraction	High-pressure water reheater	Live steam reheater	Low-pressure superheated steam	Separately superheated steam
Turbine stop-valve pressure, lb. per sq. in. abs.	420	420	420	210	380
Turbine stop-valve temperature, deg. F.	450	450	450	450	900
Reheater inlet pressure, lb. per sq. in. abs.	—	100	100	—	—
Reheat temperature, deg. F.	—	450	400	—	—
Back pressure, lb. per sq. in. abs.	0·75	0·75	0·75	0·75	0·75
Final feed temperature, deg. F.	315	315	315	270	315
Overall thermal efficiency, per cent	27·8	27·8	27·3	24·3	32·3
Steam rate to turbine, lb. per kW.-hr.	13·28	11·65	11·84	13·9	9·04
Electrical output, MW.	139·0	139·0	136·5	123·0	213
Steam to turbine, lb. per hr.	1,850,000	1,620,000	1,620,000	1,705,000	1,850,000
Dry steam to condenser, lb. per hr.	1,140,000	1,140,000	1,140,000	1,188,000	1,368,000
Exhaust area, sq. ft.	139	139	139	145	167

COMPARISON OF THE FIVE STEAM CYCLES

The cycle efficiencies have been calculated using assumptions which are generally similar to those made by Baumann (1946) and London (1952–53). In each case the back pressure has been taken as 0·7 lb. per sq. in., and the leaving losses at 20 B.Th.U. per lb. flow through the last stage. The turbo-generator mechanical and electrical losses have been taken as 3 per cent. These assumptions are arbitrary and have been made for comparative purposes only.

A summary of the results is given in Table 3. Owing to the relatively low initial temperatures and moderate heat drops, the specific output of all the cycles is low if viewed in the light of modern high-temperature steam-plant practice. Even the least efficient cycles, however, can

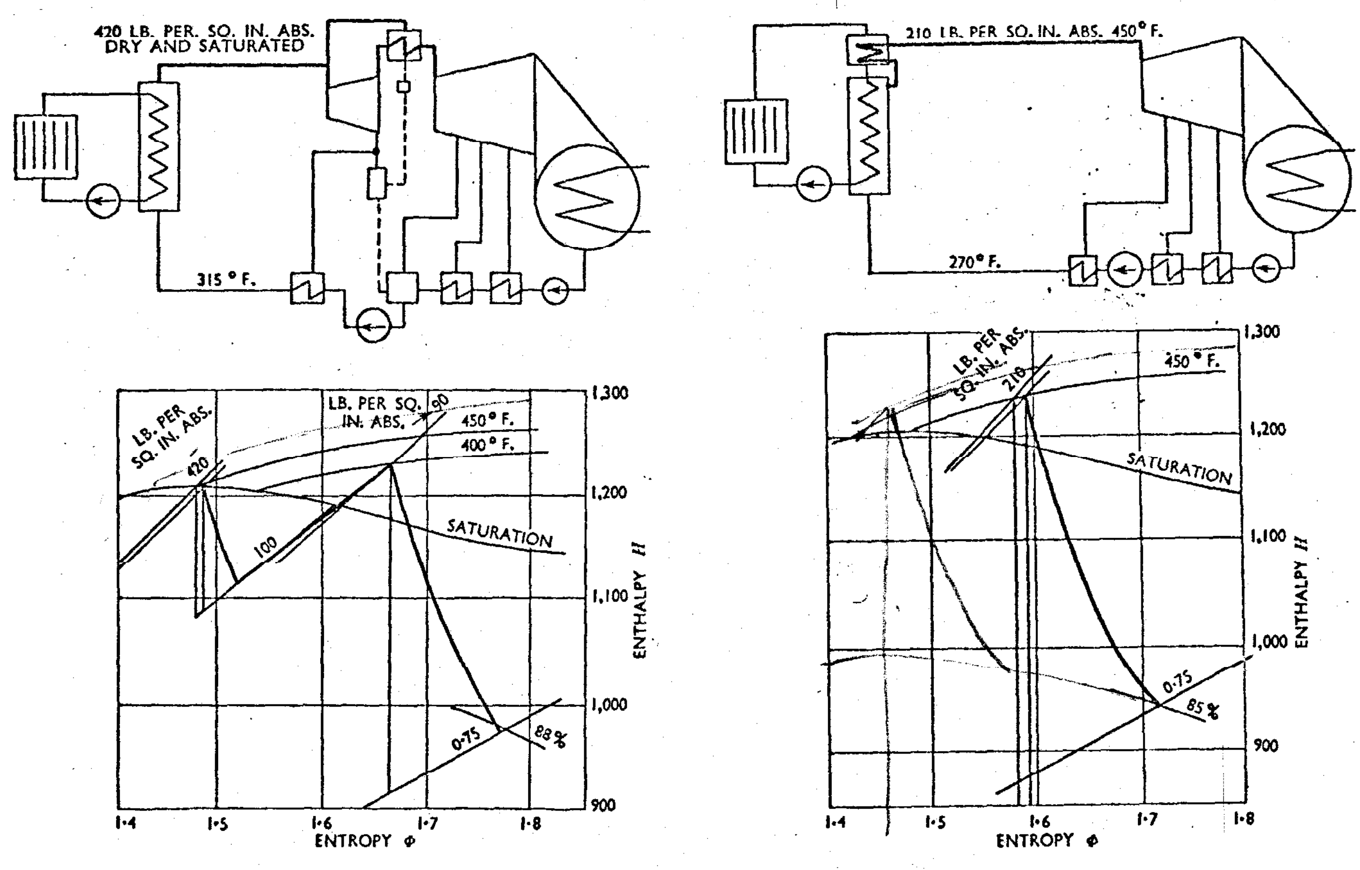
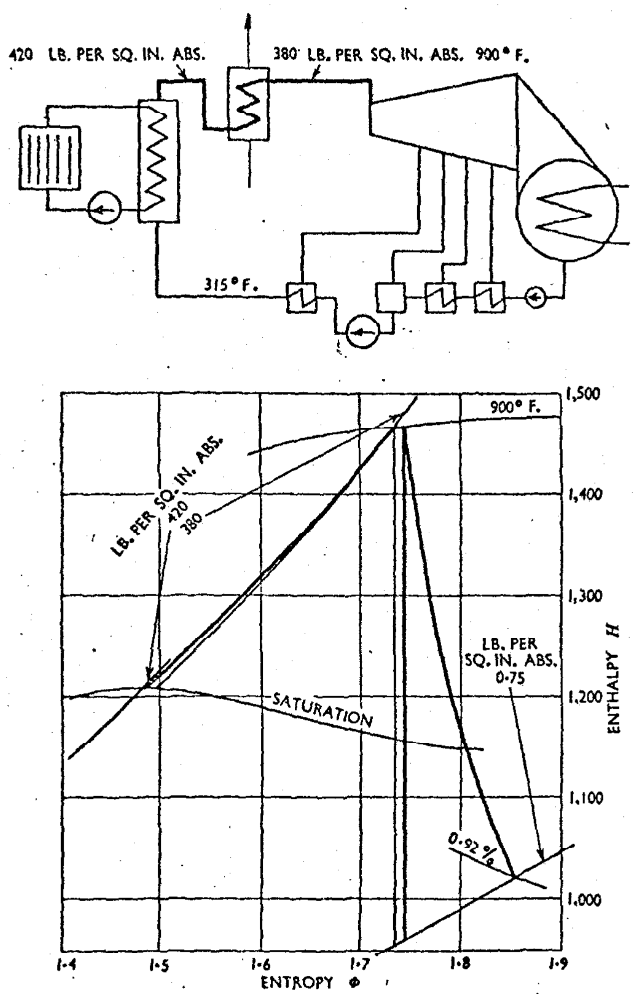


Fig. 10—continued

produce over 120 MW. of electrical power from 500 MW. of reactor heat. While the size of turbo-generator required for such plant is within the normal range of existing designs, some problems arise in connexion with the low-pressure end of the turbine, which would not be encountered in a high-pressure machine of equal output. A triple exhaust arrangement is to be used on the first 200-MW. high-pressure machines of speed 3,000 r.p.m. now on order from British manufacturers. The water extraction or reheat cycles shown in Table 3 might just be covered by using a turbine having a similar frame. The other two cycles, particularly the 213-MW. separately-superheated cycle, would require four such exhausts if developed using a single machine of speed 3,000 r.p.m. By using a turbine of speed 1,500 r.p.m., however, an exhaust area of at least 75 sq. ft. and possibly as much as 100 sq. ft. could be accommodated. The 213-MW. separately-superheated cycle could therefore be developed using a single-shaft machine running at a speed of 1,500 r.p.m. and with a double exhaust. Alternatively, there should be no difficulty in building two



e

Fig. 10—continued

machines of speed 3,000 r.p.m. of approximately 100 MW. each, two such machines being associated with a single 500-MW. reactor.

If the separately-superheated cycle is compared with the unsuperheated water-extraction cycle it will be seen that an additional electrical output of 74 MW. is obtained. The combustion heat required for superheating the steam, assuming a figure of 88 per cent combustion efficiency, would be 163·5 MW. The overall efficiency of the combustion end of the cycle is therefore 45·2 per cent. The two cases are set out for comparison in Table 4, taking 500-MW. reactor heat in each instance.

Table 4. Comparison of Efficiency of Two Steam Cycles

	Water-extraction cycle	Separately-superheated cycle
Reactor heat, MW.	500	500
Combustion heat, MW.	—	163·2
Electrical output, MW.	139·0	213
Overall efficiency, per cent	27·8	32·3
Efficiency of utilization of combustion heat, per cent	—	45·2

When using a coal-fired or oil-fired superheater it would be possible to provide some of the feed heating from the combustion heat. It will generally be found preferable, however, to make use of regenerative feed-heating as shown in Fig. 10*e* and to provide a large air preheater at the exhaust end of the combustion plant.

When considering the use of a separate superheater it may be of interest to investigate the possibility of varying the electrical output of the plant by altering the heat input to the superheater with no change of the heat loading on the reactor. If the turbine could be operated over the full temperature range at inlet, that is, between 450 and 900 deg. F., it would be theoretically possible to vary the output of the plant to about 70 per cent of full load by control of heat input to the superheater alone. It is highly improbable, however, that rapid load changes could be dealt with in this way without causing damage to the machine through thermal shock.

If the use of an auxiliary combustion fuel is excluded, because of restrictions with regard to the site or for other reasons, the problem remains of choosing the best method for handling saturated steam in the turbine. The ideal would be to expand the saturated steam through the turbine without recourse to any external reheater or water separator. Various methods of water drainage from individual turbine stages are employed in existing turbine designs. A proportion of moisture present in the steam can be collected from the outer periphery of a stage and the collected condensate can be discharged through suitable ducts. This condensate can then be drained to a feed-heating stage or to the main condenser. As much as 40 per cent of the moisture present can be extracted in this manner in favourable circumstances. It does not seem impossible to design a satisfactory turbine, incorporating a number of individual drainage points to enable saturated steam at a pressure of 420 lb. per sq. in. to be used, without excessive trouble from blade erosion. A typical steam condition line for such an arrangement is shown in Fig. 11.

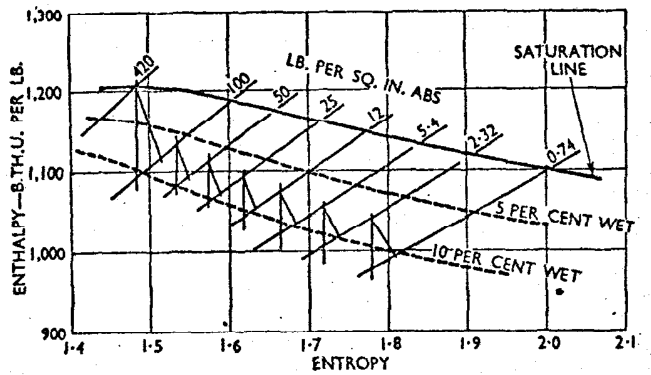
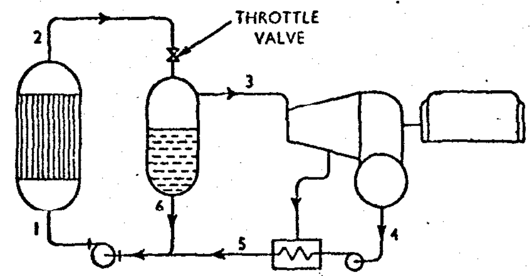


Fig. 11. Expansion of Wet Steam with Water Extraction

BOILING-WATER REACTORS

If water is to be used in a reactor as the coolant and moderator, there is a natural incentive to use this water directly for the generation of steam. The external heat-exchanger would then be discarded and the steam pressure in the boiler drum could approach the pressure inside the reactor vessel. There are two main alternatives: (*a*) a flash-boiling system in which bulk boiling of the water would be just suppressed inside the core and the generation of steam would take place mainly outside the reactor in a separate flash vessel, and (*b*) a boiling-water reactor in which bulk boiling is allowed to occur inside the reactor vessel itself. In either case there is the problem of the possible contamination of the turbine with radio-active material. Under normal operating conditions the level of activity at the turbine should not be serious, but in the event of a fuel element failure some difficulty might be experienced in decontaminating the external circuit. The feasibility of the direct use of steam from a reactor will depend both on the speed with which a rise in activity can be detected and the steam flow diverted from the turbine, and on the successful design of a fuel element which will not deteriorate rapidly in the event of a minor fault developing in the sheath.

The flash steam cycle, which was first put forward by Sir Christopher Hinton in his James Clayton lecture (Hinton 1954) is shown diagrammatically in Fig. 12. Sub-cooled water at state 1 enters the reactor and is heated as a single-phase fluid to state 2, that is, up to the boiling point at the reactor pressure. This implies that the surface temperature of the fuel elements is maintained at some value higher than the boiling point of the water and some local nucleate boiling will therefore occur adjacent to the hottest parts of the fuel element heat-transfer surface. Bulk boiling in the core, however, would be just suppressed. Hot water emerges from the reactor at the boiling point corresponding to the reactor pressure, and is then throttled to the



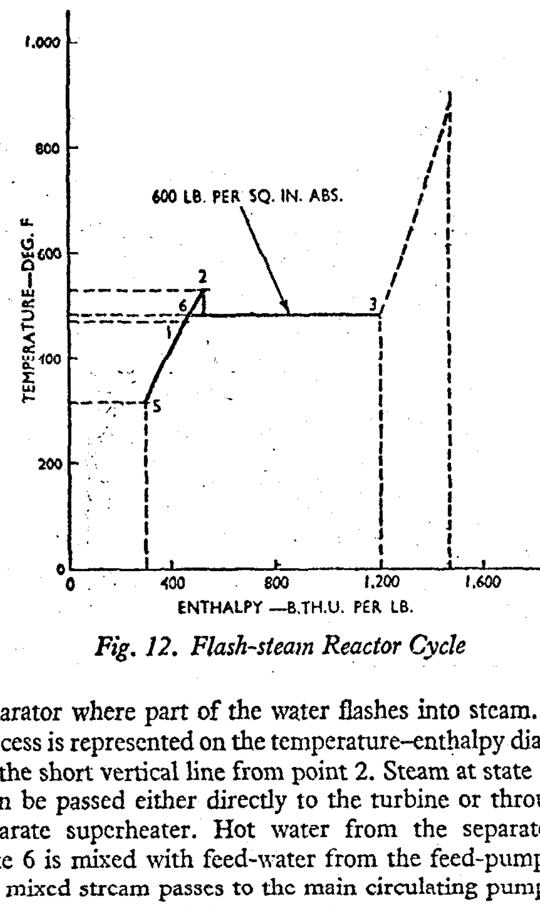


Fig. 12. Flash-steam Reactor Cycle

separator where part of the water flashes into steam. This process is represented on the temperature–enthalpy diagram by the short vertical line from point 2. Steam at state 3 can then be passed either directly to the turbine or through a separate superheater. Hot water from the separator at state 6 is mixed with feed-water from the feed-pump and the mixed stream passes to the main circulating pump and hence to the reactor inlet at state 1.

The full boiling system is shown diagrammatically in Fig. 13. Bulk boiling now takes place inside the reactor core and a steam–water mixture emerges from the core and passes to the boiler drum. The reactor coolant channels thus take the place of the evaporator section of a conventional boiler. Forced circulation may be employed to maintain the necessary velocities and heat-transfer rate from the core. Apart from the drop of pressure occurring in the main steam and water lines, the boiler pressure in the boiler drum is now identical with the pressure inside the reactor vessel.

Thermodynamically the full boiling system is undoubtedly the ideal one, and a fairly high steam pressure can be achieved at the turbine stop-valve without the necessity to maintain a much higher pressure in the reactor itself.

There is, however, an additional problem with this arrangement in providing sufficient space inside the reactor core for steam formation without undue loss of reactivity. It is, perhaps, unfortunate that the light-water reactor requires a closer lattice arrangement than the heavy-water reactor. The geometry of the heavy-water reactor core is more suited to boiling of the coolant than is the geometry of the light-water reactor, but the prospect of a boiling *heavy-water* reactor system raises fresh problems in the use of heavy-water as the working substance in the turbine. It remains to be seen whether it would be possible to reduce gland losses, etc., to a sufficiently low figure to permit the use of such a valuable working substance.

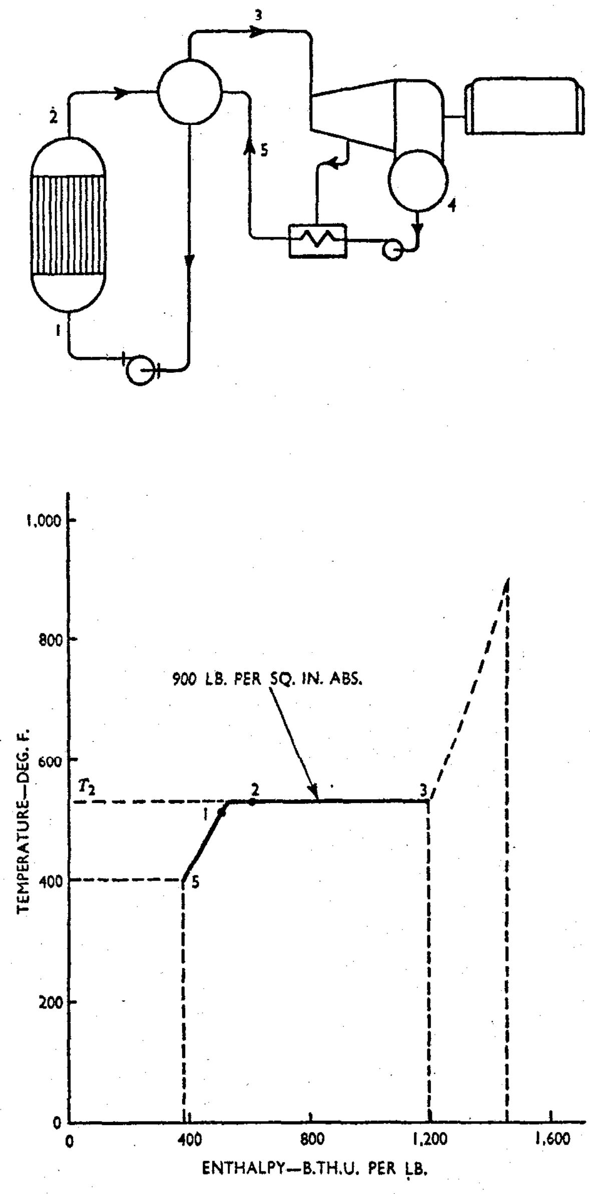


Fig. 13. Boiling-water Reactor Cycle

THE PRESSURE VESSEL AND PLANT ARRANGEMENT FOR A FLASH-STEAM REACTOR

One reason for considering the flash steam or boiling cycle is the desirability of easing the pressure-vessel design problem. In the flash-steam reactor heat is removed from the core without change of phase of the coolant. It can be assumed in the limiting case, however, that the water is heated up to the boiling point corresponding to the reactor pressure. In these circumstances, the permissible temperature level may be fixed by the maximum temperature in the fuel elements rather than by the design pressure for the reactor vessel. To illustrate this point a figure of 900 deg. F. may be taken as a conservative upper limit for the central metal temperature in the fuel elements. With a thermal rating of 20 million B.Th.U. per hr. cu. ft. and with an equivalent diameter of $\frac{1}{2}$ inch for the fuel elements, the radial temperature difference (from Table 1) would be 210 deg. F. giving a maximum surface temperature of 690 deg. F. Taking account of the fact that lower water velocities must be employed when handling water at the boiling point, and allowing a total of 150 deg. F. for the difference between the maximum fuel element surface temperature and the coolant outlet temperature, the maximum permissible value of T_2 would be 540 deg. F., corresponding to a pressure of about 960 lb. per sq. in. In the following discussion a reactor working pressure of 900 lb. per sq. in. will be assumed.

Thermodynamically, the pressure for the flash vessel should be as close as possible to the pressure in the reactor vessel, since throttling involves irreversibility and consequent loss of availability of the heat energy. However, for a given heat output, the circulation rate through the reactor increases rapidly as the flash vessel pressure is raised, and a limit will be set by the permissible velocity and flow area for the coolant. The circulation rates and theoretical cycle efficiencies are given in Table 5 for a range of flash vessel pressures. The corresponding figures for the full boiling cycle are also given for purposes of comparison.

While the pumping head decreases as the steam pressure approaches the reactor pressure, the volume flow of the circulating water increases rapidly. The net effect is a slight increase in pumping power. However, the pump design will be less difficult at the larger rates of flow and lower heads. The flash vessel could take the form of a large steam-drum mounted above the reactor vessel, although two or three drums might be required to provide sufficient surface area for disengagement of steam from the water. Both with the flash-steam and boiling reactors, it would be desirable to provide a larger flow area for the coolant than with the orthodox pressurized water reactor. It may be necessary in these circumstances to increase the diameter of the core and to reduce the length of the channels so far as the nuclear physics of the reactor will permit. This leads to the consideration of a nearly spherical shape for the pressure vessel as an alternative to the orthodox cylindrical form. One possible scheme for a pressure vessel for a reactor of moderate size is shown in Fig. 14.

Table 5. Circulation Rates and Theoretical Cycle Efficiencies

	Flash cycle				Boiling cycle
Reactor pressure, lb. per sq. in.	900	900	900	900	900
Flash-vessel pressure, lb. per sq. in.	500	600	700	800	900
Circulation ratio	9·8	13·3	20·2	40·7	—
Feed temperature, deg. F.	330	345	360	375	375
Heat from reactor, B.Th.U. per lb. steam	887	871	851	834	847
Turbine work, B.Th.U. per lb. steam	283	282·5	282	282	287
Pump work, B.Th.U. per lb. steam	16·7	17·3	17·6	17·9	—
Net work, B.Th.U. per lb. steam	266	265·2	264·5	264·1	287
Theoretical cycle efficiency, per cent	30·0	30·5	31·1	31·7	33·8
Steam flow for 500-MW. reactor heat, lb. per hr.	$1{\cdot}92 \times 10^6$	$1{\cdot}96 \times 10^6$	$2{\cdot}0 \times 10^6$	$2{\cdot}04 \times 10^6$	$2{\cdot}01 \times 10^6$
Water circulation rate for 500-MW. reactor, lb. per sec.	5,240	7,240	11,250	23,100	—

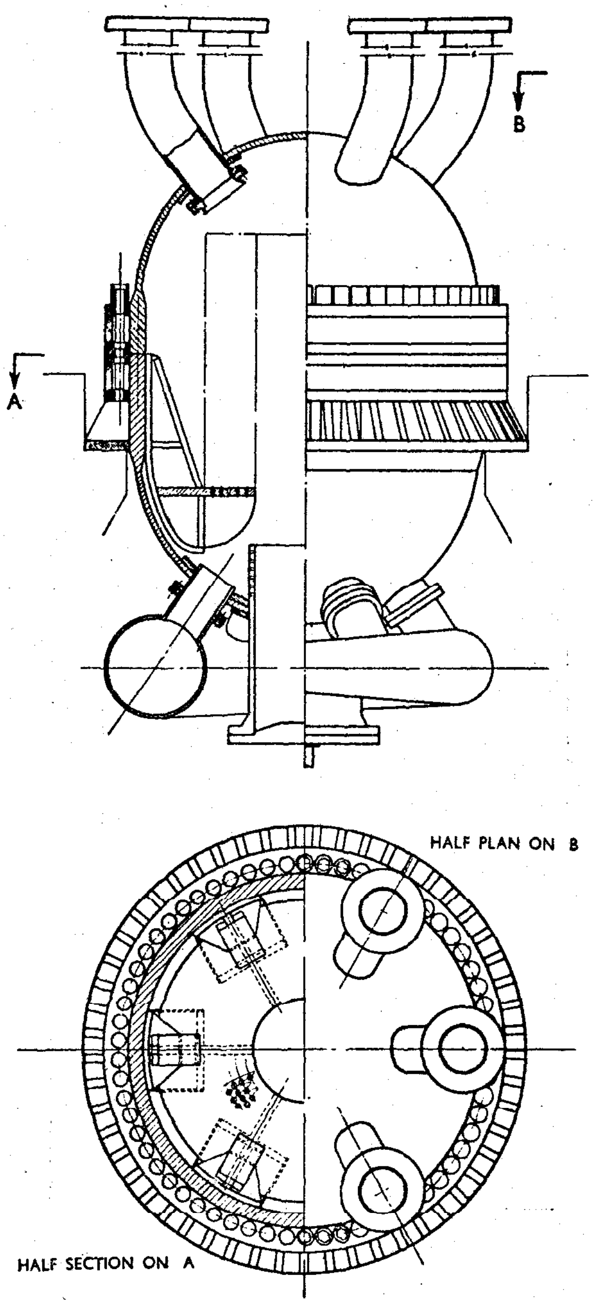


Fig. 14. Pressure Vessel for Reactor

THE PRESSURE VESSEL AND PLANT ARRANGEMENT FOR A BOILING-WATER REACTOR

With full boiling of the coolant occurring inside the core, heat will be extracted from the reactor almost entirely by the agency of the latent heat of the steam. A natural circulation system would be possible with this arrangement, but to maintain sufficiently high flow velocities forced circulation would probably be desirable. With water returned to the reactor at the saturation temperature, however, trouble would be experienced with cavitation on the suction side of the pump and a certain degree of subcooling would in fact be necessary.

A definite limit will be placed by the requirements of the nuclear physics on the permissible proportion of steam formed inside the coolant channels. This will necessitate the use of relatively larger channels, a high rate of circulation of the steam–water mixture, and some degree of subcooling of the water returning to the reactor. To meet these requirements it may in fact be necessary to design a reactor which is really a compromise between the true boiling reactor and the flash-steam reactor, and thus to arrange matters so that some of the boiling takes place outside the core. The plant arrangement for the boiling reactor would therefore be essentially similar to that for the flash-steam reactor, but the circulating pumps would be smaller. The steam drum or drums would again be mounted above the reactor vessel, but they could be reduced in size if sufficient space is provided above the core inside the reactor vessel to enable some separation of steam and water to take place before the steam passes out of the reactor vessel to the drums.

COMPARISON OF STEAM CYCLES FOR THE BOILING-WATER REACTOR

Two different arrangements are shown in Figs. 15 and 16.

(1) Saturated Steam with Water Separation

In this arrangement saturated steam is supplied to the turbine from the steam drum at 900 lb. per sq. in. Water separation is carried out at two points in the expansion to prevent the degree of moisture in the turbine from becoming excessive. The diagram in Fig. 16 implies that external separators are used, that is, the entire flow of steam and moisture is discharged from the turbine at the separation point to an external steam–water separator. As an alternative to this method, however, a multi-stage internal water-extraction system might be employed to achieve the same result. Five stages of feed heating are employed, and the feed water is raised to 372 deg. F. before being delivered to the boiler drum.

(2) External Superheating from a Combustion Source

This arrangement is shown in Fig. 16. The arrangement here is generally similar to that of Fig. 10 for the orthodox pressurized water reactor. Steam at a pressure of 900 lb. per sq. in. is supplied to the oil-fired or coal-fired superheater, and it is assumed that superheated steam at a pressure of 850 lb. per sq. in. and 900 deg. F. is available at the turbine stop-valve.

The two cycles in Figs. 15 and 16 are compared in Table 6.

These figures refer to a reactor with full boiling taking place inside the core. Some reduction of output and efficiency will be incurred if either the flash-steam reactor or

Table 6. Comparison of Two Steam Cycles

	Water extraction	Separate superheater
Boiler pressure, lb. per sq. in. abs. .	900	900
Turbine stop-valve pressure, lb. per sq. in. abs.	900	850
Turbine stop-valve temperature, deg. F.	532	900
Back pressure, lb. per sq. in. abs. .	0·75	0·75
Final feed temperature, deg. F. .	372	372
Overall thermal efficiency, per cent	31·5	35·4
Electrical output, MW. . .	157·5	230
Steam to turbine, lb. per hr. . .	2,000,000	2,000,000
Dry steam to condenser, lb. per hr.	1,072,000	1,320,000
Exhaust area for assumed leaving loss, sq. ft.	131·5	161·5

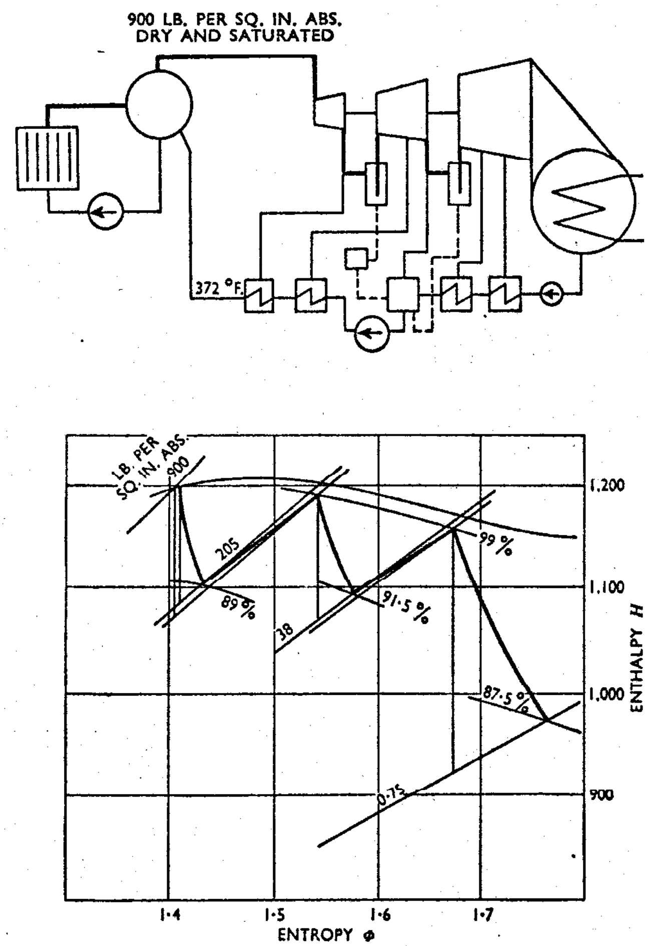


Fig. 15. Arrangement of Steam Cycle for Boiling-water Reactor—Saturated Steam with Water Separation

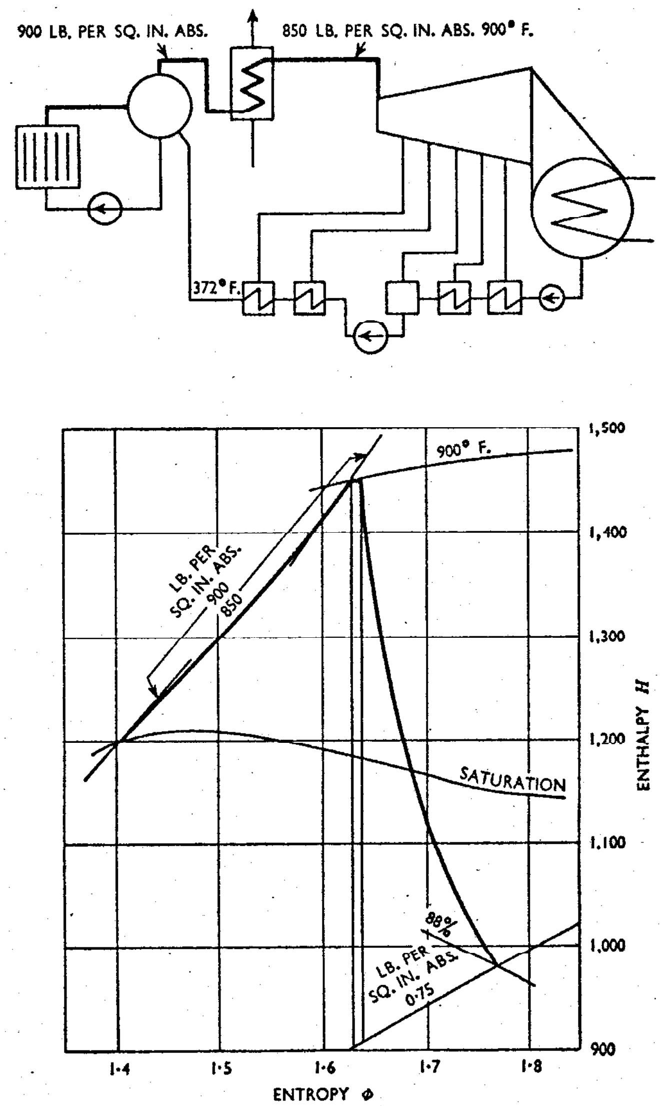


Fig. 16. Steam Cycle for Boiling-water Reactor—External Superheating from a Combustion Source

some compromise between the boiling-water and the flash-steam reactor is employed. The maximum extent of this reduction will be apparent, however, from the figures for the pumping power shown in Table 5. If the separately superheated cycle is compared with the unsuperheated extraction cycle it will again be evident that the combustion heat is used at very high efficiency.

CONCLUSIONS

Pressurized-water reactors are sometimes compared unfavourably, with either gas-cooled or sodium-cooled reactors, on the basis of the available coolant *outlet* temperature. This, however, is not a valid basis for comparison. With water-cooled reactors temperatures throughout the core are more nearly uniform than in any other known type of reactor. The boiler pressure is determined by the coolant *return* temperature and not by the outlet temperature. In terms of steam conditions and cycle efficiencies, the pressurized-water reactor appears in a much more favourable light. The estimated cost of power generation is discussed in Appendix I.

The principal merit of the water-moderated reactor is the prospect that it offers of meeting a wide range of power-station requirements with a single basic type of design. Once having solved the characteristic engineering problems of a pressurized-water system, a number of variations in design become possible without incurring new problems on each occasion. If heavy-water is available, natural uranium may be used as the fuel. If enriched fuel is available, ordinary water may be used as the moderator and, by increasing the degree of enrichment successively, smaller reactor cores may be employed. A very wide range of power output can thus be covered from the same basic design. Finally, there is the prospect of development from the orthodox pressurized-water reactor to some form of boiling-water reactor.

The real criterion as to whether one type of nuclear reactor is better than another will be reliability and convenience of operation rather than the efficiency or even the cost of the plant. A decision on this point can be made only in the light of actual operating experience. The best type of liquid-cooled reactor will probably be the reactor having the simplest type of fuel element, and preferably a fuel element which will deteriorate only slowly in the event of a minor imperfection or failure in the sheath.

ACKNOWLEDGEMENTS

The authors wish to thank the partners of Messrs. Kennedy and Donkin for permission to publish this paper. They also would like to acknowledge the help they have received from discussions with many of their friends both in industry and in the United Kingdom Atomic Energy Authority. They particularly wish to acknowledge the assistance of Mr. H. S. Napier and Mr. R. Vaux in preparing the paper. Special acknowledgement is also due to Babcock and Wilcox, Ltd., for Figs. 6 and 8, Hayward Tyler and Co., Ltd., for Fig. 9, and Markham and Co., Ltd., for Fig. 14.

APPENDIX I

COST OF POWER GENERATION FROM A PRESSURIZED WATER-MODERATED REACTOR

It is hazardous to attempt a precise estimate of the cost of power generation from any type of nuclear reactor until a prototype plant has actually been built and operated. The permissible level of irradiation and the reliability of the fuel elements cannot be predicted in advance with any degree of certainty. The following analysis, however, is based on similar assumptions to those made by Jukes (1955), and the figures should serve for purposes of comparison with other estimates.

Two cases are considered, each with a reactor heat rating of 500 MW.: (1) orthodox pressurized-water reactor, electrical output 135 MW. at a capital cost of £100 per kW. installed; and (2) boiling-water or flash-steam reactor, electrical output 150 MW. at a capital cost of £90 per kW. installed. In each case the total capital cost of the plant is therefore £13·5 million. The annual capital charges are assessed on the assumption that two-thirds of the capital cost represents conventional equipment having a life of thirty years, the remaining one-third having a life of fifteen years. Following Jukes an interest rate of 4 per cent is assumed.

For the fuel elements, a degree of enrichment is assumed equivalent to a U235 content 50 per cent greater than that of natural uranium. Taking a charge of 45 tons of fuel, the initial cost of the fuel elements is assessed at £20,000 per ton (for natural uranium) plus the cost of enrichment at £12,000 per kg. of plutonium. The latter figure is the price of plutonium quoted by Jukes (1955) which justifies the early graphite reactors.

Capital costs may therefore be summarized as follows:

	Capital cost, £ million	Annual cost, £
Reactor items	4·5	405,000
Other plant	9·0	520,000
Total construction	13·5	925,000
Cost of initial fuel charge	2·83	113,000
Total cost	16·3	1,038,000

It is assumed that the reactor is operating as part of a large combined system comprising other reactors together with the appropriate fuel element fabrication facilities and chemical separation plant. It is also assumed that by-product plutonium is recycled within the system. This procedure has the advantage of making the economics of the reactor independent of the book-keeping price of plutonium. A make-up of fissile material is obtained by feeding a supply of natural uranium either to a diffusion plant or to natural uranium reactors. Rejection of depleted uranium from the system is assumed to be possible at a concentration 0·3 times that of natural uranium.

For the 500-MW. reactor under consideration operating at 80 per cent load factor, taking an irradiation of 3,000 MWD. per ton of fuel, an enrichment 1·5 times that of natural uranium, and a conversion factor of 0·7 for by-product plutonium, the fuel quantities may be expressed as:

Annual throughput of fuel	48·6 tons
Annual production of recycled plutonium	107·5 kg.
Fresh supply of natural uranium to the system	9·02 tons

The cost of the fresh supply of natural uranium fed into the system is taken at £20,000 per ton. For the recycled uranium and plutonium a combined processing and fabricating charge of £10,000 per ton of fuel elements has been assumed. On this basis the annual fuel costs would be:

	£
Fresh uranium at £20,000 per ton	180,000
Fabricating and processing charge for recycled fuel at £10,000 per ton	486,000
Total	666,000

The total cost of power generation can then be expressed as follows, in *pence per kW.-hr. sent out*:

Load factor, per cent .	50	60	70	80
(*a*) *Orthodox reactor*				
Capital charges . .	0·420	0·350	0·301	0·263
Fuel cost . . .	0·169	0·169	0·169	0·169
Operating cost . .	0·100	0·100	0·100	0·100
Total . . .	0·689	0·619	0·570	0·532
(*b*) *Boiling reactor*				
Capital charges . .	0·378	0·314	0·269	0·236
Fuel cost . . .	0·153	0·153	0·153	0·153
Operating cost . .	0·100	0·100	0·100	0·100
Total . . .	0·631	0·567	0·522	0·489

The above figures refer to the saturated steam case using reactor heat alone.

With an external coal- or oil-fired superheater at an incremental cost of £40 per additional kW. of installed capacity, and taking an increment of 74 MW. at 45 per cent efficiency of utilization of combustion fuel from Table 4, the cost per kW.-hr. of the extra power generated would be as follows with coal at £4 per ton:

Load factor per cent .	50	60	70	80
Capital charges . .	0·127	0·106	0·091	0·079
Fuel cost . . .	0·309	0·309	0·309	0·309
Other operating costs .	0·075	0·075	0·075	0·075
	0·511	0·490	0·475	0·463

External superheating in this case would be economic for an orthodox water reactor provided fuel could be obtained at £4 per ton for coal or an equivalent figure for oil.

APPENDIX II

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Discussion

Mr. R. V. Moore, G.C., B.Sc. (*Member*), M.I.E.E., said that considerable attention was being directed to the development of a pressurized-water nuclear-power plant both in the United Kingdom and Canada and also in the United States of America. In the United States a pressurized light-water reactor was actually under construction, and had possibly been developed from the propulsion unit developed for the submarine *Nautilus*. It would go into commission about a year after the British Calder Hall plant. In the United States also a small-scale experimental boiling reactor had been built.

In heterogeneous reactors, in which the fuel was in some solid form and in which the heat produced in the reactor was conveyed out of the core to the steam-raising plant by a forced-convection process, the heat transference might be accomplished currently in one of three ways: (1) by pressurized gas; (2) by liquid metals; and (3) by pressurized water. None of those methods was ideal. Perhaps the two major shortcomings of the pressurized-water method were the following: first, although water was considered normally to be a non-corrosive agent, at high temperatures it was the reverse; in particular, its attack on uranium metal was violent; secondly, at atmospheric pressure it boiled at 100 deg. C., and the effect of that on the design of the plant was shown by the input to the turbines being low-pressure saturated steam. Water, while being generally an excellent low-temperature coolant, had many shortcomings for high-temperature applications.

The authors had set themselves a very difficult task in surveying a potentially important type of nuclear-power plant in one paper, and, in writing it from the aspect of the power-plant engineer, perhaps the problem of core design, which was where 80–90 per cent of the research and design problems were concentrated, had been somewhat under-emphasized.

One effect of that approach was, he considered, that the heat output of the reactor postulated in the paper had been somewhat inflated. That could be substantiated by the fact that the American reactor which was under construction, and which was a pressure vessel 9 feet in diameter, pressurized to 2,000 lb. per sq. in.—a very similar design to the one discussed in the paper—had a reactor output of 290 and 85 MW. electric power as target figures, which were considerably less than the 500 and 130 MW. of electric power referred to in the paper. That would, of course, have an adverse effect on the economic assessment of the plant. At the current stage of development, however, he thought that it was fruitless to compare alternative plants on an estimated economic basis, since the cost per unit with each alternative lay somewhere between $\frac{1}{2}d.$ and $1d.$ per unit, and was obviously related to some basic law arising from the fact that comparable quantities of heat at around the same temperature were being raised and utilized. It was unlikely, therefore, that that approach would clarify the selection of types of reactor for further development. Comparisons would have to be on a more subtle and detailed basis than that.

He had mentioned that 80–90 per cent of the research and design effort was centred on the reactor core. It was possible to segregate four major groups of problems: (1) the fuel element design and performance; (2) the heat rate of the reactor of specific size; (3) the neutron economy of the core; and (4) the accommodation of operational requirements, and particularly (*a*) arrangements for loading and unloading the fuel, (*b*) the control of the reactor in the steady state and in the transient states associated both with normal load changing and fault conditions, and (*c*) the detection, localization and methods of dealing with fuel elements which had become faulty.

Those factors were completely interlocking, and it was essential in reactor design to strike the right balance, to find the right simultaneous solution for all those problems. For example, in the paper it was quite rightly suggested that uranium rods of $\frac{1}{2}$ inch diameter would be advantageous for obtaining a high fuel rating. If, however, the design was being considered in regard to neutron economy, it would be beneficial to use rods of 1 inch diameter with a consequent reduction in fuel rating. That might be one of the reasons why the heat output postulated in the paper seemed to be on the high side. He would interject the thought that when designing for use of enriched fuel it was a misconception to think that neutron economy became less important. It might be less critical, but the effects of poor neutron economy would always appear in the final answer and must be offset by another factor, such as reduction in the capital cost of the plant.

The highly corrosive properties of high-temperature water on uranium constituted a major embarrassment to the successful development of a satisfactory fuel element. Alloys of uranium which were more resistant to attack had been found, but they were expensive in neutrons and seemed to detract from the ability of the fuel to withstand radiation damage. It was interesting to note that, in the pressurized-water reactor under construction at Shippingport, the

Americans were using uranium oxide as the fuel, the core having a circumferential zone of zirconium fuel elements in which almost pure fissile material was uniformly dispersed. That fuel element behaved basically as zirconium. The uranium oxide had a much lower density than the metal and was difficult to process after irradiation, while the zirconium fuel elements were very expensive, so that a substantial price had to be paid for using those types of fuel element.

Capital costs of plant per kilowatt sent out were always reduced by increasing the rating of the fuel, with the subsequent reduction in reactor size for a given output. Some 20 per cent of the capital cost of the plant could be affected in this way. If that increase in rating, however, could be achieved by an increase in the maximum temperature of the system, the reduction in capital cost extended to the steam plant as well as to the reactor, and a much larger proportion of the capital cost would be favourably affected. There seemed little prospect of achieving that in pressurized-water plants. It was perhaps ironical that zirconium, a metal with a low capture cross-section and a melting point of 1,900 deg. C., must be introduced into pressurized-water plants to resist the corrosion effects, and yet there appeared to be little chance of using its high melting point to raise the maximum operating temperature of the system.

The neutron economy of a practical light-water system did not permit the use of natural uranium as fuel; to achieve that it would be necessary to use heavy water, or D_2O. The implications of that on a full scale might be that early in 1970—by which time virtually all the new generating capacity in the United Kingdom might be met by the construction of nuclear power stations—the annual rate of installation of new capacity might be increased to 3,000 MW. The capital cost of building a large D_2O production plant, given present techniques, he believed to be about £70 per lb. per year. The adoption of a D_2O reactor as the standard commercial reactor would therefore imply a large production programme, since about 1 ton of D_2O was required for each megawatt of electrical capacity installed. Based on the quoted assumptions it would mean that by 1965–70 a D_2O plant of capacity 2,000–3,000 tons per year would be needed, and the capital investment which that would involve would be some £300–450 million. However sceptical might be the feeling about forward planning, those figures were so large that their implications required very careful reflection.

Time would not permit him to comment on the fourth factor affecting core design, but he would like to emphasize the importance of a satisfactory method of dealing with fuel elements which had become faulty. A solution of that problem was proving rather elusive with water reactors. It was particularly important in boiling cycles where the reactor coolant was the working fluid.

With the boiling cycles the frontiers of present knowledge on pressurized-water plants were reached. Although many of the problems in core design, some of which he had already mentioned, were intensified, such as the corrosion problems relating to the fuel elements, the problem of steam-water separation, the wide range of reactivity changes associated with changes in moderator density, and a worsening of the neutron economy, that cycle was of great interest because it reduced the scale of the reactor pressure-vessel problem. It was necessary, however, to qualify that statement, and that might be done by taking an example from the paper. It appeared that at the present time the rating of light-water boiling reactors might be limited by the space restrictions in the core arising from the nuclear physical requirements. A rating of 40 kW. per litre might be possible, in which case the 500-MW. reactor cited by the authors would require a core diameter of $13\frac{3}{4}$ feet, which, at the recommended pressure of 900 lb. per sq. in., meant a pressure-vessel thickness of $6\frac{1}{2}$ inches. That could not be regarded as an easing of the pressure vessel design problem. Analysis seemed to show that for reactor heat outputs above about 250 MW. it might be necessary to employ heavy water, not primarily to improve the neutron economy but to obtain wider spacing of the fuel elements to give reasonably high fuel ratings.

Mr. D. R. Griffiths, B.E. (Harwell, Berks.), said that there were four matters he wished to discuss.

First, although in their introduction the authors had stated that core design and the problems of the fuel element were beyond the scope of the paper, they had dealt with the size and rating of fuel elements, temperatures, and so on, and, therefore, he felt justified in saying a few words about the core design problem.

Any reactor which would require fuel of appreciable enrichment to operate it (and that applied to all light-water-moderated reactors and sodium-graphite reactors for that matter) would, it seemed, have to be operated on a plutonium recycling basis in order to reduce fuel costs to an acceptably low figure. On such a cycle many factors entered into the fuel cost and all those factors should be considered in the reactor and power plant design in order to obtain a satisfactory power producer. Thermodynamic efficiency which had been so adequately dealt with in the paper was but one factor; another equally important one was nuclear efficiency, as that governed the amount of feed material which would have to be supplied, and it was in attempting to obtain high nuclear efficiency—in other words a high conversion factor consistent with low fuel enrichment—that most of the difficult problems associated with core design arose. However, so long as nuclear fuel and fissile material commanded their present high prices such considerations were important and could not be ignored.

To illustrate that, he would point out that the Russian Atomic Power Station, although undoubtedly a sound engineering job, required uranium of sevenfold enrichment and had a conversion factor of only 0·3, and therefore could not generate power at a reasonable cost and could not be considered a satisfactory power producer.

Likewise, the figures given in the paper for a 500-MW reactor had been based purely on engineering considerations, the nuclear aspect having been completely ignored.

The figures for heat fluxes, temperatures and stresses given in Table I had been based on assumptions of constant radial heat flux and $\frac{1}{32}$-inch stainless steel sheathing, providing structural and mechanical strength. While a reactor designed in that way had a lot to commend it from the engineering design and operating standpoints, it was, unfortunately, not a good one if fuel costs were a consideration.

The economics given in the Appendix did not relate to such a core. With rods of $\frac{1}{2}$ inch diameter, sheathed in stainless steel $\frac{1}{32}$ inch thick, plus additional structural material in the form of coolant tubes, etc., the calculated enrichment for such a core, based on a certain amount of experimental work, was at least $2C_0$ (i.e. twice the U235 concentration as in natural uranium) and the conversion factor on a plutonium recycling basis with non-alloyed fuel elements was less than 0·65. That made no allowance for flux flattening, but with more or less ideal flattening, as suggested, and the use of alloying materials to make the fuel elements resistant to water corrosion in the event of a can failure, the enrichment factor would be much higher and the conversion factor appreciably lower, possibly 2·5 and 0·6 respectively.

While those figures were appreciably better than those quoted for the Russian Atomic Power Station they would still not be acceptable at Harwell. In fact it was very doubtful whether such a system could operate on a plutonium recycling basis with a feed of natural uranium only.

To achieve a reasonable conversion factor with low fuel enrichment much larger fuel elements had to be used than were desirable from heat transfer considerations. The core had also to be designed for low parasitic absorption and low neutron leakage and, with fuel rods of the comparatively small diameter necessary for high specific ratings, stainless-steel sheathing of the thickness suggested and complete flattening of the radial flux could not be seriously considered. Some degree of flattening might be justifiable, but if no flattening whatsoever were assumed the heat ratings, fluxes, and stresses given in Table 1 would be increased by a factor of 2, and in practice would certainly be higher than those given.

Parasitic absorption could be reduced by the use of such materials as Zircaloy 2 instead of stainless steel, but that would introduce additional problems, particularly in the fabrication and support of the fuel elements, because of the large difference in the coefficient of expansion between that alloy and uranium. Neutron leakage could be reduced by 'spiking', as in the case of the American pressurized-water reactor, but that would introduce peaks in the flux curve—which was just what the authors had tried to avoid—and thereby considerably complicate the cooling problem.

There was no simple solution to the design of a satisfactory core when nuclear considerations had to be taken into account. Also nuclear and engineering aspects of core design could not be considered independently. Their requirements frequently conflicted and usually some compromise was necessary in order to arrive at a design acceptable from all standpoints. It was problems such as those, together with those associated with reactivity control, burst slug detection gear, charge–discharge mechanisms and so on that had been and were being, considered at Harwell.

In the estimates for boiling reactors the authors had assumed that the capital cost per kW. and fuel cost per kW.-hr. varied inversely as the efficiency. From the figures presented it would immediately be concluded that boiling reactors would produce power at a lower cost than non-boiling ones. That was erroneous. It was generally agreed that the heat output obtainable from a given size of core under boiling conditions was very much less than under non-boiling conditions, because of such considerations as fuel element burn-out, moderator density variations, hydraulic stability, etc., so that for a given output an appreciably larger core would be necessary in the boiling case. Add to that the extra cost associated with reactivity control, corrosion control, steam and water separation, the use of radioactive steam and feed, and a boiling reactor could reasonably be expected to cost more per kW. than a non-boiling one, despite the fact that heat exchangers were no longer necessary. The higher thermodynamic efficiency achievable with a boiler would be partly, if not wholly, offset by the decreased nuclear efficiency. Furthermore, in the event of fuel element failures, some fission products, particularly the gaseous ones (which usually decayed to solids), would certainly be carried over with the steam and deposited throughout the plant and, because of that, plant maintenance would be higher and availability lower. The answers to all those problems were not yet known, but it should not be regarded as a foregone conclusion that a boiling reactor was a better proposition than a non-boiling one. The gains to be obtained by boiling, if any, were unlikely to be very great.

Regarding the various steam cycles, due consideration had been given to most of the alternative possibilities at Harwell, and it was considered that if a pressurized-water reactor were proceeded with in Britain it should be coupled to a separate oil-fired superheater, not necessarily because of the higher efficiency obtainable (although that was a consideration), but because it would enable more or less standard turbine plant to be used.

In the comparison of superheated and saturated cycles, it might first be pointed out that the overall efficiencies given in Table 4 were actual generating efficiencies, no allowance having been made for auxiliary power requirements. On a 'sent-out' basis those efficiencies would fall to about 25–30 per cent respectively. However, there was no denying the fact that a considerable gain in efficiency was possible by the adoption of a separately fired superheater, but whether or not it paid to superheat depended, amongst other things, on the relative cost of nuclear and fossil fuels. That problem had been examined at Harwell on the assumption of high load-factor operation and, briefly, the conclusions reached had been that on existing United Kingdom plant and fuel costs it would pay a large generating system such as the Central Electricity Authority (C.E.A.), having virtually no limit to the unit size of plant that could be connected to its networks, to build as large a reactor as was economically possible and to increase the output and cycle efficiency by the addition of a separate superheater. The position could change, however, in view of rising coal prices and falling nuclear fuel costs. For smaller systems, where the maximum permissible output from a single unit could be obtained from a reactor turbine set operating on saturated steam (say, about 100 MW.), it would not pay to use a smaller reactor combined with a superheater to give the same total maximum net output. Turbine manufacturers, with an eye on the export market, might profitably devote some effort to the development of saturated steam turbines, if pressurized-water reactors were adopted.

The auxiliary power for all highly-rated water reactors was fairly high, and the important figure was the overall efficiency on a sent-out basis. However, it was realized that without detailed designs on which to base estimates of pumping losses, etc., any figures given would be only very approximate at the best. In that connexion, it might be pointed out that the estimated pumping power given for a pressurized-water reactor was, in their opinion, very low and should have been based on a total drop of 100 lb. per sq. in. rather than 100-foot head. That would make the coolant pumping power about 7½ MW., which on a megawatt heat basis was about the same as that of the American pressurized-water reactor and their own estimates for a similar type of reactor.

He would add that, although a large number of problems was involved in the design of water moderated reactors, all of them had been or could be overcome and at Harwell they were confident that with suitable design and development, especially in the fuel element processing field, the cost of power from such reactors could be reduced to a figure approaching that given in the paper. The decision as to whether one reactor was better than another could be made only in the light of actual operating experience, as had been stated by the authors.

Dr. A. B. McIntosh, B.Sc., F.R.I.C. (Risley), said that the authors had dealt with the feasibility of the pressurized-water reactor from a steam-power standpoint, and one of their objects had been to make a case for the research and development work needed to solve the metallurgical problems. Unfortunately they had not stated what those problems were, and therefore their magnitude and difficulty might escape notice. Some of the problems might be considered in relation to the essential parts of the reactor. First, there was the pressure vessel of mild steel. The authors had written in terms of 7½-inch wall thickness fabricated to 9 or 10 feet in diameter in welded construction, and approximately 40 feet in height. It should be noted, however, that no steelmaker would guarantee to make homogeneous plate to such thickness. Because of the thickness, the size of the plate would have to be reduced, resulting in an increase in welding difficulties. The welding of such thicknesses and the inspection of the welding to ensure a standard compatible with the hazards involved would present the greatest difficulty. If a stainless steel were chosen there was as yet no technique available in Britain capable of ensuring sound welds in such thicknesses of stainless-steel plate or of ensuring proper standards of fault detection.

The selection of the material from which the shell would be made must be viewed against the conditions which could allow of the evolution of atomic hydrogen. Under such conditions that hydrogen would diffuse into the pressure-shell material. If that was mild steel there would be reaction with carbides, possibly forming methane, and in the worst cases methane blisters would result and intergranular separation might take place. That could be safeguarded against by the use of an alloy steel such as chromium steel, but it would introduce almost insuperable welding problems because of the thickness of the plate. The use of a clad steel had been mooted, but clad steels had their own difficulties. There would still be diffusion of atomic hydrogen, with the possibility of blister formation at any discontinuity between the cladding and the plate. If a cladding of stainless steel, or a plate of stainless steel were considered, another very serious problem would arise which had already been experienced in the Culcheth laboratories of the Industrial Group of the Atomic Energy Authority, that was the stress corrosion cracking of stainless steel by demineralized and pressurized water. That was particularly likely to occur with the probability of alternate wet and dry conditions; and the possibility of stress-corrosion cracking might be more serious than had yet been experienced.

The difficulties of selecting materials for fuel elements entirely depended on the extremely corrosive nature of the pressurized pure water. No satisfactory canning material was yet available. At various times stainless steel, zirconium alloys, and aluminium alloys had been considered. With stainless steel there was the problem of stress corrosion, but no doubt that could be eliminated by suitable design. In the conditions of irradiation there might be substantial and persistent dimensional changes in the fuel. The introduction of stress-corrosion conditions arising from that source would have to be considered. With alternative canning alloys, such as zirconium alloys or aluminium alloys, dangers might arise from pitting corrosion or from metallurgical defects; but whether pitting corrosion or stress corrosion occurred, the effect would be the same: there would be intense corrosion of the uranium fuel with a rapid release of fission products accompanied by pressure increase due to hydrogen evolution.

That brought into that type of reactor a hazard potentiality which seriously affected the assessment of its feasibility. The obvious way to reduce such hazard was by the use of corrosion-resistant alloys. Alloys containing percentages of molybdenum and niobium had been studied. Alternatively, non-metallic oxides of low reactivity could be considered. With either an alloy or an oxide the enrichment must be increased. To date, however, corrosion-resistant fuels had given the result that corrosion might be delayed for an induction period, but at the end of that period corrosion would take place catastrophically. That led to another difficulty: owing to the fact that corrosion was delayed and had been reduced, the difficulty of first detection was considerable. A further consequence of the use of an alloy fuel, or an alloy can, perhaps bonded to a fuel, was that the recovery of plutonium might be a great difficulty in the chemical processing.

All of those troubles could be reduced if the temperature of the pressurized reactor were reduced and the reactor were used as a plutonium producer only. That had obviously no attraction for steam-power engineering and, even though the problems were successfully overcome, there would still remain the principal hazard, which would be the catastrophic effect on the corrosion of the fuel by pressurized water. That might lead to considerations of containment of the reactor, which would seriously diminish its attractiveness.

Professor J. DIAMOND, M.Sc. Wh.Sc. (*Member*), said that the pressurized-water reactor was a very interesting type, the basic design of which might be capable of development for a variety of purposes. It appealed to the smaller countries, such as Norway; where the economic and technical resources were limited there was some advantage in that type of reactor, and it might serve for both marine and land use.

At Geneva, Weinberg had said that the enormous difficulty of choosing the proper path for reactor development was readily seen by estimating the number of conceivable reactor types. He had added that nuclear considerations greatly reduced the number of possibilities, and had thereby touched on one of the characteristics of nuclear power engineering—the breadth of the information, including operating experience—to which appeal had to be made before a reactor could be regarded as a probability for large-scale power generation.

The authors had concluded by saying that reliability and convenience of operation rather than efficiency, or even cost, were the real criteria in the choice of a type of reactor. Everyone would agree that the qualities of reliability and convenience of operation were essential, but he suggested that it might be better to say that, given reliability and convenience of operation, the cost of the power produced was the real criterion of whether one type was better than another.

The figures in Appendix I gave an indication of one direction in which a reduced cost per unit could be sought. The capital charges had been discussed and, for large power outputs at least, those reactor types were of most interest which had lower capital charges without a disproportionate increase in fuel and operating charges. The total power output of the system had an effect on the capital charges, and it seemed likely that there would be not one reactor type for all power ranges but a preferred type for each range of power, and probably each kind of location throughout the world. It would be interesting to see which range of power and which purposes would be served by the pressurized-water type of reactor.

One of the advantages given by the authors for the flash-steam reactor was that it might operate at a lower pressure than the pressurized-water reactor. Indeed, the operation of the former depended on the water temperature from the outlet being near that of the saturation temperature for the pressure. In the examples given in the paper, the pressure in the pressurized-water reactor was high enough for the saturation temperature to be in excess of the maximum fuel temperature, whereas in the flash-steam reactor the reverse was the case. If safe operation could be assured for the flash-steam reactor under those conditions, he would ask the authors whether the same conditions could not be applied to the pressurized-water reactor, thus easing the pressure vessel design problem.

He would be interested to know why the authors regarded the boiling-water reactor as a development from the pressurized-water reactor, since boiling-water reactors were now in the development stage in the United States of America.

Mr. R. H. BURDETT, B.Sc. (Eng.) (*Associate Member*), A.M.I.E.E., said that the Government White Paper outlining the probable course of the nuclear power programme had expressed the hope that by 1965 it would be possible to commission a commercial liquid-cooled reactor as the first plant in Stage 2 of the programme. The indications were that that plant might be a light-water pressurized reactor or a sodium-graphite reactor, and Sir John Cockcroft had said at the Inaugural Session of the Conference that there was no difference in cost between the two within the very wide limits of uncertainty involved in the calculations at the time. A definite probability that one type would have lower generation costs than the other would clearly have a considerable influence on the decision as to where the effort available for development should be deployed. In saying that, he did not dissent from the authors when they placed emphasis on reliability in choosing between one reactor type and another. Unreliability in a base-load power plant was a most costly feature. For example, the loss of a modern 120-MW. unit could cost as much as £1,800 per day as a result of having to make increased use of lower-merit plant. With nuclear plant, which was expected to have lower fuel costs than the base-load coal-fired plant, the losses due to unreliability would be even greater.

It was unfortunate that both reliability and operating costs were so much bound up with the behaviour of the fuel elements under long irradiation periods. As Mr. Grout had pointed out in his lecture to the Inaugural Session, it was not possible to perform accelerated tests on fuel elements, and that confirmed the authors' statement that firm assessments of reliability and precise estimates of power cost were possible only in the light of actual operating experience. The situation posed a considerable difficulty for a country of limited technical resources with an urgent need to build up a commercial nuclear power programme rapidly. Sir John Cockcroft had said that the A.E.A. could probably find the effort for only one Stage 2 prototype. If the chosen type should in the event prove to be disappointing much time would have been lost. The problem of limited technical resources was not confined to the A.E.A., and it was not easy to see a way out of that difficulty. It did, however, emphasize the importance that the C.E.A. and industry would have to attach to developing the improvements possible with the Stage 1 gas-cooled reactors, so that continuous technical and economic advances were possible, whether or not Stage 2 ran into serious teething troubles. With the Calder Hall plant coming into operation the following year, there would not be long to wait before operating experience of the pressurized gas-cooled reactor began to accumulate.

The authors had shown that, thermodynamically, the separately fired superheater was an attractive idea, and had also referred to the increased complexity of the siting problem. That type of scheme might prove to be justified in specific instances, but in general it must be remembered that, in the small and heavily populated British Isles, sites for any kind of power-station were not plentiful. In their paper to the Inaugural Session, Mr. Pask and Mr. Duckworth had shown that the capacity of coal-fired plant might well have to be doubled before 1980. As a result of that, it might be that the extreme shortage of sites would force the C.E.A. to take advantage of such freedom of siting as nuclear power permitted, and place the nuclear plants at places where supplies of fossil fuel would be extremely difficult to arrange.

From Table 4, assuming four such separately fired superheated units on one site, the combustion heat would amount to 652·8 MW. At an overall efficiency of 33 per cent, which was appropriate to modern coal-fired plant, the superheater installation would be equivalent to 215 MW. of conventional plant, so far as coal and ash handling facilities were concerned. That served to show that the latter problems would not be insignificant. From those purely practical considerations, a reactor unit which was competitive without recourse to separately fired superheaters had an advantage.

In regard to boiling reactors, the reduction in the reactor pressure-vessel duty, and the elimination of the heat-exchanger link in the energy conversion chain, were both attractive features, although it had to be remembered that the difficulties of making use of saturated steam would still be present. At the turbine end, it was probably true to say that there were no problems associated with radioactivity in the working fluid for which a possible solution could not be seen. However, work on any radioactive plant was always more costly and time-consuming than work on an equivalent inactive plant, and much would depend upon the magnitude and scope of the measures which would have to be taken. It would be unwise to form any firm conclusions about that type of scheme in advance of a detailed study of the types and levels of activity likely to arise in various circumstances and of the practical consequences on operation, maintenance, and repair. He had already emphasized the cost of set outage, and the possibility of having to perform an extensive decontamination procedure on the turbine, condenser, feed-heating plant, etc., as a result of one burst fuel element was distinctly unattractive. Nevertheless, if the use of water as a coolant proved to occupy more than a transitory stage in the advance of reactor technology, the advantages of the boiling reactor would be a strong incentive to overcome the difficulties.

Mr. J. R. Allard (*Graduate*) said that the authors had come to the conclusion that with both the pressurized-water and boiling-water reactors some type of forced circulation was necessary. That meant that for the safety of the reactor the maintenance of that circulation was essential. He would have liked to see some treatment of the subject, in that in the boiling-water reactor shown the water circulation was obviously affected by variations in steam pressure in a way in which the pressurized-water reactor was not. Since safety was likely to be an important matter in the first water reactors to be built, he would be very interested to have the authors' views on the inherent stability of the pressurized-water reactor as compared with the boiling-water reactor.

Mr. T. R. Warren, M.A., B.Sc. (Eng.), M.I.E.E., said that a study of the paper left the impression that the pressurized-water reactor using light water offered very few advantages over the more orthodox graphite-moderated gas-cooled type. Its principal advantage was that it made no demands on the limited supply of special graphite which would be in great demand when the accelerated programme of nuclear power development got under way. As against that, however, the use of light water as a moderator involved the use of enriched uranium, whereas the graphite moderator enabled natural uranium to be employed. Any large-scale adoption of that type of reactor would require large quantities of plutonium for the enrichment of the initial fuel charge and, if supplies produced by the graphite-moderated reactors were to be thus absorbed, the time when it would be possible to embark on the construction of large-scale breeder-reactor power plants, which alone could secure for the supply industry a plentiful and cheap supply of that vitally important material, would be considerably delayed.

The graphite-moderated gas-cooled reactor had, he considered, been rightly chosen to pave the way to more advanced designs. The plutonium produced could be separated by chemical means from the irradiated fuel and stored for later use in breeder reactors. Fuel replacement would then be effected by means of natural uranium, the cost of which, he understood, was about the same as the value of the plutonium recovered. The reliability of the graphite-moderated reactor had been proved by several years of experience, and any alternative design must show savings in overall cost before it could be seriously considered as a rival.

A disadvantage common to all water reactors was the temperature limitation they imposed. The adoption of pressures as high as 1,500 lb. per sq. in. resulted in steam temperatures no higher than 450 deg. F., with slightly higher values for the boiling-water reactor. Consequently, the efforts being put into metallurgical research to develop materials capable of withstanding higher temperatures would be fruitless as applied to any reactor in which the temperature was limited by thermodynamic considerations.

While the heavy-water reactor was subject to the same temperature limitation, it had the merit of requiring only natural fuel for its operation and was, therefore, better adapted to the production of plutonium. The cost of heavy water was high, and the future of the New Zealand geothermal steam project seemed to be in doubt. However, as the nuclear-power programme developed it would be necessary to look for more off-peak load to keep the plants in full operation during the night. There was relatively little pumped storage potential in England, and a suitable outlet for surplus energy could possibly be found in the large-scale electrolysis of water for heavy-water concentration, and the hydrogen produced in that way used for steam superheating. Sufficient hydrogen could be stored each day to permit the electrolytic plant to be shut down during the peak period and, provided nuclear-energy fuel costs were sufficiently low it was possible that a dual-purpose scheme of that kind might provide supplies of heavy water at a price which would render the heavy-water reactor a more attractive proposition.

In Appendix I the authors had estimated the annual fuel cost in a manner which called for further elucidation. Starting with a fuel charge of 48·6 tons of uranium having an enrichment of 1·5, they had assumed that at the end of one year's operation the fissile content could be restored by recycling the plutonium and replacing 6·47 tons of the irradiated uranium by an equal weight of natural uranium. It seemed rather extraordinary that that should be so, bearing in mind that the conversion factor was only 0·7, and he thought that they might have overlooked the weight of fissile uranium contained in the 6·47 tons of uranium which they had discarded. According to his own calculations that amounted to 0·032 ton, and the cost of providing an equal quantity of plutonium would be £390,000 at £12,000 per kg.

The authors' annual fuel cost of £550,000 then became £940,000 but, by replacing the whole of the fuel charge, recycling sufficient plutonium to give the required degree of enrichment and selling the balance, the annual fuel cost could be reduced to £275,000. It was clearly necessary to reduce the amount of initial enrichment required as well as to reduce the amount of subsequent enrichment. Both the graphite and heavy-water reactors achieved the first object, and to achieve the second it was necessary to increase the plutonium conversion factor. The heavy-water reactor, however, suffered from the same disadvantage as the light-water reactor in that the temperature was limited by thermodynamic considerations, whatever advances were made in the development of a more suitable canning material.

Of the various arrangements described on pp. 289–293, the second one would appear to possess considerable advantages over the others, but it was not clear why the authors had departed from the orthodox method of returning the steam to the boiler, which avoided the long run of high-pressure piping and the separate reheater. The adoption of superheating from a combustion source would appear to result in a surprisingly low efficiency of utilization of heat, namely, 45·2 per cent, but that figure could be improved considerably at the expense of some of the additional kW. output by adopting a lower superheat temperature than 900 deg. F., since that temperature was high in relation to the pressure of 420 lb. per sq. in.

Mr. B. L. Goodlet, O.B.E., M.A. (*Member*), M.I.C.E., M.I.E.E., said that it was important to understand that the types of reactor that could be built depended on the materials available. Given only natural uranium and no heavy water the Calder Hall type of reactor was the only possibility. Another kind of natural uranium reactor could be built with heavy water as moderator, but that meant arranging production capacity for heavy water. All other types of reactor required fuel enriched in fissile material—U235 from a diffusion plant or Pu239 or U233 from convector reactors. Before enriched reactors could be contemplated the supply of enriched fuel must be assured.

The pros and cons of the fuel-fired superheater could be argued at great length. In the feasibility study now in progress he had decided that since there were many development problems involved in the reactor it was undesirable to be faced with many additional ones arising from the use of saturated steam in turbines. He had therefore chosen an existing design of turbine and had matched the reactor to it by means of an appropriate superheater. While the use of a superheater showed excellent fuel efficiency and could well be justified on that ground alone the main reason for its use was to eliminate development problems in the turbine. When the water reactors were a proven success it might be worth while developing special turbines to suit them.

The authors had suggested that there would be no radioactivity in the steam from a boiling water reactor during normal operations. Calculations made at Harwell suggested that radiation levels around the turbine would be of the order of 20 tolerances owing to the decay of the N^{16} produced by the O^{16} (n, p) N^{16} reaction. This would not be prohibitive but any settlement of fission products in the turbine would be serious.

A problem with all pressurized-water reactors was the periodic inspection of the pressure vessel (required by the Boiler Acts), which would acquire induced radioactivity during operations. That problem looked more tractable than it had several months earlier.

Mr. Geoffrey F. Kennedy, M.A. (*Member*), M.I.E.E., said that one of the main questions to be settled was whether or not there would be any considerable advantages from the use of a pressurized-water reactor, bearing in mind the overall costs of producing electricity. If there would be considerable advantages as compared with other types, then designs and operation of the plant must be expedited, so that preparation of an overseas export programme in competition with other countries could be arranged. It seemed that fuel for a pressurized-water reactor, although not at present available, should be available within ten years, or at any rate by the time some of the gas-cooled reactors were in operation.

It stood to reason, therefore, that if the export market was to be taken seriously—and he believed that a considerable potential market existed—resources should be devoted to proving the pressurized-water reactor on the prototype scale in Britain, because it was essential to obtain actual operating experience.

In addition to the examples, quoted by Mr. Moore and other speakers, of developments in the United States, the Consolidated Edison Company of New York had recently announced their intention to proceed with a pressurized-water reactor having an electrical output of 236,000 kW. at an overall cost of $55,000,000. That represented a cost of £83½ per kW., which was substantially lower than any figure which he had yet heard quoted for a nuclear reactor generating station.

Great Britain held the lead in the nuclear field, but he believed that lead would be lost unless operating experience of new types of reactor was to be gained at an early date and the graphite-moderated gas-cooled reactor stations already authorised were put into operation in the shortest possible time. Any programme which did not provide for those considerations should be rejected.

Mr. G. A. PLUMMER (*Member*) said that there was one main point to which he wished to draw attention. During his short association with atomic power he had been a definite protagonist for the use of superheated steam. Following considerable experience of the use of waste heat and other low-temperature sources of heat, he had always found that the expenditure of some additional fuel for superheating must result in a considerable improvement in the overall cycle, because all the heat was usefully employed in the turbine and it involved no additional loss of heat to the condenser. The result, as the authors had shown, was in that particular case an overall efficiency of some 45 per cent, which was much higher than in a conventional steam station.

The authors appeared to have made a small mistake which detracted from the advantages, which would otherwise have been outstanding, in that example of using superheated steam. On the last page they had said, 'With an external coal- or oil-fired superheater at an incremental cost of £70, per additional kW. of installed capacity'. A complete power station using coal or oil fuel could be built for far less than £70 per kW. The amount of additional plant required to deal with the superheated steam, as compared with the saturated steam, was merely an addition to the high-pressure stages of the machine, a slight addition to the alternator—no additional condenser with all its associated difficulties—plus a comparatively simple separately fired superheater, which was not nearly so complicated as a complete boiler plant. He suggested, therefore, that the capital cost for the additional capacity would probably be less than half the figure which they had given. If a figure of £35 per kW. of additional installed plant, instead of the £70, were taken, the following would be the figures for the ultimate running cost: with a load factor of 50 per cent, the figure would be 0·495*d*. per kW., against the authors' figure of 0·606; with a 60 per cent load factor, the figure would be 0·477, instead of 0·569; with a 70 per cent load factor, it would be 0·463, against 0·543; and for 80 per cent it would be 0·453, instead of 0·523. That showed a very considerable gain, and an outstanding advantage with the additional superheater.

Mr. W. R. WOOTTON (London) said that he had been interested to hear Mr. Kennedy draw attention to the Consolidated Edison pressurized-water reactor in the United States, information about which had only recently been revealed. He had pointed out that that reactor had a capacity of 236 MW. and that the capital cost was low, working out at only about £83 per kW. He himself would like to link an observation on that capital cost with what Mr. Plummer had said, the advantages of separately superheating and accepting the mixing of nuclear and fossil fuels. The Consolidated Edison reactor of 236 MW. had an appreciable addition of conventional fuel heating in the cycle, and that had played a large part in making possible the low capital cost of £83 per kW.

A point of interest was that in the nuclear field a fuel was being dealt with which was capable of a temperature potential of millions of degrees, yet in the discussion the question had been argued as to whether the use of pressurized water suffered a temperature limitation, and whether the gas-cooled reactor or liquid-metal-cooled reactor was not capable of higher temperature utilization. For the time being at least the contemplation of separate superheating should put most current cycles on a similar datum.

AUTHORS' REPLY

Dr. J. M. KAY and Mr. F. J. HUTCHINSON wrote, in reply to the discussion, that they agreed with Mr. Moore that the engineering problems of the pressurized-water reactor were not easy. They had themselves particularly emphasized the problem of corrosion. Mr. Moore's remarks might, however, give the impression that the problems of the pressurized-water reactor were greater than those of other types of nuclear reactor. They considered that was emphatically not so; every type of nuclear reactor had its own peculiar problems and those of the pressurized-water reactor were at least well defined and fairly well understood which was more than could be said for many other types of nuclear reactor. The fact that pressurized-water and boiling-water reactors had been operated successfully in the United States of America would help to put Mr. Moore's comments into proper perspective. They did not agree that the figures they had quoted for power output were in any way inflated. Mr. Moore had quoted figures from the Shippingport plant, but it must be realized that that plant represented only the first large-scale experiment and occupied, perhaps, a corresponding position in the development of nuclear power to that of the Calder Hall reactors. No one would suppose that the 45-MW. installed capacity of one of the Calder Hall reactors represented the ultimate limit for the graphite-moderated gas-cooled type, and it was equally unreasonable to imagine that the 85-MW. electrical output from Shippingport represented the ultimate limit for the pressurized-water type.

They agreed with Mr. Moore that the problem of the fuel element design and the related problems of neutron economy, reactor control, and fuel element handling formed the most difficult part of any reactor design and they had stated that specifically in the introduction to their paper. Those factors had to be taken into account along with the thermal and mechanical engineering considerations when preparing the design for a reactor core. The paper was necessarily limited in length and had dealt mainly with the thermal aspects of pressurized-water and boiling-water reactors, but they had not disregarded the other considerations and were well aware of their importance. They shared Mr. Moore's doubts on the subject of heavy water, and placed their main faith on the use of ordinary light water.

They agreed with Mr. Griffiths that nuclear efficiency was important and that it was necessary to achieve a good conversion factor if nuclear reactors were to be economic for power production. Mr. Griffiths was evidently unaware, however, that the light-water reactor, if properly designed, could achieve a conversion factor higher than for most other types of thermal reactor. Mr. Griffiths had quoted a conversion factor of 0·3 for the Russian atomic power station implying that that was a water-moderated reactor. In fact the reactor to which that figure referred was graphite-moderated and therefore had no bearing on the subject of the present paper.

They had presented some heat-transfer calculations for fuel elements with $\frac{1}{32}$-inch thick stainless-steel sheathing. That had been taken as an example of the worst heat-transfer conditions through a fuel element sheath, stainless steel having a particularly low thermal conductivity. It was not to be supposed that stainless steel would necessarily be chosen for the fuel element sheathing and it was well known that a zirconium alloy, developed in the United States of America, showed substantial nuclear advantages. In regard to the conflicting requirements of fuel rods of small diameter for high heat rating and of large diameter for high conversion factor, Mr. Griffiths appeared to be unaware of the compromise solution in which fuel elements could be made up from clusters of rods of small diameter or strips.

They welcomed the contribution of Dr. McIntosh and agreed with him that the experimental development work that had to be done to make the water-moderated reactor a reliable plant was almost entirely of a metallurgical nature. They had every confidence in Dr. McIntosh's ability to solve those problems, but they could not agree with him that steelmakers would be unable to produce mild-steel plate of adequate thickness for the construction of the pressure vessels which they envisaged. They had discussed the pressure vessel problem with some of the leading heavy-engineering firms in Britain and were confident that vessels of the type which they had described could in fact be made.

In regard to Professor Diamond's comment they could see no reason why the exacting condition of complete suppression of boiling could not ultimately be relaxed in the case of the pressurized-water reactor. Operating experience alone would settle that point, but it would certainly ease the design problem if the reactor pressure could be reduced to a level at which, while bulk boiling would still be prevented, some local boiling was allowed to take place at the surface of the fuel elements. Pressurized water and boiling-water reactors had been treated together because of the essentially similar technology involved. It was true that small experimental boiling-water reactors were under development in the United States of America, but it was expected that the construction of any full-scale boiling-water reactor plant would follow that of the more conventional pressurized-water type.

They were in complete agreement with the comments of Mr. Burdett, and they shared his view that the choice of suitable sites for nuclear power stations might become unreasonably difficult if separate combustion superheaters were required and facilities had to be provided for fuel handling and ash disposal.

In regard to Mr. Allard's point it was felt that the boiling-water reactor might show some advantage in its inherent stability but the real answer would again depend on operating experience with both types of plant. In reply to Mr. Warren they did not doubt that the graphite-moderated gas-cooled reactor was the correct choice for the first stage of the nuclear power programme in the United Kingdom, which must by necessity be based on the use of natural uranium. It should not be overlooked, however, that the graphite reactor presented considerable problems in fabrication and site erection and the field of application for the graphite-moderated reactor overseas seemed to be very limited. Mr. Warren's suggestions regarding the development of heavy water production, using electrical energy during off-peak periods, were interesting, but they thought that the capital cost of such a scheme might prove excessive. The calculations in Appendix I had referred not just to an isolated reactor, but to a complete project including the appropriate supporting diffusion and chemical processing plants. The revised figures had been based on the assumption that depleted uranium could be rejected from the system at a U235 concentration 0·3 times that of natural uranium.

They agreed with Mr. Plummer that the figure of £70 per kW. installed was excessive for a coal-fired or oil-fired superheater. They had substituted a figure of £40 per kW. in their revised estimate.

They agreed with Mr. Wootton that the separate superheater was attractive economically and thermodynamically in the case of large plants but the problem of finding suitable sites had to be kept in mind.