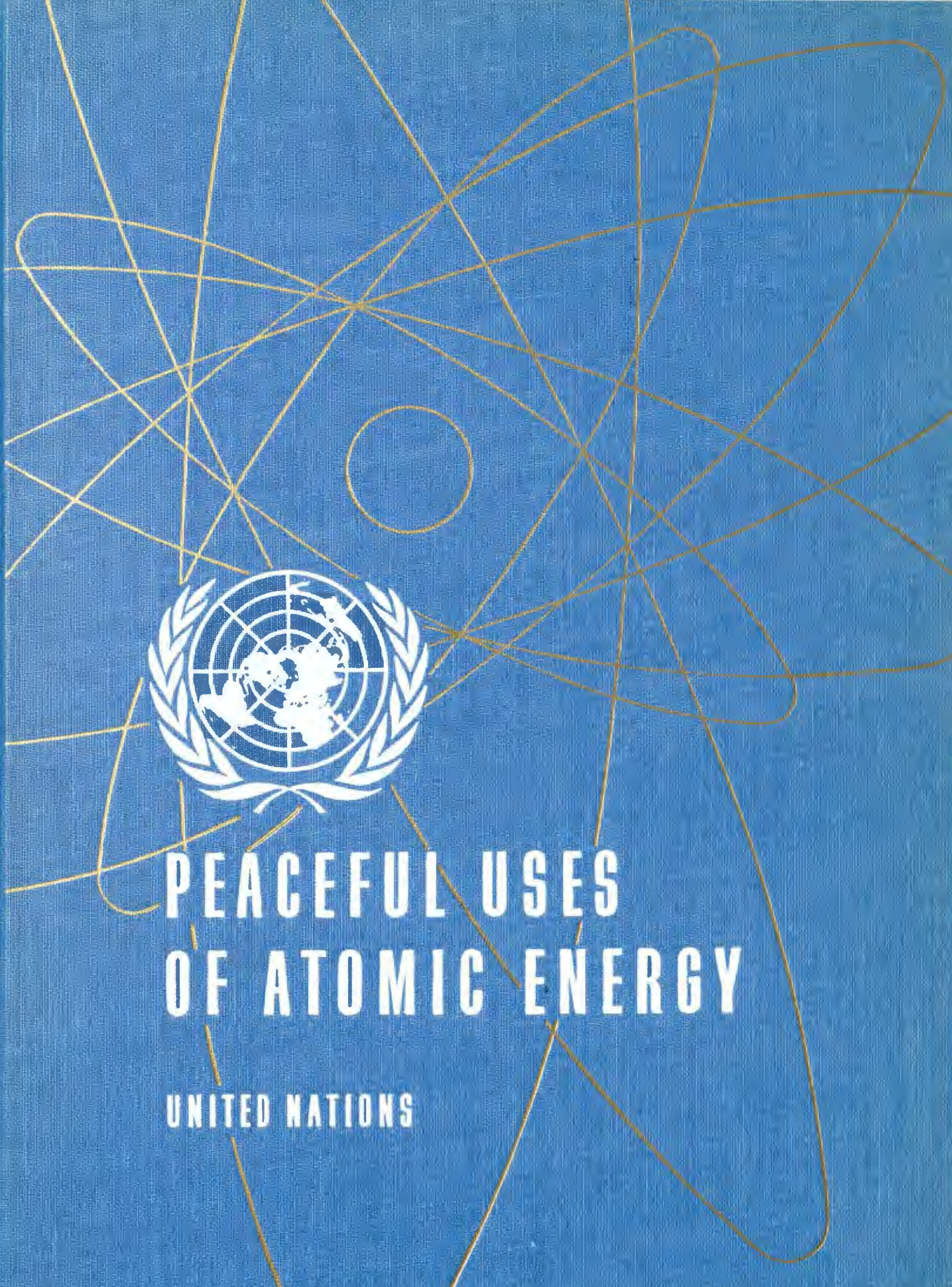

PEACEFUL USES OF ATOMIC ENERGY

UNITED NATIONS

PROCEEDINGS
OF THE
INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY

Volume 3
POWER REACTORS

Proceedings of the International Conference on the Peaceful Uses of Atomic Energy

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Volume 3
Power Reactors

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PREFACE

The Proceedings of the International Conference on the Peaceful Uses of Atomic Energy are published in a series of 16 volumes, as follows:

Volume Number	Title	Sessions Included
1	The World's Requirements for Energy; The Role of Nuclear Power	2, 3.2, 4.1, 4.2, 5, 24.2.
2	Physics; Research Reactors	6A, 7A, 8A, 9A, 10A.1.
3	Power Reactors	10A.2, 3.1, 11A, 12A, 13A, 14A.
4	Cross Sections Important to Reactor Design	15A, 16A, 17A, 18A.
5	Physics of Reactor Design	19A, 20A, 21A, 22A, 23A.
6	Geology of Uranium and Thorium	6B, 7B.
7	Nuclear Chemistry and the Effects of Irradiation	8B, 9B, 10B, 11B, 12B, 13B.
8	Production Technology of the Materials Used for Nuclear Energy	14B, 15B, 16B, 17B.
9	Reactor Technology and Chemical Processing	7.3, 18B, 19B, 20B, 21B, 22B, 23B.
10	Radioactive Isotopes and Nuclear Radiations in Medicine	7.2 (Med.), 8C, 9C, 10C.
11	Biological Effects of Radiation	6.1, 11C, 12C, 13C.1.
12	Radioactive Isotopes and Ionizing Radiations in Agriculture, Physiology and Biochemistry	7.2 (Agric.), 13C.2, 14C, 15C, 16C.
13	Legal, Administrative, Health and Safety Aspects of Large-Scale Use of Nuclear Energy	4.3, 6.2, 17C, 18C.
14	General Aspects of the Use of Radioactive Isotopes; Dosimetry	7.1, 19C, 20C.
15	Applications of Radioactive Isotopes and Fission Products in Research and Industry	21C, 22C, 23C.
16	Record of the Conference	1, 24.1, 24.3.

These volumes include all the papers submitted to the Geneva Conference, as edited by the Scientific Secretaries. The efforts of the Scientific Secretaries have been directed primarily towards scientific accuracy. Editing for style has been minimal in the interests of early publication. This may be noted especially in the English translations of certain papers submitted in French, Russian and Spanish. In a few instances, the titles of papers have been edited to reflect more accurately the content of those papers.

The editors principally responsible for the preparation of these volumes were: Robert A. Charpie, Donald J. Dewar, André Finkelstein, John Gaunt, Jacob A. Goedkoop, Elwyn O. Hughes, Leonard F. Lamerton, Aleksandar Milojević, Clifford Mosbacher, César A. Sastre, and Brian E. Urquhart.

The verbatim records of the Conference are included in the pertinent volumes. These verbatim records contain the author's corrections and, where necessary for scientific accuracy, the editing changes of the Scientific Secretaries, who have also been responsible for inserting slides, diagrams and sketches at appropriate points. In the record of each session, slides are numbered in numerical order through all presentations. Where the slide duplicates an illustration in the submitted paper, appropriate reference is made and the illustration does not appear in the record of the session.

Volume 16, "The Record of the Conference," includes the complete programme of the Conference, a numerical index of papers and an author's index, the list of delegates, the records of the opening and closing sessions and the complete texts of the evening lectures.

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FUEL CYCLES

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Some Economic Aspects of Nuclear Fuel Cycles

By W. B. Lewis, Canada

INTRODUCTION

It is generally recognized that regeneration of fissile material in a nuclear power reactor is necessary if the cost of nuclear energy is to compete with alternative means of generating large amounts of electric power. This regeneration of fissile fuel introduces a variety of possible cycles for the manipulation and processing of the fuel and of the source or fertile material from which the new fuel is produced.

It is of interest to consider the effects of choosing various fuel cycles on the cost of generating electric power. To establish the ranges which are of interest it may be noted by way of an example that in Ontario most of the electric power is generated from water power at a cost of 4 mills ($0.004) per kilowatt hour or less. Recently it has proved economic to add to the system generating stations burning imported coal that alone contributes to the cost of the electricity so generated more than 3 mills/kwh.

If every atom in a piece of uranium could be fissioned the energy yield at 200 Mev per fission would be 0.94 thermal megawatt days per gram. Assuming this energy could be converted to electric power at 25% efficiency and that uranium costs $44/kgm then this would contribute to the power cost 0.0078 mill/kwh. This is so small that it would be economically negligible and even if only 3% of the atoms are made to yield their fission energy, or, in other words, if 3% burn-up is achieved, the contribution to the cost of power from this initial cost of the uranium would be only 0.26 mill/kwh. There are, however, other costs contributed by the fuel.

On a world scale we have to envisage an installed nuclear electric generating capacity which may rise to several hundred million kilowatts within a human lifetime. A capacity of 100 Mkw operating 300 days a year at an efficiency of thermal to electrical energy conversion of 25% requires 120 tonnes to be fissioned per year. At 1% burn-up this represents a supply of 12,000 tonnes/year per hundred million kilowatts. It is apparent that for this large scale use we should be interested in fuel cycles giving rather more than 1% burn-up, and the fuel supply should not prove a major problem if 3% or more is achieved.

Considerations of both initial cost and availability of uranium or thorium for large scale power generation, therefore, separately lead to an early interest in achieving a cheap operating cycle for at least 1% and preferably about 3% burn-up. The latter is expected to remain sufficient for several decades.

FUEL FABRICATION AND REPROCESSING COSTS

Before insertion in most types of reactor, uranium has to be specially fabricated and sheathed. The costs of these operations are most conveniently considered as a component of the supply cost of the uranium.

In addition, however, uranium is likely to need subsequent reprocessing to remove fission products before it has yielded the postulated 3% of its fission energy. Moreover the radioactive fission products will require special disposal.

Of these costs, that for reprocessing appears likely to be the largest. This contribution can, however, be reduced by suitably choosing the irradiation and reprocessing cycle for the fuel.

As a target it is suggested that the fuel reprocessing cost will be acceptable if it contributes no more than 0.5 mill/kwh to the cost of the electric power generated. The chemical engineer will need to have this translated into terms of an allowance for processing given quantities of materials, so we may note that the allowance is 24 hours/day $\times$ 0.5 allowed mill/kwh $\times$ 0.25 assumed efficiency of conversion of thermal to electrical energy = $3 per gram of fission products.

If the fuel, whether natural, enriched or depleted uranium or thorium is irradiated to yield 4000 thermal megawatt days (Mwd) per tonne before reprocessing, it will contain 4 kg of fission products per tonne, and the above allowance would be $12 per kg of fuel to be processed.

It is of interest to write a more general expression for this processing allowance; it is

$$\$24\, n.\, c_p.\, e \text{ per kg of fuel}$$

where the energy yield is $n \times 10^3$ Mwd/tonne before reprocessing, c_p = allowable contribution from processing to the power cost in mill/kwh, and e = efficiency of conversion of thermal to electric power.

To the chemical engineer looking for the maximum allowance for processing this might seem to determine the allowance beyond hope of any major increase, but this is not so; for, if the fuel contained uranium-238, or thorium, some new fissile material plutonium or uranium-233 will have been produced that is chemically separable. The processing may be

designed to separate this and return it to the reactor in such a form that the fission products it subsequently yields do not cause the reprocessing of more U^{238} or thorium. Such segregation of the returned fissile material also makes it possible to eliminate the unwanted Pu^{242} isotope, which would otherwise degrade the stock of plutonium for use in thermal neutron reactors. This segregation can increase the processing allowance appreciably. This is illustrated by the example under Recycling with Natural Uranium Make-up.

For most fuel cycles involving high burn-up the processing allowance remains almost proportional to n, the thousands of megawatt days per tonne attainable before reprocessing. Physical deformation or disruption of the fuel and the poisoning effect on the reactor of neutron absorption by the fission products are the main factors setting the limit to n. The relative importance of these can vary over a wide range with the type of reactor. For reactors with solid fuel elements, absorption by the cladding or sheathing of the fuel, and by the coolant and its channelling will affect the permissible limit of fuel irradiation. It may, however, be possible to move fuel which has reached the limit for one position in the reactor to another where its contribution is still useful. Moreover, depleted uranium and thorium grow the fissile component under irradiation and so may be moved to positions where the fissile component is more valuable. By such manipulations the magnitude of n before reprocessing may be extended.

FUEL INVENTORY CHARGES AND MAKE-UP SUPPLY

There is a further contribution to the power cost from inventory charges on the fuel. These may be a major item in certain reactors, notably fast neutron breeders where the inventory of pure or highly enriched fissile material is necessarily large.

In the early years of nuclear power reactors the price at which separated fissile material is available is set by very complex considerations. Economic production demands large isotope separation plants, or combinations of large reactors and chemical extraction plants. On a small scale, with no extra revenues, cost estimates range up to many hundreds of dollars per gram in early years. This gives a marked advantage to reactors requiring little or no enriched fissile material and which, accordingly, mainly depend on natural uranium. This advantage may be maintained for several decades. Other fuel cycles, however, deserve consideration, in particular those in which only an initial inventory of enriched fissile material has to be provided; thereafter, the separated fissile fuel required is obtained from the reactor itself and only the reprocessing costs are involved. If this initial material is used at a high power rating, the inventory charges may be acceptable.

It is of interest to evaluate the price of separated fuel which would add, through inventory charges and any continued make-up required, an assigned contribution to the cost of generated electric power. Such a partial relation is shown in Fig. 1 for an assigned contribution of 0.5 mill/kwh to the cost of power, the same as suggested for fuel processing above. The uppermost line in Fig. 1 corresponds to an initial inventory with no further make-up, and carrying charges at 4% per annum.

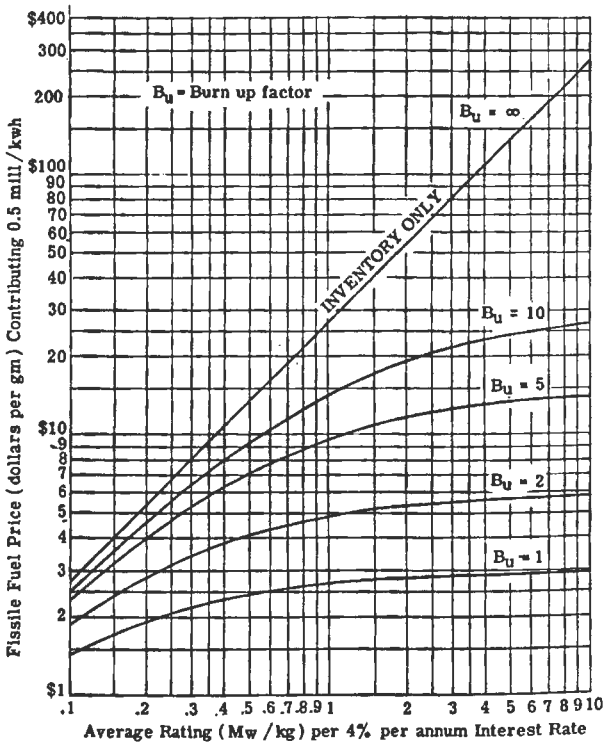

Figure 1. Value of fissile fuel

Both the "average rating" and "Burn-up Factor" in Fig. 1 have special meanings and may relate to the separated fissile fuel or the average fuel. Care must be taken in using the figure to ensure that consistent meanings are assigned. Figure 1 is derived by evaluating the contributions to power cost from the inventory and make-up supply of enriched fissile fuel as follows:

Inventory

The inventory is the total of all separated fissile fuel supplied to the reactor or system other than produced internally in the reactor, less the amount assignable to the steady rate of make-up.

The time average of the total thermal power in megawatts from the reactor per kilogram of the separated fuel inventory is the average inventory fuel rating $r_{av\,s}$ in Mw/kg. or kw/gm.

Then if $\$p_s$/gm is the price to be arrived at for the supplied separated fissile fuel and the inventory carries an annual charge of $a\%$, the charges are $(1000\ p_s)$ mills $\times\ a/100$ per year per gm.

Since the electric power generated per year of 8766 hours per inventory gm is 8766 $r_{av\,s.}$ e kwh, the contribution to the cost of power is

$$c_i = p_s.a./876.6\ r_{av\,s.}\ e \text{ mill/kwh}$$

Make-Up Fuel Supply

Setting B_{us} = burn-up factor = reactor thermal Mwd (or 24,000 kwh)/gm of separated fissile fuel supplied as make-up, the electric power is 24,000 $e.B_{us}$ kwh/gm.

The cost is 1000 p_s mill/gm of separated fuel supplied. So the contribution c_{fs} to the cost of power is

$$c_{fs} = p_s/24\ e.B_{us} \text{ mill/kwh.}$$

Combining the above contributions

$$p_s = (c_{fs} + c_i)\ e/(a/876.6\ r_{av.\,s} + 1/24\ B_{us}).$$

This same relation may be used in another way, applied to the total fissile fuel supply, it gives the contributions to the cost of power from inventory (c_i) and fuel supply (c_{fs}) in mills/kwh. Thus

$$c_i + c_{fs} = (a/876.6\ r_{av} + 1/24\ B_u)\ p./e.$$

where r_{av} and B_u relate to the average fissile fuel supplied, whether separated or not, and p is the cost per gm of this average fissile fuel as supplied to the reactor. This relation is used again and further extended later in this paper.

The required make-up supply of fissile fuel is by definition $1/B_u$ gm per thermal Mwd.

PROLONGED IRRADIATION OF NATURAL URANIUM

Before discussing fuel cycles in greater detail it is desirable to review certain characteristics of the available fuels and in particular natural uranium and the products of its irradiation. The general form of the significant changes expected in the long irradiation of natural uranium in a thermal neutron reactor is illustrated in Figs. 2, 3, 4 and 5. Due to very marked resonances in the neutron absorption by U^{238} and Pu^{240}, such curves do not have general application to any high degree of accuracy. The convention used in calculating these curves is to assume that the resonance absorption of neutrons bears a constant ratio to the thermal neutron absorption, so that "effective" thermal neutron cross-sections may be assigned. These will depend on the effective neutron

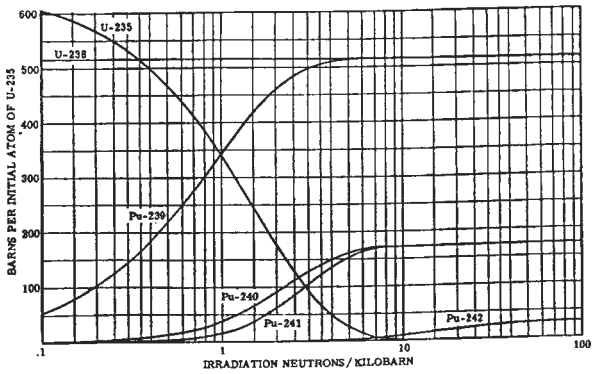

Figure 2. Long irradiation of natural uranium. Neutron changes due to growth of plutonium isotopes. Case 17 parameters

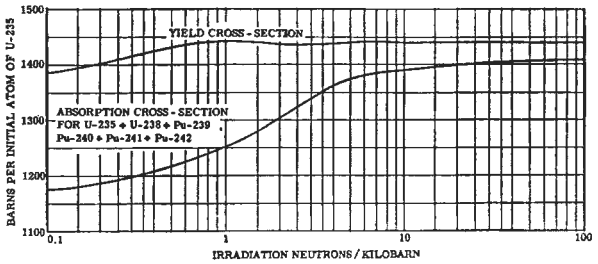

Figure 3. Conversion from natural uranium to equilibrium plutonium in thermal neutron reactor. Case 17 parameters

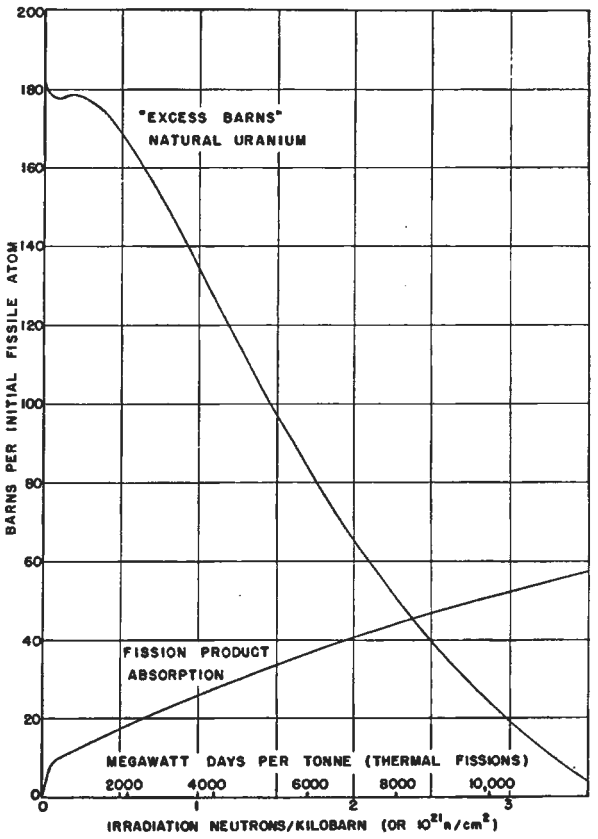

Figure 4

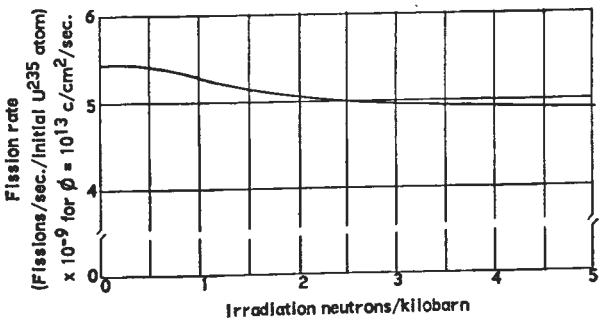

Figure 5. Long irradiation of natural uranium. Variation of heat output (fission rate) in constant flux

temperature and also, due to the large self-shielding effects at the resonances, on the form and arrangement of the uranium in the reactor, as well as the degree of irradiation.

In deriving Figs. 2, 3, 4 and 5, it is assumed that the amount of U^{238} is constant and that Pu^{239} is formed from it by the neutron flux at a maintained rate of 0.8 atoms of Pu per atom of U^{235} destroyed at the initial U^{235} concentration.

The other parameters assumed are given in Table I.

Table I. Case 17. Parameters*

$\sigma_8 = 645$ barns	$\sigma_9 = 1100$ barns	$\sigma_1 = 1700$ barns
$\sigma_{f5} = 545$ barns	$\sigma_{f9} = 730$ barns	$\sigma_{f1} = 1400$ barns
$\eta_5 = 2.112$	$\eta_9 = 2.00$	$\eta_1 = 2.35$
$\nu_5 = 2.50$	$\nu_9 = 3.0$	$\nu_1 = 2.85$
$\sigma_6 = 25$ barns	$\sigma_0 = 700$ barns	$\sigma_2 = 60$ barns

*σ is used for the absorption cross-section and σ_f for the fission cross-section. Subscripts 5, 6, 8, 9, 0, 1, 2 identify U^{235}, U^{236}, U^{238}, Pu^{239}, Pu^{240}, Pu^{241} and Pu^{242}. η = fission neutrons released per fissile atom destroyed. ν = neutrons released per fission.

This particular set of assigned parameters, designated Case 17, is one used in an extensive series of calculations. Measurements of these quantities are still in progress in many laboratories throughout the world and no one set of values is yet generally accepted. Furthermore, the relations used in the calculations are only approximations because of the great complexity involved in allowing for the self-shielding effects of the sharp resonances which exist in the neutron absorption by U^{238} and Pu^{240} in particular. The values used for the U^{235} constants differ insignificantly from the declassified values of 1950. By mass-spectrometer measurements of the ratios of Pu^{240}/Pu^{239} in the uranium rods irradiated in NRX up to 3000 Mwd/tonne, the capture cross-section of Pu^{239} (σ_{c9}) related to $\sigma_{f5} = 545$ barns was found to be 370 ± 15 barns. The values assigned to σ_0 and σ_1 are also chosen to fit the results of these measurements, but the errors may be large. Samples of plutonium have been irradiated in high flux positions in NRX and the resulting isotope ratios determined by mass-spectrometer measurements. These results were best fitted by $\sigma_0 = 500$, $\sigma_1 = 1500$, $\sigma_{c1} = 300$ barns, but in these irradiation positions the epithermal flux is only about half that in the uranium rods. The higher values for σ_1 and σ_{f1} adopted in part correspond to the higher epithermal flux in the rods and in part allow for the most marked effect of the self-shielding of Pu^{240}, which is to produce Pu^{241} first at a higher rate and then more slowly. The value of η_1 comes from guesswork—it corresponds to $\nu_1 = 2.85$ which seems not inconsistent with $\nu_9 = 3.0$. The values of σ_6 and σ_2 are from measurements made at Chalk River. It should be noted that all these values represent effective values in heavy water moderated, uranium metal reactors, they are not 2200 m/sec values.

The heat output, usually expressed in thermal megawatt days (Mwd), is correlated with the neutron flux and these parameters by assuming (1) the ratio of U^{238}/U^{235} atoms in natural uranium is 138:1, (2) the total heat in the whole reactor is 198 Mev/fission $= 3.17 \times 10^{-11}$ joules/fission $= 3.67 \times 10^{-22}$ Mwd/fission, (3) the ratio of total fissions/thermal fissions $= \epsilon^1 = 1.058$. (Note: $\epsilon^1 = 1 + (\epsilon - 1)\,\nu/(\nu_8 - 1)$ 1.058 for $\nu = \nu_8 = 2.5$, and "fast fission factor" $\epsilon = 1.035$ = ratio of total fission neutrons/neutrons from thermal fission). From these assumptions an irradiation of 10^{21} n/cm^2 or 1 n/kilobarn = 3850 Mwd/tonne. It happens that for the plutonium production assumptions above the rate of heat output in a constant neutron flux probably changes by less than 5% during prolonged irradiation. This is less than shown in Fig. 5 and follows from the parameters of Case 17 together with the assignment of $E_{f9} = E_{f1} = 206$ Mev (E_f = energy per fission). For Pu^{239} we find $\epsilon^1 = 1.07$, so that $(\epsilon^1 E_f)_9/(\epsilon^1 E_f)_5 = 1.07 \times 206/1.058 \times 198 = 1.05_2$. This partly compensates for the 10% lower fission rate of the equilibrium mixture of $Pu^{239} + Pu^{241}$, which follows from the assumptions.

In convenient numbers for an irradiation of 1 neutron/kilobarn natural uranium (with an initial regeneration factor of 0.8) yields 4000 Mwd/tonne.

It should be observed that the increase of absorption cross-section of natural uranium occurring between irradiations of 0.5 and 2.5 n/kb is largely due to the rise of Pu^{240}. An important step forward was made, and this happened with us in 1951, when it was realized that the high neutron absorption cross-section of Pu^{240} represented only an organizational disadvantage in a reactor. The role of Pu^{240} is closely parallel to that of U^{238} in providing a source for new fissile material. Its effect can be compensated either by removing U^{238} after chemical separation or by increasing the amount of fissile material in the reactor.

The neutron absorption expected from fission products is shown in Fig. 4. Not included in this is the absorption by the transient fission product Xe^{135}, which at equilibrium in a high flux reactor is believed to range from about 25 to 35 barns on the scale of Fig. 4. This major fission product poison is discussed in other papers presented to this conference. The initial step in the fission product curve of Fig. 4 is due to the long-lived fission products of very large cross-sections which rapidly reach a limiting absorption where the destruction rate balances the rate of formation. The most significant of these fission products is samarium-149. Figure 4 may be used to estimate the gain to be had from extracting the fission products in processing. Uranium-236 has been in-

cluded in the fission products, but is not removed by chemical processing.

The "excess barns" shown in Fig. 4 is in a sense a measure of the reactivity of the fuel. Its exact meaning and interpretation are discussed later.

LONG IRRADIATION OF U^{235} ENRICHED URANIUM

When U^{235} enriched uranium is used, the heat release in Mwd/tonne and the fission product build-up for a given neutron irradiation will be proportional to the U^{235} content. If the irradiation limit is set by the build-up of fission products, the processing will have to be carried out after the same heat release per tonne and therefore when the fraction of the U^{235} destroyed is smaller the higher the enrichment. For economical operation it is desirable to maintain a high fuel regeneration factor (number of new fissile atoms formed per fissile atom destroyed) and if this is the same as used in Figs. 2, 3, 4 and 5, the changes taking place will be similar to those shown by these figures, with the neutron irradiation scale changed.

A high regeneration factor is readily obtained in such an enriched uranium reactor by making the lattice more compact so that the capture of neutrons by U^{238} in the resonance range of energies at a few electron volts is promoted.

The U^{235} enrichment may be uniformly distributed through the uranium or concentrated in separate fuel elements. There are advantages and disadvantages of each method. An engineering convenience results from the steadily maintained heat output of the uniform fuel distribution. Another advantage of the uniform distribution is the greater contribution that may be obtained from the fast fission of U^{238}.

One disadvantage of the uniform distribution has already been mentioned, namely the need for the more frequent processing of the bulk of the uranium if the irradiation limit is set by its content of fission products. Another disadvantage is that U^{236} builds up in the fuel and is continually recycled because the high residue of U^{235} at each processing prevents it from being discarded with the U^{238}.

A third disadvantage is an increased inventory of U^{235} because uranium processing is liable to be delayed until U^{237} has decayed because of the associated γ-ray activity. If all the U^{235} is mixed with the bulk of the uranium it will suffer the same delay.

Another consideration, which in theory favours the uniform distribution but in practice may do the reverse, is the cycling of the fuel through an isotope separation plant. Assuming that an isotope separation plant of sufficient capacity is available, it is generally true to say that the operating cost is less if the whole supply is processed and the product is not taken to a higher enrichment than required. In practice the high capital value of the plant and the incidental handling charges may offset this theoretical advantage.

In practice the decision between these alternatives is likely to depend on more detailed considerations. The major question is whether the use of an isotope enrichment plant should be involved at all in this field of high power economical reactors. Some further considerations of this are given later.

PLUTONIUM AS A RECYCLED FUEL IN THERMAL NEUTRON REACTORS

When plutonium is extracted from irradiated uranium, the relative amounts of the different isotopes obtained will depend on the irradiation. If the extracted plutonium is returned for further irradiation, the isotope ratios will continue to change but will follow a course different from that of Fig. 2 because the Pu^{239} will be progressively destroyed and not replaced. An example of the changes that may be expected is shown in Fig. 6 which results from starting with plutonium extracted from natural uranium which has been irradiated to 1 neutron/kilobarn and producing initially 0.8 atom of Pu^{239} per atom of U^{235} destroyed. The nuclear cross-sections and other parameters for Fig. 6 are again those of Case 17 given above.

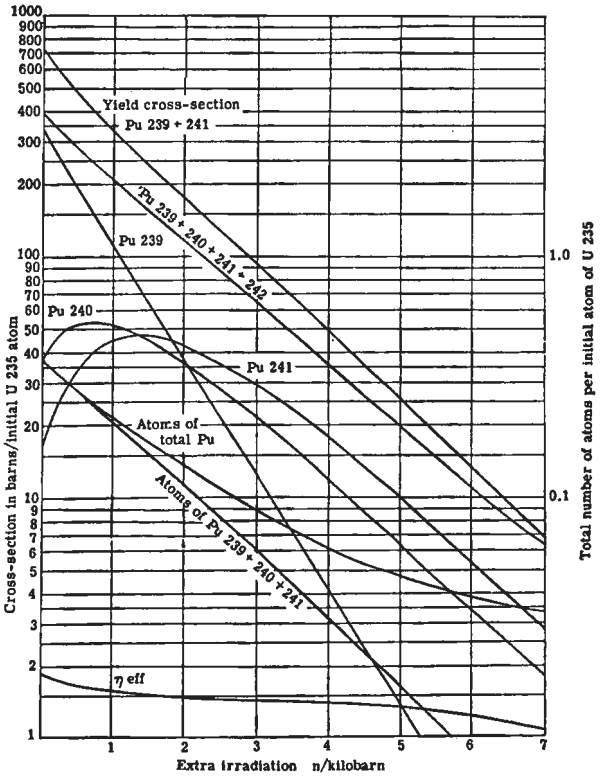

Figure 6. Irradiation of plutonium separated from uranium at 1 n/k barn. Case 17 parameters

It may be observed that after an irradiation of about 4n/kb the neutron yield from $Pu^{239} + Pu^{241}$ fission falls off at close to the same rate as the absorption by Pu^{241}, the rate being that set by the destruction of Pu^{240}. The isotope Pu^{242}, however, continues to grow throughout the irradiation range shown and this causes the total absorption cross-section to fall less rapidly, and thus the effective η ($\eta_{eff} =$

neutron yield/neutrons absorbed by all plutonium isotopes) falls. The lowest curve shows the variation of η_{eff} and shows that it remains in the range 1.5 to 1.4 from 2 to 4.5 n/kb irradiation. This steady value is due to the generation of the fissile Pu^{241} by the destruction of Pu^{240}.

These changes of neutron yield and absorption suggest that after an irradiation of 1.5 to 2 n/kb, when the yield has dropped to one-third of the original, the plutonium should be reprocessed, because it could then be returned to the reactor with one-third or less of the heat transfer surface and therefore with less parasitic neutron absorption by its sheathing. From 2 to 4.5 n/kb the neutron yield drops by a further factor of five, but the effective η does not drop much. If, however, the parasitic absorption by the sheathing and by the fission products is included in evaluating η, it would fall more steeply and would probably reach the useful limit at about 4.5 n/kb irradiation where the yield cross-section exceeds the absorption by 10 barns or about 160 barns per fissile Pu atom returned at 2 n/kb.

The plutonium, it is suggested, may be discarded at 4.5 n/kb for it will be seen that from the initial 38 atoms of Pu, only 5.3 (or 14%) remain, and of these 3.7 are atoms of Pu^{242} and the useful atoms discarded would be less than 4% of the original.

It must be remembered that Pu^{241} has a relatively short radioactive half-life, 13 years. To minimize loss, the plutonium has to be processed and recycled without long delay, and has to be irradiated at a high neutron flux.

RECYCLING WITH NATURAL URANIUM MAKE-UP

The costs and allowances associated with one particular cycle, with natural uranium as the make-up supply, will be discussed as a preliminary to assessing alternative cycles. It is envisaged that the irradiation limit of uranium is set by its content of fission products whether derived from U^{235}, Pu^{239}, Pu^{241} or U^{238}. Suppose this limit is about 4.25 kg of fission products per tonne. This results from fissions giving a heat output of 4250 Mwd/tonne.

The fuel cycle is shown in Table II.

The plutonium is returned separately as fuel to the reactor and by recycling is eventually all fissioned except that which becomes Pu^{242} (neglecting radioactive decay of Pu^{241}). The fraction becoming Pu^{242} is taken as $\sigma_{c9}\ \sigma_{c1}/\sigma_9\ \sigma_1 = 0.089$ for Pu^{239} and $\sigma c_1/\sigma_1 = 0.177$ for Pu^{240} and Pu^{241}. The total energy at 3.82×10^{-22} Mwd/fission (206 Mev) from Pu^{239} is 0.878 Mwd/gm and from Pu^{240} and ^{241} is 0.790 Mwd/gm. The total energy is then $17.05 + 0.878 \times 11.4 + 0.790 \times 4.18 = 30.35$ kMwd/tonne.

This estimate assumes a fast fission factor of 1.035, for all fissions taking place in uranium. This is likely to be an underestimate. No fast fission gain is assumed for the separated plutonium. No allowance has been made for processing losses, which should, however, be small. The energy from $Pu^{240} + ^{241}$ will have been overestimated because it is likely that some will be discarded with the Pu^{242}. The recycling of plutonium was discussed above.

The actual irradiations selected for the uranium fuels are those which yield 4000 Mwd/tonne by thermal fissions, the increase to 4.25 or 4.28 results from the contribution from fast fission of U^{238}. The calculation was made in this way to facilitate alterations later for other fast fission factors, since the irradiation limit is chosen somewhat arbitrarily, and the two are inter-related.

It may be noted that in the equilibrium condition of this cycling, almost half the power comes from the plutonium after separation and one-eighth of the power comes from each of the four types of uranium fuel. The more highly depleted uranium does not give a net gain of neutrons until it has been irradiated for some time; it is therefore expected in its early stages to occupy positions towards the outside of the reactor where neutron absorption is required to minimize leakage. These regions will also have a relatively low neutron flux, and since the irradiation of the fuel in this state has to be high, $\approx$ 1 neutron/kilobarn, the fuel will remain there relatively long, for example 1150 days at 10^{13} n/cm^2/sec. It follows that the cycle would take many years to reach its equilibrium. It is postulated that the reactor is one in which the position and type of fuel can be changed and it is therefore supposed that in the early years of operation it would be supplied with rather more natural uranium and less separated plutonium. Thorium also may be substituted in part for depleted uranium, and enrich-

Table II. Fuel Cycle for 3% Burn-up

Stage	*Fuel Type*	*Irradiation n/kb†*	*Fission Products kg‡*	*U^{235} kg*	*Pu^{239} kg*	*Pu^{240} Pu^{241} kg*	*Pu^{241} kg*
1.	Natural uranium	1.11	4.24	3.46	2.4	0.55	0.085
2.	1st depleted uranium*	1.66	4.25	1.19	2.8	0.95	0.195
3.	2nd depleted uranium	2.14	4.28	0.30	3.07	1.28	0.29
4.	3rd depleted uranium	2.33	4.28	0.066	3.13	1.40	0.33
	Total		17.05		11.4	4.18	0.90

* 1st, 2nd and 3rd depleted uranium are the products from stages 1, 2 and 3 respectively, after extracting Pu and fission products.

† kb = kilobarn = 10^{-21} cm^2.

‡ This column may also be read as kMwd/tonne heat output.

ment with U^{235} or U^{233} may be used in place of plutonium. The range of practical possibilities appears great, and the object of drawing attention to this is to guard against a too restricted and literal reading of the relations presented.

In the equilibrium condition, neglecting processing losses and the contribution to the power from fast fission of U^{238}, a thermal power of 28.8 kMwd is obtained from 1 tonne of natural uranium supplied. Making the same assumptions as previously, that the efficiency of conversion from thermal to electrical energy is 25% and the cost of uranium in the form of fuel supplied to the reactor is \$44/kg, this contributes 0.255 mill/kwh to the cost of power. If the processing adds 0.5 mill/kwh, the allowance of \$3/gm of fission products amounts to \$86.4 per kg of uranium supplied. With this allowance it is necessary to extract plutonium four times, to restore the uranium as fuel to the reactor three times, and to prepare and sheath the plutonium as fuel; and if the separated plutonium is reprocessed once in its irradiation life, as suggested above, to process and carry out this recladding of the recycled residual plutonium. In addition, the fission products extracted have to be disposed of. In essence the allowance amounts to \$21.6 for processing and cladding 4 gm of Pu and 1 kg U.

A 400 thermal megawatt reactor operating at 80% plant factor on this fuel cycle will require the processing of $320 \times 4/28.8 = 44.5$ kg of uranium per day. This is relatively small and it would be more economical to build one plant to serve ten or twenty such reactors. We have no experience yet of operating on such a scale and can only say that the allowance does not seem to present an unattainable target. It is of interest to see, however, how the allowance changes for different fuel cycles. The results of similar calculations for other cycles are shown in Figures 7 and 8. It will be seen that the processing allowance for a given fuel cost is almost proportional to the irradiation in Mwd/tonne before processing.

Not all the fuel cycles covered by Figs. 7 and 8 would prove practicable. Three important practical requirements have to be satisfied. The reactivity has to be maintained, the regeneration of fissile fuel must be sufficient and the limitations of heat removal set by the limits of fuel temperature must be satisfied. Some aspects of these are discussed in the following sections.

THE FUEL REGENERATION FACTOR

The attainable burn-up is related to the fuel regeneration factor. This may broadly be defined as the number of new fissile atoms formed per fissile atom destroyed. With this definition it will be denoted by γ, and appropriate subscripts will be added when it is necessary to indicate a more exact definition.

For 3% burn-up or, more specifically, for 30,000 Mwd/tonne of natural uranium supplied, we require (at 0.94 Mwd/gm fissioned) the fission of 3.19% of the natural uranium supplied, although this contains only 0.72% of U^{235}. For fission of 3.19 gm, we need $3.19\ \sigma_a/\sigma_f\ \epsilon^1 = 3.57$ gm of fissile material (using the U^{235} constants and D_2O and natural uranium re-

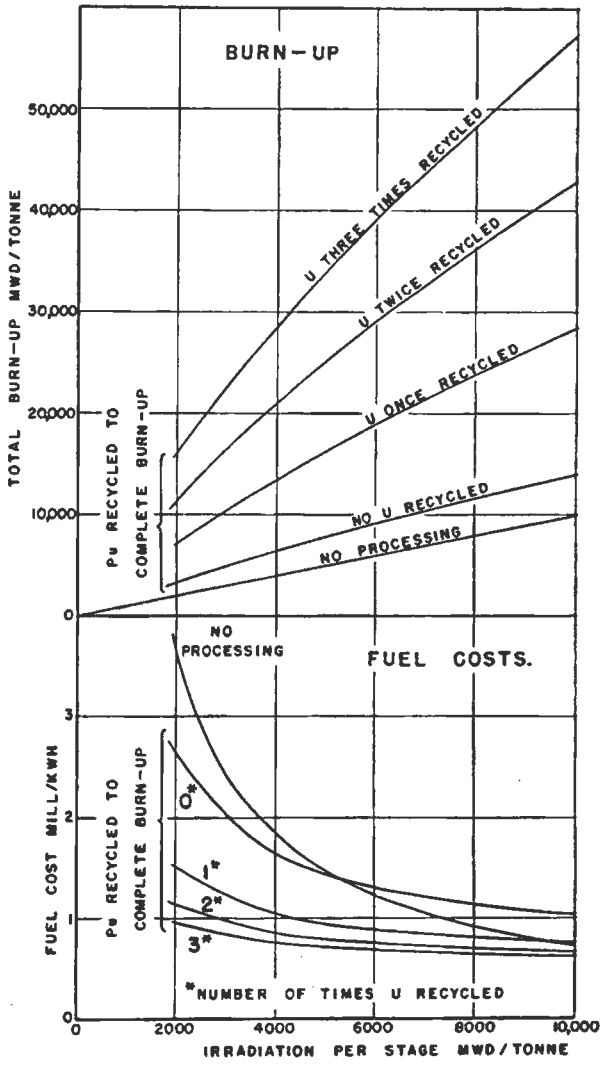

Figure 7. Burn-up and fuel costs

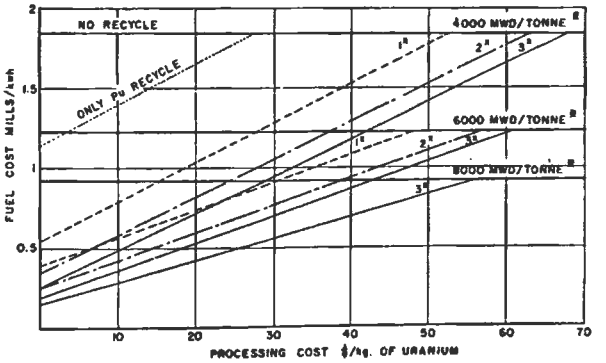

Figure 8. Fuel recycling for power reactors. Fuel costs related to uranium processing costs. Processing cost is the average for each extraction including return to reactor. Fuel cost is contribution to power cost from supply and processing of uranium at \$44/kg

actor constants); the required multiplication of fissile material is $3.57/0.72 = 5.0$. In terms of γ this multiplication is $1/1(1 - \gamma)$ so that $\gamma < 0.8$.

By the definitions $\eta\,\epsilon$ fission neutrons are released for each fissile atom destroyed. Of these, one is required to maintain the chain reaction, by definition γ produce new fissile atoms and the remainder ω are wasted by capture in the moderator, coolant, structural materials, fission products or escape from the reactor. Accordingly

$$\gamma = \eta\,\epsilon - 1 - \omega$$

The effect on the fuel cycle of changes in ϵ may be examined.

Since $\eta \approx 2$ and $\gamma \approx 1$ a 1% increase in ϵ not compensated by a change in ω gives a 2% increase in γ. The closer γ is to unity, the larger the resulting increase in the multiplication of fissile material (for $\gamma = 0.8$ the increase is 11%). Separately and in addition, a 1% increase in ϵ produces a 2% increase in ϵ^1 (for $\nu = 3.0$ and $\nu_8 = 2.5$) which gives a direct 2% increase in the heat output for a given fuel supply. In general ϵ is increased by concentrating the uranium; consequently the disposition of the uranium in the reactor has an effect on the attainable burn-up. The fuel supply cost contribution is directly affected and also the contribution from fuel processing if the reason for processing is loss of reactivity.

It does not appear necessary to achieve a value of γ much higher than 0.8, since at this level the fuel supply cost has become minor provided it is available at a price not much higher than natural uranium; on the other hand if there is value assigned to plutonium for any purpose such as starting up other reactors, increasing this by-product may contribute to lowering the net fuel costs.

So far the fuel regeneration factor has been treated as an overall factor applicable to all fuel. From considerations mentioned earlier, it is envisaged that the fuel distribution throughout a recycling reactor would be far from uniform. Earlier, a local regeneration factor was introduced in supposing that in natural uranium 0.8 atom of Pu was formed per atom of U^{235} destroyed at the initial U^{235} concentration. This initial local regeneration factor in natural uranium may be denoted γ_{a9} and may be written from first principles and definitions as

$$\gamma_{a9} = \eta_5\,\epsilon_e\,(1 - p)/(1 + l^2L_s^2) + N_8\sigma_8/N_5\,\sigma_5$$

where ϵ_e is the effective fast fission factor omitting fast neutrons which escape, p = resonance escape probability, l^2 = Laplacian or buckling and L_s^2 = slowing down length, the last two referring to the equivalent homogeneous reactor. N = number of atoms in a given mass. (Compare also $(1 + l^2L_s^2$ in next section.)

At a later stage in the irradiation the local regeneration factor γ_e may be written similarly

$$\gamma_e = \eta_{av}\,\epsilon_e\,(1 - p)/(1 + l^2L_s^2) + N_8\,\sigma_8 + N_0\,\sigma_0)/(N_5\,\sigma_5 + N_9\,\sigma_9 + N_1\,\sigma_1)$$

With the parameters assumed and for $1 + l^2L_s^2 = 1.05$, γ_e at first falls with irradiation to about 0.78 at 2000 Mwd/tonne, returning to 0.8 at 4000 Mwd/tonne and rising thereafter, but as it is envisaged that the plutonium is extracted, its later value may be of no special interest. This later rise has already been referred to above, where it was noted that the increase of absorption cross-section of natural uranium occurring between irradiations of 0.5 and 2.5 n/kb was largely due to the rise of Pu^{240}, and consequent production of Pu^{241}.

Although the above expression for γ_e involves p, it may nevertheless be written

$$\gamma = \eta_{av}\,\epsilon_e/(1 + l^2L_s^2) - 1 - \omega$$

which is independent of p.

The attainable value of the overall fuel regeneration factor can be rather greater than the local factors in the interior of the reactor. Unfortunately the values of η, particularly that for Pu^{241}, and the attainable ϵ are not yet known with sufficient accuracy to determine whether γ can be as high as 1.0 in a recycled system fed with natural uranium, but it certainly appears practicable to achieve values in the range 0.8 to 0.9, which are sufficiently high to permit 3% burn-up.

REACTIVITY REQUIREMENTS

It will be seen from Fig. 3 that in the prolonged irradiation of natural uranium the yield of neutrons does not rise as rapidly as the absorption (for initial $\gamma = 0.8$). At some point the reactivity contribution from a particular element of fuel will drop below the acceptable level. Unfortunately, for a reactor with non-uniform fuel distribution no simple relation has been found for balancing absorption at one point by an increased yield at another. A simple relation which exists for a uniform homogeneous reactor has been used as a guide to design, but its application is limited and not exact. The relation is that there would be no change of reactivity in a homogeneous reactor if a change of yield cross-section $\delta\sigma^*$ (yield cross-section = $\eta \times$ absorption cross-section of the fuel to which η applies) is compensated by a change of absorption cross-section $\delta\sigma$ at the same flux given by

$$\delta\sigma_f - \delta\sigma^* \frac{\epsilon \cdot P}{1 + l^2L_s^2}$$

where p = resonance escape factor, l^2 = Laplacian or buckling, and L_s^2 = slowing down area, with the interpretations of simple reactor theory.

This gives rise to the use of "excess cross-section," where "excess cross-section" = (yield cross-section) $\epsilon\,p/(1 + l^2L_s^2)$ — (absorption cross-section) as a measure of the "reactivity" of fuel.

This may be used to examine the irradiation limits which may be set for a given fuel although it cannot be pursued to close limits.

The effect of a given change of absorption cross-section $\delta\sigma$ on the reactivity is approximately proportional to the square of the neutron flux at the position where the change is made. If, therefore, by irradiation a fuel element loses "excess cross-section" the reactivity may be increased by moving it to a position of lower neutron flux and replacing it by an element of higher "excess cross-section."

Moreover, a recycling reactor is likely to have a considerable amount of fuel in outer regions where κ^2 is negative ($k_\infty < 1$). In the outermost regions, no fissile component is required, as this would lead to loss of neutrons, but closer in the irradiated fuel may be relatively more valuable, as indicated by the expression for excess barns for κ^2 small or negative. Fuel which has a damping effect near the centre of the reactor, where κ^2 must be large, may still be beneficial where κ^2 is smaller or negative. This however, cannot be followed through simply to a quantitative comparison, especially if the reactor is small and the fuel distribution changes within distances of the same order as L_s.

While the reorganization of fuel may be essential to preserve reactivity, and this may affect operating costs in such a way that the fuel cycle is changed, it appears that reactivity considerations do not otherwise directly affect the economics of fuel cycles.

Figure 3 indicates that in the absence of fission products the "yield cross-section" always exceeds the "absorption cross-section" for a fuel regeneration factor of 0.8. Figure 4 indicates that the fission product cross-section is smaller than this excess even to well beyond 10,000 Mwd/tonne irradiation. Other inevitable losses including absorption by xenon-135 do not change this. It follows that in theory, by reorganizing the fuel in a sufficiently large reactor, all the fuel cycles of Fig. 7 maintain an excess of reactivity. Limiting the size of the reactor and other practical design considerations may limit the irradiation before reprocessing. It appears impossible yet to be more exact in a general assessment.

The cycle appears to stay well within the limits set by reactivity considerations, provided a non-uniform fuel distribution is adopted in the reactor.

THE COST OF NEUTRON ABSORBERS

Sheathing costs have been mentioned as contributing to the cost of power through inclusion in the initial cost of fabricated fuel and also in the reprocessing cost. There is, however, one further contribution, namely through the value of the neutrons captured by the sheathing material and thus lost from the fuel regeneration cycle.

A simple way of appreciating this cost is to note that from the regeneration relation $\gamma = \eta\epsilon - 1 - w$ any one neutron added to the wastage means the loss of one fissile atom. If the system is not breeding an excess of fissile material, any fissile atom lost has to be made up in the supply at a cost of $\$p$/gm. Since 1 gm of neutrons produces 239 gm of Pu, we may assign a value of approximately \$240 p to a gram of neutrons, taking the cost of U^{238} or Th^{232} as negligible. If the make-up supply is natural uranium at \$44/kg this sets a value of \$1450/gm on neutrons wasted in the reactor. With the conditions duly noted, this relation is correct (it is indeed independent of the Burn-up Factor, B_u, although the loss affects the cost by changing B_u) but it may happen that an increase in neutron absorption necessitates buying all the fuel at a higher enrichment and a higher value of p, or may involve a drastic revision in the construction and operation of the reactor. A more complex relation also holds if a major portion of the fuel is being discarded without reprocessing; the value of a neutron then depends on its alternative fate and may become small if it was likely to be absorbed in fuel soon to be discarded for reasons other than loss of reactivity.

Assigning this value of \$1450/gm to neutrons, it is of interest to evaluate the neutron cost of irradiating various typical substances to 1 neutron per kilobarn. It is \$1450 σ_b/A per kg where σ_b is the capture cross-section in barns and A is the atomic weight. Examples are given in Table III.

Table III

Material	*Atomic weight*	*Density gm/cm³*	σ_b *barns*	*Irradiation cost per neutron/kilobarn†*	
				$ per kg	*$ per litre*
Lead	207.2	11.37	0.17	1.3	15
Zirconium	91.2	6.50	0.18	3	19
Magnesium	24.3	1.74	0.06	3.6	6.2
Aluminum	27.0	2.7	0.22	12	32
Sodium	23.0	0.97	0.50	32	31
½(D_2O)	10	1.10	0.00046	0	0
½(H_2O)	9	1.00	0.33	53	53
Iron	55.8	7.86	2.4	62	490
Stainless Steel*	55.3	7.92	2.88	76	600
Nickel	48.7	8.9	4.5	111	990

* Composition 68% Fe; 18% Cr; 10% Ni; 2% Mn; 0.8% Nb; 1% Si.

† For example, 4 months (116 days) at 10^{14} n/cm²/sec.

It is apparent that these costs which may be incurred in an irradiation of a few months or even weeks, are in some cases high compared with the normal purchase prices. They do not, however, represent the full basis for an economic appraisal, because equally large costs or savings may be involved in organizational changes of the reactor and its fuel cycle. One example of special interest but also of great complexity, is the assessment of the relative costs between ordinary and heavy water in a reactor. For this and the shielding evaluation, the analysis must be extended.

Assembling the contributions to power cost already evaluated, we have the fuel processing contribution (c_p):

$$c_p = P/24\, n_g e \text{ mill/kwh}$$

where P is the processing cost in \$/kg and the energy yield is n_g kMwd/tonne per processing including the subsequent energy yield from the fissile fuel formed by regeneration and chemically separated and recycled. It may be noted that $0.94\ P/n_g$. e is the processing cost in \$ per gm of fission products at 0.94 Mwd/gm. Also we have the inventory (c_i) and fuel supply (c_{fs}) contributions

$$c_i + c_{fs} = \left(\frac{a}{876.6\, r_{av}} + \frac{1}{24\, B_u}\right) p./e.$$

Also $$1 - \gamma = 0.94\ \epsilon^1\ \sigma_f/\sigma_a B_u$$

and $$1 - \gamma = 2 + w - \eta\epsilon$$

So we have

$$1/B_u = (2 + w - \eta\epsilon)\ \sigma_a/0.94\ \epsilon^1\ \sigma_f$$

Writing c_f = total contribution to power cost from fuel

$$= c_p + c_i + c_{fs}$$

we have $$\frac{\delta c_f}{\delta w} = \frac{\sigma_a p}{24.e. \times 0.94\ \epsilon^1 \sigma_f}$$

If an absorber introduces a parasitic cross-section σ_p barns per fissile atom, the corresponding change in w is $\delta w = \sigma_p/\sigma_a$ and the increase in the cost of power $\delta_{cf} = \sigma_p.p./24e \times 0.94\ \epsilon^1\sigma_f$.

Suppose the increased fuel requirement is met by enriched U^{235} costing \$6.1 $\times\ E_p$/gm so that E_p is a factor > 1 attributable to the cost of enrichment. Also assuming that U^{235} is the main fuel, and using the values of Table I, we find:

$$\delta_{cf} = \sigma_p.E_p/532 \text{ mill/kh.}$$

This relation gives the contribution from neutron capture in sheathing to the cost of power. With the following definitions:

Symbol	*Unit*	*Significance*
t	*cm*	Thickness of sheathing
w_s	kw/cm²	Surface heat transfer rate
ρ	gm/cm³	Density of sheathing metal
A_s	gm/6.02 × 10^{23} atoms	Atomic weight of sheathing metal
σ_s	barn	Nuclear absorption cross-section for thermal neutrons
r	thermal Mw/kg	Power rating of fissile component of fuel

An intermediate result may be noted, that the mass of sheathing in kg per kg of fissile fuel is $\rho tr/w_s$. Hence the parasitic cross-section in barns per fissile atom is $\sigma_p = \rho tr\sigma_s.\ 235/A_s w_s$. Using this in the last relation gives the cost contribution. The data for aluminum and zirconium are given in Table IV.

It is apparent that if the fuel power rating is high, the surface heat transfer rating should also be high to keep this cost contribution small.

The same relation may be applied to estimate the cost of leaving fission products in the reactor. From Fig. 4 it appears that the gain attainable by removing fission products at 1 n/kb is roughly 20 barns per fissile atom. Hence

$$\delta c_f \approx 0.04\ E_p \text{ mill/kwh}$$

The minimum amount of water required as coolant in a reactor may be arrived at by first equating the heat released by the fuel to that taken up by the coolant, thus

$$1{,}000\ r.l./v = 4.19.\theta m_w$$

where the symbols have the following significance

Symbol	*Unit*	*Significance*
r	thermal Mw/kg	Power rating of fissile component of the fuel at maximum power
l	cm	Length of cooling water channel
v	cm/sec	Linear velocity of water in cooling channel
l/v	sec	Time taken by water to flow through the reactor
4.19	watt-sec/gm°C	Specific heat of water
θ	°C	Rise of temperature
m_w	kg	Mass of water in reactor per kg of fissile fuel

Rewriting the equation to express the mass of water in the reactor,

$$m_w = 239\ r.l./v.\theta. \text{ kg/kg of fissile fuel}$$

Table IV. Sheathing Material Assessment

				For t = 0.1 and w_s = 0.2		
	A_s	ρ gm/cm³	σ_s barn	*Relative mass; sheathing/fiss. mat. kg/kg*	σ_p b/fiss. atom	*For r = 10* σc_f mill/kwh
Zr	91.2	6.50	0.185	3.25 r	1.55 r	0.03 E_p
Al	27.0	2.7	0.22	1.35 r	2.58 r	0.05 E_p

Other conditions are likely to determine r, l *and* θ so the minimum cooling water is set by the maximum practical v, for example 915 cm/sec.

The thermal neutron capture in ordinary water is due to the capture cross-section of H = 0.33 barn. Then if σ_w is the capture cross-section of the water per fissile atom (taken as U^{235}), we have $\sigma_w = 0.33 \times$ (atoms of H/atoms of U^{235}) barn which from the last equation gives $\sigma_w = 0.33 \times 239\ (l.r./v.\theta) \times 235/9$.

Suppose $l = 200$ cm, $\theta = 25°C$, $v = 915$, then $\sigma_w = 18.0\ r$.

Making the assumption that the neutron flux is the same for the water as for the fissile fuel, we may identify σ_w with σ_p and so we find

$$\delta c_f = 18\ r\ E_p/532 \text{ mill/kwh}$$

Even for no enrichment cost ($E_p = 1$) if $r = 10$ thermal Mw/kg, $\delta c_f = 0.34$ mill/kwh. In comparison with this, the cost of heavy water as coolant in the reactor, with no leakage loss, would be quite negligible, but this is irrelevant for the interest lies in considering the total cost of heavy water as moderator and coolant. Estimates suggest that 80 tonnes of heavy water may be sufficient for a reactor generating 100,000 electrical kw. If heavy water costs $66/kg and the plant averages 7000 hr/yr, the cost contribution to the power is $a_w \times 80 \times 66{,}000/700{,}000 \times 100 = 0.075\ a_w$ mill/kwh where $a_w\%$ is the annual charge for interest, losses and other expenses assigned to the heavy water. Experience is lacking on large high pressure systems, but Chalk River has experienced very low losses of heavy water and it seems reasonable to set a_w between 5 and 10%, giving $\delta c_f = 0.38$ to 0.75 mill/kwh.

It is clear that if r is made small, r_{av} will be small and c_i may become significant, particularly if the enrichment cost E_p is considerable. So far, therefore, the relative advantages seem to depend on whether heavy water losses can be kept low and what value must be assigned to E_p. This, however, is only part of the necessary comparison.

In all known designs of heavy water reactors so far constructed, ϵ is smaller than appears to occur naturally in light water reactors. It is not clear, however, that with fuel recycling the heavy water design cannot be adjusted to have almost the same fast fission contribution. We have seen that an increase of ϵ offsets an increase in w in the regeneration cycle.

The light-water design certainly requires more enriched fuel and the higher cost of this enters not only in the make-up fuel supply but also in the inventory and hold-up in processing.

The practical question must therefore be left open, but it seems that for the large power reactors considered, the heavy water reactor with recycled fuel promises a very attractively cheap range of fuel cycles.

ACKNOWLEDGEMENT

The information and assessment presented in this paper rests on contributions made by many at Chalk River that can be acknowledged only in general terms. My indebtedness will become apparent when the details of their experiments and results are published.

Fuel Cycles and Types of Reactor

By J. V. Dunworth,* UK

INTRODUCTION

The elementary features of breeding with plutonium-U^{238} in fast reactors and with U^{233}-thorium in thermal reactors are well known or will be discussed in other papers to the conference. Consequently, only brief reference will be made to them in this paper.

The first part of the present paper has been concerned primarily with drawing attention to some of the curious features of the plutonium-natural uranium fuel cycle in thermal reactors. The second part of the paper considers reactor systems and the influence of nuclear economy upon them.

A nation starting an atomic power programme without the help of any other nation must build a diffusion plant or a reactor which can operate with a supply of natural uranium, or both. In the latter case one is likely to use either a graphite or heavy water moderator. For a large heat output, large amounts of heavy water are required and the tendency is therefore to build graphite piles. For an experimental reactor, very much less uranium is needed for obtaining criticality with a heavy water system than is the case for a graphite system. These simple facts have conditioned most nations' atomic energy programmes to-date, and especially outside the USA.

The only naturally occurring fissile material is U^{235}. However, it has long been appreciated that with natural uranium as a starting point it might become possible to burn substantial quantities of the U^{238} with which it is associated. The fission of these nuclei directly is possible only to a small extent with the faster neutrons available in reactors, and so the sequence envisaged is first by conversion to Pu^{239} with subsequent fission of these nuclei.

Clearly the extent to which this can be exploited depends on the number of neutrons available for absorption in fertile material. This number, usually called the conversion factor, must be less than $(\epsilon\eta - 1)$† in any actual reactor and for a practical thermal reactor fuelled with natural or slightly enriched uranium will probably be less than unity. In such circumstances the reactor will run for a limited time on a single fuel charge, that period depending on the relative cross-sections of plutonium and U^{235} and on the respective η-values.

The possibility of extracting plutonium from this first charge and reinserting with part of the depleted uranium in the second charge, was recognized long ago (see, for example, BR7 by Halban and Kowarski, 1941). In a suitably designed reactor this process eventually leads to a fuel cycle in which natural uranium only is supplied, highly depleted uranium rejected and plutonium is recycled through the reactor (Fig. 1). This cycle with a natural uranium fuel feed is possible even if some degree of enrichment is necessary in the initial fuel charge. It is of interest to find the amount of U^{238} that can be burned and how it depends on initial enrichment and conversion factor.

NUCLEAR PROCESSES

The reactions of most importance are:

$$U^{238} + n \longrightarrow U^{239} \longrightarrow Np^{239} \longrightarrow Pu^{239}$$

$$Pu^{239} + n \longrightarrow \eta_9 \text{ neutrons} + \alpha\ Pu^{240} + \text{fission products}$$

$$\alpha\ Pu^{240} + \alpha\ n \longrightarrow \alpha\ Pu^{241}$$

$$\alpha\ Pu^{241} + \alpha\ n \longrightarrow \alpha\ \eta_1 \text{ neutrons} + \text{some } Pu^{242} + \text{fission products}$$

where η_9 refers to Pu^{239}, η_1 refers to Pu^{241} and where α is defined by the equations.

On the left-hand side we see that $2(1 + \alpha)$ neutrons are used to burn a U^{238} atom but in return $(\eta_9 + \alpha\ \eta_1)$ appear on the right-hand side. In addition, neutrons are lost in reactor components and in leakage, say P per U^{238} burned.

Then if:

$$P \leqslant (\eta_9 + \eta_1) - 2(1 + \alpha)$$

the system is self-maintaining or has a positive gain factor. Very roughly,

$$\eta_9 + \alpha\ \eta_1 = 2\tfrac{2}{3} = 2(1 + \alpha)$$

Hence as P is significantly positive, the Pu-U^{238} thermal system cannot be self-sustaining and a supply of neutrons is required to burn the U^{238}. One source of such neutrons can be fission of the U^{235} which occurs naturally with the U^{238}, but this is possible only if a balance is maintained between the number of

*Atomic Energy Research Establishment, Harwell.

†ϵ is the fast fission factor, excluding fast capture in U^{238}, and η is the net number of neutrons available per neutron absorbed in a fissile nucleus.

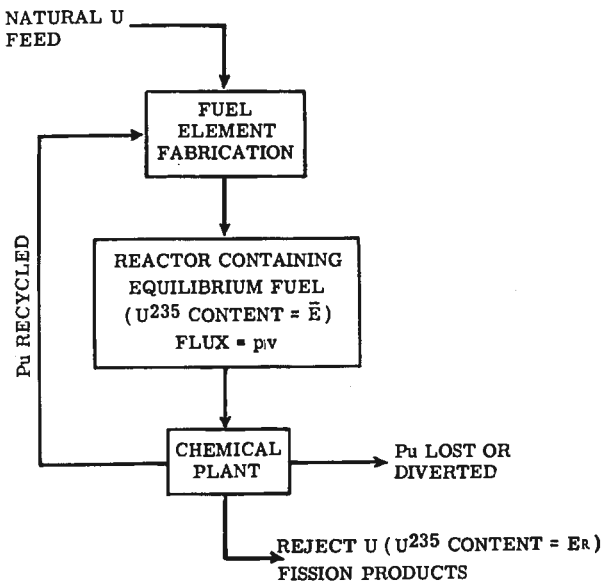

Figure 1. Fuel cycle in equilibrium system

neutrons provided by fission (i.e., from U^{235}, Pu^{239} and Pu^{241}) and the number of neutrons lost in causing the fissions, in absorption in fertile material, in fission products, moderator, coolant and structural materials and leakage from the reactor. This is the criticality condition for a reactor.

INITIAL FUEL STATE

The prediction of operating size of reactors and the fates of neutrons from a knowledge of basic nuclear data is hampered by inaccuracy or incompleteness of such data. But semi-empirical formulae which predict such quantities for reactors containing natural or enriched uranium have been obtained in various reactor physics laboratories. Such systems are therefore the natural starting point. Conditions in reactors derived from them, by recycling plutonium until equilibrium is reached, can be estimated from a knowledge of the effective cross-sections of the nuclei produced in the uranium fuel. Any attempt to do this at present leads one into extremely detailed and involved mathematics in which the essential physical principles are lost sight of. Accordingly in this document a highly simplified approach is adopted.

EQUILIBRIUM FUEL CYCLE

We will assume that η for both plutonium-239 and uranium-235 are the same and ignore for the moment plutonium-240 after formation.

Suppose that, initially, a reactor has enrichment E and, that when the equilibrium fuel cycle (Fig. 1) is reached, the plutonium content is such that the average enrichment of uranium in the pile is $\bar{E}$:

Suppose the conversion factor is C:

If the quantity of plutonium in the reactor is to remain constant, then:

$$\bar{E}C + (E - \bar{E})C = (E - \bar{E})$$

Hence:
$$\bar{E} = E(1 - C)$$

We must now consider Pu^{240} and its fissile offspring Pu^{241}. We will assume that η for Pu^{241} is the same as for U^{235}. In this case we may consider Pu^{240} and Pu^{241} as replacing an appropriate number of U^{238} and fissile atoms. A little reflection will show that for each neutron absorbed in U^{238}, a neutrons are absorbed in Pu^{240}. Hence the amount of U^{238} in the reactor at equilibrium is less by $1/(1 + a)$ than assumed above.

Hence:
$$\bar{E} = E(1 - C)\ (1 + a)$$

We have then that the average U^{235} level in the equilibrium fuel state depends: (1) on the initial enrichment required to start the system; (2) on the amount by which the conversion factor is less than unity; and (3) on the factor $(1 + a)$ which is a property of Pu^{239} and will depend on the effective spectrum in the reactor.

Now a reactor can be fueled in the equilibrium state with natural uranium if $\bar{E}<1$. The very important result is obtained therefore that a reactor, requiring an initial enrichment E, can be operated on a feed of natural uranium subsequently, if $E(1 + a)(1 - C) <1$. It is clearly possible, even, to operate on a feed of moderately depleted uranium, if $\bar{E} \leqslant 1$.

These simple arguments ignore fission-product poisoning, Pu^{242}, and the radioactive decay of Pu^{241}. However, it is believed that the result will not be improved substantially without very detailed analysis involving such problems as plutonium resonances, fine structure in the lattice, exact values of nuclear constants, design of reactor, distribution of neutron flux and processing losses.

URANIUM REJECT CONCENTRATION

Assuming the equilibrium state has been reached, the reactor may be regarded as a box with constant neutron flux, a steady input (say, m gm/sec) of natural uranium and the same output of depleted uranium plus fission products. The uranium present in the box at any one moment will have had all times of exposure between $t = 0$ and $t = T$ where T is the length of time spent in the reactor and the average U^{235} enrichment will be:

$$\bar{E} = \frac{\int_0^T m e^{-\rho\nu\sigma_5 t}\, dt}{\int_0^T m\, dt} = \frac{1 - e^{-\rho\nu\sigma_5 T}}{\rho\nu\sigma_5 T}$$

where ρv is the neutron flux in the reactor.

The reject level $E_r = e^{-\rho\nu\sigma_5 T}$

We have the relationship between $\bar{E}$ and the reject U^{235} level E_r shown in Fig. 2, and the U^{235} consumed is therefore the difference between E_r and the feed level which will normally be that corresponding to natural uranium (i.e., $E = 1$).

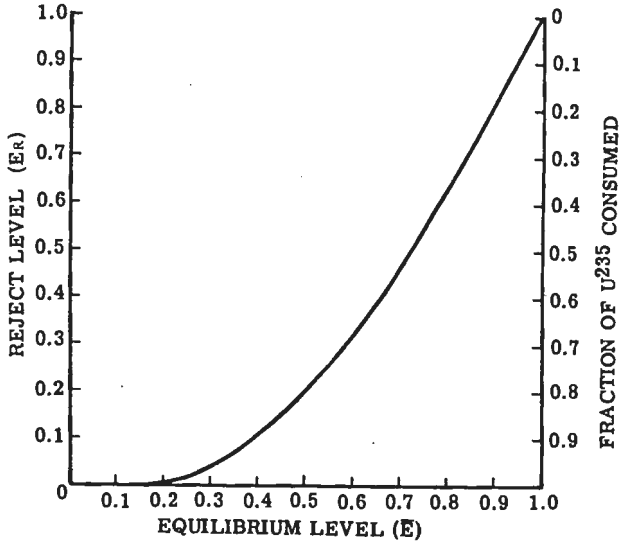

Figure 2. Reject U^{235} level E_r vs equilibrium level E

In general $e^{-\rho\nu\sigma_8 T}$ will be much less than 1 and E_r will be approximately

$$e^{-\frac{1}{E}} = e^{-\frac{1}{E(1+\alpha)(1-C)}}$$

U^{238} CONSUMPTION

Each U^{235} atom consumed produces C atoms of Pu^{239} from U^{238}. If all the Pu remains in the system it must be burned on the assumptions made earlier to produce C^2 etc. so that the ultimate U^{238} consumed must be:

$$C + C^2 + C^3 + \ldots\ldots\ldots\ldots\ldots = \frac{C}{1-C}$$

atoms of U^{238} per U^{235} destroyed.

Combining this with the U^{235} consumption we are able to deduce the proportion of natural uranium consumable in a reactor designed to start with a uranium fuel, enriched by a factor E, and giving a conversion factor C. Figure 3 shows this for a range of values of E from 1 to 3, and C from 0.3 to 0.9 with the equivalent scale of heat output in Mwd tonne of natural uranium throughput.

EFFECT OF INITIAL ENRICHMENT

It will be apparent from the treatment above that, apart from the initial fuel cost, the need to enrich a reactor fuel is not necessarily a drawback. For example, one may compare a system needing a natural uranium charge initially to give a conversion factor of 0.8 with systems designed to use two-fold enrichment and a conversion factor of 0.9. The total burn-up of the natural uranium feed would be the same.

Figure 3 shows, for example, that an initial enrichment $E \leqslant 3$ may well be tolerable if $C > 0.8$.

DIVERSION OF PLUTONIUM

It is an interesting fact that a reactor requiring initial enrichment and then converted to an "equilibrium" fuel cycle with a feed of natural uranium may still have a sizeable net output of plutonium without reducing the total fraction of U^{238} atoms consumed by too large a factor. The effect on the burn-up attainable can be assessed readily by reducing the conversion factor by the fraction of the Pu produced which is so diverted.

The diversion of plutonium can be made for several reasons:

1. It can be used with natural uranium in lieu of slightly enriched uranium from a diffusion plant as a means of fuelling a new reactor.
2. It can be used in mobile power plants in lieu of highly enriched uranium.
3. It can be converted to U^{233} in a power reactor, so as to provide make-up, in case a thorium-U^{233} fuel is used in a reactor where the over-all conversion is less than unity.
4. It can be used directly or converted to U^{233} in a power reactor so as to start off a thorium fuel cycle.
5. It can be used to fuel a new fast reactor.

PROBLEMS OF RECYCLING PLUTONIUM

Since achievement of high burn-up of natural uranium necessarily involves some plutonium recycling, it is desirable to point out some of the difficulties. Firstly there is a problem of reinserting plutonium either mixed with the natural uranium feed, in which case fuel element fabrication must be under alpha-active conditions, or separately when a low neutron absorber must be used as diluent for the plutonium, and a rapid fall off of heat production in plutonium fuel elements must be allowed for during their life.

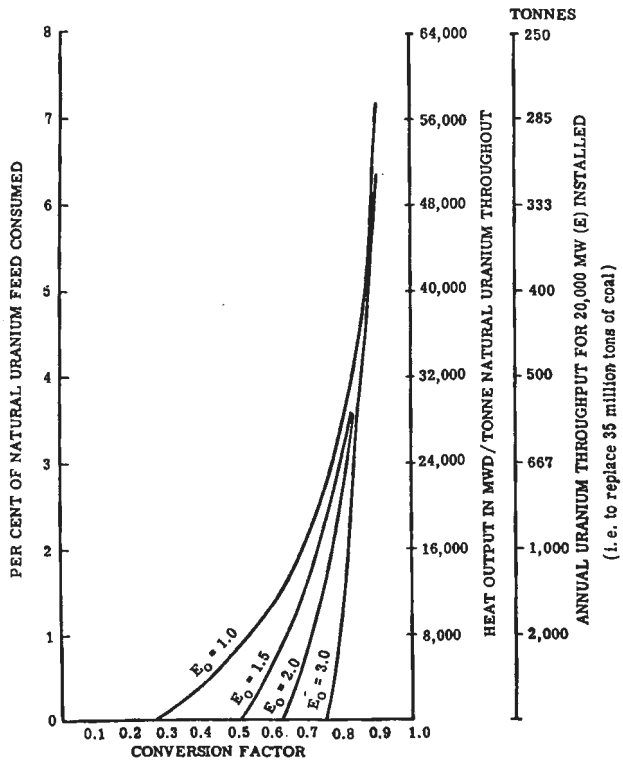

Figure 3. Natural uranium fuel requirement vs conversion factor

It should be noted that it is not necessary to process the natural uranium fed to a reactor on the equilibrium cycle shown in Fig. 1, until the uranium is rejected finally unless metallurgical considerations make it essential. Whether fission product poisoning would make earlier processing worth-while, economically, remains to be settled.

Finally, the dependence of plutonium absorption cross-section on neutron energy in the thermal region is such as to lead to positive temperature coefficients of reactivity in respect of moderator temperature (assuming no compensating density changes).

REJECTION OF FINAL PLUTONIUM

An alternative method of operating an "equilibrium fuel cycle" is to reject the plutonium left in the uranium fuel elements when they are removed from the pile. In this case, it will be necessary to have a feed of U^{235} from a diffusion plant to make up for the Pu rejected. Since diffusion plants make no use of U^{238} and extract only a proportion of the U^{235}, the over-all utilisation of uranium by reactor and diffusion plant will be much less than if the plutonium is recycled.

POSSIBLE POWER-PRODUCING REACTORS

Immediately after the war some consideration was given in the United Kingdom to natural uranium power reactors as a result of which the gas-cooled graphite moderated systems were considered in 1947. However, as it was not possible to consider a reactor of this type for power generation at that time, a small long-range effort was directed rather into systems which at the very least could be self-maintaining in fissile material. A fuel cycle based on the use of thorium was believed to be possible with thermal reactors, whilst a fuel cycle based on the use of uranium-238 was believed to require development of a fast reactor. It was considered essential to achieve high ratings to avoid the necessity of a large initial supply of fissile material or a large capital investment of fertile material (i.e., thorium or U^{238}).

About 1950 as effort became available and as the supply of uranium did not seem likely to present such a serious problem as had originally been thought, attention was directed strongly to natural or slightly enriched uranium systems. As far as was known, only four atoms could be used as moderating elements.

Of these, light hydrogen seemed most likely to be used in the form of water though molten caustic soda and organic liquids were considered as alternatives.

Heavy hydrogen seemed likely to be used as heavy water only. There seemed no reason to consider heavy caustic soda (NaOD) whilst the preparation of heavy hydrogen organic liquids seemed likely to be too expensive bearing in mind their short life in the reactor due to radiation destruction.

Beryllium seemed likely as metal or oxide.

Carbon would presumably be used in some form of graphite.

As regards coolants, there appeared to be gases, hydrogen containing liquids and sodium. Other molten metals were considered a long term possibility only. The same applied to fluidization techniques or homogeneous reactors.

The following table was drawn up to indicate likely short term possibilities.

	Moderating atom:	*Be*	*C*	*D*	*H*
Coolant:	Gas	1	5	9	13
	D_2O	2	6	10	14
	H_2O (or alternative liquid)	3	7	11	15
	Na	4	8	12	16

Beryllium and beryllia are comparatively inert chemically and resistant to high temperature and radiation. However, the cost of Be and BeO is such that high rating and thermodynamic efficiency are essential to make them economically acceptable in power reactors and so system 4 was preferred to systems 2 and 3. It seemed doubtful if adequate space could be found for gas cooling due to the small lattice pitch, but system 1 was not ruled out entirely.

The high temperature possibilities associated with gas and sodium cooling make systems 5 and 8 preferred for the graphite moderated systems. The possibility of using a gas turbine has been borne in mind either directly in system 5 or indirectly in system 8.

With water as a moderator, gas-cooling may have some attractions, but maximum permissible heat ratings seemed unlikely to approach those of systems 10 and 15, which represent a simpler engineering approach. There seemed little in favour of 11 or 14, whilst safety considerations did not favour 12 and 16. The use of an organic liquid or molten caustic soda could be considered in systems 15 and 16.

While these arguments were purely qualitative and applied only to the more immediate heterogeneous reactor for large power stations, they served to narrow the large field of possibilities for a first survey.

A further narrowing of early candidates for the UK reactor programme followed.

There seemed no advantages in pushing forward immediately on beryllium or beryllia.

The UK had no large scale heavy-water production programme. Also it saw little point in hurrying forward until a cheaper manufacturing process could be obtained bearing in mind that nearly 1 ton of water is needed in a pressurised heavy water moderated and cooled power reactor for each installed megawatt of electrical capacity.

One was then left firstly with gas cooled graphite reactors which had the great advantage that they could be designed so that they needed no initial enrichment. Sodium cooling could be an important development as a subsequent step.

Secondly it seemed clear one should develop a light water moderated and cooled reactor if reactor physics studies confirmed the theoretical predictions that such reactors had a good neutron economy. A

further corollary was that the use of an organic liquid as moderator should also be considered but serious consideration was not given to molten caustic soda.

DESIGN FOR NEUTRON ECONOMY

The accident of nature by which it is possible to make a reactor work with natural uranium and the high cost of making fissile material has led to a tendency to assume that it is a good thing economically to design reactors to use natural uranium. This may result in an unnecessary loss of neutrons in the moderator, or by leakage from the surface of the system in such a way that the neutrons cannot be absorbed usefully. Neutrons have an economic value since they can be used to convert thorium or uranium-238 into nuclear fuels. For this reason it may pay to design reactors so that these neutrons are not lost even if the reactor system must then use an enriched fuel. Reducing the lattice pitch of a graphite reactor is a clear example of this. In this case neutrons otherwise lost in the moderator go into U^{238}. However, due to an increase in neutrons captured by resonance absorption, the conversion factor becomes so high that the reactor may not go critical unless the proportion of U^{238} to U^{235} is decreased. A further point is that fast fission of U^{238} can lead to a useful gain of neutrons, if there is a closely spaced lattice, but the effect is unfortunately much less with thorium. This sort of consideration is what is meant in this paper by neutron economy.

It is obvious that the aim should be to reduce parasitic absorption by appropriate selection of low cross-section moderators, canning materials, coolants and by reductions in the quantities of these materials to the minimum compatible with engineering requirements. Low leakage of neutrons leads one to build as large a system as economic considerations permit. For a given size of system the neutron leakage will tend to decrease as the moderator-uranium ratio is decreased, reach some minimum value, then increase again. Also, as moderator-uranium ratio decreases the capture in U^{238} increases with respect to U^{235} and some compromise has to be made between the conversion factor and the degree of enrichment required to start the reactor. Further studies on the use of plutonium as a means of enrichment may lead to important modifications of these statements at a later date.

CONVERSION FACTORS ATTAINABLE

There is no definite indication at present of one of the selected systems above yielding a better conversion factor than another. All seem capable of achieving a conversion factor of 0.9 initially provided enriched uranium is used in the fuel. However, long irradiation of uranium and the formation of plutonium such as was indicated in the discussion above may lead to a reduction in conversion factor due to fission product poisons or perhaps fewer neutrons being available per fissile nucleus destroyed, but information on this point is still inadequate to be precise.

A more precise estimate of effective conversion factor in a practical case is governed by economic considerations associated with engineering design and the loss of fissile material in processing plants.

USE OF THORIUM

It seems likely that the conversion factor of a thermal reactor using the thorium-U^{233} fuel cycle will be significantly greater than that for a fuel cycle based on U^{235}, Pu and U^{238}. In consequence, one may expect that the large heterogeneous reactor may be self-maintaining when operating on thorium or at any rate nearly so. However, even if there is a negative gain factor it may still be economic to use thorium by having a continuous feed of Pu or U^{235} and rejecting uranium at a higher U^{238} concentration than would be the case on an equilibrium fuel cycle. Some speculative data are given in another paper to the conference[1] and whilst no precise conclusion has been reached it looks as though a negative gain factor of even 20% could be tolerated.

The important point, however, is that the economic dividing line between using thorium, and not using it, is not that associated with the thorium-U^{233} fuel cycle being self-maintaining.

FUTURE DEVELOPMENTS

Future high output reactor developments seem likely to be associated with two distinct types of system. On the one hand there is a thermal or near-thermal reactor in which the fuel is used in some form of homogeneous or semi-homogeneous arrangement. It seems likely that chemical and metallurgical considerations as well as nuclear ones will favour the thorium-U^{233} fuel cycle and that such a reactor will be about self-maintaining in fissile material.

The alternative route is the fast or slightly moderated fast reactor which can lead to very high gain factors using the plutonium-U^{238} fuel cycle, and a rather lower one using the thorium-U^{233} cycle. In both cases some sacrifice of gain factor is likely to ease engineering and processing problems and result in a more economic system.

ACKNOWLEDGEMENTS

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Survey of Fuel Cycles and Reactor Types

By Alvin M. Weinberg,* USA

The development of nuclear power and of fossil hydrocarbon fuel both have this in common: there are a variety of possible fuels, a variety of devices in which they can be burned, and a variety of uses to which the energy can be put. Because the development of hydrocarbon fuel utilization has been so gradual, it has hardly been necessary or even relevant to consider the comparative merits of coal, oil, or natural gas. Each is used where it is available; and practical, economic considerations determine the relative usefulness of the fuels in each situation.

Eventually, this will be the case with nuclear energy. Both raw materials, Th and U, will be used; questions of relative advantage will be settled by experience. Because the nuclear fuels are so much less bulky than fossil fuels, it is unlikely in the long run that one fuel will prevail over another simply because of local availability.

In the initial period, since a start must be made somewhere, it is worth-while to compare the two nuclear raw materials. This comparison will be based on the nuclear characteristics and on the implied engineering characteristics of systems based on U and on Th.

For countries which possess no diffusion plant facilities for re-enriching slightly used fuel, the problem of which system to start with is automatically answered: natural uranium must be the raw material. But this is a temporary situation. The plutonium produced in the natural U reactor will be burned in enriched reactors, and it soon becomes a question whether to use the excess neutrons to manufacture U^{233} or to manufacture more Pu^{239}. It will be my purpose to indicate some of the issues which determine this choice.

RAW MATERIALS

The total amount of Th in the lithosphere seems to be three times as great as the total amount of uranium.[1] Thorium exists only in the +4 valence state; almost all of its salts are insoluble. Uranium is ordinarily oxidized to the generally more soluble U^{6+}. As a result, relatively little of the earth's Th has been leached out of the original igneous rocks, while larger amounts of U^{6+} have been washed out of these rocks. Thus, although there may be more Th on earth, there is some informed opinion that workable deposits of Th may be less common than workable deposits of uranium.

The average Th content of the lithosphere is estimated to be 10 grams per ton. The electrical energy content at 25% thermal efficiency, and assuming complete burning, of 10 grams of Th, is about 60,000 kilowatt-hours. If the allowable fuel cost, exclusive of chemical processing, as 1 mill per kilowatt-hour, then the thorium in each ton of rock would be worth $60, while the uranium is worth $20!

NUCLEAR CONSIDERATIONS

Neutron economy, i.e., conversion efficiency from one fissionable isotope into another, is important for both the highly enriched breeders and the slightly or un-enriched converters. The importance of neutron economy in the breeders is obvious. In the converters, especially the heterogeneous converters, the main economic question centers around fuel burn-up: how long can the fuel element run without requiring reprocessing? It is usually considered that this requires 0.3% of all the atoms in the fuel element be burned before reprocessing; this amounts to a fuel cost of about 1 mill per kilowatt hour if fabricated fuel costs $25 per kilogram. As the reactor converts one kind of fissionable material into another, its reactivity may rise or fall depending on a sensitive balance between the nuclear properties of the various fissionable isotopes and the fission product poisons.

Number of Neutrons Produced in Neutron Reactions

The most important nuclear quantities in determining the long-term usefulness of a nuclear fuel are ν, the number of neutrons per fission, and η, the number of neutrons produced per neutron absorbed. The values of ν, η and α (the ratio of capture to fission) at thermal energy and high energies for the fissionable isotopes are summarized in Table I. The values of α are taken from boron-filtered measurements done at Knolls Atomic Power Laboratory, with theoretical interpolation between 10^4 and 2×10^5 volts. It is seen that the non-fission capture probability decreases as energy increases, but that it does not become extremely small in either U^{235} or Pu until about 5×10^5 volts. At this energy ν and η are almost equal.

Of the fissionable isotopes, Pu is seen to have the highest ν but the lowest thermal η, U^{233} a lower ν but the highest thermal η. Since the theoretical breeding gain is $\eta - 2$, the table shows that the U^{233}-Th cycle

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is the better possibility for thermal breeding, while the Pu^{239}-U cycle is the better possibility for fast neutron breeding. The possible improvement in $\eta(U^{233})$ by going from thermal to high energy,— i.e., the difference between the thermal η and ν is only about 10%, while in Pu it is almost 40%. Thus relatively little neutron economy advantage is to be gained in going to a fast breeder in the U^{233}-Th cycle; much more is gained in the Pu^{239}-U cycle.

What has been said so far ignores the possibility of the fast fission contributions in U^{238} and Th^{232}. The fast effect in the U^{238} system with fission neutrons is estimated to be about five times as large as in the Th system. In a heterogeneous, very closely packed lattice of natural or slightly enriched U it is estimated that this might increase the effective number of neutrons produced per Pu^{239} or U^{235} atom destroyed by as much as 10%; in the corresponding U^{233}-Th system this would be only about 2%. Thus for very closely packed heterogeneous systems the neutron economy of the Pu^{239}-U^{238} system may be about as good as in the U^{233}-Th system.

Table I. Value of ν, η, and α

	U^{238}	U^{235}	Pu^{239}
ν	2.54	2.46	2.88
η (thermal)	2.31	2.08	2.03
$\alpha = \frac{\sigma_c}{\sigma_f} = \frac{\nu}{\eta} - 1$ (thermal)	0.098	0.184	0.42
α (100 ev)		$\sim$ .52	$\sim$ .72
α (1000 ev)		$\sim$ .48	$\sim$ .60
α (10,000 ev)		$\sim$ .35	$\sim$ .43
α (100,000 ev)		$\sim$ .13	$\sim$ .18
α (500,000 ev)		$\sim <$ 0.1	$\sim <$ 0.1

The situation is less clear in the resonance region. Measurements of α for Pu^{239} are reported in papers in Session 18A. The results are tabulated in Table I. It is seen there that the $\eta(Pu^{239})$ drops disastrously immediately above thermal energy, and that even as high as 100 kev it is still significantly below the very fast neutron value of $\backsim$ 2.88. Less is known about U^{233}; the initial measurements show $\eta(233)$ to be pretty constant immediately above thermal energy.

The energy referred to is of course the neutron energy; in a thermal, moderated reactor this would mean essentially the temperature of the moderator. In a reactor like the Pressurized Water Reactor or the Aqueous Homogeneous Reactor in which the moderator runs at 250°C, the value of $\eta(Pu^{239})$ will have dropped well below 2; thus even with a high fast effect the likelihood of achieving a self-sustaining breeding cycle in such a system appears remote. Of course this might be improved, at the expense of engineering complication, by separating the moderator and cooling water. If the moderator were kept cool, the neutron temperature might be low enough to allow a relatively high η in the U^{238}-Pu system.

Fission Product Poisoning

The economics of all power breeder reactors is very strongly influenced by the frequency of chemical recycling. In heterogeneous reactors this may well be determined by radiation damage to fuel elements. Should fuel elements be developed which can sustain very high burn-ups, then the limiting factor becomes the neutron loss to the fission products and to the higher isotopes. The fission product losses in the resonance, fast, and thermal regions and to the higher isotopes like Pa^{233} or Pu^{240} will be discussed separately.

A. Resonance Region

E. P. Wigner has made the following estimate of the poisoning due to fission products in the resonance region. On the assumption that the reduced neutron width is proportional to the level spacing, it is possible to arrive at a definite expression for the average absorption cross section of a nucleus in the non-thermal region. If we measure the energy E, the radiation width Γ_r, and the level distance D in electron volts, the average capture cross section in barns becomes:

$$\sigma = \frac{1800\, f}{\sqrt{E}} \frac{\Gamma_r}{\Gamma_r + 4.4 \times 10^{-4} f D \sqrt{E}} \left(1 + \frac{A^{2/3} E}{10^7}\right) \text{ barns} \qquad (1)$$

where f is the ratio between reduced neutron width $\Gamma_n/\sqrt{E}$ and level spacing, i.e.,

$$\frac{\Gamma_n}{D\sqrt{E}} = 4.4 \times 10^{-4} f$$

and A is the mass number. The $A^{2/3}$ term in (1) takes care, approximately, of the higher angular momenta. A similar formula should hold for the fission cross section with the fission width Γ_f replacing Γ_r.

Recent data of Carter, Harvey, Hughes, and Pilcher[2] show that for nuclei in the light fragment group, f lies between 0.05 and 0.2; for the heavy fragments f is about twice as large, although the fluctuations in f are quite irregular. The radiation width is about 0.15 volt for the light fission fragments, about 0.1 volt for the heavy ones. These values are still uncertain by at least 50%; nevertheless they may be used to indicate the general character of the fission product poisoning.

According to (1) the *average* cross section is first proportional to $1/\sqrt{E}$ and should, in this region, have nearly the same value for every element. At a higher energy, when the second term in the denominator becomes larger than the first, the cross section should decrease as $1/E$; finally at very high energy (1 Mev) the average absorption cross section approaches the constant value

$$\sigma \sim 0.4\, A^{2/3}\, \Gamma_r/D \qquad (2)$$

The fission width is considerably greater than the radiation width; for this reason the fission cross section does not show the same strong $1/E$ high resonance region dependence.

The $1/E$ dependence of the fission product absorption sets in where Γ_r and the neutron width $\Gamma_n = 4.4 \times 10^{-4} f D \sqrt{E}$ become equal. For the light fragments, for which $D \sim 100$ ev, this would occur at about 400 ev, while for the heavy fragments, $D \sim 10$ ev, this occurs around 3000 ev. Of course these values are very rough, especially since many of the fission products are magic or near magic nuclei; for these D is larger, and the $1/E$ region sets in at even lower values. Nevertheless, this crude calculation indicates that fission product poisoning in a resonance reactor is less important above a few kilovolts than below this energy.

Unfortunately the available information on η in this low resonance region is not very encouraging. Experiments summarized in Table I show that η in Pu^{239} probably does not exceed 2 at around 10 kilovolts and is significantly less than 2 below 10 kilovolts.

Calculations of the average cross sections of light ($A = 100$) and heavy ($A = 140$) fission products, based on (1) are summarized in Table II. The column labeled $\sigma_F(U^{235})$ is an average value of the fission cross section of U^{235} at the indicated energy. The last column, labeled $(\bar{\sigma}_{a_L} + \bar{\sigma}_{a_H})/\bar{\sigma}_F$ is the ratio of the sum of light and heavy fission product cross sections computed by (1) to the fission cross section of U^{235}. It is this ratio which is a measure of the seriousness of the fission product poisoning in a resonance reactor.

Table II. Average Resonance Capture Cross-Sections

E ev	$\bar{\sigma}_{a_L}$ $(A=100)$ barns	$\bar{\sigma}_{a_H}$ $(A=140)$ barns	$\bar{\sigma}_F$ (U^{235}) barns	$\bar{\sigma}_{a_L} + \bar{\sigma}_{a_H}$	$\bar{\sigma}_F$
10^2	20	60	25	3.2	
10^3	3.7	14	7	2.6	
10^4	.5	2.6	3.8	0.8	
10^5	.07	.4	1.8	.26	
10^6	.019	.14	1.3	.12	

Parameters: For $A = 100$, $f = 0.17$, $\Gamma_r = 0.15$ ev, $D = 100$ ev
For $A = 140$, $f = 0.4$, $\Gamma_r = 0.10$ ev, $D = 10$ ev

The resonance poisoning according to the Table II is very serious below 10 kev. For example, at 10 kev, if 10% of the fuel is burned the fission products poison cross section would amount to 8% of the fuel cross section.

Thus the outlook for a high burnup resonance reactor, let alone a resonance breeder, is fairly poor unless the fission products are removed rapidly or the spectrum is kept high.

The situation is better above 10 kev: here the $1/E$, rather than $1/\sqrt{E}$, fission product cross section variation is well established, while the fission cross section falls much more slowly. Thus at 10^5 ev the fission product poisons are only one-third as important as they are at 10^4 ev.

It should be stressed that the above estimates are based on average parameters in which there is considerable uncertainty. For example, the data are also consistent with a value of f for the light fragments only one-half the value used here. This would make the light fragment poisoning less important, but since the heavy fragments cause most of the poisoning anyhow, this does not change our general conclusions.

B. Fast Region

In the very fast region the asymptotic formula (2) gives for the light fission product absorption cross sections about 12 mb and about 100 mb for the heavier fragments. These values are based on the same parameters as were used for the resonance calculation. Hence per fission there will be produced about 110 mb of poison in a fast reactor. The fission cross sections of U^{233} and Pu^{239} for fast neutrons are about 2 barns.

Now consider a very fast reactor which runs at a specific power of 1000 kilowatts per kilogram so that 0.1% of the fissionable material is burned each day. The ratio of fission product poison cross section to fission cross section increases at the rate of $0.1 \times 10^{-2} \times 0.11/2 \approx 0.6 \times 10^{-4}$ per day. The fission product poison would amount to 0.6% in 100 days. This is hardly serious since by this time 10% of the fissionable material would be destroyed and radiation damage to fuel elements would probably require purification and refabrication.

The estimate just given refers to an ideal fast reactor in which the fission spectrum has not been degraded. In any actual fast reactor, because of inelastic scattering the bulk of the fissions are likely to occur at energy between 10^5 and 5×10^5 ev. In this energy range the fission product poisoning will be more serious. According to Table II, each fission product pair will have a cross section which is about 0.2 of the fission cross section. Thus the poisoning in 100 days, in the above example, would amount to about 2%.

C. Thermal Region

The fast and resonance fission product absorption can be estimated statistically on the basis of a simple formula like (1). Moreover the cross sections are hardly expected to change as the fission product decay chain is traversed, and so the poisoning in these cases is not expected to depend on the age of the fission products. The thermal fission product cross sections, on the other hand, are not amenable to such theoretical treatment; they can be obtained only from available experimental data. In addition, since the thermal absorption cross sections of adjacent nuclides may be grossly different, the poisoning effect will depend sensitively on the fission product age.

The thermal fission product poisons may be divided into three groups. Group I comprises the rare gases, almost entirely Xe^{135}, which in a homogeneous re-

actor can be removed continuously. The equilibrium loss to Xe^{135} is essentially its fission yield divided by $1 + \alpha$; this is about 6% in both the U^{235} and U^{233} systems. The equilibrium loss is realized when the rate of destruction caused by neutron absorption is large compared with the natural decay rate. This occurs at flux levels substantially above 3×10^{13}.

Group II comprises those non-gaseous stable fission products having cross sections greater than 1000 barns, which reach equilibrium in a few days. All of these fission products are rare earths except Cd^{113}. The poisoning by these fission products is also given by their total fission yields divided by $1 + \alpha$. In U^{233} the Group II poisons contribute about 0.77%, in U^{235} about 1.5%.

Group III comprises all of the remaining stable and unstable fission products for which cross section data are available. In U^{233} Group III fission products have a yield of 126% out of a possible 200%, in U^{235} a yield of 144%, in Pu^{239} a yield of 134%. The cross sections of these fission products are time dependent.

Thirteen isotopes, with a total yield of about 45%, account for almost all of the Group III poisons. These are, with their cross sections, Nd^{143} (240 b), Tc^{99} (100 b), Sr^{89} (110 b), Cs^{133} (26 b), I^{131} (600 b, 8-day half-life), Nd^{145} (50 b), Rh^{103} (149 b), Cs^{135} (15 b), Mo^{95} (13 b), Pm^{147} (60 b), Pr^{141} (13 b), La^{139} (8 b), Eu^{153} (400 b), Sm^{152} (135 b).

The total average cross sections of the Group III fission products as a function of irradiation time at a flux level of 3.2×10^{14} have been computed by E. D. Arnold of the Oak Ridge National Laboratory and are given in Fig. 1. The cross sections of Group III initially are about $\sigma = 40$ barns per fission for U^{235}, about 32 barns per fission for U^{233}; after 110 days at 3.2×10^{14} flux these values fall to about 16 barns per fission and 12 barns per fission, respectively. For Pu^{239} a compilation of K. Way and Nancy Dismuke gives an initial cross section of 44 barns, and the long-term value is about 18 barns per fission. Most of the fission products of Group III have such low cross sections that they do not saturate; hence the average poison cross section per fission of most of the stable isotopes in the group is independent of irradiation time. On the other hand, I^{131} and Sr^{89} have relatively short half-lives and do saturate. Almost all of the deviations from constancy in Fig. 1 at low irradiation time are attributable to these two nuclides. For practical calculations a constant fission product loss of 18 barns per fission in U^{233}, 16 barns per fission in U^{235}, can be used with good accuracy except at very short irradiation times.

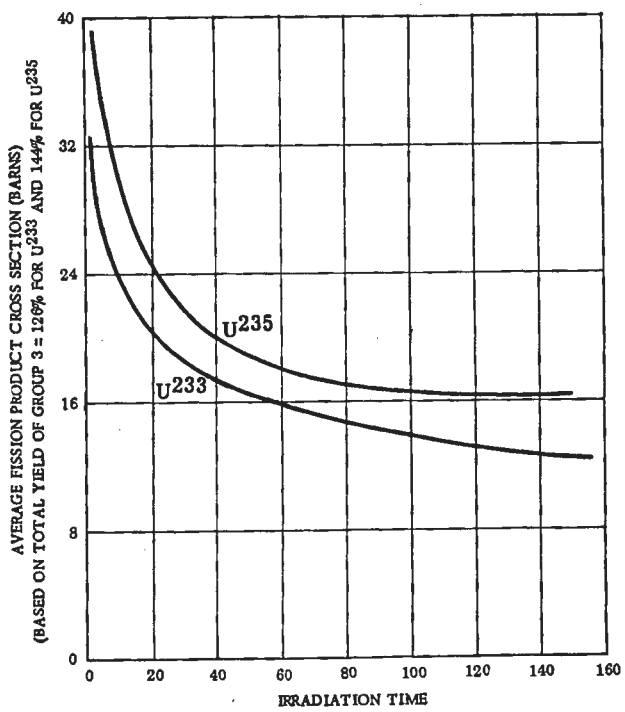

Figure 1. Effect of irradiation time on group-3 fission product poisons in U^{233} and U^{235} systems at flux level of 3.2×10^{14} neutrons/cm²-sec

The Group III fission products are of some significance in any high burnup thermal reactor. For example, in an MTR or homogeneous-type reactor operating at specific power of 10,000 kilowatts per kilogram, the fission product poisons would reduce the conversion ratio by roughly 3×10^{-4} per day. In a natural uranium reactor the reduction in thermal utilization caused by Group III fission product poisons by the time the reactor has burned 0.3% of the original atoms—i.e., 50% of the original U^{235}—would be about 0.8%. This loss increases roughly linearly with time and becomes 2.5% by the time 1% of all the atoms have been burned.

Of course there are many unknown cross sections left out in estimating the Group III fission product effects, although it is usually believed that the missing cross sections are unknown precisely because they are either small or belong to short half-lived or low-yield nuclides, in which case they would not be important poisons.

It is interesting to note that the ratio of fission cross section to Group III fission product cross section is, asymptotically, about 30 in a thermal U^{233} or U^{235} reactor. This is a little larger than the ratio of about 20 which prevails in a very fast reactor. Thus the Group III fission products are hardly as serious in a thermal as in a fast reactor, and are much less serious than in the low resonance energy reactors. Of course in the thermal systems the loss to the Group I and II fission products must be added but these losses saturate quickly.

HEAVY ISOTOPE LOSSES

In addition to the fission product poisons, there are built up in the reactor heavy intermediate isotopes which often are serious poisons. The most important known slow neutron capture cross sections are summarized in Table III.

The relative importance of the heavy isotope poisons depends on the reactor system. In the thermal Th-U

Table III. Important Thermal Neutron Capture Cross-Sections

Isotope	*σ_c barns*
Pa^{233}	~150
U^{234}	92
U^{236}	6
Np^{239}	~ 80
Pu^{240}	510

system, Pa^{233} with a capture cross section of more than 100 barns is the most serious poison. It is particularly troublesome since capture in Pa^{233} removes 2 neutrons from the breeding cycle—one being the neutron which formed the Pa^{233}, the other the neutron captured by the Pa^{233}.

Since Pa^{233} has a 27-day half-life, the loss due to Pa^{233} capture can be kept low by keeping the Pa always in a relatively low flux. Thus at 2×10^{13} flux the Pa half-life against capture is 100 times as long as its half-life against decay, while at 2×10^{15} the two half-lives are about equal. In the former case the Pa loss will be negligible; in the latter, fully half of the bred U^{233} will be destroyed by neutron capture in Pa.

In the thermal, natural, or slightly enriched uranium systems it is the higher isotopes of U and of Pu which are most important. Since the thermal fission cross section of Pu^{239} (~ 800 b) is considerably larger than the fission cross section of U^{235}, the tendency initially in thermal converters is for the reactivity to increase as the U^{235} is replaced by Pu^{239}. With further irradiation the strongly poisoning Pa^{240} grows in; this, along with the accumulation of fission products, U^{236} and burnup of the Pu^{239} itself, has the effect of reducing the reactivity at long irradiation times.

The relative gain conferred by the growth of the Pu^{239} depends upon the conversion ratio: the higher the conversion ratio, the more pronounced will be the initial reactivity gain. The conversion ratio is usually lower in slightly enriched reactors than it is in natural uranium reactors. Thus the engineering flexibility made possible by using slightly enriched instead of natural uranium tends to be balanced by the lower burn-ups which, in general, are possible with the lower conversion ratio, slightly enriched systems.

While the situation is too complicated to summarize in detail, it appears likely that reactivity losses would not make burnups of 3000 megawatt days per ton impossible in many thermal reactor systems.

In general, the rate of manufacture of heavy isotopes and the consequent poisoning due to them is proportional to α, the capture-to-fission ratio. Since α is so small in the very fast reactors, heavy isotope buildup in these should be less important than in thermal reactors. In the resonance systems, characterized by very large α, heavy isotope buildup will be much more serious. An estimate of the average resonance cross sections of the heavy isotopes can be made from equation (1). The heavy isotopes are characterized by fairly small D, small Γ_γ and fairly large f. Taking, for example, $D = 10$ ev, $\Gamma_\gamma = 0.03$ ev, $f = 0.4$, it is found that the average heavy isotope cross section is about equal to that of the heavy fission fragment cross sections. Thus as far as poisoning is concerned the heavy isotopes may be included with the heavy fission fragments, and their "fission" yield is of the order $\alpha/(1 + \alpha)$. Thus for $\alpha \approx \frac{1}{2}$, value characterizing the low resonance energy reactors, the heavy isotopes add about 30% to the poisoning due to the heavy fission fragments.

SUMMARY OF NUCLEAR CHARACTERISTICS

We can summarize the nuclear situation very briefly in the following statements:

1. Thermal breeding in the U^{233}-Th cycle has a fairly good outlook, with breeding gains of 10% to 15% probably being achievable.
2. The homogeneous breeder is not subject to the 6% Xe^{135} poison loss; if fission products can be removed continuously, the fission product loss would be further reduced.
3. Fast breeding in the Pu^{239}-U cycle has a good outlook; breeding gains of 0.5 ought to be achievable. To achieve this, however, the spectrum must be kept well above 0.1 Mev.
4. Fast breeding in the U^{233}-Th cycle, while possible, has little nuclear advantage over the slow U^{233}-Th cycle.
5. The probability of successful breeding in the resonance region is low, although not enough is known about η in this region to be certain of this. Fission product poisoning is worst in the low resonance region.
6. Reactivity losses are probably tolerable at burnups of 3000 megawatt days per ton in natural or slightly enriched uranium systems if the design is such as to give a large initial conversion ratio.

ENGINEERING CONSIDERATIONS

The enormous difficulty of choosing a proper path for reactor development is easily seen by estimating the number of conceivable reactor types. In Table IV are listed some of the choices which are more or less at the disposal of the reactor designer.

Table IV by no means lists all possible choices; yet, the total number of combinations in Table IV is $3 \times 2 \times 3 \times 5 \times 2 \times 5 = 900$ possible reactors! Of course not all of these are sensible; for example, a fast reactor could hardly be cooled with H_2O. Even so, there are probably at least 100 combinations which are not obviously unfeasible. The central issue in reactor development is to trace out of this welter of possibilities the dozen or so which are most likely to succeed.

Fortunately, the nuclear considerations already outlined greatly reduce the number of possibilities. The slow neutron breeding cycle almost certainly requires

Table IV. Choices to Be Made in Reactor Design

Fuel	*Fertile material*	*Neutron energy*	*Coolant*	*Geometry*	*Moderator*
U^{233}	Th	Fast	Gas	Heterogeneous	H_2O
U^{235}	U	Resonance	Liquid Metal	Homogeneous	D_2O
Pu^{239}		Slow	H_2O		Be
			D_2O		BeO
			Hydrocarbons		C
			etc.		etc.

U^{233}-Th; the fast neutron cycle can use either U^{233}-Th or Pu^{239}-U, but the latter has the nuclear advantage.

There are a few general principles which can be used to reduce further the number of possibilities. These will now be discussed, though not necessarily in order of importance.

Material Efficiency and Thermal Efficiency

If all else is equal, that reactor plant is best which operates at the highest thermal efficiency. However, since in each reactor there is always held up a significant amount of expensive material — either fissionable material or moderator — and this held-up material represents a continuing expense, it is also important to extract as much energy from a given amount of fuel as possible. The electrical energy extracted per unit amount of fuel or expensive moderator might be called the "material efficiency." In a chain reactor high material efficiency, as well as high thermal efficiency, leads to low over-all power cost.

As an example consider a water-moderated U^{233}-Th reactor which operates at thermal efficiency of 20% and fuel efficiency of 10,000 kilowatts per kilogram of U^{233} with, say, a liquid metal-cooled fast reactor which operates at thermal efficiency of 35% but fuel efficiency of only 1000 kilowatts per kilogram of Pu^{239}. The electrical output in the one case is 2000 kilowatts per kilogram of U^{233}, in the other case only 350 kilowatts per kilogram. It is quite possible that such a disparity in material efficiencies could overcome the economic advantage of the high breeding gain and high thermal efficiency in the fast system.

The economic use of very expensive heavy water finally depends on the extraordinarily high material efficiencies which it makes possible. For example, because D_2O absorbs so few neutrons, a chain reaction in a solution as dilute as one-half gram of U^{233} per liter of D_2O is quite possible. By careful attention to external equipment design a thermal power output of 20 kilowatts per liter of heavy water or 40,000 kilowatts per kilogram of U^{233} in a heavy water system is conceivable.

The choice between gas and liquid coolants in some measure hangs on this issue of material versus thermal efficiency. For example, the Brookhaven reactor operates at about 30,000 kilowatts with a loading of about 100 tons of natural uranium or a material efficiency of 300 kilowatts per ton of natural uranium. With high-pressure water cooling, this could be increased perhaps ten-fold. Thus, although the gas-cooled system may ultimately have a higher thermal efficiency, its material efficiency can hardly match that of the liquid-cooled systems. This fact largely accounts for the absence of gas-cooled systems in the US reactor development program.

Homogeneous and Heterogeneous Systems

The aqueous homogeneous system represents a final rationalization of the heterogeneous pressurized water systems: if the fuel elements are made smaller and smaller they finally culminate in a dispersion or solution; i.e., a homogeneous system.

The advantages of the homogeneous systems—simple chemical processing, good heat transfer, good nuclear control—must be weighed against the manifestly formidable problem of containing the radioactivity and the related problem of corrosion. From the point of view of thermal breeding, U^{233}-Th homogeneous systems have considerable advantage over the heterogeneous ones. Certainly the 6% Xe loss is eliminated, and there is a good chance that the remaining fission product poisons can be removed before they become serious. The only nuclear advantage the heterogeneous systems have is the higher fast effect. To achieve this, however, requires very close lattice spacing and high U^{238} or Th-to-fuel ratio. In a power reactor like PWR (Pressurized Water Reactor) the close packing would make awkward any attempt to segregate coolant and moderator, and this would then lead, in the Pu system, to the previously-mentioned difficulty of low $\eta(Pu^{239})$ at high temperature.

The advantage of the heterogeneous system—relatively secure containment of radioactivity—must be weighed against the likelihood of achieving high burnup before reprocessing. To see the magnitude of this problem consider a highly enriched MTR-type breeder in which a 20% burnup and 25% thermal efficiency is achieved. For every fissionable atom burned, five chemical reprocessings are required. This means that, in order for chemical reprocessing to add less than 2 mills per kilowatt hour to the cost of the electricity, a cost of not more than $2.40 per gram can be allowed for each chemical and metallurgical recycle. As the technology now stands it seems more likely that the chemical costs will reach this low value before the metallurgical costs do. It is basically the belief that the metallurgical costs, if

not the chemical ones, will always be substantial that has motivated the whole homogeneous development.

Fast Reactors versus Thermal Reactors

Since both the U^{233}-Th and the Pu^{239}-U cycles can be used as bases for a fast neutron breeding cycle, it may not be obvious why there should be any incentive to develop thermal U^{233}-Th breeders. The main reason of course is that thermal reactors are intrinsically more dilute than are fast reactors; hence the material efficiency of a thermal reactor will almost certainly always exceed the material efficiency of a fast reactor. The example of the MTR-type breeder compared with the fast breeder points up this. It may therefore well turn out that U^{233} will generally be burned in thermal neutron reactors even though a somewhat higher breeding gain is attainable in the fast reactor.

The engineering problems of the fast reactors center mostly around the fact that any diluent in the reactor tends to decrease the neutron energy while the high $\eta(Pu^{239})$ accrues only if the neutron energy is kept very high—of the order of several hundred kilovolts. Diluents in a fast breeder are of three kinds: structural material, coolant, and fertile material. Of the three, the fertile material, having the highest inelastic cross section, is most important in depressing the neutron spectrum. Yet there is strong incentive to have a large proportion of, say, U^{238} atoms to Pu^{239} atoms since the larger this ratio is, the longer such a reactor would operate before serious radiation damage sets in and reprocessing is required. For example, in some fast breeders a U^{238}-to-Pu^{239} atom ratio of 10:1 is contemplated. If the alloy requires reprocessing after 2% of all its atoms are destroyed, 20% of the Pu^{239} could be burned before each recycle. Thus five chemical and metallurgical recycles are required to burn one Pu^{239} atom, and the problem of high reprocessing costs must be faced. Of course the large breeding gain in the Pu^{239}-U^{238} system tends to offset the economic effect of the manifold reprocessings.

In much the same way there is a limit to the amount of structural and coolant material which the fast system can tolerate. In all cases the result is that the fast reactors always suffer from low material efficiency. Whether this deficiency can be balanced by the high breeding ratio it is too early to say.

Will one or two reactors emerge as unique choices? I think every worker in reactor design must have wondered whether, in the long run, any one reactor type will emerge as so distinctly superior to the others that it will render the rest obsolete. The history of hydrocarbon-burning devices suggests that the technology will develop a succession of "most desirable" types: the reciprocating steam engine was followed by the steam turbine—which may ultimately be replaced by the gas turbine. Within each class—say the steam turbine—there has been a tremendous development and corresponding high rate of obsolescence; for example, the heat rate on the most modern turbines is less than half the heat rate of turbines only 20 years old.

But the main reason for obsolescence of conventional power generating devices—low thermal efficiency—will hardly operate to render nuclear power plants obsolete. Rather nuclear plants ought to be much more like hydroelectric plants: if they have sufficiently low *over-all* operating costs, and this is a sum of costs determined by thermodynamic efficiency, material efficiency, maintenance, etc., then it is at least not obvious why they should become obsolete any more than dams become obsolete.

Thus in trying to choose the most likely lines of development, the aim will always be to reduce the operating costs. It is perhaps on this account that ingenuity in the method of heat removal as, e.g., in the direct boiling scheme, may prove ultimately to be less significant than ingenuity in fuel handling—that schemes which aim at low operating costs may be more important than schemes aimed at low capital costs.

What I have said refers mostly to the long run. In the meantime, with highly enriched material something of a rarity, it is inevitable that power will be extracted from the natural uranium systems in more or less conventional reactors. Yet once a reactor is constructed with natural uranium fuel rods, there will be strong nuclear incentive, after the first few loadings with ordinary uranium, to switch to the U^{233}-Th cycle. The strong possibility, even in the heterogeneous system, of making the U^{233}-Th thermal system self-sustaining will always be an attractive goal, no matter how plentiful natural uranium may be. Thus it may not really be relevant to argue whether breeders ought to be developed as soon as possible. It seems to me entirely likely that most heterogeneous power reactors which begin their lives as thermal, natural U converters will gradually convert to the U^{233}-Th cycle simply because operation with this fuel is probably independent of raw material.

In any event certainly two nuclear types, the slow cycle based on U^{233}-Th and the fast cycle based on Pu^{239}-U will become accepted reactor types. Beyond this—whether the reactors will be homogeneous or heterogeneous, whether the moderator ultimately will be D_2O, Be, C, etc., what the coolant will be—these are difficult issues whose outcome no one can foresee. It is of great advantage to our budding technology that it is being pursued—with such little overlapping—in so many different countries, for only out of such diversity in effort will come the answer to the question: "Which reactor type is best?"

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Record of Proceedings of Session 10A.2

FRIDAY AFTERNOON, 12 AUGUST 1955

Chairman: Mr. M. Oliphant (Australia)
Vice-Chairman: Mr. L. Kowarski (United Nations)
Scientific Secretaries: Messrs. D. J. Littler and J. Ulehla

PROGRAMME

P/4 Some economic aspects of nuclear fuel cycles.....W. B. Lewis
P/403 Fuel cycles and types of reactors.....J. V. Dunworth
DISCUSSION
P/862 Survey of fuel cycles and reactor types.....A. M. Weinberg
DISCUSSION

Mr. LEWIS (Canada) presented P/4, concluding with the following remarks:

The practical question as to whether a light water or a heavy water system would prove cheaper must be left open, but it seems that for the large power reactors considered, the heavy water reactor with recycled fuel promises a very attractively cheap range of fuel cycles.

I am suggesting that it is entirely reasonable and quite economically justifiable to utilize 3 per cent of the energy in uranium and to be satisfied with 25 per cent thermal to electrical energy conversion.

Because such considerations are unfamiliar and it is so much easier to seize upon an ideal—such as 100 per cent burn-up and 100 per cent efficiency—it goes against our greedy nature to accept this apparently low objective. I would like, therefore, in closing to put the case in another way.

To generate 1 megawatt-day of power, the station I propose consumes 4 gm of uranium and discards 133 gm of U^{238}. A hydro-electric plant operating with a 40 meter head of water generating 1 Mwd discards 240,000 tons of water. If this contained 1 part in 10^9 of uranium, and few waters contain less than this, it discards twice as much uranium as the nuclear plant. I do not want to follow this up to the limit but it is just an idea.

Again, to generate 1 Mwd by burning coal consumes about 10 tons of coal; a residue of 133 gm would be only 15 parts per million.

Moreover, this 133 gm of U^{238} discarded by the nuclear power station can be left in a neat stockpile for use by future generations if they need it.

Mr. J. V. DUNWORTH (UK) presented P/403 as follows: The elementary features of breeding with the U^{238} plutonium fuel cycle in a fast reactor is described in other papers to the Conference and detailed nuclear data is given therein. It seems fairly clear that the intermediate energy region U^{238}-plutonium fuel cycle and also the intermediate energy region U^{235}-U^{238} conversion cycle are of no great interest and some detailed measurements are given later in this session in Mr. Weinberg's paper.

I will go on, therefore, to consider the thermal fission energy U^{238} plutonium fuel cycle. The first point is that, as is well known, this is likely to have a negative gain factor in any practical system. However, it is possible to feed fresh fissile material into the system as natural uranium, and it is possible to reject U^{238} as depleted uranium. This results in a practical fuel cycle with some very curious features. At present, we do not have all the nuclear data necessary for evaluating this system. In this paper, attention is drawn to some basic features. In the other papers, Mr. Lewis of Canada has attempted to bring out a considerable amount of detail whilst Mr. Weinberg has endeavoured, in particular, to estimate the effect of fission product poisoning.

The first question, however, is how to commence the fuel cycle. Now, a nation starting an atomic energy programme without the help of any other nation must build either a diffusion plant or a reactor which can operate with a supply of natural uranium or, of course, both. If it builds a reactor it must use either a graphite or heavy water moderator. For large heat output, large amounts of heavy water are required, and the tendency is to build graphite piles. For a high flux experimental reactor very much less uranium is needed to obtain criticality with a heavy water system than is the case for a graphite system. Therefore, a high flux is obtained for a low total power output with a heavy water system. These simple facts have conditioned the atomic energy programmes of most nations to date.

It has long been appreciated that with natural uranium as a starting point, it might become possible to burn substantial quantities of U^{238} which occur in it. The fission of the U^{238} nuclei directly is possible only

to a small extent with the fast neutrons available in reactors, and so the sequence envisaged is by conversion to Pu^{239}, with subsequent fission of these nuclei. The possibility of extracting plutonium from the first charge and re-inserting with part of the depleted uranium in the second charge was recognized long ago and became of economic importance when the cost of extracting and using such plutonium became competitive with the price of U^{235}. It is a matter of historical interest that there was a British report, No. 7, written in 1941 on this subject by Mr. Halban and by the Vice-Chairman of this afternoon's session, Mr. Kowarski.

In a suitably designed reactor, this process eventually leads to a fuel cycle in which natural uranium only is supplied, highly depleted uranium is rejected, and plutonium is recycled through the reactor. This cycle, with a natural uranium fuel feed, is possible even if some degree of enrichment is necessary in the initial fuel charge.

It is of some interest to find the amount of U^{238} which can be burned and how it depends on initial enrichment and conversion factors.

Slide 1 (Fig. 1, P/403) shows the fuel cycle. Some of the reactions involved are:

$$U^{238} + n \longrightarrow U^{239} \longrightarrow Np^{239} \longrightarrow Pu^{239}$$

$$Pu^{239} + n \longrightarrow \eta_9 \text{ neutrons} + \alpha\, Pu^{240} + \text{fission products}$$

$$\alpha\, Pu^{240} + \alpha\, n \longrightarrow \alpha\, Pu^{241}$$

$$(\alpha\, Pu^{241} + \alpha\, n) \longrightarrow \alpha\, \eta_1 \text{ neutrons} + \text{some } Pu^{242} + \text{fission products}$$

where η_9 refers to Pu^{239}, η_1 refers to Pu^{241} and where α is defined by the equations.

U^{238} captures a neutron, going to Pu^{239}. When Pu^{239} absorbs a neutron, roughly one-third of the atoms undergo capture to Pu^{240} and the other two-thirds undergo fission. The Pu^{240} captures a neutron to form Pu^{241}. A considerable fraction undergoes fission; a smaller fraction leads to Pu^{242}. The latter can then be destroyed and removed from the system by the capture of a neutron to give Pu^{243} which decays to Am^{243}. If one adds up the number of neutrons in this cycle, one finds that almost as many are produced as are consumed. However, allowance must be made in any practical system for losses of neutrons parasitically in the reactor and also from the surface of the reactor. When allowance is made for this, the cycle, in fact, has a significant negative gain factor. However, in the centre of the pile and if there is very little parasitic absorption, the fuel cycle may be nearly self-maintaining, but this cannot apply for the system as a whole.

Reactors containing natural or enriched uranium are a starting point, and conditions which exist by recycling plutonium until equilibrium is reached can be estimated in theory from a knowledge of the effective cross-sections. Any attempt to do this at present leads one to extremely detailed and involved mathematics, and in this document I have adopted a highly simplified approach so as to bring out the main physical principles.

I am assuming that the reactor has an initial enrichment E (which is unity, of course) if one is using a system such as Calder Hall, which does not require an initial enrichment. I am assuming that the average level of the U^{235} in the reactor when the whole system has reached equilibrium is $\overline{E}$. I have supposed that the conversion factor is C, and that this is roughly unchanged when the U^{235} is replaced by plutonium. I will ignore, for the moment, Pu^{240} after its formation.

$$\overline{E}\,C + (E - \overline{E})\,C = E - \overline{E};\ \overline{E} = E(1 - C)$$

If the quantity of Pu^{239} in the reactor is to remain constant, then the plutonium formed from U^{235}, which is given roughly by the average level of U^{235} in the cycle times the conversion factor, plus the plutonium made from plutonium, which is proportional to the difference between initial enrichment and average enrichment times the conversion factor, will equal the plutonium destroyed, which is given by the initial enrichment minus the average enrichment. That simple formula then leads to the formula $\overline{E} = E(1 - C)$. However, I have ignored Pu^{240} and its fissile offspring, Pu^{241}. I will assume that η for Pu^{241} is the same as for U^{235}, and in this case, we may consider that Pu^{240} and Pu^{241} will replace an equivalent number of U^{238} atoms and fissile atoms.

If so, a little reflection will show that for each neutron absorbed in U^{238}, alpha neutrons are absorbed in Pu^{240} where alpha is the ratio of the capture in Pu^{239} to the total cross-section in Pu^{239}.

In this case, we arrive at a second formula in which, in the equilibrium cycle, the average level of the U^{235} content given by $\overline{E}$ equals $E(1 - C)\ (1+\alpha)$, where α is the cross-section for capture in Pu^{239} divided by the total cross-section. And that is quite a simple formula.

Such a reactor can be fueled in the equilibrium state with natural uranium if the average enrichment of the U^{235} in the reactor, $\overline{E}$, is less than unity. And we therefore have the important result that a reactor requiring an initial enrichment, E, can be operated subsequently on a feed of natural uranium if $\overline{E}$ is less than 1. If it is substantially less than 1, then we could even operate on a feed of depleted uranium, despite the fact that the reactor needed initial enrichment.

These simple arguments ignore fission-product poisoning. They ignore some of the details of Pu^{242} and the radioactive decay of Pu^{241}. Also, one has to consider, if one is going to do a detailed analysis, the many problems presented by plutonium resonances, fine structure in the lattice, the exact value of nuclear constants, the engineering design of the reactors, parasitic absorption, distribution of neutron flux in the reactor, and also, in a practical system, the losses in the processing plant. All of these introduce many complications into the cycle.

I will not go into the argument, but the reject concentration of the uranium in the fuel is given by

$$E_r\, e - \phi\, v\, \sigma_5 T$$

where $\phi\, v$ is the neutron flux in the reactor, σ_5 is the absorption cross section of U^{235} and T is the irradiation time.

For each U^{235} atom consumed, there is produced C, i.e., the conversion factor, atoms of Pu^{239}. If these remain in the system and are consumed, one gets C^2 produced, C^3, and so on. As a crude approximation, then, $C/(1 - C)$ atoms of U^{238} are consumed for each U^{235} atom consumed in the reactor.

Slide 2 (Fig. 3, P/403) has been prepared by Mr. Fenning and shows diagrammatically some of the results of these formulas for a reactor operating on an equilibrium cycle with a feed of natural uranium.

On the left-hand side is the percentage of the whole of the uranium consumed, which goes from 0 up to about 10 per cent at the top. The horizontal line is the conversion factor, going from 0 at the left to unity at the right. The vertical line on the right shows the heat output in megawatt days per ton for the natural uranium placed in the pile. The line on the right shows the annual uranium throughput if the whole of the United Kingdom electricity were provided by reactors running on such a fuel cycle. The figures there start at 250 tons at the top, coming down to about 2000 tons towards the bottom of the diagram, and the four curves correspond to building reactors with an initial enrichment of unity, initial enrichment 1.5, initial enrichment 2.0, and initial enrichment 3.0.

Slide 3 (Fig. 2, P/403) simply presents graphically the relationship between the equilibrium level of U^{235} in the fuel cycle, which goes from 0 up to unity in a horizontal direction, and the reject level of the uranium when it is finally removed from the system on the left, which goes from unity at the top to 0 at the bottom of a linear scale.

It will be apparent from what I have said, I think that the need to enrich a reactor fuel is not necessarily a drawback. For example, one may compare a system needing a natural uranium charge initially to give a conversion factor of 0.8 with systems designed to use twofold enrichment and a conversion factor of 0.9. The total burn-up of the natural uranium feed would be the same if the simplifying assumptions made earlier are not too much in error.

It is an interesting fact that a reactor requiring initial enrichment and then converted to an equilibrium feed cycle with a feed of natural uranium may still have a sizeable net output of plutonium without reducing the total fraction of U^{238} atoms consumed by too large a factor. The effect on the burn-up can be assessed readily by reducing the effective conversion factor of the system by the fraction of the Pu produced which is so diverted.

The diversion of plutonium can be made for several reasons: It can be used with natural uranium in lieu of slightly enriched uranium from a diffusion plant as a means of fueling a new reactor which requires an initial enrichment. It can be used in small mobile power plants in lieu of highly enriched uranium from a diffusion plant. It can be converted to U^{233} in a power reactor so as to provide make-up in case a thorium-U^{233} fuel cycle is used in a heterogeneous reactor where the over-all conversion is less than unity. The plutonium can also be used directly or converted to U^{233} in a power reactor so as to start off a thorium fuel cycle. It can be used to fuel a fast reactor.

It is perhaps desirable to point out some of the difficulties of recycling plutonium. There is the problem of reinserting plutonium mixed with a natural uranium feed, because in this case fuel element fabrication must be under alpha-active conditions, though possibly remote control methods of making fuel elements may prove the cheapest in the long run.

If it is used separately from the uranium, the plutonium must have a diluent and, also, the production of heat in the plutonium fuel element will fall off rapidly during its life in the pile; allowance must be made for this in the engineering design.

It should be noted that, in theory, in the fuel cycle I have mentioned, it is not essential to process the natural uranium feed to a reactor until the uranium is finally rejected, unless metallurgical considerations make it essential. I shall leave to later speakers a discussion of the question whether fission product poisoning would demand this on economical grounds.

I must draw attention to the fact that the plutonium absorption cross-section increases in an unfavourable way as the neutron energy increases. It is possible that the η for plutonium falls rather sharply as one begins to move out of the thermal region by increasing the moderator temperature, for instance. This must be allowed for in estimating the utilization of uranium.

There is also the point that the variation of the total plutonium absorption cross-section is such as to lead to positive temperature coefficients of reactivity in respect of moderator temperature, assuming no compensating density changes.

An alternative method of operating a so-called plutonium equilibrium fuel cycle is to reject the plutonium left in the uranium fuel elements when they are removed from the pile. In this case, if one is to have a long burn-up, it will certainly be necessary to use an enriched fuel initially, and this fuel will have to be enriched uranium from a diffusion plant. The amount of enriched U^{235} which will have to be provided will correspond to the plutonium which has been rejected. Since diffusion plants make no use of U^{238} and since they extract only a portion of the U^{235} present in natural uranium, the over-all utilization of uranium by reactor and diffusion plant will be very much less than if the plutonium is recycled. This is, however, a fuel cycle which has often been described in the literature.

Finally, in this lecture I should like to say a few

words about the reactors which we have considered in the past in the United Kingdom.

We gave initial consideration to the Calder Hall type of reactor as long ago as 1947, but at the time it was not possible to consider a reactor of this type for a variety of reasons, and a small long-range effort was directed rather into systems which would make more economical use of nuclear fuel. A fuel cycle based on the use of thorium was considered, as was also the fast reactor development utilizing U^{238}. About 1950, as effort became available and as the supply of uranium did not seem likely to present such serious problems as had originally been thought, attention was directed strongly to natural or slightly enriched uranium reactors.

As far as was known, only four moderators could be considered. These are well known: light hydrogen, heavy hydrogen, beryllium and carbon.

As regards coolants, the obvious ones—gases, hydrogen-containing liquids and sodium—were allowed for. Other molten metals were considered a long-term possibility, and the same was considered to apply to fluidization techniques or homogeneous reactors.

A table (in P/403) was drawn to indicate likely short-term possibilities. The four moderating atoms are listed at the top, and the four coolants are listed down the side of this table. We rejected beryllium and beryllia in favour of graphite, on the grounds at that time of supply and cost. We rejected sodium cooling of the water systems and water cooling of graphite-moderated systems. This left us with the Calder Hall line of development with the very interesting possibility of going on to sodium cooling. It left us, alternatively, the two water reactors; that is heavy water cooled and heavy water moderated, and light water cooled and light water moderated. In addition, with light water we considered the possibility of organic liquids and molten caustic soda.

The United Kingdom had no large-scale heavy water production programme. Also, it saw little point in hurrying forward with such a programme until a cheaper manufacturing process could be obtained, bearing in mind that nearly one ton of heavy water is needed in a pressurized heterogeneous heavy water system for each installed megawatt of electrical capacity. We therefore narrowed the field further, initially, to the Calder Hall reactor which had the great advantage that for the early stages of our power programme it could be designed to work without any initial enrichment. Sodium cooling could then be an important development as the next step.

Secondly, it seemed clear that one should develop a light water moderated and cooled system if reactor physics studies which were commenced about that time confirmed the theoretical predictions that such reactors should have a reasonably good neutron economy. We rejected caustic soda, but we left in an organic liquid.

As regards initial enrichment, the accident of nature by which it is possible to make a reactor work with natural uranium and the high cost of making fissile material have led to a tendency to assume that it is a good thing in the long run economically to design reactors to use natural uranium. This may result in an unnecessary loss of neutrons in the moderator, or by leakage from the surface of the system. Neutrons have an economic value, since they can be absorbed in thorium or U^{238} to make nuclear fuels. For this reason it may pay, for example, to reduce the lattice pitch of a reactor in order to achieve this objective. Neutrons otherwise lost in the moderator go into U^{238} in that case. Owing, however, to an increase in neutrons captured by resonance absorption, the conversion factor in a reactor can become so high that the reactor may not go critical unless the proportion of U^{238} to U^{235} is decreased.

A further point is that fast fission of U^{238} can easily lead in a light water reactor to a useful gain of neutrons if there is a closely spaced lattice, but the effect is unfortunately much less with thorium. Also, there may be a deterioration in the plutonium economy, for the reasons I have mentioned. We have, therefore, a rather difficult economic compromise between size of system, initial enrichment, nuclear design and fraction of uranium fuel consumed.

The other day I made a few remarks in my lecture about thorium. All I should like to say today is that in a heterogeneous reactor the economic dividing-line between using thorium and not using it is not necessarily the point where such a cycle becomes self-maintaining. When plutonium becomes plentiful and as cheap as the U^{235} in the natural uranium, it will almost certainly be preferable to use plutonium to consume thorium rather than to consume U^{238}. The exact point when this occurs will be governed very largely, I think, by metallurgical and processing costs rather than necessarily by nuclear or engineering considerations.

For the future, I am fairly certain that with thermal reactors we shall go in the long run to the thorium-U^{233} fuel cycle but, as I mentioned in my previous lecture, we shall need uranium in order to provide the fissile material so that such a system may be gotten under way. The alternative for the future is the fast reactor which does make it possible to consume U^{238} by means of the plutonium-U^{238} fuel cycle. I believe many statements have been made to the effect that we have a fuel cycle here which doubles the amount of plutonium each time we burn the plutonium in the reactor. This is more or less correct for our small zero-energy system, but I should like to qualify the statement by mentioning that the engineering design and, also, the processing of the fuel will almost certainly be much cheaper if we sacrifice some of this gain. Hence, there will be in this case an economic compromise with a gain factor somewhat less than that which I mentioned before.

Mr. WEINBERG (USA) presented P/862.

DISCUSSION OF P/4, P/403, P/862

Mr. DAVIS (USA): Mr. Dunworth has said quite a bit about the possibilities of using thorium. Considering that natural uranium, containing 0.72 per cent of U^{235}, is worth about $40 per kilogram, I wonder whether Mr. Dunworth could comment on the value of thorium in comparison in view of the fact that it contains no fissile material. In other words, how much is the U^{238} or the Th^{232} worth by itself as a fertile material as distinguished from the fissile material content?

Mr. DUNWORTH (UK): I think that obviously that is a question I cannot possibly answer. I am sure that our raw materials people would be very annoyed if I did.

Mr. GAST (USA): In view of the fact that Mr. Lewis' paper was not available to the United States delegation earlier, I should like to ask him if he could briefly outline his method of maintaining reactivity when using recycle fuel.

Mr. LEWIS (Canada): In reply to that question, I would say this: That is done, firstly, by taking advantage of heavy water as a moderator so that the neutron losses are kept low; secondly, by not making the reactor too small, so that the leakage losses are not too great; and, thirdly, by moving the fuel inside the reactor—this has to do with a question that I just touched on but this is not after a short irradiation. I am not moving any fuel at a smaller irradiation than 1.1 neutrons per kilobarn. This action of moving the fuel is really to postpone the processing, so that the processing costs are kept down. By making a heavy water reactor of sufficient size, we think it will be practicable—although, as yet we have not got sufficiently accurate information on Pu^{241} and perhaps Pu^{239}—to achieve a fuel regeneration factor as large as 0.9 and this seems to be all that is needed.

Mr. WIGNER (USA): Mr. Weinberg has emphasized the importance of a high η. He also emphasized that the value of η—that is, the number of neutrons produced per fissionable atom destroyed—is particularly low in the resonance region—in the energy region between 1 and about 1,000 electron volts. In view of this fact, I am wondering whether there are any means to decrease the number of neutrons absorbed in this energy region—possibly by lumping the fissionable material—or by some other means that one could think of.

Mr. WEINBERG (USA): I should have made clear that the figures which I gave were effective values of η for a very thin slab. As Mr. Wigner implies, if the fuel is in thicker slabs, of course the effective value of η will be somewhat improved. This, I believe, will be gone into by Mr. Kanne in his paper to be presented later on.*

Mr. BROOKS (USA): I believe that my comment has already been covered by the discussion of Mr. Wigner's question. I was merely going to point out that there is evidence that the preferential self-absorption in capture resonances leads to an effective value of η in a resonance reactor which will be somewhat higher than the values of η corresponding to the capture-to-fission ratios which were shown in Mr. Weinberg's table.

Mr. TARANGER (France): Of necessity, I prepared my question before hearing Mr. Weinberg's oral statement. The change from the printed paper that he has made in his oral statement partially answers my question, but I will ask it just the same.

Mr. Weinberg condemns gas-cooled piles because of their low material efficiency. He bases this observation on a comparison between a pile cooled by a poor gas at low pressure and a pile cooled by a good liquid at high pressure. Has he considered the case of a pile cooled by a good gas at high pressure?

Mr. WEINBERG (USA): The figures which I quoted in my paper were naturally chosen to make my argument seem as convincing as possible. That, as you all know, is characteristic of every reactor designer. It was very kindly called to my attention by Mr. Dunworth, however, that the British expect to extract 4000 kilowatts per ton of uranium from their advanced Calder Hall reactors. This begins to approach some of the figures that one gets with liquid-cooled systems. I, however, am inclined to believe that the figures which Mr. Dunworth quotes, while they are fairly impressive, are still not as high as those which one can get with a very good liquid under most favorable conditions. Hence, if one compares—which I think is the only natural thing to do—a good gas under most favorable conditions with a good liquid under most favorable conditions, even the most enthusiastic proponents of gas cooling will, I think, have to concede that there is probably a difference of at least a factor of 5 between the ultimate material efficiencies.

Mr. LEWIS (Canada): I should like to add one comment since in my presentation I made very slight reference to thorium.

I should merely like to point out that the rate of breeding gained from thorium is rather slow, and, if the world is going to move into atomic power development on the scale and in the time that I suggested, a very large supply of uranium will be needed to get going.

Mr. DUNWORTH (UK): I should like to say that I certainly had no intention of being disparaging to Mr. Lewis about his fuel cycles. In fact, I am really very envious of a country whose Chairman has such immense reserves of uranium at his disposal.

For the United Kingdom, however, if all our electricity were at the present moment derived from nuclear power and if we used only 1 per cent of the uranium, we should actually be consuming 1,500 tons a year. If we, as we must, look 20 or 30 years ahead

*P/595, Session 18A, Volume 4, these Proceedings.

and perhaps have 4 or 5 times as much electricity, one can easily see that we start having an imports bill for our country which assumes quite appreciable proportions. Hence, anything that we can save on this by improving the efficiency of the utilization of the fuel is obviously worthwhile. I would certainly fully agree with Mr. Lewis that anything like 100 per cent utilization would obviously not be worthwhile at all in the next decade.

Mr. WEINBERG (USA): I should make clear to Mr. Lewis that nothing in my remarks implied that I had anything but the highest admiration for him in going ahead with his cycle. Of course, if we cannot get the thorium or the U^{233} that we need by breeding, nuclear energy will not stop. I think, however, that what I was doing, mainly, was making a prediction of where Mr. Lewis would find himself in some 30 to 40 years from now, quite irrespective of the obvious merits of the U^{238}-plutonium system.

Mr. LEWIS (Canada): Thank you very much. I have been foolish enough to make a sort of forecast, which I have not published, for 50 years ahead, and it does not include thorium.

The CHAIRMAN: Are there any further contributors?

Mr. GEOGHEGAN (UK): I should like to ask Mr. Dunworth whether he agrees that, since the likely over-all rate of breeding of U^{233} in a thorium-uranium system may only be below 5 per cent per annum, such a fuel system could never play a large part in the supply of power in which the over-all demand for power was growing by more than 5 per cent per annum?

Mr. DUNWORTH (UK): I fully agree that one has to get additional supplies, and that this unfortunately does involve one in consuming considerable quantities of uranium in order to make the fissile material unless one can devise some other method of making neutrons. However, it may be, as Mr. Weinberg has suggested, that we can get 10 per cent, in which case we are just about all right once we have got our supplies. The other possibility is, if one is going to be very academic, that we can go to the fast reactor where we may be able to breed at a much more rapid rate. There is a point about thorium too, that if one can get a roughly self-maintaining system, and if the fuel elements really can be made to last a long time and the fission product poisoning is not too serious, one may have a situation where one can get very long irradiations, much longer at least than in the heterogeneous uranium system.

Mr. GEOGHEGAN (UK): I was referring to 5 per cent per annum, assuming that with 10 per cent you only achieve one cycle in every two years; but surely with the rapid rate of increase of total power consumption per annum you have to continue to establish new self-maintaining systems at a high rate, and this will require a large fraction of the other power stations to be U^{233} converters.

Mr. DUNWORTH (UK): This point was covered in Sir John Cockcroft's paper—that one does need a considerable inventory in any reactor—and therefore I am certainly not suggesting that we can forevermore just import 15 tons of fuel a year. I think I made this clear in my paper on thorium, and it was certainly made quite clear, too, in Sir John Cockcroft's paper on the United Kingdom power programme presented a day or two ago.

Mr. DAVIS (USA): I should like to make the point that it is not just the breeding gain that is important here. Since one has difficulty achieving high specific power in a fast breeder even though it may have a high breeding gain, the thermal breeder, even though it may have a low breeding gain with high specific power, may do just as well in increasing fuel supplies.

Session 3.1

EXPERIENCE WITH NUCLEAR POWER PLANTS

LIST OF PAPERS

The First Atomic Power Station of the USSR and the Prospects of Atomic Power Development

By D. I. Blokhintsev and N. A. Nikolaev, USSR

INTRODUCTION

The construction of the first industrial atomic power station in the USSR with an output of 5000 kw was completed in 1954, and by 27 June 1954, this power station was already generating electricity from the energy of fission of uranium nuclei.

The first atomic power station was built through the efforts of a large body of physicists, designers, heat engineers, technologists and many other specialists, and has become a reality as a result of the extensive cooperation of science and industry.

Numerous difficulties had to be overcome before the atomic station was finally completed; most of them could not be foreseen in the beginning and sprang up only as the work progressed. It would have been impossible to cope with these difficulties if it had not been for the versatile preliminary work carried out by our scientists, designers and technicians. By the time work on the design of the power station was begun, a vast amount of experience had been accumulated on the design and calculation of atomic reactors.

A number of papers of major importance for the realization of the atomic power station were presented to the July session of the USSR Academy of Sciences,[1] and others will be brought before this Conference.[2]

Out of this great diversity of studies, those carried out at the pilot reactor of the Academy of Sciences of the USSR[3] are of special importance. This reactor was designed expressly for the investigation of problems in physics and heat engineering connected with power reactors.

The main task in the construction of the first atomic power station in our country was the solution of the scientific and engineering problem of building an industrial atomic station that would be reliable in operation.

Appraisal of the cost of an atomic power station from project estimates, prior to the accumulation of actual experience in the construction of such enterprises, would probably lead to great errors, because each new type of station requires considerable expenditures on experimental work and on the organization of the production of new kinds of materials. Such expenditures, obviously, cannot be decisive when nuclear power development has reached a high level; on the other hand, they may be quite considerable in the early stages.

The first atomic power station in our country was constructed in order to accumulate technical and economic experience with atomic power plants and to serve as a base for training personnel.

At present this station provides a real basis for nuclear power development in our country, and its operating experience may be of help also to other countries interested in the use of atomic energy for peaceful purposes.

Figure 1 shows a general view of the building in which the reactor of the atomic power station is located.

FLOW SHEET OF THE ATOMIC POWER STATION

The heart of the operating atomic power station is a pressurized water-cooled thermal uranium-graph-

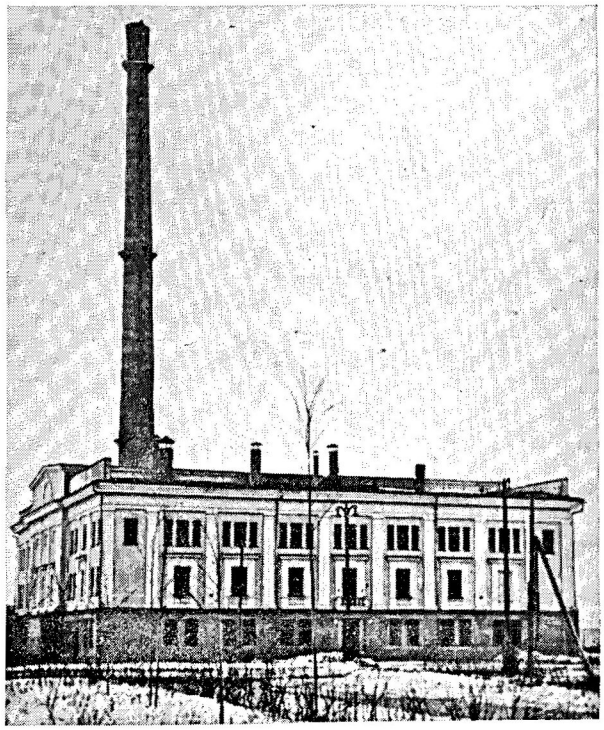

Figure 1. General view of atomic power station building

Original language: Russian.

ite reactor with a rated heat-generating capacity of 30,000 kw. The average flux of neutrons is 5×10^{13} neutrons/cm^2/sec.

The atomic fuel is enriched uranium containing 5 per cent of U^{235}, the total charge being 550 kg. The heat transfer system consists of two circuits: the water of the primary circuit, which circulates through the reactor, is under a pressure of 100 atmospheres and gives up its heat through a system of heat exchangers to the water of the secondary circuit, which, transforming into steam, drives a 5000 kw turbogenerator (see diagram, Fig. 2).

The use of a two-circuit heat transfer system eliminates the possibility of radioactive contamination of the turbine and its connections. This makes maintenance of the turbine and all its auxiliary equipment exactly the same as the conventional heating and power plants, biological protection (from radioactive radiations) being unnecessary for this equipment.

PHYSICAL PECULIARITIES OF THE REACTOR AT THE ATOMIC POWER STATION

Water is one of the best coolants, and its properties as such are well known. The selection of water as the coolant for the atomic power station reactor simplified a great number of engineering problems. However, the presence of water in the reactor required giving special attention to control of the chain reaction.

The reason for this is that ordinary water is at once the best moderator and a strong absorber of

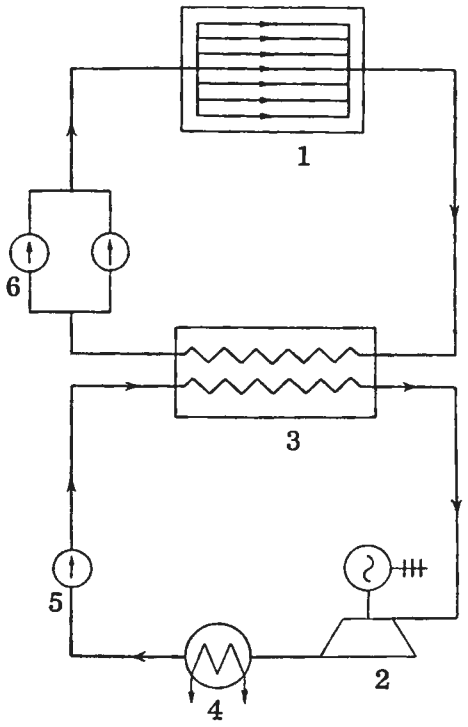

Figure 2. The plan view of the atomic power station. 1, Reactor. 2, Turbogenerator. 3, Heat exchanger (steam generator). 4, Condenser. 5, Feed pump. 6, Circulating pumps

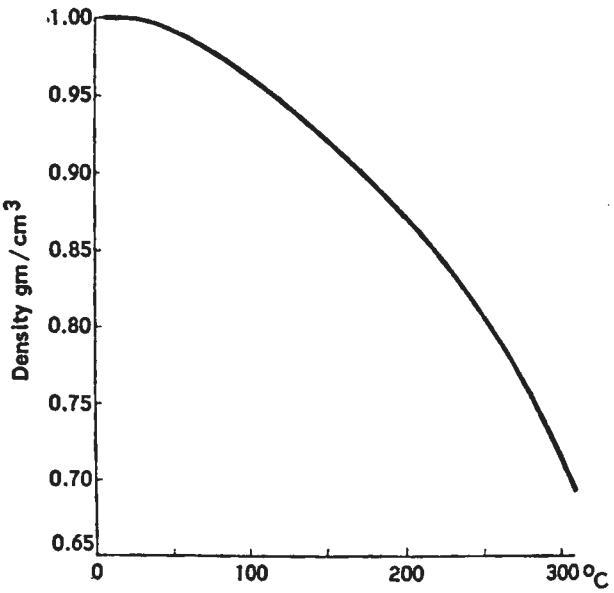

Figure 3. The dependence of water density on temperature at a pressure of 100 atm

neutrons. The moderating length in water is 5.7 cm and the diffusion length 2.8 cm, the corresponding values for graphite being 19 cm and 50 cm. For this reason the chain reaction is very sensitive to the water content of the reactor. But the water content in the reactor changes inevitably with the temperature, due to the change in density, which is especially marked at high temperatures (see Fig. 3 shown above).

Change of the water content may also be due to accidental leakage from the fuel channels. In this case the water may penetrate the graphite brickwork of the apparatus. The effect of the change in the amount of water in the reactor will depend largely on the ratio of uranium and graphite and the normal water content. If the quantity of graphite is so large as to slow down the neutrons itself, the additional water will act essentially as an absorber and therefore will reduce the reactivity of the reactor.

If the graphite-uranium ratio is smaller, the additional water will act so as to improve the moderating properties of the reactor, and therefore, will increase its reactivity.

There is a certain ratio between the quantities of uranium, graphite and water at which the reactor is least sensitive to changes in the water content. However, actual use of this ratio in the reactor would lead to a considerable increase in the quantity of structural steel inside the reactor. For this reason the quantity of water used in the atomic power station reactor is less than that corresponding to the least sensitivity.

Figure 4 shows the changes of reactivity in this reactor with the water content for both the cold and hot conditions. The point of normal water content in the cold reactor has been adopted as the origin of the co-ordinate system.

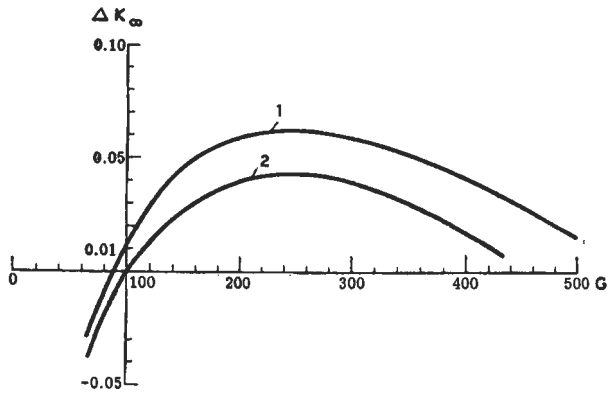

Figure 4. Variation of reactivity of the reactor in dependence on the quantity of water in the active zone (Δk_∞). X-axis: quantity of water G in the active zone of the reactor. Y-axis: neutron multiplication factor increment. Curve 1 corresponds to the cold conditions (20°C). Curve 2 to the operation temperature $\Delta k\infty = 0$ corresponds to a cold reactor with a normol content of water (G = 100 kg)

The diagrams show that the stability of the reactor during heating in the case selected for the power station, is higher than at the point of maximum.

As these are the cases involved in the run away of the reactor, the highly negative temperature coefficient is very useful for the stabilization of the chain reaction. The part of the curve to the right of the working point, corresponding to an increase of the water content in the reactor, has a smaller slope. It is more important, however, that the cooling of the reactor is a slow process.

Rapid filling of the reactor with water, in case of leakage in the channel, is prevented by special measures (drainage and automatic cut-out of the leaking channel). Hence, the reactivity changes to the right of the working point cannot be rapid. It should be noted also that the reactivity increase in the apparatus due to the cooling of the water partially compensates the reactivity loss owing to xenon accumulation (the so-called "iodine pit") in the reactor after shut-down. The curve in Fig. 5 is typical of the reactivity change in the reactor in time, after shut-down. The figure shows a temporary rise in the reactivity as a result of the water cooling, followed by a decline, due to the accumulation of xenon and then a slow rise owing to the decay of the latter.

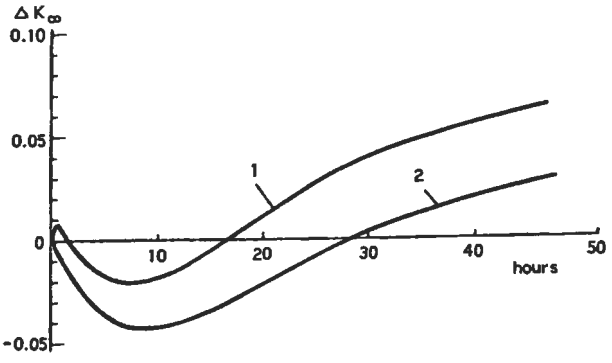

Figure 5. Changes in the reactivity of the reactor after shut-down X-axis: time after shut-down. Y-axis: neutron multiplication factor increment (Δk_∞). Curve 1 shows the reactor reactivity caused by the temperature effect and by the formation of xenon from iodine (the "iodine pit"). Curve 2 is the same in the imagined absence of a temperature effect. In this case the "iodine pit" is deeper

Another curve in the same figure shows the change in reactivity if the temperature effect is not taken into consideration.

In the process of cooling the temperature effect partially compensates the "iodine pit," making it possible to operate with smaller initial excess reactivity.

Thus, selection of the working point to the left of the maximum guarantees high stability of the chain reaction with respect to changes in the water content in the reactor.

The chain reaction in the atomic power station reactor is characterized by a rather high specific uranium heat removal. The heat flow from the surface of uranium is about 1.5×10^6 kcal/m²/hr.

The high specific heat removal and the large heat flow as well as the considerable intensity of the neutron field lead to rigid requirements for the endurance of the uranium fuel elements. Therefore, the problem of producing reliable fuel elements was one of the most important tasks in the whole project of the atomic power station.

In this reactor the burnout of uranium-235 is 15 per cent. Owing to the low resonance capture the production of plutonium from uranium-238 is low and equals only 0.32. Fast neutron multiplication is practically absent.

Thus, the reactor of the atomic station described operates almost exclusively by burning uranium-235, the degree of enrichment decreasing during the operation of the reactor from 5 to 4.2 per cent.

CHOICE OF MATERIALS FOR THE POWER STATION REACTOR

Proper selection of materials is a basic requirement for the reliability of the atomic power station. In this connection, mechanical strength, corrosion, thermal expansion, fatigue and other factors well known to designers had to be supplemented by knowledge of the effects of prolonged neutron irradiations on the properties of the materials used.

The problem of building the reactor for the atomic power station could be solved successfully only through the preliminary work in studying the behavior of materials in a neutron field carried out in various institutes of the USSR.

The most important components used in the atomic power station reactor are graphite, used as the moderator, and uranium enriched in the U^{235} which is the fuel. Therefore the behavior of these materials in a neutron field had to be ascertained.

As to graphite, it was established that graphite expands under the action of neutron irradiation due to crystal lattice distortion, the relative change of its

dimensions depending on the integral dose of fast neutron irradiation. Operation of the graphite at high temperatures counteracts this phenomenon. It is important to note that the increase in the dimensions of graphite is very anisotropic owing to the peculiarities of its crystal structure and texture.

Therefore, in order to secure stability of the graphite brickwork and to make it possible to extract the channel from the reactor whenever desired, clearances had to be provided which, on the one hand, had to be small enough to guarantee good heat conductivity in the brickwork as a whole, and, on the other hand, had to compensate not only for the thermal expansion of the graphite, but for its possible swelling under the action of the neutrons as well.

The second important phenomenon established for graphite was a decrease in its thermal conductivity. This decrease is a function of the integral dose of irradiation and the temperature, and is also anisotropic. Wherever graphite is subjected to bombardment by fast neutrons, i.e., in the immediate vicinity of the uranium rods, its thermal conductivity may fall to a value several times smaller than its usual magnitude.[4]

This must be taken into consideration in the heat transfer calculation of the reactor.

As a structural element uranium is in many respects even more troublesome than graphite. Three modifications of uranium are known: the rhombic, the tetragonal and the cubic, the transition points being 660°C and 800°C, respectively. Special investigations[5] have shown that repeated heating from 20°C to 500°C, followed by cooling causes substantial changes in the dimensions of the uranium rods.

This phenomenon is evidently connected with the anisotropic growth of the uranium grains during recrystallization. X-ray analysis has shown that it is accompanied by relief of internal stresses.

Neutron irradiation and fisson fragments cause similar changes in the dimensions of the uranium rods.

It has been demonstrated by special experiments that irradiation increases the plasticity of uranium by almost two orders of magnitude. Under irradiation the surface of uranium becomes globular in structure, as shown in the photographs in Fig. 6. The photographs were taken with the aid of an electron microscope under a magnification of 15,000. These properties of uranium make the manufacture of fuel elements a difficult problem, which has been solved successfully in several ways after diverse investigations.

It was required to find a strong and corrosion-resistant material for the tubes through which the high-pressure cooling water was to circulate; moreover, the material had to be stable under neutron bombardment. These requirements are met satisfactorily by stainless steel.

A study of the behavior of this steel in a neutron

Figure 6. The surface of irradiated uranium (photographed with an electron microscope, magnification 15,000 times)

field showed that the austenite in it does not decompose to form the ferrite constituent.

Certain mechanical properties of the steel improve under neutron bombardment (the ultimate strength and hardness increase).

The elongation decreases considerably, but remains within tolerable limits.[6]

These peculiarities of steel made it eligible as a reliable material for the high-pressure water tubes.

Since steel is a strong absorber of neutrons, all the tubes inside the reactor had to be thin-walled, which, by the way, led to smaller thermal stresses.

The most important parts of the reactor are its fuel elements. The heat transfer from uranium to water through the walls of a steel tube was studied on special test stands (see photograph, Fig. 7), where the heat production was imitated by means of electricity. The same test stands were employed subsequently for inspection of fuel elements manufactured by production methods.

However, the most important tests of the fuel elements were those carried out in the water loops of the reactor of the Academy of Sciences of the USSR. These tests made it possible not only to check the heat production but also to establish the stability of the fuel elements under the influence of an intense neutron field, and, hence, with the fission process going

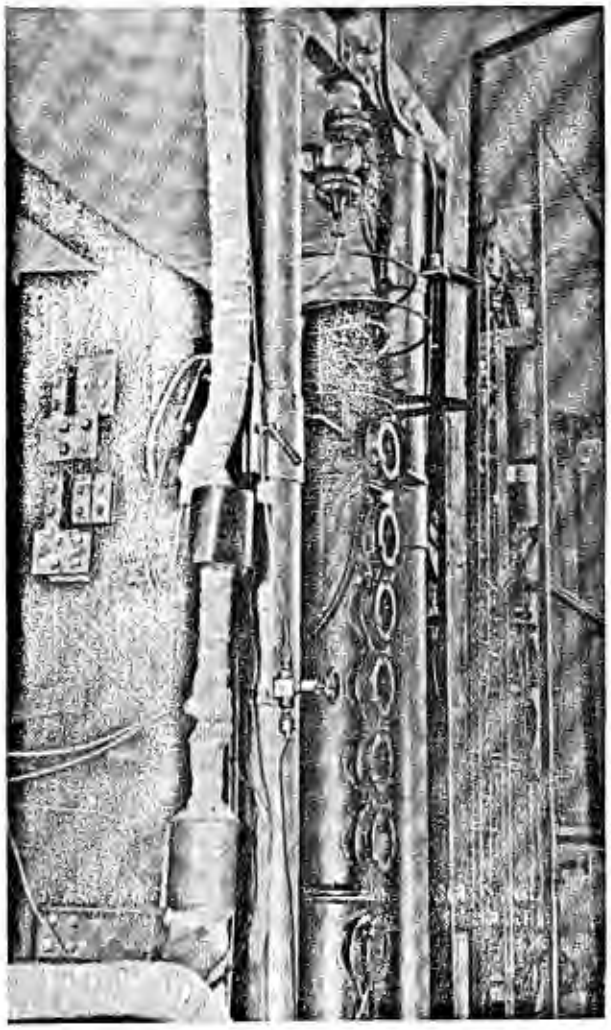

Figure 7. Stand for measuring heat transfer from uranium to water. The figure shows a column with windows permitting observation of the external surface of a fuel element

on in the uranium at the temperature in the future atomic station reactor.

The degree of enrichment was intentionally increased in the experimental specimens in order to reproduce fully the conditions that would exist in the APS reactor. The experiments lasted thousands of hours, after which the fuel elements were studied[7] in the "hot" laboratory of the Academy of Sciences of the USSR. These tests enabled the selection of the most reliable and stable type of fuel element.

The fuel elements are formed by the annular space between the inside steel water circulation tube of the first circuit and the thin-walled outside tube which keeps the fission fragments from penetrating into the cooling water for the reactor. The annular space contains enriched uranium in the form of a special alloy, which guarantees good thermal contact and stability in the neutron field.

ATOMIC POWER STATION REACTOR

The principal features of the reactor design were determined by the properties of the materials chosen.

A general idea of the design is given in Fig. 8. The reactor is encased in a hermetically sealed cylindrical steel jacket mounted on a concrete foundation. The jacket is filled with graphite brickwork having proper clearance. To avoid burnout of the graphite the jacket is filled with helium (or nitrogen).

A total of 128 fuel channels pierce the central part of the graphite brickwork. Each fuel channel is a long graphite cylinder containing thin-walled steel tubes carrying the primary circuit water. The water enters through the upper end of the channel connected with the inlet and outlet headers, then runs down through the tubes and returns, flowing up over the surface of the uranium fuel elements.

The uranium fuel elements form the cylindrical active zone of the reactor, 150 cm in diameter and 170 cm high, enclosed by a graphite reflector. In Fig. 8, the active zone is indicated by a dotted line.

Figure 9 shows the arrangement of the channels. The cover of the upper part of the reactor is removed.

The flow and temperature of the water are recorded for each channel. The indicating instruments are located on the central panel and are connected with the safety rods which stop the chain reaction when the water flow exceeds the prescribed limits or when the temperature rises above the allowed level.

At a pressure of 100 atmospheres water boils at 309°C. If the temperature of the water in a channel reached this level, its boiling would cause a sharp heat-removal decrease and, consequently, overheating and possible destruction of the fuel element.

It is, therefore, very important to control the outlet temperature of the water since this reflects the conditions in the channel. In practice, this is effected by means of a scanning light signal indicating the channels where the temperature of the water is above the prescribed limit. There is, in addition, a separate plug panel for measuring the absolute temperature of the water in any desired channel.

Temperature measurement is supplemented by water flow measurement. The latter is effected by flowmeters mounted on a separate board. The warning light signals which indicate that the water flow has exceeded the prescribed limit are sent to the central panel. A decrease in the flow of water in a channel leads to a rise in the outlet temperature. An increase in the water flow in a channel indicates a rupture in tubes of the channel or its connections.

All these deviations are dangerous and lead to a shutdown of the reactor (termination of the chain reaction) by means of the safety rods.

To prevent the rapid filling of the reactor with water in case of a rupture of channel tubes, which would lead to a rapid rise in reactivity, each channel is equipped with a cutoff device and a non-return valve to interrupt the flow of water from the header. A small water flow remains to remove the residual heat produced by radioactive decay of the fission products.

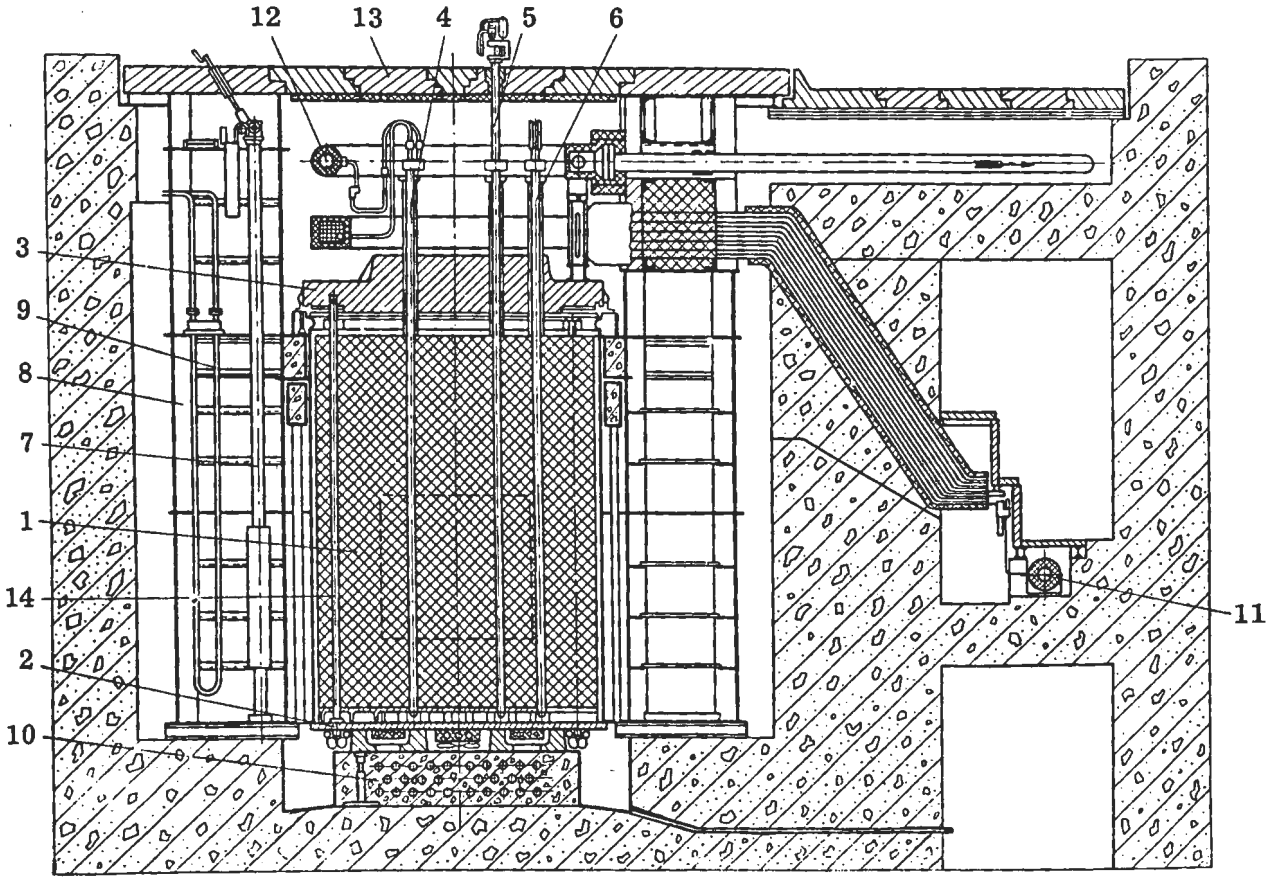

Figure 8. Atomic power station reactor: 1. Graphite brickwork. 2. Lower plate. 3. Upper plate. 4. Fuel channel. 5. Safety rod channel. 6. Automatic control rod. 7. Ionization chamber 8. Side shielding (water). 9. Refrigerator. 10. Refrigerator. 11. Distribution (inlet) header. 12. Outlet header. 13. Top shielding (cost iron). 14. Cooled reflector stand

The radioactivity of the water and gas in the reactor is measured constantly so that any rupture in the fuel elements would be indicated by a rise in activity. By drawing off the gas near the channel it is possible to determine, by measuring the moisture content, which channel is ruptured if the leak is not big enough for the flowmeters to record it.

In addition to channel flowmeters the central panel has indicators showing the total flow of water through the reactor, its pressure and temperature in the inlet and outlet headers.

To compensate for the excessive activity of the reactor, 18 boron carbide rods are installed in the reactor, 6 near its center and 12 at the extremities of the active zone. These rods move in special water-cooled channels with a separate cooling system. There is a separate cooling system also for the graphite reflector.

By means of wire ropes and servomotors these rods are moved vertically. Their position is registered at the central panel on a special board. Special precautions have been taken to prevent the operator from accidentally lifting these rods too quickly. With these rods it is possible to regulate the outlet temperature of the water in the fuel channels.

The total submerged length of all the shim rods, in the reactor at any one time, indicates the total excess reactivity, and consequently, the total excess of atomic fuel available.

Therefore, the fuel quantity may be expressed in linear centimeters. A special graduated scale indicates the value of each centimeter of rod with regard to its influence on the reactor reactivity.

Besides the shim rods there are four automatic regulating rods located in the reflector. Their position is changed automatically by the servomotors activated by ionization chambers.

If the chain reaction increases the automatic regulation rods fall; if it decreases they rise. One pair of rods works at a time, while the other pair is held in reserve.

These regulators maintain the chain reaction at a prescribed level to within 3 per cent accuracy. Finally, there are two safety rods with servomotors installed directly over the reactor's upper shield, to insure prompt action. In case of an emergency these rods fall freely into the active zone of the reactor and thereby stop the chain reaction. In all, there are 12 different emergency signals. The most important of these are: signals indicating a 20 per cent rise above the prescribed power level, a signal indicating a too rapid increase of the chain reaction, a signal indi-

cating the shut-off of electric current to the main circulation pumps, signals of changes in the flow and temperature of the water in the fuel channels.

The reactor has a number of devices for physical investigations and tests of materials.

There are in the reactor six curved channels, extending through the reflector into the boundary of the active region, for studying the behavior of samples of various materials under conditions of intensive neutron flux

$$(2\text{-}8 \times 10^{13}\ n/cm^2/sec)$$

and high temperature, and three straight horizontal channels. One of the latter running to the center of the reactor, permits the study of the radial neutron density distribution. Beam intensity at the outlet from the straight channels reaches 10^7 and 10^9 $n/cm^2/sec$. The channels are designed for nuclear research. Finally, there is a graphite thermal column adjacent to the reflector. The thermal neutron flux in this column is approximately 10^9 $n/cm^2/sec$. The column is equipped with remote-controlled plugs, so that instruments, and the materials under investigation, may be changed without interrupting the operation of the reactor.

Biological shielding is insured by a side water shield 100 cm thick and a concrete wall 300 cm thick. In the upper part of the reactor, biological shielding is insured by increasing the thickness of the graphite reflector, by the upper steel cover of the apparatus and by a cast-iron plate. The shielding attenuates both the fast neutrons which escape from the reflector and the gamma radiation from the radioactive fragments and the thermal neutron capture in the structural iron. Since the shield is only calculated approximately, special attention in designing the reactor was devoted to avoiding any possibility for the channelling of neutrons or gamma-rays.

Fast neutron inelastic scattering at a large angle in the steel jacket of the reactor caused an almost vertical neutron beam. An additional concrete shield was set up in the way of this beam.

Figure 10 gives a view of the reactor in operation, showing the top shielding plate and two servomotors of the automatic safety rods.

PHYSICAL DESIGNING AND EXPERIMENTAL STUDY

Considerably greater quantities of uranium have been loaded into the reactor of the atomic power station than is necessary to obtain the critical mass which makes the chain reaction possible. The chain reaction in this reactor was started for the first time on 9 May 1954. It was established that the critical mass is reached with only 60 uranium rods. The complete operating loading consists of 128 uranium rods. The 18 compensating boron carbide rods mentioned above absorb the excess reactivity of the reactor.

When the reactor reaches a sufficient power level its reactivity will drop considerably due to the heating

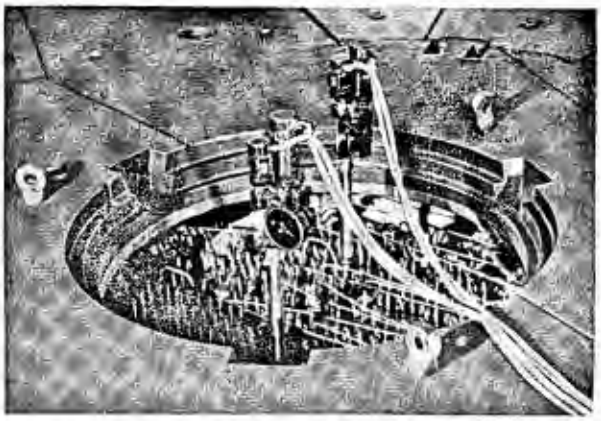

Figure 9. Upper part of reactor without cover showing the servomotors of the safety rods and their cables. The water connexions can be seen below

of the water. Moreover, xenon with its tremendous resonance absorption is formed which also leads to a drop in its reactivity (xenon "poisoning").

Lastly, its reactivity drops slowly during the operation due to the burning out of the uranium-235 and the formation of the fission products absorbing the neutrons.

All these effects make it imperative that the quantity of uranium should considerably exceed the critical mass of the cold reactor.

One of the most important tasks in making a physical calculation of the reactor was to establish the value of the critical mass of the reactor and that of the additional load of uranium, which was essential to ensure its prolonged operation and to compensate temperature and xenon poisoning effects.

The second task was to calculate the compensating capacity of the boron carbide rods. Account had to be taken of the fact that boron rods with the chosen diameters could absorb completely only the neutrons of energies less than 10-20 ev. The boron carbide rod is partly transparent for the neutrons of greater energy. It would be advisable to take account of the fact that the compensating capacity of these rods depends on the temperature of the neutron gas.

A third group of tasks was connected with the calculation of non-stationary states and probable

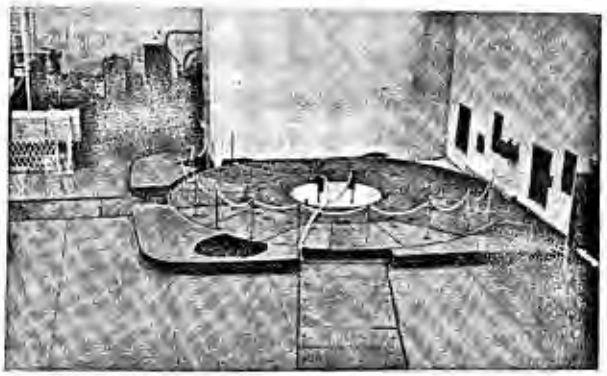

Figure 10. Upper part of reactor in operation, showing top shielding plate and safety rod servomotors

Table I. Rated and Measured Values of Reactivity of the Reactor. The Reactivity Is Given in $K_{eff} = a\ K_{\infty}$, where a Is Neutron Leakage Coefficient

	Calculated	*Measured*
Critical mass (number of fuel channels)	~ 59	~ 60
Total reactivity excess	0.13	0.11 ± 0.005
Reactivity excess compensating the temperature effect	0.029	0.025 ± 0.002
Reactivity excess compensating the poisoning	0.039	0.40 ± 0.003
Reactivity excess compensating the burn-up	0.068	0.045 ± 0.004
Compensating ability of one rod of the internal circle	0.012	0.013 ± 0.001
Compensating ability of one rod of the external circle	0.007	0.007 ± 0.001
Compensating ability of two safety rods	0.020	0.018 ± 0.002

emergency states of the reactor. These tasks called for a simultaneous calculation of the criticality and thermal states of both the apparatus and the entire primary circuit.

The general principles of the physical calculation of the heterogeneous reactors, the calculation of the resonance absorption of neutrons as well as the calculation of the partly transparent ("grey") absorbing rods have been outlined in a number of reports submitted to the International Conference on the Peaceful Uses of Atomic Energy.[2,8,9]

It turned out that it was possible, without making gross errors, to use a simple two-group model according to which the neutrons are divided into two groups: thermal neutrons and epithermal neutrons. At the same time it is important to formulate boundary conditions on the interface between the active zone and the reflector.

Calculations were made also on the basis of the precise neutron slowing down equation with the use of the finite differences method which is more consistent and less laborious.[10]

The resonance absorption of neutrons was calculated on the basis of the previously evolved theory.[11,12,13] According to this theory the effective breadth of the energy interval ΔE within the limits of which the resonance neutrons are strongly absorbed is proportional to the square root of the thickness of the uranium slug.

The calculated value of the coefficient of resonance absorption is 0.87. The experimental data, preliminary as they are, point to a somewhat greater value.

Both calculation methods were checked with the experimental reactor RPT, which was already in operation during the designing of the atomic power station, and also with a special graphite-water physical reactor which was a more or less exact replica of the reactor of the atomic power station.

It turned out that the results of calculations of the critical mass and neutron distribution were in good agreement with the experiment in the case of no water being present in the calculated reactor. In the presence of water the agreement is less precise due to the heterogeneous effect of water, i.e., the spatial dependence of the influence of water on the chain reaction.

Owing to the presence of water, the application of the age approximation in calculating the reactor cell gave rise to serious doubts. That is why a separate cell of the reactor consisting of a fuel channel and surrounding graphite moderator was calculated on the basis of a kinetic equation which makes it possible to take into account the non-age character of the neutron slowing down in water. This calculation showed that the age approximation was justified in the case of the reactor of the atomic power station. The same calculation proved to be helpful in finding the effective constants (range and capture length of the neutrons) for an equivalent homogeneous reactor.

The comparison of the results obtained in the various calculations leads to the conclusion that the critical mass of a graphite-water reactor can be calculated with an accuracy of 15 per cent. Actually, however, the theoretical value of the critical mass (59 channels) coincided with the experimental value (60 channels) more precisely.

Table I gives both theoretical and experimental values of the temperature effect, xenon poisoning effect and the compensating capacity of the boron carbide rods expressed in terms of the effective multiplication factor.

As regards the excess reactivity intended for burn-up and fission products poisoning, it proved to be sufficient to insure the reactor working at full power for two and a half months which is considered quite satisfactory.

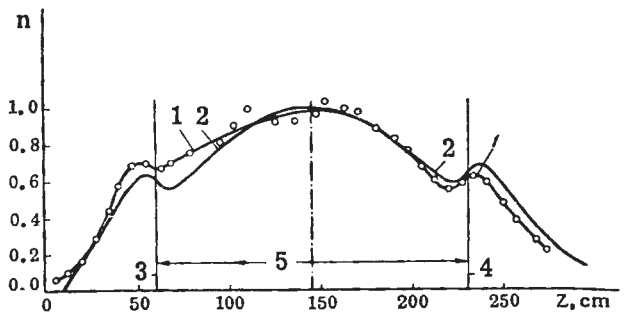

Figure 11. Axial distribution of thermal neutrons density: 1. Experimental curve, made with absorbing rods of the internal circle lowered and rods of the outer circle raised. 2. Theoretical curve, determined on the assumption that the substance of the absorbing rods is uniformly distributed in the active zone of the reactor. 3. Boundaries of the active zone and of the lower reflector. 4. Same for the upper reflector. 5. The active zone.

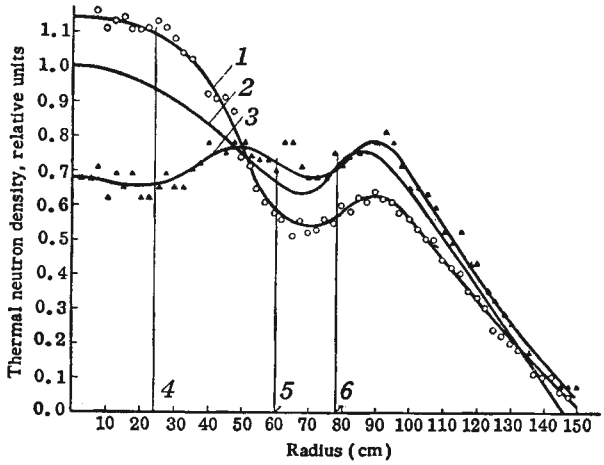

Figure 12. Radial distribution of thermal neutron density. 1. Experimental curve; the absorbing rods of the internal circle raised, the rods of the outer circle lowered. 2. Theoretical curve, determined on the assumption that the materials of the absorbing rods are uniformly distributed in the active zone. 3. Experimental curve; the absorbing rods of the outer ring lowered. 4. Absorbing rods (internal circle). 5. Absorbing rods (outer circle). 6. Boundaries of the active zone

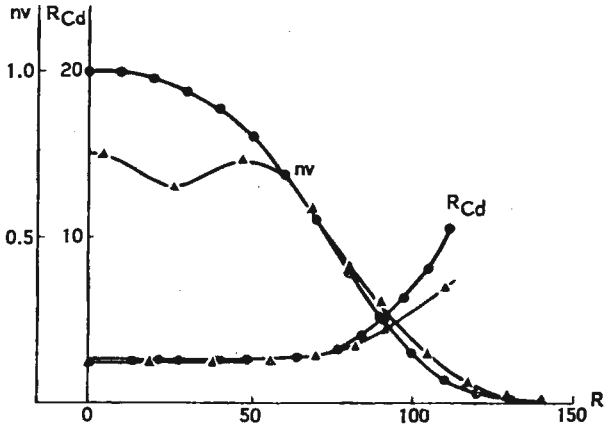

Figure 13. Distribution of resonance neutrons $n\ v$ according to an Au indicator ($E_2 = 4.9$ ev) and radiol distribution of the cadmium ratia R_{cd}. Circular points are calculation (for uniform distribution of the substance of the absorbing rod); triangular points are experiment (with absorbing rods of the internal circle lowered)

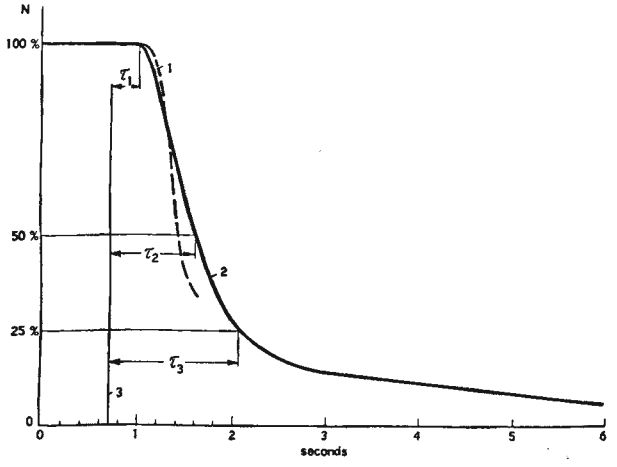

Figure 14. Decrease of reactor power after shut-down. N is power level, t time. 1. Theoretical curve. 2. Experimental curve, recorded by oscillograph. 3. Time of the emergency signal. The decrease in power to one-half takes about 0.90 seconds

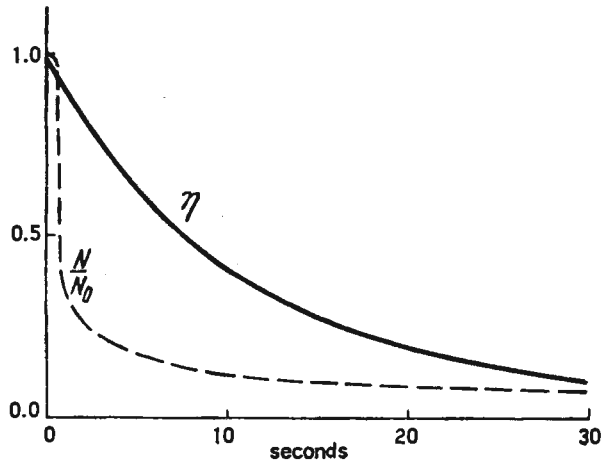

Figure 15. Changes in the water flow and reactor power after shutting off the electric supply of the circulating pump motors. X-axis: time; Y-axis: relative change in consumption of water (η) and the relative power N/N_0

The comparison of the theoretical and experimental data for the axial distribution of neutron density in the reactor is made in Fig. 11, and for the radial distribution in Fig. 12. In the latter case the theoretical calculation is made on the assumption that the absorbing substance of the compensating rods is distributed homogeneously in the reactor.

The theoretical and experimental data on the spatial distribution of the resonance neutrons and on cadmium ratio are compared in Fig. 13.

To illustrate the transient response, Fig. 14 shows the curves (theoretical and experimental) of the power of the reactor on the decrease when the emergency protection rods have fallen.

If the emergency system shuts down the reactor upon receiving a signal that the main circulating pumps are cut off from their current supply, the presence of the remaining radioactive heat could result in the water boiling in the reactor. Figure 15 shows the curve of heat production after the safety rods are put in and the curve of the decreasing mass velocity of water after the pumps are cut off.

The comparison of these curves shows that the former falls more steeply than the latter, except for a very short period of time. The momentum gained by the water will maintain the mass velocity of water at a level sufficient for heat removal for dozens of seconds. This is more than enough time for the circulating pumps to be connected to the storage battery, which is done automatically.

It has been mentioned above that the reactivity of the reactor rises when the water content is increased. Therefore, the reactivity of the reactor will rise in the case of a leak occurring in the fuel channel, and the level of chain reaction will rise.

If the regulators do not cope with this rise the safety rods shut down the reactor.

To illustrate the kinetics of this process the curves showing the changes in the power of the reactor for

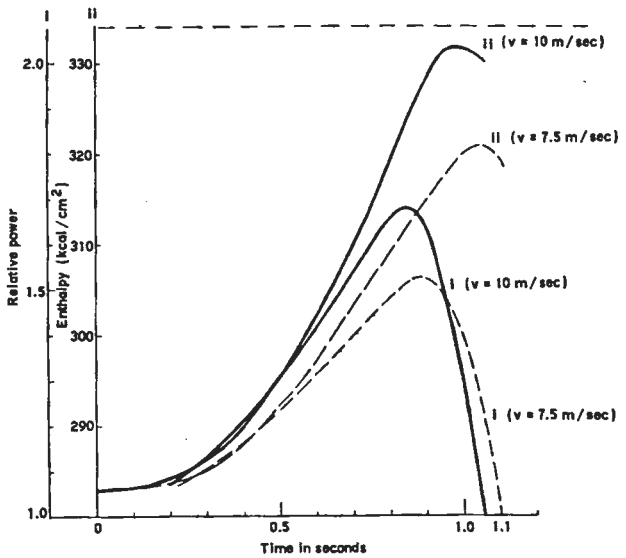

Figure 16. Changes in the power of the reactor N/N_0 and in the enthalpy of the water i in fuel channels at different velocities of the supply of water to the graphite brickwork of the reactor. X-axis: time t; Y-axis: enthalpy i, relative power N/N_0. The horizontal line at the top shows the enthalpy value (saturation) at 100 atm

two velocities of water supply are given in Fig. 16. Time is along the abscissa, while the power of the reactor is plotted along the ordinate. The power increases at first because of water entering the graphite brickwork and then drops owing to the action of the safety rods. The curves showing the changes in the amount of heat contained in the water in the fuel channels during this process are drawn in the same figure.

It follows from these curves that water in the fuel channels will boil if it is supplied to the graphite brickwork at a speed of ten liters per second. Special experiments have shown, however, that the speed of water supply in the reactor will not exceed two liters per second even in the case of the complete rupture of the tube of a fuel channel. The cut-off devices in the fuel channel will bring this figure down to 0.1 liters per second.

The calculations of the rise in reactivity have been made on the assumption that water supplied to the reactor is distributed uniformly throughout the graphite brickwork. However, it should be expected that the effect will depend on the location of the water concentration to a considerable extent.

If surplus water concentrates close to the uranium the rise in reactivity will be greater than in the case of water concentrating far from the uranium, say, in the graphite, where it may even have a negative effect. Consequently, the calculations mentioned above give only a rough picture of the actual process. This prompted the carrying out of special experiments in order to measure the influence of water on the reactivity of the reactor.

To this end a special fuel channel was made with greater holdup of water, and water channels (without uranium) were made too. These channels helped to confirm the heterogeneous character of the action of water and made it possible to establish that the theoretical calculations based on the assumption that water is distributed evenly overestimate its influence a bit. Although measures had been taken to prevent any possible flooding of the apparatus with water (drainage and switching off systems), it was, nevertheless, important to determine experimentally the possible rise of reactivity when the reactor is filled with water.

For this purpose measurements were made of the rise of reactivity in the reactor without water, one of its channels being filled with the normal quantity of water at first and later with the surplus quantity.

The dependence of the radial effect on the location of the channel was also measured.

This experiment was repeated in the reactor with the fuel channels filled with the normal quantity of water.

By using the data obtained in the previous experiments we can evaluate the reactivity of the fuel channel in the reactor filled with surplus water, and consequently estimate the internal reactivity excess of the reactor completely flooded with water because of a breakdown.

The measurements have shown that the possible rise in reactivity of the reactor will not exceed 0.01 (in $\Delta k/k$), which can always be compensated for by the available reserve of the shim and safety rods.

The comparison of the calculations and the results of the experimental study of the reactor shows that calculation and experiment agree with each other to a reasonable degree of precision. Consequently, these calculational methods can be taken as a sufficiently reliable basis for designing more powerful reactors of the same kind.

A DESCRIPTION OF THE TECHNOLOGICAL FLOW SHEET OF THE ATOMIC POWER STATION

As has already been mentioned the most characteristic feature of the station scheme is the two-circuit system.

The heat-transfer medium is water which circulates in the primary circuit under high pressure. This water transfers the heat from the reactor to the water of the secondary circuit. In the steam generators the water is converted into steam which goes to the turbine. The entire primary circuit is made of stainless steel and is situated behind the biological shielding.

The water of the primary circuit is heated in the reactor channels to a temperature of 260–270°C and passes to the steam generators from the header (see Fig. 17). There are eight such steam generators coupled in pairs in the shielded boxes (see Fig. 18). At peak power output, three pairs of steam generators are in operation and the fourth pair is in reserve. Each steam generator consists of a water heater, a water evaporator and a steam-superheater. The water of the primary and secondary circuits in the steam

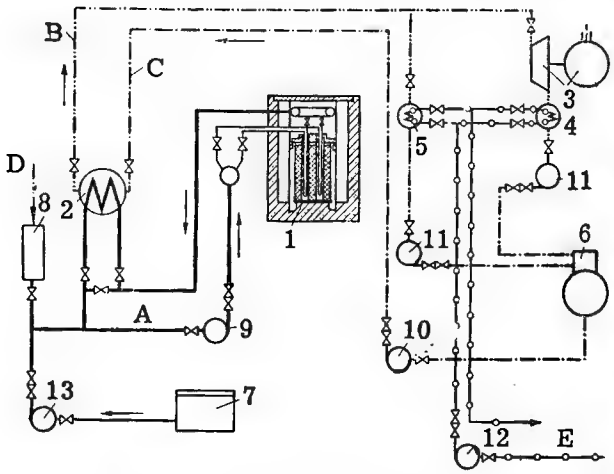

Figure 17. Technological flow-sheet of the atomic power station. 1. Reactor. 2. Steam generator. 3. Turbogenerator. 4. Turbine condenser. 5. Starting condenser. 6. Deaerator. 7. Supplementary water tank. 8. Volume compensator. 9. Primary circuit circulation pump. 10. Secondary circuit feed pump. 11. Condensate pump. 12. Cooling water pump. 13. Feed pump of the primary circuit. *A* is primary circuit water (100 atm). *B* is steam (12.5 atm). C is secondary circuit condensate. *D* is compressed air. *E* is cooling water

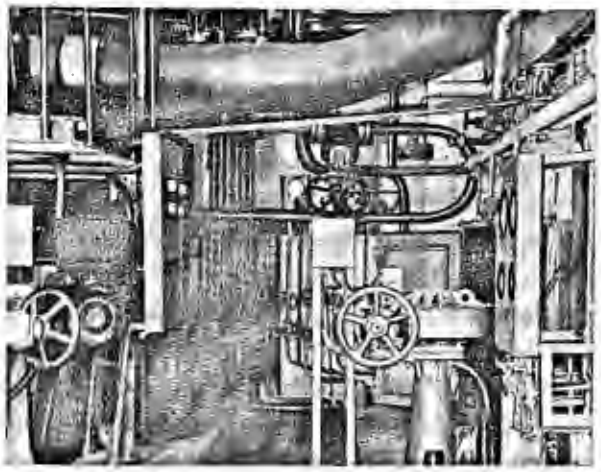

Figure 18. Part of the steam generator corridor. The open door of one of the four steam generator shielded boxes can be seen. At the top is the steam pipeline

Figure 19. General view of one of the main circulating pumps

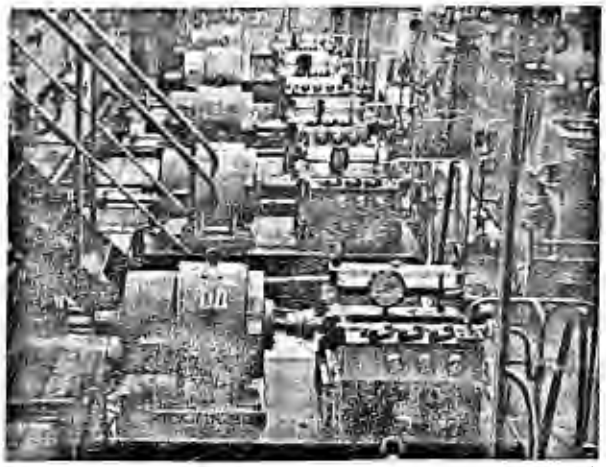

Figure 20. A group of primary circuit feed pumps

generators moves according to the countercurrent principle.

The water of the primary circuit leaves the steam generators at a temperature of about 190°C and enters the inlet pipe of the main circulating pumps. A general view of a main circulating pump is given in Fig. 19. There are four such pumps. Three are in operation at peak power output, insuring water consumption of about 300 tons per hr; the fourth pump is in reserve.

There are special feed pumps for adding fresh water to the primary circuit. Figure 20 shows a photograph of a group of such pumps. These same pumps, with the aid of an automatic device, feed the water to seals of the main circulating pumps in such a way that the pressure of this water is always 0.5 atmospheres above the water pressure in the pump. This insures a hydraulic seal which prevents the radioactive water in the main pumps from penetrating along the shaft.

It should be noted that as the water of the primary circuit moves away from the reactor, its activity gradually falls. The activity of the water coming out of the reactor is due to oxygen (0.2 curie/liter) with a short decay period (7 sec), whereas at the inlet of the reactor the activity of the water (2×10^{-5} curie/liter) is due basically to activity of impurities.

To insure compensation for changes in water volume during heating or cooling and to maintain for a certain period of time the pressure in the circuit in the case of a leak, the cooling circuit is supplied with volume compensators, which are circular tanks connected with cylinders of compressed gas.

A change in the volume of water in the primary circuit is compensated by either a fall or rise in water level in the volume compensators. The level reading is shown at the central control panel and is connected with the safety rods.

Along the primary circuit line, on the bypass, there are contact units for burning the explosive mixture

which may arise through water decomposition due to neutron irradiation.

From the pressure line of the main pumps, the water enters the input collector of the apparatus, from which it is fed to the apparatus channels.

The water of the secondary circuit might become radioactive only as a result of mechanical mixing with the water of the primary circuit, for example, as a result of cracks in the tubes of the steam generators. However, this is highly improbable. Therefore the equipment of the secondary circuit is of the conventional type.

The water of the secondary circuit (the condensate) is fed by feed pumps to the heater of the steam generator; it then enters the evaporator, in which the level is maintained constant by a special automatic device; from the evaporator the steam enters the steam superheater. At peak power output of the station, the three operating groups of steam generators produce 40 tons of steam per hour at a pressure of 12.5 atmospheres and at a temperature of 255–260°C.

This steam is directed along the main steam line to the turbogenerators (Fig. 21). From the turbine condenser the water returns to the steam generators.

An atomic reactor may also operate without producing steam (a water-water cycle). In this case, hot water, the heat of which is removed by ordinary refrigerators is produced in the steam generators. This scheme is used during the startup period only and for special experimental work.

The station is provided with various components of electric equipment, including a storage battery capable of feeding the electric motors of the main circulation pumps at low speed for several hours, which might be needed in an emergency (shutting off the electric supply). The sole purpose of the equipment is to insure a reliable source of electricity for the control instruments of the reactor and the main circulating pumps. A slight circulation of water in the reactor is needed even when the chain reaction is stopped in order to remove the heat due to decay of long-lived radioactive fission products.

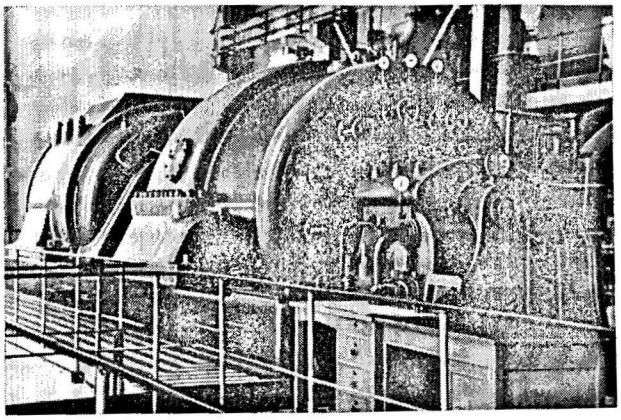

Figure 21. The turbogenerator of the atomic power station

All the station equipment is controlled from a central panel (Fig. 22). The central control panel gives the following important readings: the position of the shim and control rods, the temperature of the water at channel outlet, water consumption in the primary circuit and water pressure, the power of the reactor, the quantity and quality of the steam produced, etc.

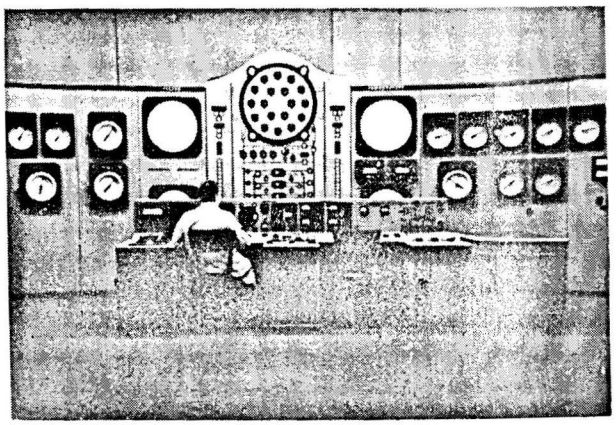

Figure 22. Central control panel. The simplified flow sheet is in the center of the panel. The switchboard is in the center of the picture

For safeguarding personnel from irradiation all rooms in the station are equipped with ionization chambers for measuring gamma radiation and with air-intake lines for controlling the radioactivity of the air.

In case the activity of gamma radiation or the activity of the air in any room is above normal the man on duty at the central dosimetric panel (Fig. 23) is warned by a light and sound signal. This same signal is also given in the room where the activity has exceeded the normal level. All rooms at the station are well-ventilated and the air escapes through a high chimney (Fig. 1). When the station is working normally, a slight amount of radioactivity is inevitable due to the argon of the air near the reactor.

Radioactive water (which results when the water is let out of the primary circuit) is discharged to storage tanks where its activity is reduced to a safety level by dilution and by decay.

ATOMIC POWER STATION OPERATION EXPERIENCE

Since the atomic power station was started up on 27 June, 1954, it has been tested in various operating conditions and has produced about 15 million kwh of electric energy. This operational experience has made possible the following important conclusions:

1. During operation of the station there has not been a single case of a fuel element failing. This proves that the heat-transfer from the uranium elements to the water proceeds regularly without disturbances, and that the stainless steel selected for the tubes is a reliable material for work in the active zone of the reactor. Despite the fact that preliminary tests of the

Figure 23. Part of the central dosimetric panel

fuel elements permitted us to expect such a result, still a bulk test, in the reactor of the station, of these most important parts might have produced undesirable results. The durability of the fuel elements insured reliable operation of the electric power station as a whole.

In view of the changes that graphite undergoes in a neutron field, it might have been feared that the holes in the graphic brickwork would become clogged with fuel elements.

Experience in channel replacement shows that the fuel channels are easily removable.

The temperature of the graphite proved to be above the rated temperature and at the hottest points reached 650–700°C.

Water flow through the fuel channels of the reactor was stable during operation. Figure 24 shows the radial distribution of power, temperature and water flow. As may be seen, it is possible to reduce the spread of water outlet temperatures in the channels to ±7° by controlling the water flow and equalizing the neutron density along the radius of the reactor.

Operational experience has shown that control makes it possible to insure rated power of the apparatus for a long period (from loading to loading).

Once the temperature and steam pressure are fixed the operation is characterized by a high degree of stability and the parameters may be maintained without requiring attention from the personnel of the station.

The automatic control system and the breakdown protection system of the reactor worked without a mishap.

Given definite operation conditions the activities of the station personnel consist in taking instrument readings and observing signals.

A startup of the reactor from zero to the rated power takes approximately half an hour and is the most important operation, especially when the reactivity of the reactor is not precisely known. At present, work is proceeding on an automatic startup project.

Unloading the uranium fuel elements from the reactor is done by sections with a view to economizing atomic fuel. The reactor is simultaneously loaded up with fresh heat-generating elements, and the less burned up peripheral elements are moved to the center of the core. This results in extending the exposure of the fuel elements and increasing the burn-up of U^{235} to 20 per cent.

The elements are replaced once every two months, requiring two to three days.

2. The main technological equipment of the station (main circulating pumps, booster pumps, steam generators, volume compensators, etc.) worked without failure. This is due, to a considerable extent, to the fact that during installation of the high-pressure pipe lines, great attention was paid to the quality of welding. It should be noted that the total number of welded joints is several thousand. Therefore the quality of the welded seams was carefully checked. Samples of welded joints were studied to check for the absence of crystallite corrosion. The welding was done by highly-skilled workers.

During the operating period of the station there were only a few cases of cracks and pits arising.

High purity of the water of the primary circuit is necessary for successful operation of the reactor. Bi-

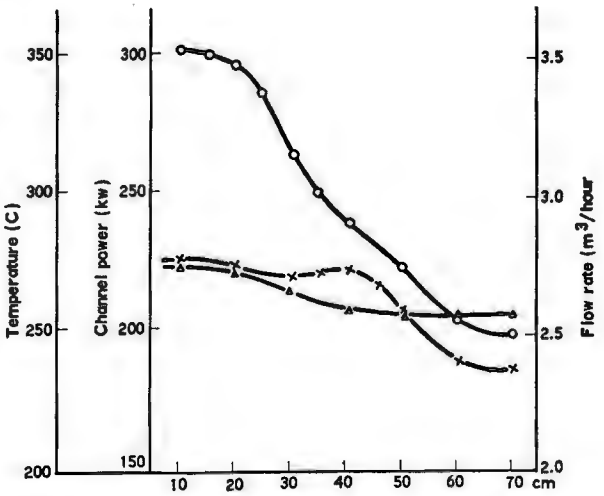

Figure 24. Radial distribution of power of the fuel channels, the flow and temperature of the water in the channel outlet. Circles are channel power. Triangles are outlet temperature; crosses are flow of water in the channel

distillate, with which the primary circuit is filled, has a dry residue of less than 1 mg/liter. In the primary circuit, the content of dry residue is maintained at a level of 3 mg/liter by constant removal of the primary-circuit water and by adding up to 300 liters/hr of fresh bi-distillate. An increase in the dry residue is probably caused by washing away of copper gaskets and asbestos-graphite seals in the valves.

Experience shows that the presence of this residue in the primary circuit does not lead to the formation of scales in the fuel channels to the extent of influencing heat transfer. Nevertheless, the station personnel consider a further reduction of impurities in the water of the primary circuit to be desirable. This would also make unnecessary the constant removal of water from the primary circuit.

Preliminary experiments[14] established the fact that at the temperature and pressure most common in the primary circuit, we find the reverse process of ion recombination which practically eliminates the formation of a detonating mixture in the water. This conclusion was completely confirmed during operation and made it possible to dispense with a contact apparatus for burning the detonating mixture.

3. The equipment of the secondary circuit is of conventional type, but it was important to know how flexible the control of the whole scheme of the station was. At first, there was a little uncertainty in the controllability of the whole reactor—steam generator—turbine setup, especially in transient operating conditions. No study was made of the startup system of the turbine, permissible heating-up speeds of the steam generators, and speeds of steam pressure rise.

As an example, mention may be made of the hydraulic shocks that occurred in the steam generators when they were heated during the first startups. It later became clear that this was due to steam generation in the water heater. It was due to an excess heating speed and to low steam pressure in the steam generator.

A study of startup conditions established the fact that startup from a cold state, for example, after repairs, may be carried out in 1.5 to 2 hours; and the steam in the first period is delivered to the startup condenser. After operating parameters have been attained steam is fed to the turbine. Such a startup system has proved to be convenient in operation.

The automatic setup of steam generator control is so stable in operation that the steam parameters are quite constant. As a result, the turbogenerator is absolutely stable in operation. Figures 25 and 26 show graphs of steam temperature and pressure before the turbine.

Station operating conditions were studied in the case of various loads on the turbogenerators. The station was also tested for a steam cycle used for the heating of building. The station was entirely stable for all of these conditions.

The station personnel succeeded in switching over to the reserve steam generators and circulating pumps without any loss in the power of the turbogenerator, which is very important for insuring normal operation of the station.

The time loss for repair work conducted during the year was small, since it consisted mostly in examination, revision of equipment, and various slight repairs.

4. During operation of the station, a detailed study was made of the efficiency of the designed biological protection. It turned out that the gamma radiation intensity in the main hall, at nominal reactor power, did not exceed 0.1 μr/sec, which is considerably less than the biological tolerance dose. When the appa-

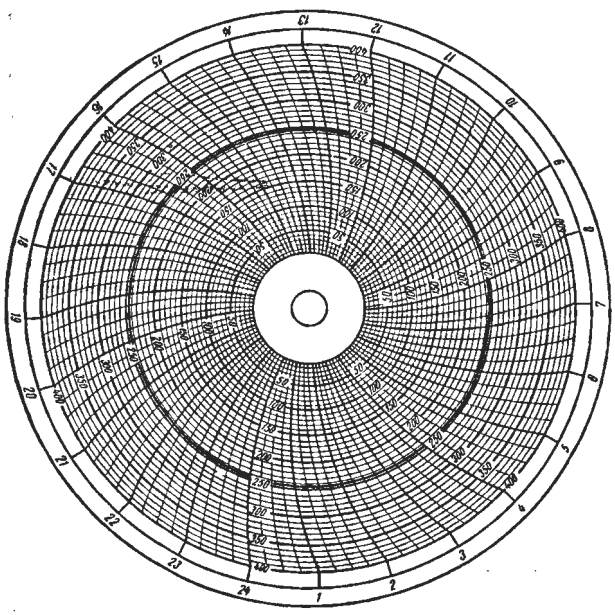

Figure 25. Inlet steam temperature

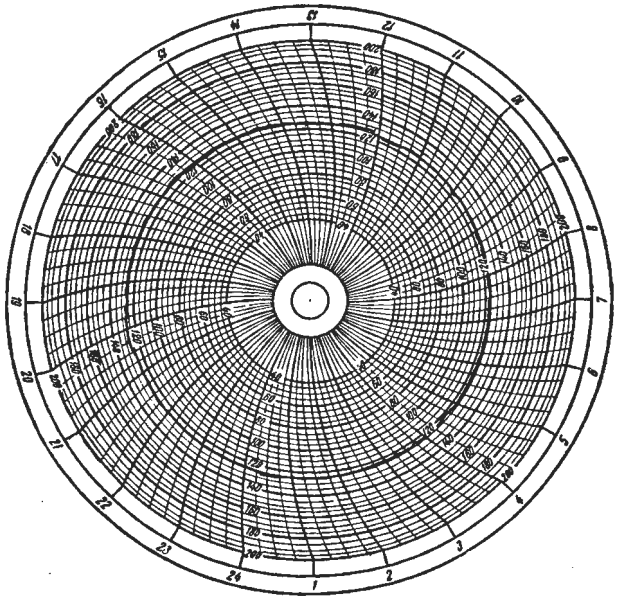

Figure 26. Inlet steam pressure

ratus is being unloaded the shielding walls of the hall safeguard completely work in the surrounding rooms. Shielding of the rooms containing the steam generators, the pumps and pipelines of the primary circuit likewise safeguards the service rooms from radiation.

For a limited period of time one may also enter the halls where this equipment is located (radiation does not exceed 6 μr/sec).

Gas activity is not observed in the main hall of the station and in the other rooms, which points to a satisfactory system of exhaust ventilation.

The activity of the gas passing into the ventilating funnel was due basically to the activity of the argon of the air. Measurements show that the selected height of the stack is sufficient insurance of safe conditions for the surrounding population.

The activity of the water drained out of the primary circuit is 10^{-5} curie/liter due to Na and Mn. After its retention in storage tanks and after dilution with pure distilled water, the activity of the drained water is easily brought down to the biologically safe level.

In the case of rupture in a channel, the presence of intermediate reservoirs excludes the possibility of fission fragment activity getting into the water.

Operational experience of the station showed that the system of local and central warnings signals of elevated gamma and gas activity works without a hitch.

During the time the station has been operating there has not been a single case of over-irradiation of personnel.

Each worker at the station has control dosimetric photo plate-holders and undergoes periodic medical examinations. No harmful influence of the work of the station on the health of the workers has been observed.

It should be noted that the most important means of combatting irradiation of the workers, is strict discipline in observance of operating and safety regulations.

A general conclusion after a year's experience in operating the station is that an atomic electric power station having as its source of energy an atomic reactor similar to the one described above will be extremely stable and reliable in operation.

A COMPARISON OF AN ATOMIC AND A COAL ELECTRIC POWER STATION

The operation of the 5000 kw industrial atomic power station of the USSR for nearly a year gives much data for designing and building large atomic power stations and makes it possible to evaluate economic factors on the basis of actual technical experience.

The cost of one kilowatt hour of electric energy produced by the first atomic power station exceeds considerably the average cost of one kwh in large thermal stations in the USSR. In 1953, the cost of electricity in the thermal power stations of the Ministry of Electric Power Stations was 10 kopeks per kilowatt hour.[15]

However, the cost of a kilowatt hour in the atomic station is comparable in cost with that of a similar-type low-power thermal power station (1–5-thousand kw).

An analysis of the cost of one kilowatt hour of energy produced by the first atomic station shows that its high cost is due first of all to the small size of the station, to the large outlay for manufacture of fuel elements, to greater relative consumption of uranium-235 because of the small size of the atomic reactor, and also to a number of peculiarities in the design of the station aimed at creating greater reliability of operation. Experience shows that these peculiarities may be dispensed with.

Consideration has been given to an atomic power station with 100,000 kw useful power equipped with two reactors of the same type as those of the first atomic station of the USSR, each of them having a thermal rating of 200,000 kw.

An analysis of the cost of one kilowatt hour of electric energy has been carried out for the case of an atomic station of 100,000 kw taking account of expected improvements. These calculations show that the cost of electric energy is considerably reduced for this case.

An increase in the reactor size will permit a decrease in the enrichment of the uranium loaded into the reactor. Calculations show that for the two reactors of this large station 20 tons of uranium a year will be required. The uranium will have to be enriched to 2.5 per cent in the uranium isotope 235 instead of the 5 per cent enrichment used at the first atomic power station.

Reduction of enrichment lowers considerably the fuel component of the cost of one kilowatt hour of energy produced by the first atomic power station.

The economics of the improved 100,000 kw atomic power station were compared with those of a modern thermal power station of the same capacity working on medium-grade coal burned in a powdered state.

It turned out that the fuel cost per kilowatt hour at the atomic station was greater but still comparable to the cost of the fuel component of one kilowatt hour at the coal-fired station of the same capacity working on low-quality coal.

It may be expected that the number of workers operating the atomic reactor and the steam generators will be two to three times less than in the boiler room and in the auxiliary structures of a coal station of the same capacity.

Energy consumption for the needs of such an atomic power station itself is less than in the case of a coal-fired station.

These favorable circumstances improve the technical and operational factors of the atomic power station and simplify its operation, thereby compensating for the necessity of installing equipment for the

Table II. Comparative Data for 100,000-kw Atomic and Coal Power Stations

Features compared	Units	Coal power station	Atomic power station
1. Weight of machinery and mechanisms	tons	2700	700
2. Weight of metal structures	tons	1250	900
3. Weight of pipelines and armature	tons	300	200
4. Weight of refractory masonry (graphite brickwork for atomic station)	tons	1500	500
5. Weight of mechanisms of coal storage	tons	2500	—
6. Weight of rolling stock	tons	300	—
7. Volume of concrete members	m^3	4000	9000
8. Volume of buildings (without turbine room and electric equipment)	m^3	75,000	50,000
9. Size of site	hectares	15	5
10. Inner power needs	kw	8000	5000

removal and storage of used radioactive fuel elements, radioactive gases and drained waters.

A comparison of the quantity of equipment, materials and maintenance required in the building of a 100,000 kw power station with atomic reactors of the type used in the first atomic power station of the USSR shows that the material outlay in the construction of the power station changes considerably and is more favorable for the atomic power station.

Table II shows the weights of the machines, mechanisms, and metal structures, as well as the volume of iron and concrete work plus some other data for the building of one type of the 100,000 kw atomic power station in comparison with a coal-fired power station of the same capacity. The figures relating to the turbine portion and its electrical auxiliaries are not given in the table since they are the same for both types of stations.

As may be seen from Table II, the expenditures for the construction of an atomic power station are less in many respects than for the construction of a coal station; this is due to the absence of large fuel storage space, fuel delivery systems, coal-grinding mills, ash removal equipment and other cumbersome structures that are common at coal-fired power stations.

Figure 27 shows an atomic and a coal-fired power station with capacities of 100,000 kw.

The coal station is equipped with two steam boilers, each with a capacity of 200 to 240 tons of steam per hour and two steam turbines of the condenser type, 50,000 kw each with a steam pressure of 30 to 35 atmospheres. The steam generators of the atomic station are rated for the same steam parameters.

A comparison of the size of the buildings of two such power stations shows that the atomic station is more compact.

A comparison of possible characteristics of an atomic and coal power station of 100,000 kw capacity shows the possibilities and future of atomic stations, similar to the first industrial atomic power station of the USSR, and the system of heat transfer from atomic reactors with the aid of water under pressure is a reasonable technical procedure, which may be widely used in atomic power engineering.

Depending on local conditions, for example the cost of electric energy in a given region and the volume of work required for mining and enriching the uranium, the capital operating costs of an atomic power station may vary widely.

The over-all cost of one kilowatt hour of electric energy produced by an improved larger atomic electric station of any given type will not be much more than the cost of the electric energy produced at a good coal-fired power station.

An atomic power station is now already more profitable than either a coal station in regions situated far from coal mines or coal stations operating on low quality fuel.

This opens up possibilities for atomic power stations to meet the growing electrical energy requirements of both industry and agriculture. An atomic power station may also be used for heating buildings in cities.

At present, in the USSR, various-type atomic power stations with a capacity of 50–100-thousand kw and more are being designed, including the above type station. Economic calculations show the possibility of producing electric energy at such stations at a cost of between 10 and 20 kopeks per kwh.

PROSPECTS OF ATOMIC POWER DEVELOPMENT

Conditions already exist, not only in large and rapidly developing countries such as the USSR, but also in many other countries, both large and small, which make it appropriate to build atomic power stations in specific areas remote from other sources of power.

It must not, however, be assumed that new atomic power stations must necessarily be built on the design of the first atomic power station of the USSR. The first station which was built on the basis of the most highly-developed methods and materials showed that economically efficient results could be obtained on those lines. At present, however, we are still at the beginning of atomic power development and accordingly there are many paths to be trod in setting up atomic power plants based on different types of reactors.

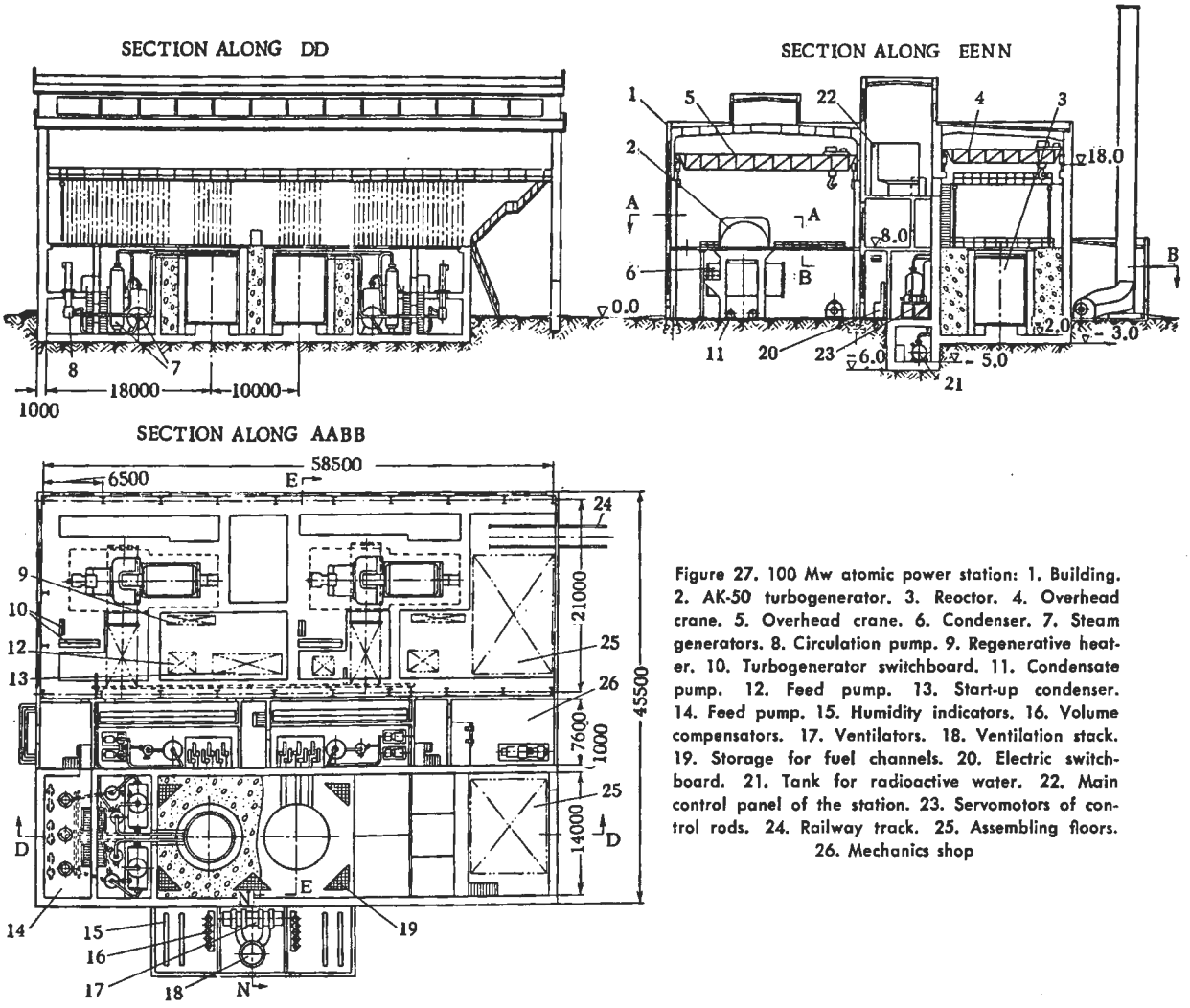

Figure 27. 100 Mw atomic power station: 1. Building. 2. AK-50 turbogenerator. 3. Reactor. 4. Overhead crane. 5. Overhead crane. 6. Condenser. 7. Steam generators. 8. Circulation pump. 9. Regenerative heater. 10. Turbogenerator switchboard. 11. Condensate pump. 12. Feed pump. 13. Start-up condenser. 14. Feed pump. 15. Humidity indicators. 16. Volume compensators. 17. Ventilators. 18. Ventilation stack. 19. Storage for fuel channels. 20. Electric switchboard. 21. Tank for radioactive water. 22. Main control panel of the station. 23. Servomotors of control rods. 24. Railway track. 25. Assembling floors. 26. Mechanics shop

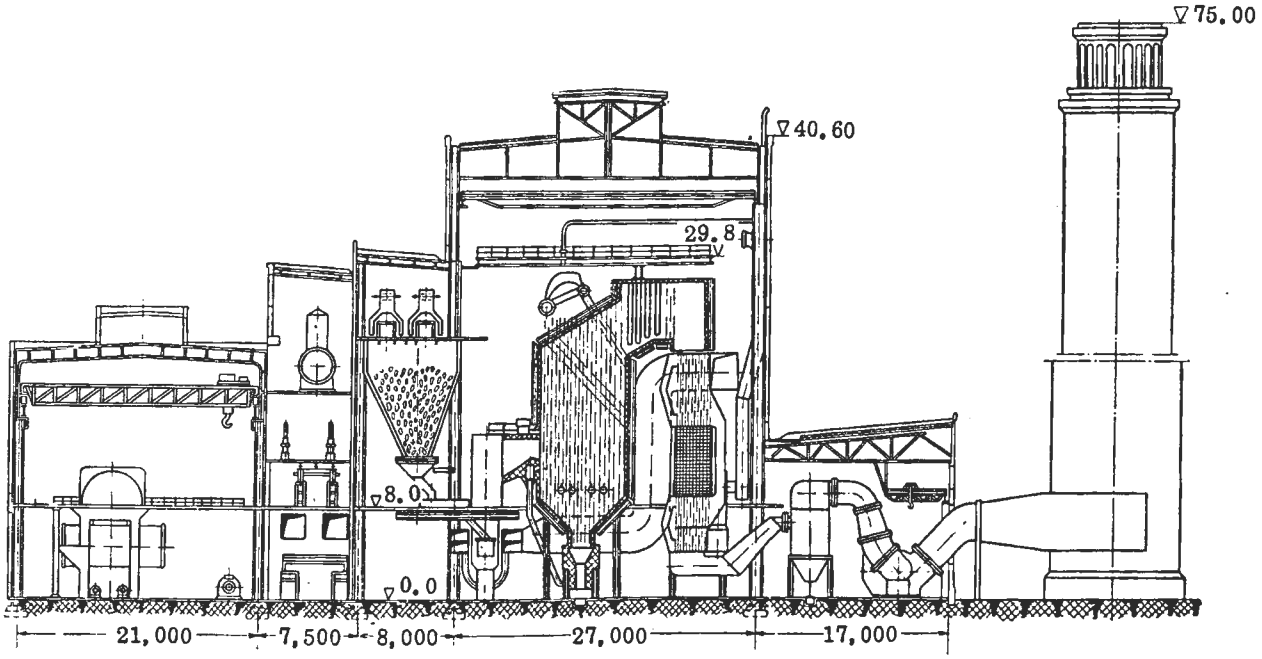

Figure 28. 100 Mw coal power station

From the physical point of view, the possible types of reactor must be classified mainly according to the energy spectrum of the neutrons which cause the fission of uranium or of other fissionable materials. On this basis, reactors can be classified into thermal-, intermediate- and fast-neutron reactors. Thermal-neutron and intermediate-neutron reactors must have a neutron moderator. The substance used as a moderator is a further criterion for classifying reactors. The fissionable material may be distributed in the moderator homogeneously or in slugs. Hence, reactors may be subdivided into heterogeneous and homogeneous reactors. Reactor types may be further distinguished by the choice of coolant and, finally, by the structural features of the reactors.

We shall consider, therefore, the comparative features of a number of reactors which may serve as the heat sources for atomic power stations.

First, we shall consider heterogeneous, thermal-neutron, graphite-moderated reactors.

1. Water Cooling

If the heat is removed from the reactor by pressurized water, as is the case in the atomic power station reactor described in this paper, the temperature of the coolant is too low to obtain high working-steam parameters in the turbo-generator which is a great drawback of this type of reactor.

Thus, for instance, to obtain steam at a temperature of 375°C the pressure in the primary circuit must exceed 225 atmospheres. To achieve this, a considerable quantity of structural material, most probably steel, would have to be introduced into the reactor, which would make it necessary to increase the uranium-235 enrichment very greatly, thus causing an increase in the cost of the atomic fuel used which will not be compensated by an improvement in the coefficient of thermal efficiency of the power station. Nevertheless, as was said above, atomic power stations based on reactors of this type can be built to work quite efficiently.

2. Gas Cooling

In the case of gas-cooling, this interdependence between the temperature and the pressure of the coolant does not exist. This represents an advantage of gas-cooling as against water-cooling. It may be used at higher temperatures. From the point of view of heat capacity, however, gas cannot compete with water and gas blowers of high power are essential to maintain a high rate of heat removal.

Gas-cooled systems may occasionally prove practicable. Hence British physicists and engineers have chosen for the immediate future to build graphite-moderated power reactors with gas-cooling. This choice makes it possible to utilize the experience gained from reactors already developed in Britain and at the same time to find a practical solution of the problem of setting up a nuclear power system on a considerable scale in a short time. In France too, apparently, the lines on which atomic power research is proceeding are explained by similar considerations.

3. Liquid-Metal Cooling

This combines the advantages of gas and water cooling. The high boiling point of molten metals means that high pressures in the primary circuit can be avoided and the considerable heat capacity of molten metals compared with gases eliminates the need to pump large quantities of coolant through the reactor. Sodium is the most acceptable coolant of this type.

The chief difference between sodium and water from the point of view of their effect on the chain reaction is that water is a far better moderator. Hence, in a water-cooled reactor in which the proportions of water and graphite are such that the water plays practically no part in neutron moderation but acts as a neutron absorber, water may be replaced by sodium. In these circumstances, the temperature in the primary circuit at low coolant pressures (5–10 atmospheres), can be raised considerably, thereby increasing the working-steam parameters. It is quite possible to achieve an efficiency exceeding 30 per cent. When the system is large in size, slightly enriched uranium (less than 1 per cent) can be used, and if zirconium or very thin steel is used for cladding the uranium slugs, it is even possible to operate on natural uranium. Thus, graphite-sodium reactors seem to be promising and probably economic as power reactors.

A disadvantage of sodium cooling is the high radioactivity of sodium, which decays relatively slowly (the half-life is 14 hours). This fact makes a sodium circuit difficult to handle. The heat-exchange system is also complicated by the need to pre-heat the sodium and to take a number of precautions to avoid contact between the water and the sodium.

Research on liquid-metal-cooled reactors is proceeding in many countries, including the USSR.

4. Water-Water Reactors

Another type of atomic power reactor is the reactor in which ordinary water acts as both moderator and coolant. Water has a number of distinctive features as a moderator: both the slowing-down length and the scattering free path are exceptionally short (5.7 cm and 2.8 cm respectively). Its most interesting peculiarity, however, is the dependence of its neutron scattering cross-section on the neutron energy; the cross-section is small for high energies and large for low energies. If the uranium slugs are distributed in a close lattice, some of the neutrons will penetrate neighboring uranium slugs where they may cause additional fission of uranium-238, which can only be done by fast neutrons. As a result of this, the multiplication factor is increased, so that some increase may be permitted in resonance absorption of neutrons as they slow down. Experiments and calculations carried out in the USSR indicate that

for certain lattice geometries a fast-neutron multiplication of as much as 15 per cent can be obtained.

Greater resonance absorption means that more plutonium is accumulated in this reactor than in thermal-neutron reactors using other moderators. This is the most important property to be found in reactors of this type.[16]

Exposures of varying length can be obtained, depending on the structure of the reactor. Physical calculations indicate that the length of the exposure may be extremely long. Whether the fuel elements can withstand a long exposure without substantial damage is another question, which can be answered only by experiment. If long exposures prove feasible, a system of this type will be an exceedingly economic source of atomic energy.

For that reason, the diverse studies necessary for building water-water reactors are being carried out in the USSR. Research workers in the United States are also paying a good deal of attention to this type of reactor, for the same reasons, of course. It is interesting to note that atomic power development in the USA and in the USSR is proceeding on very similar lines. The great expanses of land and the diversified economic areas which are characteristic of both countries call for the fullest possible consideration of the various types of atomic power station available to our countries on the basis of their scientific and engineering resources.

We would observe in conclusion that the disadvantage of water-water reactors is the difficulty of compensating for temperature and other effects due to the slow neutron-absorbing capacity of the rods and the short scattering path and slowing-down length which are characteristic of water. A factor which requires special attention is the action of radiation produced by the reactor on the material of the casing, which has to withstand the high pressure of the water.

5. The Heavy-Water Reactor

The efficiency of thermal-neutron reactors may also be improved by reducing neutron absorption by the moderator. Heavy water is a moderator which is an extremely weak neutron absorber and therefore suitable for this purpose.[17] If the heat in a reactor of this type could be transferred without the use of strongly neutron-absorbing structural materials and if, in addition, a heat-transfer fluid with a low absorption power could be chosen, such a reactor could operate on natural uranium, which must certainly be regarded as a distinct advantage in a reactor of this type. For this reason, gas-cooled and heavy-water moderated systems are undoubtedly suitable from the point of view of neutron economy. Power development on this basis is, however, tied to the use of large quantities of expensive heavy water and complicated gas-blower systems, and this is the disadvantage of the type of reactor under consideration.

6. Homogeneous Reactors

The reactors which we have been considering so far have been heterogeneous reactors. The uranium in them is arranged in slug form in a spaced lattice of one sort or another. These reactors have low resonance capture. Another possible course is that of having the uranium evenly distributed in the moderator, either in solution or in suspension. Resonance capture in such a reactor will be high, but the resulting reduction in the multiplication factor may be offset by the possibility of removing the xenon, which inhibits any reactor operating on thermal neutrons, from the reactor core. Unless this is done, a substantial increase in uranium-235 enrichment is necessary. A reactor of this type has no fuel elements, which always present a definite problem. The chemical preparation of the fissionable material is also simplified in this type of reactor.

In one of the papers presented to this conference by the USSR[18] a design for a boiling homogeneous reactor is described; in this connexion, an exceedingly important result of the investigations carried out in our country is evidence that it is possible to eliminate radioactivity from the steam from the boiling reactor to a very high degree of decontamination. Although many questions remain to be cleared up in connexion with the development of boiling homogeneous reactors, it is reasonable to believe that the difficulties we are encountering can be overcome completely.

The reactors described above have the defect that the energy they develop is derived almost exclusively from the burn-up of uranium-235, which occurs in nature 140 times less frequently than uranium-238.

In thermal-neutron reactors, uranium-238 is only partially used by conversion under radiative capture and subsequent radioactive decay into plutonium-239, which is also fissionable by thermal neutrons, like uranium-235. The conversion factor for plutonium from uranium-238, i.e., the number of plutonium atoms derived from a single split atom of uranium-235, is not greater than unity in thermal-neutron systems, according to all calculations and experimental data. It is, therefore, doubtful whether the problem of the complete utilization of natural supplies of fissionable materials can be solved solely by building reactors operating on thermal neutrons.

One of the fissionable isotopes of uranium is uranium-233, which does not occur in natural uranium. This isotope can be obtained artificially from thorium-232 by neutron capture. By the use of this isotope, it is possible to design a thermal-neutron reactor which will produce uranium-233 from thorium with a conversion factor somewhat greater than unity (1.1). Hence, a reactor of this type will not only produce energy but will also create the fissionable isotope uranium-233 from thorium. It is essential that the losses in the chemical treatment of the thorium should be considerably less than 10 per cent. In one

of its alternative designs, the above-mentioned boiling homogeneous reactor is based on this particular technique. This renders it possible to use thorium as raw material for atomic fuel. Thorium occurs fairly widely in nature, so that a "thorium cycle" of this sort is one of the interesting possibilities in the field of atomic power.

It is highly probable that countries having large supplies of thorium will find it advantageous to concentrate on the development of homogeneous heavy-water reactors with extensive uranium-233 breeding. The use of atomic fuel in a fuel cycle, as described in one of the papers by Canadian scientists, should be investigated further and may represent a substantial economy in the consumption of natural supplies of fissionable material. We note that uranium-233 may prove to be a very suitable fissionable material for intermediate-neutron systems. We shall not, however, deal with reactors of this type in greater detail, since in their physical characteristics they are intermediate between thermal-neutron reactors and fast-neutron reactors, so that their features are not so sharply defined as those of the extreme types. The main feature of these reactors is that they can be built in moderate sizes, which may be advantageous in some instances.

7. Fast-Neutron Reactors

There is no moderator in fast-neutron reactors. The fissionable material (uranium-235 or plutonium) is only slightly diluted with the coolant which is to carry away the heat. The core of a reactor of this type may be extremely small, and any increase in its size would be needed only to accommodate the heat-transfer system.

The coolant must be as poor a neutron moderator as possible, so that liquid metals are the natural coolants to choose for these reactors. The reactor reflector must also be made of material which does not slow down neutrons. Fast-neutron reactors have a number of physical characteristics which distinguish them radically from thermal-neutron reactors.

Firstly, with fast neutrons it is possible to cause fission in the main isotope uranium-238 (fission threshold 1 Mev). The process competing for neutrons with fission will be inelastic scattering in both the uranium and the various structural materials. Inelastic scattering softens the neutron spectrum and brings the neutrons into the region below the fission threshold. It is therefore desirable to minimize the part it plays in the chain reaction.

Secondly, the non-productive capture cross-sections of the materials decrease with increasing neutron energy more rapidly than the fission cross-sections of uranium-235 and plutonium. Thus, the ratio between fission and radiative capture for fast neutrons changes in favour of fission. Because of this, and the substantial reduction of the poisoning effects which are characteristic of thermal systems, a considerable proportion of the fission neutrons can be used for absorption in uranium-238 and subsequent conversion of uranium-238 into fissionable plutonium. This conversion may take place both in the reactor core and in the reflector, which is of course made of natural or even depleted uranium.

Experimental data were lacking at the outset of our research, and our estimates gave a reproduction factor of about 1.3–1.4. In fact, however, our expectations were exceeded. Experiments extending over two years on a plutonium research reactor and measurements obtained by other methods indicated that we could reckon on achieving plutonium-uranium systems with a reproduction factor of up to 2, or not less than 1.5 after allowing for all losses. Our experiments showed that for the hardest spectrum measured $\nu = 2.72$, for neutrons of 900 kev., $\nu = 2.9$, and for neutrons of 250 kev energy, $\nu = 2.6$. Our measurements of the contribution of uranium-238 fission to the reproduction factor indicated that it might be about 0.5.

The results obtained at the Argonne Laboratory on the EBR-1 reactor and published at this Conference and also the work done at Harwell agree with our own findings.

The data available in the USSR indicate that despite the need to load the reactors with large quantities of active materials, they may still have an extremely bright future, particularly when allowance is made for the fact that the thermal-neutron power reactors planned and under construction may in the future produce the necessary quantities of plutonium.

The development of fast-neutron reactors presents new and difficult technological problems, such as, for instance, the behaviour of structural and fissile materials in a flux of the order of 10^{16} neutrons per cm^2 per sec and a total flux-time of the order of 10^{23} neutrons per cm^2 per exposure. This means that in the course of an exposure almost all of the atoms of the structural materials would be knocked out of place.

Since experience so far has been limited to exposures thousands of times smaller than this, it is difficult on the basis of this experience to predict how the materials in fast-neutron reactors will behave. The burn-up of active materials will also considerably exceed the limits with which we are familiar in practice.

We anticipate that a certain percentage of the total uranium and plutonium charge will be converted into fragments. It is still difficult at present to predict what effect that will have on heat conductivity, volume and shape, and consequently on the working capacity of the fuel elements. Finally, the problem of extracting a very large quantity of power from the small volume of the core is not a simple one.

If it proves technically possible to produce fuel elements which will operate in conditions of high thermal stress and in intense neutron fluxes for as long as a year or more, and also to eliminate high losses in chemical processing, such reactors will not

only produce power but will also convert the main isotope of uranium, uranium-238, into the fissionable material, plutonium.

Thus, fast-neutron reactors afford the possibility of completely utilizing the natural supplies of all uranium, and not simply of uranium-235.

The actual economic conditions in each country call for an individual approach to the solution of the problems facing that particular country.

It is very difficult to predict at present which line of development will prove to be most appropriate for various times and various conditions, and because of this it is necessary to carry on the work on a broad front. The scale of atomic power development work is so great that every country has a vital interest in the experience and achievements of other countries.

Because of this, international co-operation in the field of atomic power will be of particular value in the development of this new and fascinating field of human activity.

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Design and Operating Experience of a Prototype Boiling Water Power Reactor

By J. R. Dietrich, H. V. Lichtenberger and W. H. Zinn,* USA

The development of a nuclear reactor in which steam is generated directly in the reactor vessel has involved theoretical and experimental studies on the boiling phenomenon and some experience with several test reactors. In order to gain experience under conditions of an operating power plant, a test facility, consisting of a reactor, whose heat rating is approximately 15,000 kw, and the associated control equipment necessary for continuous operation, was constructed at the National Reactor Testing Station by Argonne National Laboratory. Steam generated by the reactor either is vented to the atmosphere or is delivered to a turbine generator unit of 3500-kw electrical rating.

Since the problems connected with fuel element life and fuel element fabrication are adequately under investigation in connection with other projects, the fuel elements for this reactor were made of materials which allowed the easiest construction. This is an important consideration, because many of the problems under investigation in this test facility require a change in the dimensions of the fuel elements. Fuel elements which can be fabricated rapidly and cheaply and reprocessed are therefore an important item in the operation of the facility.

The material of construction of the fuel elements was aluminum. Corrosion experiments by Draley[1] and his co-workers have demonstrated that aluminum can have an adequately long life for reactor use in temperatures well over 200°C. For this boiling reactor test facility, an operating pressure of 300 psi was chosen, leading to a fuel element surface temperature of 215°C. Experience at higher pressures is highly desirable. The pressure chosen for the test facility was a compromise to permit fairly rapid and inexpensive construction.

The uranium used was enriched in U^{235} to about 90 per cent and was incorporated in plates to provide the heating surfaces.

The basic variables available in the design of a boiling reactor using enriched fuel are as follows: (1) U^{235} per unit volume of reactor core; (2) ratio of water to fuel element metal in the core; (3) plate spacing. The choices made for these quantities determine the geometric buckling of the reactor through the criticality relation. The shape and size of the reactor must then be chosen to be consistent with this geometric buckling.

FUEL ELEMENT

In the design of a fuel element for an enriched boiling reactor, not only must the quantity of U^{235} required for criticality be known, but the physical size of the reactor core must be specified. In such a reactor, leakage of neutrons to the exterior of the core is the mechanism by which the reduction in reactivity of the reactor and the amount of void in the core due to steam are connected. Therefore, the physical size of the core becomes an important factor in the design. In arriving at the dimensions of the fuel element for the test facility reactor, the following points have been taken into consideration:

1. The condition is imposed that the void coefficient must never be positive, including the case of the moderator being subcooled to room temperature. The void coefficient is defined as the fractional change in reactivity divided by the fraction of water volume occupied by steam.

2. The area of a diametral plane of the reactor tank is arbitrarily divided into two parts. The central area to be occupied by the fuel elements, called the core, is chosen as large as possible; the remaining annular area is just sufficiently large to provide for the downflow of water. This arbitrary division of the area of the reactor vessel is only necessary when natural circulation is employed. Of course, the annular water space also serves as a neutron reflector for the core.

3. The diameter of the core being determined, the condition that at room temperature the void coefficient is to be close to zero is used to determine the length of the reactor core. Too long a core does not give sufficient leakage to yield a negative void coefficient; too short a core will give too large a void coefficient at elevated temperatures.

4. Considerations of the flow of a mixture of water and steam through channels of height as determined above are used to specify the plate spacing. The plate spacing being determined, the concentration of ura-

*Argonne National Laboratory. Including work by S. A. Bernsen, R. A. Cameron, J. J. Dickson, R. O. Haroldsen, J. M. Harrer, W. C. Lipinski, F. D. McGinnis, M. Novick, R. L. Ramp, O. A. Schulze, J. A. Thie, G. K. Whitham, Argonne National Laboratory, and C. B. Zitek, Commonwealth Edison Company.

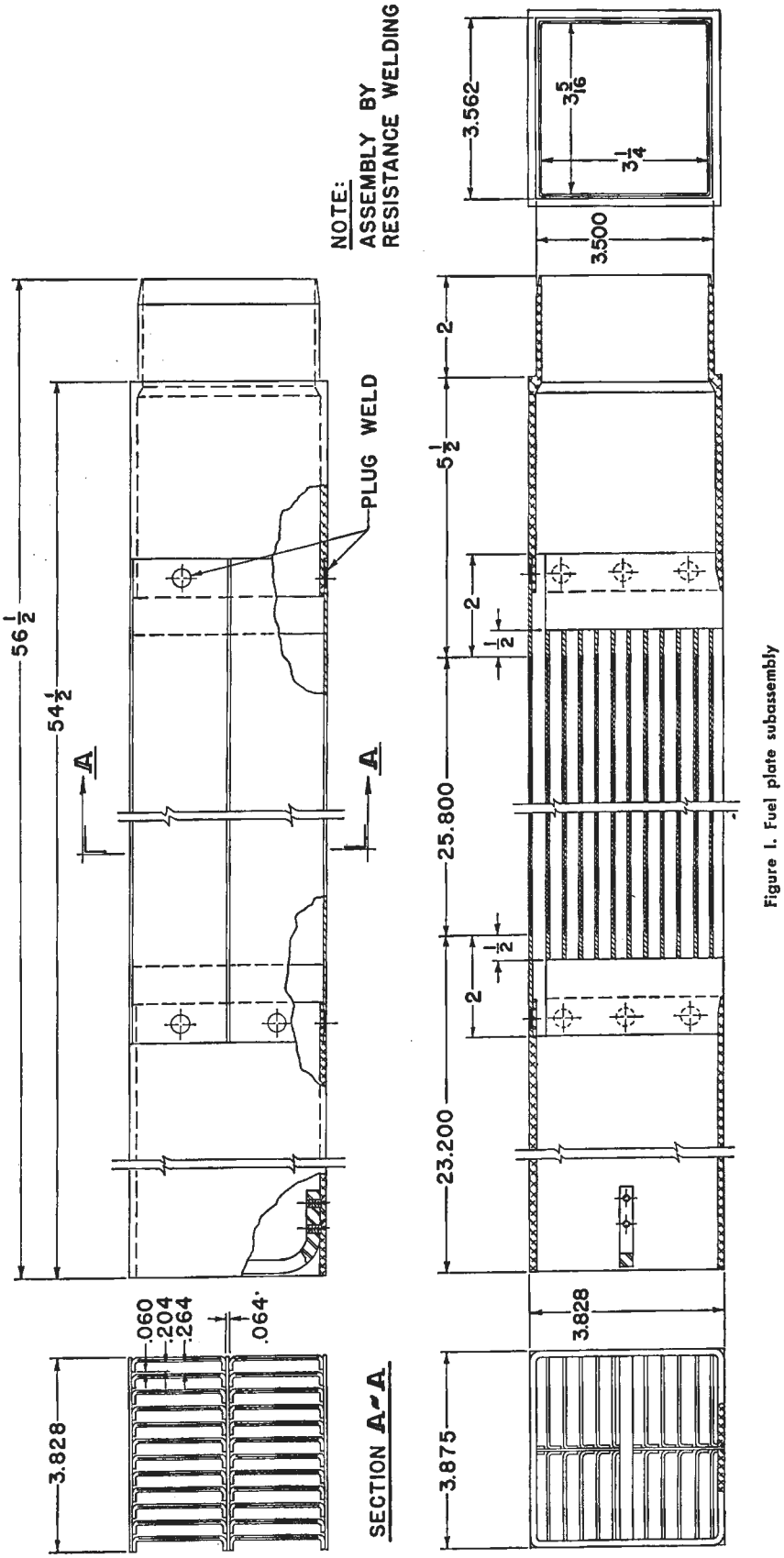

Figure 1. Fuel plate subassembly

nium in the plates follows directly from the critical size (mass of U^{235}) of the reactor.

In the following, a description of a typical fuel element subassembly is given. In the work which has been done with the test facility, the plate spacing and length have been varied. Each plate consists of a core of uranium-aluminum alloy jacketed on two sides with aluminum. Figure 1 gives the dimensions of the plates and indicates the manner in which 24 of them are assembled into a box-type fuel element subassembly. Because the formation of steam in the channels exerts considerable force on the plates, the unsupported span of the plate must be made fairly short. A cooling channel is 0.264 inch thick, 1.27 inches wide, and 26.8 inches long. In order to fit into the core support structure, the subassembly shown is fitted with a bottom extension and a top section as illustrated in Fig. 1. By changing the length of the top extension, the heated length of plate can be changed without affecting the clamping arrangement of the core structure. Also, it provides above the heated fuel plates a tube which, in operation, is filled with a mixture of steam and water and, therefore, tends to increase the circulation rate of water through the core. Figure 2 is a photograph of the partially assembled subassembly. Several concentrations of uranium in the fuel plates are available. The lowest concentration provides 137 grams of U^{235} per subassembly, the highest, 233 grams of U^{235} per subassembly.

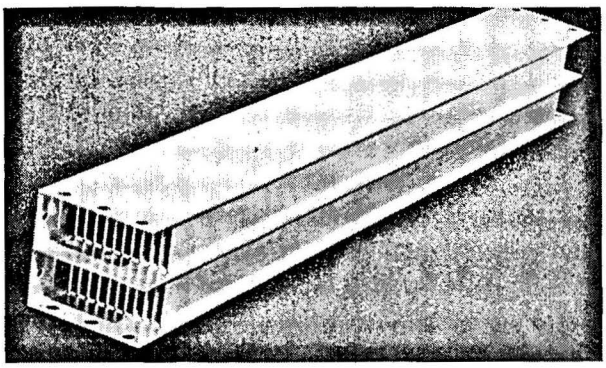

Figure 2. Partially assembled subassembly

The heavily loaded fuel elements are used to adjust the void coefficient of the reactor. Placing heavily loaded elements at the periphery of the reactor increases the leakage and, therefore, can increase the negative void coefficient. In order to determine the void coefficient, it is necessary to carry out transient experiments of the kind discussed in the paper by J. R. Dietrich, *et al.*[2]

REACTOR CORE ASSEMBLY

Since this is a test facility, it was necessary to make arrangements to permit complete change of not only the fuel elements, but of the whole reactor core assembly, including the control rods. The reactor vessel is a stainless steel tank, 15 ft 11¼ in. high and 52½ in. internal diameter, constructed of stainless steel plate, ¾ in. thick, conforming to the code for unfired pressure vessels. The core assembly is carried on a framework which is supported in the reactor vessel from a fixture which is bolted to the reactor vessel at the top, just below the flat lid closure. This construction permits removing the whole core assembly even after the reactor has been operated and core materials have been activated. Figure 3 is an illustration of the arrangement of the core assembly in the reactor vessel. Figure 4 is a cross section through the core, giving the dimensions of the four blade-type control rods and the central cross control rod. Cadmium, boron, and hafnium are the absorbing materials in the control rods. Figure 5 is a photograph of the core assembly before installation in the reactor.

The control rod drives are situated above the dense concrete shield plug which is rolled over the top of the reactor. In order to change fuel in the reactor, it is necessary to disconnect the control rod drive rods, roll back the shield plug and remove the lid of the reactor vessel. To make this possible, the reactor vessel and the pit in which it stands are flooded with water. The insulation on the outside of the reactor vessel is of a type which does not deteriorate due to immersion in water.

Figure 6 is a schematic diagram of a single control rod drive. These drives are so arranged that a control rod at any time can become a rapidly operating shutdown rod. A control rod which is driven by a shaft sealed through the wall of a pressure vessel, such as this, has two severe difficulties. First, the internal pressure acts to lift the control rod out of the core, and, secondly, the friction of the packing gland opposes any motion of the rod. In the present design, the lifting force due to the steam pressure in the reactor vessel is balanced out with a steam cylinder and piston in which the drive rod terminates. Steam for the balancing cylinder is supplied from the reactor vessel.

A large force for moving the control rod is supplied by a pneumatic piston which exerts a force of 800 pounds to drive the rod. This is much larger than any friction force in the packing gland. When the rod is in use as a regulating rod, the pneumatic cylinder holds a stop on the drive shaft in contact with a large nut which straddles two lead screws, which are driven in synchronism. Driving the nut up or down imparts the same motion to the control rod. Rapid insertion of the control rod is achieved when the air pressure is switched to the side of the pneumatic cylinder which drives the rod down.

PLANT LAYOUT

Figure 7 is a flow diagram for the whole plant. Three principal circuits are indicated.

The power circuit consists of a feed water pump which can pump into the reactor either condensate returned from the condenser or freshly de-ionized

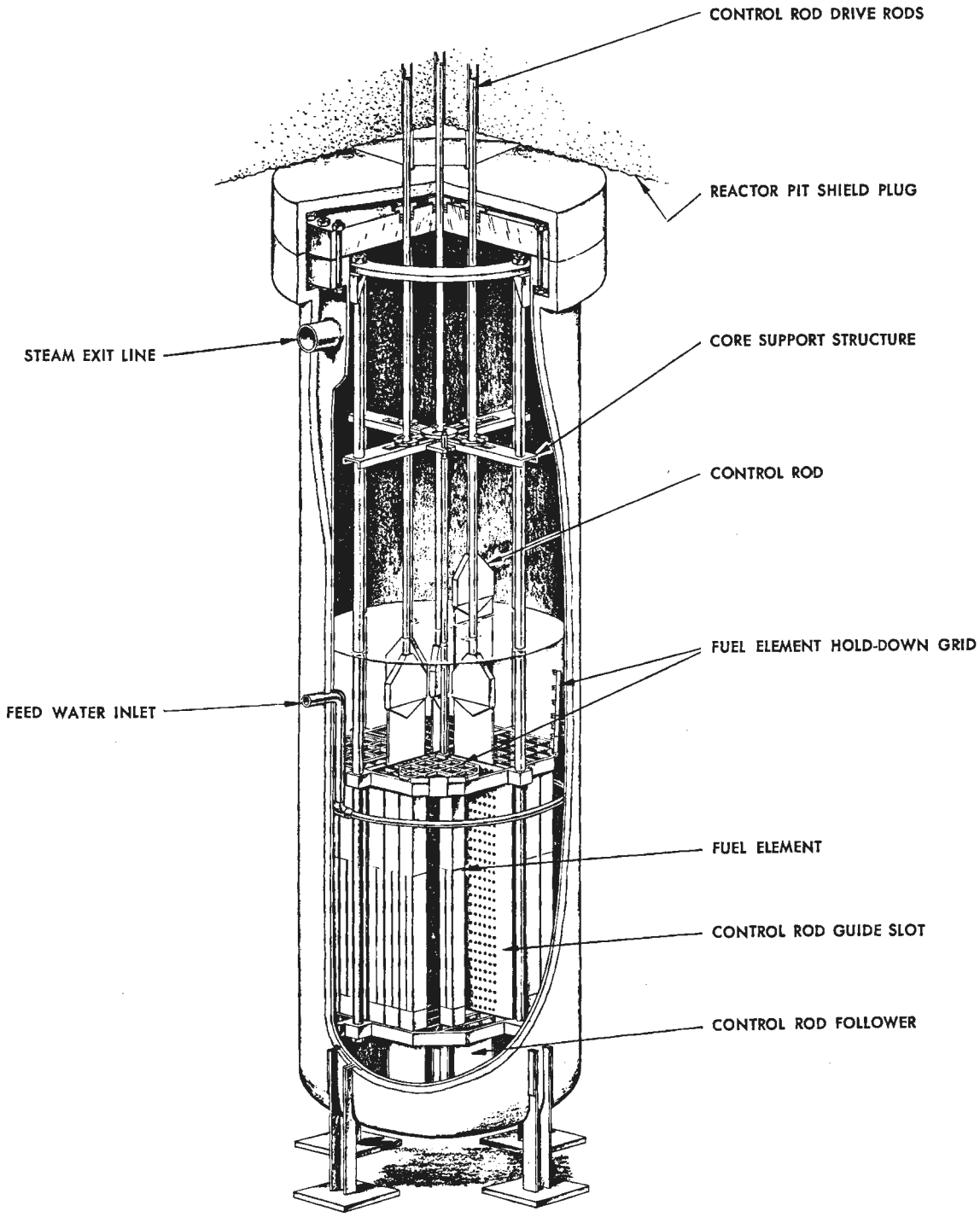

Figure 3. Core assembly in reactor vessel

water from the makeup system. The feed water is discharged into the annular downcomer of the reactor. This helps to cool the downcomer water and promotes the natural circulation in the reactor vessel. A large volume in the upper part of the reactor vessel contains only steam. This large volume promotes the separation of steam and water and acts as a cushion, preventing rapid pressure changes. Steam from the reactor vessel is carried by a six-inch line either to the turbine or to a vent to the atmosphere. Valves suitable for these operations are shown. The condensate from the condenser is pumped to the feed water supply tank.

The second circuit consists of a pump and electrical heaters. This permits pressurizing the reactor vessel without operation of the reactor. It is useful for testing for leaks and also is used to raise the temperature so that the reactor will not have to be run with the control rods inserted most of their length.

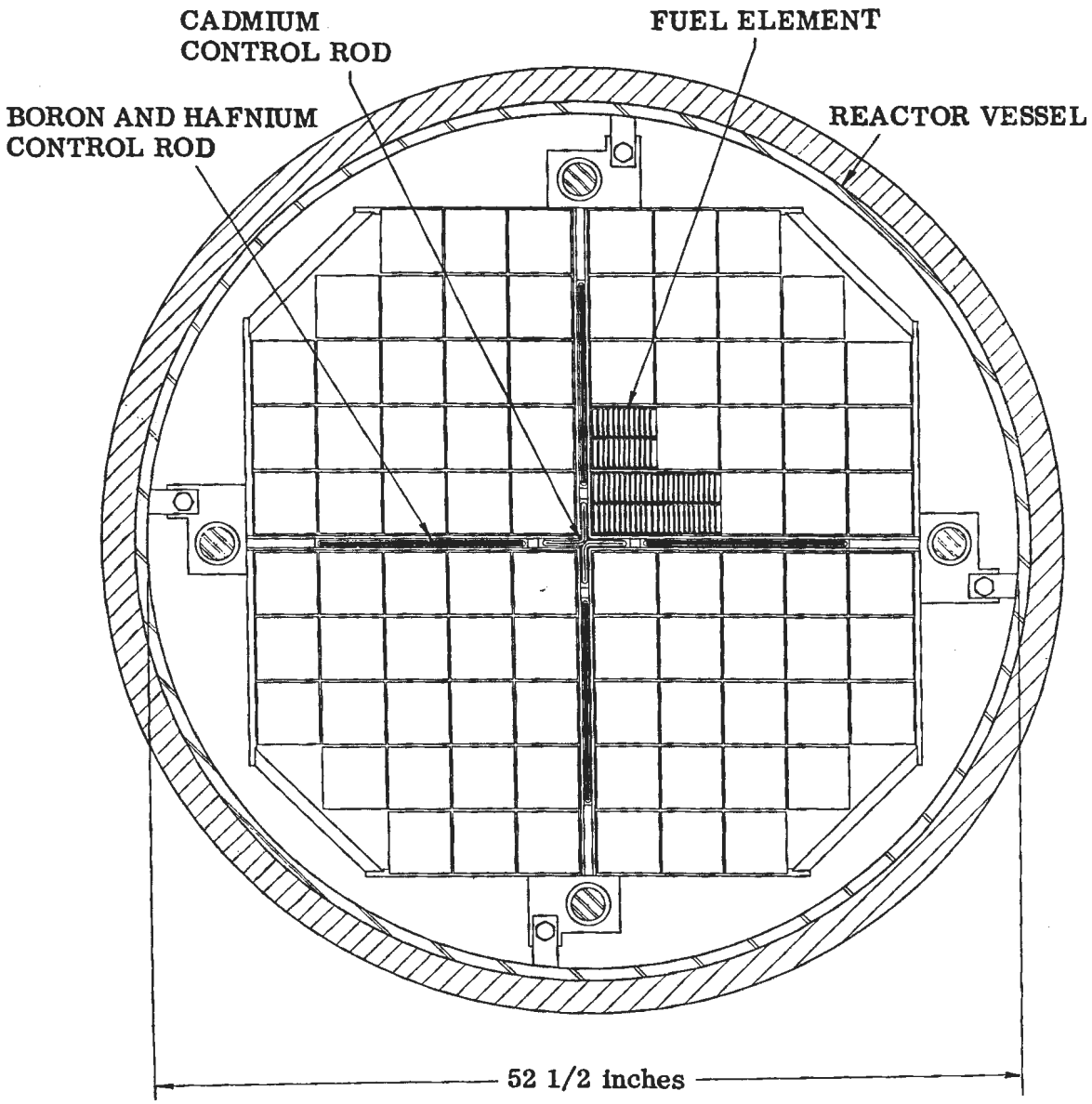

Figure 4. Cross section through reactor core

The third circuit is the cleanup circuit and consists of filters and ion exchange columns. In operation these will become quite radioactive; consequently they are installed in the fuel element storage canal so that the water of the canal will provide the necessary shielding. Figures 8 and 9 give the general arrangement of the equipment in the plant.

INSTRUMENTATION

Since this facility is also used for transient testing of reactor cores, the control room is situated some distance from the reactor itself. All indication of operating conditions, therefore, must be made by electrical means. The level of the water in the reactor, which normally is about 4 feet above the active core, is closely controlled. Instruments for recording the level of the water are installed in a standpipe exterior to the reactor vessel and in the shielded pit which contains the pumps and other auxiliary equipment. Level control is by means of a float-type level controller installed at the operating level. A television camera is arranged so that by means of it the level in a sight glass can be read directly. The television camera also can be used to read directly a Bourdon-type pressure gauge which measures the steam pressure in the reactor vessel. In the control room the rate of flow of feed water as well as the steam flow are recorded.

Ionization chambers recording neutron flux are installed in tubes which extend close to the core, but outside the reactor vessel. Gamma-ray intensity at the top of the reactor vessel is recorded. The neutron intensity is measured by three types of instruments differing in their time constants. For normal operation, an amplifier recorder system having a response time of 0.05 second is used. In order to observe transient effects, a Brush amplifier recorder with a time constant of 0.002 second, or a Heiland

galvanometer-oscillograph with a time constant of 0.0004 second, is used for neutron intensity.

STEADY OPERATION

It has been possible to operate the reactor under conditions which are stable for all pressures from atmospheric to 300 psi. A necessary requirement for reasonably smooth operation is that the amount of

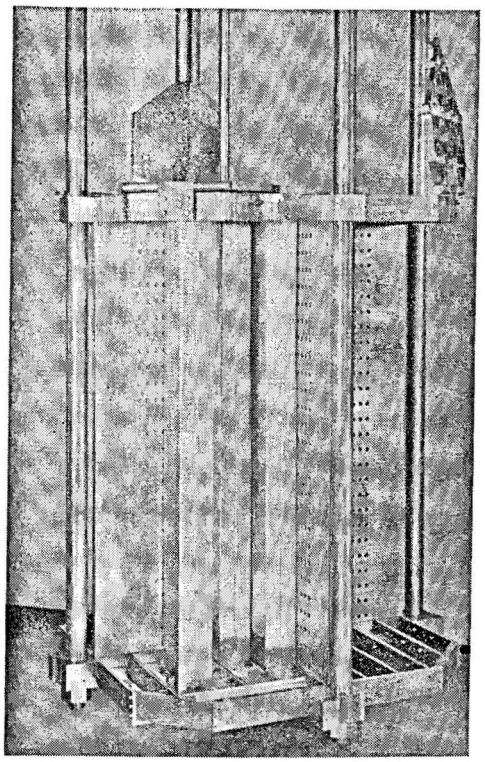

Figure 5. Core assembly before installation in reactor

reactivity corresponding to the steam void not be excessive. In all operations, however, the power level is subject to rapid variations. These clearly are connected with the turbulence created by the formation of steam. The degree of roughness of the power is directly related to the excess reactivity which has been subtracted by the formation of the steam voids and replaced by withdrawal of the control rods.

A series of experiments has been performed to determine the relationship between the excess reactivity in voids and the power density, and this has been done for a number of pressures. It has been observed that for any given pressure the degree of roughness of the power varies with the excess reactivity in steam voids. If the excess reactivity is raised to too high a figure, oscillations in the power become apparent, and further increases in excess reactivity in steam may bring small excursions in the reactor power.

Figure 10 contains curves of power density versus excess reactivity in steam for two different reactors (II and III). Reactor III was that illustrated by Figs. 4 and 5, whereas reactor II, although it used the same vessel, was considerably smaller and had

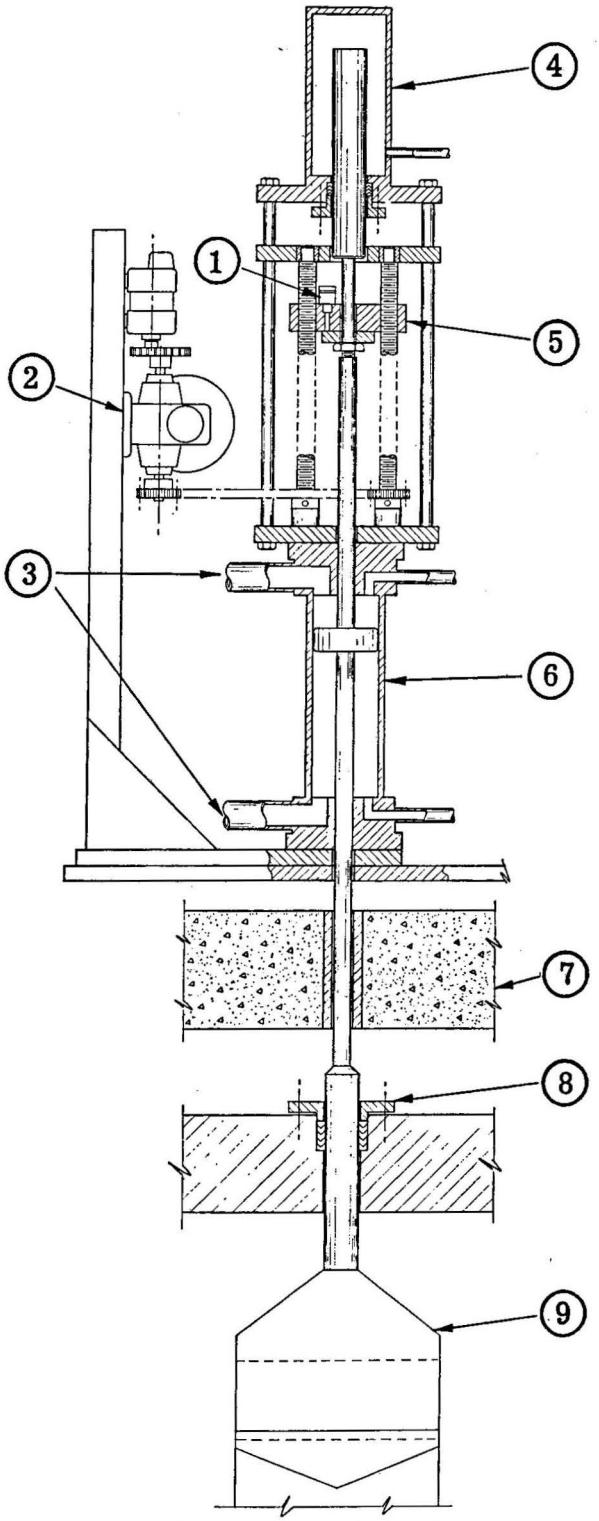

Figure 6. Diagram of control rod drive. (1) Control switch; (2) Drive motor; (3) Compressed air; (4) Balancing cylinder; (5) Travelling nut; (6) Pneumatic cylinder and piston; (7) Shield plug; (8) Packing gland; and (9) Control rod

a higher (negative) steam coefficient of reactivity. Consequently, a given reactivity in reactor III corresponds to a higher steam content per unit volume than the same reactivity in reactor II, and the power density of reactor III is correspondingly higher.

Figures 11a and 11b are reproductions of sections of the power traces of reactors II and III, as obtained by a neutron-sensitive ion chamber and fast recorder. Both reactors were operating at 300 psig and at roughly the same power density (about 22 kilowatts/liter). This power density corresponded to 2.6 per cent reactivity in steam for reactor II, and about 1.5 per cent for reactor III. The short-period power fluctuations, which amount to as much as about 15 per cent for reactor II, are not greater than 4 per cent for reactor III. In neither case did these power fluctuations produce detectable variations in reactor pressure or steam flow.

The curve for reactor II in Fig. 10 includes points at several operating pressures. There is no great variation with pressure in the power density attainable from a given addition of reactivity, at least within the pressure range investigated. This insensitivity to pressure is qualitatively consistent with the results of laboratory boiling experiments.[3] Although the energy content per unit volume of the steam increases almost linearly with pressure, the velocity of slip of the steam past the water in the fuel element decreases with pressure. The steadiness of operation of the reactor increases markedly with pressure, however, and it is therefore feasible to operate the reactor with more reactivity in steam voids and consequently at higher power at high pressure. In Fig. 10, the highest plotted points for the 75 psi and the 150 psi conditions represent approximately the upper limits of satisfactory operation at these pressures. At 200, 250, and 300 psi, however, the maximum reactivity added was limited by the amount available in the reactor; at these higher pressures the curve could have been extended to somewhat higher power density if additional reactivity had been available.

Table 1 summarizes the results of power density studies which have been obtained for three reactor cores which differed in size, plate spacing, and the U^{235} loading of the plates.

Item 1 in the table shows that the second reactor had the plate spacing increased by 83 per cent with respect to reactor I. From item 2, it is seen that this increased the ratio of water to metal in the core by the factor 1.26. The effective neutron lifetime is not changed much, as is indicated in item 4.

As in all water reactors, the temperature coefficient is sizable and represents a troublesome problem for control. In item 6, the over-all reactivity loss observed in heating from shutdown to near operating temperature is given. It should not be assumed that the temperature coefficient of reactivity is at all constant over this range of temperature.

In item 7, the calculated void coefficients for the reactors are given. A certain fractional void (less than 10%) was assumed. The corresponding re-

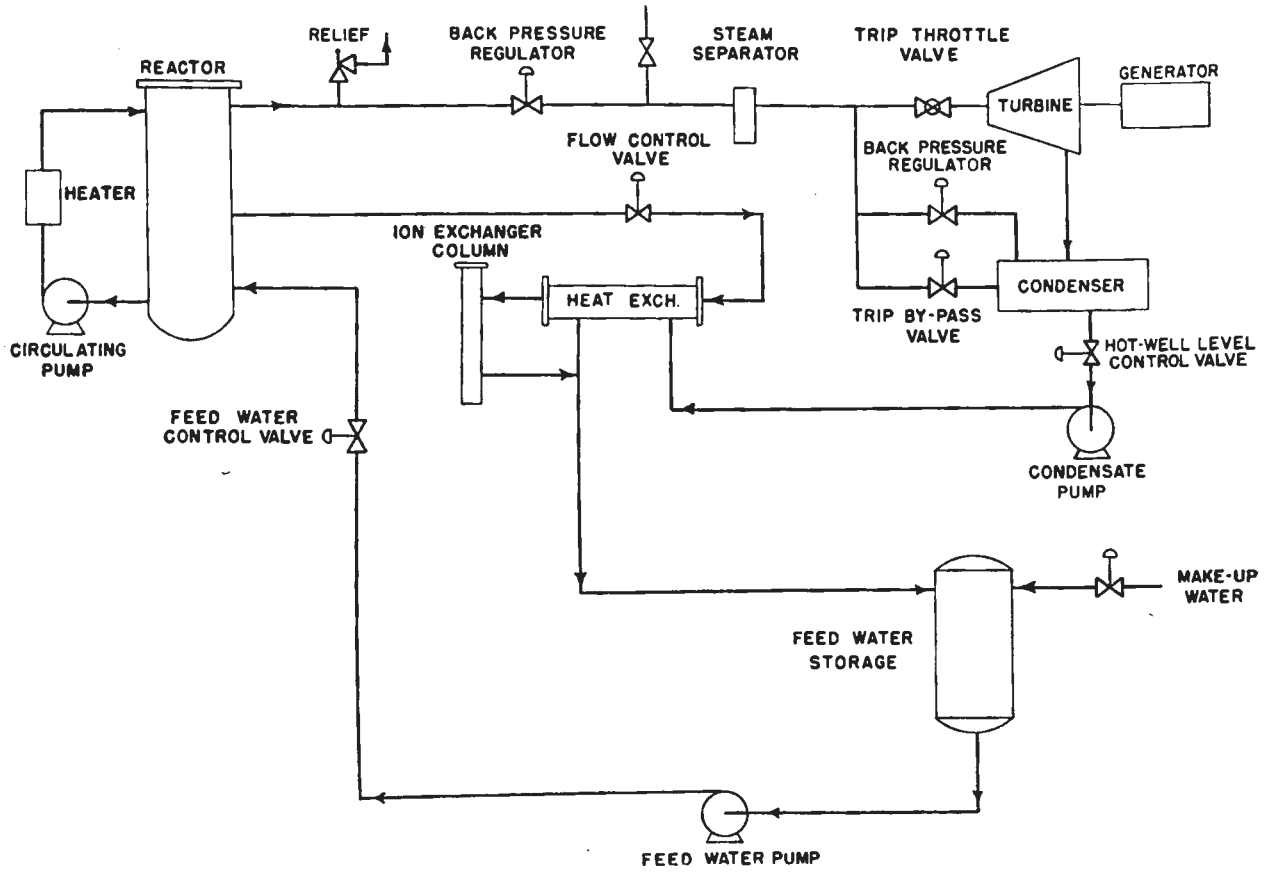

Figure 7. Flow diagram for complete plant

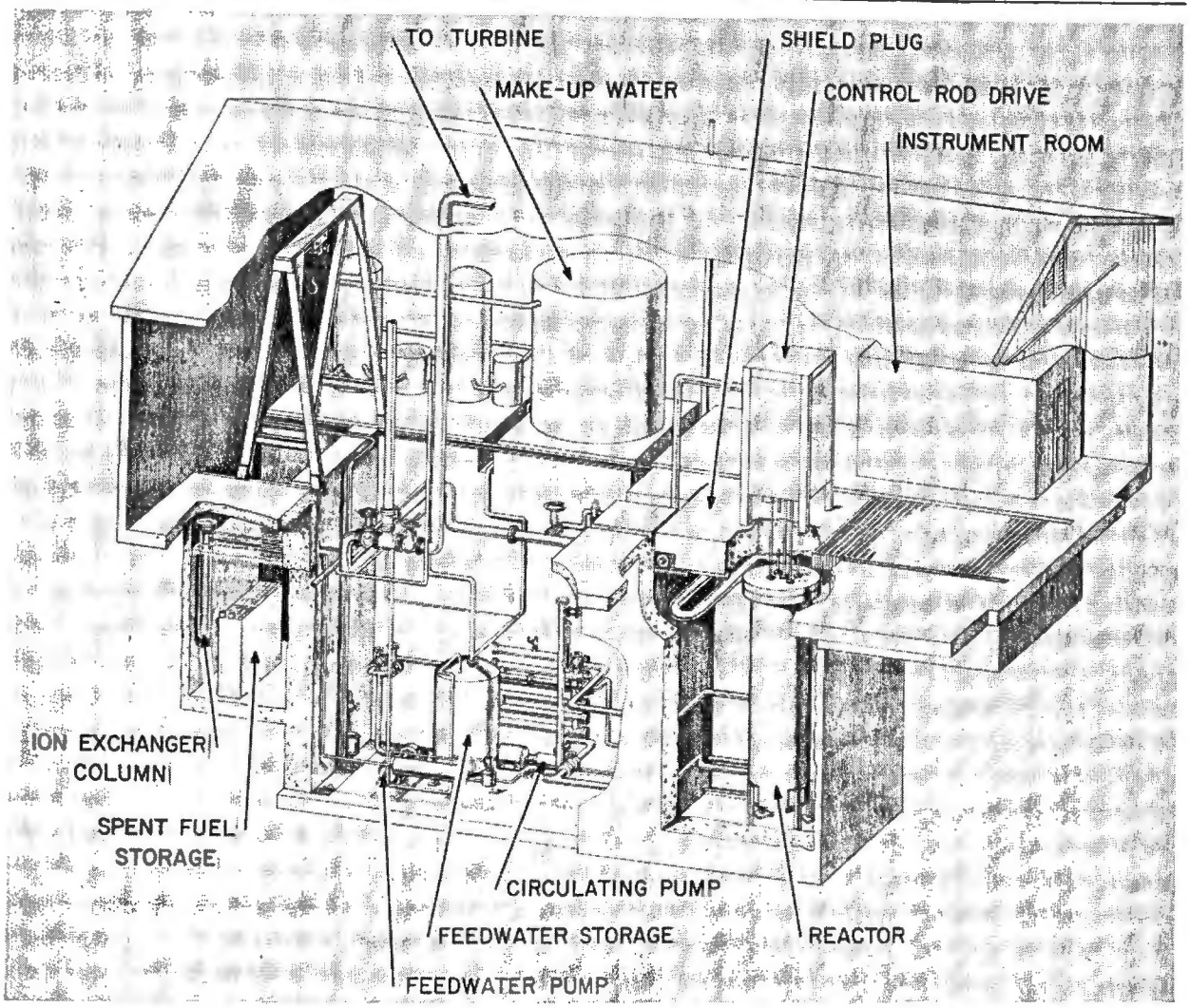

Figure 8. Reactor building

activity change then was computed and recorded.

In item 8, the power density for per cent reactivity subtracted by the voids is given. This quantity is measured by the total power of the operating reactor and by the amount of reactivity added by the calibrated control rods. It would be of great interest to be able to measure directly the actual steam void in the operating reactor.

Laboratory experiments have been done in which the fractional void in a heated pressurized channel has been measured as a function of operating power.[3] The void fraction was measured by the attenuation of γ rays passing through the channel. It can be said that the quantity [power density/$\%k_{eff}$ (void)] in the reactor is not in disagreement with the above experiments and the calculated void coefficients.

The void coefficient of reactivity apparently is the principal factor affecting the performance of the several reactor cores so far studied. Obviously from the point of view of obtaining the maximum power density the smallest void coefficient is wanted; considerations of the degree of inherent self-limitation for power excursions will play a large part in the choice of void coefficient for any practical power reactor.

It is interesting to note the power level at which a typical fuel element operates. The power density of 22 kw/liter, which resulted in the power traces of Figs. 11a and 11b, may be taken as representative. The active section of the fuel element of reactor III has a volume of 6.26 liters, and therefore the average fuel element delivers 138 kw. Fully loaded, the reactor contains 87 fuel elements (one fuel element position is reserved for the installation of a start-up source) and hence, at this power density, the total power is 12,000 kw. The surface area of heated fuel plate in an element is 12,000 cm^2. The average heat flux is therefore 3 cal/cm^2 sec, and the maximum about 7 cal/cm^2 sec. It is clear that in boiling reactors which employ only natural circulation of the water the heat rates attained are not high; the limiting factor on power output is the rate of flow of steam out of the reactor core rather than the heat flux from the fuel plates. This characteristic is, of course, favorable from the standpoint of reactor safety; at the same time power densities in the range 20 to 30 kilowatts

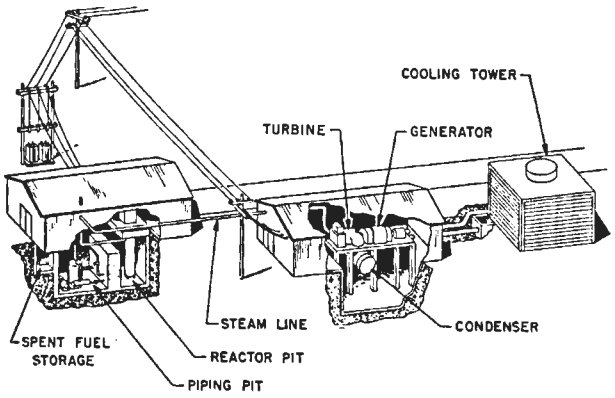

Figure 9. Diagram of complete plant

per liter of core (or 30 to 40 kilowatts per liter of water in the core—see Table I), which have been experimentally attained, are sufficiently large for many power reactor applications.

The actual loading of the reactor is complicated by the fact that, for the experimental program on attainable power densities, core sizes less than the maximum must be used because the extra fuel allowance required for burnup in continuous operation is not wanted. It also is complicated by the fact that the experiments must be performed either without appreciable xenon poison or with saturated xenon poison. Great care in loading has been used to avoid arriving at erroneous conclusions, because in one experiment the full height of the reactor was used whereas in another the effective height was reduced by operating with the control rods partially inserted.

For continuous operation, where burnup will steadily reduce the available excess reactivity, burnup plates which incorporate a certain amount of boron are affixed to the subassembly of Fig. 1. The time interval between successive additions of fuel can be substantially increased by the use of such burnable poison plates.

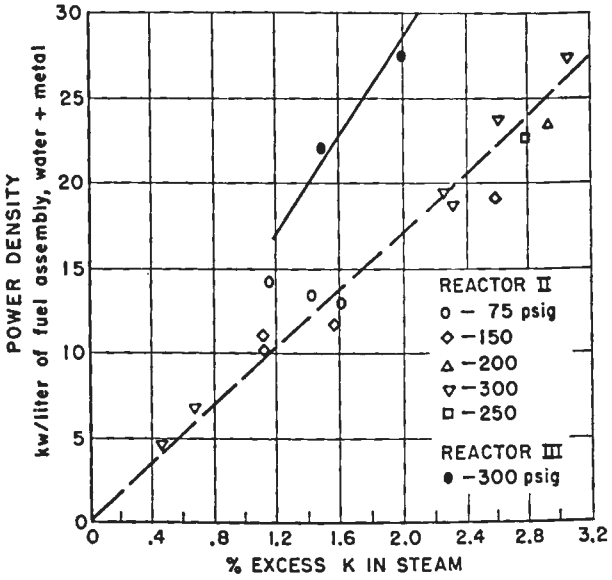

Figure 10. Effect of reactivity on power density

STABILITY UNDER VARIOUS OPERATING CONDITIONS

As was indicated earlier, reactivity can be added safely to this boiling reactor in amounts which are dependent upon the pressure of operation. Experiments to date indicate that at 300 psi with a water channel width of 0.324 inch, steady power operation is obtained even with 3.2% reactivity subtracted by the steam voids. Experiments at lower pressures clearly indicate that there is an amount of excess reactivity above which instability sets in. This is not a matter of great concern, because as more and more reactivity is added the instability manifests itself by oscillations in the power level. These oscillations have a period of about 0.4 second and as successive amounts of reactivity are taken up by the steam voids, the amplitude grows. There is a considerable region of added excess reactivity where these oscillations

Table I. Characteristics of Two Boiling Reactors

Item	*Reactor I*	*Reactor II*	*Reactor III*
1. Plate spacing in subassembly	0.177 in	0.324 in	0.324 in
2. $\frac{\text{Volume of Metal}}{\text{Volume of } H_2O}$ in core	0.63	0.49	0.36
3. Number of fuel plates per subassembly	18*	10*	24
4. Effective neutron lifetime	6.5×10^{-5} sec	7.5×10^{-5} sec	7×10^{-5} sec
5. U^{235} per subassembly	138.6 gm	93.4 gm 157.3 gm	137 gm 233 gm
6. Temperature coefficient of reactivity, 27° to 182°C	3.5% k_{eff}	1.16% k_{eff}	0.8% k_{eff}
7. Void coefficient (calculated)	—0.24% k_{eff}/% steam	—0.10% k_{eff}/% steam	—0.07% k_{eff}/% steam
8. $\frac{\text{Power density}}{\text{Per cent } k_{eff} \text{ (void)}}$	4.1 kw/liter/% k	8.6 kw/liter/% k	13 kw/liter/% k

*These subassemblies were of a different design than that shown in Fig. 1. The width of the box was less, which accounts for the smaller number of plates.

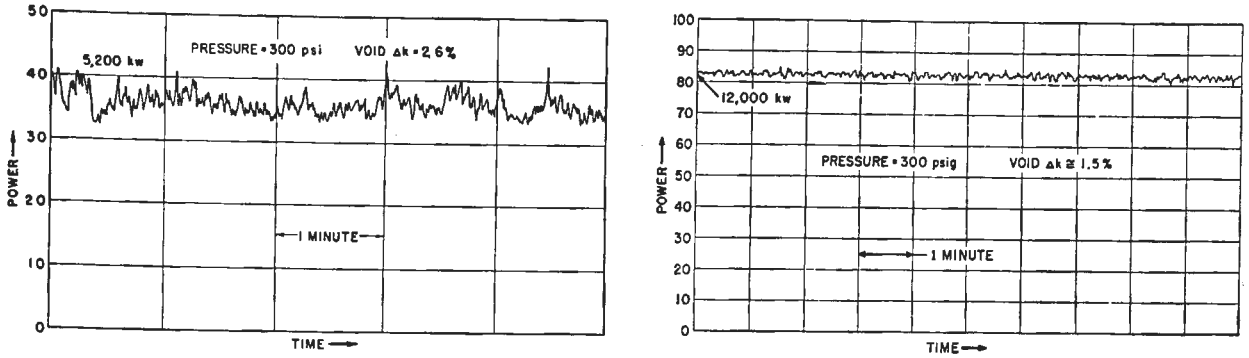

Figure 11. Reproductions of typical power records from two different boiling reactors: Figure 11a. reactor II. Figure 11b. Reactor III

are stable and can be made to increase in amplitude only by a further increase in reactivity.

Experiments have been performed in which reactivity has been added at a fixed rate by the withdrawal of the control rods. In Fig. 12 the amount of excess reactivity added by the rods is plotted versus time. For the more or less straight portion of this curve, the rate of addition was 0.075% k/sec. In Fig. 13, the behavior of the reactor as shown by the neutron power trace is shown. At the start of the withdrawal of the control rods, the power level was 485 kw and the reactor pressure was 300 psi. In approximately 40 seconds, 2.5% excess reactivity was added. The power level rose with an approximately 6.5-second period and leveled off at 5400 kw without any indication of unstable operation.

In a second experiment, the addition of reactivity took the form shown in Fig. 14. An excess reactivity of 2.6% was added in a time of 160 seconds. In this case, the pressure was 300 psi and the power of the reactor at the beginning of the experiment was only 1.2 kw; in other words, the amount of void volume in the reactor due to steam formation was essentially zero. Reactivity addition was at the rate of 0.016% k/sec. The reactor power increased with an approximate period of 3.3 seconds, as is shown in Fig. 15. The power leveled off at 4300 kw.

For a boiling reactor which is producing steam at an appreciable rate, there has been considerable conjecture as to the consequences of a sudden closure of the steam exit line. A boiling reactor connected to a turbine would experience this if the trip throttle valve of the turbine should close. For a boiling reactor, sudden closure of the steam delivery line is not possible if a trip valve is arranged to bypass the steam to the condenser in all cases when the trip throttle valve to the turbine closes. Nevertheless, it was thought worthwhile to perform experiments in which the steam delivery line suddenly was closed. Figure 16 indicates what happens. In this case, the reactor was operating at 150 psi and at a power level of 1400 kw. Under these circumstances, 2% reactivity had been subtracted by steam voids. Closing the valve initiated an increase in power which then leveled off at a little over 3000 kw. Figure 17 is a replot of this same experiment showing the behavior of the steam pressure in the reactor vessel. As was expected, the pressure rose steadily. But no instability was introduced in the power by the sudden closure of the valve in the steam delivery line. This experiment was repeated a number of times and indicated that there is no particular hazard connected with this circumstance.

The behavior of the reactor under circumstances where the pressure for some reason is allowed to rise until the safety valves open has been investigated. Normal safety valves open at a pre-determined pressure and close again when the pressure has fallen by a pre-determined amount. The safety relief valves in use for this experiment were set to open at slightly over 250 psi and closed again when the pressure had dropped approximately 20 psi. This caused no difficulty at all in operation. The relief valve would open, the pressure would fall with a small corresponding fall in reactor power, and then when the relief valve closed the reactor power would readjust itself to approximately its initial value.

It is clear, however, that if a relief valve is opened and allowed to remain open a considerable length of time, resulting in a large drop in pressure, the reactor power will also fall by a large amount. The volume of the steam voids in the vessel will increase materially and, as the temperature of the reactor water falls, the amount of excess reactivity subtracted by the void will steadily increase. This eventually should lead to unstable operation according to the earlier

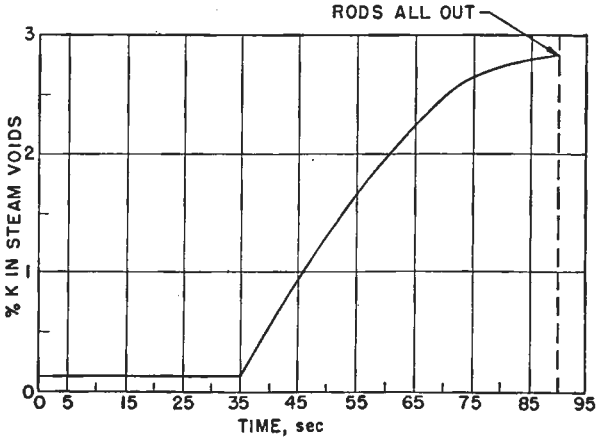

Figure 12. Reactivity addition during run of Fig. 13

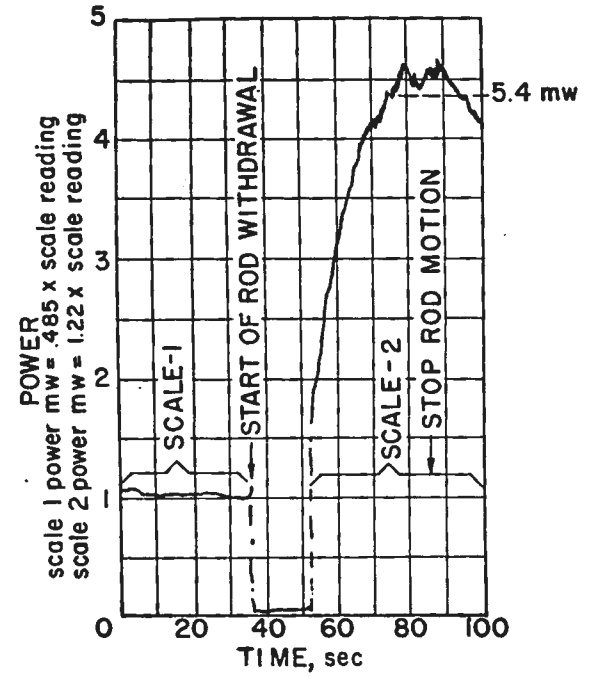

Figure 13. Power variation during reactivity addition: initial power 485 kw

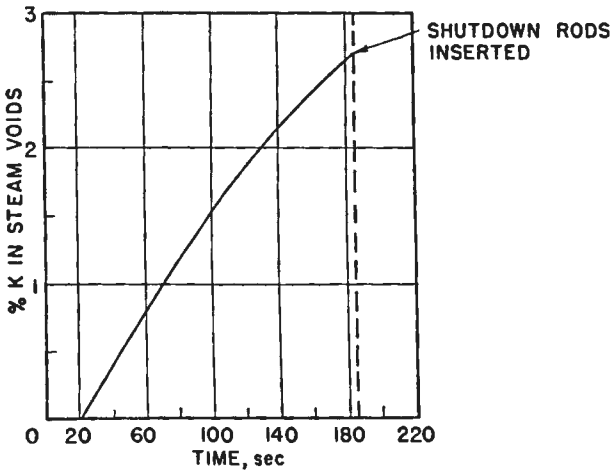

Figure 14. Reactivity addition during run of Fig. 15

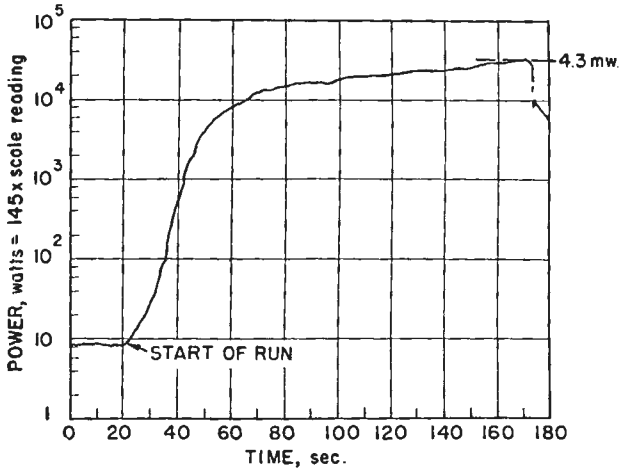

Figure 15. Power variation during reactivity addition: initial power 1.2 kw

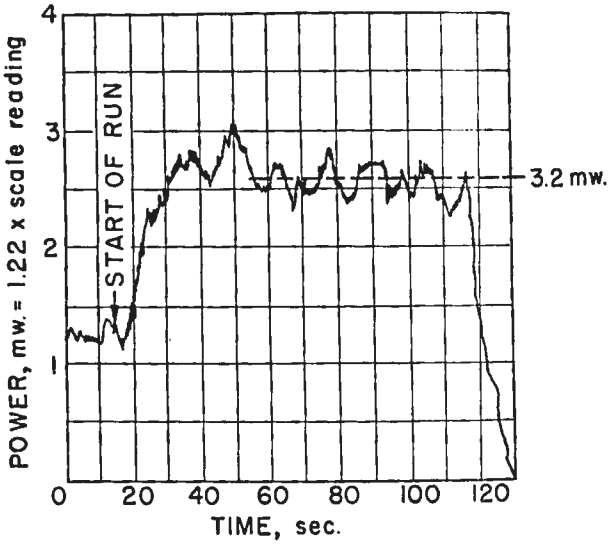

Figure 16. Power variation after sudden closure of steam valve

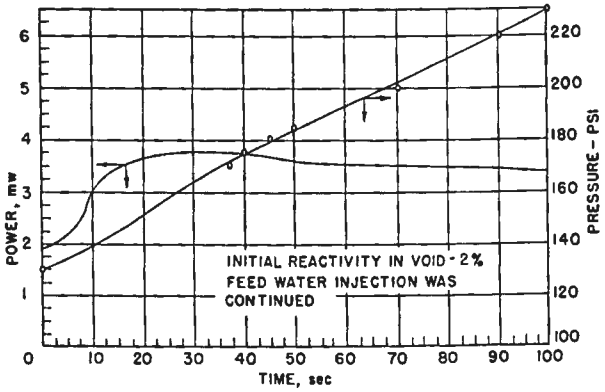

Figure 17. Power and steam pressure variation after sudden closure of steam valve

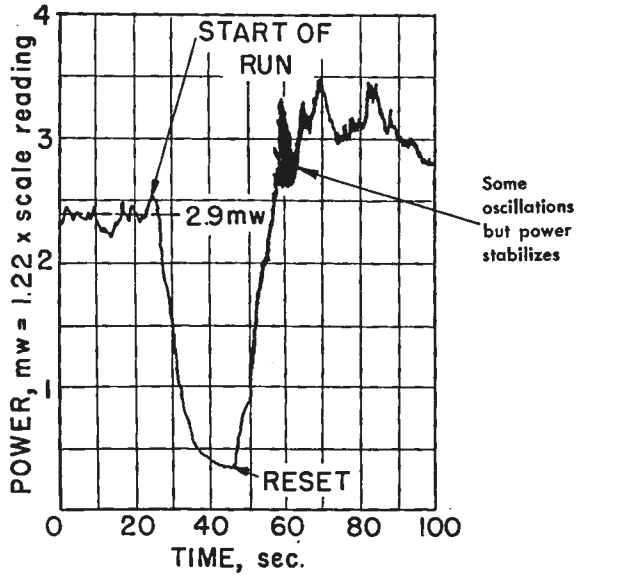

Figure 18. Power variation caused by sudden opening of steam valve followed by reset

description of the behavior when an excessive amount of reactivity is subtracted by the steam voids. Figure 18 is a power trace of an experiment of this type. The reactor was operating at 2900 kw and at a pressure of 265 psi. Reactivity amounting to 1.8% had been subtracted by the steam voids. A control valve in the steam line suddenly was opened to the atmosphere. The reactor power fell rather rapidly to a few hundred kilowatts. The reactor pressure decreased steadily and, when it had fallen to 190 psi, the control valve was closed rapidly. The reactor now had appreciably more reactivity to be subtracted by steam voids than it had initially. The power rose rapidly to a value in the neighborhood of 3000 kw, a value corresponding to the new void situation. It should be noted that the additional reactivity which was required to be subtracted by the steam voids had been acquired because the temperature of the reactor had fallen. The reactor now stabilized at this new power level, but it was clear that it was not far from the power oscillation situation described earlier. In other experiments, with a more vigorous blowdown of the reactor, the oscillation region actually was entered. These experiments indicate that in a boiling reactor any situation which causes a rapid decrease in reactor pressure should call for the immediate insertion of the shutdown rods. The shutdown does not have to be very fast because the flashing into steam of the moderator and the cooling of the reactor cannot be a very rapid event. It should, however, be a normal operating procedure to shut off the reactor whenever there is more than a pre-determined decrease in reactor steam pressure.

The operation of a boiling reactor differs in a very marked manner from the operation of non-boiling reactors. In non-boiling reactors, the motion of the control rod ordinarily results in a change of power of the reactor with an exponential period. The amount of reactivity added by the motion of the rods must be judged carefully in order to prevent short power periods which rather easily can produce temperature effects of a deleterious nature in the reactor. For the boiling reactor, addition of reactivity by the withdrawal of rods does not result in an exponential increase to another power level. The amounts of reactivity which can be added without undue concern for the boiling reactor are considerably greater than the amounts which can be treated in a similar way for a non-boiling reactor. It is only necessary to regulate the changes so that the region of oscillating instability is avoided. This it has been very easy to do during many hours of power operation.

In practice, once the conditions of operation for the reactor have been established, the startup of the reactor is as follows. In order to avoid a situation where reactor power changes are introduced when the moderator is subcooled to room temperature, electric heaters are employed to raise the pressure in the reactor vessel to an arbitrary value, say 40 psig. Use of auxiliary heating is not necessary; it merely avoids using the reactor in a non-boiling manner as a water heater. The reactor is then made critical and immediately brought to a power level of approximately a thousand kilowatts. This is high enough so that boiling takes place in the channels of the core. The energy produced heats the water moderator while the steam bubbles in the core are condensed, probably in the cooling channels. Meanwhile, the pressure in the reactor vessel rises steadily, but the power level remains constant and smooth. As the temperature and pressure rise, it is necessary to slowly withdraw the control rods in order to add the reactivity lost because of the increased temperature. When the reactor pressure approaches the operating point, the control rods are further withdrawn to cause the addition of the operating amount of reactivity which then is subtracted by the formation of steam voids. The power rises accordingly but is smooth all the while. When the operating power level and pressure are reached, the steam delivery valve is opened and the machine is in normal operating condition. Figure 19 is a photograph of the reactor delivering steam to the atmosphere when operating at 8000 kw. For such venting to the atmosphere, it is necessary to feed deionized water to the reactor at the required rate.

No attempt has been made to deliver the feed water at any particular temperature to the reactor. Lowering the temperature of the feed water, of course, raises the reactor power. Turning the feed water pump on and off brings a corresponding increase and decrease in reactor power, but the reactor accommodates itself to these changes in a completely stable manner. Stopping the feed water flow while keeping the reactor at power lowers the level of water in the reactor vessel. Boiling off most of the upper reflector, keeping the control rods in a fixed position, reduces the power to a very low value.

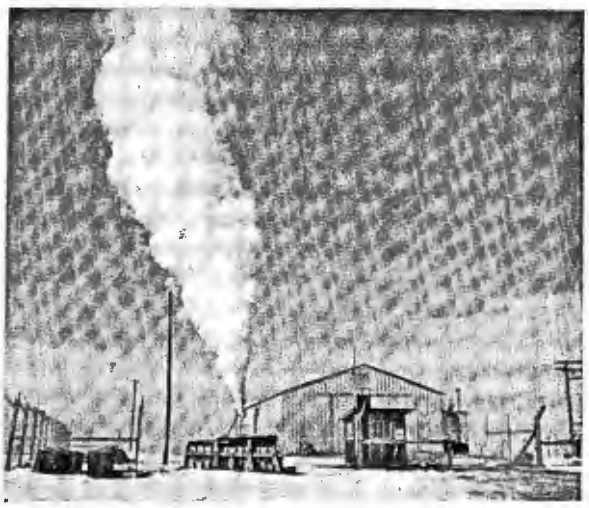

Figure 19. Reactor delivering steam to the atmosphere

STEAM QUALITY

A boiling reactor will deliver wet steam to the turbine. Steam separators must be used. The large

steam volume in the top of the reactor vessel itself acts in a very effective way to cut down the entrainment of water. Measurements on the quality of the steam issuing from this reactor with a steam calorimeter show a value of 98% or better. The radioactivity of the steam due to entrained particulate matter is also very low. Laboratory experiments show that steam issuing from a vessel under the conditions of this reactor will show a separation of 10^5 for particulate radioactive materials. In a qualitative way measurements of the radioactivity of the steam from the reactor after the short-lived nitrogen activity has decayed verify these laboratory observations. It will take long continued operation to establish the magnitude of the radioactive maintenance problem encountered in the turbine where steam directly from the reactor is employed.

COST ANALYSIS

The test facility has been constructed, having in mind that its useful life will be only a few years. Actually, however, the reactor and its auxiliary equipment and the power conversion equipment do not differ in any material way from standard construction. The savings in cost are mostly in the buildings which house the equipment. These buildings, while utilitarian, are fully equipped and would be entirely suitable for a power plant constructed in a remote area or in connection with mining or manufacturing operations. For these reasons, there is an incentive to analyze the cost for the plant and to come to a conclusion as to what would be the cost of power from a power plant constructed along these lines and operated with enriched uranium as fuel.

The total cost of the facility is $550,000. More than half of this is in the power conversion equipment and the cooling facilities. With an 80% load factor and an efficiency of 20%, the total electrical energy produced per year is 1.4×10^7 kwh. If the capital charge is written off at a rate of 15% per annum, the contribution of the capital investment to the cost of the power is 5.9 mills/kwh.

Fully loaded, the reactor requires 11.8 kg of U^{235}. This is sufficient to provide reactivity for the loss due to the temperature coefficient and the operating xenon with a margin of about 2 kg for burnup. Assuming that all of the fuel must be changed when 2 kg have been consumed, 1.8 fuel charges will be required per year and 3.6 kg of U^{235} will be consumed per year.

In order to allow for radioactive decay of discharged fuel, about 1.5 reactor charges will be committed to the plant at all times. The following costs are computed for prices of U^{235} ranging from $15 per gm to $30 per gm.

At $15 per gm and 4% the inventory cost is $10,600 per year. At $30 per gm and 4% the inventory cost is $21,200 per year. The corresponding contributions to the power cost are 0.76 mill/kwh and 1.52 mills/kwh. The 3.6 kg of U^{235} which are burned in a year represent a charge to the power cost of 3.85 mills/kwh at $15 per gm and a charge of 7.70 mills/kwh at $30 per gm. Each fuel element costs $425 to fabricate. Fabrication of the 1.8 fuel charges per year adds 4.7 mills/kwh to the power cost.

In order to recover the unburned U^{235} and make it available for reuse, chemical processing is required. Various estimates indicate that this would be entirely practical at the present time at $5 per gram and that, if any considerable volume were available, this figure could be reduced. At $5 per gram, the processing charge adds 6.2 mills/kwh to the power cost.

Experience to date in operating this plant indicates that a staff of about twelve is required. Operating costs consequently add another 8.6 mills/kwh to the power cost.

The following table summarizes these various costs:

	U^{235} price	
	$15/gm	*$30/gm*
Capital charge	5.9	5.9
Inventory	0.76	1.52
Burnup	3.85	7.70
Fabrication	4.70	4.70
Processing	6.20	6.20
Operation	8.60	8.60
	30 mills/kwh	34.6 mills/kwh

Almost half of this cost is due to the capital investment and the operation cost. It is doubtful that for a small size plant these can be improved very much. Of the fuel costs, the major part is for fabrication and processing. For both of these considerable improvement is possible.

The experience acquired with this facility strongly suggests that where a new type of reactor is to be investigated a test facility such as here described is of great usefulness. It is believed that results have been obtained which would have been difficult or impossible to achieve by either calculation or experiments with other reactors. More important, however, is the conviction that the results have been obtained in a shorter time and with the use of fewer reactor technologists than otherwise would have been possible.

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2. Dietrich, J. R., Zinn, W. H., *et al.,* P/481 *Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors,* Session 6.2, Vol. 13, these Proceedings.
3. Lottes, P. A., *Boiling Studies at Argonne Relative to Boiling Reactors.* Proceedings, 1955 Conference on Nuclear Engineering, University of California, Los Angeles; California Book Co. Ltd., Berkeley, California.

Record of Proceedings of Session 3.1

TUESDAY MORNING, 9 AUGUST 1955

Chairman: Sir John Cockcroft

Scientific Secretaries: Messrs. F. de Hoffmann and N. Dobrotine

PROGRAMME

P/615 The first atomic power station in the USSR and the prospects of atomic power production ..... D. I. Blokhintsev and N. A. Nikolaev

DISCUSSION

P/851 Design and operating experience of a prototype boiling water power reactor ..... J. R. Dietrich *et al.*

DISCUSSION

Mr. D. I. BLOKHINTSEV (USSR) presented paper P/615.

DISCUSSION OF P/615

Mr. LEWIS (Canada): In one respect the reactor described [in P/615] is similar to the NRX reactor in that it has very many monitoring devices. Our experience with the NRX reactor has shown that about half the shut-downs that occur are attributable to the statistical fault rate of these many monitoring instruments. I would like to hear what has been the experience in the USSR with these many monitoring instruments.

Mr. BLOKHINTSEV (USSR): Irregularities in the operation of the electronic equipment occurred only during the early days of operation (during the first and second months). Since then, over an extended period of time, we have had no instances of reactor shut-downs due to defects or errors in the electronic equipment. We have taken a number of steps. Several instruments which particularly interfered with the working of the reactor have been replaced or placed under the control of others. This has made the emergency precautions system function more regularly.

Mr. LIBBY (USA): This very interesting and competent paper [P/615] has raised a number of questions, one of which I would like to put at this point. Most of the United States' program to obtain economic atomic power is based upon high conversion ratios. We note with interest that the USSR atomic power plant has a low conversion ratio. In essence the plant burns costly separated uranium-235. Do you plan in your large plants to have conversion ratios nearer unity? If so, how do you plan to alter your fuel elements to obtain these?

Mr. BLOKHINTSEV (USSR): Your observation is perfectly correct. This reactor does in fact have a low conversion factor. We intend in the larger model to make a number of changes in the structural design of the reactor and in the materials. We want to use alloys we did not employ originally. By so doing we shall reduce neutron absorption and increase plutonium breeding from uranium-238. We think we can bring the uranium-235 enrichment factor in a design of this kind down to, say, 1.5 per cent. This will increase the conversion factor from 0.3 to 0.5. In systems of this kind, however, it is hardly possible to improve the conversion factor much further, and we of course consider systems that yield a higher conversion factor to be unquestionably more useful. It is important to establish that an alternative type working with a conversion factor of 0.5 can also be suitable. We in the USSR are also considering systems which yield an even higher conversion factor.

Mr. WENT (Netherlands): I should like to put one technical question about your most interesting power reactor. You are using a special uranium alloy. Might it be that you are using molybdenum for metallurgical reasons?

Mr. BLOKHINTSEV (USSR): An alternative material of that kind undoubtedly could be used.

The CHAIRMAN: Mr. Blokhintsev will also reply to questions submitted in writing by Mr. Dunworth.

Mr. BLOKHINTSEV (USSR): Yes, I have Mr. Dunworth's questions in written form. I shall read them out:

1. Do you in Russia intend to build a large number of 5000 kw reactors of the type you have described here, or are you considering building reactors of some other type?

My reply to this question is that we do not propose to build any 5000 kw stations. We are, however, planning to build stations of this type with higher capacities, incorporating a number of technical improvements.

2. It was recently stated by Mr. Malenkov that Russia is building a 100 mw reactor. Is this correct? Are you planning to build other types still, and how do you consider that the Russian reactor-building programme compares with ours?

My reply to this question is that Mr. Malenkov's statement was correct. A reactor of that kind is being

built and will probably be put into operation in a year's time. I did not include any data on this reactor in my paper.

With regard to comparing power reactor programmes, my view is that our conference will bring about a more complete exchange of information about our work in future, and this is evidently the only way in which we can compare the results of our work and gain useful and valuable experience from each other.

Mr. SHANKAR (India): Is there much decomposition of water in the primary circuit due to intense radiation? If so, how is it taken care of?

Mr. BLOKHINTSEV (USSR): At low pressures and temperatures, large quantities of electrolytic gas are in fact evolved in the powerful neutron flux, but experiments carried out in the Soviet Union show that electrolytic gas should not be formed at a pressure of 100 atmospheres and a temperature of 300°C. A paper on this subject by Messrs. Dolin and Ershler is to be presented at this Conference.

Mr. WEINBERG (USA): I have two questions to put to Mr. Blokhintsev.

The first is this: If I properly understand the description of the fuel element, that element is cooled only internally. That would imply a tendency for the fuel element to be unstable, since the fuel would heat up to a greater extent than the jacket. How do you cope with this instability in the fuel element?

Mr. BLOKHINTSEV (USSR): You're quite right, Mr. Weinberg, you have understood me correctly; the cooling takes place internally only. Outside there is only a thin jacket. The materials, details of which were not given in the paper, are flexible, so that in this case it would be wrong to proceed from the idea of coefficients of expansion: since the materials are flexible, no rupture occurs between the jackets and the fuel.

Mr. WEINBERG (USA): Does that mean that the jacket is not bonded to the fuel?

Mr. BLOKHINTSEV (USSR): No, there is no bonding.

Mr. WEINBERG (USA): The second question has to do with the very interesting and significant comments that were made on the fast breeding cycles. The conversion ratios, if I understood Professor Blokhintsev's remarks correctly, that were found in the Soviet Union were of the order of 2.6. Now, this I take it was with a system in which the U^{238} to U^{235} ratio was low, and therefore the neutron spectrum was high. Would you venture to guess what neutron spectrum and what conversion ratio you might expect in the actually operating fast neutron chain reactor?

Mr. BLOKHINTSEV (USSR): Evidently there is some misunderstanding with the interpretation here. We have never observed a conversion factor greater than 2. I introduced a figure relating to the number of neutrons per fission, what is called "$\bar{\nu}_{eff}$". This figure was 2.72 for the hardest spectrum and 2.9 for neutrons of 900 kev energy, while for neutrons of 250 key energy, the figure for $\bar{\nu}_{eff}$ was 2.6.

The CHAIRMAN: Well, I think that brings us to the end of this very interesting discussion which I am sure could have gone on for a long time, and I hope it will be continued in smaller private discussions.

Mr. W. H. ZINN (USA) presented paper P/851 as follows: Mr. Chairman and delegates to the Conference: it is my privilege to describe to you the first atomic power plant in which steam formed directly in the reactor core is used to propel a turbine. It is my purpose to relate to you our operating experience and to state some of the advantages and the difficulties of operating a heterogeneous nuclear reactor as a water boiler.

The mechanical and other complexities of most power reactors readily can be appreciated by an inspection of the many fine models on display in this building. You will note that a single reactor can have primary and secondary cooling systems, gas-handling systems, emergency power systems and a whole complex of control apparatus. Such complexity is costly. In the United States, the fact that mechanical energy and electricity can be generated successfully with an atomic reactor is accepted. The goal is to generate electricity at a cost entirely competitive to the cost of electricity from coal. Natural resources and technical advances have combined—I could even say conspired—in the United States to make the cost of electric power from coal remarkably small. It is the task of our reactor technology to equal this low cost and to improve on it. The boiling water reactor dispenses with much of the complexity in design, and as a result is expected to be advantageous with respect to both the capital investment and the cost of operation. There are numerous reasons why the boiling reactor improves the situation with respect to competitive power.

The reactor pressure vessel and the temperature of the fuel elements are two points of major importance. It is assumed that in order to produce economically useful power the reactor must operate with a high conversion ratio, in other words, it is not acceptable to burn only uranium-235. This means that, insofar as possible, neutron-absorbing material must be eliminated from the core of the reactor. In a water reactor, moderated and cooled with water, this most readily is done by pressurizing the whole core, including the fuel elements and the moderator. This calls for the construction of a pressure vessel of large dimensions. This is a difficult technical problem and may be very costly. In a pressurized water reactor—that is, one cooled with solid water—the vessel may be required to operate at a pressure of 100 atmospheres. Because of the loss of temperature in raising steam with this pressurized water, it is

feasible to produce steam for the turbine at a pressure no greater than 20 atmospheres.

The first point on which the boiling reactor is distinctly superior is that, in producing 20 atmospheres pressure steam for the turbine, a reactor vessel of only 20 atmospheres operating pressure is used. The over-pressure which water reactors have required to prevent boiling is not needed. This reduction of pressure in the reactor vessel permits sizeable cost reduction and, indeed, makes feasible the construction of pressure vessels for reactors of very large heat rating.

The second point on which the boiling reactor presents a subtle but very important advantage concerns the temperature of the fuel element. Water reactors of high conversion ratio probably will have as the base fuel a metal—either uranium or thorium. These materials are very corrosion-sensitive in hot water and must be protected with materials of low neutron absorption. Aluminum and zirconium most likely are the metals for the protective jackets. These metals also are subject to corrosion, which is temperature sensitive. There is then a real advantage in keeping the temperature of the fuel element low. In the boiling reactor where the steam from the reactor is used directly in the turbine, the temperature and pressure losses that an intermediate steam boiler demands are eliminated. In effect, this lowers the operating temperature of the fuel element.

These two reasons alone are sufficient justification for pursuing the development of direct cycle boiling reactors. In addition, there are simplifications in the circulating system and in the controls. For two years, experiments in the boiling reactor have been under way. A great many of these experiments have dealt with the inherent safety characteristics of such a reactor. It is not my purpose today to give details of such experiments. You are referred to a paper which is to be given at session 6.2 tomorrow by my colleague, Dr. J. R. Dietrich. As had been predicted by S. Untermyer, the boiling reactor has a relatively favourable performance with respect to safety.

A number of reactor cores have been examined. The one which is referred to in my paper as reactor III has been operated at a thermal power of 15,000 kilowatts, and has been used with the turbo-generator unit to produce electric power of 2300 kilowatts. Many of the problems raised by the use of reactor steam in a turbine can only be answered by actual experience, and some perhaps only by prolonged experience. The turbo-generator has been in operation for only a short time but it has received more than 3 million kilowatt-hours of energy as steam, and experience in operating the whole plant is being acquired steadily.

As mentioned earlier, the fuel element material for a power reactor should be mostly uranium or thorium. Such fuel elements clad with zirconium have been developed and are on display in the United States exhibit in this building. For the test plant described here it has been convenient to use a fuel element the construction of which is both easy and cheap. This is so because experiments in the properties of a boiling reactor require changes in the physical dimensions of the fuel element, and ease of construction is of primary consideration.

This reactor is operated at a pressure of 300 pounds per square inch. Its fuel elements operate at 215°C. This permits the use of aluminum as a construction material. The basic heat-producing element is a plate of aluminum uranium alloy clad with aluminum. The uranium is enriched to about 90 per cent.

Slide 1 is a photograph of a partially assembled fuel element (Fig. 2 of P/851). You see that it is made up of 24 plates which are spot-welded into a square structure having a dimension along this section of 10 centimeters. The space between the fuel plates is half a centimeter and is the space which is filled with water and steam. The length of the part of the fuel element which contains uranium is 65 centimeters. In actual use the fuel element has extensions fastened to it on both ends.

In operation, each fuel element must produce on an average 170 kilowatts of heat, and due to the peaking of the neutron flux those at the center of the reactor must produce in the neighborhood of 250 kilowatts of heat.

I will now show a series of slides illustrating the construction of the reactor.

Slide 2 is a cross section of the reactor vessel (Fig. 4 of P/851). This is the reactor vessel with its insulation on the outside. The diameter of the vessel is 1.3 meters, and it contains 88 of the fuel elements just illustrated. There are five control rods. Four of these are blade type rods being half an inch thick, made of compressed boron, and about one foot wide. The fifth one is in the form of a cross and contains cadmium. Water and steam formed in the channel shown here between the active plates pass upwards, and water passes downwards in this space which surrounds the core.

Slide 3 is a photograph of the core assembly (Fig. 5 of P/851). Here is one fuel element supported bottom and top. It is firmly held by this stainless steel plate here and a similar stainless steel plate below. The channels shown here with the holes cut into them provide spaces for the control blades to slide up and down.

Slide 4 (Fig. 3 of P/851) shows the reactor core installed in the pressure vessel. The core assembly which we have just seen in Slide 3 is suspended from the top of the reactor vessel on stainless steel bars. It is arranged in this way so that it is possible to remove the whole structure and replace it by one of a different design. The active section is shown here in red. Water enters the bottom of one of these fuel elements and is first heated, thus forming steam. The steam and water rise in this chimney which stands above the active part of the fuel element. This chimney of reduced density promotes the natural

circulation of water down the outside of the core and the mixture of steam and water through the core. The control rods are actuated by shafts which go through the flat lid of the reactor vessel and through a shield block which is above the reactor vessel. The large space above the water level in the reactor vessel acts as a steam chest or steam dome. Steam rises, and water droplets fall out of the steam, so that the steam emitted from this exit pipe is relatively free of entrained water.

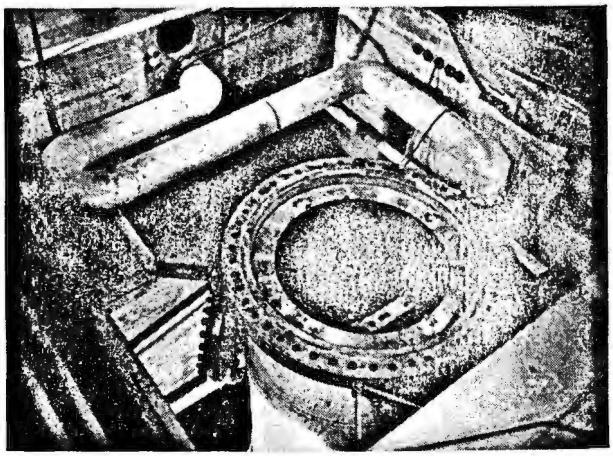
Slide 5 (Fig. 7 of P/851) shows the flow diagram of the whole plant. It is clear in looking at this flow diagram that the boiling reactor has produced major simplifications. In red we have the steam line which goes to the turbine. First we have a safety valve or relief valve; then a back pressure regulating valve, a valve which permits the venting of the steam to the atmosphere if it is desired, then a steam separator for further drying the steam; then the conventional trip throttle valve of the turbine. This valve leading to the condenser is a back pressure regulator which takes that steam which is produced by the reactor and which is not demanded at the moment by the turbine. The third valve leading directly to the condenser is interlocked with a trip throttle valve of the turbine so that in the event of the loss of load on the generator causing this trip throttle valve to close quickly this one will just as quickly open and let the steam proceed to the condenser. In this way the reactor does not know that the path of the steam has been altered.

The condensate is sent by a pump back to the reactor via a feed-water tank and an injection pump, or a feed-water pump, which puts the water back into the reactor. On the way it is used to cool the water leaving the reactor, which is destined to pass through the filter and ion exchange column required to clean the water, which is then returned to the feed-water tank. The third water circuit is an auxiliary one, which consists merely of a pump and some electrical heaters which circulates water in the reactor vessel and permits bringing the reactor to some temperature by electrical means rather than using the nuclear means. It is a circuit which could easily be dispensed with.

Slide 6 (Fig. 8 of P/851) is a diagram of the whole reactor arrangement. First, we note that the main components are below ground. The reactor vessel, which is some 16 feet tall, is installed in a pit shielded on all sides by concrete and earth. The control rod drive mechanism is housed in this case above. This shield, when the reactor is to be unloaded, can be rolled to one side, after first disconnecting the control rod drives and flooding this whole pit with water for shielding purposes. At that moment the reactor vessel top can be removed and the fuel elements, one by one, can be drawn into a coffin and carried by the crane to a storage basin at this point. In this second pit there is all of the machinery which was shown on the flow diagram. Here is the feed-water tank; here is the feed-water pump; here is the circulating pump and, above, the electrical heaters.

The steam line then proceeds to the turbine.

Slide 7 is a photograph of the reactor vessel in its pit, and you can see here the support ring on which the whole reactor core is supported.



Slide 7. Photograph of top of reactor vessel

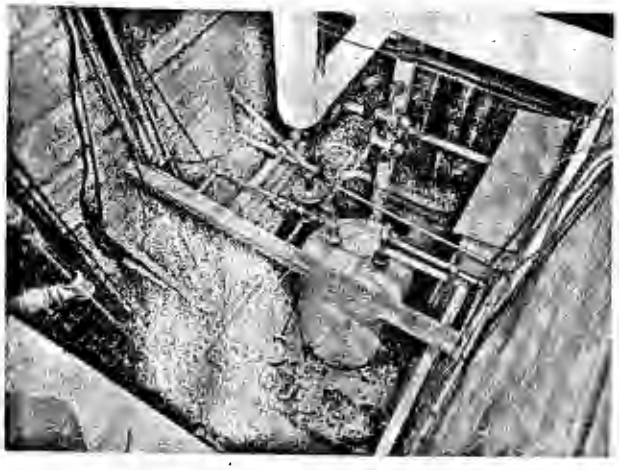
Slide 8 is a photograph of the machinery pit, and you can see directly below, in the foreground, the feed-water pump having a motor of 50 horsepower capacity and the flow capacity maximum of 90 gallons per minute.

It is one of the features of a boiling reactor that the auxiliary machinery is small and does not require a great deal of electric power. The tank is the feed-water tank. The vertical column in parallel with the reactor vessel carries all of the devices for measuring the level of the water in the reactor vessel. This is the one quantity which it is most important to know accurately at all times, and numerous devices for measuring the level and recording it, and using it to control pumps are shown on this column.

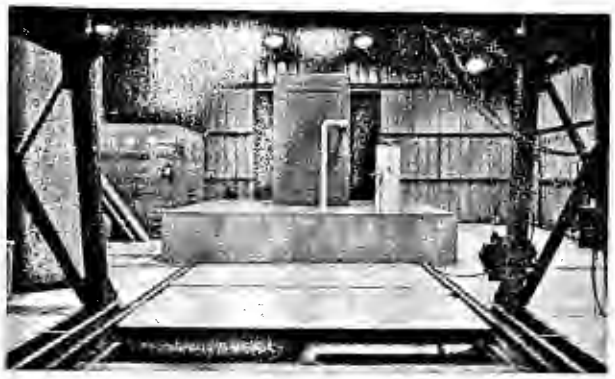
Slide 9 is a general view of the reactor. The control rod housing contains some auxiliary shielding which is required above the moveable shield. The pipe in front of the housing delivers reactor steam to a part of the control mechanism. I will refer to this in a moment. In the foreground we have the cover over the machinery pit.

Slide 10 is a view of the control rod equipment inside the housing I just mentioned. You can see that it is quite crowded because one must have the equipment to move five control rods packed in a very small space. In order to overcome the internal pressure of the reactor vessel which would tend to lift the control rods out of the reactor the upper ends of the control drive rods terminate in steam pistons which are fed with steam from the reactor vessel, thus balancing out the force of the steam.

Slide 11 gives us a close-up of the operating panel, showing all of the instruments required to control the neutron reaction of the reactor. The five center



Slide 8. Photograph of machinery pit



Slide 9. Photograph of inside of reactor building

Slide 10. Photograph of control rod drives

dials are the indicators for the five control rods. This recorder records the neutron intensity, as does this one. At the right we see a television camera showing the pressure on a gauge fastened to that stand-pipe in the machinery pit, and we see that it reads the pressure of the reactor at three hundred pounds per square inch. The control knobs are all that are required to operate the reactor.

Slide 12 is a general view of the control area showing all of the instruments required to operate the plant. The control panel I just referred to is in the background. On this control panel at the left we have steam flow, steam temperature, water temperature, radioactivity of the steam and temperatures in all parts of the reactor system.

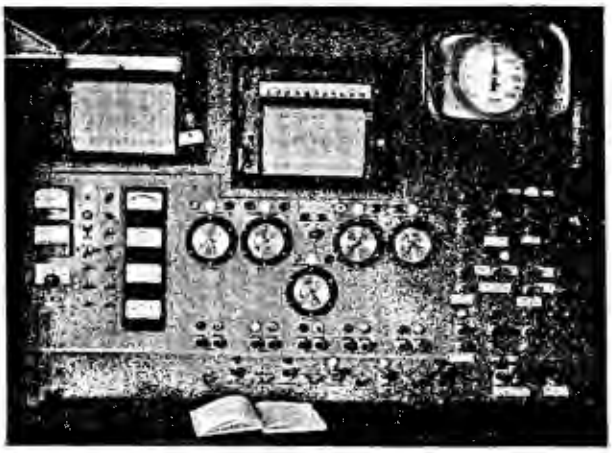
Slide 13 is a general view of the turbo-generator which has a rating of somewhat over three thousand kilowatts. Presently it is generating something around two thousand kilowatts of electricity. We can see the turbine here, the generator and the condenser below.

Slide 14 is a general view of the buildings, with the reactors housed in one and the turbo-generator in the other. They are connected by an overhead steam line and in the foreground there is the cooling tower. These buildings, of course, are of utilitarian construction since it is not expected that this plant will be operated for a long time.

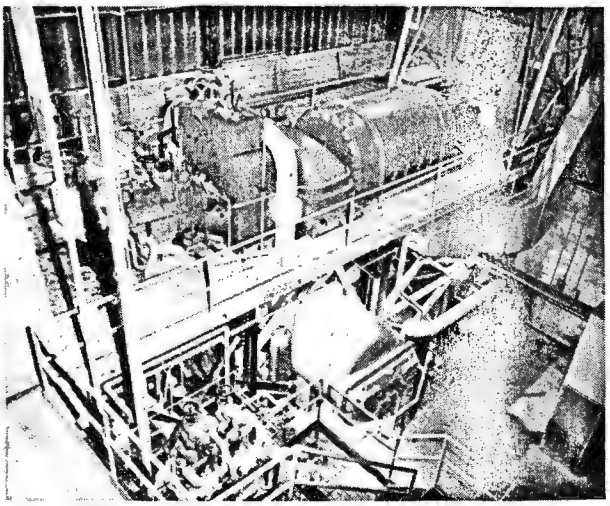
Now that we have seen something of the construction and, I hope, something of the simplicity of this power reactor, I wish to state something about the operation. It has been possible to operate the reactor under conditions which are stable for all pressures from atmospheric to 300 psi. A necessary requirement for reasonably smooth operation is that the amount of reactivity corresponding to the steam void not be excessive. In all operations, however, the power level is subject to small rapid variations. These clearly are connected with the turbulence created by the formation of steam. The degree of roughness of the power is directly related to the excess reactivity which has been subtracted by the formation of the steam voids and which has been replaced by withdrawal of the control rods.

A series of experiments has been performed to determine the relationship between the excess reactivity in voids and the power density, and this has been done for a number of pressures. It has been observed that for any given pressure the degree of roughness of the power varies with the excess reactivity in steam voids. If the excess reactivity is raised to too high a figure, oscillations in the power become apparent, and further increases in excess reactivity in steam may bring small excursions in the reactor power.

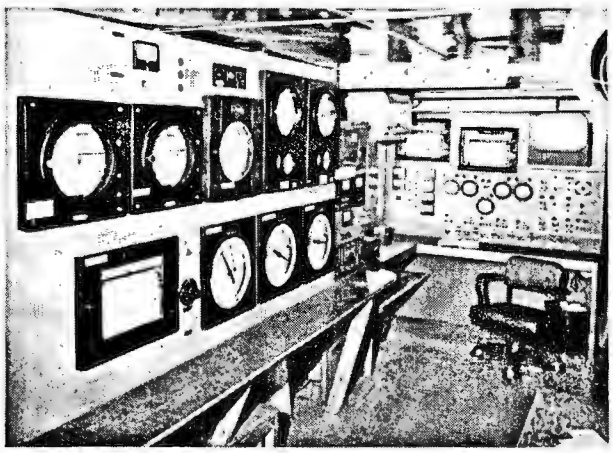
Slide 15 (Fig. 10 of P/851) contains curves of power density versus excess reactivity in steam for two different reactors labelled II and III. Reactor III was that illustrated earlier, whereas reactor II, although it used the same vessel, was considerably smaller and had a higher (negative) steam coefficient



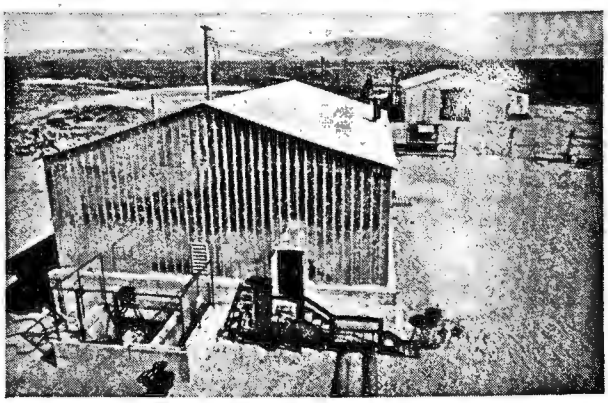
Slide 11. Photograph of control panel



Slide 13. The turbine-generator installation



Slide 12. Photograph of interior of control room



Slide 14. View of reactor and turbine buildings

of reactivity. I might say that the steam coefficient of reactivity is defined as the fractional volume of the core occupied by steam bubbles, divided by the fractional change in reactivity caused by that void. Consequently, a given reactivity in reactor III corresponds to a higher steam content per unit volume than does the same reactivity in reactor II, and the power density of reactor III is correspondingly higher.

In other words, the steam void coefficient is smaller in III than in II. That is why we get a higher power rating. Vertically, there is plotted the power density in kilowatts per litre of the core and horizontally there is plotted the excess reactivity in steam in per cent.

The curve for reactor II includes points at several operating pressures. There is no great variation with pressure in the power density attainable from a given addition of reactivity, at least within the pressure range investigated. The points we refer to are these here showing that you do not gain power density by increasing the pressure. This corresponds to certain laboratory experiments which have been done.

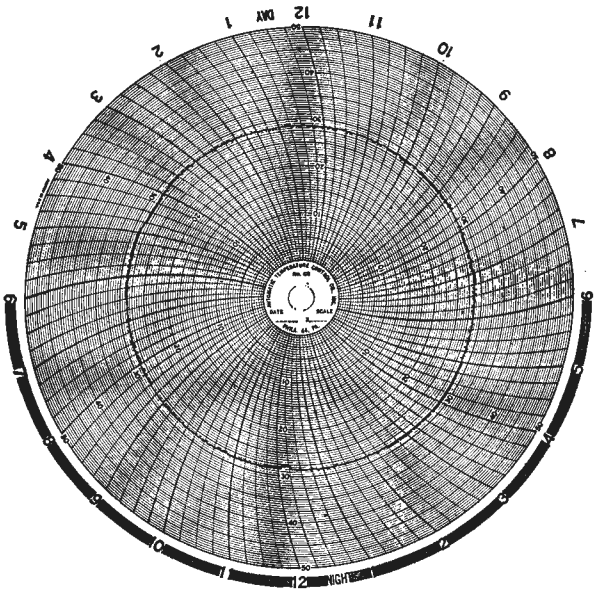
Slide 16 (Fig. 11 of P/851) reproduces the power traces of reactors II and III obtained with an ionization chamber and a rapid recorder. In reactor II, which was the small reactor, with 2.6 per cent excess reactivity in steam voids the trace has *this* form. Again, plotted vertically is the power on a relative scale and horizontally the time. Here the reactor was operating at 5200 kilowatts but, because of the greater steam coefficient, one has a considerable degree of roughness which may amount to as much as 15 per cent. In reactor III, where we have for 12,000 kilowatts of operation 1½ per cent in steam voids, the roughness is very considerably less, and we see that the variations amount to at the most 4 per cent in power. This small roughness in the power is something to which one rapidly becomes accustomed, but of course is a new thing when one has been looking at the power tracers of more conventional reactors.

Slide 17 is a photograph of the steam pressure, in this case at the turbine. The pressure at the turbine is 290 and this a record for 24 hours. You will see that there are very few major fluctuations in the steam pressure.

Slide 18 is a photograph of the trace of steam flow, in this case at 39,000 pounds per hour, again for 24 hours, and you see that there is very little difficulty in getting a uniform and even steam flow. I could

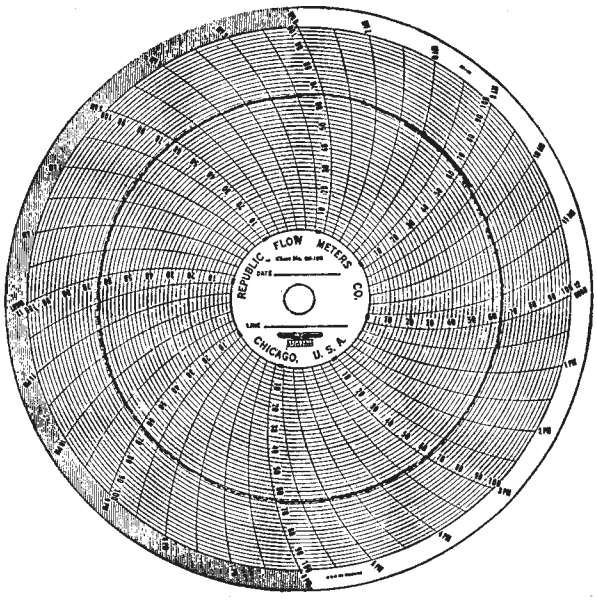
comment that, in getting one of these curves, the control rods are hardly moved at all after the xenon has come to equilibrium.

Slide 19 (Table I of P/851) summarizes the results of power density studies which have been obtained for three reactor cores which differed in size of plate spacing and the U^{235} loading of the plates. We see in the first item the spacing of the plates, that is to say, the thickness of the water channel plus the fuel element. The aluminum-clad fuel element plate is 1.5 mm thick. In the second line we see that the ratio of volume of metal to the volume of water has been successively decreased from reactors I, II to III. The third item shows that the number of plates in a fuel box was varied, but this is unimportant since the design of the fuel box was also changed. The effective neutron lifetime has not altered very much in changing ratios by the amount indicated. Item 5 shows the amount of U^{235} in each fuel box and what can be noted is that for reactors II and III we have two weights of uranium available. This is done so that we can use some of the heavier loaded plates to adjust the void coefficient, the steam coefficient. By placing some of the heavier loaded plates nearer to the periphery of the reactor, leakage can be increased, and of course it is the leakage which determines the void coefficient. As in all water reactors the temperature coefficient is sizeable, and it becomes smaller as the amount of metal is decreased from reactor to reactor. This is the over-all temperature coefficient from room temperature to operating temperature. It must not be assumed that this coefficient is uniform over this whole range. Item 7 shows the void coefficient, which has to be calculated since we have no way of actually measuring the fraction of the volume which is steam, and this shows that the void coefficient becomes successively smaller as the reactor is changed. In the eighth item is shown the power density per per cent reactivity in voids. This, of course, is the critical number for the reactor and is a number which one would like to have as large as possible. It was increased successively as is shown here. But, however, it should be noted that it was done at the expense of a smaller and smaller void coefficient. Here the reactor designer must choose between how much inherent safety he wishes in the reactor and how great a power density he wishes to have. It is clear that, as this coefficient becomes smaller the inherent safety situation is deteriorated a bit, and this is done by getting a higher power density. The void coefficient of reactivity apparently is the principal factor affecting the performance of the reactor core so far studied. It is clear that without a careful study of the excursion characteristics of the reactor it would not be possible to make a cogent decision on the magnitude of void coefficient in the design of the power reactor. The average heat flux for a fuel plate is between 3 and 4 calories per cm^2 per second, while the maximum is 7 calories per cm^2 per second. It is clear that in boiling reactors which employ only natural circulation of the water the heat rates attained are not high: the limiting factor on power output is the rate of flow of steam out of the reactor core rather than the heat flux from the fuel plates. As was indicated earlier, reactivity can be added safely to this boiling reactor in amounts which are dependent on the pressure of operation. Experiments to date indicate that at 300 psi with a water channel width of 0.5 cm steady power operation is obtained even with 3.2 per cent reactivity subtracted by the steam voids. Experiments at lower pressures clearly indicate that there is an amount of excess reactivity above which



STEAM PRESSURE ~ 290 PSI

Slide 17. Steam pressure record



STEAM FLOW ~ 39,000 LBS/HR

Slide 18. Steam flow record

instability sets in. This is not a matter of great concern because as more and more reactivity is added the instability manifests itself by oscillations in the power level. These oscillations have a period of about four-tenths of a second and as successive amounts of reactivity are taken up by the steam voids, the amplitude grows. There is a considerable region of added excess reactivity where these oscillations are stable and can be made to increase in amplitude only by a further increase in reactivity. Experiments have been performed in which reactivity has been added at a fixed rate by the withdrawal of the control rods.

Slide 20 (Fig. 12 of P/851) shows the amount of excess reactivity added by the rods as a function of time. In the vertical direction is plotted the reactivity added—1, 2 and 3 per cent—and in the horizontal direction the time in seconds. We see that the rod was steadily withdrawn from *here* to *here;* in roughly four seconds an amount of reactivity of 2.6 per cent was added. This gave the results shown in the next slide.

In Slide 21 (Fig. 13 of P/851), the rod withdrawal was started when the reactor was operating at 485 kilowatts. Seconds are shown horizontally, power vertically. The power rose with a period of roughly 6½ seconds and then stabilized at 5.4 megawatts. Let me repeat, this is an experiment in which almost 3 per cent excess activity was added in some 40 seconds. The reactor power rose but stabilized despite this rather brutal treatment.

In Slide 22 (Fig. 14 of P/851) is shown an experiment of a similar nature. This time the same amount of excess reactivity was added, almost 3 per cent, in a time of 160 seconds, but the difference is shown in Slide 23 (Fig. 15 or P/851) in that now the reactor began at only 1.2 kilowatts; in other words, the water was not boiling. The reactor power increased with a period of roughly 3 seconds and stabilized at 4.3 megawatts, as shown. We have found it is practically impossible by the manipulation of the controls to get into any difficulty. One can manipulate the controls blindly, if one wishes.

For a boiling reactor producing steam at an appreciable rate there has been considerable conjecture as to the consequences of a sudden closure of the steam exit line. A boiling reactor connected to a turbine might experience this if the trip throttle valve of the turbine should close and if there were not the by-pass valve which I mentioned earlier.

As shown by Slide 24 (Fig. 17 of P/851), this experiment has been tried without the by-pass valve. On the power scale we have the point at which the valve was suddenly closed when the reactor was operating at roughly 1900 kilowatts. The power rose and stabilized, whereas the pressure, which began at roughly 130 psi—naturally, since it was a closed vessel in which boiling was taking place—rose along a fairly straight line. Why did the reactor power stabilize as indicated? Because the pressure increased the temperature also had to increase, and there is a negative coefficient of reactivity, and this tended to stabilize the power. This action could not be kept on indefinitely. As a matter of fact, that experiment has also been tried. The reactor is equipped with pressure relief valves as safety valves. The pressure relief valves open up and hold the pressure at whatever pressure the valves are set at.

On several occasions all the electricity supplied to the plant has been cut off over a period of hours and quite suddenly at a time when the reactor was at full power and when no emergency source of power was supplied. The shut down heating was easily accommodated by boiling, and no danger resulted. This suggests that the boiling reactor requires little or no source of emergency power or apparatus.

The radioactivity of the fluids supplied to the turbine is of paramount interest. The radioactivity of the steam is overwhelmingly due to N^{16} which is formed by an (n,p) reaction on O^{16}. The half-life is 7½ seconds. The following values for the radioactivity of the steam are characteristic: on the surface of a 6 inch line carrying 290 psi inch steam, 100 milliroentgens per hour; on the surface of the turbine casing, 20 milliroentgens per hour; on the surface of the condenser hot well, when the hot well is filled with water, 150 milliroentgens per hour. These activities on shut-down of the plant die out to negligible values except for the hot well where there is some longer-lived activity in the water.

The reactor water has an activity of 390 microcuries per liter. More than 0.8 of this radioactivity is due to Na^{24}. Because of the relatively large activity due to sodium, the plant was examined to determine the source of sodium. Two sources were identified. The demineralizer which supplies make-up water to the plant produces water of about one part per million sodium. This is much too high for such a water supply, and is a point which needs to be and can easily be corrected. Also there is evidence that there is some leakage in the condenser tubes. The cooling water for the condenser water is raw well water having a very high sodium content. Rather small leaks in the condenser would supply enough raw water to account for the sodium activity of the reactor water.

The gases removed from the condenser by the steam ejectors are radioactive. The principal activity of course is short-lived N^{16}. There is, however, in this activity a component which has been identified as A^{41}. This is not surprising since the make-up water supplied to the reactor is not de-aerated and it is the argon in this air which is the source of the radioactive A^{41}. The gas withdrawn from the condenser by the steam injector contains 1.4 microcuries per liter of A^{41}. The total A^{41} vented to the atmosphere per day is only 240 millicuries.

This plant has been constructed having in mind that its useful life will be only a few years. Actually, however, the reactor and its auxiliary equipment and the power conversion do not differ in any serious

way from standard construction. The savings in cost are mostly in the buildings which house the equipment. These buildings, while utilitarian as I mentioned, are fully equipped and would be entirely suitable for a power plant constructed in a remote area or in connection with mining or manufacturing operations.

For these reasons there is an incentive to analyze the cost for the plant and to come to a conclusion as to what would be the cost of power from a power plant constructed along these lines and operated with enriched uranium as fuel.

Slide 25 gives a summary of the costs (see P/851). The plant, as so described, cost $550,000 to construct. At a rate of 15 per cent per annum the capital charge against the power is 5.9 mills. The enriched uranium has a cost of somewhere between $15 and $30 per gram. Using the two extremes one can find the total of power cost. Inventory at 4 per cent amounts to 0.76 mills *here* and double that value in the second case. Burn-up is 3.85 mills per kilowatt-hour, and 7.7 mills per kilowatt-hour. Fabrication can be derived because the fabrication costs for these fuel elements was $425 per fuel element. This adds 4.7 mills. Processing costs are estimated to be $5 per gram, and this amounts to 6.2 mills. It takes twelve people to operate the plant, and this is estimated to be 8.6, giving the totals which are shown. Most of this cost is in the fuel element.

In conclusion, I would like to say that these power costs are less than a factor of 2 greater than the cost of power from a coal station of comparable size. In estimating power costs for atomic energy there is a great temptation to make the calculation for reactors of very large heat rating because this produces the most favorable result. For many countries, however, large capacity power plants are impractical at the present time. We are greatly encouraged to find that a small plant has actually been constructed for which the power cost begins to approach a useful value.

DISCUSSION OF P/851

The CHAIRMAN: I will now ask Mr. Kruzhilin, of the USSR, to ask his question.

Mr. KRUZHILIN (USSR): Professor Zinn's report on the heterogeneous boiling-water reactor is of great technical interest. On the basis of industrial-scale experiments, the report gives a clear account of the most difficult problem of the reactor's stability. Direct experiments have shown that the use of a boiling-water reactor to generate power is feasible. As a result, the alluring prospect of using a reactor in which steam is raised in the reactor vessel itself will become a reality.

In connection with this report, I will take the liberty of dwelling on one purely engineering question, which in my opinion is of real importance in the case in point. I refer to the question of keeping the steam free from droplets of water. One can fully agree with Professor Zinn and the other authors that in this case very pure steam can be obtained. What I have in mind is experience with the operation of steam boilers. We can reckon that for this purpose it is essential that the amount of steam raised through the "evaporation mirror" must not exceed certain limits. According to the information available to the Academy of Sciences of the USSR, with pressures of 30 to 40 atmospheres about 20 tons of steam of the purity indicated can be raised per hour per square meter of evaporating mirror, since in the case in point we cannot have in mind a reactor vessel more than 2 meters in diameter.

According to the information available to the Academy of Sciences of the USSR, not more than 60 tons of steam per hour can be obtained direct from the reactor. With a greater load, the amount of moisture entrained in the steam will become appreciable.

If the load on the evaporating mirror is increased still further, the natural circulation of the water in the reactor will become unstable. The more correct conclusion would be that a reactor of the boiling-water type with natural water-circulation could not be very powerful. This conclusion can be applied to the design of Professor Zinn's reactor. The desired capacity of the reactor is 250 to 500 tons of steam an hour. Even with the extremely valuable and important data contained in Professor Zinn's report, the design of a powerful reactor of the boiling-water type will still present theoretical difficulties. I would ask Professor Zinn to ponder the question: what capacity is at present realistic for a reactor of this type? I would like to say that the other engineering problems, namely the transfer of heat and the circulation of water in a reactor of the boiling-water type, are more or less clear, and that from this point of view the design of a reactor of this type is interesting.

In the Soviet Union fairly extensive experiments have been carried out on heat transfer, and have shown that for a reactor of this type a flow of heat amounting to 1,500,000 kilocalories per square meter per hour may be contemplated. That is a considerable load. But, as we have learned from Professor Zinn's report, the load is not the only crucial factor in the reactor. In the Soviet Union there is a fairly large amount of experimental and theoretical data concerning the circulation of water in natural conditions, which we are putting to successful use in the design of equipment of this type.

Mr. ZINN (USA): I shall attempt to answer the questions the representative of the USSR has posed.

First, on the capacity or size of a boiling water reactor, we believe that there are practical designs which go up to 1000 megawatts of heat, which ought to be large enough to satisfy anyone.

It was stated that one cannot expect to make a pressure vessel of more than 2-meter diameter. We certainly expect to be able to do this because, as I explained, the boiling water reactor does not require an overpressure. The pressures are modest, and we can envisage pressure vessels of perhaps as much as

4 meters in diameter. This, of course, gives a considerable increase in the steam rate.

We agree that the problem which has been proposed, namely the removal of steam from the core, is more crucial than the heat transfer rate from the fuel elements.

Secondly, we do not propose for large-scale reactors that natural circulation be used. Our experiments in the laboratory on boiling show very clearly that natural circulation is limited. However, it is very useful for reactors in the small and intermediate scale.

When one goes to forced circulation, it must be recalled that the pump required for the forced circulation does not have nearly the strain on it and it does not require as much power as the pump which is needed in a circulating solid water reactor. The water channels in a boiling reactor are naturally wide, and therefore the pressure head required in the pump is small, although the capacity will be large.

With forced circulation, we do not see this limitation in the amount of steam that can be obtained from the core.

I shall point out another feature, which has been proposed by the General Electric Company, namely that some of the water can be withdrawn from the reactor and flashed into steam and used in a further stage of the turbine; thus increasing the power obtained from the reactor without depreciating its safety characteristics.

We agree that these points need verification by actual experiments, and steps along these lines are under way.

However, we do not agree at all that boiling reactors are limited in their size.

I might interpose this remark: that the reactor which I described, being fuelled with enriched uranium, has size difficulties which are removed when slightly enriched uranium is used, because then the leakage of neutrons is into the fuel elements rather than to the outside of the reactor.

The CHAIRMAN: I think at this point I will take five minutes from the Chairman's remarks on the next paper. I will now call upon Mr. Goodlet, of the United Kingdom.

Mr. GOODLET (UK): I must confess, Mr. Chairman, that I have a certain agnosticism about boiling reactors because there is a fundamental contradiction which has never yet been resolved for me. If you wish to obtain a large amount of power from a boiling reactor you have to make it produce a large amount of steam. If you produce a large amount of steam you have a large amount of voidance in the core and you run into oscillation difficulties which are very well described in Professor Zinn's paper. I hope that his work will result in the resolving of those difficulties in fact and not only in argument.

Now I have one question to ask on ultimate safety. It seems to me that if a steam main of a boiling reactor fractures you let pressures down to something like atmospheric, and a large amount of flash steam will form, but the flash operation will also cool what water remains in the reactor, and you will be left with a reactor full of water at about 100°C. That will also happen in pressurized water reactors and from that point of view as regards ultimate safety and fractures resulting from water and steam I can see very little difference between these two reactors. Possibly there is something that Professor Zinn has up his sleeve.

Mr. ZINN (USA): Your observation is correct. I am only sorry that time did not permit me to show the experiment that we have done. In that case we actually opened the steam lines which allowed the reactor to depressurize itself. In these cases we have found that this is a relatively slow operation for any steam line that is attached; it may vary from 20 to 30 to 40 seconds, during which time you can insert your shut-off rods before you get the temperature down where the reactor, with the lower temperature, begins to have excess reactivity due to flashing.

Another experiment has been done, and that is, steam has been allowed to flow out of a small opening while the reactor is not shut off, and after a while the top reflector vanishes, and that, too, acts to shut off the reactor; so that the situation you mentioned exists, but it is by no means a hazardous one because of the long time constants involved.

The CHAIRMAN: I have a written question from Dr. Dunworth, of the United Kingdom, which I will now read out in order to save time: "Is there any evidence for a reaction between water and aluminum?"

Mr. ZINN (USA): Of course there is no evidence in our work; but I would say that wherever one deals with these fuel elements I think one should bear in mind the possibility of chemical reactions, since, of course, these are permitted thermodynamically, and many experiments have been done on this point. Where the aluminum had a thickness and condition similar to those used in a reactor, no reactions were observed. Where special efforts have been made to condition the aluminum either by contaminants or by sub-dividing it in a particular way, then a chemical reaction has been observed.

Session 11A

DESIGN OF REACTORS FOR POWER PRODUCTION

LIST OF PAPERS

Recovery of the Energy Produced in Air Cooled Graphite Reactor G1

By P. Chambadal* and M. Pascal,† France

The five year plan which has been carried out under the authority of the French Atomic Energy Commission since 1952, includes, among other objectives, the construction of the first two French graphite moderated reactors. The building of G1, which started in May of 1954 at Marcoule, a place in the south, 30 km to the north of Avignon, is progressing in such a way that this reactor, the purpose of which is plutonium production, could start up at the beginning of 1956 and reach full power at the beginning of the second half of the same year.

G1 REACTOR

It is known that this first reactor will use natural uranium as a fuel, graphite as a moderator, and air under a pressure close to that of the atmosphere as a coolant. The total thermal power expected from the reactor is approximately 40 Mw for a load of 100 tons of magnesium clad natural uranium made up of 3.80 m long rods. The uranium is made up into slugs 100 mm long, 26 mm in diameter; the cladding metal is provided with 8 fins and its outer diameter is 62 mm, the channels in the graphite having a diameter of 70 mm.

The graphite pile, the general form of which is that of a cylinder 8.840 m long (with 0.660 m of reflector material on the average) and 9.600 m in diameter (with 0.660 m of reflector material on the average), is pierced in the central section by an 80 mm slit. It is perforated by 1337 horizontal channels for each half of the pile, which will receive the uranium.

The cooling air, which is drawn in at the top of a 25 m stack, goes over a set of intake filters which remove, according to the preliminary tests, 97% of the dust contained in the atmosphere. The groups of electro-blowers are on the intake side of the reactor, immediately following the filtering system. These blowers driven by 5500 v engines of 2600 kw unit power, have been designed to deliver 80 m^3 of air per second under a pressure of 2500 mm of water. Before reaching the reactor, the air goes through cooling towers (thermal exchange air-water—water flowing over metal plates—water output: 200 m^3/hr—filter at the cooled off air outlet) which brings its temperature down to 18°C at the reactor intake.

A manifold then takes the cooling fluid to the slit provided in the reactor. Tests have been carried out by the Rateau Company on a model, in order to check on the theoretical distribution per channel.

Under these conditions, the air which has gone through the reactor comes out at a temperature close to 220°C. Electricité de France and the Atomic Energy Commission computed, once the data on the project were known, that it would be worth while to design and build a power recovery installation. This, let it be made very clear, was undertaken well after the reactor as such was being designed. It can be said that the whole project is conservative: G1 is the first French production reactor and its aims are limited; the power recovery plant is, fundamentally, a prototype meant to enable the French technicians to study, as early as possible, how a power producing installation which derives its heat from a nuclear source operates.

POWER RECOVERY

The air which comes out of the reactor goes through an air-water exchanger, the features of which are indicated below:

Exchanger surface: 58,000 m^2
Pressure rating: 350 kg/cm^2
Gas: Temperature at intake: 220°
Temperature at outlet: 80°
Output: 257 kg/sec
Pressure at intake: 300 kg/m^2
Average speed of flow through the exchanger: 6.95 m/sec
Water: Temperature at intake: 60°
Temperature at outlet: 200°
Output: 61 kg/sec
Pressure at intake: 20 kg/cm^2
Speed of water: 1.18 m/sec

The amount of heat exchange will reach 31,500,000 kcal/hr, with losses of 315,000 kcal/hr on the walls, for an efficiency of 99%.

Thus, no vaporization takes place in the exchanger, which rules out the accumulation of salts, inevitable in an ordinary boiler, with the risk of corrosion and a reduction of the heat transfer coefficient. In addition the heat exchanger is very simple, and it can be hoped that the maintenance and repair operations, which are

Original language: French.
*Electricité de France.
†Industrial Administration, CEA.

all the more critical since the hot fluid is activated by going through the reactor, will be greatly reduced. Finally, the connections between the heat exchanger and the equipment room (where the boilers and the turbine are grouped) are limited to a small diameter pipe, making it possible to set the equipment room as far from the reactor as needed.

Following heating in the heat exchanger, the water goes through flash boilers, where simultaneous reduction in pressure and temperature causes partial vaporization of the liquid, after which that stream gives its pressure up to a turbine. A study of the thermodynamic cycle made it possible to establish, for a final temperature of 30°C, some curves which give the theoretical utilizable energy in kcal/kg, as a function of the intake temperature and of the number of boilers.

Three series-connected boilers will be provided: they will supply steam to three stages of a turbine, the main features of which are given here, for normal operation:

1. Boiler Circuits
   Water Output
   - Intake, 1st boiler ..... 61.2 kg/sec
   - Intake, 2nd boiler ..... 55.67 kg/sec
   - Intake, 3rd boiler ..... 50.99 kg/sec

   Quantity of water vaporized
   - In 1st boiler ..... 5.53 kg/sec
   - In 2nd boiler ..... 4.68 kg/sec
   - In 3rd boiler ..... 3.85 kg/sec

   Enthalpy of the water
   - Intake, 1st boiler .....203.2 cal/kg
   - Outlet, 1st boiler .....156.8 cal/kg
   - Outlet, 2nd boiler .....112.5 cal/kg
   - Outlet, 3rd boiler ..... 69.94 cal/kg

   Water pressure
   - 1st boiler ..... 5.64 kg/cm²abs
   - 2nd boiler ..... 1.57 kg/cm²abs
   - 3rd boiler ..... 0.318 kg/cm²abs

   Water temperature
   - 1st boiler .....155.7°C
   - 2nd boiler .....112.2°C
   - 3rd boiler ..... 70°C

2. Temperature of the circulating water ..... 20°C

3. Turbine
   Steam output
   - Intake, 1st group of stages...19,950 kg/hr
   - Exhaust, 1st group of stages.19,500 kg/hr
   - Intake, 2nd group of stages..36,400 kg/hr
   - Exhaust, 2nd group of stages.34,380 kg/hr
   - Intake, 3rd group of stages..48,240 kg/hr
   - Exhaust to condenser.....46,140 kg/hr
   - Titer at exhaust..... 975 kg/hr

   Water pressures
   - Intake, 1st group of stages... 5.64 kg/cm²abs
   - Intake, 2nd group of stages.. 1.57 kg/cm²abs
   - Intake, 3rd group of stages.. 0.318 kg/cm²abs
   - Exhaust to condenser..... 0.035 kg/cm²abs

   Enthalpies of steam
   - Intake, 1st group of stages... 657 cal/kg
   - Intake, 2nd group of stages.. 643 cal/kg
   - Intake, 3rd group of stages.. 627 cal/kg
   - Exhaust to condenser.....608.6 cal/kg

4. Condenser
   - Vacuum .....0.035 kg/cm²abs
   - Temperature of condensed water 27°C
   - Enthalpy of condensed water.. 27 cal/kg

5. Extraction of condensed water in the process of undergoing a drop
   Output of condensed water extracted
   - 1st group of stages..... 680 kg/hr
   - 2nd group of stages..... 2300 kg/hr
   - 3rd group of stages..... 2100 kg/hr

The alternating current generator, driven at 3000 rpm, develops 5750 kw under normal conditions, with 5500v between terminals. Its efficiency is 96.75% under a power factor of 0.8.

The turbo-alternator group (Rateau turbine, Alsthom generator) is of the vertical shaft type, which makes it easier to dispose of the water condensed in the various stages of the turbine.

Figure 2 is an over-all schematic of the installation:

It shows a hot water accumulator which, in case there is a breakdown in the current supply that might cause the reactor to shut down, could feed the boilers directly. The accumulator and the turbo-alternator group are so designed that, for an exceptional overload time, limited to 30 minutes (time during which the water needed for the operation of the refrigeration towers comes from a 2000 m³ tank located on a hill

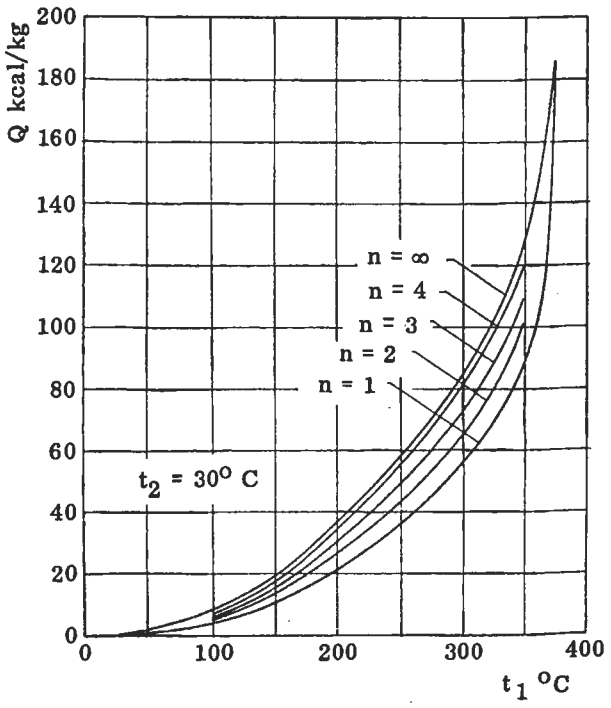


Figure 1. Variation in the useful energy supplied as a function of the number of boilers and initial temperature

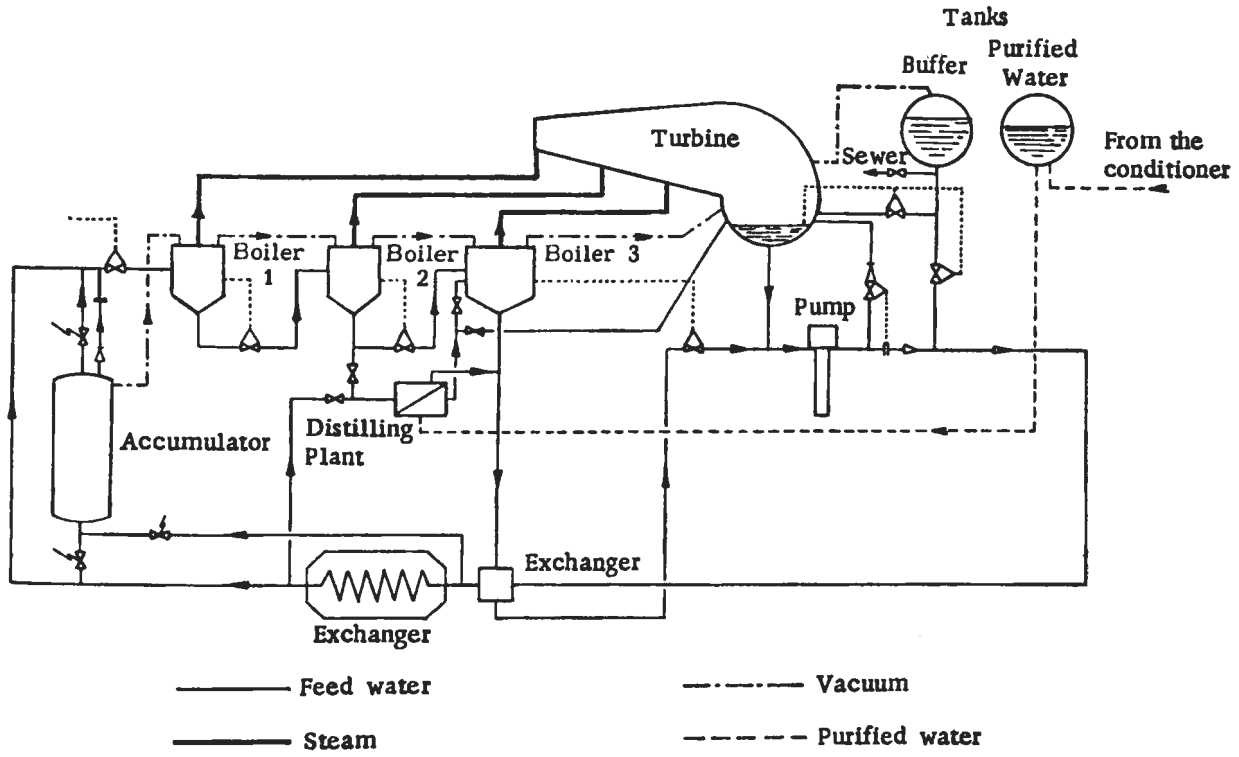


Figure 2. Schematic of the G1 arrangement

close to the site of Marcoule) the average power can reach 8000 kw. During this time, it is possible to drive three electro-blowers, and thus to maintain the reactor at its normal power. With this arrangement, the power recovery system for the G1 reactor is a stand-by feeding system which will doubtless make it possible for current supply breakdowns lasting more than 30 minutes to occur, in the region considered, at an average frequency of once a year, considerably reducing the number of reactor shutdowns due to the lack of current.

ECONOMICAL ASPECTS

Since it must be remembered that the installation here described is not guided by any economical feasibility considerations in its first planning stages, it is stipulated that the investments to which it leads amount approximately to 676 million francs.

Thus, independent of the value as a stand-by, mentioned above, the installation should make it possible to study, at the time when installations on an industrial scale are being planned, how to run a power plant utilizing the heat developed in the coolant used in a nuclear reactor. The technique thus developed should find application in the future, whenever the problem of recovering electrical power from the outlet side of a large reactor, the main purpose of which would be to produce fissionable material (and which thus would operate at moderate temperatures), would present itself.

Design for a Dual Purpose Reactor (G2)

By P. P. Ailleret,* P. Taranger,† J. Yvon,‡ France

In order to come, as rapidly as possible to peaceful applications of atomic energy, namely industrial power plants, the French Atomic Energy Commission chose to resort to large natural uranium reactors using graphite as a moderator and a gas under pressure as the coolant.

By concentrating France's efforts in that direction, namely by avoiding, from the start, their dispersal on the separation of uranium isotopes or the mass production of heavy water, the ground was prepared for reactor G2 by successively building the Saclay reactor (P2) and reactor G1, which are described in other papers from France. In other words, experience in pressurized gas cooling was acquired in an experimental heavy water reactor while a study on the use of graphite was being approached by means of a simple type of air cooled reactor.

The second French graphite reactor, G2, is now being built at Marcoule, in the Gard department: it will be in operation in two years.

We are dealing here with a reactor which uses natural uranium as a fuel, graphite as a moderator, and compressed carbon dioxide as a coolant.

The design arrived at is the outcome of studies carried out jointly by: The French Atomic Energy Commission (Reactor Planning Department), The Electricité de France (Nuclear Equipment Subdivision), and French Industry at large (Société Alsacienne de Constructions Mécaniques, Société des Forges et Ateliers du Creusot, Société Rateau, Alsthom).

ACTIVE CORE

The fuel (100 tons of uranium) is used in the form of cylindrical slugs 26 millimeters in diameter, 300 millimeters in length, made up of very low aluminum content binary U-Al alloy. Each slug is clad in a finned magnesium sheath, to which it is mechanically bonded.

These slugs rest in horizontal channels 70 millimeters in diameter, provided in the graphite for the purpose: this horizontal arrangement, which may be less satisfactory than the vertical system in the mind of some designers, has been used, in view of the simpler metallurgical problems encountered with respect to fuel behavior.

The graphite in which the channels are cut has a specific gravity of 1.7, and a 4 mb capture cross-section (for 2200 m/sec neutrons).

It comes in square cross-section blocks. The side of each square (200 mm) is exactly equal to the dimension of the lattice.

These graphite blocks are piled up to form a right prism, the bisector plane of which is vertical. To the dimensions of the active core (height 8.45 meters, diameter: 7.80 meters), which correspond to 1200 channels, should be added those of a 0.500 m lateral peripheral reflector.

Under these conditions, the reactivity available, including the reserve needed for full power operation (temperature effect) and poisoning compensation (xenon, samarium . . .), of approximately 3500×10^{-5}, leaves a margin of some thousands of 10^{-5} units which will make it possible, in particular, to make U^{233} as a thorium by-product, to make use of slightly depleted uranium, to flatten the neutron flux or to achieve combinations of these three factors.

CONTROL AND SAFETY

The control and safety devices are those to be found in any reactor, and are provided in order to make it possible to measure and adjust the power and reactivity, to measure temperatures and pressures, the intensity of the radiation, and preventively to detect cracks in the cladding.

RENEWING FUEL

This is done without any need for a shut down: the slugs are inserted into the reactor on the outgoing side of the coolant fluid by means of a movable sealed chamber. Each fresh slug is automatically pushed into a channel, where it displaces the slugs loaded earlier in that same channel.

The warm slugs, as they come out, fall by gravity on 45 degree sloped chutes: this slope is the factor which determines the manner in which the graphite blocks are piled up.

All the chutes lead to a helical arrangement in which the fall of the slugs is slowed down, making it possible for them to get outside the sealed vessel at a

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Original language: French.

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†Industrial Director, Atomic Energy Commission.

‡Chief, Reactor Planning Department, Atomic Energy Commission.

low speed independent of the position they occupied in the reactor.

These devices are behind the biological shield.

BIOLOGICAL SHIELD: TANK

The whole of the reactor, the fuel renewal and thermal protection devices must be housed in a gas-tight pressure resistant vessel (carbon dioxide is used for cooling under an average pressure of 150 kg/cm^2), in which a great many holes are provided for fuel loading, gas circulation, control rods, etc.

The fabrication of a cylindrical steel tank, the height and diameter of which would have considerably to exceed 10 meters in order to meet these conditions, would have raised, quite apart from price and time considerations, difficult technical problems of supply and manufacture (welding and annealing in particular).

On the other hand, it was only natural to attempt to make use of the positive need for a biological shield around the reactor to build a gas-tight cement tank.

In this case, however, conventional concrete does not appear to be suitable, for the paradoxical reason that the construction problem involved, mechanically speaking, is too simple: the shape of the structure and the load conditions are reminiscent of those met in a power plant pressure pipe line, and do not specifically call for the essential qualities of conventional concrete.

The solution to which we resorted was prestressed concrete, in which the whole resistance of the cylindrical core is achieved by the prestressing by means of shrinking on forms which plays an important part in the stability of the lower parts of the assembly, and particularly in assembling them with said cylindrical tank.

The choice of moderate prestressing makes it possible to retain one of the advantages of standard concrete over a steel tank: that of reducing an explosion to mere leaks in cases where the wear due to temperature variation and radiation would not make it possible to resist a pressure excursion.

Finally, in order to rule out any leak from the concrete tank, this one is protected internally by a steel liner. By circulating cold carbon dioxide between the liner and thermal shield, the cement can be kept at an acceptable temperature.

COOLING CIRCUIT

The flux in the uranium (2.5×10^{13} neutrons per square centimeter per second at the center of the reactor) decreases toward its periphery. Two temperature limits, according to whether the center or the periphery of the reactor are considered, determine the cooling arrangement: 550°C may not be exceeded in the center of the uranium, nor 400°C over the magnesium liners.

On the other hand, proper balancing of the two purposes of the reactor, namely the production of plutonium and that of energy, requires that a colder fluid be admitted to the central channels.

Accordingly, the intake side for the carbon dioxide has been divided into three concentric sectors where the gas is admitted at different and adjustable temperatures. Laminating, by a channel, guarantees fine adjustment at the outlet.

The gases which come from the three sectors are mixed at the outlet at an average temperature of more than 300°C. However, utilization of the peripheral gas which is warmer, makes it possible to superheat the steam: it is generated in a system of boiler exchangers, and drives an ac generator and carbon dioxide compressor through conventional turbines. A small auxiliary boiler is necessary for the start up of an installation which, at full power, will deliver about 30,000 kw net.

Detection of Breaks in the Cladding in Gas Phase Cooled Reactors

By Jacques Labeyrie* and André Roguin,† France

INTRODUCTION

In a nuclear reactor using rods or plates cooled by a gaseous fluid, breaks or cracks may appear in the cladding which provides gas-tight separation between the coolant (gas) and fissionable material.

Radioactive fission products may escape through these breaks, and thus grossly pollute the outlet side of the reactor. Furthermore, in some instances, the gas which penetrates through the break may react with the metal and cause the slugs or plates to swell, making their removal and subsequent manipulation difficult. In view of this, it is necessary to detect breaks at the earliest possible time.

Some of the breaks or cracks remain small, and thus present no great danger, while others develop more or less rapidly. Consequently, it is also necessary that their development as a function of time be followed.

We shall describe in this paper a device designed to keep a check on all the fuel channels in a reactor. In addition, the main features and construction of such a device are indicated, for the case of a reactor using natural uranium cooled by air circulation at approximately atmospheric pressure.

PRINCIPLE

The physical phenomenon utilized for the detection of the development of a break is the appearance in the stream of coolant gas, on the outlet or exhaust side, relative to the slug or plate affected, of dispersed fission products. The radioactivity of the polluted gases can readily be used, outside the reactor, to reveal the formation of the break, or to follow the development of such a crack. The activity of the gas is sufficient for the detection of fissures having a size of about a square millimeter, even if only a fraction of the coolant is investigated, and if it is observed several seconds after its passing over the crack.

PRINCIPAL FISSION PRODUCTS FORMED

Table I shows the main products of U^{235} fission which may be of interest for this detection, as well as their parent and daughter products, the relevant half lives, the yields of the various products, and the energies of the β and γ rays which they emit.

DEFINITION OF A CRACK

A crack is an opening in the side of the cladding placed around the fissionable material. In the rare instances where the cladding remains in very close contact with the fissionable material, namely where there is no contact between the coolant (gas) and the area of the fissionable material located just at the level of the break, the only fission products which reach the coolant are those ejected from the surface of the fissionable material directly into the gas. The thickness of the layer from which they come is some 4 microns and, accordingly, the number of atoms produced by each uncovered square millimeter of this surface will be relatively very limited.

In the general case, the area of fissionable material which comes into contact with the coolant is much greater than that of the surface of the break or crack proper. There is a certain amount of space between the cladding and the fissionable material due to their being separated, so that the gaseous fission products emitted by this surface can be scattered toward the crack, and thence to the coolant. Thus, summing up, a break or crack will cause a large or small number of radioactive atoms to reach the coolant, according to whether the cladding, at this particular point, is in loose or close contact with the fissionable material.

In the following, when a "1 mm^2" crack is mentioned, this means that a total area of 1 mm^2 of fissionable material is caused to come into contact with the coolant (Figs. 1 and 2).

METHOD USED FOR COLLECTING THE FISSION PRODUCTS DISPERSED IN THE COOLANT GAS

A fraction f of the gas which goes into the channels where the fissionable material, covered by its cladding, is located, is collected by means of tube a and, after a travel which may vary by a few meters, it goes from this tube a on to the radioactivity detector. This detector includes:

1. a collector c which recovers as large a fraction as possible of the fission products suspended in the gas.

Original language: French.

*Chief of the Electron Physics Section, Atomic Energy Commission.

†Engineer in the Electrical Construction Department, Atomic Energy Commission.

2. a radiation detector d which measures the radiation emitted by these products, in as good a yield as possible, so as to increase the sensitivity of the detecting process.

On the other hand, the detector is chosen as to be as insensitive as possible to the parasitic radioactivity of the coolant created by its irradiation in the reactor, since this always is present, even in the absence of any crack in the cladding.

Following passage through the detector, the cool-

Table I. Main Fission Products of U^{235} of Interest for the Detection of Cracks in the Cladding: Parents, Daughters, Half-lives, Fission Yield, β and γ Energies in Mev

Families of the Kr's

A \ Z	35 (Br)	36 (Kr)	37 (Rb)	38 (Sr)	39 (Y)	Fission yield %
85	3min β 2.5	4.5hr β 0.9, γ 0.37				0.24
86			19.5d β 1.6, γ 1.0			~ 1.5
87	55sec β 2.6, γ 5.4	75min β 4	6×10^{10}yr β 0.13			~ 2
88	16sec β	3hr β 2.5, γ 0.027	18min β 5, γ 1.8			~ 3
89	4.5sec β	3min β, γ 4	15min β, γ 4.5	54d β 1.5		4.6
90		33sec β, γ 3.2	2.7min β, γ 5.7	20yr β 0.6	61hr β 2.2	~ 5
91		10sec β 3.6	1.7min β, γ 4.6	9.7hr β 3.2, γ 1.3	61d β 1.5, γ 1.2	6
92		3sec β	short β	2.7hr β	3.5hr β 3.5, γ 0.7	5
93		2sec β	short β	7min β	11hr β 3.1, γ 0.7	~ 6
94		1sec β	short β	2min β	17min β 5.4, γ 1.4	5

Families of the Xe Isotopes

A \ Z	53 (I)	54 (Xe)	55 (Cs)	56 (Ba)	57 (La)	Fission yield %
131	8d β 0.6, γ 0.6					2.8
132	2.4hr β 2.2, γ 0.7					4.4
133	21hr β 1.4, γ 0.5	5.3d β 0.34, γ 0.081				6
134	52min β 1.6, γ 2.2					~ 6
135	6.7hr β 1.0, γ 1.3	92hr β 0.9, γ 0.25	3×10^{6}yr β 0.2			7
136	1.8min β 6.5, γ 2.9		13d β 0.3, γ 1.2			7
137	22sec β 0.7	3.5min β 4	37yr β 0.5			6
138	5.9sec β	17min β	30min β 2.6, γ 1.2			6
139		41sec β 2.1	9.7min β	85min β 2.27, γ 0.16		6
140		16sec β	66sec β	12.8d β 1.02, γ 0.5	40hr β 1.3, γ 1.6	6
141		3sec β	short β	18min β, γ 2.8	3.7hr β 2.4, γ 1.5	5.7
142			1min β	6min β	77min β	~ 5.5
143		1sec β	short β	0.5min β	20min β	5.4
144		short β	short β	short β	short β	5.3
145		0.8sec β	short β	short β	short β	~ 5

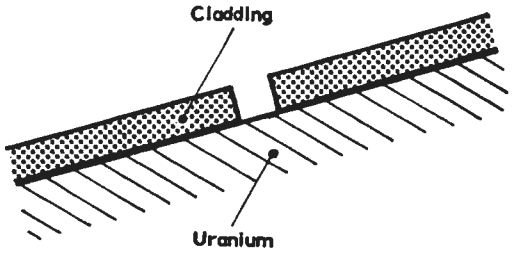

Figure 1. The type of breaks or cracks

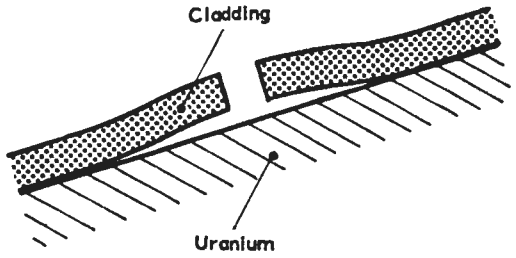

Figure 2. The type of breaks or cracks

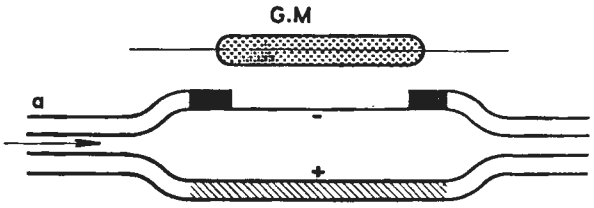

Figure 3. Collection on a charged window
The type of electrostatic collection

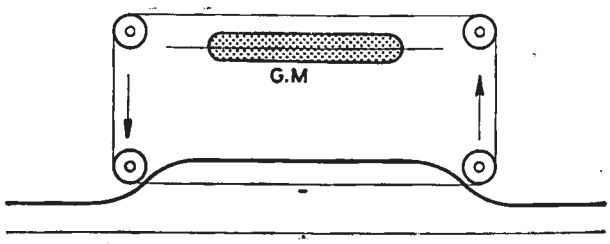

Figure 4. Collection on a charged wire
The type of electrostatic collection

ant is ejected to the atmosphere, if it is not costly (air), or recovered, in order to be returned to the reactor, if it is a costly gas (CO_2).

Collection of the fission products dispersed in the gas in the vicinity of the detector can be carried out in various ways: when the detector is not far from the reactor, the electrical charges carried by the fission product particles in suspension in the gas can be collected. We show, on Figs. 3 and 4, various collecting methods in which use is made of these electrical charges.

Fission particles also can be collected by mechanical filtration, by means of a high yield filter made up of small diameter fibers about one micron across. We show, on Fig. 5, the yields given by these collecting methods.

It is obvious that the mechanical filtration method is interesting only in such cases as there are fission products in the gas, in the vicinity of the detector, made up of solid elements such as, for instance, rubidium or cesium. The ratio of such fission products to those which are gaseous, such as xenon and krypton, is important, at the level of the detector, only if the crack in the cladding enables the fission product atoms to be projected directly into the gas without striking a wall. It is a known fact that the distinguishing feature, whether they appear in an atomic or molecular form, between the particles of a gas and those of a solid or a liquid, is that the former can rebound from a wall, whereas the latter adhere to it on the first impact.

For the solid element particles to reach the detector, it is further necessary that the distance covered inside the sampling tube *a* be sufficiently short and sufficiently large in diameter, and that this be done sufficiently rapidly that most of the particles present in the gas do not hit the tube walls (and remain stuck to them). For instance, in a tube having an internal diameter of 1 cm, through which air circulates under atmospheric pressure at a speed of 10 m/sec, the proportion of the solid particles which disappear due to this scattering effect is approximately 20% per meter of tube length.

It should be noted that, besides the solid particles which come directly through the crack, there are others which appear in the mass of the gas by disintegration of the gas atoms, but their proportion is small, and using them is not worth while. Collection of the xenon and krypton atoms is difficult. Although, in most cases where cracks appear, they make up the bulk of the fission product atoms at the location of the detector. The simplest method for using them is to "bell" the sampling tube *a* at the location of the detector, in order to form a large volume around it. Figure 5 gives some interesting values of activities due to solid particles and gaseous products.

BACKGROUND ACTIVITY OF THE COOLANT GAS

As indicated above, the gas used as a coolant always shows some activity due to its irradiation in the reactor. For instance, in the case of air, one liter under atmospheric pressure, which remains for one second in the flux of 10^{12} n/cm^2/sec will acquire an activity, immediately after irradiation, of approximately 15,000 disintegrations per second, due to A^{41}, N^{16} and O^{19} formed by the capture of thermal neutrons. Table II gives the properties and nature of the radiation emitted.

It is difficult to design, from the outset, the best detector, namely that which would give most contrast between the normal activity of the coolant and that due to the presence of fission products, in view of the great variety of the latter.

For a given reactor construction, sampling tube length, pressure and output of gas, it is possible to choose experimentally the optimum radiation detector. As an example we show, in Table III, the various

Table II. Radioactive Gases Found in the Air, following Passage through a Neutron Flux

Volumetric Composition of the Air

	% of volume	*Specific mass, gm/l*
N_2	78.03	1.2506
O_2	20.99	1.429
A	0.94	1.784
CO_2	0.03	1.977
H_2	0.01	0.0899
Ne	0.0012	0.9002
He	0.0004	0.177
Kr	0.00009	3.708
Xe	0.00001	5.851

Argon 41 — due to the activation of argon-40. Isotopic abundance of A^{40} 99.6% δ = 0.6 barn

T = 1.82 hr — β: 1.24 (2.5); γ: 1.3 (0.7%)

Nitrogen 16 — due to the activation of nitrogen-15. Isotopic abundance of N^{15} 0.365% δ = 0.08 mb.

T = 7.3 sec — β: 3.8 (40%), 4.3 (40%), 4.6 (2%), 10.5 (18%); γ: 6.2, 6.7

Oxygen 19 — due to the activation of oxygen-18. O^{18} isotopic abundance 0.365% δ = 0.22 mb

T = 29 sec — β: 2.9 (70%), 4.5 (30%); γ: 1.6

Neon 23 — due to the activation of neon-22. Ne^{22} isotopic abundance 8.8% δ = 0.04 b

T = 40 sec — β: 4.21 (93%), 1.18 (30%); γ: 3

For 1 liter of air which remains for one second in a flux of 10^{13} neutrons/cm²/sec, the respective activities are approximately the following, immediately after irradiation.

Activity of argon-41	14,000 dps
Activity of nitrogen-16	1000 dps
Activity of oxygen-19	100 dps
Activity of neon-23	400 dps

counting rates obtained with two types of Geiger-Muller counters of conventional design, one having a thick wall, the other a thin wall, using, as a coolant, 18 liters per second of air under atmospheric pressure, a 1/10 sampling fraction, a 40 meter sampling tube length and a 1.6 liter gas volume, around the counter. The geometry is shown in Fig. 6.

EXAMPLE OF A LEAKAGE DETECTOR

We describe here, as an example, a device which is designed for the detection of leaks in an air cooled reactor having approximately 2600 fuel channels, in which the fissionable matter is natural uranium, and the moderator, graphite.

In each of the channels, a tube having a diameter of 10 mm takes out a fraction of the cooling air which is approximately 5% of the total volume going through the channel. The 2600 tubes used are grouped in bundles of 135 units, which feed a single collector

Table III. Counting Rates of Two Types of Geiger-Muller Counters Placed in a Given Volume of Cooling Air

Type of counter	10 B 12	3 G 3
Wall thickness	26 mg/cm²	500 mg/cm²
Activity due to the air alone: A_1	43	20
Activity due to the air loaded with fission products	110	4.3
A_2/A_1 Ratio	2.5	87

The source of fission products was a 9 mm² surface of uranium metal placed in a neutron flux of 7×10^{11} n/cm²/sec.

The cooling air stayed in this flux for approximately 0.2 seconds, and the time needed by the air to reach the detector was 2 to 3 seconds.

through 135 electromagnetic valves. The single collector leads to a detector made up of a cylindrical volume of 1.6 liters, approximately 10 cm in diameter. A halogen filled Geiger counter, with a 2 mm thick glass wall, 10 cm long, is placed axially in this volume. The readings of the counter are fed into an electronic device which includes an integrating circuit and a single channel recorder having a band width of 25 cm, of the quick responding type, namely 3 seconds for full scale deflection. The 135 electromagnetic valves send air into the detecting volume, from the 135 sampling tubes in succession. Each valve remains open

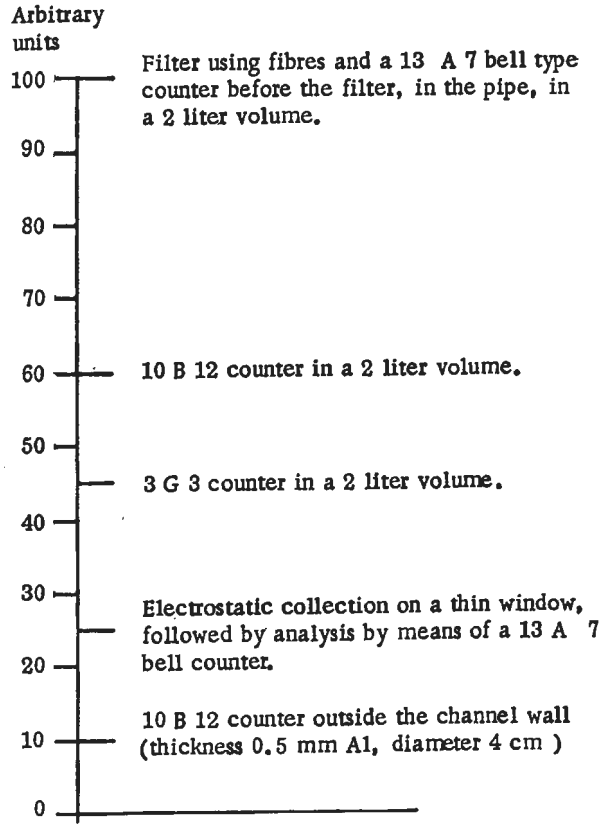

Figure 5. Yields. Comparison between the various systems used for the measurements of activity of fission products. Distance between the point of activation and the point at which the measurement is made: 5 metres. Air output: 15 liters. Speed: 13m/sec.

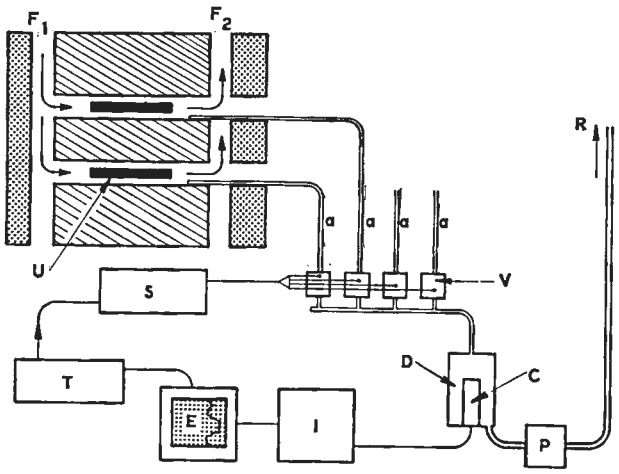

Figure 6. General schematic. *U*: clad uranium bars. F_1: intake, coolant gas. F_2: output, coolant gas. *a*: sampling tubes. *V*: electromagnetic valves. *D*: radioactivity detector. *C*: Geiger-Mueller counter. *P*: vacuum pump. *R*: return to stack. *I*: integrator. *E*: recorder. *S*: selective valve control. *T*: signaling and alarm panel

for about 20 seconds in order for the air to travel the length of the sampling tube (some 40 meters). The time required for a check-up on 135 channels is approximately 45 minutes. The whole of the reactor is explored by a set of 20 similar devices. All these operate constantly, night and day.

When a signal indicating that the activation of the air from a given channel exceeds a preset threshold, a contact on the recorder stops sample analysis from the tube serving that channel. Two situations are possible: 1, Activation already was substantial at the time the detector was connected with the cell, namely the leak offers immediate danger, and the reactor is shut down; 2, the leak is not very extensive and, in this case, no reactor shutdown is required. In such a case, the tube from the channel where the leak is located is manually connected, by means of a set of valves, with a special "follower" detector, similar to the first, and the object of which is to record permanently the activity of the channels in which there is a leak. For the whole of the reactor, there are four follower detectors. Once the channel containing a leak is connected with the follower, continuous check-up by the normal service detector is started again on the other 134 cells.

Various electronic warning devices, which make it possible to insure that the whole device operates properly, both electronically and as regards the output of gas in the various tubes, are also provided. Two vacuum pumps having an output of some 70 m^3 per hour aspirate air through the 20 sampling tubes which operate simultaneously, as well as through the tubes of the follower detectors.

The whole system is housed in two special rooms which have a total volume of 300 m^3; the general control panel of the reactor carries only the recorders connected with the follower detectors, the leak warning signal indicators and the signaling devices showing that the equipment, as a whole, is operating properly.

The sensitivity of the equipment is adequate, allowing for a normal background activity of the cooling air, to detect a leak having an area of only 5 mm. As indicated above, in view of the separation which generally takes place on the edges of a crack and in view, also, of the porous nature of the uranium oxide formed, the device actually can detect much smaller cracks.

Figure 7 shows a sample recording made during tests carried out on P2 (Saclay) for the development of the device, under conditions identical to those indicated above. It will be seen, in particular, that the activity shown with an observation time of 20 seconds in air coming out of the channel where there is a crack, is some two thirds of that found following a very long observation time (5 minutes). It will also be seen that for a total crack area of 9 mm^2 activity of the air coming out of a channel is five times greater than when there is no leak.

Allowing for the fact that the various channels of the reactor do not all offer the same signal to noise ratio (due to variations in air output, in the duration of the stay of this air and local intensity of the neutron flux), the detection sensitivity achieved in the course of industrial operation of the reactor varies between some 4 and 25 mm^2 of total "apparent" crack surface.

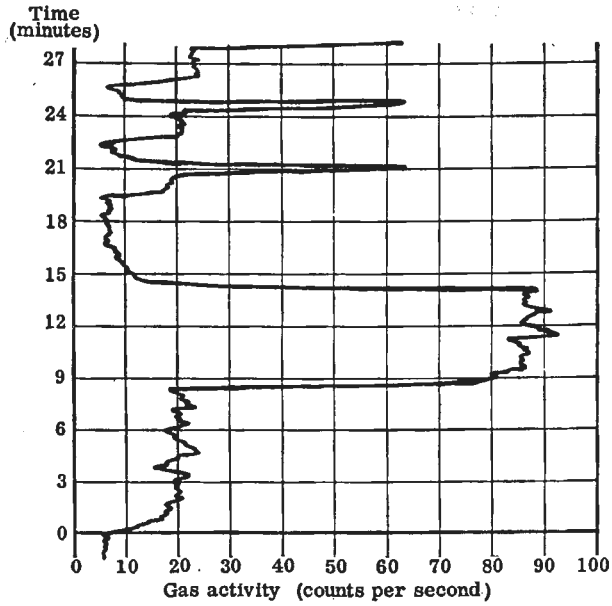

Figure 7. Sampling recording. a. At C time: non-active atmospheric air. b. 0 to 8 period: active air in a cell, no crack in cladding. c. 8 to 14 period: air from a cell with a break having an area of 9 mm^2 (metallic uranium). d. At times 21, 24.5 and 28, aspiration for 20 seconds from the cell with a crack in the cladding. Neutron flux at level of crack: $7 \times 10^{11}/cm^2/sec$

A Graphite Moderated Nuclear Power Plant Design*

By R. K. Andersen, A. B. Carson and J. E. Love, USA

INTRODUCTION

This paper describes a nuclear power plant of broad potential application. The reactor concept is based on extrapolation of technological, engineering, construction, and operational information gathered in more than ten years of experience with similar operating reactors. With the anticipated successful development of long-exposure fuel element designs, this plant is capable of producing electrical power at an economically competitive level in many parts of the world. The design scope presented is for a heterogeneous, graphite moderated, high pressure water cooled reactor of 800 megawatt thermal rating, combined with a power generation system capable of producing 223,000 kilowatts of net electric power. The plant also produces fuel-grade plutonium and other isotopes for use in research, medical, and industrial applications. The concept is presented in terms of its major subdivisions: (1) reactor and reactor cooling system, (2) power generation system, and (3) water treatment and supply system. The pressurized reactor cooling water is contained in a forced circulation primary loop; the heat generated in the reactor is given up in heat exchangers. Steam is formed in the heat exchangers and is passed to the turbine in a separate secondary loop. Water is supplied for condenser cooling, primary and secondary loop make-up, and to other auxiliary uses. The plant operating and flow diagram is shown in Fig. 1.

* From work performed by the staff of the Reactor Design and Development Unit, Engineering Department, Hanford Atomic Products Operation, General Electric Company

REACTOR

The reactor moderator consists of a rectangular stack of graphite blocks, 18.4 feet wide, 18.7 feet high, and 24 feet in length. Graphite is of special interest as a reactor moderator since it combines good

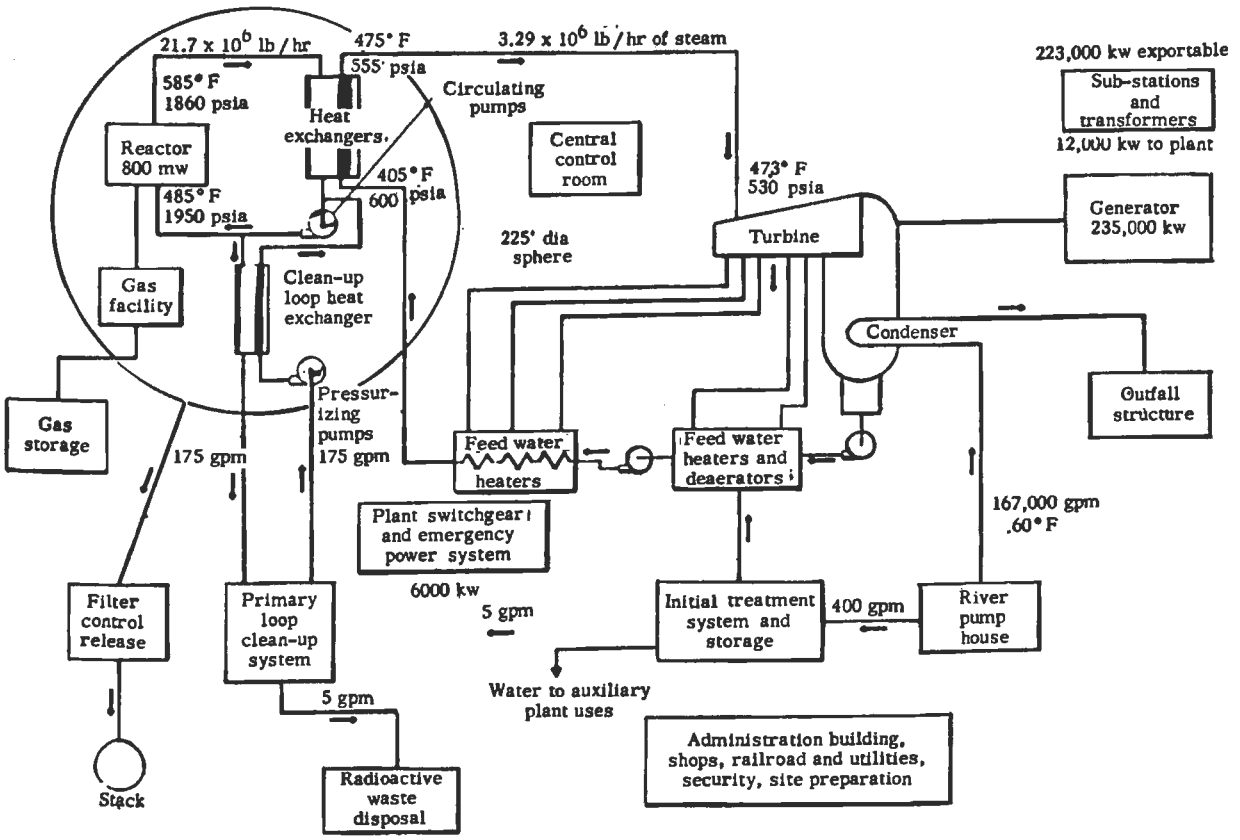

Figure 1

nuclear properties with other advantages of nuclear control, instrumentation, and piping geometry. In addition, there is a relatively complete material technology available in the industry, as well as extensive operating experience in large scale reactors in the United States and Great Britain.

The graphite moderator is perforated by 1521 zirconium alloy process tubes spaced in a uniform rectangular array of 39 tubes square on 4.5 inch centers. The process tubes have an outside diameter of 1.38 inches. Longitudinal ribs on the inside support cylindrical fuel elements and provide a nearly uniform cooling water annulus of about 0.075 inch.

The lattice geometry is arranged for a fuel requirement of 2.9 pounds of slightly enriched uranium per foot of process tube. Although fuel element performance under the design operating conditions has not been demonstrated, this fuel requirement is less than the maximum attainable density of uranium, and thus permits considerable flexibility in the choice of geometry and composition. This flexibility is an advantage in the application of alternate designs for optimizing fuel element performance. As one example, the fuel material can be pellets of uranium metal, 0.92 per cent U^{235}, uniformly distributed in a magnesium alloy matrix to obtain 65 per cent uranium and 35 per cent magnesium alloy by volume. The individual fuel elements, in this case, are externally cooled hollow rods having an outside diameter of 0.9 inch, an inside diameter of 0.3 inch and an overall length of 7.91 inches. Each is clad in a zirconium alloy jacket with wall and end cap thicknesses of 0.020 inch and 0.045 inch respectively. Thirty fuel elements are centered in each process tube to obtain an active column length of twenty feet. The total reactor fuel weight is 45 tons. The total length of each process tube is 38.6 feet, since each must extend through the thermal and biological shielding to connect with the external cooling water system. The fuel column is centered in the process tube by means of inert elements of the same external dimensions as the fuel.

The reactor core is completely surrounded by a cooled cast iron thermal shield and a heavy aggregate concrete biological shield. The purpose of the thermal shield is to reduce the heat load on the biological shield. The biological shield is designed to provide adequate shielding for personnel protection in the immediate reactor area during shutdown periods and to prevent radioactivation of materials and equipment adjacent to the reactor during operation. Personnel access to the immediate reactor area for any but the shortest periods will not be possible during operation due to the radioactivity and temperature of the recirculating cooling water. However, secondary shielding walls are provided in the building structure to reduce radiation levels to acceptable limits outside the immediate reactor area at all times. The two sides and the top of the biological shield consist of structural steel framework filled with heavy aggregate concrete and are water cooled to control maximum shield temperature as well as the temperature differential across the shield. The front and rear shields are constructed of a more elaborate steel framework containing accurately positioned sleeves for each process tube. The space between the tube sleeves is filled with a dense mortar consisting of steel shot, magnetite sand, portland cement and water. The front and rear shields are keyed to the side and top shields for lateral support, but are free to expand and contract with changes in pile coolant temperature independently of the sides and top. The bottom shield, which is the reactor base, is of ordinary concrete.

Outside the biological shield at both the inlet and outlet faces of the reactor are large high pressure water chambers which encompass the entire process tube pattern. Thermal expansion of reactor process piping is accommodated by roller bearings at the base of these chambers.

An assembly of tube protecting sleeve, concentric shielding rings, gas seal, and cooling water fitting and connector is provided for each tube at both these chambers or "water walls." The process tubes extend through the water walls and are connected to the fittings which are anchored to the outside surface of the water walls. The cooling water enters and leaves the process tube through individual connectors between each fitting and the appropriate water wall. Valves are provided in the inlet face fittings to permit individual tube flow regulation during charging operations. A perspective view of the reactor block and process piping is shown in Fig. 2.

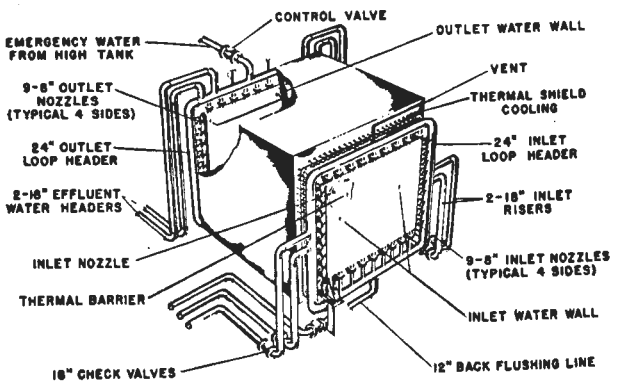

Figure 2

Facilities are provided for charging fuel elements into the process tube at the inlet face and discharging them from the outlet face. These include a platform elevator extending across the inlet face upon which are mounted hydraulic manual charging machines with self-contained power units. Sufficient shielding is provided around the charging machine cab to permit operation in the presence of the radioactivity expected at the inlet face piping. A platform elevator is also provided across the outlet face. It is equipped

with a remotely operated machine for performing the discharging function and a limited amount of maintenance work on the outlet face where radiation conditions will not permit manual operation.

Upon discharge from the outlet end of the process tubes, the irradiated fuel elements fall directly into a water filled basin. They are guided by a discharge chute to an underwater pick-up station which is separated from the reactor by a shielding wall. The water depth is 18 feet which provides adequate shielding for protection of operators located on platforms immediately above the surface of the water. The irradiated material is transferred under water into storage buckets. These buckets are suspended under the water from yokes which travel on a monorail system and permit movement of the irradiated material from the pick-up station to storage and shipping facilities. The storage basin is provided with 150 buckets which give a storage capacity of approximately 75 tons of fuel.

Facilities are provided for maintaining a controlled inert gas atmosphere for the graphite moderator. For this design the reactor atmosphere is helium. The gas is introduced between the graphite stack and the thermal shield at the inlet face of the reactor. A system of baffling provides for uniform distribution of the gas, which circulates slowly through small passages formed at the adjoining edges of individual blocks in the graphite stack. The effluent gas is monitored for moisture content and the moisture is removed by use of a dessicant. In event of a process tube rupture, large quantities of steam enter the moderator. The reactor must then be shut down, the faulty tube replaced, and the bulk of the water removed from the moderator by means of the gas system. This can be done with the reactor shut down by recirculating hot water in the primary coolant system, and educting the steam that is formed to the exhaust system.

The design incorporates two basic systems of nuclear control for the reactor, one for regulating reactor power level during normal operation, and a second for insuring rapid and positive reactor shutdown. The power level regulation is accomplished by a system of 26 water cooled 1.38 inch diameter aluminum rods, packed with boron carbide. These rods can be inserted into the reactor core through horizontal channels extending 16.5 feet into the moderator from one side of the reactor. The rod pattern is 6 by 5, without corners, spaced 37 inches apart front to rear, and 22.5 inches apart top to bottom. The position of each rod can be varied individually by means of direct current electric motors with gear drives. This system has a variable operating speed from 0.1 inch to 8 inches per second. The control rod system is connected to the reactor safety instrumentation so that in an emergency, all rods are automatically driven into the reactor at maximum speed. Battery power is provided to the rod drive system to insure rod operation if the normal power source fails. The control rod system is capable of reducing the reactor reactivity by about 4.7 per cent which is adequate capacity for reactor power regulation through all normal operating transients.

The safety control system consists of 26 dual hoppers in the top shield filled with 0.375 inch diameter steel balls containing 1.5 weight per cent boron. When the hopper gates are released, the balls drop into and fill 26 vertical slots, 2 by 4 inches, extending through the reactor moderator. The slot pattern is 6 by 5, without corners, spaced 38 inches apart front to rear and 31.5 inches apart side to side. To insure operability, each hopper has two storage bins and automatic releasing mechanisms so that if one mechanism fails to operate, the other will function approximately 0.25 second later. The safety control system is capable of reducing reactivity by 7.5 per cent which is sufficient to shut down the reactor and hold it subcritical under all credible conditions. The system is highly reliable, and will function effectively even though distortion of the moderator may have occurred as a result of earthquake or other abnormal condition. The balls are removed from the reactor by opening valves provided at the bottom of each slot. Inclined ducts lead the discharged balls into storage bins located at the base of the reactor from which they are subsequently returned to their original hoppers in the top shield by means of an electromagnetic conveyor and distribution system. The reserve charge of balls in each hopper is available for operational use during the time period in which the first charge is being returned to the top of the reactor.

Both the reactor and the power generating system are controlled from a single center, located in the control building external to the reactor building. All process information necessary for safe and efficient operation of the plant is transmitted to the control center. This information includes: neutron flux level and distribution within the reactor; radiation levels in critical zones outside the reactor; primary coolant system pressure, temperature, and flow; moderator temperature and gas atmosphere composition; steam generating system temperatures, pressures, and flows; and electrical generator loads. Instrumentation is provided for certain system variables critical to the safety of operation such that warning is given when a dangerous condition is approached, and automatic reactor shutdown is initiated if the variable exceeds predetermined limits.

REACTOR COOLING SYSTEM

Heat is removed from the reactor by recirculation of 21,700,000 lb/hr of high purity water through the process tubes to heat exchangers. At full power the reactor outlet water temperature is 585°F and the pressure is 1860 psia. This pressure is sufficiently high to prevent boiling in the process tubes under

conditions of limited power and flow transients. The heated water is piped to four shell and tube heat exchangers in which steam is generated in the secondary loop. The heat exchangers and the entire cooling system external to the reactor are constructed of carbon steel. The reactor cooling water leaves the heat exchangers at 485°F and 1840 psia.

Circulation is maintained by four primary pumping units, each consisting of a pump, a flywheel and an electric motor. The pumps are operated at a suction pressure of 1840 psia and a discharge pressure of 1950 psia. Normal rotation speed is 1800 rpm. The 1000 horsepower induction type motors are equipped with two-speed windings to permit reduced flow operation during reactor shutdown.

If the electric power to the primary pumps is lost, the reactor is shut down. The flywheels allow the pump speed to gradually decrease from the 1800 rpm normal speed to the 600 rpm secondary winding speed at which time the emergency power source would supply the pump motors to maintain shutdown flow. The flywheels are sized to provide sufficient energy to maintain adequate reactor cooling during the shutdown transient period. The primary pumps are single stage centrifugal pumps, and are equipped with special shaft seals to minimize leakage of the radioactive cooling water. Each seal system consists of a high pressure water chamber bracketed by carbon disc seals and labyrinth breakage glands. A separate supply of high purity, non-radioactive water is fed to the water chambers at a pressure slightly above the primary loop pump suction. Proper adjustment of this pressure allows operation with essentially zero leakage into the primary loop pump, and less than 5 gpm of non-radioactive water past the outer disc seal to a low pressure return system.

Three high pressure water make-up pumps are provided for the primary cooling loop. Normally, two of these pumps are in operation to provide 150 gpm to replace water withdrawn for purification, and approximately 25 gpm to supply the leakage through all the primary pump seals. The third pump serves to increase the make-up system capacity to 650 gpm for emergency conditions such as a process tube rupture or complete breakdown of a primary pump shaft seal. These pumps are equipped with wound rotor motors for speed control and throttling valves for system pressure variation. The water supply for these pumps comes from a 10,000 gallon high purity water storage tank. A 500,000 gallon standby water storage tank is available to supply the make-up system by means of automatic valving in the event that the demand on the normal supply tank becomes excessive due to malfunctioning of the primary coolant loop.

POWER GENERATION SYSTEM

A single turbine-generator set is used to generate a gross power of 235,000 kw from the 3,290,000 lb/hr of saturated steam formed in the four heat exchangers at rated power. Each heat exchanger has 2750 square feet of feed water heating surface, and 15,750 square feet of evaporating surface. The steam loop water enters at 404°F and leaves as saturated steam at a pressure of 555 psia and a temperature of 478°F. The saturated steam is delivered to the turbine throttle, passed through the turbine, and then to the condenser. The turbine is a tandem-compound, double flow type, designed to operate with saturated steam at 530 psia. Five extraction stages are provided for the removal of moisture and steam for feed water heating. The condenser is a two-pass unit mounted under the turbine with a surface area of 185,000 square feet and a heat load of 1850 million BTU/hr. The condenser is operated at a turbine exhaust pressure of 1.5 inches of mercury absolute with a condenser cooling water temperature of 60°F. The 168,000 gpm condenser cooling water flow will be taken directly from the normal water source.

The generator is rated at 278,000 kva at a 0.9 power factor with 45 psia hydrogen pressure. A 14,400/230,000 volt three-phase transformer (water cooled) and associated switchyard for outgoing transmission lines are provided. The scope of this design concept does not extend beyond the transformer and switchyard equipment.

During normal reactor operation, two sources of primary pump power are available; these are the main plant generators and the electrical network to which the plant output is fed. During shutdown or while starting up power is provided by a 6000 kw generator driven by an oil-fired gas turbine.

Since the steam loop is not radioactive, the generating equipment can be altered to include an oil-fired superheater of conventional design. The overall heat rate can be improved in this manner and the unit cost of the power produced may be reduced, depending on the relative cost of the nuclear and the chemical fuels.

WATER TREATMENT AND SUPPLY SYSTEM

Water of several different degrees of purity is supplied for the various plant requirements. Approximately 168,000 gpm of water is required from the normal water source; the principal requirement is for condenser cooling. This is supplied by a pumping station equipped with four low-head pumps which will deliver 42,000 gpm each. Two 8000 gpm pumps provide water for emergency reactor and steam loop cooling, fire fighting, and miscellaneous requirements.

Approximately 400 gpm of water is diverted from the condenser cooling stream for filtration and treatment in a conventional manner common to the area of the site location. After filtering, 200 gpm is used for non-process sanitary water requirements. Process requirements for filtered water include about 50 gpm interruptible supply for the fuel element storage

basin and a similar amount for the preparation of chemical solutions. About 100 gpm of filtered water is processed through the first stage of demineralization to obtain water which has no more than one to three parts per million total solids content and is relatively free of silicates. A closed storage tank of 500,000 gallons capacity is provided for water of this quality. The major portion of this water is used to make up normal losses in the steam generation system where high water purity is necessary to insure the high heat transfer rates called for in the primary loop heat exchanger design. Filtered water is also used as make-up supply for the shield and horizontal control rod recirculating cooling systems. Under normal conditions about 10 gpm is supplied to the final mixed bed demineralization system to be used for primary loop make-up, including pump seal supply.

A mechanical separator system is utilized to simplify the problem of continuous clean-up of the primary loop system. A 1500 gpm flow is diverted from the approximately 43,000 gpm discharge of the primary loop pumps and is fed to a system of centrifugal separators capable of diverting particles in the 2 to 20 micron range by way of a 150 gpm underflow. The cleaned overflow from this system is returned to the suction side of the primary pumps. The underflow is cooled to less than 200°F by passing through a regenerative heat exchanger. At this point the water stream may be diverted to the waste disposal system if the radioactivity is too great, or may be sent on to a 500,000 gallon standby storage tank for radioactivity decay. In addition to its decay time function, this standby storage serves as an additional emergency water source for the primary coolant loop and also provides a source of water for single pass cooling during charging and discharging operations. About 150 gpm is normally bled from the standby storage tank to another system of centrifugal separators. The 5 gpm contaminated underflow from this system is sent to the waste disposal system while the cleaned overflow is combined with the primary pump seal leakage water, and passed through a 200 gpm mixed bed demineralizer. The demineralizer effluent is pumped to the primary loop by way of the 10,000 gallon high purity water storage tank which supplies the loop make-up system.

To discharge irradiated fuel, the reactor is shut down, the loop pressure reduced, and the outlet ends of the process tubes opened. Under these conditions the low, shutdown flow is routed to a 100,000 gallon storage tank from which it is passed through a heater-deaerator, and returned to the suction side of the primary pumps. The piping arrangement will also allow direct routing of the entire shutdown flow to the waste disposal system for flushing the loop after a fuel element rupture.

All radioactive waste is routed to a 500,000 gallon retention tank. This includes primary loop clean-up underflow, control rod and shield loop bleeds, fuel element storage basin overflow, and miscellaneous contaminated sewer wastes. After sufficient hold-up for decay of short-lived activity, controlled release is made through a 50 gpm ion exchange system to ground or to the normal water source for dilution to a safe level of activity. The ion exchange resin will be removed and buried when exhausted.

PLANT OPERATION

The safety of plant operation and maintenance is an important consideration in this design. Instrumentation, piping arrangements, pumps, and reactor control and safety systems have been designed to minimize the probability of the initiation or continuance of an unusual nuclear incident. However, it is recognized that some combination of equipment and human failure could lead to inadequate cooling of the reactor fuel and emission of radioactive material from the reactor or from the primary loop. For this reason, the reactor, all components of the primary loop system, portions of the primary loop water purification equipment, the reactor gas facility, and the miscellaneous equipment and services housed in the reactor building are enclosed within a 225 foot diameter spherical vapor container. A cross-sectional view of the reactor building and its components, enclosed in the sphere, is shown in Fig. 3. The sphere is constructed of carbon steel plate in accordance with accepted engineering standards for unfired pressure vessels. The sphere has been designed to withstand an internal pressure of 45 psia. The bottom of the sphere is 40 feet below the ground level and the reactor operating floor level is 20 feet above the ground level. An accident involving rupture of the main cooling loop would result in a 4 psi rise in the internal pressure of the sphere from the cooling water flashing to steam. Available reactor control is adequate to stop the nuclear reaction, but there may be additional energy release from the chemical reaction between molten metals and water if the continuity of cooling is completely disrupted. The interior of the sphere is equipped with a 2000 gpm cooling spray system which will absorb the heat released by radioactive decay and chemical reactions. The maximum sphere pressure that would occur under these conditions is 40 psia.

The control rod system in this design will permit power changes consistent with typical electrical demands on large base-load power stations, as can be seen from the following table.

Heat Load Variations

Decreasing power level	*Elapsed time min*	*Increasing power level*	*Elapsed time min*
800 M/w to 600 M/w	8	400 M/w to 600 M/w	4
800 M/w to 400 M/w	16	400 M/w to 800 M/w	8
800 M/w to 200 M/w	24		

By establishing anticipatory scheduling of output requirements, manipulation of the steam cycle will

not be necessary. However, for sudden loss of load the reactor is shut down immediately and excess steam is bypassed directly to the condenser.

Periodic infrequent reactor shutdown will be required for reactor charging and routine maintenance requirements. The exact schedule will depend on the goal exposure for the fuel elements which in turn will depend on the market for the products formed in the irradiated fuel and the attainable exposure of the particular fuel element design. In the event of a fuel element rupture, it is planned to continue to operate until the next scheduled shutdown. Approximately one-sixth of the total reactor fuel load can be readily replaced at each shutdown which results in the following shutdown schedule:

Exposure limit (Megawatt-days per ton of uranium)	*Time between discharges (days)*
2000	18
5000	45
10,000	90

The plant is designed for 10,000 Mwd/t operation, but it is intended that advantage be taken of any possible economic benefit from variations in goal exposure with proper consideration being given to continuity of operation.

Shutdown reactivity transients in this reactor are such that the minimum outage time will be on the order of 38 hours. This is adequate time for the complete charging operation as well as routine maintenance, but still may be fitted into low electrical load periods.

CAPITAL AND OPERATING COSTS

The forecasts which have been made of the costs and returns for a power plant of this type will be discussed only briefly, since these factors vary greatly with project sponsorship, type of contract rendered, method of financing, construction site, labor rates, productivity factors, and general economic conditions prevalent during the period of erection.

Based on actual construction experience at the Hanford production reactor site, it is estimated that the complete power plant erected at this site would require a capital expenditure of $65,000,000. In addition to direct construction costs and contingencies, this estimate includes typical indirect charges such

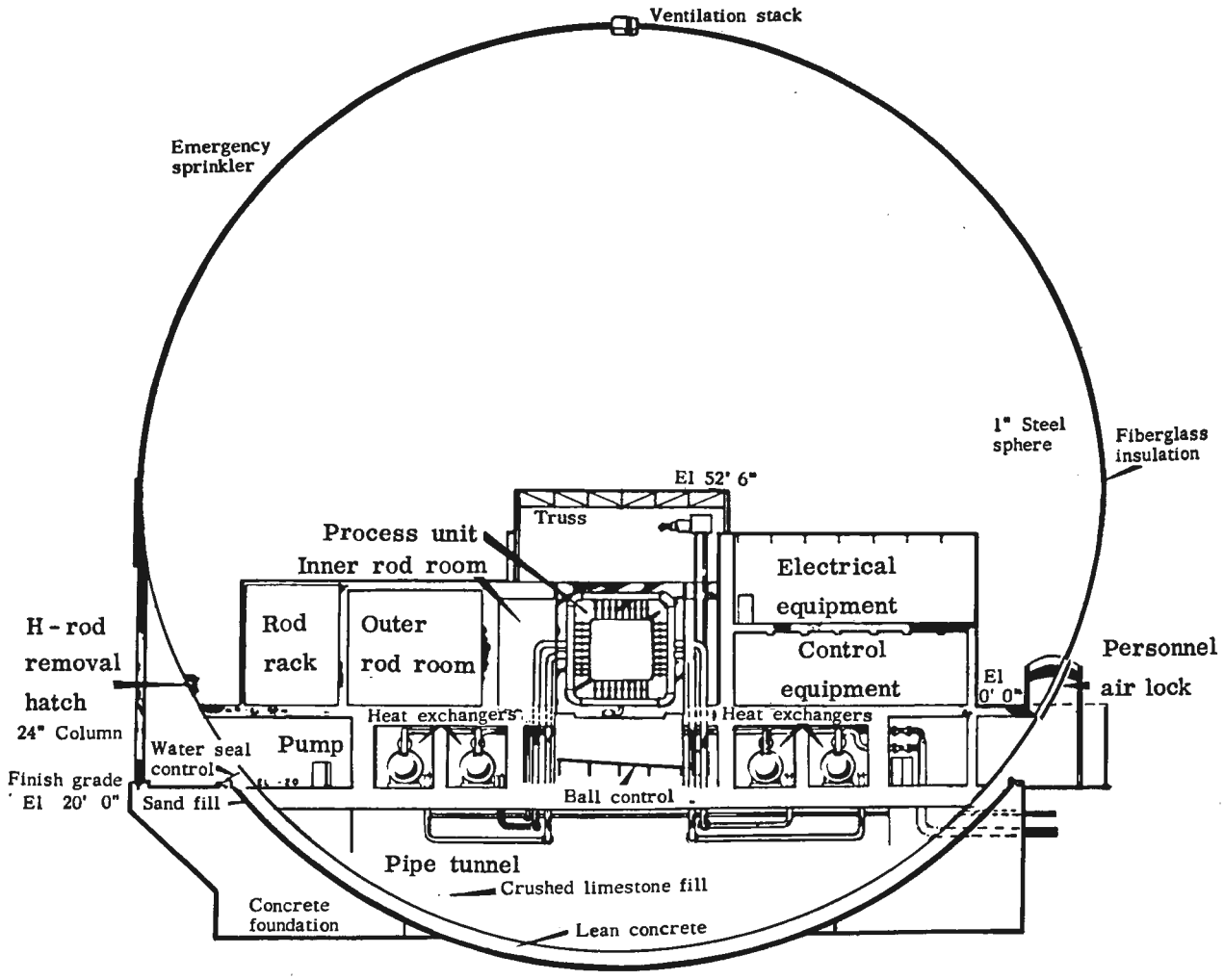

Figure 3

as engineering costs, interest on capital funds, and general overhead. However, the estimate does not include any allowance for the overall guarantee risk to be borne by the general design and construction contractor.

The operating costs of the plant will depend on the level of performance attained with the fuel element designs to be developed for these operating conditions. The existence of relatively efficient facilities for fuel element fabrication and for reprocessing spent fuel, including recovery of fuel grade plutonium, has been assumed. For predicted levels of fuel element performance, cost, and recovery value, the electric power can be marketed at a financial profit for less than seven mills per kilowatt-hour. These electric power costs are competitive with conventionally fired steam plants in many parts of the world.

Sodium Graphite Reactor 75,000 Electrical Kilowatt Power Plant

By Chauncey Starr,* USA

One of the promising approaches to economical nuclear power is the liquid metal cooled, high temperature, non-pressurized, reactor type. A present embodiment of this concept is the Sodium Graphite Reactor (SGR) system. The choice of this approach to nuclear power as part of the United States Atomic Energy Commission's program is based on both its immediate technical feasibility and confidence that advances in technology will result in major improvement in performance. The important features of this system are the following:

1. A high coolant temperature—925°F (~ 500°C) at present, 1200°F (650°C) projected—is possible without requiring pressurization of the reactor heat extraction system. This permits good steam conditions and high thermal efficiency in the power conversion equipment.

2. The reactor can be adapted to a variety of fuel elements and operating conditions. Both slightly enriched uranium or a Th-U^{235} fuel may be used.

3. There are no chemical incompatibilities in the fuel element, coolant, structure combination.

4. The absence of releasable potential energy from pressurization or chemical reactions increases the inherent safety of this reactor system and permits the plant configuration to be easily arranged to contain radioactivity under all circumstances.

The use of a liquid metal such as sodium as a heat transfer fluid has many advantages. While solid at room temperature, it melts at 208°F (97.8°C) and is therefore easy to keep liquid in a power plant system. Sodium boils at 1620°F (882°C) and thus permits adequately high coolant temperatures with the system at atmospheric pressure. Because liquid metals, sodium in particular, have extremely high thermal conductivity, they are excellent heat transfer media. The fluid dynamics of liquid sodium is similar to that of water. A further advantage of sodium is the general absence of chemical corrosion in metal systems. As a result, a wide variety of steels and other alloys may be utilized as construction materials. Because of these characteristics of sodium, the engineering development of the Sodium Graphite Reactor and its heat extraction system may eventually permit (1) the production of steam temperatures which will be limited only by turbine performance, and (2) the simplification of the reactor cooling system to an extent approaching low pressure water systems.

Sodium, however, is not without certain disadvantages. It reacts chemically with oxygen and water and must therefore be isolated from these. Further, when exposed to neutrons as in a reactor, it becomes a strong radioactive gamma ray emitter with a half-life of about 15 hours, and this creates a serious problem in nuclear power plant applications. For these reasons (in the present conceptions) the steam boiler has been separated from the primary sodium, which extracts the reactor heat, by a secondary sodium system which does not become radioactive—and carries the heat from the primary system to the boiler. The vault necessary to contain the radioactive primary sodium provides the shielding required and also acts as a container for the inert gas atmosphere which surrounds the primary system.

The choice of graphite as a moderator results from the fact that graphite is mechanically easy to handle, as compared to any liquid hydrogenous moderator, and is considerably less costly than beryllium, which might also be used. The principal disadvantage of graphite is associated with its porosity, which causes it to act as a sponge when in contact with liquid sodium. For this reason the graphite is contained in zirconium metal cans in the present design of this type of plant.

In mid-1954, North American Aviation, Inc. entered into a program with the U.S. Atomic Energy Commission aimed at expanding the known area of information on the Sodium Graphite Reactor. The sodium cooled graphite moderated reactor approach will be experimentally demonstrated through the construction and operation of a Sodium Reactor Experiment (SRE). This is a 20,000 thermal kilowatt pilot reactor, incorporating the major design features of the full scale Sodium Graphite Reactor. An experimental engineering program with the aim of attaining the potential performance of this high temperature, low pressure reactor system is an important part of the joint project. This engineering development program is now under way. The Sodium Reactor Experi-

* Vice president, Nuclear Engineering & Manufacturing, North American Aviation, Inc., Los Angeles, California. Including work by E. F. Weisner, W. M. Sybert, R. Balent, A. B. Martin, W. E. Abbott, J. O. Henrie, A. M. Stelle, T. T. Shimazaki, M. P. Heisler, all of North American Aviation, Inc.

ment will be fully operative in early 1956. Experience in construction, test, and initial operation will be available for incorporation into the initial Sodium Graphite Reactor power plant, which is presently conceived to be one capable of producing 75,000 kilowatts of net electrical power.

GENERAL DESCRIPTION

The proposed 75,000 kilowatt central station plant consists of a reactor core cooled by a primary sodium loop, intermediate heat exchangers, secondary sodium coolant loop, steam generators, turbo-generators, condensers, transformers, switchgear, and auxiliary equipment. A schematic flow diagram of the plant is presented in Fig. 1. Figure 2 shows the arrangement of the principal components of the plant. Figures 3 and 4 illustrate the arrangement of the equipment inside the reactor building.

There are two reactor core designs for this plant. One is for use with uranium fuel and the second for use with thorium-base fuel. The two reactors are identical in all design aspects. They differ only in type of fuel element employed and certain physical dimensions. The best choice of reactor design for a particular application depends on such factors as the relative costs and availability of uranium and thorium fuels, fabrication and processing costs in each case, and similar considerations.

The reactor core is contained in a large diameter steel tank located below ground and cooled by liquid sodium. The sodium coolant is separated into four independent loops and flows from the reactor by means of radial discharge pipes to four intermediate sodium-to-sodium heat exchangers. The loops are contained in individually shielded vaults located in four quadrants around the reactor tank and below ground level. The four individually shielded quadrants permit maintenance on any one of the four loops without complete shutdown of the plant.

The secondary, non-radioactive, sodium from each intermediate heat exchanger is pumped above ground to the steam generators. The steam generators are separated into two buildings located on either side of the main reactor building and are each fed by the secondary sodium from the two intermediate heat exchangers nearest that side of the building. The location of the steam generators is sufficiently removed from the main building and the electrical generating equipment to eliminate damage to these units in case of a tube failure which might result in a sodium-water reaction. The steam from each exchanger unit is piped to the turbo-generator located to the rear of the main building. The location of the electrical transformers, switchgear enclosure, and cooling tower area are only shown schematically; the exact location would be dictated by specific plant location and needs.

The present temperature limitations in these reactor cores are a result of the limited experience with zirconium and steels in sodium, and also with fuel elements, at high temperature. The physical properties of zirconium are known to be affected by trace impurities such as oxygen, nitrogen, and hydrogen. The extent to which such impurities can be controlled in large sodium systems has not as yet been determined. Further, the mechanical behavior of zirconium under practical high temperature conditions for prolonged times is uncertain. For this reason, the maximum sodium-zirconium temperature has been held to a conservative 950°F (510°C) in the present design.

The use of uranium metal fuel elements at high temperature is complicated by the alpha-beta phase transition which occurs at about 1220°F (660°C) and its accompanying volume change. While it may be possible to operate with a mixed phase situation at high temperatures, insufficient experience exists. For this reason, the maximum uranium metal temperature at the center of the fuel elements has been limited to 1200°F (649°C) in this design.

In this regard, thorium offers interesting possibilities, since a maximum metal temperature of as much as 2000°F (1093°C) may possibly be utilized due to the absence of phase transitions. Here again more experience under practical operating conditions is needed.

URANIUM FUELED REACTOR

The reactor core consists of an array of closely packed, canned, hexagonal graphite cells contained within a large steel tank (see Figs. 5 and 6). The tank is surrounded by a thermal shield and a layer of thermal insulation. Since the reactor is located entirely below ground level, the concrete foundation provides adequate biological shielding. The top shield is composed of several feet of heavy concrete and could be removed should major repairs to the core be required.

The reactor has been designed to produce a thermal power of 250 megawatts with the maximum zirconium temperature limited to 950°F (510°C), and the maximum uranium temperature limited to 1200°F (649°C). The sodium inlet temperature to the reactor has been selected as 500°F (260°C), so that the desired steam pressure may be produced. The number of fuel channels and velocity of coolant through the core were chosen so that a mean bulk outlet temperature of at least 925°F (496°C) would result while satisfying the above requirements. Table I presents the reactor design and performance data.

The lattice spacing was selected as 10 inches (25.4 cm) and the required enrichment as 1.8 per cent on the basis of a preliminary analysis. Although these quantities cannot be completely optimized until a specific application and cost basis are defined, these values are sufficiently close to an optimum to permit reasonably accurate estimates to be made of construction and operating costs.

The fuel elements consist of clusters of 19 rods. Each rod is made up of ten slightly enriched uranium fuel slugs 0.455 inch (1.156 cm) in diameter by 12 inches (30.48 cm) long. The rods are contained in a

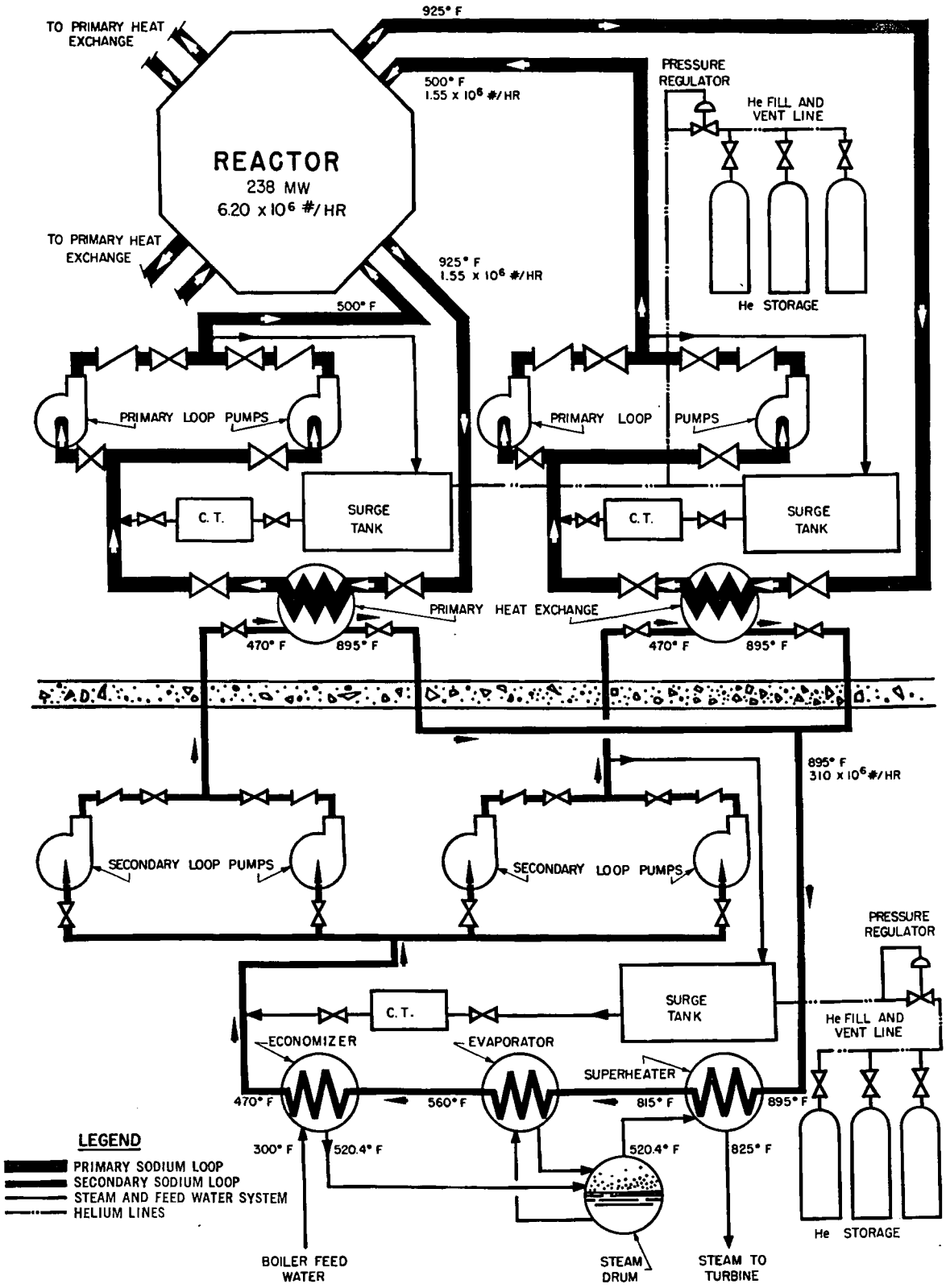

Figure 1. Schematic flow diagram

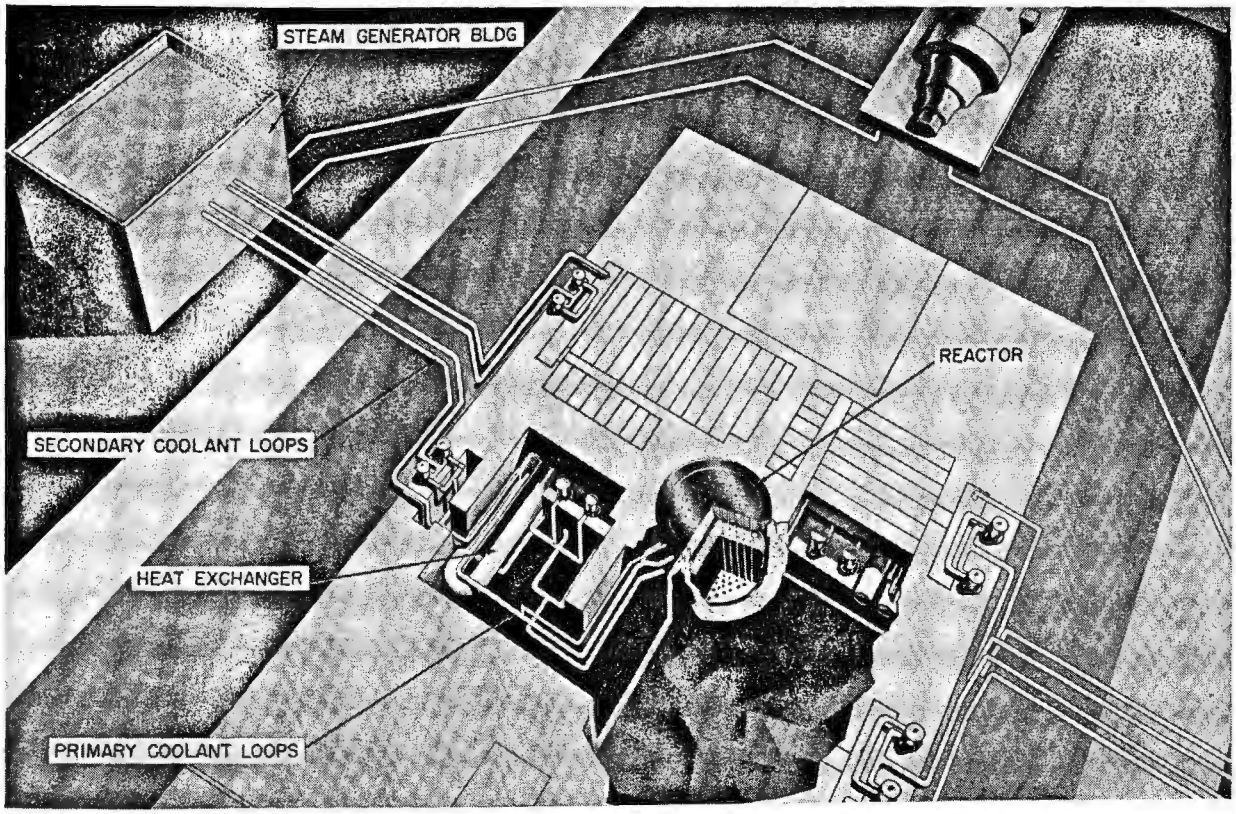

Figure 2. Aerial view of 75 megawatt reactor plant

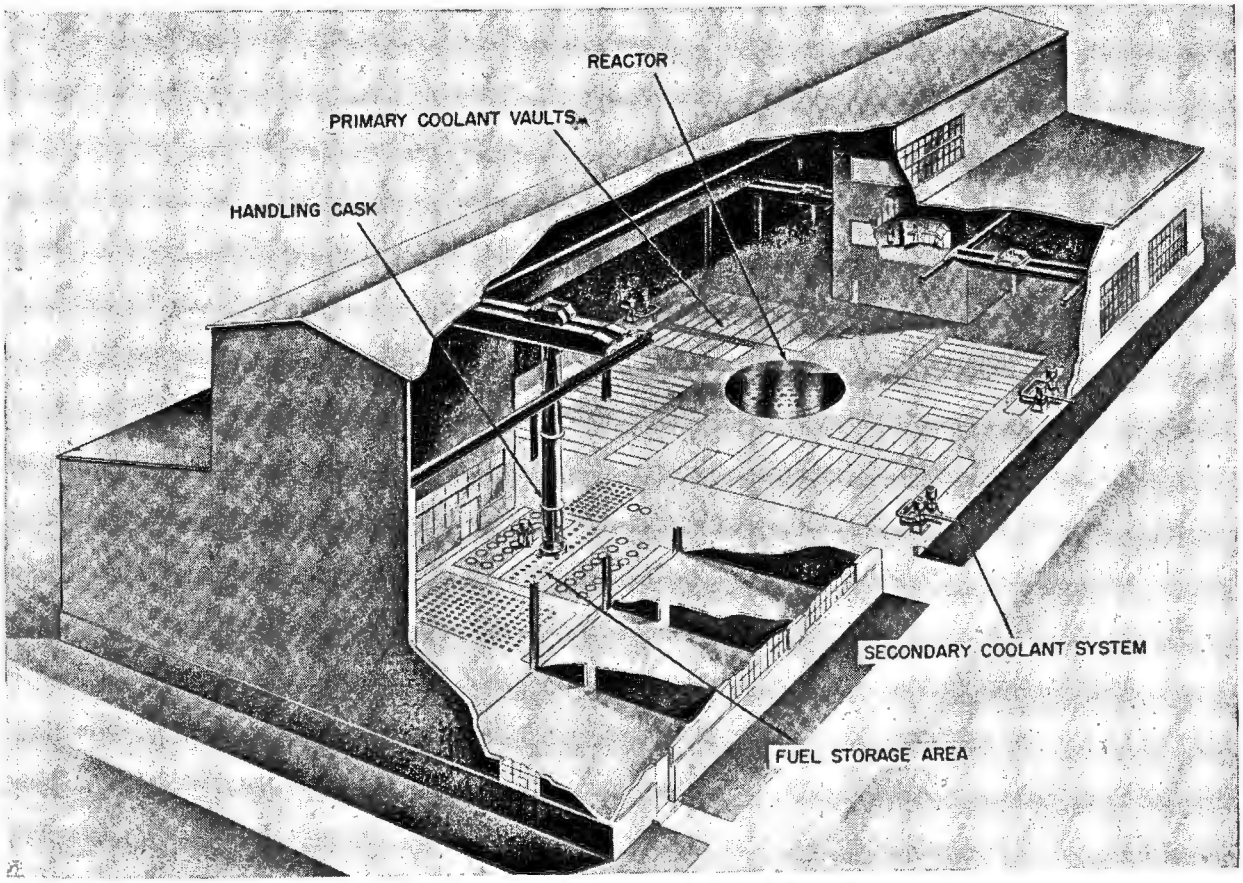

Figure 3. Aerial view of 75 megawatt reactor floor plan

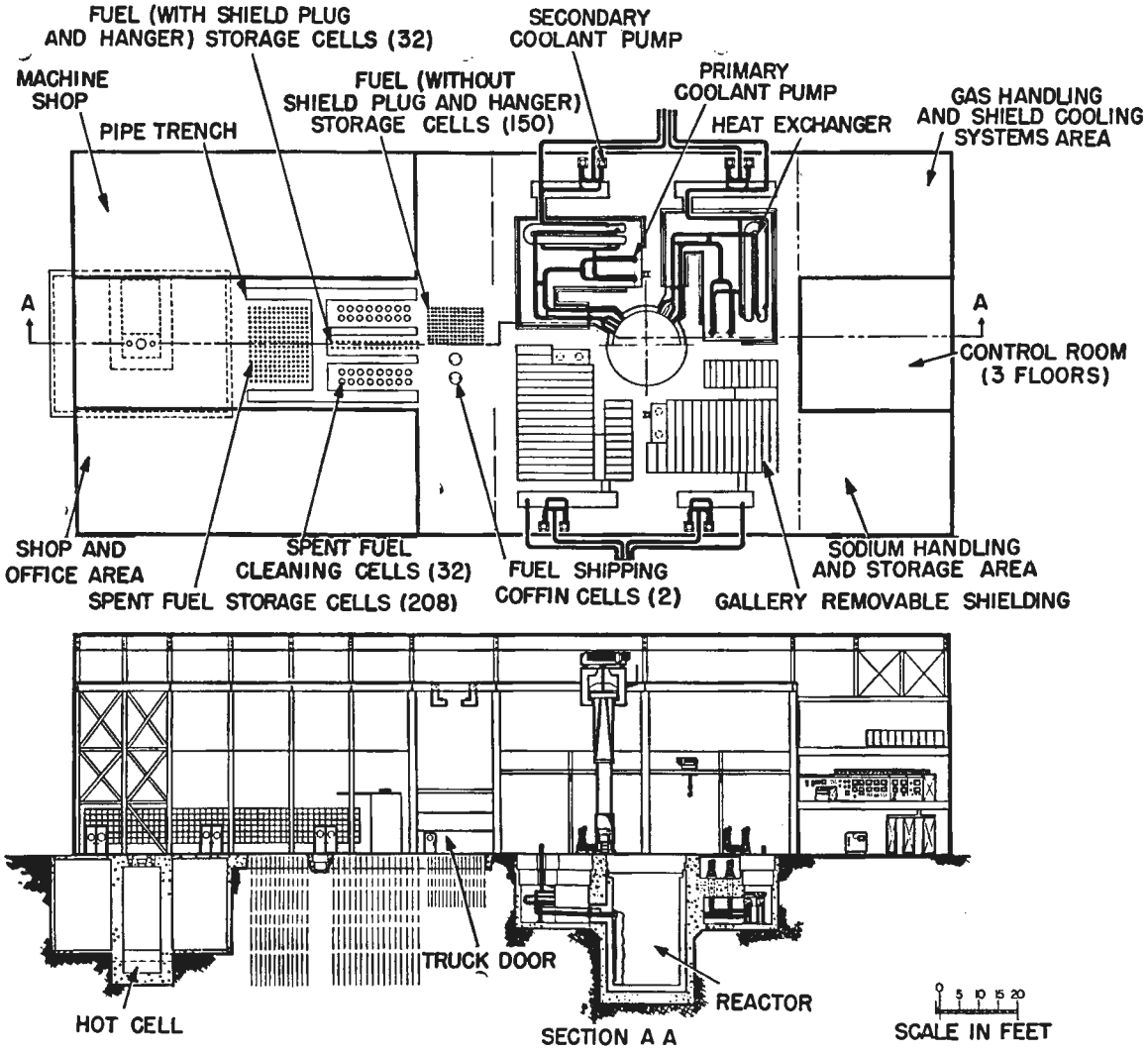

Figure 4. Cross section and plan of the reactor system

stainless steel jacket 0.495 inch (1.257 cm) outside diameter (OD) by 0.010 inch (0.0254 cm) wall (see Fig. 7). The jacket is sealed at the top and bottom with a plug and, except for a helium filled expansion space, is filled with a bonding material (either sodium or NaK). The average thickness of the bond is 0.010 inch (0.0254 cm). The rods are spaced from each other by helically wrapped steel wires.

The rods are joined to a common hanger rod by which they are suspended in the coolant channels located in the moderator elements. The hanger rod is supported from the top shield by a shielding plug. On each assembly there is an orifice plate to throttle the flow of coolant to each fuel cluster. The size of the orifice plate is determined by the location of the element in the reactor so that the bulk coolant outlet temperature may be maximized.

The moderator elements are graphite columns contained within sealed zirconium cans having 0.035 inch (0.0889 cm) wall thickness. There are small gaps between the zirconium and graphite to allow for thermal expansion, and clearance for assembly. A nominal clearance of 0.100 inch (0.254 cm) is provided between adjacent can walls to permit sodium coolant to flow between these elements.

A typical canned moderator element is shown in Fig. 8. The canned reflector elements are similar except that they do not contain fuel channels or coolant tubes.

There are 233 moderator elements forming an active core 13.3 feet (4.054 m) in diameter by 10 feet (3.048 m) high. A 2-foot (61.0 cm) reflector surrounds the core both in the radial and axial directions resulting in overall dimensions of 17.3 feet (5.273 m) in diameter by 14 feet (4.267 m) high.

The core tank is 17.5 feet (5.334 m) inside diameter (ID) with 2-inch (5.08 cm) walls made from stainless steel plate. Near the bottom of the tank is a core support plate which locates and supports the moderator and reflector elements. Beneath this plate is a second plate or tube sheet, forming two plenum chambers, one between the two plates and one between the lower plate and the bottom of the core tank. The coolant channels extend through both plates so that the main sodium flow is from the lower plenum chamber up through the coolant tubes. The

Table I. Performance and Design Data, Uranium Fuel

A. Reactor	
1. Reactor Type	Thermal heterogeneous
2. Reactor Operating Data	
Nominal rated output, Mw (thermal)	250
Average specific power, fuel, Mw/ton of metal (1 ton = 2000 lb)	9.2
kw/kg of U^{235}	565
Average specific power, moderator, Mw/ton graphite	3.5
Average thermal flux in fuel, neutrons/sec-cm^2	2.5×10^{13}
Peak thermal flux in fuel, neutrons/sec-cm^2	5.4×10^{13}
Initial at. % U^{235} in the fuel for 2% k excess	1.80
3. Materials	
Fuel	Uranium metal, slightly enriched in U^{235}
Moderator and reflector	Graphite, each cell canned in 0.035 in. (0.0889 cm) zirconium
Fuel element cladding	0.010 in. (0.0254 cm) type 304 stainless steel
Coolant tube	3.250 in. (8.26 cm) OD × 0.035 in. (0.0889 cm) wall zirconium
Primary coolant	Sodium
Secondary coolant	Sodium
Thermal shield	Mild steel
Biological shield	Barytes concrete
4. Core Nuclear Data	
Lattice type	Triangular
Lattice spacing, in.	10.0 (25.4 cm)
Number of cells, total	233
for fuel	215
for control and safety rods	18
Nominal reflector thickness, ft	2 (61. 0 cm)
Core diameters (including reflector), ft	17.3 (5.273 m)
Core height (including reflector), ft	14.0 (4.267 m)
Core composition, per cent by volume	
Graphite	87.05
Sodium	6.72
Stainless steel	0.43
Uranium	3.57
Zirconium	1.81
NaK	0.32
Void (gas)	0.10
U^{235} content (initial loading), kg	443
Lattice constants (initial loading with equilibrium poisons) 2% k excess at 600°C. Uranium enrichment 1.8%	
ϵ	1.026
p	0.7564
f (for uranium)	0.8551
η (for uranium)	1.6878
$k\infty$	1.120
L^2, cm^2	134
τ, cm^2	405
B^2, cm^{-2}	205×10^{-6}
5. Fuel Element Data	
Type of fuel element	19-rod cluster
Number of elements in reactor	215
Fuel slug material	Uranium
Diameter of uranium slugs, in.	0.455 (1.156 cm)
Length of fuel slug, in.	12.0 (30.48 cm)
Initial fraction U^{235} in fuel, atom %	1.80
Weight of uranium in reactor, kg	24,600
Fuel element length, ft	10.0 (3.048 m)
Jacket wall thickness, in. of SS	0.010 (0.0254 cm)
Bond, in. of NaK	0.010 (0.0254 cm)
6. Control and Safety Rod Data	
Number of control rods	10
Type of rods	Boron steel rod operating inside of steel thimble
Diameter of rods, in.	1.92 (4.88 cm)
Total Δk in control rods, %	~10
Number of safety rods	8
Total Δk in safety rods, %	~9
B. Cooling System Data	
1. Reactor	
Coolant	Sodium
Temperature in, °F	500 (260°C)
Mixed mean outlet temperature, °F	~925 (496°C)
Highest temperature out, °F	950 (510°C)
Maximum coolant velocity, ft/sec	9 (274 cm/sec)
Pressure drop across core, psi	6 (0.43 kg/cm^2)
Flow rate, lb/hr	6.5×10^6 (3.0×10^6 kg/hr)
Maximum heat flux, BTU/hr-ft^2	350,000 (950,000 kg-cal/hr-m^2)
Maximum fuel temperature, °F	1200 (649°C)
Total power, Mw	250
2. Primary Loops	
Number of parallel circuits	4
Inlet temperature to primary heat exchanger, °F	925 (496°C)
Outlet temperature from primary heat exchanger, °F	500 (260°C)
Flow rate in each circuit, lb/hr	1.625×10^6 (0.74×10^6 kg/hr)
Pumps in each circuit	2
3. Secondary Loops	
Number of parallel circuits	2
Inlet temperature to superheater, °F	895 (479°C)
Outlet temperature from economizer, °F	470 (243°C)
Flow rate in each circuit, lb/hr	3.25×10^6 (1.48×10^6 kg/hr)
Pumps in each circuit	4

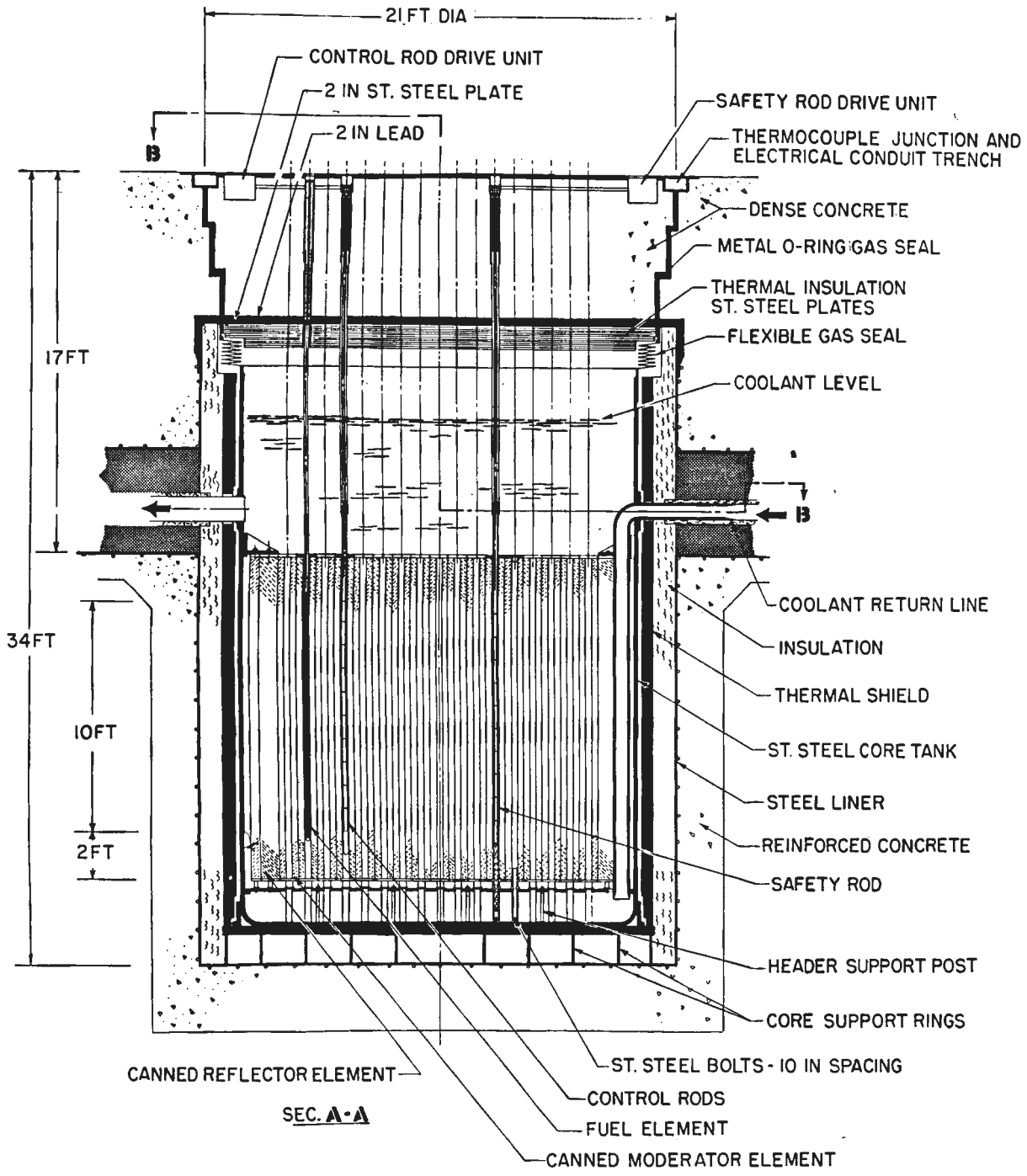

Figure 5. Reactor elevation

flow between the moderator and reflector elements is from the upper plenum chamber. Each chamber is force-fed by separate downcomers located just inside the core tank wall with the division of flow being adjustable.

The primary sodium coolant removes heat from the fuel elements, the moderator and reflector elements, and the thermal shield. This useful heat represents more than 98 per cent of the total energy produced in the reactor. The heat generated in the neutron shield and losses through the thermal insulation are removed by a separate low temperature cooling system. The low temperature coolant can be an organic liquid such as toluene, which is chemically nonreactive with sodium.

The sodium cooling system is divided into primary and secondary sections so that the radioactive primary coolant is contained in a small volume and is separated physically from the steam generator units. There are four parallel primary loops (see Fig. 1). The coolant in each loop leaves the core tank through pipes welded to the tank at a level just above the top of the moderator and reflector units. The coolant then flows to the inlet end of a vertically-mounted centrif-

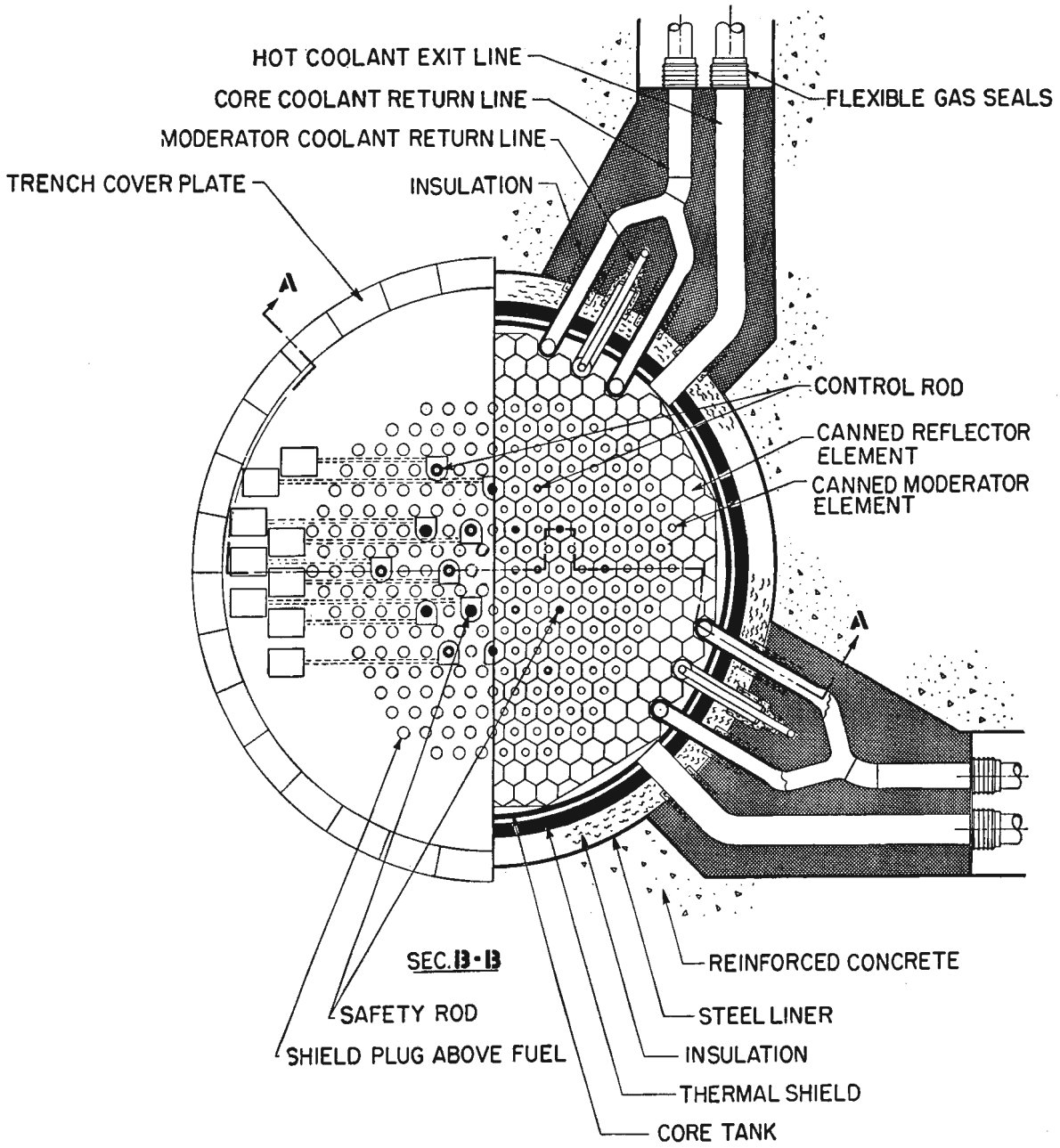

Figure 6. Reactor plan

ugal pump. Two pumps in parallel are provided in each loop. Frozen seal sodium pumps now being developed are suitable for application here.

Because of the radioactivity induced in the sodium, that portion of the primary sodium system not inside the reactor must be surrounded by a gamma shield. The sodium-to-sodium heat exchangers, primary system piping, and pumps associated with each loop are located in a vault adjacent to the reactor and below ground level. There are four such vaults, one for each loop. Each vault is covered by 6-foot (1.829 m) deep removable concrete shielding.

The secondary non-radioactive sodium system receives its heat from the primary system with approximately a 30°F (16.5°C) loss in temperature. There are two secondary loops with four pumps in parallel in each loop. The secondary sodium gives up its heat to steam in the superheaters, evaporators and economizers.

A total of 18 control and safety elements are provided. Each operates within a stainless steel thimble, which extends through the core in regular coolant channels.

Each control rod is made up of a stainless steel tube surrounded by boron rings. The rod size is determined by the amount of heat generated within the rod, the maximum temperature limits, and heat transfer consideration. The rod is cooled by conduction and radiation across a 10-mil (0.0254 cm) helium annulus between the boron steel rings and the

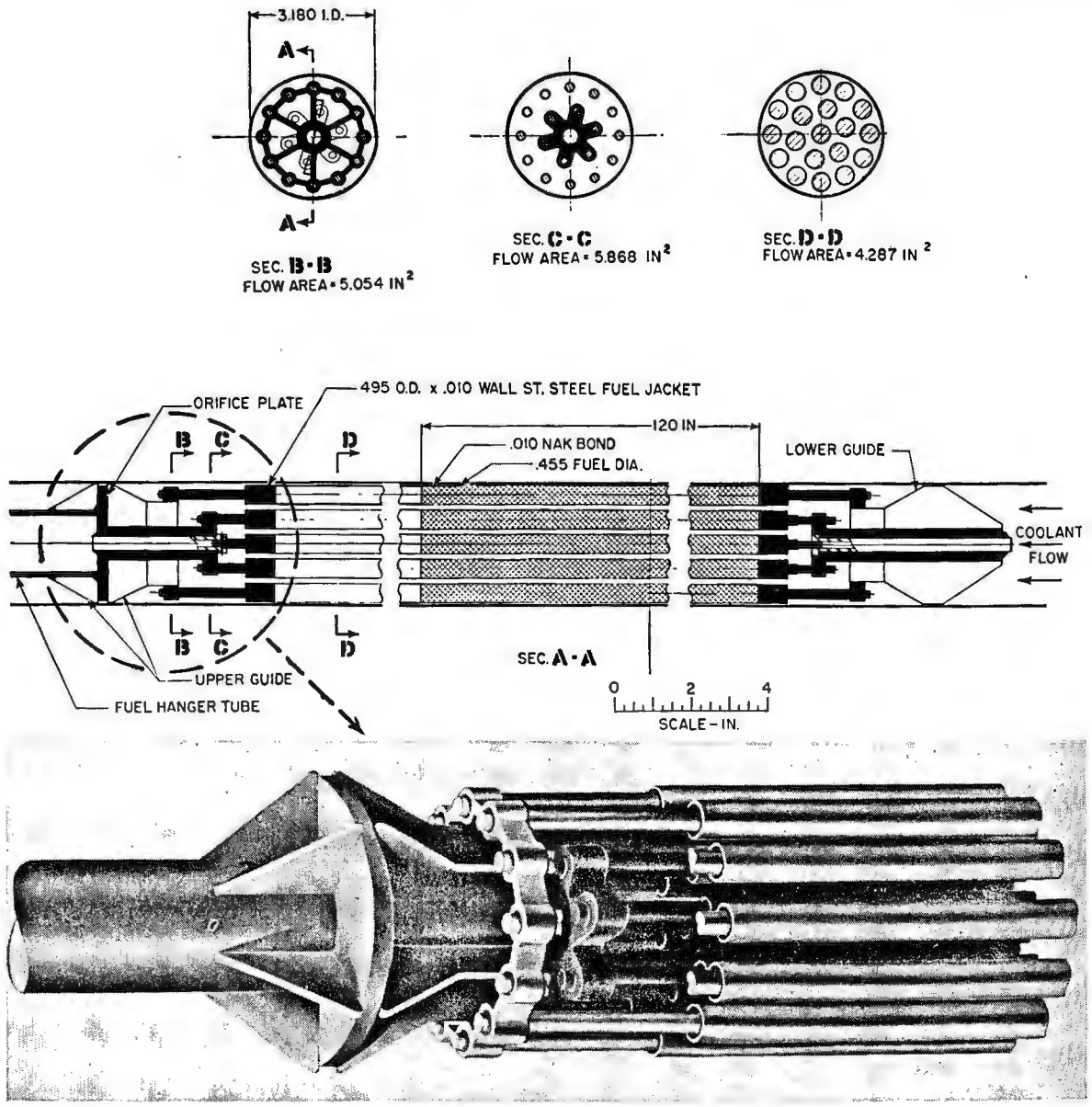

Figure 7. Nineteen-rod fuel cluster

thimble. The thimble wall is cooled by the primary sodium flowing between the process channel and thimble. The purpose of the thimble is to exclude sodium and sodium vapors from the drive mechanisms. It would be desirable from a nuclear point of view to avoid the use of the steel thimbles, but the physical properties of zirconium at these temperatures are not sufficiently known to permit zirconium thimbles to be specified at this time.

The safety elements are as large in diameter as the channel dimensions will permit since they normally must be out of the reactor during operation and therefore do not present the heat transfer problem encountered in the control rod design. The safety elements may either be boron steel rods or boron-containing balls. The final selection will depend upon the results of development work now in progress for the Sodium Reactor Experiment.

The core tank itself actually serves as a partial thermal shield but is normally not designated as such. The thermal shield proper is made of mild steel interlocking slabs, approximately 6 inches ($\sim$ 15.24 cm) total thickness located adjacent to and just outside of the core tank. The upper thermal shield is supported from the top biological concrete shield. The top shield is pierced by vertical holes, aligned with the fuel channels, and normally filled by shield plugs which support the fuel element hanger rods.

Thermal insulation is placed between the thermal shield and the biological shield or foundation walls to reduce the heat losses. Most of the heat generated within the thermal shield is radiated or conducted back into the core tank and removed by the sodium coolant.

The temperature of the thermal shield will vary from around 500°F (260°C) below the reactor to

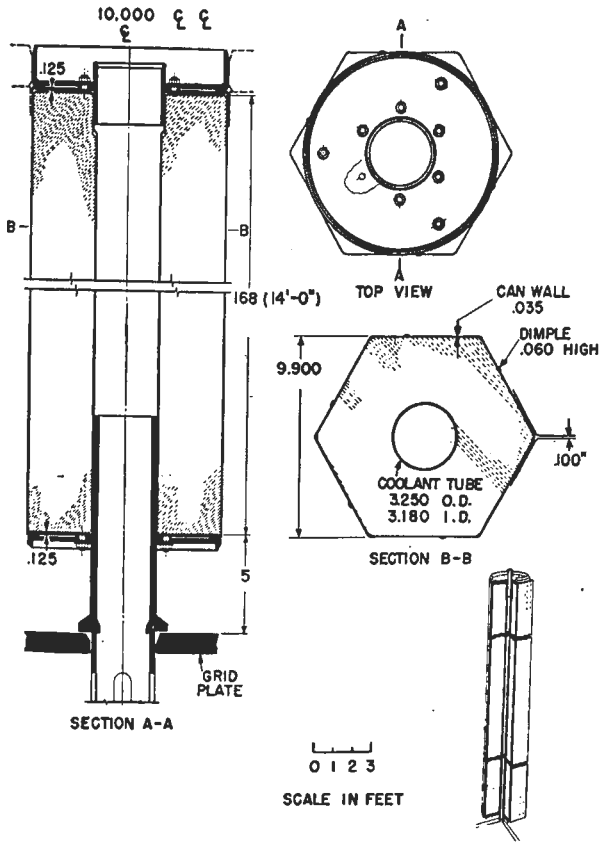

Figure 8. Moderator can

approximately 1200°F (649°C) in the central portion of the upper thermal shield.

The foundation concrete is lined with a mild steel plate. Cooling tubes welded to this plate contain an organic liquid which is circulated to remove the heat conducted across the insulation and generated in the foundation concrete and steel liner. A design temperature of 150°F (66°C) has been selected as a maximum for the concrete. Since the reactor is entirely below ground, the earth and foundation concrete provide adequate biological shielding. The foundation material is ordinary reinforced concrete. The thickness and reinforcing requirements will depend on the subgrade conditions at the site.

Instruments monitor the reactor neutron flux during start up and operation. Figure 9 is a block diagram of the proposed reactor control system. Three types of chambers are used to cover the wide range of neutron flux involved in going from shut down conditions to full operating power. These are fission chambers, gamma-compensated ion chambers, and uncompensated ion chambers. This selection has been made so as to provide continuous information over the range of approximately 10 decades which is attainable with a neutron source.

The fission chamber signals which are amplified and read on a counting rate meter can provide neutron flux indication over a range of four decades. The location of these four decades of range with respect to the power level may be altered by remotely positioning the fission chambers. The gamma-compensated ion chambers each cover a range of approximately six decades. ffhey are arranged so as to overlap each other and also overlap the coverage provided by the fission chambers. The signal from this type of chamber is passed through a logarithmic amplifier and differentiated to obtain an output proportional to reactor period. The amplifier output is also displayed on a recording instrument with a six decade strip chart. The output from the uncompensated ion chambers, of the parallel circular plate type, is used primarily for flux monitoring at power levels from about one decade below to one-half decade above full power level. By the use of scale changing resistors in the amplifier input circut, this range may be made to fall in any portion of about seven decades of range from normal full power level downward. This feature is particularly useful in the initial start up of the reactor.

Instrumentation is also provided in each of the sodium circuits to measure the flow rate, temperature, and pressure at several different points. The sodium level in the reactor and in the surge tanks is continuously measured. The outlet temperature of the sodium in each coolant channel is also continuously monitored.

The sodium service system consists of equipment required for melting, filtering, and transferring sodium to the primary and secondary coolant systems. An inert gas, either helium or nitrogen, is used wherever an inert atmosphere is necessary to prevent oxidation of either graphite or sodium.

The fuel handling system replaces the irradiated fuel elements with new fuel elements while the reactor is shut down. In addition, this system is designed to store, inspect, clean, and transport new and spent fuel elements (see Figs. 3 and 4). While the reactor is shut down, the fuel transfer cask is used to transfer new fuel elements to the reactor and to remove spent fuel elements from the reactor and then transfer them to a cleaning cell. After reactor operation is resumed, the cask is used to transfer the irradiated elements from the cleaning cells to the storage cells. The design permits making and breaking gas-tight connections between the cask and the reactor loading face, the new fuel storage cell, or the cleaning cells.

The neutron economy which is given in Table II for the start up conditions is approximately the same throughout the core life. The initial conversion ratio is 0.72. The number of loaded channels is 215. This loading provides an initial excess reactivity of 2 per cent, which should be sufficient to allow for at least 3000 Mwd/t irradiation.

THORIUM URANIUM-235 FUELED REACTOR

The reactor design discussed in this section differs in only one major aspect from the design discussed in the preceding section; namely, the use of Th-U^{235} fuel elements. The thermal rating of the reactor has

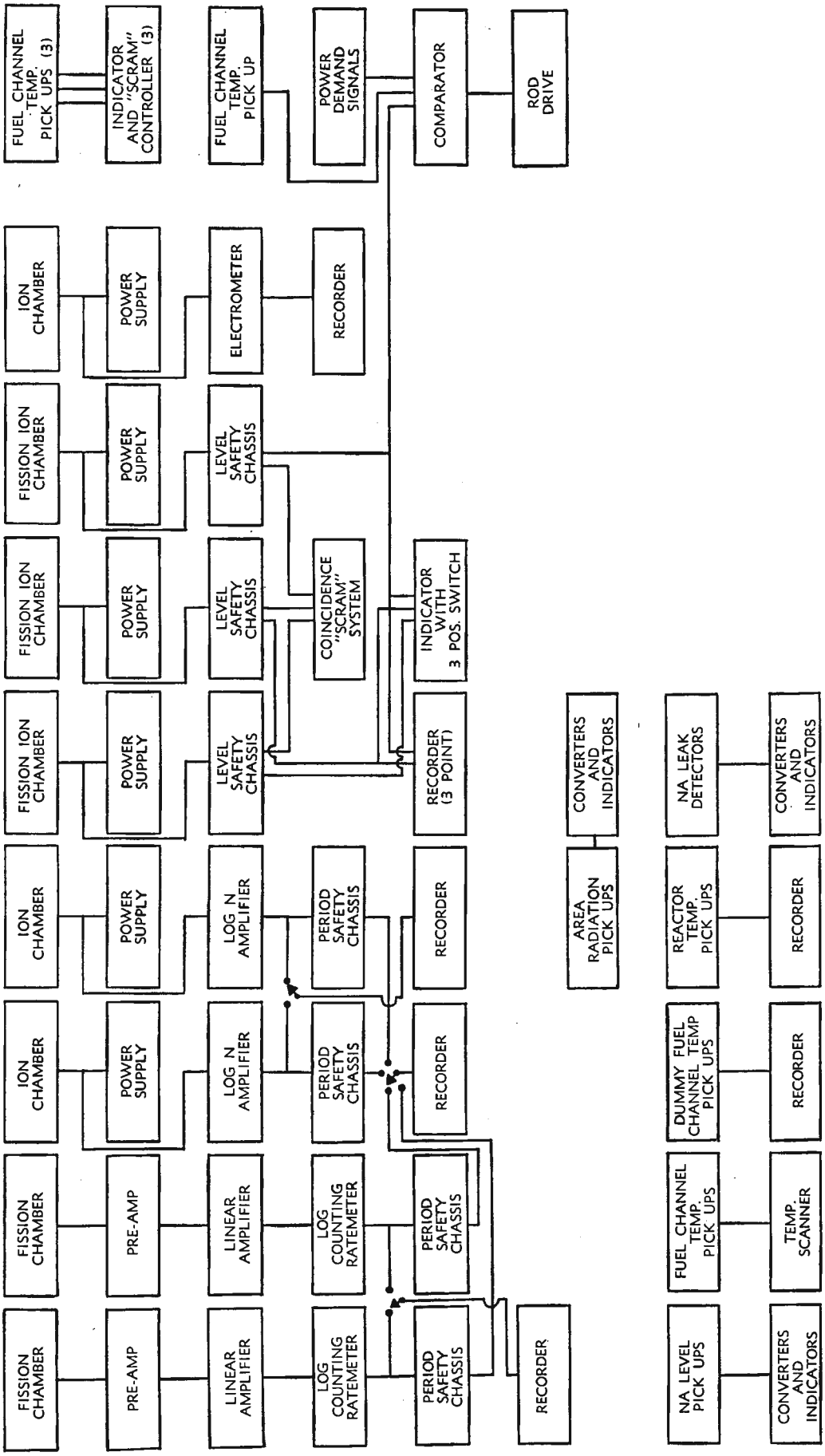

Figure 9. Instrumentation

Table II. Neutron Economy of Reactor Core, Uranium Fuel

VIRGIN FAST NEUTRONS		100.00
Fast leakage	5.946	
U^{238} resonance captures	22.912	
Thermal leakage	1.412	
Moderator captures	1.887	
Outer coolant tube captures	0.135	
Moderator can captures	0.878	
Moderator coolant captures	1.370	
Control rod captures	1.857	
Fuel element captures	63.603	
	100.000	
FUEL ELEMENT CAPTURES		63.603
U^{238} captures	10.866	
Xe and Sm captures	2.401	
Na coolant captures	1.346	
SS cladding captures	1.976	
NaK bond captures	0.132	
U^{235} fission captures	39.620	
U^{235} non-fission captures	7.262	
	63.603	

$$\text{Initial conversion ratio} = \frac{22.912 + 10.866}{46.882} = 0.720$$

$$\text{Second generation neutrons} = 39.620 \times 1.026 \times 2.46 = 100.000$$

been taken as 250 megawatts with the maximum zirconium temperature limited to 950°F (510°C). The sodium inlet temperature to the reactor is 500°F (260°C). The calculated central thorium temperature is 1600°F (871°C), but this is not considered to be limiting. The core size was determined mainly from nuclear considerations. While the required amount of power could be removed from a smaller core, the neutron leakage would be quite high.

The lattice spacing was selected as 8.0 inches (20.32 cm) and the required atomic fraction U^{235} in thorium as 3.70 per cent on the basis of a preliminary analysis. See Table III for reactor design and performance data.

The fuel elements consist of 7-rod clusters. Each rod is made up of ten slugs of Th-U^{235} alloy 0.648 inch (1.646 cm) in diameter by 12 inches (30.48 cm) long. The slugs are contained in a stainless steel jacket 0.688 inch (1.747 cm) OD by 0.010 inch (0.0254 cm) wall. The jacket is sealed at the top and bottom with a plug and, except for a helium filled expansion space, is filled with a bonding material (either sodium or NaK). The average thickness of the bond is 0.010 inch (0.0254 cm).

The moderator elements are graphite columns contained within sealed zirconium cans similar to those used in the uranium fueled reactor except for the overall dimensions. The can walls and coolant tube wall thickness are each 0.035 inch (0.089 cm). The coolant tube inside diameter is 2.493 inches (6.332 cm). There are 203 moderator elements forming an active core 10 feet in diameter by 10 feet (3.048 m) high. A 2-foot (61.0 cm) reflector surrounds the core both in the radial and axial directions resulting in overall dimensions of 14 feet (4.267 m) in diameter by 14 feet (4.267 m) high.

The core tank is identical to that previously described except that the inside diameter is 14.2 feet (4.328 m). The core cooling system, the control and safety elements, the thermal shield and insulation, the foundation, shielding and shield cooling, the instrumentation, the auxiliary systems, and the reactor building are the same as those described for the uranium fueled core.

The reactor has been rather conservatively designed. The maximum sodium-zirconium temperature has been limited to 950°F (510°C). The size of the reactor, the fuel element dimensions, and the lattice spacing were selected from nuclear and economic considerations. Since the maximum required reactor power is 250 megawatts, the center temperature of the thorium will not exceed 1600°F (871°C) with a maximum sodium velocity of 16.2 ft/sec (494 cm/sec) through the central channel. (See Table III.)

The nuclear characteristics change with burn-up since the U^{235} originally present is replaced by U^{233}, but not on a one-to-one basis. The fissionable material required for make-up is highly enriched U^{235}. The neutron economy of the core is given for the first two complete cycles in Table IV. The conversion ratio improves from an initial value of 0.675 to 0.782 at the end of the second cycle. The conversion ratio should ultimately level off at about 0.81. It is assumed that 40 per cent of the fissionable material is burned each cycle (this is approximately 10,000 Mwd/t). The irradiated material is removed from the reactor after 40 per cent burn-up of the fissionable material and sent to chemical processing. The fission products are removed, the Pa^{233} allowed to decay to U^{233}, and sufficient U^{235} and thorium are added in order to restore the initial excess reactivity. It is necessary to change fuel by zones so that the reactor will be critical at all times.

POWER PLANT CYCLES

The selection of an initial turbine cycle for a nuclear steam electric plant involves a compromise between two competing factors. These are (1) maximum heat generation from a given reactor, and (2) maximum conversion of this heat into electric energy, i.e., maximum thermal efficiency.

Heat removal from a reactor fuel element depends on the temperature drive from the center to the surface. Due to the excellent heat transfer characteristics of sodium, the temperature gradient between the fuel element surface and the sodium coolant is quite small. Therefore, heat removal from the reactor depends primarily upon the difference between the center temperature of each fuel element and the average temperature of the liquid sodium flowing up and through each fuel channel. The present state of metallurgical development sets an upper limit upon fuel element center temperature. Therefore, an increase in heat generation, at constant sodium velocity, can only be accomplished by lowering the average sodium temperature.

Table III. Performance and Design Data, Thorium-U^{235} Fuel

A. Reactor	
1. Reactor type	Thermal heterogeneous
2. Reactor operating data	
Nominal rated output, Mw (thermal)	250
Average specific power, fuel, Mw/ton of metal (1 ton = 2000 lb)	23.3
kw/kg of U^{235}	694
Average specific power, moderator, Mw/ton graphite	6.2
Average thermal flux in fuel, neutrons/sec-cm²	3.1×10^{13}
Peak thermal flux in fuel, neutrons/sec-cm²	6.7×10^{13}
Initial at. % U^{235} in the fuel for 2% k excess	3.70
3. Materials	
Fuel	Thorium-uranium alloy (highly enriched uranium)
Moderator and reflector	Graphite, each cell canned in 0.035 in. (0.0889 cm) zirconium
Fuel element cladding	0.010 in. (0.0254 cm) type 304 stainless steel
Coolant tube	2.563 in. (6.51 cm) OD × 0.035 in. (0.0889 cm) wall zirconium
Primary coolant	Sodium
Secondary coolant	Sodium
Thermal shield	Mild steel
Biological shield	Barytes concrete
4. Core nuclear data	
Lattice type	Triangular
Lattice spacing, in.	8.0 (20.32 cm)
Number of cells, total	203
for fuel	185
for control and safety rods	18
Nominal reflector thickness, ft	2 (61.0 cm)
Core diameters (including reflector), ft	14.0 (4.267 m)
Core height (including reflector), ft	14.0 (4.267 m)
Core composition, per cent by volume	
Graphite	86.24
Sodium	6.50
Stainless steel	0.43
Thorium-uranium	4.16
Zirconium	2.25
NaK	0.26
Void (gas)	0.10
U^{235} content (initial loading), kg	360.4
Lattice constants (initial loading with equilibrium poisons) 2% k excess at 600°C	
ϵ	1.000
p	0.7936
f (for uranium)	0.6892
η (for uranium)	2.0769
k_∞	1.136
L^2, cm²	95
τ, cm²	435
B^2, cm⁻²	240×10^{-6}
5. Fuel element data	
Type of fuel element	7-rod cluster
Number of elements in reactor	185
Fuel slug material	Thorium-uranium alloy
Diameter of thorium slugs, in.	0.648 (1.646 cm)
Length of fuel slug, in.	12.0 (30.48 cm)
Initial fraction U^{235} in fuel, atom %	3.70
Weight of thorium in reactor, kg	9740
Fuel element length, ft	10.0 (3.048 m)
Jacket wall thickness, in. of SS	0.010 (0.0254 cm)
Bond, in. of NaK	0.010 (0.0254 cm)
6. Control and safety rod data	
Number of control rods	10
Type of rods	Boron steel rod operating inside of steel thimble
Diameter of rods, in.	1.8 (4.57 cm)
Total Δk in control rods, %	~9
Number of safety rods	8
Total Δk in safety rods, %	~8
B. Cooling system data	
1. Reactor	
Coolant	Sodium
Temperature in, °F	500 (260°C)
Mixed mean outlet temperature, °F	925 (496°C)
Highest temperature out, °F	950 (510°C)
Maximum coolant velocity, ft/sec	16.2 (494 cm/sec)
Pressure drop across core, psi	16 (1.1 kg/cm²)
Flow rate, lb/hr	6.5×10^6 (3.0×10^6 kg/hr)
Maximum heat flux, BTU/hr-ft²	850,000 (2,300,000 kg-cal/hr-m²)
Maximum fuel temperature, °F	1600 (871°C)
Total power, Mw	250
2. Primary loops	
Number of parallel circuits	4
Inlet temperature to primary heat exchanger, °F	925 (496°C)
Outlet temperature from primary heat exchanger, °F	500 (260°C)
Flow rate in each circuit, lb/hr	1.625×10^6 (0.74×10^6 kg/hr)
Number of pumps in each circuit	2
3. Secondary loops	
Number of parallel circuits	2
Inlet temperature to superheater, °F	895 (479°C)
Outlet temperature from economizer, °F	470 (243°C)
Flow rate in each circuit, lb/hr	3.25×10^6 (1.48×10^6 kg/hr)
Number of pumps in each circuit	4

The conversion of thermal power to electrical power (i.e., thermal efficiency) decreases with a decrease in the average sodium temperature. For a given heat sink, the efficiency of any heat cycle is determined by the average temperature of heat addition to the system. Thus, in order to obtain a high thermal efficiency, a fairly high average sodium temperature is required.

As discussed in the previous sections, a reactor using metallic uranium fuel elements and zirconium-clad moderator presently has two temperature limitations. The first applies to the uranium metal which limits the center temperature of the fuel element. The second limitation pertains to the zirconium metal which limits the sodium outlet temperature.

Heat generation varies radially from a maximum for the center fuel elements to a minimum for the peripheral elements due to the neutron flux distribution. Equal sodium flow through each channel would result in a low mixed mean sodium outlet temperature due to the greater number of outer fuel elements. In this situation only the inner fuel elements are at the maximum permissible center temperature. Therefore, the sodium flow through each fuel channel is adjusted by throttling such that the outer channels have the least flow. This causes the sodium outlet temperature to be a maximum consistent with the aforementioned temperature limitation. As a result, each fuel element operates at a temperature limit, either in maximum center temperature or sodium-zirconium surface temperature.

The non-reheat regenerative steam cycle appears to be the optimum choice at this time. In conventional fuel fired stations fuel economy can be considerably increased by the use of five or more extraction feedwater heaters, which raise the temperature of the feedwater to about 400°F (204°C). In a nuclear plant, however, a high feedwater temperature has a detrimental effect. It either lowers the maximum possible steam pressure with a consequent loss in cycle efficiency, or it decreases reactor power output. The decrease in reactor power output does not result in any significant reduction in either capital expenditures or operating costs, whereas the higher feedwater temperature is only obtained by the installation of five or more heaters at a cost of about $50,000 per heater. Previous studies have shown that the best regenerative feedwater temperature for nuclear plants is in the range from 250°F (121°C) to 300°F (149°C). The attached "Temperature-Difference (or Pinchpoint)" diagrams (Figs. 10, 11 and 12) show graphically how higher steam conditions push up the cold end of the reactor coolant temperature with a consequent decrease in reactor power for a constant hot end temperature.

In this design the mean temperature of the primary sodium leaving the reactor has been limited to 925°F (496°C), and a non-reheat regenerative cycle with

Table IV. Neutron Economy of a Thorium Uranium-235 Fuel Cell

	*Beginning of 1st cycle**		*End of 1st cycle*		*Beginning of 2nd cycle*		*End of 2nd cycle*	
Virgin fast neutrons		100.000		100.000		100.000		100.000
Fast leakage	8.303		5.955		8.303		6.040	
Th^{232} resonance captures	18.930		19.202		18.930		19.185	
Thermal leakage	1.354		1.084		1.406		1.106	
Moderator captures	1.457		1.646		1.513		1.644	
Outer coolant tube captures	0.105		0.118		0.108		0.121	
Moderator can captures	0.607		0.632		0.614		0.645	
Moderator coolant captures	1.183		1.230		1.196		1.257	
Control rod captures	1.535		0		1.787		0.017	
Fuel element captures	66.526		70.133		66.143		69.985	
	100.000		100.000		100.000		100.000	
Fuel element captures		66.526		70.133		66.143		69.985
Th^{233}	13.540		16.037		14.268		16.587	
Xe and Sm	2.380		2.439		2.382		2.440	
Na coolant	0.816		0.958		0.851		0.985	
SS cladding	1.611		1.891		1.679		1.945	
NaK bond	0.080		0.093		0.083		0.096	
U^{235} fission	40.650		29.328		29.692		20.991	
U^{235} non-fission	7.449		5.376		5.443		3.848	
U^{233} fission	0		10.966		10.614		19.041	
U^{233} non-fission	0		1.069		1.034		1.857	
U^{234}	0		0.076		0.067		0.211	
U^{236}	0		0.033		0.030		0.056	
Pa^{233}	0		0.296		0		0.303	
Fission products	0		1.571		0		1.625	
	66.526		70.133		66.143		69.985	
Conversion ratio	0.675		0.754		0.710		0.782	

*A cycle represents 10,000 Mwd/ton irradiation.

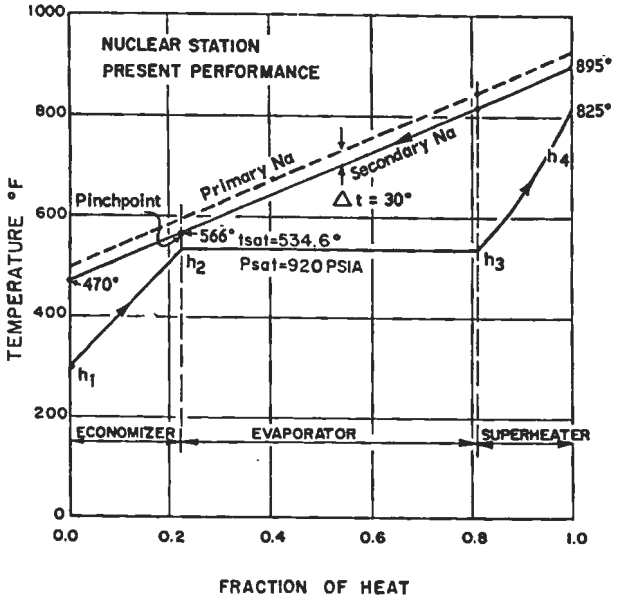

Figure 10. Temperature difference (pinchpoint) diagram. Steam conditions: 800 psig/825°F

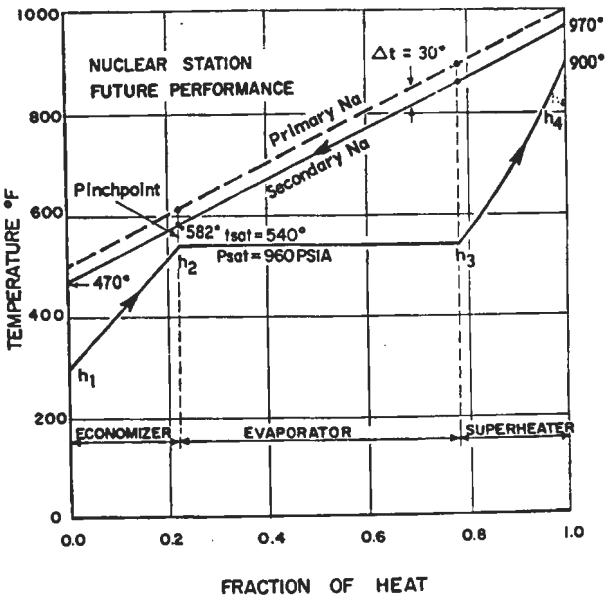

Figure 11. Temperature difference (pinchpoint) diagram. Steam conditions: 850 psig/900°F

800 psig/825°F (5.6×10^7 dynes-cm^{-2}/441°C) throttle conditions and a feedwater temperature of 300°F (149°C) has been chosen. The choice of 800 psig/825°F (5.6×10^7 dynes-cm^{-2}/441°C) represents what appears to be the best compromise between a low cycle heat rate, adequate temperature drives in the sodium to water heat exchangers, and permissible moisture in the turbine exhaust.

A three-stage feedwater-heating arrangement was selected to obtain a final feedwater temperature of 300°F (149°C). Preliminary heat balance calculations show a turbine cycle heat rate of 10,050 Btu/kwh (2532.6 kg-cal/kwh) for a gross electrical output of 80,800 kw. A simplified heat balance for this cycle is shown in Fig. 13.

In the present reactor design the maximum primary sodium and zirconium temperature has been limited to 950°F (510°C) which brings about a mixed mean sodium outlet temperature of about 925°F (496°C). A 30°F (17°C) temperature difference in the primary-to-secondary sodium heat exchangers results in a secondary sodium temperature of 895°F (479°C) which, for a 70° approach in the hot end of the superheater, permits a steam temperature of 825°F (441°C). These relationships are shown in the temperature-difference diagram for this case in Fig. 10.

There exists a reasonable expectation that operating experience with the SRE and future metallurgical developments will raise the present temperature limit. In order to estimate the gains that may result from the expected improvements in reactor performance, two successively higher steam conditions have been examined. The first is the standard steam condition 850 psig/900°F (5.96×10^7 dynes-cm^{-2}/482°C), which represents the least departure from the initial design conditions of 800 psig/825°F (5.6×10^7 dynes-cm^{-2}/441°C). Thus, the additional investment required is probably quite small; yet the net plant heat rate drops from an estimated 10,800 Btu/kwh (2721.6 kg-cal/kwh) to 10,500 Btu/kwh (2646.0 kg-cal/kwh), a 2.8 per cent reduction.

The attainment of 850 psig/900°F (5.96×10^7 extension from the basic design. It may be possible dynes-cm^{-2}/482°C) steam requires a relatively modest at the same time to increase the superheater surface and decrease the temperature approach from 70°F (39°C) to 45°F (25°C). If this change should be

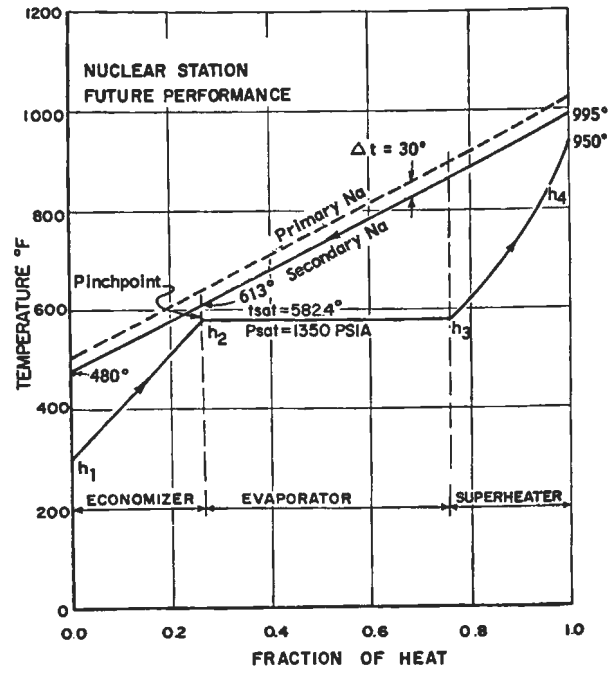

Figure 12. Temperature difference (pinchpoint) diagram. Steam conditions: 1250 psig/950°F

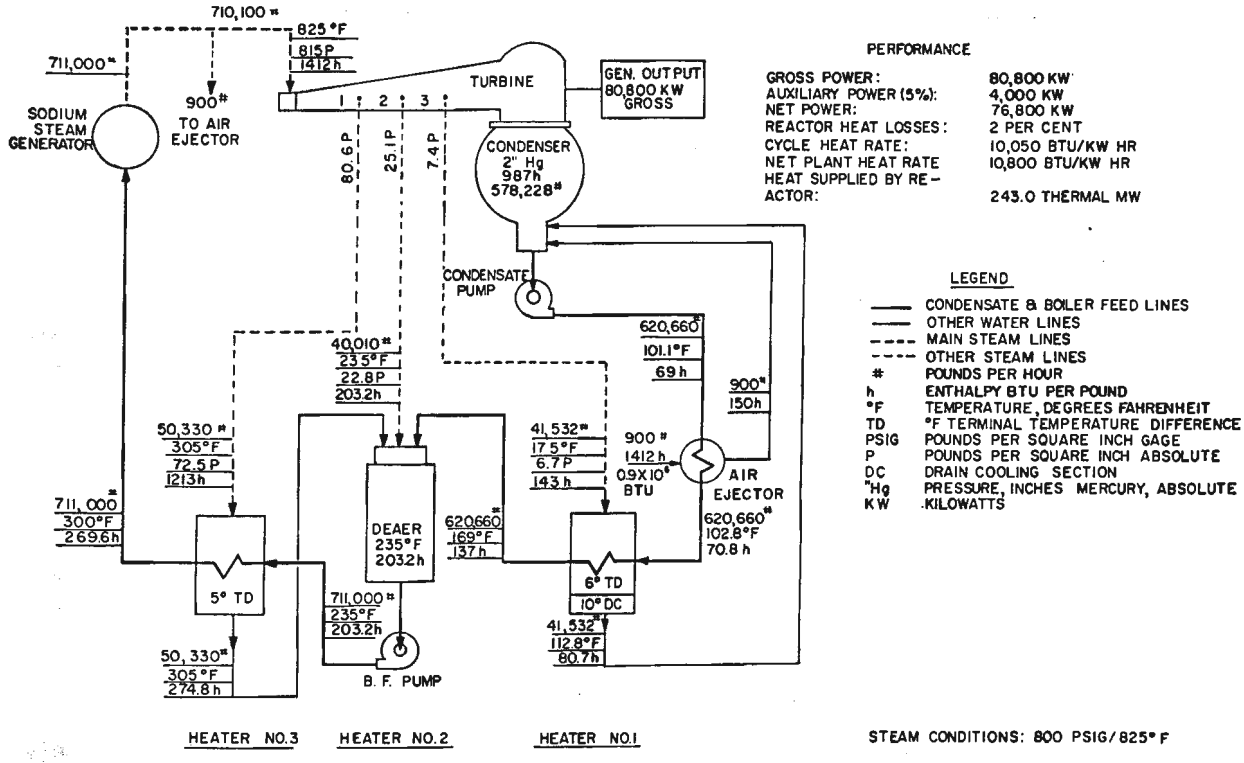

Figure 13. Preliminary heat balance

economically justified, then the attainment of 850 psig/900°F (5.96×10^7 dynes-cm^{-2}/482°C) conditions requires a primary sodium outlet temperature of 975°F (524°C), an increase of only 50°F (28°C) over the initial condition. This reduction of drive in the superheater does not cause a significant change in the pinchpoint. (Pinchpoint is the temperature difference at the entrance to the evaporator.)

In the same manner, the attainment of the standard steam condition of 1250 psig/950°F (8.7×10^7 dynes-cm^{-2}/510°C) requires a mean sodium outlet temperature of 1025°F (551°C), an increase of 100°F (56°C) over the outlet temperature of the present basic design (see Fig. 12). The estimated net plant heat rate drops to 10,160 Btu/kwh (2560.3 kg-cal/kwh), a decrease of 3.1 per cent from the first step (850 psig/900°F) (5.96×10^7 dynes-cm^{-2}/482°C) and a decrease of 5.9 per cent from the initial heat rate of 10,800 Btu/kwh (2721.6 kg-cal/kwh). This condition is summarized in Table V.

Table V. Summary of Conditions for Various Turbine Cycles

Basic data
- Plant gross generation ..... 80,800 kw (approx)
- Plant aux power consumption .... 4000 kw (assumed)
- Plant net generation ..... 76,800 kw (approx)
- Condenser pressure ..... 2 in. (5 cm) Hg abs.
- Number of feedwater heaters 3
- Boiler feed temperature ..... 300°F (149°C)

Heat balance assumptions

The following simplifying assumptions have been made:
- Auxiliary power consumption 5%
  - Assumed constant for all cases, no allowance has been made for the extra power requirements of the auxiliary combustion heater.
- Heat losses in reactor and coolant systems .. 2%
- Efficiency of auxiliary combustion heater 75%
  - No provision has been made for utilization of the hot flue gases except for air preheating.
- Boiler feed pump work has been neglected.

Performance

	Nuclear station			*Nuclear station with aux. combustion heater*	
Steam conditions, psig/°F	800/825	850/900	1250/950	1450/1000	1450/1000/1000
Cycle heat rate, Btu/kwh	10,050	9785	9465	9145	8860
Net plant heat rate, Btu/kwh	10,800	10,500	10,160	10,130	10,100
Net plant thermal efficiency, %	31.6	32.5	33.6	33.7	33.8
Improvement in net heat rate, % (800/825 base)	-	+2.8	+5.9	+6.2	+6.5
Heat supplied by reactor, %	100.0	100.0	100.0	90.0	80.0
Heat supplied by reactor, thermal megawatts	243.0	236.3	228.6	205.2	181.8

POWER COSTS

The cost of power is made up of (1) capital charges on the investment in the reactor system, steam and electrical generating equipment, buildings and site, and charges against fuel hold-up, and (2) operating charges for fuel fabrication, chemical decontaminations, and re-enrichment and charges for operating personnel and plant maintenance.

The capital charges for construction in the United States of the 75,000 kilowatt plant described here have been estimated to be approximately $165 per installed kilowatt of electrical capacity for the reactor portion, $15 per installed kilowatt for the sodium-steam generator and $120 per installed kilowatt for the electrical portion of the plant. These three items total $300 per installed kilowatt for this particular nuclear plant. As a comparison, the average for similar sized conventional hydrocarbon fired plants in the United States is about $200 per kilowatt. Using 15 per cent annual charges against the capital investment and assuming an 80 per cent load factor, the contribution of capital charges to the power costs is 6.5 mills per kilowatt-hour.

The total fuel costs have been estimated to be 2.0 to 3.2 mills per kilowatt-hour (corresponding to values of $15 and $30 per gram of U^{235} in highly enriched material) and the fixed operating and maintenance charge as 2 mills per kilowatt-hour. Thus, the total operating charges of the nuclear plant contribute 4 to 5.2 mills per kilowatt-hour for the cost of power.

The total cost of power from this plant is thus estimated at 10.5 to 11.7 mills per kilowatt-hour. The corresponding average power cost for this size conventional fueled plant is 6 to 8 mills per kilowatt-hour.

EXPECTED FUTURE PERFORMANCE

The power costs given in the previous paragraph are based on (1) conservative design limitations and (2) cost data extrapolated from experience obtained in construction of the Sodium Reactor Experiment. Thus, these power costs indicate what can be expected from "first generation" sodium-graphite nuclear power plants, based upon the present state of engineering information. Reductions in power costs from sodium graphite plants will come from improvements in three broad areas; namely, (1) design and manufacturing simplication, (2) improved utilization of nuclear fuels, (3) increased plant thermal conversion efficiency. The capital costs used in the previous section reflect laboratory type fabrication and instrumentation, and unusually rigid specifications. These present costs will undoubtedly be reduced as industrial experience in this field is developed. Although the extent of such cost reduction is obviously uncertain, it is not unreasonable to assume that the capital cost of nuclear plants may eventually be about the same as present conventional plants.

Improvements in nuclear performance may be more quantitatively estimated. The steady state conversion ratio of the thorium-uranium reactor described for the 75,000 kilowatt electrical plant was given as 0.81. This is the internal conversion ratio; that is, conversions which take place within the core. Examination of Table IV shows the largest single loss of neutrons in this design is leakage neutrons from the core. Approximately 10 per cent of the neutrons produced are lost by leakage. The use of a blanket of thorium can substantially reduce the non-productive leakage loss of neutrons from this type core. A blanket with a catching efficiency of 65 per cent would thus yield an external conversion ratio of 0.14, approximately. Due to the reduced reflector savings from the use of a blanket, the core diameter would have to be increased about two feet to maintain criticality. The total conversion ratio would then become 0.95, provided the nuclear performance is actually as good as estimated. There is every reason to believe that a slightly larger reactor with improved core design and utilizing an effective blanket arrangement will yield conversion ratios about unity. Thus, with such a design, after the initial fuel loading, the *only feed material* to the plant would be thorium metal.

As mentioned previously, the size of the reactor, number of elements, etc., were determined from nuclear and economic considerations. The reactor is capable of producing considerably more power than the amount required to generate 75 megawatts of electricity. If the center temperature of the thorium is increased to 2000°F (1093°C) which appears to be only slightly less conservative than 1600°F (871°C) and the sodium velocity through the central channel is increased to 25 feet per second (7.62 m/sec), the capability of the reactor would be nearly 400 megawatts thermal, or approximately 125 megawatts electrical, with 950°F (510°C) limit retained for the maximum sodium temperature. This improvement in performance does not entail the solution of a basic metallurgical problem as is required to overcome the uranium alpha-beta phase transformation limitation. The absence of any significant phase transformation in thorium and the generally superior metallurgical properties of this material lend confidence to the belief that the improved performance indicated above will be readily attained.

The present development program is expected to raise the 950°F (510°C) limitation on maximum sodium temperature. Advances in this direction will permit a further increase in thermal power from the reactor described here and also will permit improved steam conditions. In fact, the thorium-base fuel element is uniquely capable of taking advantage of higher sodium temperatures, since very high center temperatures in the metal can be sustained. Even now, if one were willing to sacrifice nuclear performance for thermal efficiency, stainless steel could be used instead of zirconium, and a sodium outlet tem-

perature of 1200°F (649°C) could be obtained from a thorium fueled reactor.

With the eventual availability of higher sodium temperature, the use of a reheat-regenerative steam cycle, or a supercritical pressure cycle will become desirable. The latter cycle is especially suitable for a high temperature nuclear steam station. The above possibilities are summarized in Table VI.

The improved performance indicated is closely associated with the fact that the sodium cooled graphite moderated reactor is basically capable of taking advantage of the higher temperatures and powers which metallurgical developments in fuel elements and structures will permit. The history of steam temperatures in conventional stations, over even so short a period as the last decade, provides a striking illustration of metallurgical advances permitting ever higher temperatures. Thus, it would not seem unreasonable to expect similar advances in reactor technology.

Table VI. Expected Progress of Sodium Graphite Nuclear Power Plants

	Present →		→ Future
Fuel	Uranium	Thorium	Thorium
Fuel temperature limit (°F) (°C)	1200 (649)	1600 (927)	2000 (1093)
Sodium-structural T. limit (°F) (°C)	950 (510)	950 (510)	1200 (649)
Steam conditions (psig/°F) (dynes-cm^{-2}/°C)	800/825 (5.6×10^7/441)	800/825 (5.6×10^7/441)	145/1000 (10×10^7/537.4)
Electrical capability (Mw)	75	100	125
Conversion ratio	0.72	>0.9	1.0
Capital costs (dollars/installed kw)			
Reactor system	135	100	80
Cooling system	45	45	20
Electrical generating equipment	120	120	100
Totals: $/kw	300	265	<200
Power costs (mills/kwh) (80% load factor)			
Capital costs (15% fixed charges)	6.5	5.5	4.0
Fuels costs	2.0 – 3.2	1.5 – 2.5	1.0 – 2.0
Operating and maintenance	2.0	1.5	1.0
Totals: mills/kwh	10.5 – 11.7	8.5 – 9.5	6.0 – 7.0

The Design of a Small Scale Prototype of a Homogeneous Power Reactor Fueled with a Uranium Oxide Suspension

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1. INTRODUCTION

The important and valuable properties of liquid fueled homogeneous thermal reactors are well known. The simple construction of the reactor vessel proper without metallic fuel elements, combined with the large energy output per unit volume, the inherent temperature safety with respect to temperature due to a large negative temperature coefficient of the reactivity, which is connected with the fission process without any appreciable time lag and the possibility of a continuous purification are perhaps the most important characteristics. In our particular homogeneous suspension reactor only slightly enriched fuels will be used. Nevertheless the conversion ratio will be high. If we are using ThO_2 suspensions enriched with U^{233} in heavy water, combined with a suitable blanket even a thermal breeder reactor may be possible. However, up to now only uranium is available to us in an enriched form and therefore the first experiments will be carried out with uranium oxide suspensions.

It is possible to reduce the dimensions of a homogeneous reactor considerably‡ using enriched fuel without changing substantially the characteristics and behaviour of the reactor. One can avoid therefore building an expensive large scale high flux and, consequently, high powered reactor for investigating the behaviour and properties of such a suspension reactor. We will therefore start with a small scale slightly enriched UO_2 suspension reactor using light water and a BeO reflector for a rather high neutron flux (10^{13}) operating at a temperature of 250°C.

The following properties can be investigated equally well in a relatively inexpensive small scale reactor as in a large power reactor.

1. The stability of the reactor, especially if the concentration of the suspension fluctuates from point to point or with time.

2. The colloid chemical stability of the suspension under normal reactor power conditions.

3. The erosion and corrosion of the reactor circuit under irradiation.

4. The mechanical breakdown of the suspended particles by attrition or irradiation with fission products.

5. The decomposition of the water inside the reactor.

6. Continuous and batch type purification methods which are possible with a suspension and which are especially important for the removal of Xe^{135}, Sm^{149} and the long-lived radioactive fission products.

Although all these properties can be investigated in a small scale reactor, in certain respects the circumstances are more difficult than in a large one. For reasons of safety the delayed neutrons must be liberated inside the reactor. This limits the linear velocity of the suspension in small reactors to such a value that the settling velocity of the suspension particles is of the same order as the water velocity. Therefore the concentration must be regulated more carefully and the direction of flow must be upwards to ensure that the gases resulting from the decomposition can easily be removed from the reactor vessel.

Secondly the tubes of the outer circuit are relatively long in a small reactor, resulting in either a large outer circuit, or a high flow resistance. To overcome this resistance easily a rotating pump must be used instead of a gas lift pump.§

Finally, in a large reactor it is easy to use such a fuel concentration that the reactivity decreases both with an increasing and with a decreasing concentration. However this is not possible in a small scale reactor fueled with highly enriched material.

Before a description and flow sheet of the small suspension reactor are given in section 3, a few remarks are made in section 2 concerning the suspensions used, the possible purification procedures, the decomposition of the water and the special flow system inside the reactor vessel.

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‡At least if there exists no severe limitation in the fuel concentration as might be the case in the U liquid Bi solution.

§We still believe that instead of canned rotor pumps, the use of multiple gas lift pumps which combine the functions of the pump and the heat exchanger is desirable. Therefore extensive research on the stability and efficiency of such devices is being carried out.

2. CHEMICAL PROBLEMS

2A. The Suspension

Although many uranium oxides are known, e.g., UO_2, U_3O_8, UO_3 and UO_4, we expect to use UO_2 or U_3O_8 for the following reasons.

1. It is rather simple to maintain UO_2 in a reducing atmosphere which oxide cannot easily be reduced further.

2. The oxides UO_3 and UO_4 can be reduced easily and are therefore not very stable at higher temperatures.

3. UO_3 forms several hydrates with water at elevated temperatures. Therefore the number, dimensions and shapes of the original particles will be changed when the suspension is heated. $UO_2 \times H_2O$ is already dehydrated under water above 285°C.[1]

4. UO_2 and U_3O_8 behave similarly from the point of view of colloid chemical stability, so that small deviations of the oxygen content of UO_2 (between UO_2 and $UO_{2,67}$) seem to be harmless in this respect. These suspensions are stable between pH 6.3 and over 10.[2]

5. The crystal parameter of uranium oxide is constant between compositions corresponding to $UO_{2,0}$ and $UO_{2,2}$ but increases from $UO_{2,2}$ to $UO_{2,67}$ ($= U_3O_8$).[3] Small changes of the oxygen content of UO_2 are therefore harmless from the point of view of mechanical stability.

6. UO_2 is stable in water at 250°C under hydrogen pressure.[4]

7. With UO_2 the smallest concentration by volume can be obtained for a given concentration by weight (spec. weight is 10.75 compared with 8.2 for U_3O_8).

The size of the uranium oxide particles will be limited between about 5 and 12 μ. For particles with dimensions smaller than the mean stopping path of the fission products inside this oxide (about 10μ),[5] there is a good chance that both fission products of a uranium fission will leave the original particle of the fissioning nucleus. If the volume concentration of the oxide is low as will be the case for the 20% enriched light water moderated small reactor as well as for the slightly enriched heavy water moderated large reactor (always a few vol %), it is to be expected that the greater part of the fission products will be stopped in the water and not re-enter the particles. This means that a part of the fission energy will be transferred immediately to the water stabilizing the reactor automatically through the large negative temperature effect of the expanding water. Furthermore, an automatic separation of poisoning fission products and fuel will result. If the particles are not too small, e.g., $> 1\mu$, the recoils of the U^{238} nuclei transformed into Pu^{239} will be too small to leave the particle. Neutron irradiation experiments with a UO_2 suspension have confirmed these ideas. From thoroughly stirred suspensions with 4 vol % UO_2 in water, irradiated with thermal neutrons at least 90% of several fission products (Ba, Mo, Cs and I have been investigated) are transferred to the water phase. Of course the separated fission nuclei will be partly readsorbed on the surface of the particles. In a pure water suspension at room temperature the fraction of iodine adsorbed on the surface of the particles compared to the fraction of iodine in the water is determined to be 1.3 : 1.0. Therefore even by purifying the water phase only, part of the iodine can be readily removed.

As a matter of fact, the adsorbed fission products can be easily removed as well by the addition of suitable electrolytes. This, however, makes the cleaning process somewhat more complicated.

For this reason it seems to be desirable to make the surface of the particles as small as possible, which means that the particles should not be too small (5–12μ) and that also the surface area of the particles is as small as possible.

We have determined the particle surface area of several heat-treated particles according to the well known B.E.T.[6] method with the following results (Table I).

When considering the construction of power reactors, great attention must be given to the elimination of Xe^{135} and Sm^{149} particularly if a high conversion factor is desired, as is the case in the thermal Th breeder. Xe^{135}, the most important poison in a power reactor, can be removed as a gas together with decomposed water and adsorbed on active charcoal[7] or silica. Moreover, it is possible to remove the mother I^{135} which is already from the water phase. Calculations have shown that for a small gas let off ($< 6\%$) the removal efficiency of I^{135}, and for a large let off ($> 6\%$), the removal of Xe^{135}, is more efficient. A combination of both methods is of course most effective. In the small scale prototype reactor

Table I. Internal Active Surface of UO^2 Particles with Different Reduction and Heat Treatments Determined by Nitrogen Adsorption According to the B.E.T. Method

Reduction temperature from U_3O_8 to UO_2	*Heat treatment after reduction*	*Measured surface/gm*	*Calculated external surface/gm*
550°C in NH_3	no	5 m^2	
900°C in H_2	no	1.1 m^2	
900°C in H_2	2 hr at 1400°C in H_2	0.3 m^2	0.1 m^2

(section 3) the possibility of applying both methods is foreseen.

2B. The Decomposition of the Water

It is very difficult to predict from experiments outside a reactor the amount of decomposition of the water or heavy water. The small scale reactor experiment will give us the answer. A few estimates, however, can be made. The decomposition is mainly caused by the highly ionising fission particles during the stopping process in the water. Since on an average the fission products are passing half of their mean stopping path through the uranium oxide the other half of the stopping path in the water, it is very probable that the decomposition will be about a factor two smaller than in the homogeneous solution reactor. There are two more reasons for a smaller decomposition. In the first place practically no electrolyte is present in the water and many electrolytes especially those with ions which easily undergo oxidation or reduction promote the formation of H_2 and O_2. In the second place an excess of hydrogen or deuterium gas will be maintained in the reactor vessel with the result of decreasing the H_2 and O_2 formation.[8]

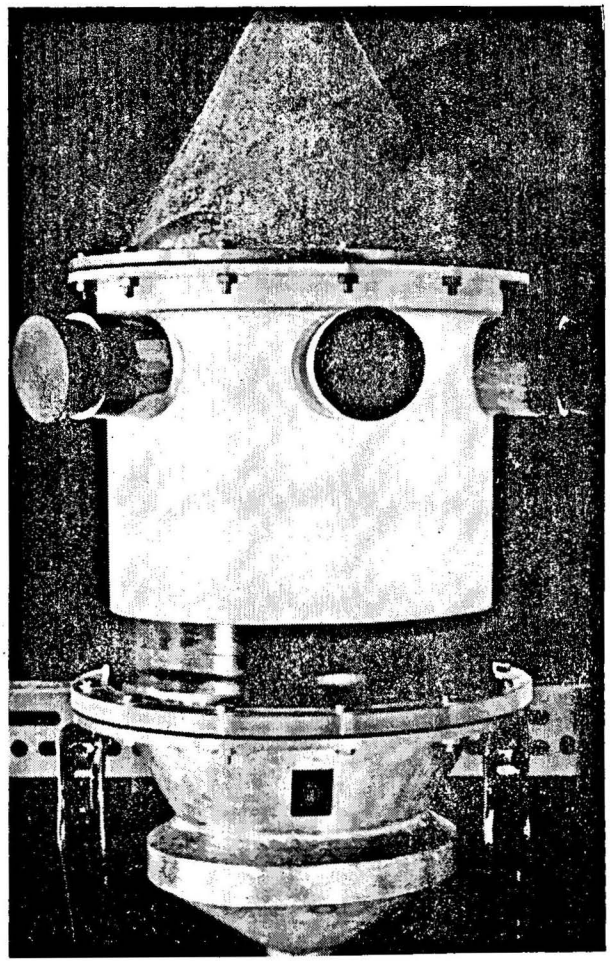

Figure 1a. Model of the small-scale reactor vessel. The side tubes are only present for experimental purposes

2C. The Reactor Vessel

In the reactor vessel a constant concentration and sufficiently homogeneous suspension must be maintained. Model experiments (Fig. 1) have resulted in the construction of a reactor vessel of the type represented in Fig. 2. The suspension is mixed thoroughly in a turbulent mixing chamber above which a set of vanes impart to the suspension a slowly rotating flow, while at the top of the vessel the suspension is collected again in a small space to enter the tubes. Measurements with light absorption have shown that the distribution of the suspended particles is fairly homogeneous inside such

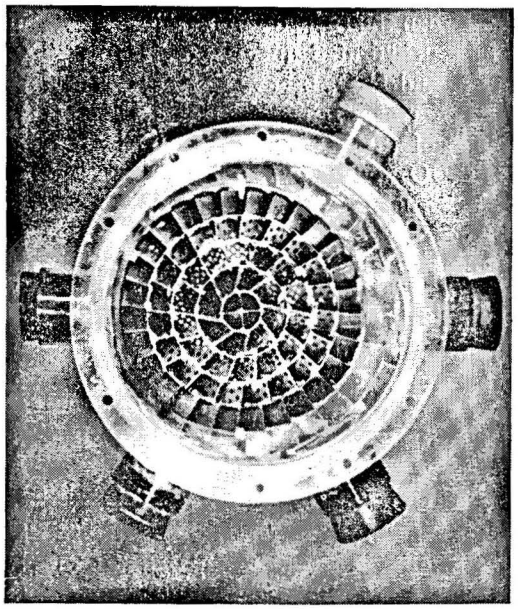

Figure 1b. Inside view of the reactor vessel with the straightening device on the bottom

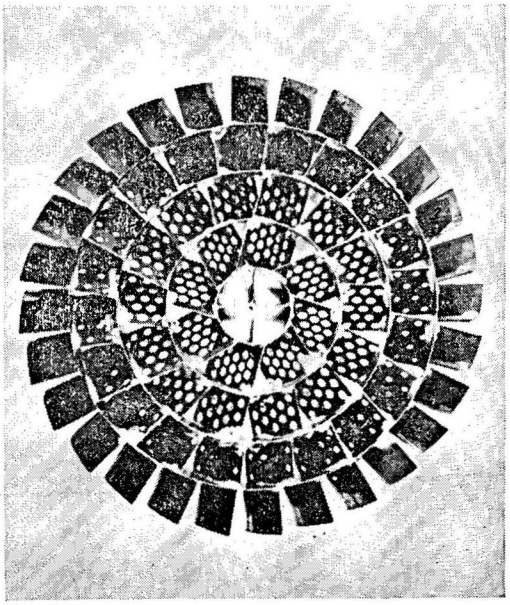

Figure 1c. Straightening device

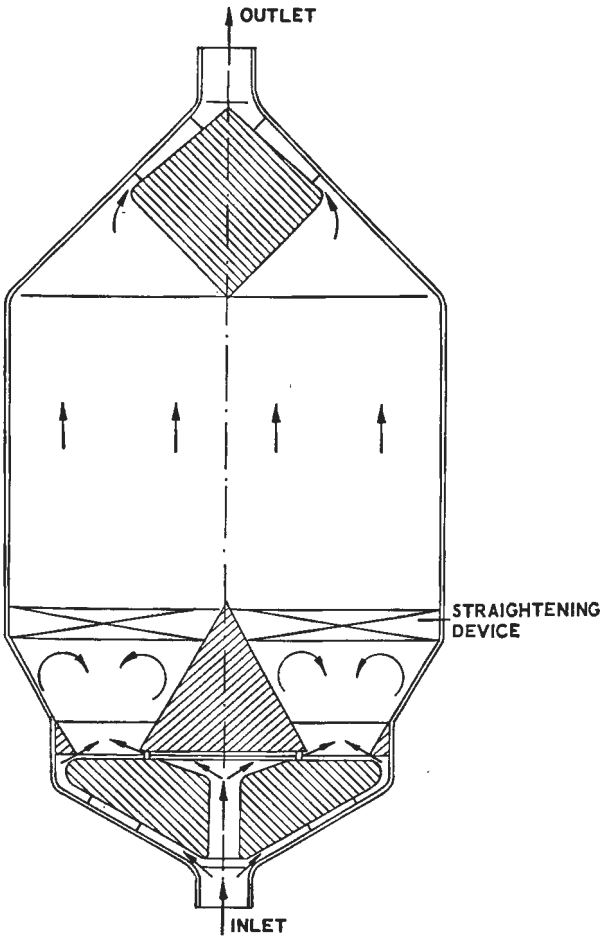

Figure 2. Drawing of the small-scale reactor vessel. Inside the cylindrical vessel, the suspension is slightly rotating

a vessel. In the stationary state the concentration is constant. It is very difficult, however, to formulate quantitatively our demands in this respect. A zero-energy experiment, now under way, will have to give the answer. If sudden and large fluctuations in the reactivity occur which result in dangerous pressure waves, provisions will be taken for a surge volume at the top of the reactor vessel.

3. FLOW SHEET OF A 250 KW HOMOGENEOUS SUSPENSION REACTOR WITH CONTINUOUS PURIFICATION

Figure 3 represents the flow sheet of the small scale high flux, high temperature reactor. This reactor will be built after a satisfactory zero-energy experiment, which will start this year, has been made.

We will not give a detailed description of this flow sheet, but a few remarks may be useful. The primary circuit consists of the reactor vessel *A*, a gas separator *B*, a canned rotor suspension pump *C* and a heat exchanger *D*. In the gas separator of special design, the decomposed water (H_2 and O_2) is removed and mixed with such a large amount of gas that even at elevated temperatures and high pressure (about 50 atmospheres) no explosion can take place. In the gas circuit a catalytic recombiner and condenser are provided together with a canned gas pump. A by-pass makes it possible to remove continuously some of the poisoning Xe-gas present in the gas circuit. No valves are provided in this primary circuit. In the whole system only one suspension valve is used, viz. in a by-pass *E* in which a storage vessel *F* is present for the fuel. This valve has a special design to ensure a reliable operation. When it is opened a stream of pure water dilutes the settled suspension inside the storage vessel to lower the viscosity and to wash the suspension going to the primary circuit. A second by-pass *H* which has no storage vessel is provided to allow a continuous purging of the water for purification in ion exchange columns.

The regulation of the temperature of the reactor is carried out by adjusting the concentration of the fuel with the storage vessel and suspension valve. The energy production, and therefore the neutron flux, is regulated by heat extraction through the secondary cooling circuit *G*.

The entire system is under gas pressure — both the vessel containing the suspension and the reflector — with two exceptions. The dump vessel *K* will always have a somewhat lower pressure than the reactor system proper to ensure a very fast discharge of the whole reactor content in an emergency. The other exception might be the ion exchange columns *L*. They will probably be used at normal pressure making the design this part of the purification system. However, in that case a small compressor pump will be needed between the outlet of the ion exchange columns and the water inlet in the primary reactor circuit.

For the small scale reactor described in this paper, it is not necessary to combine a continuous purification system with the reactor, a batch purification from time to time would be quite adequate. However, in order to gain experience with these systems, we hope that an intermediate power reactor can be constructed immediately after this small scale prototype.

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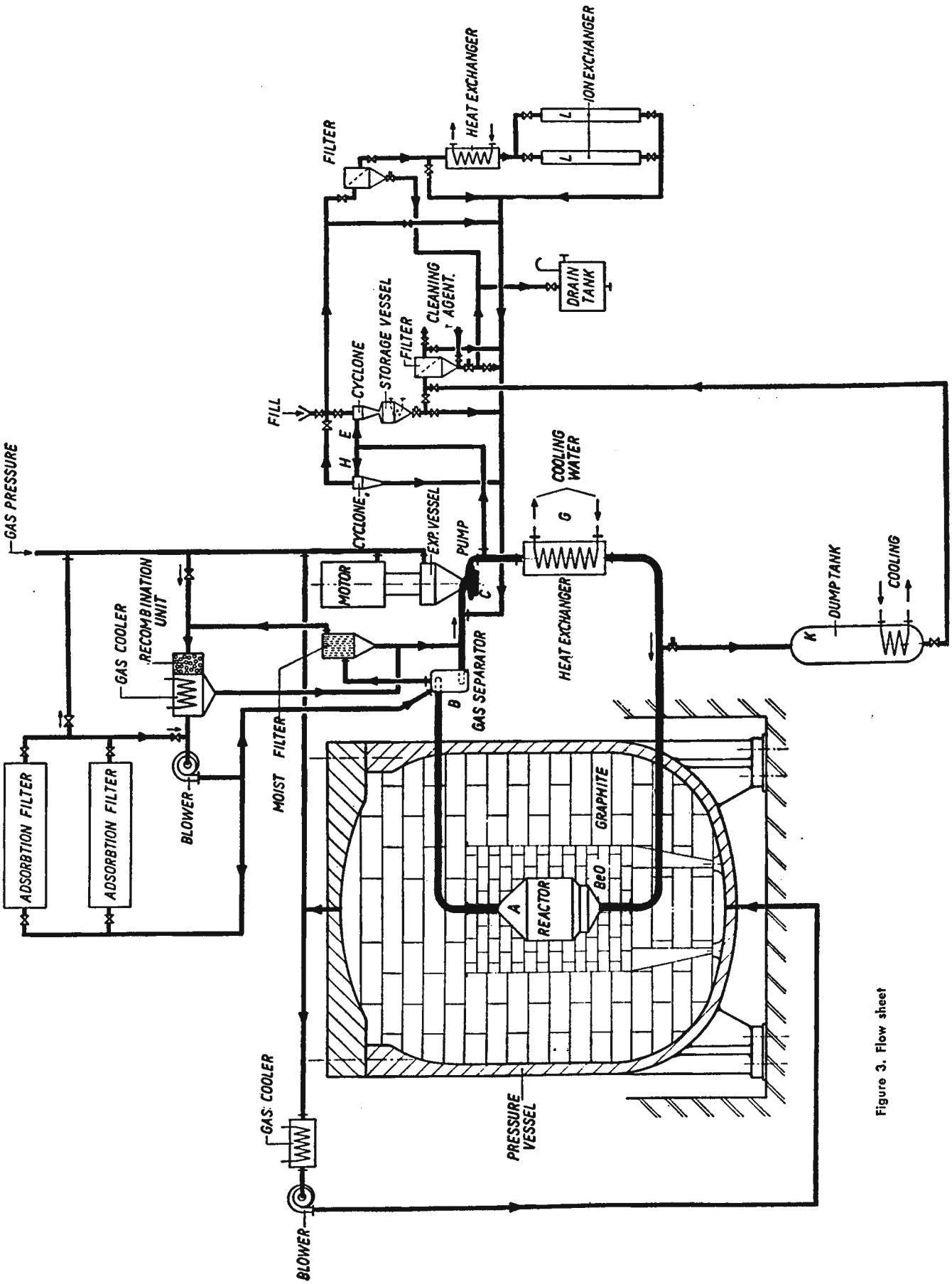

Figure 3. Flow sheet

A "Dry" Suspension of Uranium Oxide for a Heterogeneous Power Reactor

By H. de Bruyn,*† B. L. A. v. d. Schee,* J. J. Went,* The Netherlands

I. INTRODUCTION

The application of suspensions of uranium dioxide as a liquid fuel for power reactors was discussed by two of us at the Ann Arbor conference about a year ago.[1]

In the present paper we shall extend the discussion on the "dry" suspensions (gas as a medium) whereas in another paper presented to this conference a description will be given of a small scale prototype of a homogeneous "wet" suspension (heavy water as a medium) power reactor.

Three main types of power reactors fueled with "dry" suspensions of uranium dioxide will be distinguished: a homogeneous reactor, a quasi-heterogeneous reactor‡ and a heterogeneous reactor.

Of these the heterogeneous type of reactor will be treated in more detail as in our opinion it has more valuable qualities than the other two.

II. THE HOMOGENEOUS REACTOR

At first sight the homogeneous reactor type offers the best prospects because its construction would be the simplest.

The reactor vessel is filled with compound grains of uranium dioxide and beryllium oxide (with an atomic ratio of Be/U of about 120). The grain size distribution is well chosen to ensure a rather close packing (70–75% by volume). The mean grain size will be about 200–250μ. The cylindrical vessel is surrounded by a beryllium oxide reflector 70 cm thick. The grains move by gravity in a downward direction through the reactor vessel and are let off through holes in the bottom of the reactor vessel. By choosing suitable geometric form of the bottom of the reactor vessel it is possible to establish a velocity distribution in the flow of grains down the reactor which corresponds to the buckling of the reactor.

The grains leaving the reactor vessel are transported pneumatically to a heat exchanger in which the bed of compound grains is fluidized by means of a gas stream. Helium is used. The gas velocity will be adjusted properly to ensure a moderately boiling state of the grains. Intensive boiling must be avoided to prevent too much attrition of the grains. In this boiling bed of grains the degree of packing will be about 35% by volume. The heat transfer coefficient of such a boiling fluidized bed is rather high (about 500–600 joule/m^2-°C-sec). After the passage through the heat exchanger the grains are returned to the reactor vessel by simple overflow of the boiling bed.

Inside the heat exchanger the heat is transferred to the water circulating in the boiler tubes. At a pressure of 100 kg/cm^2 steam of 310°C is produced. Part of the heat exchanger may be used as superheater, to produce high quality steam of 550°C.

We calculated the necessary dimensions of the reactor and heat exchanger for a 300 Mw power reactor assuming a maximum temperature of 1200°C of the grains inside the reactor and 700°C in the boiling bed. The size of the reactor vessel proper could be 1.85 m in diameter and 1.85 m in height when the maximum flow velocity is limited by the number of delayed neutrons necessary for smooth regulation of the reactivity. The weight of compound grains (moderator and fuel) inside the reactor comes to 10.2 tons BeO and 0.9 tons UO_2. The hold up in the heat exchanger without superheater amounts to 31.5 tons BeO and 2.9 tons UO_2, with superheater to 50.8 tons BeO and 4.7 tons UO_2 (boiler tubes of 2″, superheater tubes of 1½″ and a tube spacing of 1½ × tube diameter).

It is doubtful whether the production of high quality steam (better efficiency of the turbogenerators and smaller cost of turbine installations) will pay when superheating is connected with an extra hold up of compound grains. Not only is the beryllium oxide rather costly but the uranium has to be fairly highly enriched (about 6% U^{235}) and therefore the necessary amount of UO_2 will be costly as well. Apart from the fact that moderately enriched uranium must be used as a fuel the conversion ratio will be rather low in a reactor of this type. From the point of view of fuel economy this reactor is not very attractive since we may expect strong attrition of the costly compound grains.

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‡ A quasi-heterogeneous reactor is homogeneous from neutron physical point of view although it has a heterogeneous structure.

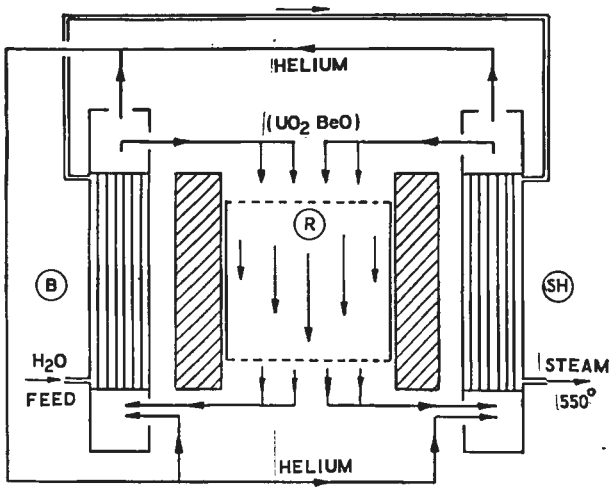

Figure 1. Flow sheet of homogeneous reactor fuelled with a dry suspension of uranium oxide. *R* is reactor vessel surrounded by a BeO reflector; *B* is boilerpart of heat exchanger; *SH* is superheater part of heat exchanger

III. THE QUASI-HETEROGENEOUS REACTOR

The large hold up of costly compound grains of enriched uranium oxide—beryllium oxide in the heat exchanger *outside* the reactor being one of the severe drawbacks of a homogeneous power reactor fueled with "dry" suspensions — we examined the possibilities of a reactor type with *internal* heat exchanger. The boiler tubes (1½") are placed inside the reactor; no circulation of the compound grains is necessary for the transfer of heat. Still the grains can be removed for reconditioning or replacement.

With a boilertube spacing of 2¼" the reactor can be treated as a homogeneous reactor. Between the tubes the space is filled with compound grains of uranium oxide and beryllium oxide (with an atomic ratio Be/U of about 80). This time the grains are much larger, namely 3 mm. A stream of helium is passed through the fixed bed of grains to ensure the transfer of heat from the grains to the tubes. Although lower than in case of a boiling bed of grains the heat transfer cofficient remains large.

When aiming at a power reactor of 300 Mw thermal power and assuming a fuel temperature of 1200°C and a steam temperature 250°C (40 kg/cm^2 pressure inside the tubes), the dimensions of the reactor without reflector will be: diameter 2.50 m, height 2.50 m. From a constructional point of view it is necessary to *pressurize* the whole reactor. The helium used for the transverse conduction of heat inside the reactor is heated up to 1200°C and can be used for the production of dry steam at 310°C in a separate heat exchanger outside the reactor. When steel tubes are used as boiler tubes the required enrichment of the fuel will be rather high. Therefore it was suggested that we use graphite tubes. In that case an enrichment up to 4% U^{235} seems to be sufficient.

This type of reactor offers certain advantages over the homogeneous reactor. First of all the amount of fuel and moderator required for a power reactor of 300 Mw is much smaller, namely 1.5 tons UO_2 and 11.5 tons BeO. Moreover, attrition of the rather costly compound grains and erosion of construction materials is eliminated. Finally the control of the heat extraction will be much easier. The crucial valves for the control of the let-off of grains will be of the same device as is discussed in the next paragraph. However, the fact that this reactor has to be pressurized may be a disadvantage.

IV. THE HETEROGENEOUS REACTOR

In both reactor types which are briefly discussed in the preceding paragraphs moderately enriched uranium must be used as a fuel. If fuel economy is an important consideration, it is worth while to consider a reactor type which can be fuelled with natural uranium and which ensures the highest possible conversion ratio.

These conditions can only be fulfilled in a dry suspension reactor if heavy water of a relatively low temperature is used as a moderator (temperature below 50°C) and if a heterogeneous construction of the reactor is chosen.

A. The Reactor Vessel

With a 300 Mw thermal power reactor the reactor vessel may have a diameter of 3.40 m and a height of 2.50 m. In this vessel, filled with about 20 tons of heavy water, 250 tubes containing uranium oxide grains as reactor fuel, are placed. The diameter of these tubes, made of graphite, is 9 cm outside, 8 cm inside.

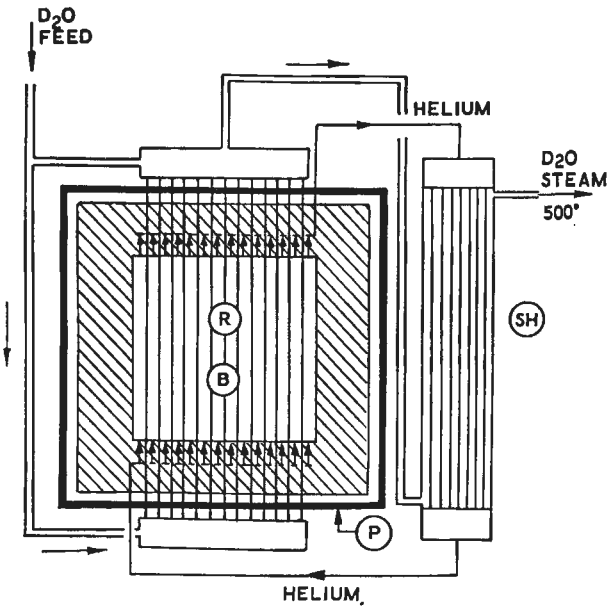

Figure 2. Flow sheet of quasi-heterogeneous reactor fuelled with a dry suspension of uranium oxide. *R* is reactor vessel containing large compound grains of UO_2BeO in fixed bed condition; *B* is boiler inside the reactor vessel: *SH* is superheater; and *P* is pressure vessel enclosing reactor vessel and BeO reflector

The pure uranium oxide grains of 250 μ diameter move in a downward direction. When no gas is flowing between the grains the heat conductivity is poor and the loss of heat through the graphite to the heavy water is rather low (about 2% of the power production). At the bottom of the reactor the hot uranium oxide grains (1200°C) are transferred to a stream of helium and transported pneumatically to the heat exchanger. Inside the reactor vessel a 40 cm thick graphite reflector is placed which is cooled by the circulating heavy water.

B. The Heat Exchanger

Inside the heat exchanger the bed of grains is fluidized. The concentration by volume of the grains will be about 35%. The temperature of the boiling bed is 700°C. The hold up of fuel in the heat exchanger has been calculated when 2 in. tubes are used as boiler tubes and 1½ in. tubes for superheating the steam to 550°C (100 kg/cm² pressure). It turned out that for the boiler a hold up of 92.5 tons uranium oxide and for the superheater an additional hold up of 57.5 tons was necessary, while the tubes inside the reactor are filled with 24.4 tons uranium oxide. Therefore the total amount of fuel necessary is still rather high: *175 tons*. Once again the question arises whether superheating of the steam is advisable although in this case the cheaper pure natural uranium is used as a fuel.

When the heat exchanger containing a large amount of uranium oxide is built around the reactor vessel it will serve as a rather efficient γ-shield. The heat developed from the absorption of the γ-rays will be used for steam production. Therefore the problems of preventing damage to the concrete shield which encloses the entire power reactor will be reduced.

The heat exchanger will also operate as a very efficient neutron shield for the neutrons without considering the graphite reflector. All escaping neutrons will be used in the heat exchanger for Pu production.

C. Details of Construction

Although at the moment the investigations with regard to the various constructional problems are still in a preliminary stage and no details can be published, we will make one exception for the regulation of the flow of the uranium oxide through the tubes. As a matter of fact, the flow rate has to be highest in the tubes placed in the centre of the reactor vessel. The distribution of the flow rate over the different tubes must correspond to the buckling of the reactor.

One of the devices proposed for the remote control of the flow rate in each tube consists of a vibrating cone of special form shown in the lower part of Fig. 4. This cone, constructed of beryllium oxide, is vibrated rapidly. The flow rate of uranium oxide grains depends on the amplitude of the vibrational movement of the cone.

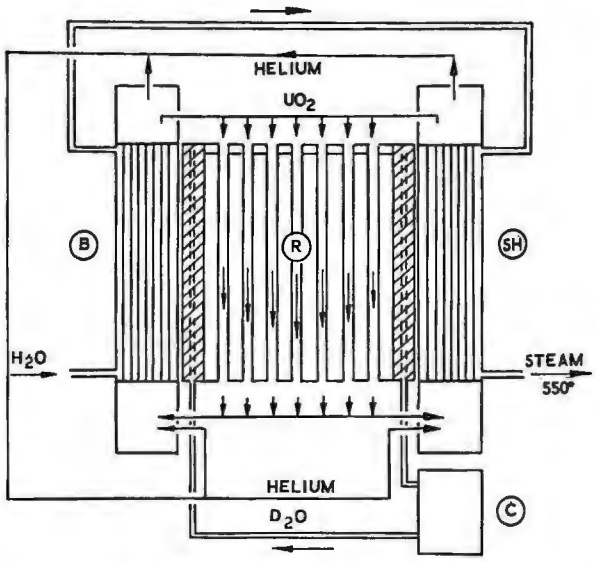

Figure 3. Flow sheet of heterogeneous reactor fuelled with a dry suspension of uranium oxide. *R* is reactor vessel containing graphite fuel tubes, cold heavy water as moderator and a graphite reflector; *B* is boilerpart of heat exchanger; *SH* is superheater part of heat exchanger; and *C* cooler of heavy water circuit

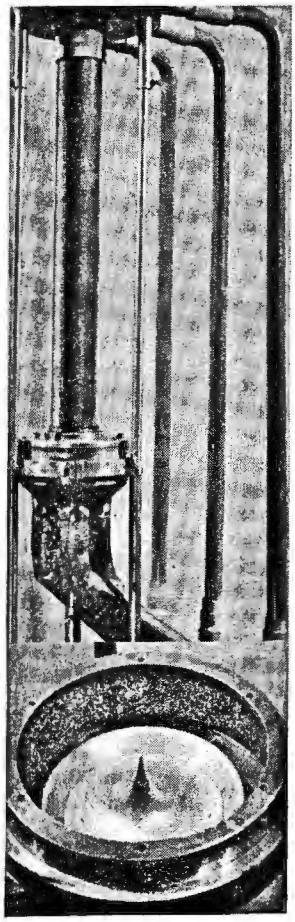

Figure 4. Flow rate regulator for "dry" suspensions. Upper part showing valve connected with tube of 9 cm diameter. Lower part showing inside of valve consisting of special shaped vibrating cone

In another proposition each tube has a special fixed resistance at the outlet opening adjusted to the desired flow rate.

D. Reactor Control (Power Level, Emergency Cases)

The reactivity of a heavy water moderated reactor is strongly dependent on the temperature. Therefore the power level of the reactor can be regulated conveniently by the adjustment of the temperature of the heavy water, in other words by the *circulation rate of the heavy water* through the cooler necessary to transfer the heat accumulated in the heavy water owing to the loss of heat through the graphite tubes.

The temperature of the uranium oxide inside the reactor tubes will correspond to the power level of the reactor when the circulation rate of the uranium oxide is constant.

If for any reason the passage of uranium exide is blocked in one of the tubes it will be noticed immediately owing to a rise of the temperature in this particular tube and safety measures can be taken. For shut-down, cadmium rods or boron steel plates will be used.

In case of stagnation of the gas compressors the reactor automatically shuts down because the supply of uranium oxide at the top of the tubes immediately stops.

In case of stagnation of the heavy water pumps the reactor will stop because the temperature of the heavy water rises.

E. Neutron Economy

In a reactor with cold heavy water as a moderator and graphite as construction material the conversion ratio will be high. However the fission products accumulated in the fuel during operation will gradually increase the non-productive absorption of neutrons. In our laboratory the diffusion of iodine in uranium oxide at high temperature has been studied. In his preliminary experiments v. d. Plas noticed a sufficiently high diffusion rate already at 1000°C.

We may expect therefore that xenon^{135}, the reactor poison "par excellence," can be eliminated for the greater part by continuous purification of the helium gas circuit.

On the other hand, samarium^{149} and rare-earth fission products cannot be removed in this way. However, we mentioned the attrition taking place when "dry" suspensions are used as "liquid" fuel. The dust has to be removed from the fuel circuit continuously. It can be reconditioned by a simple sintering process but when the poison level rises too high it can be replaced by new fuel whereas the dust is reprocessed in batch.

CONCLUSION

After a brief description of three types of power reactors fuelled with "dry" uranium oxide suspensions one gets the impression that the heterogeneous power reactor offers the best possibilities. It has the following attractive qualities:

1. The possibility of producing high quality steam *without* the necessity of *pressurizing* the reactor;
2. The *cheapest fuel* can be burnt in this reactor, namely natural uranium in the oxide form;
3. A *high conversion ratio* and probably the possibility of breeding with thorium oxide are ensured.

However in case moderate enrichment of fuel can be accepted and a high conversion ratio is not necessary, the quasi heterogeneous type of reactor will be the most attractive one from a point of view of heat extraction.

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Liquid Metal Fuel Reactor

By F. T. Miles and Clarke Williams,* USA

Since the early days of reactor technology the concept of a fluid fuel which could be circulated for cooling and processing has been an intriguing one. A solution of uranium in bismuth was suggested because of the low melting point and low neutron capture cross section of bismuth. The principal development of the U-Bi fuel in the USA has been at the Brookhaven National Laboratory.† A variety of reactor designs can be based on the liquid metal-fuels developed here.

This paper presents a design for a central station power plant which will produce 210,000 kw of electricity. A high thermal efficiency is achieved with the U-Bi solution fuel as its use in a reactor will produce steam at 482°C and 88 kg/cm^2. The U^{233} which is used as the fuel is produced in the reactor from thorium which surrounds the reactor core in a "breeder blanket." Both the U-Bi fuel solution and the thorium bearing fluid are circulated for heat removal and chemical processing.

Figure 1 is a schematic design showing the principal cooling and processing streams leading from the reactor core and breeder blanket. First, the main fuel stream is circulated through a sodium heat exchanger, the sodium carrying the heat to the steam boiler. The thorium bearing blanket stream is composed of a slurry of Th_3Bi_5 suspended in molten Bi and is also circulated through a heat exchanger for cooling. Volatile fission products are carried by a helium stream from the free surface of the U-Bi solution and stored in traps. A small part of the main fuel stream flows through the processing system where the fission products are removed and additional U^{233} added. Finally, a process stream flows from the blanket for removal of the Pa and U^{233} (produced in the blanket), as well as fission product removal. The advantages of fluid fuels, in general, with respect to handling for processing and cooling, have been discussed.[1] The additional advantages of uranium-bismuth solution fuels are stability to radiation as well as temperature.[2]

REACTOR ARRANGEMENT AND CONSTRUCTION

The reactor consists of a graphite core through which the U-Bi fuel is pumped, an inner graphite vessel which separates the fuel stream from the thorium bearing breeder blanket, the blanket and an exterior containing vessel to which are attached the inlet and outlet piping for the fuel and blanket streams. Figure 2 shows a vertical section through the reactor. It is desirable to keep the concentration of uranium in the fuel stream low as a considerable volume is held up in the heat exchanger and external piping. By having a thermal reactor the concentration of uranium-233 required to maintain a chain reaction under operating conditions is only 670 parts per million by weight (ppm). This is well below the solubility limits of uranium in bismuth at 400°C, the minimum temperature in the operating cycle. Because of its relatively low cost as well as its ability to withstand high temperatures, graphite was chosen as a moderator for slowing the neutrons. As it has a low neutron capture cross section and retains its strength at the temperatures in the reactor, it is also used as the material for the interior structure of the reactor. No reaction between graphite and the uranium-bismuth fuel has been observed at temperatures up to 1000°C with exposures of 100 hours. At higher temperatures up to 1200°C there is some reaction. However, in the presence of a few hundred parts per million of Zr there is a preferential formation of ZrC and no evidence of UC is found. On all counts the operation of the reactor with the U-Bi fuel in direct contact with the graphite is quite feasible.

This reactor design uses very small quantities of metallic absorber materials for structural purposes in the core and blanket which results in a high neutron economy. The reactor core is pure graphite; the barrel-shaped vessel separating the core fluid from the blanket fluid is, with the exception of the alloy bonding rings, pure graphite. (See Fig. 2.) The core is made of a solid right circular cylinder of high density graphite 5 feet in diameter (152.4 cm) and 5 feet high (152.4 cm). This cylinder is perforated with holes 2 inches (5.1 cm) in diameter and paral-

*Brookhaven National Laboratory.

† A solution of uranium in bismuth as a fluid reactor fuel was first suggested by von Halban and Kowarski in 1941. Wigner, Weinberg, and Young made preliminary calculations on a breeder of this type in 1944. Work at Brookhaven National Laboratory on this system was started by W. E. Winsche and D. H. Gurinsky, and has been going on since 1947. Among others who have made significant contributions are D. W. Bareis, J. Chernick, O. E. Dwyer, I. Kaplan, C. J. Raseman, T. V. Sheehan, R. J. Teitel, and R. H. Wiswall, Jr.

The design for a liquid metal fuel reactor presented in this paper is mainly the work of W. A. Robba, D. J. Sengstaken, T. V. Sheehan, and L. D. Stoughton of Brookhaven National Laboratory. The design for the power production cycle was suggested by a group of utility engineers as part of a course in nuclear engineering at Brookhaven National Laboratory.

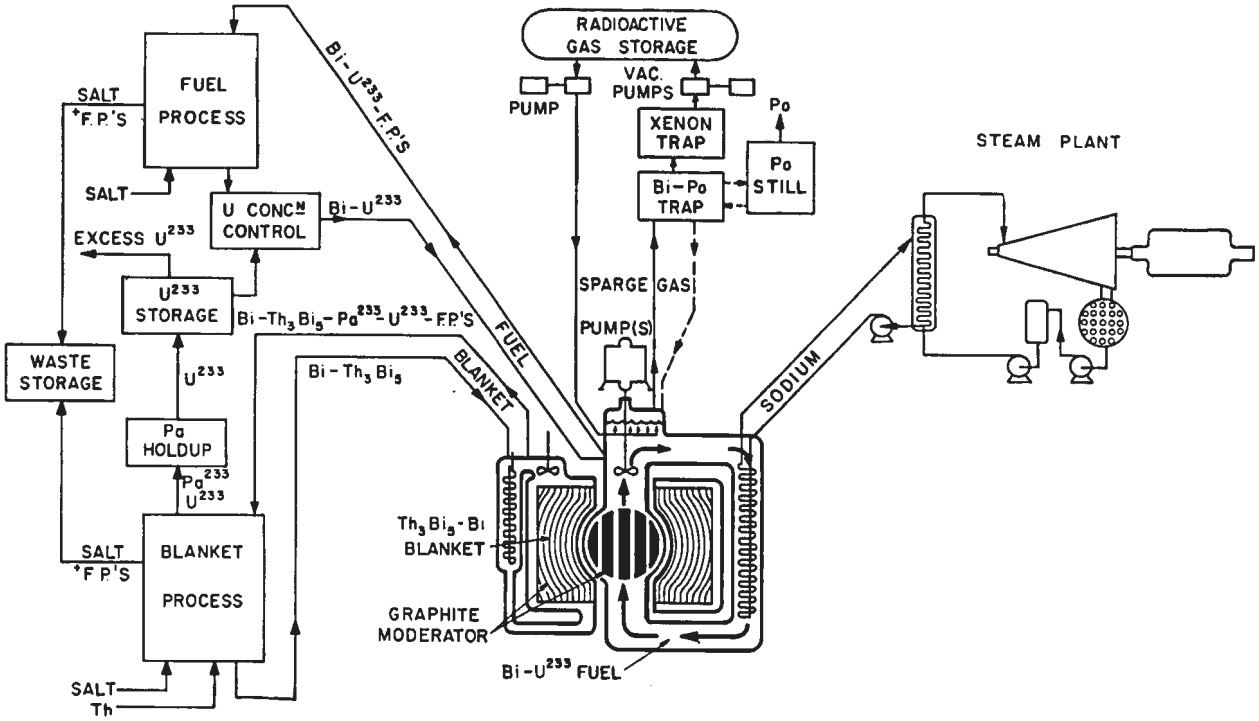

Figure 1. Liquid metal fuel reactor. Schematic diagram showing reactor, steam plant, and chemical processing

lel to its axis. The holes are spaced 2.7 inches (6.9 cm) center to center in a triangular array so as to give the volume ratio of U-Bi to graphite of 1:1 as required by the nuclear design. Under full load condition the U-Bi solution, flowing at a rate of 36,000 gpm (137,500 liter/min), enters the core at 400°C. Fission takes place in the core generating heat at a rate of 500 Mw and raises the temperature of the fuel to 550°C at the exit to the core. The average flux in the core is 10^{15} neutrons per cm^2 per sec.

The internal graphite vessel is made of longitudinal staves, like a barrel, machined so that they will fit over the core, and, at the same time, fit together to form a cylindrical vessel. The abutting edges of the staves are dovetailed together, and all such edges, as well as the surfaces of the core and the staves where these join, are coated with carbon forming paste. The entire assembly is then temporarily clamped together. The alloy bonding rings are thermally shrunk onto the temporary assembly, forming a tight final assembly. This assembly is then heated to a sufficient temperature to carbonize the paste and firmly join the parts essentially leak tight.

End connections from the graphite inner vessel to the outer containment vessel are made by means of transition pieces which make a connection between the alloy clamping ring attached to the graphite and the outer containment vessel. The alloy clamping ring is made of metals or alloys having substantially the same coefficient of thermal expansion as does graphite. By using a sufficient number of transition pieces of varying alloy content stresses due to differential motion can be kept within bounds. The connection at the bottom firmly anchors the graphite inner vessel, while that at the top allows differential axial motion. The slight leakage that may occur through this joint will not affect the operation of the core or blanket.

As a moderator is required in the blanket, graphite rods 3.2 inches (8 cm) in diameter are placed in the blanket volume outside the graphite vessel described above. The rods rest in grids which fix the space at 4.3 inches (11 cm) center to center, the bottom grids containing a socket for the lower end of the rod. The top grid contains a hole to permit the assembly of rods. A plug is then screwed into the top end of the hole to retain the rod.

Fuel is pumped directly through the inner vessel and core from bottom to top. The thorium-bismuth slurry is pumped through the blanket, flowing upward around the rods and out at the top.

The entire graphite core and blanket assembly is contained in a steel pressure vessel 12 feet (366 cm) in diameter and 15 feet (457 cm) in height. The material used in the container vessel and connecting piping is low chrome steel containing 2¼% Cr-1% Mo, known as "Croloy 2¼." This is a relatively inexpensive steel used for boiler tubing which has excellent creep and strength properties at the operating temperature of 550°C. The chief corrosion problem in a liquid metal system, such as the U-Bi solution in steel, is that of mass transfer—that is, the deposition of dissolved metal at the cold region of circulating liquid metal stream. This has been successfully solved by the additions of Mg to pick up any traces of oxygen, and of Zr to inhibit both the rate of dissolution of the steel in the high temperature region and the deposition in the low temperature region of

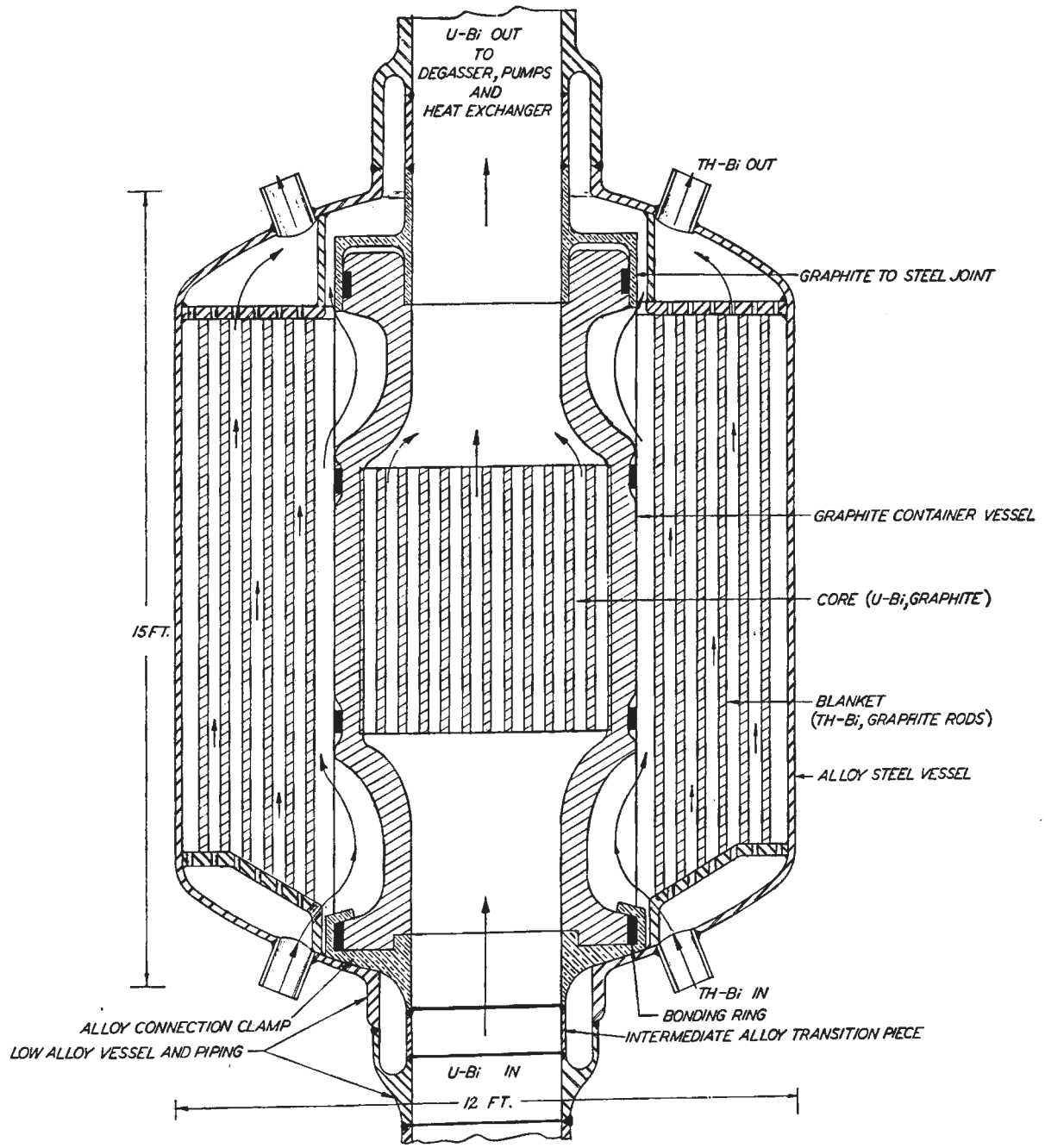

Figure 2. Reactor vessel, vertical section

such systems. These effects are discussed in more detail in one of the papers submitted to this conference by D. H. Gurinsky *et al.* of Brookhaven Laboratory.[3]

We have circulated U-Bi solutions under these conditions in the temperature ranges between 400°C to 550°C for more than 18,000 hours without any appreciable signs of this mass transfer plugging. Most of these loops have relied on thermal convection for circulating the U-Bi solution, but in a few of them an electromagnetic pump has been used. A description of the loops and their operation has been published elsewhere.[3,4] The successful operation of these loops indicates that power reactors should operate for 10 to 20 years without trouble due to corrosion or allied difficulties.

HEAT REMOVAL SYSTEM

At full load the reactor will consume 560 gm of U^{233} per day in the core producing heat at a rate of 500 Mw. In order to remove this heat from the reactor the fuel is circulated through the core and the main reactor cooling loop as shown in Fig. 3 which is a schematic diagram of the entire heat cycle, and Fig. 4 which is a sectional view showing the reactor heat exchangers, piping, boiler and auxiliary

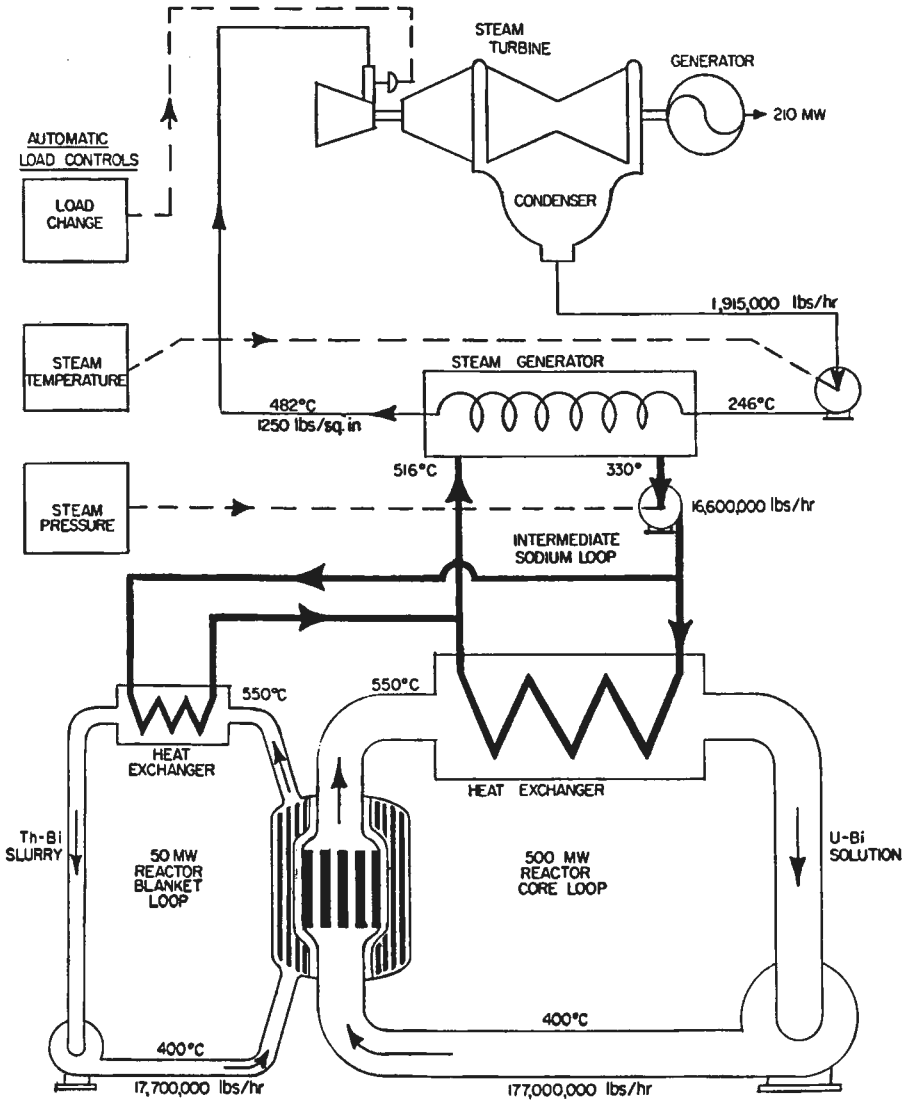

Figure 3. Heat cycle

equipment. This loop, consisting of a 36 in. pipe of 2¼ Croloy, forms a closed loop from the reactor core through the uranium-bismuth to sodium primary heat exchanger to the degasser, to the fuel circulating pumps and back into the core. Two 26 in. lines from the chamber which re-enter the 36 in. return line in a wye-connection, contain the main circulating fuel pumps. This piping is designed to a minimum volume (consistent with allowable pressure drop) for fuel inventory considerations. The U-Bi hold-up in the fuel system is approximately 867 cubic feet (25,000 liters). High temperatures in the loop which create stress conditions and expansion problems necessitate expansion joints and a minimum length for the hot leg. Welded construction is necessary throughout the entire loop, which must pass a test for tightness using helium mass spectrograph.

HEAT EXCHANGERS

The 36 in. line from the loop of the reactor connects with the U-Bi solution heat exchanger where the heat is transferred to sodium. Inlet U-Bi temperature is 550°C and exit temperature, 400°C. The heat exchanger is of U-shaped construction fabricated of 2¼ Croloy and consists of ¾ in. (1.8 cm) tubes.

The breeder blanket system contains 746 cubic feet (231 tons) of bismuth containing 10% by weight thorium in the form of a suspension of Th_3Bi_5. The neutrons leaking from the chain reacting core produce heat in the blanket through capture of the thorium by fission of some of the U^{233} formed in the blanket. This plus the heat due to gamma rays and other radiations absorbed in the blanket amounts to about 50 Mw of heat. This is removed by pumping the blanket fluid through a closed loop of heat exchanger and processing plant in the same manner as the main reactor loop.

Fuel pumping requirements for the main reactor loop indicate a maximum flow of 36,000 gpm (137,500 liter/min) corresponding to a velocity of 11 feet per second (3.3 m/sec). Two pumps each of 18,000 gpm (70,000 liter/min) capacity and each

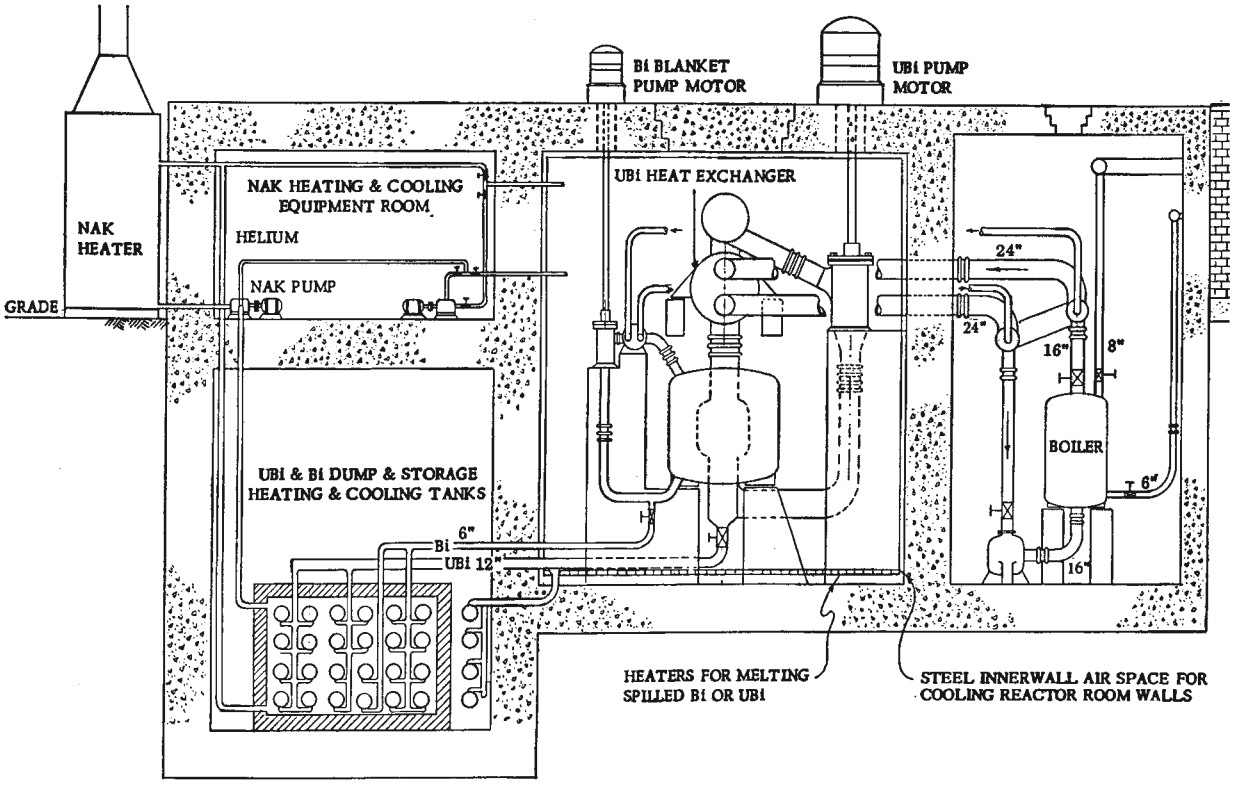

Figure 4. Liquid metal fuel reactor power plant, sectional view

driven by a 3000-hp motor will be located in the low-temperature, 26 in. (72 cm) lines emerging from the gaseous removal chamber. Similarly, the breeder blanket loop will require two 1800 gpm (7000 liter/min) units each driven by a 400-hp motor. These pump motors as shown are a canned motor, gas sealed shaft type which appear to be most feasible for this service at present. However, both canned rotor and electromagnetic pump are under consideration and if the electromagnetic pumps are developed in time they would be used as they require no opening of piping for maintenance.

The material selected to contain the liquid metals (U-Bi, Na and NaK) is 2¼ Cr-1 Mo steel (Croloy 2¼) for all vessels and transport lines that are in contact with these metals. Croloy 2¼ is the most promising of the alloys tested for service at these temperatures. Research indicates that it satisfactorily resists corrosion of Na and NaK at 600°C provided oxygen does not exceed 0.02%. As already indicated Croloy 2¼ is satisfactory for U-Bi when 350 to 500 ppm Zr and Mg are added to the fuel solution.

STEAM CYCLE

A 1250 psi (88 kg/cm²) 482°C steam cycle was chosen to permit the use of a conservative design of conventional turbine-generator equipment. The 482°C steam temperature provides an ample temperature differential margin below the 550°C discharge temperature of the U-Bi to permit the use of relatively small U-Bi to Na and Na to steam heat exchangers. This arrangement provides full steam temperature to below half load operation with simplified controls with no operational controls for the reactor.

Because of the available temperature differential of about 34°C between the U-Bi and Na, the heat exchanger is relatively small, making it feasible to use a single unit. A single U-Bi to Na heat exchanger is desirable since such a unit should operate as an integral part of the reactor. The single unit provides a simplification of piping and valve arrangement as well as reducing the inventory of the costly U-Bi.

The steam generators are of the forced flow once-through type. The once-through flow provides a very simple design principle of boiler-superheater arrangement. It, in effect, provides a variable surface relationship between the boiler and superheater which serves to maintain constant steam temperature over a large load range. That is, at high load when the temperature spread between reactor discharge and return temperature is at maximum, superheater surface, with respect to boiler surface, is a minimum. For lesser loads, the superheater to boiler surface relationship increases, thus making it easier to maintain full steam temperature for low loads.

Four steam generators connected in parallel are used so as to provide reasonably sized units. They are designed to provide 482°C steam at 1250 psi (88 kg/cm²). The rate at which sodium can be pumped through each generator can be varied. This provides more flexibility in operation and maintenance. The

water side of the steam generators has a conventional feed pump supplying water through a series of continuous tubes in which boiling water is changed to superheated steam. This type of boiler has no conventional drum for separation of steam and water. Orifices will be required in each tube to assure proper distribution of water flow among the tubes. For partial loading the number of steam generators may be reduced.

TURBINE GENERATOR

On the basis of 550 Mw total heat liberation in the reactor core and blanket, the heat cycle has been designed to meet the steam requirements of a 3600 rpm turbine generator unit with a guaranteed rating of 210 Mw. Steam conditions at this rating would be 1250 psig (88 kg/cm^2), 482°C and 1.5 in. (3.8 cm) Hg exhaust pressure. The turbine-generator is a tandem-compound, double-flow, unit with the last few stages having buckets grooved for removal of moisture. Steam flow to the 210 Mw unit will be 1,915,000 lb/hr.

A conventional steam cycle, employing six stages of feedwater heating and a deaerating condenser will preheat feedwater to 246°C. The surface condenser would have 145,000 ft^2 (13,500 m^2) of surface of single-pass design using two vertical condensate pumps and two vertical circulating water pumps.

Because of conditions peculiar to liquid metal fuel system operation, the reactor complex requires several unique auxiliary systems, the NaK system, the helium system, and the cooling system for the reactor room. The NaK system maintains the temperature of the uranium-bismuth solution and sodium systems, within definite limits to prevent freezing of the metals, and also provides for emergency cooling of the U-Bi. The NaK system was selected as a cheap and practical arrangement for maintaining the system temperature within the specified limits of the liquid metal systems. Jacketed lines have long been used in the chemical industry for various purposes which are similar to the liquid metal arrangement used with the LMFR system. The helium system is devised to pressurize the U-Bi, sodium and NaK circuits, and to preheat the reactor core whenever it is recharged. The cooling system utilizes closed air circulation to remove sufficient heat from the concrete reactor room walls to prevent dehydration of the concrete with accompanying deterioration of its shielding effect. Air is circulated between the steel lining and concrete walls and cooled externally with spray water. This closed system of air circulation is used to prevent the possibility of discharging radioactive air to the atmosphere. The steel lining serves the additional purpose of protection of the concrete walls from contamination of U-Bi should a leak occur. In this event the steel lining will be relatively easy to decontaminate or remove.

NEUTRON ECONOMY

Table I shows the ultimate destination of neutrons produced by fission of U^{233}. Because of the low capture cross sections of bismuth and carbon the losses to these materials are small. Continuous removal of fission products, by processes described below, maintains a low level of neutron losses to these poisons. As a result of this it is possible to obtain a breeding ratio of 1.05, even though with thermal neutrons the value of η is only 2.31 neutrons produced per neutron absorbed by U^{233}. This positive breeding gain is of considerable significance to the operation of the LMFR, which can be operated independently of any isotope separation plant, given enough U^{235} to produce one charge of U^{233} from the blanket. This results in low fuel costs as the only raw material required is thorium.

Table I. Neutron Economy in LMFR

				Neutrons per neutron captured in U^{233} in core
In core:	Neutrons absorbed in U^{233}			1.00
	Neutrons produced from fission, η			2.31
	Neutrons absorbed in:	Bi	0.09	
		C	0.03	
		fission products	0.05	
	Wasteful captures, a		= 0.17	
	Total absorptions in core, $1 + a$		= 1.17	
	Neutrons leaking to blanket, $l = \eta - 1 - a$		= 1.14	
In blanket:	Neutrons absorbed by Th			1.12
	Neutrons absorbed by:	Bi	0.04	
		C	0.02	
		Fe	0.02	
	Wasteful captures			0.08
	Neutrons absorbed in Pa^{233}			0.01
	Neutrons absorbed in U^{233}			0.06
	Total absorptions in blanket			1.27
	Neutrons produced by fission, 0.06×2.31		= 0.13	

Breeding ratio, $g = 1.12 - 0.01 - 0.06 = 1.05$ atoms of U^{233} produced per atom of U^{233} destroyed.

FUEL PROCESSING

The breeding gain shown in Table I will be obtained only if the concentration of fission product poisons is kept low by rapid processing. One advantage of any fluid fuel reactor is that the fuel can be circulated and processed continuously or in small batches. There need be no shut-downs or costly handling equipment for changing rods in order to add fuel to make up for that consumed. The processing is an integral part of this reactor.

The fuel is processed in three different ways. The volatile fission products are removed continuously from the main fuel stream. This is accomplished by passing the fuel stream into an evacuated chamber in a fountain which allows the Xe and other volatile fission products to escape. The bismuth and polonium vapors are condensed in a water-cooled condenser. If there is a demand for the polonium it can be distilled from the bismuth; otherwise it can be remelted and returned to the fuel stream. The fission products are absorbed on charcoal and may be used as a radiation source. Alternately the sparging gas is reused or allowed to decay until it can be discharged to the atmosphere.

The non-volatile fission products fall into two groups. The first includes Sr, Ce and the rare earths which form the worst fission product poisons. This group can be removed by extraction into a fused salt.[5] In order to keep their poisoning effect below 1% they will be processed continuously at a rate that holds them down to 15 ppm. A small stream of fuel, about ½ gpm (2 liter/min), is diverted from the reactor as shown in Fig. 5, and circulated through a counter current column where the fused salt stream, consisting of the eutectic mixture NaCl, KCl, and $MgCl_2$, reduces the fission product concentration to around 5 ppm.

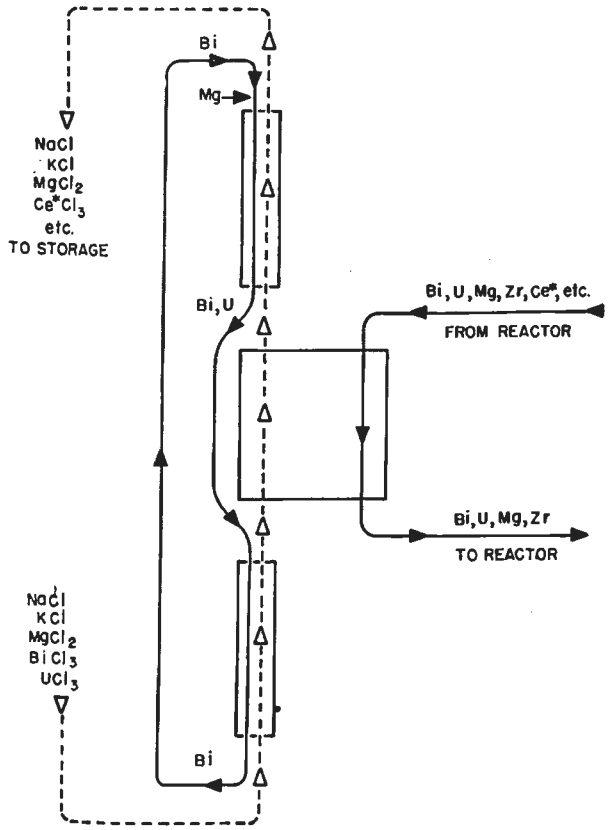

Figure 5. Fuel process

Most of the uranium stays in the fuel stream, but a small amount transfers to the salt. This is scrubbed from the salt stream in the upper column where a small stream of Bi containing Mg reduces the U from the salt into the Bi. This Bi stream flows to the bottom column where the U is returned to the salt by oxidation with $BiCl_3$. In order to add U to the fuel stream to make up that consumed in the reactor, the salt stream carries UCl_3 into the processing column.

The flows of salt amount to around 1½ gpm (5½ liter/min) and the Bi scrubbing stream around 1½ gph (5½ liter/hr). It can be seen from the flows involved that the equipment will be small, the columns around 2 or 3 inches (5 to 6 cm) in diameter and not more than 3 feet high (92 cm).

The second group of non-volatile fission products cannot be separated from uranium by the above process. These fission products have much smaller cross sections on the average and their poisoning effect builds up much more slowly. Therefore, it only is necessary to process at about 1/20 the rate required for the salt soluble fission products. This could be done by a batchwise process in which these elements were removed from the Bi with the uranium and then separated in a small process. This also would clean out any corrosion products which would have some slight poisoning effect if allowed to accumulate indefinitely.

The blanket consists of Th_3Bi_5 suspended in liquid Bi.[6] If the concentration of uranium formed in the blanket is kept below 100 ppm, the power generated by fission in the blanket will be less than 10% of the total. This concentration level can be maintained if the blanket fluid will be processed at a rate of 20 tons (18,000 kg) per day. The processing cycle is shown in Fig. 6. A stream of about 20 gph (75 liter/hr) of Bi containing 10% Th as Th_3Bi_5 is diverted from the blanket cooling system. This is first diluted with an equal amount of pure hot Bi. In order to get the U^{233} and Pa into solution without completely dissolving the Th_3Bi_5 particles, this diluted mixture is cycled thermally between 850°C and 350°C. During this treatment 60% of the solids go in to solution and reform on the undissolved portion of the crystalites allowing about 30% of the U^{233} + Pa to stay in solution in the Bi. The mixture goes into a phase separator which concentrates all the Th_3Bi_5 particles into one stream to return to the reactor. The other liquid contains in solution 50 ppm of Th and 30 ppm of U^{233} and Pa which are extracted from the Bi. The latter is purified and re-

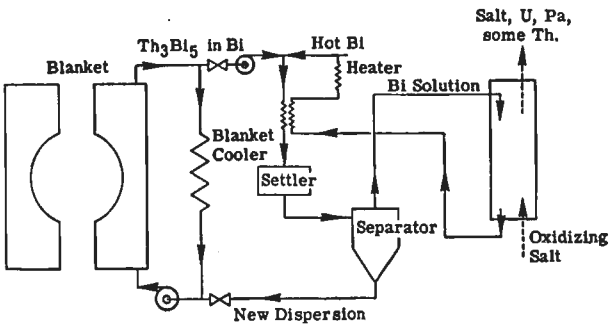

Figure 6. Blanket process

cycled to the processing stream. The Pa is allowed to decay to U^{233} and is available for use as a reactor fuel. Again the quantity and equipment size are small. For a 500 Mw core power about 550 gm of U^{233} and Pa will be recovered, per day.

A more complete discussion of the chemistry involved in the processing has been published previously.[5] Further details are also found in an article "High Temperature Processing Systems for Liquid Metal Fuels and Breeder Blankets" by Dwyer, Teitel and Wiswall which has been submitted to this conference.

REACTOR KINETICS

Fluid fuel reactors whose fuels expand on heating have large negative temperature coefficients of reactivity. As a result the reactor is very stable and adjusts quickly to changes in reactivity and load. If the reactivity of the core is increased by adding uranium to the bismuth, the average core temperature will rise until a new steady temperature corresponding to the new loading is reached. For example, a sudden addition to the fuel inventory of enough U^{233} to raise the concentration by 2.75% and give 1% excess reactivity, will cause the reactor to stabilize at an average temperature 55°C higher than before.

The fuel concentration thus fixes the steady state average temperature, while the inlet and outlet temperatures adjust themselves to meet the power demand. A sudden increase in load causes a drop in the inlet temperature and an equal rise in the outlet temperature keeping the average unchanged.

PLANT CONTROL

Although for greatest economy the nuclear power plant should be base loaded at its maximum capability, the plant is designed to operate over the range from full capacity down to approximately 10% load. The response of the equipment for load changes will be equivalent to that of a conventional unit. Actually, the reactor is capable of responding somewhat faster than the conventional related equipment. The operation of this nuclear plant may be completely automatic or by remote manual control similar to the method followed by conventional power plants.

The rate of heat release in the LMFR is determined by the temperature rise of the fuel across the reactor and the rate of flow through the reactor. The concentration of U^{233} in the fuel controls the average temperature, and, for any given power demand, the inlet and exit temperatures. Therefore, the concentration of the U^{233} must be sufficiently high to provide a reactor heat level that will assure a minimum temperature of the returned fuel. This temperature should be at least 370°C to prevent precipitation of the uranium and possible freezing of the bismuth. At the same time the load conditions of the 210 Mw power unit selected for this application must be met. That is, the U^{233} concentration is established to provide an average temperature of the reactor of 475°C. Then at a fixed fuel flow rate, the reactor discharge and return temperatures are established for all loads from zero to 100%. (Actually because of the heat released by fission product decay, a sudden shutdown only drops the power to about 10% of the previous level after which the power level drops off more slowly. Therefore means for cooling the reactor at all times must be available.)

These load-temperature characteristics provide features that can be utilized to make the reactor self-controlled automatically for all load ranges. That is, a decrease in power demand decreases the steam flow and corresponding reduction in heat absorption from the sodium circuit, which in turn, reduces the heat absorption of the fuel circuit. This reduced heat absorption of the fuel circuit results in a narrowing range of the discharge to return temperatures to maintain the 475°C average temperature.

The operation controls for the LMFR power unit are comparatively simple. As for any power plant, there are several different arrangements of controls that are satisfactory. The selected arrangement will depend on specific requirements and perhaps somewhat on the personal preference of the operating personnel. The arrangement selected for this application, illustrated by Fig. 3 provides complete flexibility in operation, minimum metal temperature of the high pressure superheater, and high thermal efficiency for the full load range.

The controls function as follows:

1. Steam temperature is maintained by the feedwater flow by means of a feed control valve or by variable speed pumps. The control elements are: a. steam temperature—for final adjustment, and b. steam flow (representing load)—anticipating element.

2. Steam pressure is maintained by the heat absorption in the steam generators and is controlled by the sodium heat flow through the generators. The control elements are: a. Steam pressure—final adjustment, b. steam flow—anticipating element, c. sodium flow times temperature difference—representing heat flow for stabilization element.

3. Reactor heat release is self-correcting as described above.

Safety controls differ from but are comparable to

those of conventional power plants. They operate in the following manner:

1. Failure of the boiler feed pumps opens the sodium by-pass control valve. The fuel and sodium circulation pumps are kept operating but under no load condition. The reactor will attempt to maintain its heat level of 475°C.

2. Failure of the sodium pumps results in an automatic dumping of the load due to loss in steam generation. The feedwater flow is automatically shut off by the steam temperature control mechanism.

3. Temperature limits of the reactor will reduce the load as follows: a. 595°C fuel discharge temperature opens the sodium by-pass control valve, b. 370°C fuel return temperature opens the sodium by-pass control valve, c. fuel temperature differential of 167°C opens the sodium by-pass control valve.

4. If the operation of the above safety equipment does not function properly, or if the residual heat of the reactor is excessive, the fuel is dumped to the storage tanks by remote manual operation of the dump valve. The storage tanks are provided with an external source of combination heating or cooling to maintain fuel temperature at the allowable limits.

ECONOMICS

While a detailed cost estimate of this plant has not been made, rough estimates on this and similar LMFR designs indicate that the entire capital investment will run around $50,000,000. The best guesses as to the operating and maintenance costs for the power plant are around $1,000,000/year. Because this system is a breeder the fuel cost is negligibly small. In its place there is the cost of operating the chemical processing plants for the fuel and blanket. These costs are estimated to be $2,000,000 for this system per year. Assuming a 16% annual charge on the capital investment and an 0.8 use factor, we would get the cost of electricity at 7.8 mills per kwh. While there are other ways of figuring proper charges on the capital investment of the plant, these operating costs are rather conservative in that it is believed that with some experience a small staff would be required to run the chemical processing plants, reducing this charge appreciably. However, until such a plant is actually built it is doubtful as to whether any really significant changes in the estimates of the costs will be possible.

STATUS

The present status of this plant is the following. Most of the basic chemical and metallurgy problems have been solved. Many small loops have been operated outside of neutron radiation fields. At present one chemical processing loop is being operated at a flux of 10^{10} neutrons per cm^2 per sec, in the Brookhaven reactor. Engineering developments have been started on pumps, heat exchangers, and chemical processing columns. Exponential assemblies of uranium, bismuth, and graphite will be made beginning sometime in 1956. It is planned to push the testing of the various large scale components so that in late 1957 or 1958 construction can start on some form of experimental or prototype reactor. If it were considered desirable to operate such a reactor as a U^{235} burner without breeding it might be possible to achieve reasonably significant power levels, 10 to 50 Mw, at this time. It is believed, however, that to achieve large scale plants such as we have described with breeding ratio approaching 1 it will take several years after the operation of the reactor experiment, that is, sometime in the early 1960's. However, no obstacle of a fundamental nature has yet appeared that stands in the way of eventually operating this reactor and achieving the thermodynamic efficiency, economy and safety of operation possible with a continuously processed liquid metal fuel.

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A Developmental Fast Neutron Breeder Reactor

By A. Amorosi,* A. P. Donnell,* and H. A. Wagner,† USA

A detailed proposal for a 100,000 kilowatt (electric) fast neutron breeder reactor has been prepared by Atomic Power Development Associates, Inc. A group of private companies, including some of the APDA, Inc. member companies (see Appendix, Table I), has submitted this proposal to the United States Atomic Energy Commission in connection with the AEC Reactor Demonstration Program. This represents a reactor steam-electric plant which these companies are willing to design now, to construct with private funds, and to own and operate the completed plant. This paper contains a brief discussion of the background and reasons for the choice of the fast breeder reactor as a most promising one for economic atomic power, and a general description of the reactor design.

APDA, INC.

Atomic Power Development Associates, Inc., consists of a group of twenty-five electric power systems, four manufacturing enterprises, and four engineering organizations (see Appendix, Table II). This is a non-profit membership corporation, the objective of which is to apply the knowledge and experience of its members and associates to develop atomic power into a commercially practicable means of power generation. This group had its origin as a proposal made by The Dow Chemical Company and The Detroit Edison Company to the United States Atomic Energy Commission late in 1950. It was proposed that industry be permitted to study the feasibility of using nuclear heat in the operation of steam-electric power plants.

In March 1951, an agreement was signed by Dow and Detroit Edison with the AEC to undertake a study of the practicability of using nuclear power for industrial purposes. The main objective was to consider whether it was feasible and economic to build a commercial-size reactor to produce power and fissionable materials as joint products.

RESULTS OF INITIAL STUDY

The report containing the results of the first part of the study, submitted in December 1951, concluded that the use of nuclear heat for industrial processes and the production of steam for thermal-electric power plants was feasible. Certain design objectives and specifications were outlined which were considered essential in the successful development of a commercial power reactor. These were as follows:

1. The reactor should utilize low-cost fuel, i.e., it should be a breeder reactor which within itself converts thorium or depleted uranium to fissionable material for fuel. The fast reactor is preferred because it provides a higher margin of excess neutrons. (Throughout this discussion, breeder reactor is taken to mean a machine which produces more fissionable material than it burns, whether it is the same or a different material.)
2. The reactor should be a high-temperature machine to permit a high efficiency in the production of power.
3. The reactor should require a small enough exclusion area so that it can be located close to the power demand.
4. The reactor should be of a design to permit integration with a metallurgical type extraction process to make possible fast low-cost separation of fissionable materials and fission products.
5. The reactor should use a fuel which is easy to load into the reactor. It appears that the most reliable and lowest cost reactor will prove to be one using a mobile or fluid fuel.
6. The reactor should be inherently self-regulating. It is essential that the design will be such that the reactor will inherently shut itself down before a reactor incident becomes serious.

CHOICE OF REACTOR

It was felt that the foregoing specifications would best be met by a liquid-metal-cooled fast neutron breeder reactor. Dow-Detroit Edison proposed that a research and development effort on such a reactor be undertaken jointly with the US AEC to determine its engineering practicability and whether or not it was commercially economic. The proposal was accepted in substance by the AEC in April 1952.

Since that time, work has proceeded from the study through the preliminary design state, including a substantial research and development effort. Prior to this year, about $4,000,000 of private money has been spent. This year, $3,815,000 has been budgeted for research and development work. This effort has resulted in the proposal to build and operate a devel-

*Atomic Power Development Associates, Inc., 2000 Second Avenue, Detroit 26, Michigan.
†The Detroit Edison Company.

opmental fast neutron breeder reactor referred to above.

GENERAL CHARACTERISTICS OF FAST REACTORS

The following are features which characterize the sodium-cooled fast neutron reactors:

a. As expressed in other papers, the fast reactor operates at a neutron energy above 0.1 Mev. It has no moderator. Because of the relative cross-section of uranium-238 and uranium-235 at this energy, it must operate on partially enriched fuel. The core is essentially uranium and coolant and is, therefore, small.

b. Reactivity changes are primarily caused by the change in neutron leakage accompanying core expansion, particularly fuel expansion. The reactor is required to accommodate only very small reactivity change in normal operation. The build-up of fission products has a negligible effect on reactivity.

c. The prompt neutron lifetime is short—of the order of 10^{-7} seconds. However, because of the small reactivity to be accommodated, it appears that the reactor can always be operated within the delayed neutron fraction. The delay fraction is greatest for uranium-235 of the possible fuels, so that extremely short periods are avoided if uranium-235 fuel is used.

d. High η and ϵ (fast fission of uranium-238); therefore, a good conversion ratio can be obtained even with uranium-235 fuel.

e. Low neutron absorption for most material which permits use of as much as 20 per cent by volume of a wide variety of structural materials without serious loss of neutron efficiency. For the same reason, fission product build-up is not serious (so that large burn-up of the fuel can be tolerated) from the standpoint of neutron economy.

f. Because of its small size and high power density, the fuel must be finely subdivided to limit the temperature gradient across it. As a result, coolant passages are quite small, around ⅛ inch diameter. Growth and deposition may tend to plug these passages. Plugging due to mass transport does not seem to be a problem, since the dissolved metal seems to preferentially deposit on cold surfaces—not hot. Care must be exercised to prevent particle accumulation in the small passages.

In some respects, the small reactor size appears to be a help—not a problem. With a reasonable coolant velocity of 30 feet per second, and a ΔT of 300°F, a three-foot diameter reactor will give 500 megawatts of heat, a desirable power output. It is also about three feet long. A greater length would not help the heat transport problem. This small size greatly simplifies the structural design and unloading problem.

g. The use of sodium coolant permits operation at high temperature without high pressure, providing an opportunity for a relatively high steam cycle efficiency.

h. Many structural materials are corrosion resistant in sodium. Low carbon iron, chromium iron, stainless steel, nickel, and inconel are all corrosion resistant at the temperature of interest, providing the oxygen content is kept down. As a result, it is expected that the system equipment will not be very expensive.

i. In many ways, the most serious problem in a fast reactor is the fuel reprocessing charge. An example will illustrate the problem. In a thermal reactor, perhaps only 1 to 2 per cent of the fuel element is fissionable material, and the conversion ratio is above 0.7. It is envisioned that, if 1 per cent burn-up is achieved, a large fraction of the initial fuel will be burned before reprocessing.

In the case of a fast reactor, about 25 per cent of the fuel element is fissionable material, and the internal conversion ratio is under 0.5. If 2 per cent burn-up is achieved, only 8 per cent of the fuel is burned before reprocessing. Thus, since a fuel atom may be reprocessed 12 times before it is burned, there may be high inventory and loss charges associated with reprocessing. However, there are some compensating circumstances. Fission products can be allowed to build up to high levels without seriously affecting neutron absorption so that high burn-up and crude reprocessing can be tolerated. The requirements here, too, are less stringent since the corrosion resistance of uranium in sodium permits the use of a less perfect job of cladding than is necessary in water-cooled reactors.

These factors contribute to the possibility of using simple methods of reprocessing and refabrication. It is envisioned that pyrometallurgical reprocessing consisting of a simple melting operation with oxide slagging by the crucible will be used. It is expected that this process will remove 80 per cent of the fission products. This will be followed by a simple radioactively hot fabrication. In the distant future, it is expected these reactors will be operated with mobile fuel as discussed later.

j. As stated in the foregoing, the reactor has a poor internal conversion ratio. To avoid handling large reactivity changes, portions of the core should be changed at frequent intervals—about once every two to four weeks—even though the average core life is about three to six months.

APDA, INC. FAST REACTOR

With these general remarks on fast reactors, the APDA version will be described to give an illustration of some of the design features and problems of fast reactors. A tabulation of the design conditions of the reactor is given in Table III.

The reactor is designed to operate on uranium-235 fuel, not plutonium, because of (1) safety, (2) availability of materials, and (3) lack of plutonium technology. Operation of this plant, with results from the Argonne National Laboratory's Experimental Breeder Reactor No. 2, are expected to yield information which will permit construction of a full-scale

Table III. Performance Data Tabulation

Plant capacity	
Gross electric capacity, Mw	100
Net electric capacity, Mw	90
Turbine-generator rating, kw	100,000
Reactor specifications	
Power (heat), kw	300,000
Core diameter, ft	~3
Core length, ft	~3
Over-all height of reactor vessel, ft	~14
Outside diameter of reactor vessel, ft	~10
Thickness of reactor vessel, inches	~1
Core sodium flow rate, lb/hr	11.22×10^6
Sodium velocity, ft/sec	27.6
Liquid metals and steam systems	
Net thermal efficiency, %	30.0
Sodium temperatures, °F	
Leaving reactor	800
Entering reactor	550
Sodium flow, lb/hr	13.2×10^6
NaK temperatures, °F	
Entering boiler	750
Leaving boiler	500
NaK flow, lb/hr	16.1×10^6
Steam pressure, psig	600
Steam temperature, °F	730
Feedwater temperature, °F	400
Steam flow, lb/hr	1×10^6

plutonium breeder. The reactor which is described will produce about 300 megawatts of heat with a corresponding gross electrical output of 100 megawatts. In size, it will be about three feet in diameter by three feet high, surrounded by a two-foot blanket of depleted uranium. A perspective view of the reactor is given in Fig. 1.

CORE DESIGN

The reactor core, Fig. 2, is an assembly of partially enriched uranium alloy plates. Plutonium is produced in the reactor core and in a surrounding breeder blanket of depleted uranium rods. Insertion of boron-containing poison rods provides shim and safety control. The core consists of numerous square fuel-element sub-assemblies arranged to approximate a right cylinder. These sub-assemblies contain the fuel plates and axial blanket elements, Fig. 3. The radial sub-assemblies are the same size as the core sub-assemblies. They contain cylindrical rods of depleted uranium, Fig. 4. Both the core and the blanket are cooled by sodium that is pumped into the bottom of the reactor vessel, which flows upward through these sections and flows out near the top of the reactor vessel. By using up-flow through the core, decay heat can be removed by natural circulation.

CONTROL

Shim control is provided by a boron control rod located at the center of the reactor. Very little reactivity must be accommodated by this control rod so the loss in neutrons is very small. Moving poison was chosen instead of moving fuel because the heat dissipation requirements are much less and because much more reactivity can be controlled per unit of volume. The control rod is, therefore, lighter, permitting use of a relatively inexpensive drive. The safety or shutdown rods are also made of boron. Although these rods contain some 6 per cent Δk, they do not absorb neutrons in normal operation because, under operating conditions, they are located out of the core at the upper edge of the top blanket. They are scrammed by gravity with an initial spring assist.

MECHANICAL HANDLING

Unloading of the core and blanket is accomplished by an element handling mechanism, Fig. 5. It consists of a rotating shield plug and an offset handling crane mounted in the plug. A hold-down plate is suspended below the plug which maintains radial alignment of the core sub-assemblies and holds them down against the pressure drop forces caused by flow through the

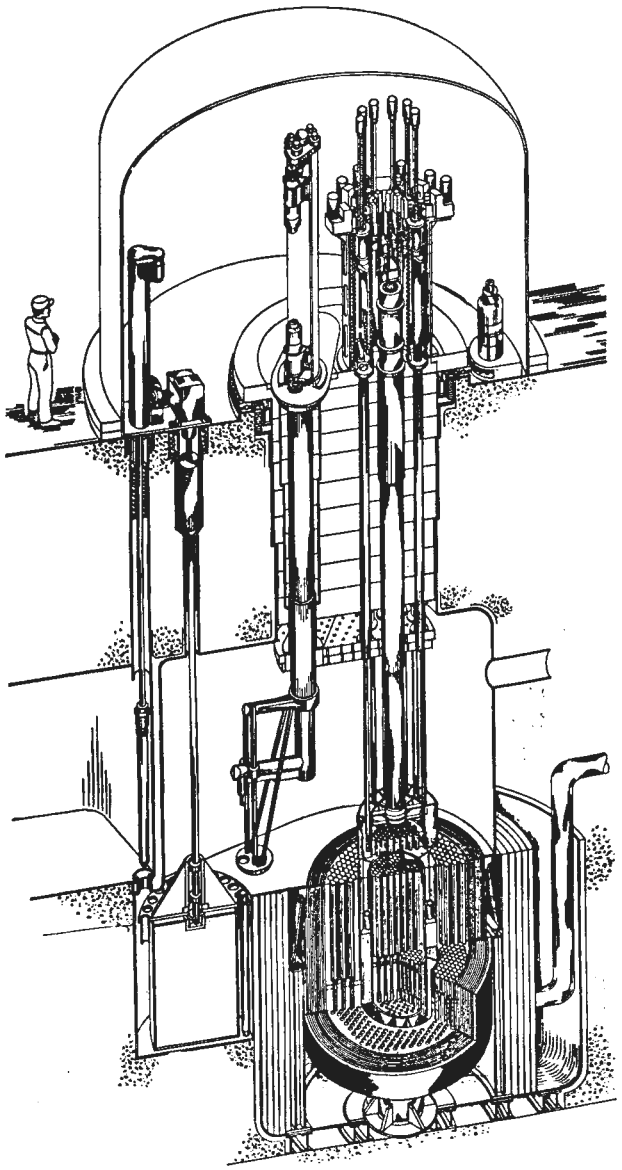


Figure 1. Perspective view of reactor

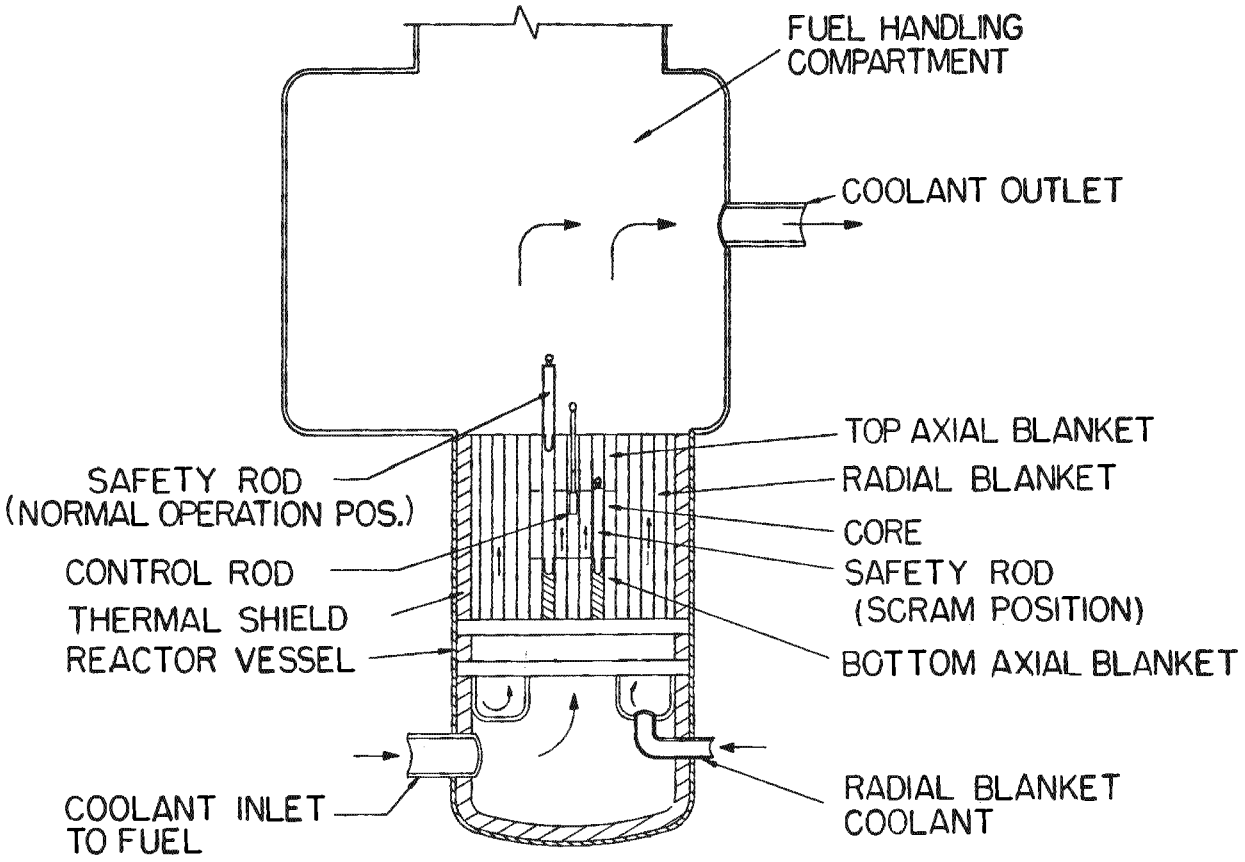


Figure 2. Reactor arrangement

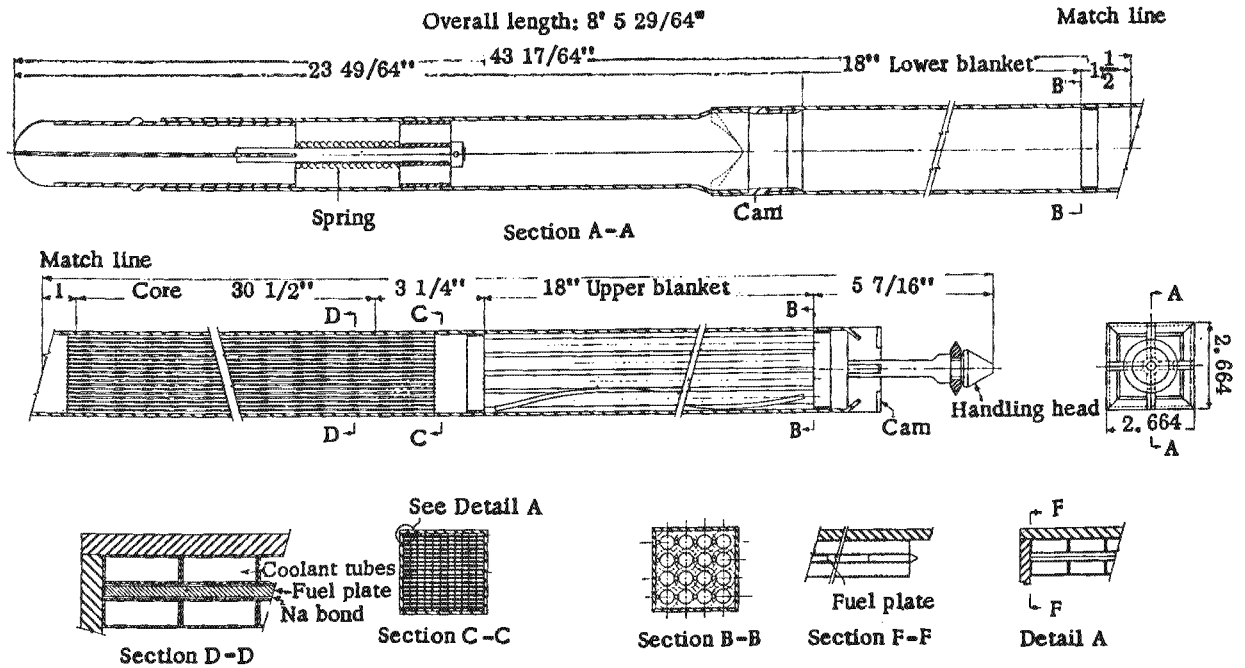


Figure 3. Flat plate sub-assembly

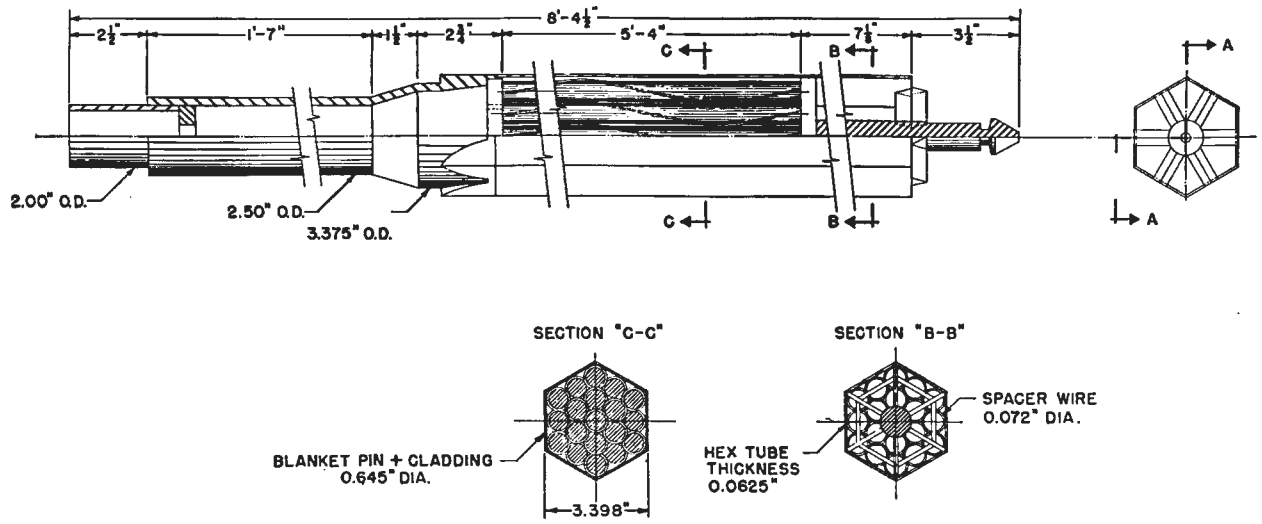


Figure 4. Radial blanket sub-assembly

elements. It also acts as a guide for the control element drives.

The sodium level is maintained some eleven feet above the core as indicated in Fig. 5. This is done to: (a) keep the fuel elements under sodium during unloading to dissipate decay heat, and (b) provide natural circulation cooling of the core itself during unloading or on pump failure.

HEAT TRANSPORT AND UTILIZATION SYSTEM

Heat is removed from the reactor core and blanket by circulating sodium. The heat is transferred to an intermediate sodium-potassium alloy system and is then transferred to water and steam in a once-through-type steam generator, a schematic arrangement of which is shown in Fig. 6. Steam is utilized in a conventional steam turbine that is directly connected to an electric generator. The intermediate circuit is used to prevent a NaK-water accident from releasing radioactivity. An elevation of the plant is shown in Fig. 7. The only equipment in the primary system which may require removal for inspection and repair are the pumps and the tube bundles of the heat exchangers. These are made as sump type units with a free surface above the active portions and can be lifted out without cutting the piping and without draining the system. No isolating valves are used in the primary or secondary systems since they are considered more vulnerable than the piping, heat exchanger housing, and pump housing which they would be called on to isolate. The valves would also unnecessarily increase pressure drop which it is desirable to keep to a minimum since the coolant flows by gravity from the reactor tank to the heat exchanger and pump. A syphon break is provided on the piping at the reactor so that a rupture in the piping will not drain the reactor tank. Surrounding the reactor vessel is a so-called primary shield tank which provides containment for sodium in the event of a reactor vessel leak.

The large sodium pool above the reactor contains some 15,000 gallons of sodium and contributes the largest part of the 40-second cycle time for the sodium flow through the primary system. The pool minimizes thermal shocks to the piping system which tend to accrue during reactor scram.

Thermal shock to system parts has also been minimized by (*a*) providing inlet cooling to the high pressure (100 psi) parts of the reactor vessel; (*b*) limiting the rise through the reactor to 250°F; (*c*) designing the top pool and piping of thin walls, piping is ⅜ inch thick and the pool tank is ½ inch thick, and (*d*) practically eliminating need, and hence call for scram.

SHIELDING

The thermal shield composed of metal sheets is positioned against the inner wall of the reactor vessel, Fig. 5. Its purpose is to reduce the fast neutron flux reaching the reactor vessel in order to minimize the damage that results from neutron embrittlement and neutron-induced thermal stress.

Compartment shielding is used to provide flexibility in the design and location of the system components. The radial shielding close to the reactor is primarily neutron shielding to reduce induced activation of the secondary coolant system. It consists of two parts, primary neutron shielding of borated graphite, and secondary neutron shielding made of Borax (a trade name for $Na_2B_4O_7 \cdot 10\ H_2O$). The primary neutron shielding is located in the primary neutron shield tank so that it can be confined and cooled by forced gas circulation. The secondary neutron shielding does not require cooling. Borax can be used as this secondary shielding because it is not exposed to high neutron flux, high temperature, or sodium.

The space between the Borax shield and the building is the heat exchanger and pump machinery compartment. The personnel shielding around this compartment is to attenuate primary gammas and the

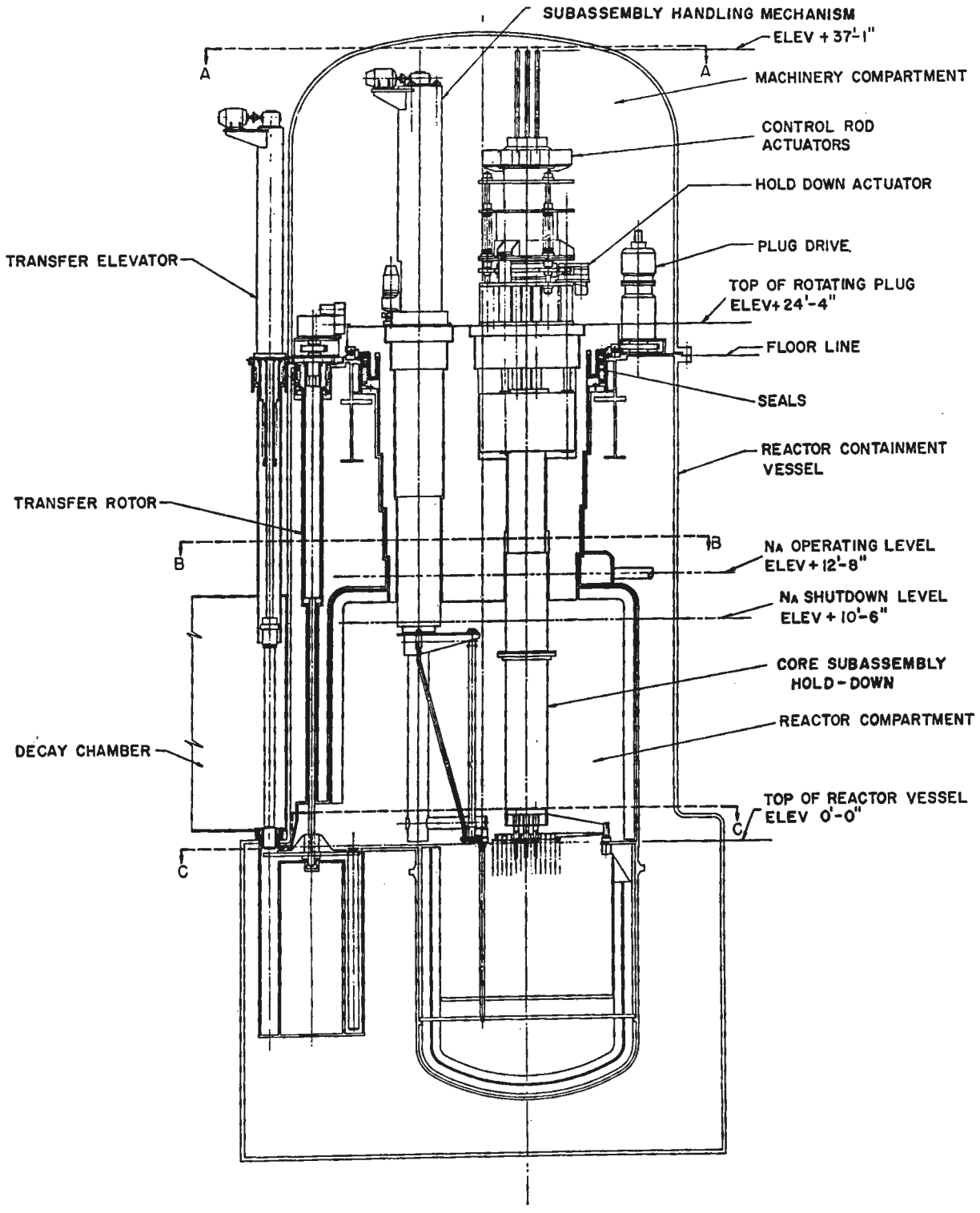


Figure 5. Element handling mechanism (elevation)

secondary gammas which are produced in the sodium. There is about 0.3 curie per cm^3 of radioactivity in the sodium, and it is mainly generated in the sodium pool above the core. Much of the shielding is located outside of the building where it acts as structural support and is cheaper to construct than if it were located inside of the building.

REACTOR SAFETY

Emphasis has been placed on reactor safety. There seems to be no positive temperature coefficient of reactivity. Uranium expansion alone seems to be fast enough to handle any conceivable fast reactivity transient. The operating temperature has been chosen low enough so that, even if all of the control rods are re-

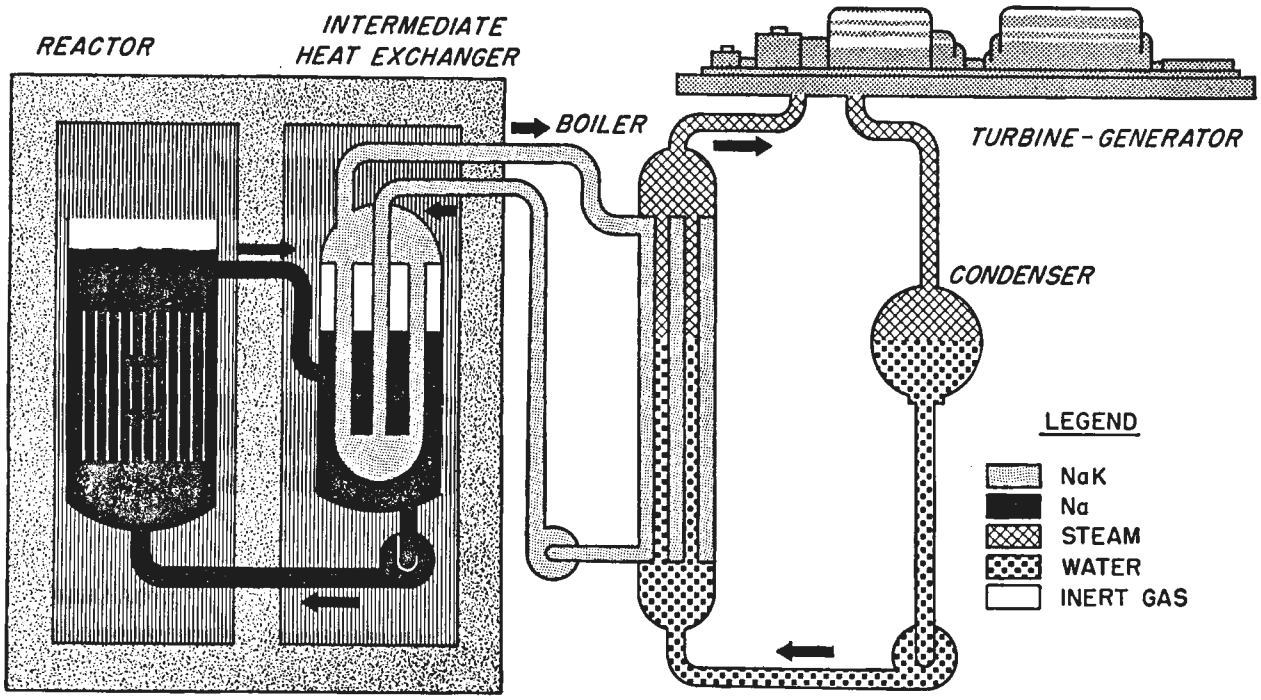


Figure 6. Schematic arrangement of plant

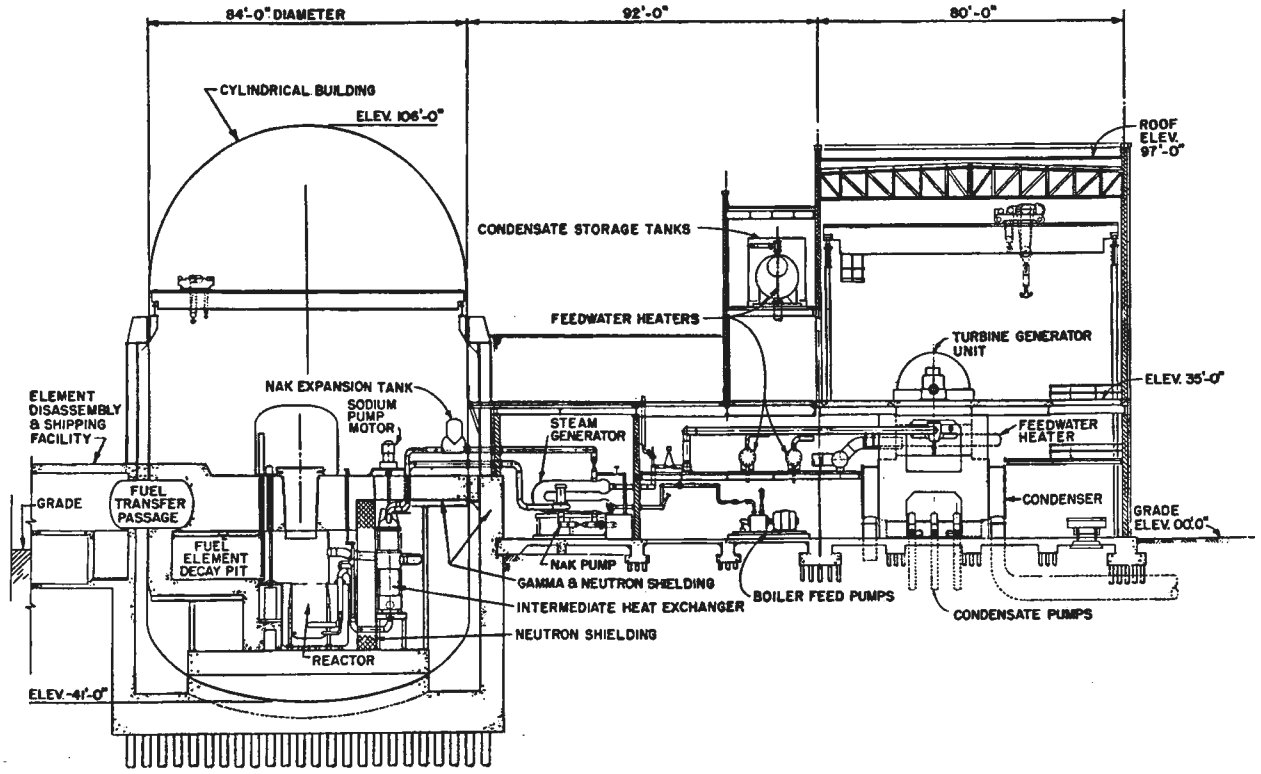


Figure 7. Elevation of plant

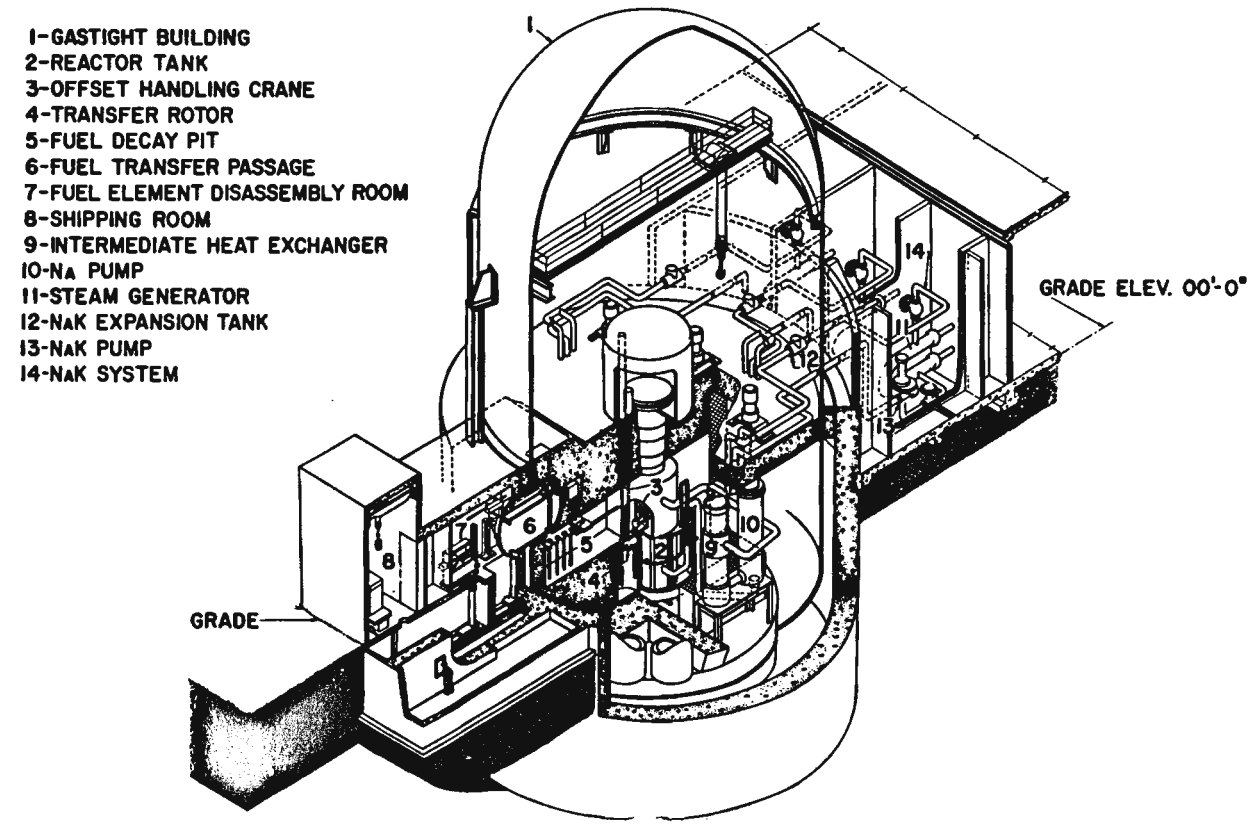


Figure 8. Perspective view of reactor building

moved, the reactor would shut off short of melting. A serious cold sodium slug accident is avoided by the low operating temperature, and by the fact that the system cannot be isolated. The reactor is unloaded at temperature so that little reactivity is required in the shim control. The number of fuel elements has been chosen so that the maximum reactivity of one element is less than that equal to the delayed neutron fraction. Therefore, if a fuel sub-assembly were accidentally dropped into the core when the reactor is critical, it would not get into a prompt period. The maximum speed of the control rods is 0.007 per cent Δk per second, so that during the start-up it cannot get into a prompt period. The fuel elements are also inserted at this rate so that, even if the core were accidentally critical during loading, the prompt period accident would be avoided.

CONTAINMENT

An airtight steel cylindrical reactor building, Fig. 8, encloses the reactor, the fuel handling mechanisms, the intermediate heat exchangers, and the sodium pumps, piping, and storage tanks. It is approximately 84 feet in diameter and has a wall thickness of about 1½ inches. The purpose of this building is to contain the products of any possible reactor accident resulting in a release of fission products and radioactive sodium. All air is de-humidified and enriched in nitrogen to minimize the likelihood of a sodium-oxygen reaction should a leak occur in the liquid metal system.

GENERAL DESIGN CONSIDERATIONS

Flexibility has been designed into the system. All mechanisms are removable. The rotating plug is removable in layers. The control drive, rods and thimbles are removable. The fuel element support plates can be removed. The fuel and blanket sub-assemblies are made the same size so that they are interchangeable. Therefore, the core size can be adjusted during initial start-up to achieve criticality. Furthermore, after a number of years of successful operation at design power, the core size can be increased to increase the power output.

The design of the reactor has been established and a full-scale mechanical test facility of the reactor and one primary loop is being constructed. This facility will not contain fissionable material. It is designed to prove out the mechanical and hydraulic operation of the reactor. This facility will be completed by the fall of 1956. The reactor plant is expected to go into operation in 1959.

There are many design features of the reactor which remain to be demonstrated, but there appear to be no problems for which a solution is not foreseeable.

ECONOMICS

Detailed estimates of the reactor cost, based on informal price quotations from equipment manufacturers have been made. The $29,100,000 reactor cost was established as follows:

a. Reactor plant, estimated cost, including erection	$17,750,000
b. Construction overhead (17 per cent of Item "a")	3,020,000
c. Engineering and drafting (8 per cent of Items "a" plus "b")	1,660,000
Subtotal	22,430,000
d. Contingencies (30 per cent of subtotal)	6,670,000
Total	$29,100,000

Approximately 70 per cent of reactor plant costs shown above were determined from manufacturers' estimates of the equipment involved. Our most recent estimates indicate that the cost would be somewhat less than $29,100,000. The foregoing does not include site costs, working capital, and preliminary operation. The actual amount of money budgeted for construction of this plant is considerably greater than shown above primarily to provide for development and design costs of first units of a kind. It may be observed that a generous contingency allowance has been included. Because of the conservatism in design and the large allowances provided in estimates for capital costs, it is not anticipated that this reactor plant will be competitive with modern thermal plants.

FAST REACTOR POTENTIAL

The future potential for the fast reactor is great. It appears possible to utilize a liquid fuel for the blanket, and eventually for the core. Use of a liquid fuel will tend to simplify processing problems and will enable continuous operation of the reactor without need for shutdowns for refueling. Research and development work is under way to achieve these goals, which are not being designed into this first reactor plant.

It is believed that, with the higher performance which will be obtained with the second and subsequent plants, coupled with lower capital costs, as a result of experience gained with this first plant, power costs competitive with present modern thermal power plants will be achieved.

APPENDIX

Table I. Companies* Proposing to Design, Construct, Own and Operate a Developmental Fast Neutron Breeder Reactor

Electric power systems
- Central Hudson Gas & Electric Corporation
- The Cincinnati Gas & Electric Company
- Consumers Power Company
- Delaware Power and Light Company
- The Detroit Edison Company
- Long Island Lighting Company
- Philadelphia Electric Company
- Potomac Electric Power Company
- Rochester Gas and Electric Corporation
- The Southern Company
  - Alabama Power Company
  - Georgia Power Company
  - Gulf Power Company
  - Mississippi Power Company
- The Toledo Edison Company
- Wisconsin Electric Power Company

Manufacturing enterprises
- Allis-Chalmers Manufacturing Company
- The Babcock & Wilcox Company
- Burroughs Corporation
- Holley Carburetor Company

*Those intending to participate as of 10 June 1955.

Table II. Atomic Power Development Associates, Inc. Membership

Electric power systems
- Atlantic City Electric Company
- Baltimore Gas and Electric Company
- Boston Edison Company
- Central Hudson Gas & Electric Corporation
- The Cincinnati Gas & Electric Company
- The Cleveland Electric Illuminating Company
- The Connecticut Light & Power Company
- Consolidated Edison Company of New York, Inc.
- Consumers Power Company
- The Detroit Edison Company
- General Public Utilities Corporation
  - Jersey Central Power & Light Company
  - Metropolitan Edison Company
  - New Jersey Power & Light Company
  - Pennsylvania Electric Corporation
- The Hartford Electric Light Company
- Long Island Lighting Company
- New England Gas and Electric Association Service Corporation
- New England Power Company
- New York State Electric & Gas Corporation
- Niagara Mohawk Power Corporation
- Philadelphia Electric Company
- Potomac Electric Power Company
- Public Service Electric and Gas Company
- Rochester Gas and Electric Corporation
- Southern Services, Inc.
  - Alabama Power Company
  - Georgia Power Company
  - Gulf Power Company
  - Mississippi Power Company
- The Toledo Edison Company
- Wisconsin Electric Power Company
- Wisconsin Power and Light Company

Manufacturing enterprises
- Allis-Chalmers Manufacturing Company
- The Babcock & Wilcox Company
- Bendix Aviation Corporation
- Ford Motor Company

Engineering organizations
- Commonwealth Associates, Inc.
- Jackson & Moreland
- United Engineers & Constructors Inc.
- Vitro Corporation of America

Record of Proceedings of Session 11A

SATURDAY MORNING, 13 AUGUST 1955

Chairman: Mr. O. Dahl (Norway)
Vice-Chairman: Mr. K. Wirtz (Germany)
Scientific Secretaries: Messrs. F. de Hoffmann and N. A. Dobrotine

PROGRAMME

P/333 Recovery of the energy produced in air cooled graphite reactor G 1 ..... P. Chambadal and M. Pascal

P/337 Design for a dual purpose reactor (G 2).....P. Ailleret *et al.*

P/492 A graphite moderated nuclear power plant design .....R. K. Andersen *et al.*

P/493 A sodium graphite reactor 75,000 electrical kilowatt power plant .....C. Starr

DISCUSSION

P/936 The design of a small prototype of a homogeneous power reactor fuelled with a uranium oxide suspension.....H. de Bruyn

P/938 A "dry" suspension of uranium oxide for a heterogeneous power reactor .....H. de Bruyn *et al.*

P/494 Liquid metal fuel reactor .....F. T. Miles and C. Williams

DISCUSSION

Mr. CHAMBADAL (France) presented paper P/333.

Mr. TARANGER (France) presented paper P/337 as follows: In its concern to harness atomic energy for peaceful purposes as quickly as possible, that is mainly in industrial power stations, the Atomic Energy Commission (*Commissariat à l' énergie atomique*) has decided in favor of large natural-uranium piles using graphite as moderator and a gas under pressure as coolant.

By concentrating the resources of France in this direction the ground for the G2 pile has been prepared by the completion successively of the Saclay pile, known as P2, and the G1 pile described yesterday by Mr. Yvon and just now by Mr. Chambadal.

The second French G2 graphite pile, the building of which has begun at Marcoule, Gard, will be in operation in two years' time.

The scheme is based on research carried out jointly by the reactor research department of the Atomic Energy Commission, the nuclear equipment division of *Electricité de France* and French private industry, mainly the *Alsacienne de constructions mécaniques*, the *Société des forges et ateliers de Creusot*, the *Société Alsthom* and the *Société Rateau*.

Before I describe the plant, I should like to point out that since this paper was originally prepared we have had an opportunity of reading a number of papers from other countries, in particular Mr. Marvin Fox's recent paper on the Brookhaven reactor and the paper to be presented shortly by Sir Christopher Hinton on the Calder Hall piles.

In reading these papers we have sometimes observed that parallel lines of thought have been followed independently in various countries; for example, the *Société Rateau* and the Atomic Energy Commission in France have specified for the G1 pile just described by Mr. Chambadal a central gap practically identical with that of the Brookhaven reactor.

It is reassuring to have the results of our research confirmed in such cases. On the other hand we have also encountered lines of thought which are radically opposed to our own, and this puzzles us. I shall give you an instance shortly.

In the G2 pile, the fuel takes the form of cylindrical elements of natural uranium alloyed with a very small quantity of aluminum. Each element is canned in a finned magnesium tube mechanically bonded to it.

These slugs rest in 70 mm diameter channels of circular section hollowed out horizontally in the graphite. Although this arrangement, in the view of some engineers, may be less satisfactory than a vertical one, it was adopted because the metallurgical difficulties presented by the fuel element were fewer.

Thus, our experience at Saclay with a pile having vertical channels in which compressed carbon dioxide circulates did not prevent us from building G2 with horizontal channels.

On the other hand, our British friends, who have had a great deal of experience with horizontal channels at Harwell and Windscale, are building vertical

channels at Calder Hall. Here is one puzzling point, or at least we find it so.

The graphite in which the channels are cut has a density greater than 1.7 and an effective capture cross-section of 4 mb for neutrons of 2200 m/sec., allowing for nitrogen.

Even after some weeks of operation at full power, there is a reactivity surplus which permits, *inter alia,* collateral production of uranium-233 from thorium, the use of slightly impoverished natural uranium, a flattening of the reactor flux, or combinations of these three factors.

The fuel can be renewed without shutting the pile down, and this seems to us of fundamental importance, for, as has been mentioned several times from this platform, a power station of this kind must operate as a central station. Again, the need to produce as much plutonium as possible, in order to bring down its price, requires the maintenance of the maximum permissible power, other things being equal, of course. In practice the only limitation on the power is the frequency of slug bursts, which would quickly become intolerable if it were necessary to shut down the reactor, reduce the gas pressure before changing the damaged fuel element and then build up the pressure and power again. These operations, which are particularly time-consuming make it necessary to avoid thermal shocks, which will be eliminated in G2.

The entire unit consisting of the pile structure, the fuel renewal arrangements and the heat production equipment must be enclosed in a gastight housing capable of resisting an average pressure of 15 kg/cm^2 and perforated by a large number of holes.

To prevent any leakage from the concrete housing, its inside walls were covered with a sheet-steel lining. The flow of relatively cold carbon dioxide between the lining and the thermal shield allows the concrete to be kept at an acceptable temperature.

I shall deal briefly with the cooling circuit, which is similar to G1 so far as energy utilization is concerned. The flux in the uranium decreases towards the periphery so that there are two temperature limits, one at the center and one at the periphery of the reactor which determine the cooling arrangements. We decided not to exceed 550°C at the center of the uranium core or 400°C on the magnesium cans.

The face where the carbon dioxide is introduced was therefore divided up into concentric zones, into which the gas is admitted at different controllable temperatures.

On emerging from the various channels, the gases mix at an average temperature exceeding 300°C. By tapping part of the hotter peripheral gases, the steam can be superheated; the steam is raised in a unit consisting of heat exchangers and boilers and rives through the turbines, an alternator and the carbon dioxide compressors. A small auxiliary boiler is therefore needed to start up the plant, which can then supply, at its full capacity, a net power of about 30 Mw.

It has been decided that two piles of this kind, to be known as G2 and G3, will be built at the same time and their rated capacity of 60,000 kw will supply the first electric current produced by nuclear energy to the French grid.

As Mr. Ailleret has mentioned earlier, projects of higher capacity will follow; these projects have been decided on by the French Government at the suggestion of an advisory board for the production of electricity from nuclear energy.

This scheme is, in effect, a minimum programme, based on graphite-moderated, carbon-dioxide-cooled, natural-uranium piles, and it is scheduled to begin in 1959 with the 60 Mw of the EDF1 station; it then provides for the power installed each year to be doubled rapidly—about every three or four years.

The minimum programme will expand still further as the Atomic Energy Commission continues to perfect improved types of pile and if, as is possible, it is extended for technical reasons or by the establishment of power stations in France's overseas territories.

Mr. GAST (USA) presented paper P/492.

Mr. STARR (USA) presented paper P/493.

DISCUSSION OF P/333, 337, 492 AND 493

Mr. VALLARTA (Mexico): I have two questions to ask in connexion with Mr. Chambadal's paper P/333. The first question is: what is the capital cost per kilowatt of electric power produced? The second question is: what is the cost per kilowatt-hour of the electric power from the G1 plant?

Mr. CHAMBADAL (France): My reply to the first question is as follows: As I mentioned in my paper, the capacity of the plant is 5500 kw and its cost was about 676 million francs, and the capital cost per kilowatt of rated capacity can be deduced from these figures. This is, of course, for power generation alone, and does not include the reactor. The sum of 676 million francs covers the cost of the heat-exchanger, boilers, turbo-alternator sets and the various accessories.

The second question I have been asked concerns the cost price per kilowatt-hour of power from this plant. This is, obviously, a very general question. There are two possible answers; firstly, no one yet has enough experience to determine the cost price; secondly, every plant of this kind is intended to produce, and does produce, not only electric power but also plutonium. Hence, the cost price of the power depends on the proportions of the over-all cost attributed to plutonium and to the kilowatt-hours produced.

Mr. SPINRAD (USA): I should like to ask what gross and net thermal efficiencies of the G2 and G3 plants are expected—that is, the ratio of electric power generated to heat generated in the system, and

also the ratio of electric power available for distribution externally to the system when you take away the power for fans, and so on, compared to the heat generated.

Mr. TARANGER (France): The heat capacity of the G2 and G3 piles is 150 Mw. The thermodynamic efficiency of the heat cycle is about 25 per cent. Allowing for the requirements of the blowers and carbon-dioxide compressors, the net efficiency is only 20 per cent.

Mr. STAUB (Switzerland): I should like to know what method is used for loading the reactor while in operation. Is there an arrangement of gastight locks? I should also like to know what accessory apparatus is used.

Mr. TARANGER (France): A gastight chamber containing the new element is applied to the required channel. A push-rod arrangement places the element inside the reactor and the new element expels the worn-out element which is to be replaced at the other end of the channel, where it drops into a gastight disposal system also equipped with locks.

The CHAIRMAN: We shall now have some questions on Paper 492.

Sir John COCKCROFT (UK): Could Mr. Gast tell us how far the United States has gone towards achieving a 10,000-megawatt-day per ton burn-up?

Mr. GAST (USA): That is a rather difficult question for me to answer, because in this design we are depending upon results achieved at other laboratories in the United States. We are not ourselves carrying out an extensive program of developing fuel elements for use in power reactors. I believe that the question should really be answered by persons in the other laboratories who are doing this type of work.

Mr. BENEDICT (USA): I should just like to have a few specific points of information with regard to the design of the graphite-moderated reactor described by Mr. Gast.

First, what enrichment of uranium-235 was used in the design described to us?

Secondly, what conversion ratio was expected from this design?

Thirdly, in arriving at the power cost of 7 mills per kilowatt-hour, what rate of return on plant investment was assumed?

Fourthly, what would be the general effect of loss of cooling water on the reactivity of the system?

Mr. GAST (USA): The exact enrichment and the exact conversion ratio will, of course, depend upon the final form of the fuel element. Our initial calculations, however, have indicated that enrichment in the neighbourhood of 0.9 per cent of U^{235} would be required, and the conversion ratio would be in the neighborhood of seven-tenths.

As regards the percentage of return on investment, this worked out—in computing the cost—to be 13.9 per cent. This figure was achieved by adding up individual charges on the investment.

The consequences of loss of water from the reactor will always depend on the exact form of the fuel element. We have chosen here a lattice and an amount of uranium going into it which would give only a very slight change in reactivity upon loss of water from the reactor. This change in reactivity may be either positive or negative, depending upon the amount of plutonium which is built into the fuel elements at the time that the incident occurred and on the attainable exposure in the fuel elements. Immediately upon loss of water we feel that the situation would be well in hand. If for some reason the graphite went on heating up after loss of water, the insertion of the safety control system would be necessary in order to keep the reactor from again going critical.

Mr. BLOKHINTSEV (USSR): I should like to know why steel was used as a cladding for the uranium elements, when zirconium is sufficiently stable in a sodium medium and would be preferable from the point of view of neutron economy. That is my first question.

My second question is, what is the maximum temperature of the graphite in this device?

Mr. STARR (USA): Steel was used for the fuel element rather than zirconium because the strength of zirconium was not considered high enough to carry the full load of the fuel element structure. The uranium rods or slugs that go into the stainless steel tube are supported by the bottom of the tube so that the whole uranium load is carried by the thin wall tube.

As regards the second question, we do not know what the graphite temperature will actually be. We know that it will be quite high—much higher, of course, than the liquid sodium temperatures. There are, however, many problems of heat transfer as well as energy absorption of the graphite. Hence, as I have said, we do not know exactly what the temperature will be.

Mr. BLOKHINTSEV (USSR): I have several questions for Dr. Gast, but I think I shall meet him in the corridor and have a talk with him. I should like only to ask a question about his design for a reactor. Dr. Gast's plan of a reactor closely resembles the plan of our operating station. I should be interested to know whether it is intended to build this reactor or whether it merely marks a stage in the development of reactor design.

Mr. GAST (USA): This is just a design on paper. There has been no beginning of construction, or anything of that sort, on the reactor. This is just a proposed design.

Mr. GOTT (UK): The use of central entry for the coolant must be a balance of the neutron loss and the pressure drop in the slit. There is also a problem of gas distribution to the channels. I do not think that

that is terribly difficult. The central entry appears to give a system with a minimum pressure drop, and it is therefore very suitable for a reactor which produces plutonium; but it gives a lower temperature rise and, consequently, may not be quite so useful for a power reactor. In order to recover this situation, it is necessary to reduce the cross-sections of the cooling channels. The ones mentioned are rather smaller than I should have expected. In doing this, however, you again put up the pressure drop in the cooling channels. My question is the following: having these things in mind, would Mr. Taranger have used central entry of coolant if he had set out in the first place to design a power station plant, or does his proposal arise from the plutonium production requirement for which the reactor was at first designed?

Mr. TARANGER (France): I think there has been a little confusion. In speaking of G2, I referred to the central gap in the G1 reactor, but there is no such gap in the G2 power pile itself. The question is simply where you are going to put the money or the neutrons. Although we may have thought that it would pay to have a gap in an atmospheric-air-cooled pile, because there would be a considerable reduction of pumping power, this gain in pumping power becomes negligible when you are dealing with a gas at 15 kg pressure. Again, a central gap obviously has the disadvantage of keeping the graphite temperature down at the very point where the fast-neutron flux is highest.

The CHAIRMAN: The next question relates to paper 492.

Mr. GOODLET (UK): I should like to ask four questions relating to the design of the water walls.

Would the author care to tell us more about the method of support of this flat wall against a very high pressure?

Would he care to tell us more about the method of handling the uranium charge through the water wall?

Would he care to tell us more about the method to be used for replacing defective cooling tubes in the pile?

Finally, would he tell us something about the method of locating a defective can, if such an event occurred?

Mr. GAST (USA): Before answering those questions, I should perhaps make it clear that I am not the author of the paper; I am merely the speaker.

The water wall design is of the following type. For each tube that pierces the water wall, there is a cylinder which is welded to both the front and back sheets of the wall, so that we have many internal braces to keep the two faces of the wall in place. Then, the tubes which go through the reactor pass through these cylinders and are attached on the outside surface of the wall, so that the tubes themselves furnish the support which holds the front and rear walls together. This supports the structure against the pressure which would normally force the walls apart.

The temperature of each tube is individually monitored at the outlet end, so that one can determine here any flow blockage; also, we have a radioactivity monitoring system which picks up the presence of fission products in water, and by a system of elimination we can locate the tube from which these products are coming. If you would like more detail on that, I think I could arrange to furnish it to you privately, but it is rather detailed.

Mr. PREECE (U.K.): In view of the relatively high cross-section of sodium, I should like to ask Mr. Starr if he would care to comment on the alternative possibility of using either bismuth or the separated isotope of lithium, lithium-7, in reactors of this type. The point I have in mind is that one could afford to use twelve times the volume of bismuth or eight times the volume of lithium-7 for the same total parasitic capture.

Mr. STARR (USA): The choice of sodium was primarily an engineering choice in terms of building a plant. The melting point of bismuth is not quite as satisfactory as that of sodium. It gives handling problems in production of radioactive isotopes, and it is our feeling that sodium would be a better engineering choice. In the case of lithium, there is the problem of availability, and that of the purity of the lithium; there is also the very limited experience in terms of its engineering handling in a system of this type.

Mr. LARSON (USA): I should like to ask Mr. Starr what initial cost of U^{235} or U^{233} he was assuming in predicting a 6–7 mill cost of power from the thorium reactor.

Mr. STARR (USA): We were assuming no specific number for U^{233} since that would be produced in the cycle, and the only costs involved were the costs of chemically reprocessing the fuel. The costs for U^{235} were taken as between $15 and $30 a gram, and this gave us the range of 6–7 mills for the cost of the power from the thorium cycle in the future.

Mr. GAST (USA): I should like to ask what the consequences would be of getting moisture into the carbon dioxide gas system as, for example, from a leak in the heat exchangers.

Mr. TARANGER (France): Judging by the experience we have had on the antipathy of magnesium for water, the consequences would be horrible. We therefore intend to build a gastight heat exchanger, but, (more important) to raise the steam at a lower pressure than that of the carbon dioxide so that any leaks in the heat exchanger will be from the gas to the water and not from the water to the gas.

Mr. LEWIS (Canada): In the paper, P/492, it is stated that, in the event of a fuel element rupture, it is planned to continue to operate until the next scheduled shutdown. Operating in this way, I should

like to ask what monitoring devices remain operative then to guard against any more serious failure in the fuel element such as, for example, the melting of magnesium in one of the proposed types of fuel also mentioned.

Mr. GAST (USA): There is a temperature monitor on each tube which monitors the outlet water temperature, and there is also a flow monitor, so that in the event of serious blockage of the tube by the consequences of a fuel element failure it would be necessary to shut down.

Mr. GOODLET (UK): Could Mr. Starr say how a defective can is first of all detected and, secondly, located?

Mr. STARR (USA): We have no means of detecting a defective can. The effect of a defect in the can is to allow sodium to enter into the pores of the graphite, and this reduces the neutron efficiency of the machine and shows up as a change in reactivity. In a 75,000 kilowatt plant, a defect of this kind would hardly be noticeable normally. If it occurred suddenly, it might be noticed by the operators. The problem of locating defective cans, in the event that enough of them were defective to require removal, could be solved by temperature distributions. The effect of the sodium in the can is to change the neutron distribution. This would change the temperature of the fuel rods, and if enough of this occurred over any section of the reactor, the thermocouples which are on each fuel rod would give evidence of a maldistribution.

Mr. GOODLET (UK): The can I was referring to was the fuel element can. I am sorry I did not make myself plain.

Mr. STARR (USA): We have no way of detecting a rupture in a fuel element can either. Each rod, with a stainless sheet on it, has sodium on one side of it, and either sodium or sodium potassium alloy on the other side. In the case of a rupture, there would be a slow mingling of these two liquid metals, but this would have no consequences on the system except perhaps to permit a small amount of surface fission products from the uranium slugs to leak into the sodium. Since the sodium system is already radioactive, there would have to be a considerable leakage of this kind to have any serious effect.

Mr. FENNING (UK): I notice that Mr. Starr's neutron economy breakdown depends on the assumption of a value of η of 2.09. In other papers presented to this Conference it appears that there is some uncertainty on the value of this quantity. Secondly, I assume that the amount of burn-up that is assumed requires the consumption of plutonium-239, which again has a lower value of η. It seems to me, therefore, that it may well be that, instead of a conversion ratio of 0.72 as quoted, it may well be as low as 0.6, bearing in mind the neutron spectrum. I should like to ask, therefore, if Mr. Starr has considered how this sort of phenomenon would affect his costs of power.

Mr. STARR (USA): We have not taken any of the new values presented at this meeting and recalculated the effect of these on the cost of power. Obviously any impairment of neutron economy will have some effect on increasing the cost of power.

Mr. LILJEBLAD (Sweden): I should like to ask Mr. Starr if he considers both a primary and a secondary cooling system to be absolutely necessary. Is there any risk that the steam would be contaminated if you used one cooling system only?

Mr. STARR (USA): The two systems, the primary system and the secondary system, may not be essential. However, there are several considerations that go into this choice, and safety is only one of them. It is apparently possible to design steam boilers where, by intermediate liquids or intermediate structures in the boiler, the safety of a sodium water boiler can be increased, and I think it might be possible in the future to remove the secondary system with safety. However, the steam boiler is very large compared to the size of a metal to metal heat exchanger, since the primary sodium must be shielded because of its gamma ray activity. The cost of the structure—I am talking now about the concrete vault that holds the primary system—would be considerably increased if the steam boilers were put on the primary system, so there may be a very serious economic reason for having both a primary and a secondary system. The secondary system, of course, requires no shielding.

Mr. BOWDEN (UK): Would Mr. Taranger say what is the total pressure drop in the gas circulating system for G2? What type of circulator or blower is used, what type of drive to the circulator, and what total speed regulation is obtained or demanded of the circulatory system?

Mr. TARANGER (France): I am sorry that I cannot answer the first question as I do not carry the exact figures in my head. The type of blower used is a kind of steam-driven compressor centrifuge. The superheated steam is dried by tapping some of the peripheral gases, which are hotter than those reaching almost every other part of the heat exchanger.

With regard to the degree of regulation, we are waiting until we have some experience with the first small power station connected with the G1 pile before drawing conclusions in regard to running an electric power station associated with a pile.

The CHAIRMAN: There are still some questions on these papers we have been dealing with up to now but I think we have to go on with our session.

Mr. WENT (Netherlands) presented P/936 as follows: In the Norwegian-Netherlands co-operation, the FOM-KEMA reactor development team working in the KEMA laboratories in Arnhem, has the

task to investigate the possibilities of non-metallic liquid fuels and to develop, if possible, power reactors with such fuels. We believe that, if we can solve the various problems present in liquid fuel reactors, a very reliable and economic power reactor can be built.

Two different types of liquid fuels are of special interest to us, both types using oxides, namely uranium or thorium oxides. The first type is an oxide suspension in heavy water, a so-called "wet suspension"; the second type is an oxide suspension in a gas, a so-called "dry suspension."

I should like to start with a few remarks considering the wet suspensions. The research and development in this field is concentrating on a homogeneous reactor and is now so far advanced that we believe it is justified as a first step to the power reactor, to build a small-scale reactor with a suspension of intermediate enriched uranium (20 per cent U^{235} in ordinary water). The first experiments, mainly with respect to the kinetic behaviour of such a reactor will be carried out in a zero energy experiment, but a high neutron flux and high temperature version of the reactor will follow as soon as possible.

As liquid fuels, we, in the Netherlands, are considering specifically oxide suspensions in heavy water. We believe they will give us the best chances in our difficult raw materials situation, as no uranium or thorium is found in the Netherlands, nor can be expected to be found. We need to develop reactors with, if possible, only low enriched fuel and a high conversion ratio for uranium, or a real breeding ratio in the case of thorium. The best thermal breeder would be a homogeneous Th-oxide reactor moderated with heavy water, without construction materials and with continuous purification possibilities. With such low enrichment, solutions would almost inevitably give too high neutron losses.

In the case of thorium there is only one oxide to be considered (ThO_2), but we must start at least with uranium oxide as it is impossible for us to enrich the thorium sufficiently. In the case of uranium the whole range of

$$UO_2 \longrightarrow U_3O_8 \longrightarrow UO_3 \longrightarrow UO_4$$

should be considered. It is very interesting that in the United States much attention has been paid to UO_3 and that in the homogeneous boiling reactor described by the USSR, the possibility of a U_3O_8 suspension is mentioned whereas we advocate UO_2.

The arguments in favour of UO_2 are manifold and I should like to mention some of them.

In the first place, even in a strongly reducing atmosphere UO_2 cannot easily be reduced further; in the second place, UO_2 is stable in water at 250°C. In the homogeneous reactor considered here a serious decomposition, however, of the water will take place, hydrogen and oxygen are formed and it is probable that in these circumstances UO_2 is oxidized up until UO_3. We are, however, adding an excess amount of hydrogen to the suspension at the outlet of the reactor vessel to avoid a detonation of the hydrogen oxygen mixture after the separation of the suspension and the gas. With such an excess of hydrogen UO_2 will remain stable. What is happening under normal reactor conditions, when apart from oxygen and hydrogen gas also radicals, peroxides and superoxides are present, must be investigated in our planned small-scale reactor.

In the third place, UO_2 does not form hydrate unlike UO_3. Such hydrates are undesirable since they lead to recrystallization effects and formation of non-spherical particles.

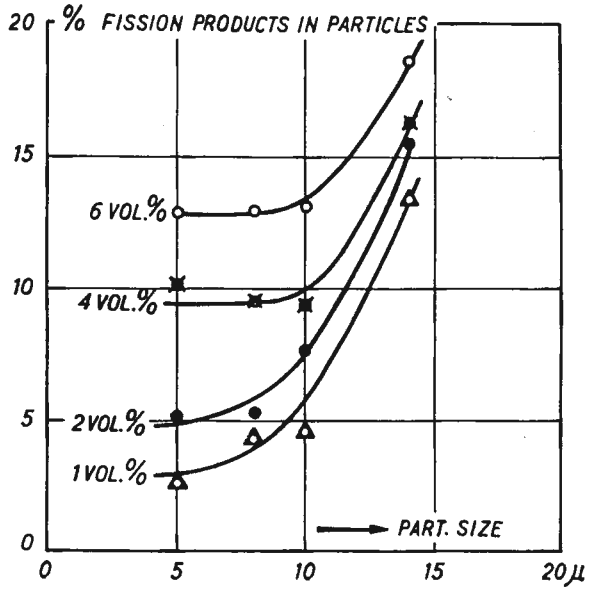
In the fourth place, the crystal parameter of the uranium-oxide system is as follows. The crystal parameter remains constant between the oxygen contents of 2.0 and 2.2 and increases from 2.2 to 2.7. Consequently small changes in the oxygen content of UO_2 will not result in additional stresses in the particles.

In the fifth place, the colloid chemical properties of UO_2 and U_3O_8 suspensions are equal. Small changes in the rate of oxidation do not effect the colloid chemical stability. All these suspensions are stable for pH values >6.3.

With a suspension it is possible to separate the fission products from the fuel in a very efficient way as soon as the diameter of the suspended particle is smaller than the mean free path of the fission nucleus.

In Slide 1 you can see this separation effect in irradiated suspension particles in the case of iodine. For other elements like Ba, Mo and Cs we obtained the same results. As can be seen the separation effect is rather sensitive to the particle size; for particles with a dimension exceeding 10μ it is decreasing rapidly. Furthermore the separation effect is of course a function of the concentration. For high volume concentrations the chance that a fission atom is recaptured in a neighboring UO_2 particle is great. In the slide you can see this effect too for different concentrations. Now in the slightly enriched power reactor as well as in the small-scale reactor we are planning to build now with fuel enriched up till 20 per cent U^{235} we need the same volume concentration of about 4 per cent.

Although the separation is very efficient, this does not mean that the fission atoms remain free in the water phase, on the contrary, they will be absorbed at least partially on the UO_2 particle surface. The concentration of the fission products and the active surface of the particles are therefore important. Therefore it is not sufficient to produce particles of about 10μ. They should be if possible without an internal surface. Fortunately this can be achieved by sintering the appropriate particles at relatively low temperature. A few remarks should be made about the decomposition of the water inside the reactor. Here again the small-scale reactor must give the answer, but with the figures known from the Oak Ridge Homogeneous Reactor Experiment, we



Slide 1

can make some rough estimates. There are three reasons to expect that in our case the decomposition will be considerably less. In the first place about half of the kinetic energy of the fission products, which are mainly responsible for the decomposition, is dissipated inside the oxide particle and only the other half inside the water. For this reason, as well as in order to reduce the active surface, the particles are made as large as is compatible with the efficient separation of the fission products from the fuel. In the second place the water phase itself does not have a high content of electrolytes avoiding therefore ions which can be readily reduced or oxidized. In the third place in general the decomposition into oxygen and hydrogen molecules is reduced by an excess of hydrogen which is already present for other reasons.

For the stability against gravity it is important to know that the settling velocity in heavy water of 250°C will be about 1 cm/sec for the suspensions we should like to use. At flow velocities ten times larger no settling is to be expected. In the power reactor these conditions are always fulfilled, but in the small-scale reactor that will not be so and special precautions must be taken.

One possible solution is shown in Slide 2 (Figs. 1, 2 of P/936). I should like to draw your attention first to the design on the right-hand side. The suspension after entering the reactor vessel at the bottom is divided in two directions, one central and one outer flow. Afterwards these two flows are impinging on each other resulting in a turbulent mixture in this mixing chamber. This is necessary as the flow velocity in the outer circuit is so high that there the suspension does not remain homogeneous. After homogenization of the suspension a set of vertical partitions is used to guide the suspension flow in the vertical direction through the cylindrical reactor. However under these circumstances the flow is not completely stable and therefore a slow rotation is superimposed over the vertical flow. In Figs. 1b and 1c the special straightening device is shown. Finally in Fig. 1a a model of the reactor vessel, almost on scale, is shown with a suspension flowing through it, with settling properties equal to the settling properties of 10μ UO_2 particles in water of 250°C.

I should like to draw your attention to this: the side tubes in this model are not essential for the reactor vessel; they are only used to measure by optical methods the concentration through the vessel.

Before showing you a simplified flow-sheet for such a homogeneous reactor I should like to make a short remark on the pumps to be used. In a reactor with a highly radioactive outer-circuit very reliable pumps are needed. For the power reactor therefore gas lift pumps are being developed in which the driving force is due to the difference in the over-all specific weight between a suspension and a suspension mixed with a large volume percentage of gas (again the hydrogen gas or in the case of the power reactor deuterium gas). The volume of a gas lift pump, however, is rather large and therefore we will combine this gas lift pump with the heat exchanger by splitting up the pump in a large number of parallel gas lift pumps with a large heat exchanging surface. To secure a stable functioning of these parallel pumps it is necessary to introduce several special technical features concerning the gas inlet system and the method of starting the pumps.

However, in our small-scale experiment, we will use a normal, canned rotating pump mainly because the flow resistance in this system is relatively large and under such circumstances a rotating pump is far more efficient. Moreover, the life of the pump in this experimental reactor does not need to be so long.

In Slide 3 (Fig. 3, P/936) you can see a simplified flow sheet of the reactor. The primary fuel circuit contains the reactor, the gas separator, the rotating pump and the very simple heat exchanger. Then we have the gas circuits. They start by the gas separator. In the by-pass circuits water flows except in one special case.

The reactor is surrounded by a BeO reflector and a second graphite reflector. The combination of the reactor vessel and the reflectors is enclosed in a pressure vessel. The gas pressure on top of the suspension system is connected with the gas inside the pressure vessel. Therefore, the reactor vessel itself does not need to withstand a pressure difference and it can be constructed with a thin wall. Behind the gas separator are the normal recombination unit and a pump circulating the excess amount of hydrogen which is always used in this system. As far as xenon is formed, this will be taken up mainly in the gas circuit.

Between the pressure and the pumping side of the rotating pump, a by-pass circuit is built mainly to adjust the equilibrium temperature in the reactor

vessel itself. With a cyclone at this point, the suspension particles are separated from the liquid, the liquid is returned to the primary circuit and the suspension particles can settle in this storage vessel. If it is desired, it is possible afterwards to allow the suspension particles to go from the storage vessel again to the primary circuit. I should like to draw your attention to the fact that in the primary circuit we do not have any valve at all. It is rather unsatisfactory to have a suspension valve. However, underneath the storage vessel we need to have a suspension valve and we have special valves which we believe will be reliable.

Apart from this by-pass system, a second by-pass system for chemical purification is planned. In the second by-pass system there is a cyclone here in which you have a separation of the suspension return to the primary circuit and a fuel purification possibility. Finally, a dump vessel is present to drain the whole system in case of emergency.

The zero-energy reactor does not require these secondary circuits except the circuit for adjusting the concentration. On the other hand, both regulating rods and a very accurate temperature control are needed to maintain the reactivity at a constant level: in a zero energy reactor the large negative temperature coefficient of reactivity cannot be used for inherent stability.

I should like to finish by stating that we are aware that constructing these reactors successfully is a difficult task for a small team of scientists such as ours and we shall value criticism.

Mr. DE BRUYN (Netherlands) presented P/938 as follows: Instead of summarizing the paper on the use of dry suspensions as a reactor fuel, I prefer to make a few additional remarks. In this paper, three different types of reactors are discussed.

The first is a homogeneous reactor in which compound particles consisting of the moderator BeO and the fuel UO_2 are used [Slide 4 (Fig. 1, P/938)]. The reactor vessel contains the compound particles of 250 microns which move in a downward direction. The particles are transported pneumatically to the boiling bed heat exchangers. They are returned back to the top of the reactor vessel by simple overflow of the boiling beds. Helium gas is used to transport the particles. The vessel is surrounded by a beryllium oxide reflector.

We discussed a heterogeneous reactor in which pure UO_2 particles are used [Slide 5 (Fig. 3, P/938)]. The reactor vessel contains a number of graphite tubes filled with pure uranium dioxide particles again of 250μ diameter. The heavy water between the tubes is kept at a rather low temperature of about 50°C. It acts purely as a moderator. The hot uranium dioxide particles of up to 1200°C move in a downward direction and are transported pneumatically to the boiling bed heat exchangers outside the reactor. A graphite reflector will be used. The moderating heavy water is cooled outside the reactor at this point. Because cold heavy water is used as a moderator and graphite as a construction material, this reactor can be fuelled with natural uranium; moreover, the conversion ratio will be high.

The third type of reactor we have discussed is a so-called quasi-heterogeneous reactor in which large compound grains of about 3 mm diameter consisting of BeO and UO_2 are used [Slide 6 (Fig. 2, P/938)].

In this type of reactor a great number of boiler tubes, made of graphite, are placed inside the reactor. Heavy water under forced circulation is passing through the tubes. The compound grains of fuel and moderator are in a fixed bed condition between the tubes. Helium gas is passing through the bed of grains. The hot gas leaving the reactor vessel can be used for drying or superheating of the heavy steam produced. The reactor vessel is surrounded by a beryllium oxide reflector and a pressure vessel.

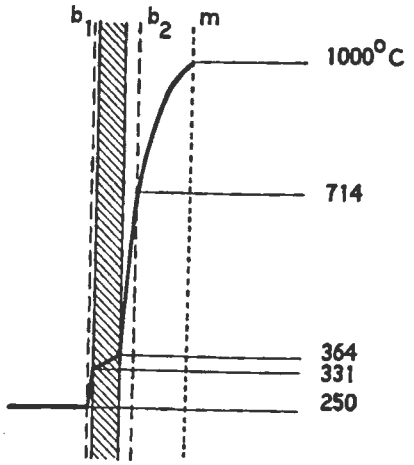
In both cases, the homogeneous and the heterogeneous reactors, the dry suspensions are used for the transport of heat out of the reactor into the heat exchanger. It turns out that in the case of a large power reactor, 300 Mw thermal power, the hold-up of fuel outside the reactor vessel is about three or four times the reactor content. This means that high investments are necessary even in case of the heretogeneous reactor in which natural uranium can be used as a fuel. Moreover, we may expect in both types of reactors, severe attrition and erosion problems as the particles have to be transported pneumatically. It may be that by shaping the particles correctly, the attrition can be reduced considerably.

In the third system, the quasi-heterogeneous reactor, the particles are at rest. The heat transfer from compound grains to the coolant inside the tubes within the reactor vessel is excellent even when the grains are in fixed bed condition, provided hydrogen, deuterium or helium gas is flowing between the grains to achieve transversal heat conduction.

At this point it is necessary to make an additional remark to the original paper. After the paper went to press, this type of reactor has been recalculated more accurately. It turned out that the heat transfer will be much better than was assumed in the paper. Therefore the dimensions of the reactor will be much smaller (2.50 m diameter and 2.50 m height) and the amount of fuel and moderator will be much lower, namely 1.5 tons moderately enriched uranium oxide and 11.5 tons beryllium oxide. We calculated for a 300 Mw reactor the temperature profile between coolant and grains. See Slide 7.

On one side of the graphite tube wall, we are dealing with boiling heavy water of 250°C and 40 atmospheres. On the other side, a temperature gradient exists in the fixed bed of compound grains up to 1000°C. On both sides of the wall we will have transition zones—a liquid film on one side and the first layer of grains on the other.

We examined the relative investment of fuel, moderator and coolant. For the three types of dry



Slide 7. Temperature gradient across boundary layers; b_1, boundary liquid film; b_2, boundary first layer of grains; m, maximum temperature

suspension reactors, the preliminary results are given in Table 1.

The figures in column I are for the homogeneous type of reactor, those in column II for the heterogeneous type of reactor and those in column III for the quasi-heterogeneous type of reactor. With power is meant heat production. With respect to the percentage of U^{235}, this is only an approximate calculation because of the rather complex structure of the moderator.

Because the prices of natural uranium, U^{235} and heavy water are still disputable, we expressed the investments of fuel, moderator and coolant only in tons of material.

We left out the dollar investment figures. The only thing we can say is this: that the investment will be the lowest in the case of the quasi-heterogeneous reactor and about twice as much in the other two cases. However, the quasi-heterogeneous reactor type, although it has the lowest investment figures, has two disadvantages. The conversion ratio will be much smaller than that of the heterogeneous reactor type, because of higher neutron absorption and, secondly, the whole reactor has to be pressurized.

Table 1

Reactor type	I	II	III
Power (megawatt)	300	300	300
Volume reactor vessel (m^3)	5	13	31.5
Pressure reactor vessel (atm)	—	40	—
Maximum grain temperature (°C)	1200	1000	1200
Steam temperature (°C)	310	310	310
Percentage U^{235}	6	4	0.7
Quantity UO_2 (natural U) (tons)	2.75	1.45	117.5
Quantity UO_2 (U^{235}) (tons)	0.15	0.05	—
Quantity BeO (tons)	31.5	11.5	—
Quantity D_2O (tons)	—	10	30

With respect to the removal of poisonous fission products, the quasi-heterogeneous reactor type offers excellent possibilities, the same as the other two reactors. As a matter of fact, the xenon and other volatile fission products will diffuse out of the grains already at the high temperatures used inside the reactor, whereas reprocessing will be necessary to remove samarium-149, the rare-earth elements, and so on. Therefore the compound grains will be purged continuously and slowly without any danger for erosion or attrition. The let-off valves will be of the same design as has been described in the paper.

In these crucial valves, necessary for the remote control of the flow of the grains, the principle of fluidizing a system of grains by means of vibrational movement is applied. See Slide 8 (Fig. 4, P/938).

The upper part shows the valve in connexion with one of the tubes of a heterogeneous dry suspension reactor for which it was originally developed. The lower part shows more details. A specially shaped cone is vibrated more or less intensively. The frequency is determined by the size of the particles.

A special feature of using an internal heat exchanger is this: the fuel grains will move through the reactor vessel very slowly without mixing. The purified grains are introduced at the top of the vessel, whereas the contaminated grains are purged at the bottom.

We have come to the conclusion that the heterogeneous power reactor type still offers the best possibilities. It has the following attractive qualities: The possibility of producing high-quality steam without the necessity of pressurizing the reactor; the cheapest fuel can be burned in this reactor, namely natural uranium in the oxide form, and a high conversion ratio and probably the possibility of breeding with thorium oxide are ensured.

However, in case moderate enrichment of the fuel can be accepted and a high conversion ratio is not necessary, the quasi-heterogeneous type of reactor will be the most attractive one from the point of view of heat extraction.

Mr. WILLIAMS (USA) presented P/494.

DISCUSSION OF P/936, 938 AND 494

Mr. RIEZLER (Germany): Is there any danger that rapid changes occur in the concentration of the suspension so that dangerous changes in reactivity occur?

Mr. WENT (Netherlands): We do not believe it will be very serious, but we are planning to build a small-scale reactor and want to investigate this problem. In general, even if you have changes in concentration we believe it should not be too serious with regard to the large negative temperature coefficient of such types of reactors as we are using. I should like to add one point: as soon as one considers a slightly enriched liquid fuel reactor, we can adjust the concentration, and both a decrease and increase of the concentration would give a decrease of the

reactivity. In that respect we believe it might always be completely stable.

The CHAIRMAN: Mr. Vladimirsky of the USSR would like to put a question to Mr. Went or Mr. de Bruyn.

Mr. VLADIMIRSKY (USSR): My question refers to the reactor using the suspension of oxides in heavy water. First of all I should like to point out that this approach offers very favorable prospects for the production of cheap energy. I have two questions to ask. First, what is the maximum concentration of uranium oxide in heavy water by weight? Secondly, I should like to learn details of the construction of the pump used to transfer the suspension, and especially the condensate. How is the suspension separated from the gas which occurs above it?

Mr. WENT (Netherlands): Concerning the concentration of uranium oxide suspension, this will be about four volume percentages for large power reactors. Therefore, we are using in the small-scale reactor experiment an enrichment of U^{235} where we can use just the same concentration of four volume percentages. Using such a concentration, the radiological properties of the water itself are not very much changed.

In the case of the power reactor, we are considering constructing so-called gas lift pumps. As I mentioned in my lecture, we should like to combine this construction of the gas lift pump with the heat exchanger. That means that we must build a set of parallel gas lift pumps. If you do that, it might be that one of the tubes of the gas lift pump is pumping in the right direction and the second tube is flowing downwards. That might be caused by the gas inlet in the system that is all right on one tube but not all right on the second tube. For that reason we are introducing a resistance in the gas inlet system, and this resistance is so high that the pressure of the gas just before this resistance point is sufficient to overcome the pressure of the height of the complete pump. Secondly, by starting the pump it may be that one of our tubes is running upwards and the second tube is running downwards, even if the gas inlet in that system at the bottom is all right. In that case, the inlet system at the bottom would also flow downwards with the gas and up through another tube.

To avoid that, you need to have a separate gas inlet system on the top of the tube during the starting of the pump. It is possible to construct this gas inlet on the top of the tube in a very simple way without special gas inlet points by using the gas coming from the other tubes flowing down through the tube which is not working correctly.

Mr. HAWTHORNE (UK): I should like to ask Mr. Williams what proportion of the neutrons leaking from the core are lost to the reactor as a whole by leakage through the inlet and exit pipes up which the uranium bismuth solution flows into and out of the core.

Mr. WILLIAMS (USA): As we have not fixed the exact geometry of the inlet and outlet in this design I cannot give you a specific answer. However, from some calculations on some other designs that we have made, we do not believe that this will be sufficient to cut your losses down to a point where you will not be able to get a breeding ratio of one.

The CHAIRMAN: Mr. Manson Benedict of the United States has a question to ask Mr. de Bruyn on paper P/938.

Mr. MANSON BENEDICT (USA): I wanted to inquire about the pressure of heavy water in the quasi-heterogeneous reactor that you described, and I wanted to inquire how leakage of heavy water or gas through the graphite tubes separating them was prevented.

Mr. DE BRUYN (Netherlands): The pressure inside the tubes will be about 40 atmospheres when the reactor is operating at a temperature of 250°C. We believe, with respect to the tightness of the graphite, that it will be sufficient. There is no question of a large pressure difference between the inside of the boiler tubes and the outside. The whole reactor vessel is at the same pressure.

Mr. FRY (UK): Has Mr. Williams any evidence to show for how long the graphite components in the core of the reactor will stand up to the operating conditions? Could they be replaced?

Mr. WILLIAMS (USA): With regard to how long they will stand up, we have some evidence but not at high temperatures on the effects of corrosion of a metal solution on graphite which would indicate that there will be no problem here. As far as the other conditions are concerned, graphite has excellent mechanical properties at the temperatures about which we are talking, 550°C, and we do not believe we shall get into any difficulties on this score. The third possibility is that there may be a penetration in through the graphite of fission product recoils, or something of that nature, due to bismuth uranium solution seeping in through the graphite. There is work being done in the United States and elsewhere, I believe, on producing fairly high purity graphite with rather impermeable surfaces, and this looks quite promising. Our evidence to date would indicate that it may be possible to build the reactor core of such material, in which case we would not anticipate any difficulty.

The CHAIRMAN: Mr. Swartout of the United States would like to ask Mr. Went a question on paper P/936.

Mr. SWARTOUT (USA): I should like to ask Mr. Went whether he has considered the problem of determining the material balance for uranium throughout his reactor while it is operating. Secondly, if he is planning to do this by analysis of samples withdrawn from the reactor, how will he obtain repre-

sentative samples? I might clarify my first question by saying that it is prompted by the difficulty we encountered with our first aqueous solution reactor in the United States in finding exactly at a given time where the uranium was throughout the rather complex system consisting of pipes and tanks and so on.

MR. WENT (Netherlands): Our suspension is only in the primary circuit and in the container vessel that is used for starting up and shutting down the reactor, and eventually for changing the temperature of the reactor. In principle, therefore, the idea is to have the uranium oxide content in the primary circuit. It is not completely constant through the primary circuit as the flow velocity in the tubes is much higher than the flow velocity in the reactor vessel itself, and you must combine that with the settling velocity. We intend to measure the over-all concentration separately in our reactor vessel by measuring the pressure difference between the inlet and the outlet of the reactor vessel. This pressure difference is related, of course, to the resistance of the flow through the reactor vessel, but it is also related in a very sensitive way to the specific weight of our reactor content, and the specific weight is mainly related to the uranium oxide content: that is so heavy compared with the heavy water.

Mr. ROLLIER (Italy): I should like to ask Mr. Williams what is the order of magnitude of the by-product polonium-210 produced in the liquid metal fuel reactor of the planned size.

Mr. WILLIAMS (USA): The amount of polonium —I am assuming that you are interested in its radioactivity—that is produced in this reactor when it is present in the reactor at equilibrium is of the same order of magnitude of activity as the fission products themselves. I do not have the number in terms of parts per million at my finger-tips, but it is something of the same order of magnitude of activity.

The CHAIRMAN: Mr. Liljeblad of Sweden would like to ask Mr. Williams a question.

Mr. LILJEBLAD (Sweden): I should like to ask Mr. Williams a question about the sodium graphite moderator reactor. If you use double cooling circuits, and the primary circuit in any case is highly radioactive, have you not contemplated completely omitting the canning of the fuel elements? Canning is costly, and there are not only the fixed costs to be considered; there are variable costs which have to be repeated every time the reactor is reloaded. There is also the problem of the temperature drop from uranium to the coolant. Our investigations in Sweden have indicated that if you use really purified sodium it does not chemically affect the uranium.

Mr. WILLIAMS (USA): The use of uncanned fuels has been considered by us. We have hesitated to put these into our first experimental machines, primarily because of the difficulty of maintaining mechanical tolerance of fuel elements in the fuel channel. The problem of getting long rods of uranium which will stay unwarped in the fuel chamber and maintaining the sodium flow pattern has not been solved by us and we have hesitated to do this initially. We agree there is a very interesting possibility of not having to can the fuel element at all and this is one of the experiments we hope to conduct with our experimental reactor after it is in operation.

Mr. SPINRAD (USA): In the paper P/938, the density of the solid appears to approach that of ceramic materials in bulk form. I assume that this packing density is not the actual bulk density and I would like to find out what the bulk density of the solid is in these flowing grain channels.

Mr. DE BRUYN (Netherlands): I do not know whether I got your question right, but the volume percentage of the filling is about 70 per cent. Does that answer the question?

Mr. SPINRAD (USA): I would assume that this 70 per cent is the sum filled with grains. What I was wondering about was, for example, the density of uranium oxide in this channel.

Mr. DE BRUYN (Netherlands): In the case of the homogeneous reactor we think we will use compound grains of beryllium oxide and uranium dioxide and the density of the grains will be about 3.2. It depends on the ratio of uranium and beryllium. In the case of the heterogeneous reactor the density of the particles will be 10.6.

Mr. SPINRAD (USA): We ought perhaps to enlarge on this point after the session.

Mr. DALTON (Australia): I would like to ask Dr. Williams if he is at all concerned about the loss of delayed neutrons in the core from the point of view of control and also from the point of view of their emission in the heat exchangers and pumps. I should also like to ask him the effect of filling possible cracks in the graphite with the fuel solution on the reactivity of the core.

Mr. WILLIAMS (USA): We realize that there will be fewer delayed neutrons in the reactor, but because of its high negative temperature coefficient we do not believe this will pose any serious problems so far as control of the reactor is concerned. This will produce radioactivity in the heat exchanger. As a matter of fact we ourselves have been amused in this reactor design in that the neutron flux in our heat exchanger for this reactor is the same as in our research reactor at Brookhaven in the maximum flux area.

Mr. DALTON (Australia): The other question I would like to ask you is: are you concerned about the effect of filling possible cracks in the graphite of the core with the uranium-bismuth fuel solution?

Mr. WILLIAMS (USA): We hope to prevent most of this, by using this impregnated regraphitized graphite. If there is a little penetration locally here and there I do not think that will be serious.

Session 12A

DESIGN OF REACTORS FOR POWER PRODUCTION

LIST OF PAPERS

Heavy Water Reactors for Industrial Power, Including Boiling Reactors

By H. P. Iskenderian, L. E. Link, M. Treshow, and J. M. West,* USA

Heavy water comes very close to meeting technical specifications for an ideal moderator material. Of the materials which are commonly considered as moderators for reactors, the slowing down power of heavy water is second only to light water. The combination of good moderating power and low absorption cross section for thermal neutrons gives heavy water a moderating ratio (slowing down power divided by absorption cross section) far greater than any other material.

Heavy water is the only moderator which can be seriously considered in a high performance power reactor using natural uranium. With reasonable care in the design to prevent excessive amounts of neutron-absorbing structural materials in the core, a large, natural uranium power reactor should be able to operate with a conversion ratio near unity. This high conversion ratio assures maintenance of reactivity for the high fuel burnup which is necessary for economic reasons.

The intermediate slowing down power and low absorption cross section of D_2O encourage core lattice design in which the volume ratio of D_2O to uranium is very high. Common values of this ratio for D_2O and H_2O reactor cores are in the neighborhood of 20 and 2, respectively. The small volume of uranium makes the core design, fuel handling, and control of the D_2O reactor much simpler than its H_2O counterpart. The small quantity of uranium and low absorption cross section of the moderator account for a prompt neutron lifetime more than 10 times as long in a D_2O reactor as in the H_2O type. This is a very important factor from the safety standpoint inasmuch as short period excursions need not be considered in D_2O reactors. Since a significant part of the capital expenditure for a nuclear power plant is usually associated with safety provisions, the relative safety of a D_2O reactor is expected to have an important bearing on the cost of electricity.

The chief disadvantage of D_2O as a moderator has been its high cost. A few years ago price quotations of up to \$100/lb were common. Estimates of the amount of D_2O required per megawatt of heat output varied from 200 to 1000 lb. At \$100/lb and with a 12 per cent annual carrying charge, the cost of this D_2O per kilowatt of saleable electricity amounted to between 1.4 and 7 mills. To this must be added additional hardware charges for minimizing D_2O leakage and charges for actual D_2O leakage. That these discouraging cost figures should represent a major deterrent to construction of D_2O-moderated power reactors was not surprising.

It is now clear that the cost of producing D_2O can be decreased markedly by mass production techniques. The present quotation for D_2O from large production plants in the USA is \$28/lb. At this price and with reactor designs which minimize the D_2O investment per unit of heat output, the direct charges against D_2O can be reduced to less than 1 mill per kilowatt hour of electricity produced. The cost of D_2O becomes significantly less than the cost of fuel in the D_2O reactor and very much less than the cost of fuel in reactors using other moderators in conjunction with enriched uranium.

Direct cycle boiling reactor power plants are particularly well suited to use of D_2O as moderator and coolant. The only D_2O required outside the reactor is a very small amount in the form of steam and condensate. In a pressurized water type the amount of D_2O required in external pipes, boilers, and pumps approximates that in the reactor itself.

The advantages of D_2O moderator are best realized in reactors of large physical size. To achieve criticality and operate at high temperature with natural uranium fuel, D_2O-moderated reactor cores must be 10 ft or more in diameter. When the reflector is taken into account, a pressure vessel having a diameter of at least 13 ft is required. For a boiling reactor generating 600 psi steam, the thickness of the pressure vessel wall would be about 4 in. For a pressurized watere reactor operating at 2000 psi in order to generate 600 psi steam in external boilers, the wall thickness of the pressure vessel would be so great that fabrication would be extremely difficult, if not impossible.

The choice of an optimum reactor design for power production depends upon the assumptions which must be made regarding availability of slightly enriched uranium and on the amount of power desired per reactor. If only natural uranium is available, D_2O reactors are necessarily large, and H_2O reactors are excluded entirely. A D_2O natural uranium reactor,

*Argonne National Laboratory.

which is large in physical size and capital investment, must produce large quantities of heat if the unit cost of this heat is to be reasonable. The design of a large boiling D_2O reactor, utilizing natural uranium fuel, capable of producing 1000 megawatts of heat will be described.

1000-Mw D_2O BOILING REACTOR

To obtain high heat outputs per unit of investment cost in uranium, D_2O, and reactor appurtenances, the uranium must be subdivided to provide a large amount of surface area. This subdivision would result in capture of too many resonance neutrons in the uranium, leading to a prohibitively large core size, if the individual fuel elements were not grouped in clusters for self-shielding purposes. By clustering, the portion of the core area through which cooling water must be circulated is also minimized.

Figure 1 shows a fuel element for the 1000-Mw D_2O boiling reactor. Natural uranium plates clad with an alloy of zirconium (zircaloy-2) are bound together in clusters surrounded by a 6-inch diameter zircaloy-2 tube. Each plate is 0.15 inch thick, including 0.04 inch of cladding. The water passage between plates is 0.35 inch wide. Additional area for coolant flow exists between the ends of the fuel plates and the zirconium tube.

The particular fuel assembly illustrated in Fig. 1 supports and spaces the fuel plates at the top. The spacing is maintained over the full length of the plates by corrugated plate aligners. The latter are pressed against the edges of the fuel plates by semicircular metallic springs and clips. The fuel plates are free to grow downward if they are inclined to do so as a result of irradiation and/or thermal cycling.

The length and total number of fuel assemblies in the reactor determine the power output. It is estimated that in a boiling reactor operating at 600 psia the power output can be about 0.28 megawatt per foot of length of the fuel assembly shown in Fig. 1. This corresponds to an average heat flux of about 160,000 Btu/hr/ft^2. Thus, for 1000 megawatts total output, a total of 3550 lineal ft is required. The total weight of uranium is 43 tons. For various reasons, which will become clearer as other features of the design are presented, this was divided into 295 assemblies, each having an active length of 12 feet.

Figure 2 shows a horizontal section through the reactor. The fuel assemblies previously described are located on a triangular lattice. The 8-inch spacing between assemblies represents a compromise between nuclear, mechanical, and economic factors. The distance between 6-inch diameter fuel assemblies could not be decreased much below 8 inches without complicating the mechanical design and increasing the resonance capture of neutrons to a point where the critical size of the core would be too large. To use a lattice spacing much larger than 8 inches would excessively lower the heat output per unit volume of the reactor and increase the capital charges for D_2O and reactor hardware per unit of heat output accordingly. For very large lattice spacings, formation of steam in the fuel clusters would increase the reactivity and present a real danger of an autocatalytic nuclear runaway.

The 295 assemblies on an 8-inch triangular lattice form a core about 12 feet in diameter. A 1.25-foot thick D_2O reflector surrounds the core. Y-shaped control rods are interspersed in the lattice as shown. A stainless steel shroud around the reflector acts as a thermal shield to absorb gamma radiation, thereby reducing thermal stresses in the pressure vessel, as well as serving as a partition to separate the upflow and downflow streams of cooling water. The thickness of the annular downcomer is 3.5 inches. The pressure vessel itself has an outer diameter of 16 feet and a wall thickness of 4 inches.

A vertical section through the reactor is shown in Fig. 3. Cooling water is pumped into the plenum chamber below the core through three 24-inch pipes at the rate of 75,000 gal/min. From the plenum, the water is distributed to the 295 fuel clusters, through which it flows with a velocity varying from 3 feet per second at the bottom to 16 feet per second at the top. This water is slightly below the boiling point as it enters the reactor. At the top of the fuel assemblies it contains about 80 per cent steam by volume.

Feedwater from the power plant condenser at a temperature of 90°F enters the reactor pressure vessel through a separate pipe and is distributed by a header system in the spaces between the fuel clusters at an elevation just above the bottom grid. This cold feedwater warms up gradually as it rises due to absorption of gamma rays and moderation of neutrons and to conduction of heat through the walls of the 6-inch diameter fuel tubes. By the time this water reaches the top of the core it will have increased in temperature to about 300°F. The average temperature of the moderator and reflector is about 200°F. By keeping the moderator relatively cool in this manner, the resonance capture and the leakage of neutrons are reduced, thereby giving more reactivity and a slightly higher conversion ratio. It is also advisable to keep this moderator below the boiling point to avoid flashing upon reduction of pressure in the steam dome due to an increase in the steam demand of the turbine.

Upflowing streams of water from the fuel tubes and the moderator mix above the core. This slightly subcooled water is then returned through the annular downcomer and the outlet pipes to the circulating pumps.

The reflector thickness below and around the core is 1.25 feet. A portion of the space between the bottom head of the pressure vessel and the lower grid is filled with steel to reduce the inventory of D_2O. Above the core the reflector thickness is increased to 3 feet in order to minimize the effect of turbulent

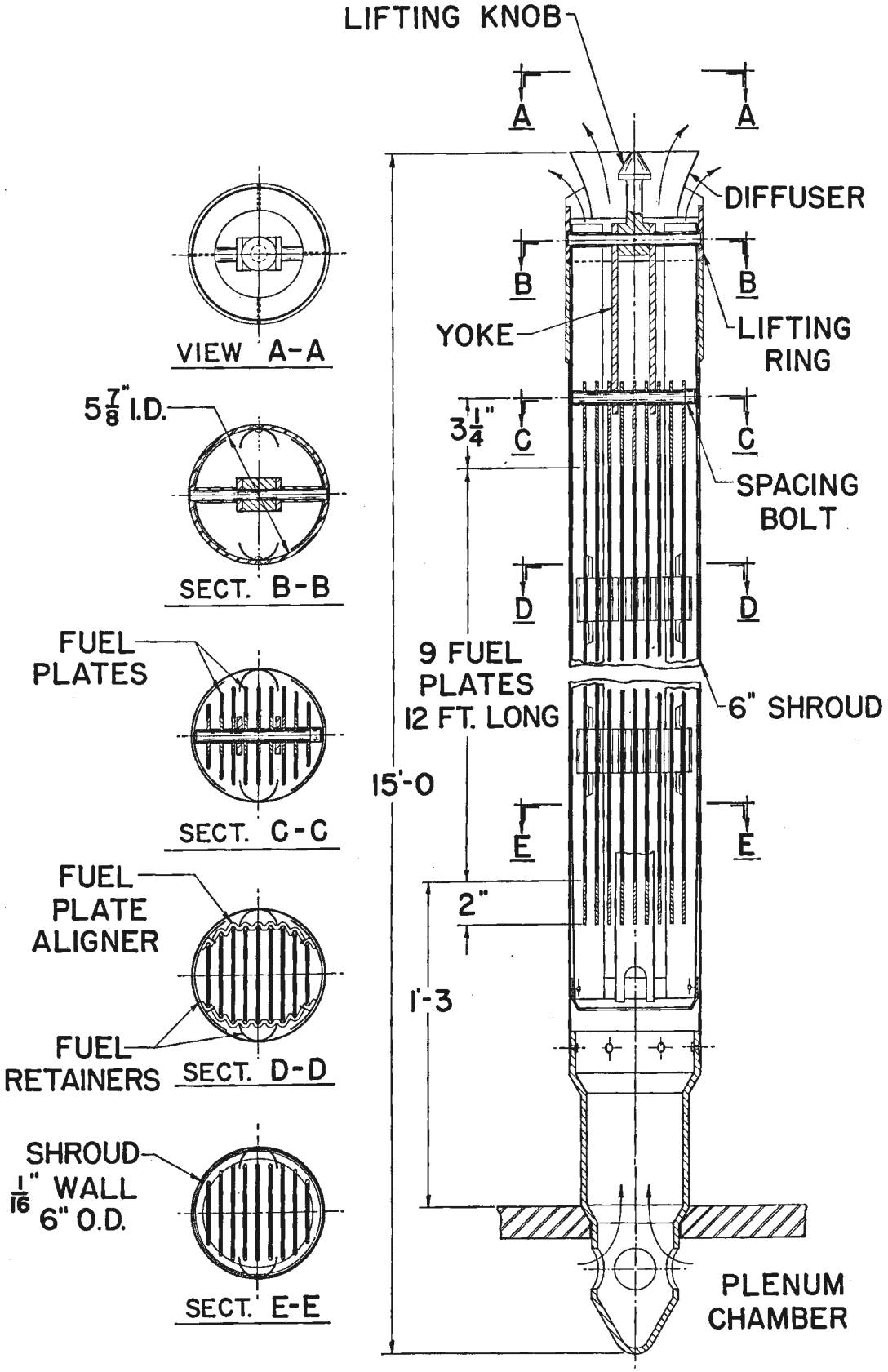


Figure 1. Fuel assembly for 1000-Mw D_2O boiling reactor

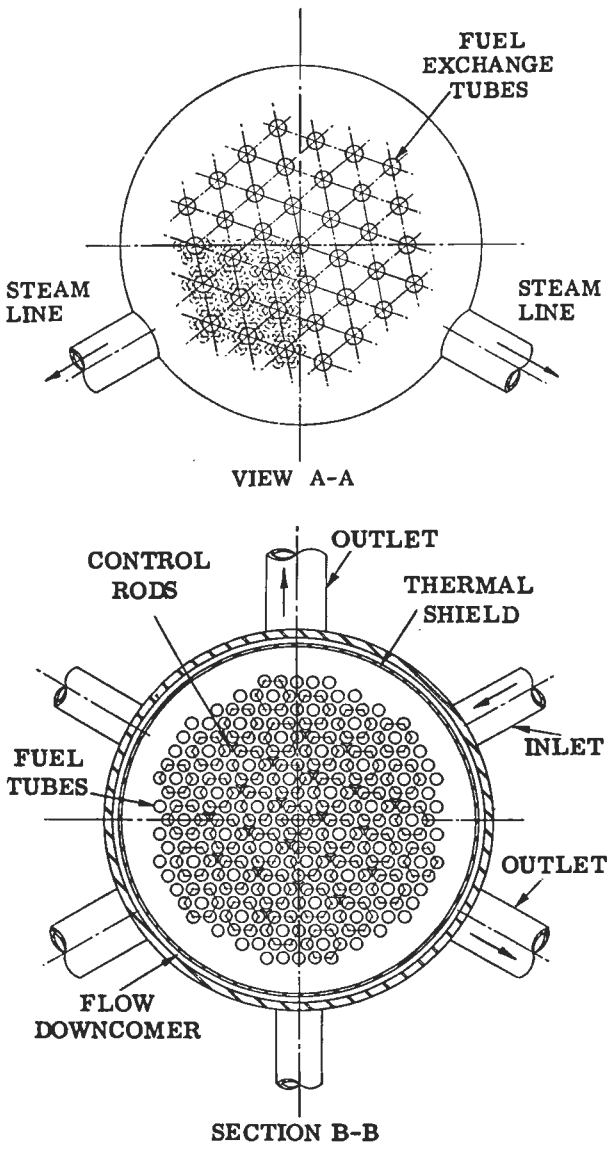


Figure 2. Plan view of 1000-Mw D_2O boiling reactor

surface conditions on reactivity. The total weight of D_2O in the reactor is about 100 tons. An additional 20 tons are held up in the power plant equipment and in reserve storage.

Extension shafts for the control rods pass through seals in individual thimbles to mechanical drive mechanisms located in a shielded room beneath the reactor. The D_2O leakage past the seals is drawn off and pumped back into the system. When withdrawn, the neutron-absorbing portions of the rods are above the core. Rapid insertion is accomplished, when required, by gravity supplemented by the steam pressure in the pressure vessel.

A large steam dome above the water line provides ample space for gravitational separation of moisture from the steam before it leaves the reactor through pipes located near the top of the dome. At 1000 Mw and 600 psia the rate of flow of steam is 2.9×10^6 lb/hr.

Since a bolted, leak-tight closure of such large diameter would be very difficult, the top head is shown welded to the pressure vessel proper. Fuel charging and discharging is accomplished through access holes in this top head. One access hole is provided at the center of each hexagon of the core lattice, providing access to seven fuel assemblies as shown in Fig. 2. The fuel loading and unloading device would have a double elbow mechanism which could be offset horizontally after passing through an access hole. In its offset position it could reach any of the six outer fuel tubes in a 7-tube group.

The general relationship of the reactor to other major plant equipment is shown in Figs. 4 and 5. Steam from the reactor passes through moisture separator drums before reaching the turbogenerators. The total gross electrical output of the turbogenerators is 270,000 kilowatts. After plant load requirements are taken into account the net saleable electricity is 250,000 kilowatts.

The conversion ratio in the 1000-Mw boiling D_2O natural uranium reactor has been calculated to be 0.90 with an initial excess reactivity of 2.3 per cent. Experimental determinations of conversion ratio in D_2O uranium lattices have given values on the order of 4 per cent higher than calculated. Long-term reactivity studies in which the effects of fission products and transmutation products are taken into account lead to the expectation that the fuel in this reactor can be irradiated to 10,000 megawatt days per ton without becoming subcritical. Some rearrangement of fuel and adjustments in total uranium content of the core, to take full advantage of the increase in reactivity of the fuel during the early part of its life to offset the decline which sets in later on, may be necessary to reach the 10,000 Mwd/ton goal.

An estimate of the cost of electricity from the 1000-Mw boiling reactor power plant is summarized in Appendix B. It is noted that a 12 per cent annual assessment against the D_2O in the system amounts to only 0.5 mill per kilowatt hour of electricity sold. If 5 per cent of the D_2O is lost in leakage per year, an additional charge of 0.2 mill/kwh is involved. The fuel cost per kilowatt hour totals 1.9 mills/kwh. Thus, the fuel cost far exceeds the D_2O cost.

The authors have followed the usual pattern of using natural uranium with D_2O moderator. The D_2O reactor appears at its best in the large sizes and correspondingly large power outputs which result from this approach. Even when smaller power outputs are desired, the D_2O reactor should not be ruled out without careful consideration. The only requirement is that slight enrichment of the uranium fuel be permitted. If enriched uranium is assumed to be available for other reactor types which may be under consideration, it follows that the smaller quantity of enriched uranium required for D_2O reactors should also be considered available. It will be shown that for heat outputs as low as 250 megawatts, the cost of electricity from a D_2O boiling reactor plant should

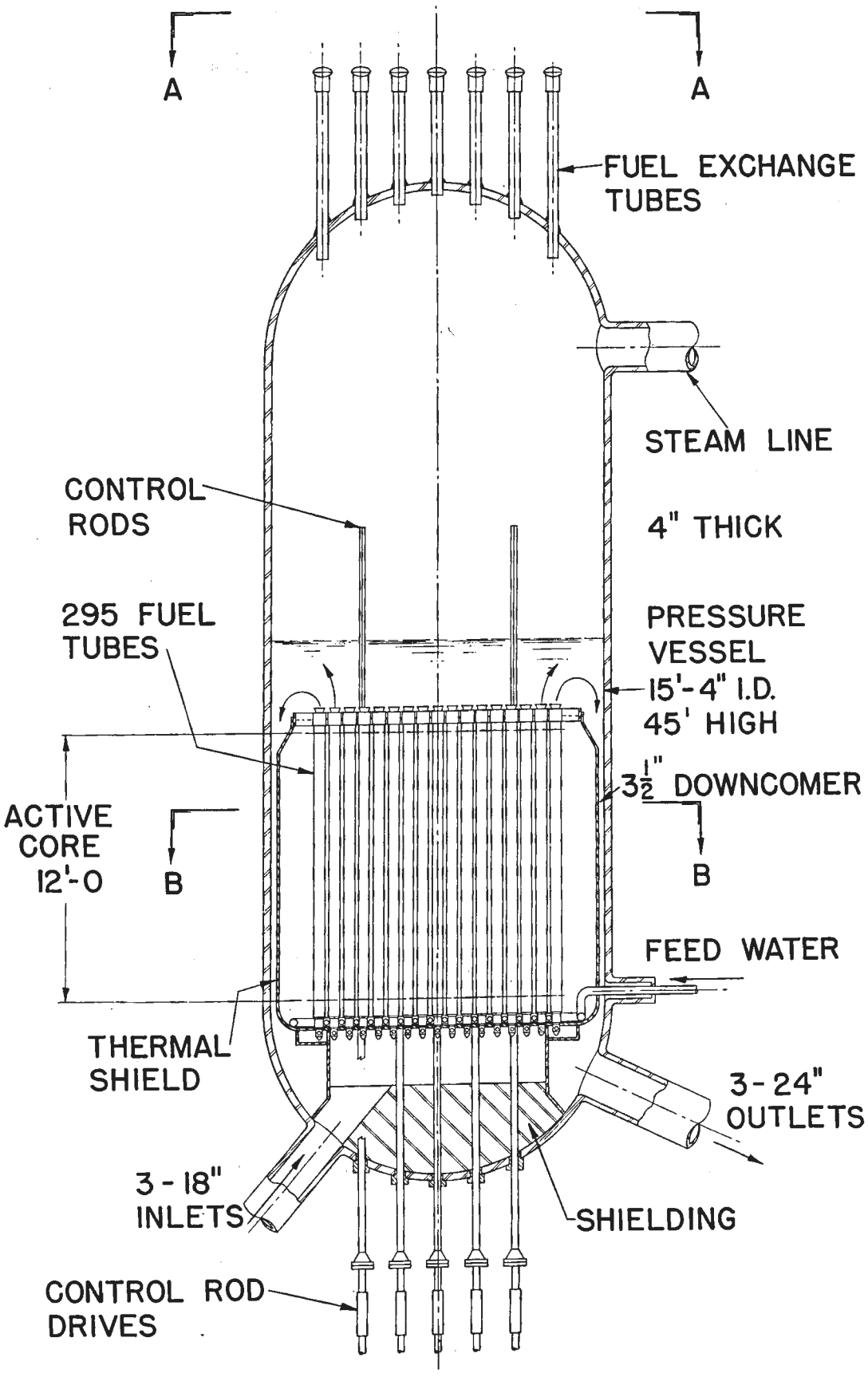


Figure 3. Vertical section through 1000-Mw D_2O boiling reactor

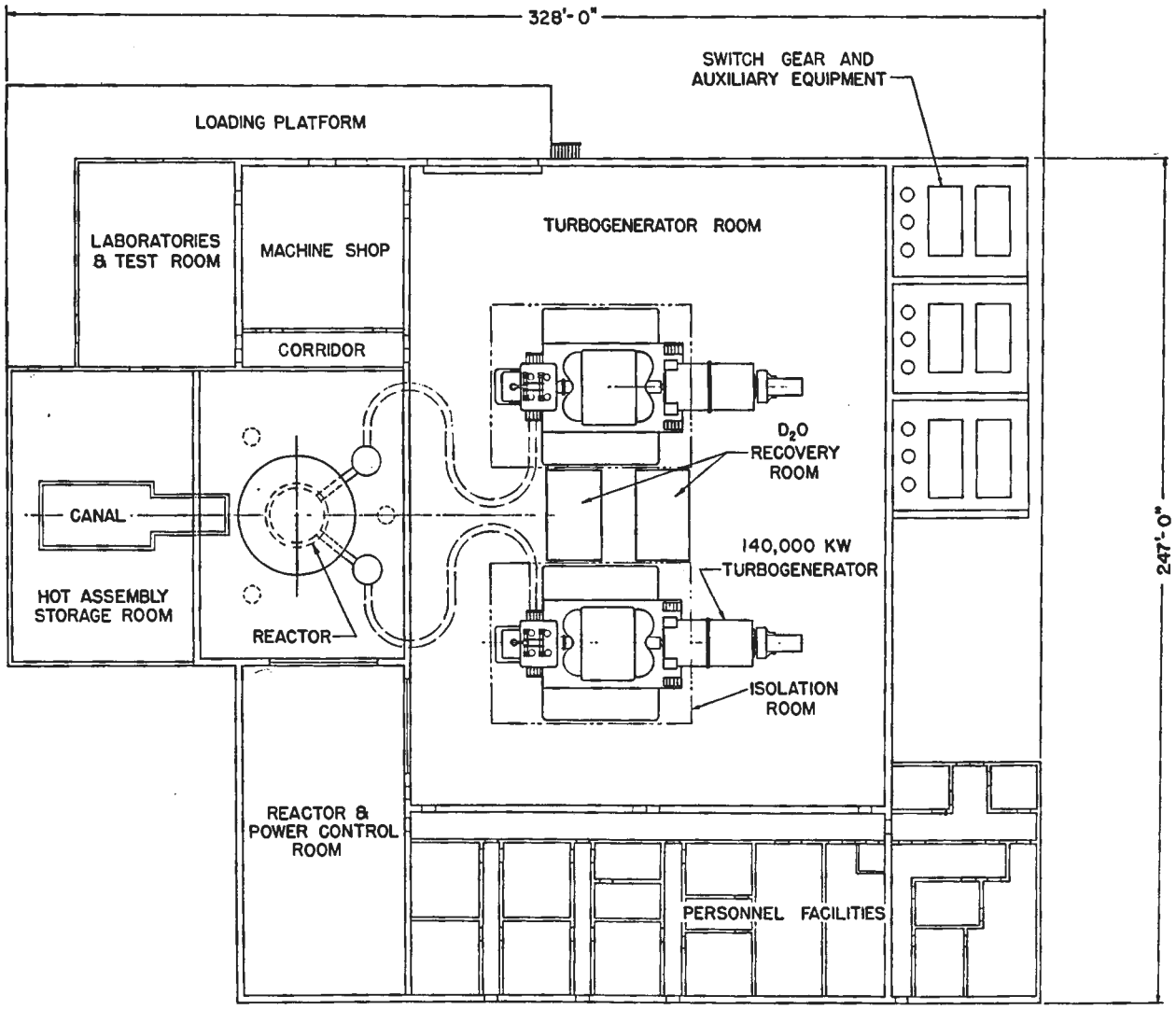


Figure 4. Plan view of central station 1000-Mw D_2O boiling reactor

be no greater than from an H_2O type of the same output.

The assumption that slightly enriched uranium is available gives considerably more freedom in the design of the core. For example, some departure from the high degree of clustering used with natural uranium fuel is permissible. The effect of slightly greater homogeneity of fuel and moderator is to decrease the resonance escape probability and require a corresponding increase in U^{235} content of the fuel to preserve reactivity. The conversion ratio increases slightly when this is done.

250-Mw D_2O BOILING REACTOR

A design study has been made on a 250-megawatt D_2O boiling reactor. The general design is identical with that previously described for the 1000-Mw reactor except that the size is reduced. A fuel element basically the same as shown in Fig. 1, but with an over-all diameter of 5 inches, rather than 6 inches, was used. These smaller clusters were spaced at 7-inch intervals on a triangular lattice. The core for the 250-Mw case is 7.5 feet in diameter and 7.5 feet high. The total uranium content was reduced in proportion to the power output, giving about 11 tons for the 250-Mw case. The outside diameter of the pressure vessel was decreased from 16 feet to 11 feet. The operating pressure was taken to be 600 psia as before.

Nuclear calculations for the 250-Mw core are described in Appendix A. The U^{235} content of the fuel is 0.92 per cent. The initial conversion ratio is 0.87 with an excess reactivity of 2.2 per cent. To provide the same degree of assurance that the fuel can be irradiated to 10,000 Mwd/ton as in the previous 1000-Mw reactor, the deficit in conversion ratio of 0.03 (0.90 — 0.87) is made up by adding fully enriched U^{235} fuel element "spikes" as the irradiation progresses.

Reference to Appendix B will show that the net result of decreasing the reactor to a 250-megawatt size and enriching the fuel was to increase the combined fuel and D_2O charges from 2.6 to the range of

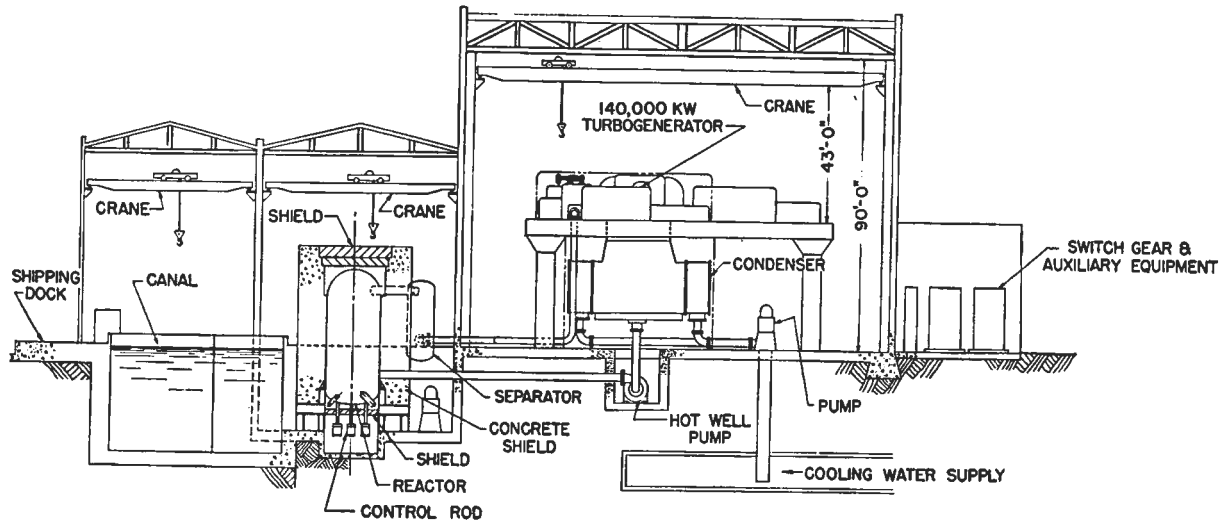


Figure 5. Elevation view of central station 1000-Mw D_2O boiling reactor

3.9 to 4.6 mills per kilowatt hour. That this 250-Mw D_2O reactor is still attractive from an economic standpoint will be shown by comparing with an H_2O boiling reactor having the same power output.

250-Mw H_2O BOILING REACTOR

Conceptual features of a 250-Mw H_2O boiling reactor operating with natural circulation are shown in Fig. 6. Rather marked differences from the D_2O reactor core are immediately apparent. Reactors using H_2O as moderator must have a much greater degree of homogeneity of fuel and moderator in order to keep the required enrichment at a tolerable level and prevent excessive hot spots. This complicates the design of the core and controls very considerably.

The plate type fuel assemblies illustrated in Fig. 6 were chosen to give minimum hydraulic pressure drop consistent with power density and heat flux limitations. In this design the power output per liter of coolant space in the core was taken as 40 kw/liter. This fixed the volume of the core. The diameter (9 feet) was made greater than the height (7 feet) because the power density obtainable with natural circulation is greater in short tubes than in long ones.

Each fuel assembly in the H_2O reactor contains five zirconium-clad uranium plates having a thickness of about 0.5 inch. These assemblies, having overall cross sectional dimensions of $6\frac{1}{2} \times 6\frac{1}{2}$ inches, are placed in individual 7×7-inch cells. Control rod assemblies containing fully enriched uranium in the lower portion and depleted uranium in the upper portion occupy cell locations as shown. The number of control rods required is reduced and the neutron economy improved by substituting fertile absorber for fissionable material instead of merely inserting parasitic absorber as is frequently done.

The maximum heat flux in the H_2O reactor is 360,000 Btu/hr/ft^2. The steam generation rate is 750,000 lb/hr at 600 psia and 250 megawatts. About 30 pounds of water are circulated for each pound of steam leaving the fuel channels, corresponding to a total coolant flow rate of 55,000 gpm. The volume percentage of steam in the exit coolant is about 50 per cent.

The total quantity of uranium in the H_2O reactor core is 72 tons. This is enriched to 1.1 per cent U^{235} content. The conversion ratio is estimated to be about 0.84. Additional U^{235} is added in the form of spikes to maintain reactivity throughout a 10,000 Mwd/ton exposure period.

The economic analysis presented in Appendix B shows that the cost of fuel per kilowatt-hour of electricity from the 250-Mw natural circulation H_2O boiling reactor plant is 6.3 to 8.6 mills. These values are more than double the combined fuel and D_2O costs for the 1000-Mw plant and 2.4 to 4.0 mills/kwh higher than the 250-Mw D_2O plant.

The foregoing comparison utilized forced circulation cooling for the D_2O reactors and natural circulation for the H_2O reactor. This permits a higher power output per unit of fuel in the D_2O cases. The clustered arrangement of fuel in the D_2O reactors is much better suited to forced circulation and to operation with a high volume percentage of steam in the fuel clusters without complicating the control of the reactor. It appears that even the H_2O reactor can profit significantly by use of forced circulation, however, as shown by Case 5 in Table IV of Appendix B.

For the forced circulation H_2O case it was assumed that the same power (250-Mw) could be obtained from a core volume only one-half as large. The core could be a cylinder 6.5 ft in diameter and 6.5 ft high. By making the fuel plates only one-half as thick, the maximum heat flux remains the same. The amount of water circulated remains the same (55,000 gal/min) to preserve the same steam content in the exit coolant.

The U^{235} content of the smaller forced circulation H_2O core is estimated to be about 1.15 per cent. The

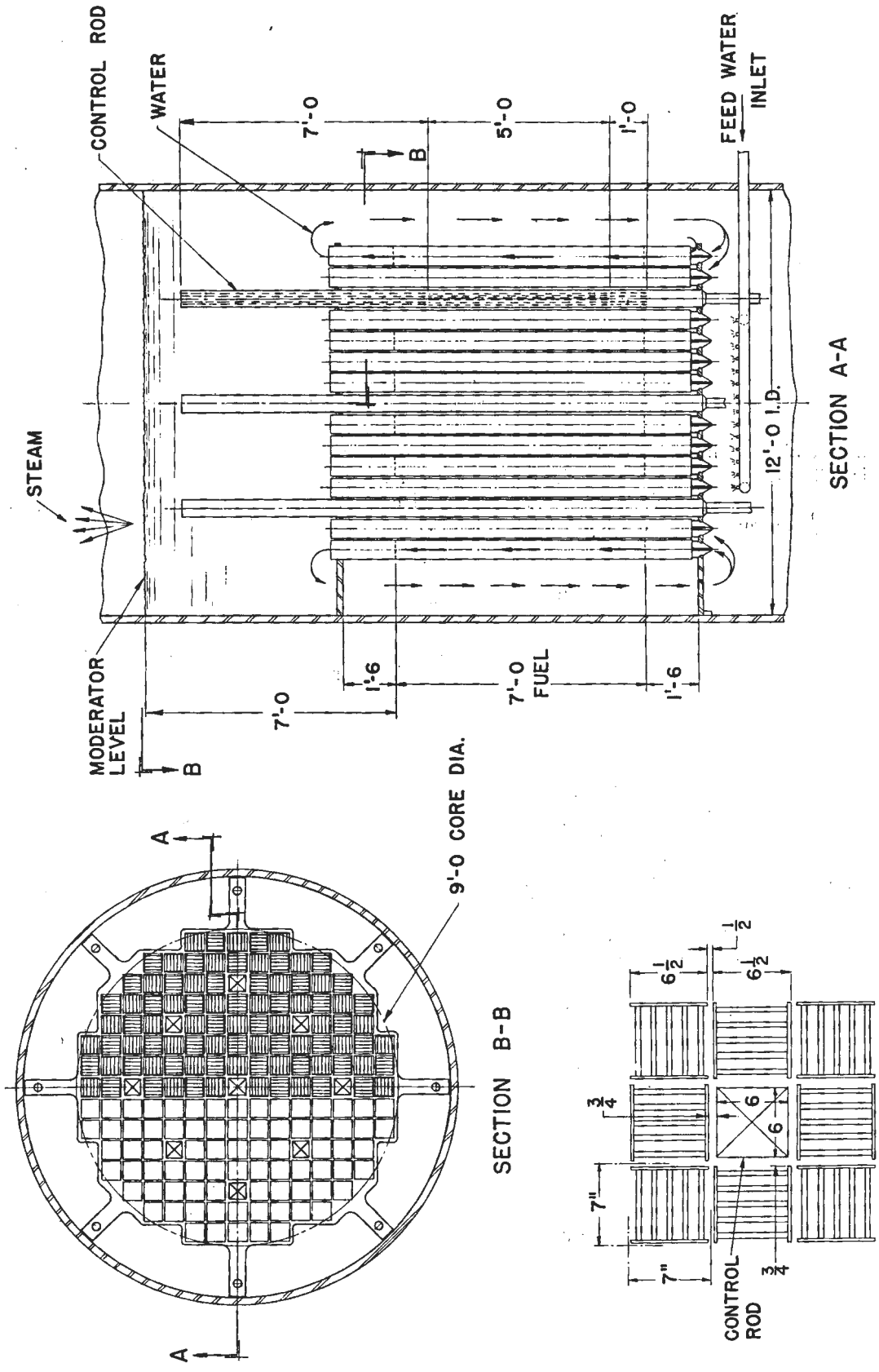


Figure 6. 250-Mw light water boiling reactor

conversion ratio would be about 0.82. The total tons of uranium would be 36. The outside diameter of the pressure vessel would be around 9 feet.

The fuel cost for the 250-Mw forced circulation H_2O reactor is calculated to be 5.2 to 7.3 mills/kwh (Appendix B). This is an improvement of 1.1 to 1.3 mills/kwh relative to the natural circulation case. Part of this reduction in fuel cost is nullified by the additional capital and operating costs associated with the forced circulation equipment. The net improvement due to forced circulation appears to be 0.7 to 0.9 mill/kwh.

From the foregoing comparisons it is evident that a 1000-Mw D_2O reactor can produce steam at a cost of only about one-half that from a 250-Mw H_2O reactor. An increase in the size of the H_2O reactor to 1000 Mw would not reduce the fuel cost per unit of steam significantly, since the required enrichment does not decrease much with further increases in core size.

Additional expenses over and above the \$25/kw used in the 250-Mw D_2O system may be required to keep leakage of D_2O in the external power equipment to the 5 per cent per year level assumed in the economic analysis. It is to be noted, however, that a 1 mill/kwh advantage in the reactor itself requires a capital expenditure of \$58/kw of electrical output to offset it. Encouraging experience with seals for D_2O systems in the USA lead the authors to believe that the leakage can be controlled for a much smaller expenditure than \$58/kw.

For very small power outputs, H_2O reactors are undoubtedly superior to D_2O types. This stems from the fact that the amount of D_2O per unit of power output and the enrichment of fuel increase, and the conversion ratio decreases rapidly, as the size of the D_2O reactor decreases. The authors believe that in the neighborhood of 200 megawatts of heat output, D_2O and H_2O reactors are roughly equivalent from an economic standpoint. In larger sizes the D_2O reactors are superior; in smaller sizes they are inferior to H_2O reactors.

Building medium-sized D_2O reactors requiring slightly enriched fuel does not necessarily imply continued dependence on a source of separated U^{235}. Plutonium can be separated chemically from the irradiated fuel and recharged into the reactor. A small amount of the irradiated uranium which is depleted in fissionable isotopes could be discarded. The remainder could be mixed with natural uranium and spiked with the recovered plutonium to form a new fuel charge. This recycling could be repeated as many times as desired, with a small amount of natural uranium being the only required feed material. A chemical separations plant, and facilities for fabricating alpha-active fuel would be required in this procedure, but the cost of these facilities should be small compared with that of a plant for separating fissionable U^{235}.

D_2O-H_2O MIXTURES

Of some interest is the fact that the required enrichment for small D_2O reactors can be decreased significantly by adding a small percentage of H_2O. In the sample case presented in Appendix A of this paper 10 per cent of H_2O was added to the 250-Mw D_2O reactor previously described. The required enrichment decreased from 0.92 per cent to 0.85 per cent. The conversion ratio dropped from 0.87 to 0.77, however, and this would be a serious handicap if integrated exposure of fuel is to be limited by reactivity rather than irradiation damage considerations. The D_2O-H_2O reactor would require relatively heavy spiking with U^{235} beyond 6000 Mwd/ton exposure. When compared with the pure D_2O case for 10,000 Mwd/ton fuel exposure, the (fuel + D_2O) cost is increased by 0.5 to 0.7 mill/kwh. For much smaller cores and with the percentage of H_2O optimized, the cost of power could actually be decreased by adding H_2O. Adding H_2O also provides a control over the temperature and steam void coefficients, which would otherwise be quite high in small D_2O reactors.

APPENDIX A. PHYSICS OF D_2O BOILING REACTORS

The physics calculations of the boiling reactors listed in Table II are obtained by simple diffusion theory.

The equivalent cell of circular cross section of these reactors (Fig. 7) consists of a fuel zone "0"

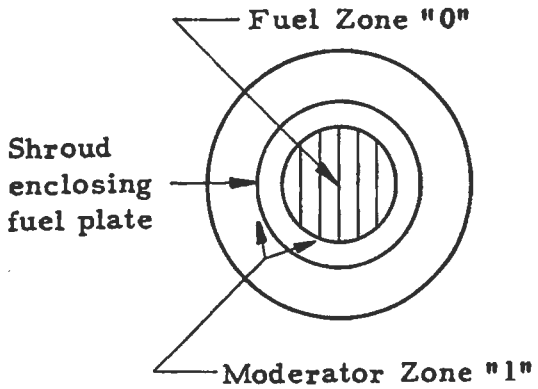


Figure 7

confined to the fuel plates, and an outer moderator zone "1", which consists in part of the coolant, but mainly of the moderator.

In determining the microscopic constants ϵ, p, f, it is assumed that the fuel and all zirconium metal are homogenized within the coolant in the fuel zone. Disadvantage factors in each zone for thermal and resonance neutrons are then obtained by diffusion theory. In obtaining p, self shielding to resonance neutrons by the parallel set of fuel plates is accounted for.

The thermal diffusion area, L^2, is obtained in the usual manner, making use of the thermal disadvantage factors referred to above. The Fermi age τ is taken to be that in the pure moderator-coolant. (In

view of the small amounts of zirconium present in the reactor, and since scattering by zirconium enhances collisions within the moderator; also, in view of the slowing down of neutrons due to inelastic collision by uranium atoms, the metals are taken to be the equivalents of the coolant-moderator in obtaining τ.)

On the basis of previous long term reactivity studies it is estimated that a reactor with a calculated initial conversion ratio of 0.90 and an excess reactivity of 2% is capable of remaining critical for an average fuel exposure of 10,000 Mwd/ton. These studies took into account the formation of the series $Pu^{239,240,241}$ and fission products poisoning. Comparison of the three reactors listed in Table II is based on this exposure of 10,000 Mwd/ton; i.e., allowance is made for the addition of extra U^{235} fuel in the form of "spikes" as required, to maintain criticality over the exposure period.

The effect of adding small amounts of H_2O to the D_2O moderator to reduce leakage flux in the smaller size reactors is illustrated in the last column of Table II. The benefits from adding small percentages of H_2O would be greater in reactors smaller than the 250-Mw unit listed. Furthermore, the 10% H_2O taken for this sample calculation is undoubtedly not optimized. This 250-Mw unit would remain critical over an exposure period of about 6000 Mwd/ton, and it is of interest if the exposure of fuel is to be limited to such a value due to irradiation effects on fuel, or other reasons.

The distribution of Δk_{eff} from cold to hot to boiling, and for (Xe + Sm) poisoning, is also shown in Table II, for the 1000-Mw D_2O reactor. The effect of adding small amounts of H_2O to the D_2O coolant-moderator would be to reduce the magnitudes of these Δk's.

APPENDIX B. ECONOMICS

The cost estimates presented are concerned mostly with the influence of fuel and moderator. Capital cost estimates are also included to arrive at approximate total power costs from reactor plants which might be built in the near future. The basic data are presented in Tables I, II and III and the factors in the power production costs are given in Table IV.

The fuel assumed for all these reactors is the multiplate type illustrated in Figs. 1 and 6. The cladding metal and the structural metal required to assemble the plates is Zircaloy-2. It is expected that \$30/lb of uranium should cover the fabrication cost for the quantity of fuel required for any of these reactors. A large portion of this cost is for zirconium. The \$30/lb fabricating cost is applied uniformly to all reactor types presented. The unit costs for the grades of uranium required are based on a price schedule of \$18/lb of natural uranium metal and \$15 to \$30/gram for uranium metal having 90 per cent U^{235}. The costs of the slightly enriched fuels are calculated on the basis of the proper blends of natural uranium and the 90 per cent U^{235} grade. This method of calculating the cost of slightly enriched uranium gives values higher than would be the case where slightly enriched uranium is obtained directly from a gaseous diffusion plant. The two values of fuel costs shown in Table

Table I. Summary of Design Data

Case number	*1*	*2*	*3*	*4*	*5*
Reactor power, Mw (heat)	1000	250	250	250	250
Moderator	D_2O	D_2O	90% D_2O 10% H_2O	H_2O	H_2O
Core diameter (ft)	12	7.5	7.5	9	6.5
Core height (ft)	12	7.3	7.3	7	6.5
Pressure vessel inner diameter (ft)	15.2	10.7	10.7	12	9
Fuel tube dimensions (in)	6 (dia.)	5 (dia.)	5 (dia.)	6.5 × 6.5	6.5 × 6.5
Lattice	Triangular	Triangular	Triangular	Square	Square
Lattice spacing (in)	8	7	7	7	7
Number of fuel assemblies	295	151	151	168	90
Total uranium (short tons)	43.4	10.8	10.8	72	36
U^{235} content of uranium	Natural	0.0092	0.0085	0.0110	0.0115
Initial conversion ratio	0.901	0.869	0.766	0.84	0.82
Fuel plate thickness (in)	0.15	0.15	0.15	0.50	0.50
Fuel clad thickness (in)	0.02	0.02	0.02	0.02	0.02
Water channel thickness (in)	0.35	0.35	0.35	0.7	0.3
Maximum heat flux (Btu/hr/ft²)	400,000	400,000	400,000	360,000	360,000
Coolant flow rate (gal/min)	75,000	19,000	19,000	55,000	55,000
Average coolant density (gm/cm³)	0.66	0.66	0.654	0.64	0.64
Average moderator temperature (°F)	200	200	200	486	486
Exit steam voids (%)	80	80	80	50	50
Steam pressure (psia)	600	600	600	600	600
Steam temperature (°F)	486	486	486	486	486
Gross electric output (Mw)	270	67.5	67.5	67.5	67.5
Net saleable electricity (Mw)	248	62	62	62.5	61

Table II. Operating Characteristics of D_2O Boiling Reactors

	1000-Mw reactor 6" fuel tubes, 8" lattice	*250-Mw reactors 5" tubes, 7" lattice*	
Moderator	*Pure D_2O*	*Pure D_2O*	*10 % H_2O + 90% D_2O*
ϵ	1.029	1.030	1.030
p	0.8364	0.7974	0.8592
f	0.932	0.942	0.874
η	1.3362	1.4531	1.4179
$k\infty$	1.0717	1.1241	1.0968
U^{235} content	0.00714	0.0092	0.0085
$\tau(cm^2)$	184	186	137
$L^2(cm^2)$	86	75	53
$(1 + L^2B^2)e^{\gamma B^2}$	1.047	1.100	1.073
k_{eff}	1.023	1.022	1.022
Initial conversion ratio	0.901	0.869	0.766
Core diameter, ft	12	7.5	7.5
Core height, ft	12	7.3	7.3
Tank inner diameter, ft	15.25	10.75	10.75
Number of tubes	295	151	151
Reflector thickness, ft			
Radial and below core	1.25	1.25	1.25
Above core	3.0	3.0	3.0
Tons of U	43.4	10.8	10.8
Moderator Temperature, °F*	200	200	200
Coolant density (gm/cm^3)†	0.66	0.66	0.654
Δk Distribution in the above 1,000=Mw D_2O Unit			
Δk_{eff} (cold − hot)	= −0.0200		
Δk_{eff} (hot − boil)	= −0.0094		
Δk_{eff} (Xe + Sm)	= −0.0377		
Total Δk	= −0.0671		

*Average moderator temperature outside of fuel tube.
†Average density of coolant moderator inside of fuel tube.

IV result from the two different costs used for 90 per cent U^{235}.

All fuels are evaluated on the basis of a fuel charge receiving 10,000 Mwd/short ton irradiation before discharge. The 1000-Mw D_2O reactor has a sufficiently high conversion ratio to achieve this without additional fissionable material being added. All the others require some U^{235} addition to reach 10,000 Mwd/short ton. It is assumed that this additional fissionable material is added in small portions in the form of fully enriched uranium in fuel plates (spiking fuel). In addition, it is assumed that a fuel element can tolerate a 50 per cent burnout of the contained spiking U^{235}. Consequently, for each gram burned, an additional gram must also be fabricated and then recovered after irradiation. It is estimated that in this recycle process the chemical processing and refabrication costs amount to $30/gram of U^{235} spike burned.

The initial charge of fuel is treated as an inventory item bearing an annual charge of 12 per cent. The charge for burnout of the U^{235} is treated as an operating cost. No credit is allowed for the irradiated fuel. It is assumed that the cost of separating the residual fertile and fissionable material from the fission products is equal to the value of the separated products. This appears to be a fair assumption for high burnup of slightly enriched fuels.

The D_2O price of $28/lb is taken from current estimates for large production plants. The D_2O held up in the power plant system is an inventory item with a 12 per cent annual charge assessed against it. The annual D_2O loss is somewhat arbitrarily assigned at five per cent of a system's holdup. Experience with D_2O research reactors indicates that this figure is a reasonable one.

The capital operating costs are taken as averages of industrial team estimates. Since the variation of capital cost with plant size was obtained by the industrial teams for reactors differing materially only in size, application of the results to the different types of systems discussed in this paper cannot be claimed to have a high reliability. Because of the necessity for special precautions to prevent leakage or contamination of D_2O, the D_2O power plants are assumed to cost $25 more per kilowatt than an H_2O plant of the same capacity. A 12 per cent annual charge is assessed against all capital costs.

Table III. Operating and Economic Data

Plant data					
Case	*1*	*2*	*3*	*4*	*5*
Reactor power, Mw	1000	250	250	250	250
Irradiation level, Mwd/ton	10,000	10,000	10,000	10,000	10,000
Fuel required, tons/year	29.2	7.3	7.3	7.3	7.3
Spiking required, gm 100% U^{235}/ton of U	—	400	1600	700	1000
Systems costs, $/net kw	250	450	450	400	425
Annual fixed charges					
Capital, %	12	12	12	12	12
Inventory, %	12	12	12	12	12
Cost of uranium metal, $/lb of U	18	34 – 50	29 – 39	47 – 77	51 – 84
Fuel fabricating cost, $/lb of U	30	30	30	30	30
Cost of 90% U^{235} metal, $/gm	—	15 – 30	15 – 30	15 – 30	15 – 30
Operating labor and maintenance $/year	1.5×10^6	10^6	10^6	10^6	10^6

Table IV. Power Production Costs, Mills per Net kwh

Reactor Number	*1*	*2*	*3*	*4*	*5*
Capital					
Plant	4.2	7.7	7.7	6.8	7.2
Initial fuel	0.3	0.4 – 0.5	0.3 – 0.4	3.1 – 4.2	1.6 – 2.3
D_2O inventory	0.5	0.7	0.6	—	—
Operating					
Labor and maintenance	0.9	2.3	2.3	2.3	2.3
Fuel, low U^{235}	1.6	2.2 – 2.7	2.0 – 2.3	2.6 – 3.6	2.8 – 3.9
Fuel, spiked	—	0.3 – 0.4	1.3 – 1.8	0.6 – 0.8	0.8 – 1.1
D_2O loss	0.2	0.3	0.2	—	—
Total—fuel + D_2O	2.6	3.9 – 4.6	4.4 – 5.3	6.3 – 8.6	5.2 – 7.3
Grand total	7.7	13.9 – 14.6	14.4 – 15.3	15.4 – 17.7	14.7 – 16.8

It should be remembered that any economic analysis contains elements tailored to rather specific situations. The 12 per cent annual capital and inventory charge is believed to be reasonable on the average but could vary considerably for particular situations. The uranium metal costs, fuel fabrication costs, heavy water costs, and separations costs are largely either assumed or are current estimates. The capital costs were estimated without the benefit of construction experience on any large nuclear power plants. With the development of the nuclear power industry certain of the elements of cost used in this paper could change radically. A change in the price of fabricated fuel will naturally affect the power production cost for all reactor types. However, the cost advantage of the D_2O reactor will remain unless one or more of the following conditions exist:

1. Uranium price and fabrication cost become extremely low.
2. D_2O price becomes very high.
3. D_2O losses or contamination with H_2O become high.

A Boiling Homogeneous Nuclear Reactor for Power

By A. I. Alichanow, W. K. Zavoisky, R. L. Serduk, B. W. Ershler and L. J. Suvorow, USSR

1. INTRODUCTION

The use of atomic energy for peaceful purposes presupposes the possibility of a broad industrial utilization of nuclear power plants, which is only possible in the event of their being simple in design, relatively cheap, and highly reliable in operation. We are of the opinion that these conditions are, to a great extent, satisfied by homogeneous reactors with water as coolant and heat transfer medium.

However, the diverse designs of homogeneous reactors, described in the literature, working with a solution or suspension of uranium compounds in a moderator, have a number of disadvantages precisely from this point of view. They demand the use of absolutely reliable pumps for pumping the active mixture through the reaction zone and the heat exchanger; the repair of these pumps must be carried out by remote control, which greatly complicates their operation. Further, the power obtained from a kilogram of fissionable material in such reactors is not high, since a considerable part of all the uranium loaded fills up the circulation circuit and the heat exchanger and does not participate in the reaction. Finally, when the heat exchanger utilizes a heat transfer medium which has been activated by neutrons, radioactive steam gets into the turbines, which is a great inconvenience.

The present work describes a reactor scheme in which these difficulties are to a great extent eliminated; this is a homogeneous pile which utilizes a boiling homogeneous heat transfer medium.

Such a pile makes unnecessary the pumps for pumping the working medium; next, the heat in such a pile is removed directly from the reaction zone by steam, which reduces the total amount of uranium, since the active mixture, in this case, is not pumped through the heat exchanger. Finally, secondary steam entering the turbine is not active.

The moderator in the pile under consideration is natural or heavy water, and the fuel is either pure fissionable material, enriched uranium, or natural uranium. In the last case, only heavy water may be used. Uranium may be distributed in the moderator in the form of a salt solution or in the form of a suspension of oxide powder.

Original language: Russian.

The boiling power reactor is equipped with a thorium reflector for fuel breeding. Below we shall speak of a homogeneous reflector filled with heavy water containing a thorium suspension.

The designing of a boiling homogeneous power reactor required the solution of a number of physical and engineering problems which are of general interest. Some are described in the present paper.

2. THE ACTIVE MIXTURE

In a homogeneous reactor working on a pure fissionable material, the core contains several kilograms of uranium per ton of water. In a reactor working on enriched or natural uranium, the content of uranium reaches several hundreds of kilograms per ton of moderator. In the reflector, the concentration of the suspension is several hundreds of kilograms of thorium per ton of heavy water.

Experience shows that suspensions, even in the case of large concentrations, retain sufficient mobility. They are easily poured and can be transported through pipes differing but slightly in this respect from water.

Suspensions passed through a colloid mill settle slowly. The shifting speed of the interface depends on the size of the particles, the concentration of the suspension and the viscosity of the water. In the case of sufficiently intensive boiling, bubbling or circulation of the steam, the suspensions are stable and do not settle. The quantity of steam production which is required to retain the uranium oxide powder in a suspended state depends, all other conditions being equal, on the form of vessel. To maintain a suspension of uranium oxide in a suspended state by boiling, three times as rapid consumption per sec of steam is required as for the suspension of thoria.

A study was made of the time behaviour of suspensions as a function of temperature. An increase in the speed of settling was observed when the suspension of uranium oxide was boiled for several hundreds of hours at atmospheric pressure. Holding the suspension in water at a temperature of 210° (without boiling) for 20 days does not bring about a considerable change in the speed of settling.

Further experiments were conducted to investigate the precipitation of the uranium oxide powder on stainless steel in contact with the suspension. When stainless steel was in contact with the nonboiling

suspension for several hundreds of hours at 210°, precipitation of the powder onto the surface of the steel was not observed. The experiments were conducted with the suspension motionless, and also when the suspension was continuously being stirred.

In experiments with a boiling suspension conducted over a period of 20–25 days at atmospheric pressure, only an insignificant precipitation and adhesion of the powder to the plates of stainless steel placed in the suspension was observed. After a given layer of powder has accumulated on the surface of the stainless steel, further adhesion ceases. When the plate of stainless steel was situated on the interface between suspension and steam, a thick layer of powder adhered to the plate.

The results obtained imply the possibility of using suspensions of uranium oxide and thoria for a boiling nuclear reactor.

3. PURIFICATION OF STEAM AND WORKING MEDIUM

It is highly desirable that the steam emerging from the reactor should not contain active impurities. Since the steam always carries with it from the reactor drops of water containing fission fragments and fissionable material powder, it is necessary to purify the steam as it emerges from the reactor. For this purpose an apparatus was designed for the purifying of water steam, the degree of purification being a factor of 10^{10}.

As the reactor operates, the water partially decomposes and an explosive mixture accumulates in the assembly. It is convenient to remove the detonating mixture from the steam. A catalytic chamber is placed between the steam purification apparatus and the heat exchanger. The catalytic removal of the detonating mixture has been studied in detail in engineering practice.

The gaseous fission fragments, including Xe^{135}, Xe^{131}, and Kr^{79}, that accumulate in the reactor are removed from the heat exchanger together with part of the steam. The steam is then condensed and returned to the system, while the gases are disposed of.

The fission fragments that remain in the working medium may be disposed of by periodically removing a certain part of the medium for reprocessing. Experiments in the elimination of rare earth impurities from uranium oxides showed that it is possible to attain a degree of purification, where harmful losses of neutrons by capture do not exceed 1–2%.

4. MODERATOR DENSITY

A knowledge of the density of the active mixture is necessary for a determination of the critical size of a reactor. In a boiling reactor, this value is not determined uniquely by the temperature, because the moderator is a mixture of steam and water. In the first approximation, we may speak of the mean moderator density for the whole volume of the reactor. A boiling reactor may operate either with an unorganized or an organized circulation of the active mixture.

In the first case, the suspension is situated in the lower part of the reactor, whereas the steam is in the upper part, above the evaporation mirror. In the case of organized circulation, the medium circulates in the external or internal circulation circuit. In this case, there is no evaporation mirror inside the reactor.

In a reactor with unorganized circulation, the mean density of the moderator is lower, and the critical size greater than in the case of a reactor with an organized circulation. However, in the first case there is no circulation circuit and therefore all of the fissionable material is retained in the core, which may appreciably enhance the heat removal per kilogram of fuel.

In order to determine the moderator density in the first approximation, methods described in the technical literature which are applied for calculating circulation in ordinary steam boilers were used (Central Boiler Turbine Institute, Calculation Standards of Water Circulation in Steam Boilers. Mashgiz, 1950).

As is well known, the density of the steam-water emulsion depends (all other conditions being equal) on the diameter of the vessel, in which the water is heated. In ordinary steam boilers the diameter of the pipes is 15–20 times less than the diameter of the reactor. We carried out experiments which showed that as the diameter of the vessel increases, the density of the steam-water mixture increases only slightly, after which it remains constant. The experiments were carried out by the bubbling method, and the density was determined by measuring the absorption of gamma rays from a cobalt source.

Further, the density of the steam-water mixture was determined as a function of the circulation velocity of the steam and water through the reactor. By comparison with the results obtained, it was possible to predict the density of the steam-water mixture in the top part of the reactor where steam formation is relatively slight and where a large portion of the steam enters from below the reactor. The data obtained make it possible to carry out calculations up to pressures of 120 atmospheres.

If steam is formed in the bulk of the liquid, the moderator density differs slightly from that obtained in the case of bubbling. Therefore, determination of the density in the lower layers of the reactor was made on the basis of experimental data concerning volume heating of water.

5. PLANT DIAGRAM

Figure 1 depicts two types of boiling homogeneous reactors and their corresponding plant diagrams.

Figure 1a shows a bare reactor diagram with an organized external circulation of the medium.

The steam-water mixture passes from reactor (1) to separation vessel (3) through the up-pipe (2).

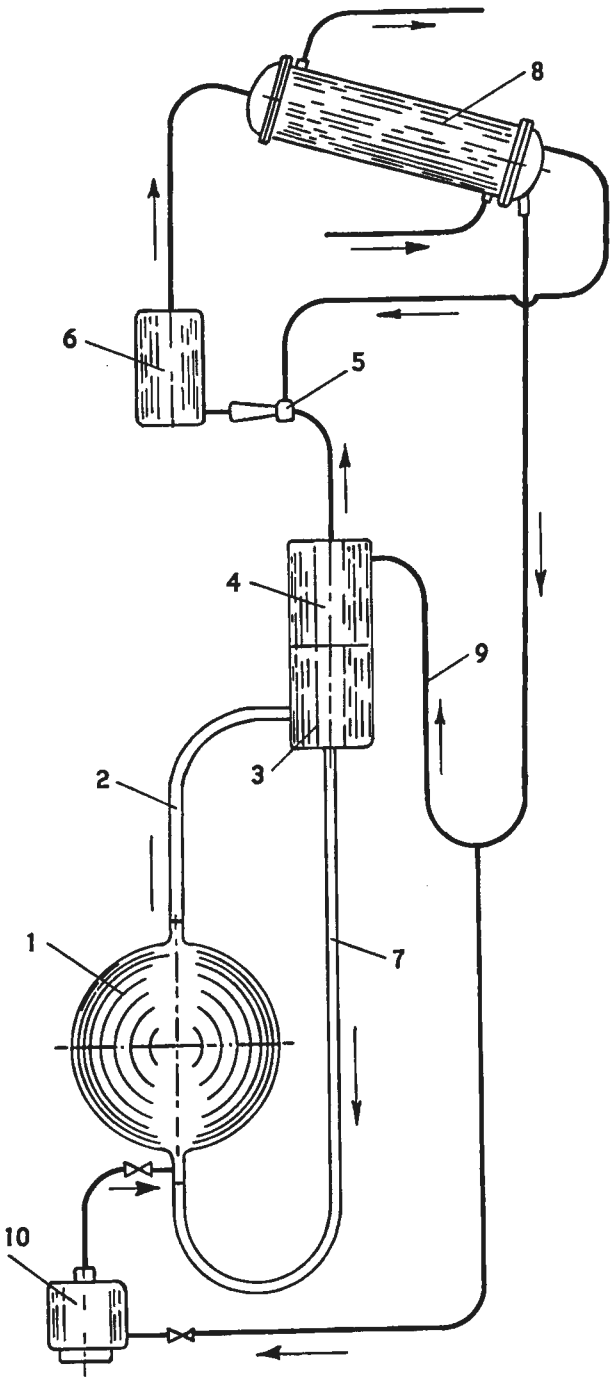


Figure 1a. Diagram of reactor without reflector: 1. Reactor. 2. Up-pipe. 3. Separator. 4. Steam purification apparatus. 5. Injector. 6. Burn chamber. 7. Down-pipe. 8. Heat exchanger. 9. Liquid seal. 10. Electric-boiler starter

In the separation vessel the steam is separated from the water and is directed to a purifying apparatus (4), then to the injector (5) and the catalytic chamber (6). The water from the separation vessel circulates back into the reactor through the down-pipe (7). Circulation of the medium is maintained by the difference in the densities of the steam-water mixture in the reactor and the up-pipe and the density of the liquid in the down-pipe.

From the catalytic chamber the steam enters heat exchanger (8) which produces the secondary steam for the turbines.

Injector (5) circulates a part of the steam through the heat exchanger and the catalytic chamber with the aim of burning up the remains of the detonating gas.

From the heat exchanger, the condensate returns to the reactor. An electrical boiler (10) is added to the scheme for maintaining the suspensions when the reactor is being started.

A reactor arrangement with organized internal circulation (no figure is given) is also possible. The whole reactor, together with the catalytic chamber and heat exchanger, may be united in one block, thus becoming very compact.

The scheme in Fig. 1b shows a reactor with fuel breeding and with the same circulation system as in the first case in scheme 1. It differs in that the reactor is surrounded by a reflector (11). The reflector is filled with a boiling (in the case of organized circulation) suspension of thorium in heavy water and is equipped with an external circulation system. The steam formed in the reflector is separated from the liquid in separator (12), it is purified of thorium powder in apparatus (13) and joins the steam going from the reactor to the burning chamber. From the heat exchanger the condensate enters tank (14) whence it is distributed between the reactor and reflector.

It should be noted that in the schemes considered residual heat generation, in the reactor, should be removed in order to avoid boiling up of the medium and upsetting of circulation. Heat removal may be achieved by adding a small heat exchanger, which is not shown in the schemes.

6. PROPERTIES OF BOILING HOMOGENEOUS REACTORS

The actual design of a boiling homogeneous nuclear power plant is determined by solutions to a number of key problems which are best considered together than separately. The following are the most important: (*a*) reactor and reflector arrangement; (*b*) mean moderator density; (*c*) rate of circulation, diameter and length of circulation pipes; (*d*) temperature of active zone and pressure in reactor; (*e*) fuel concentration; (*f*) efficiency of heat utilization; (*g*) duration of nonstop pile operation; and (*h*) breeding factor.

Of importance also is the time of the transition of the breeding reactor to secondary nuclear fuel, which is usually called the fuel doubling time.

The problem consists in finding a combination of the above-mentioned factors, for which the cost of the produced electric power is least.

A description of general methods used in solving this problem is not within the scope of this paper. Here we shall describe the more characteristic peculi-

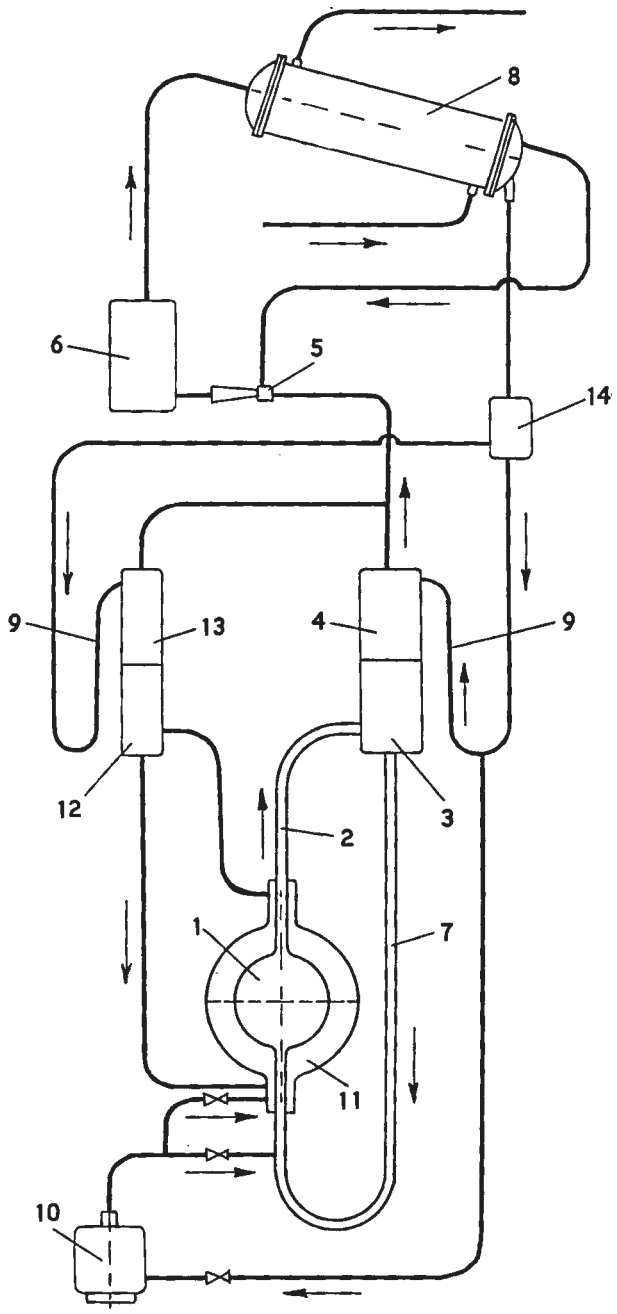


Figure 1b. Diagram of reactor with fuel regeneration: 1. Reactor. 2. Up-pipe. 3. Separator. 4. Steam purification. 5. Injector. 6. Burn chamber. 7. Down-pipe. 8. Heat exchanger. 9. Liquid seal. 10. Electric-boiler starter. 11. Reflector. 12. Reflector separator. 13. Apparatus for reflector steam purification. 14. Separator tank

arities of a boiling homogeneous reactor which were found while designing the reactor.

As an example, consider a small-size breeding reactor (Fig. 1a). U^{233} is used as the fuel. The selected external circulation system consists of an up-pipe of diameter 700 mm, a down-pipe of diameter 500 mm. The geometric head is 5000 mm.

The calculations performed made it possible to establish the dependence of the thermal power of the reactor upon the pressure in it for various concentrations of fissionable material in the core. The results of calculation are given in Fig. 2.

In the figure, the thermal power of the reactor (in thousands of kw) is plotted versus the pressure in the reactor. The solid lines correspond to constant concentrations of uranium in the operation medium. The uranium concentration is expressed in kg of U^{233} per ton of heavy water.

In a just critical reactor, the moderator density is constant and corresponds to the given uranium concentration. Therefore, in Fig. 2, every constant concentration curve is simultaneously a constant mean moderator density curve. If a reactor is operating at a temperature for which the density of the nonboiling moderator is equal to the mean density, then such a reactor maintains this temperature but does not produce steam. This state is described by the point of intersection of the solid curve and the X-axis in Fig. 2.

As the temperature falls, the moderator density in the reactor rises, but because of an increase in the volume of the steam being formed, the mean density falls to its former value. A further reduction in the temperature increases the quantity of steam, and, consequently, the power produced by the reactor. At a certain temperature, the reactor power reaches a maximum. As the temperature continues to fall, the increase in the volume of steam in the reactor is so great that the reactor power begins to fall. In Fig. 2 the pressures corresponding to the boiling temperatures of the moderator are plotted. The dotted curves ($N_{sp.}$ heat) give the constant specific thermal power in thousands of kw per kg of U^{233} in the reactor and the circulation circuit.

From Fig. 2 it may be seen that:

1. Reactor power rises with an increase in the concentration of the fissionable material.

2. With a constant concentration, higher operating pressures first increase the reactor power and then reduce it to zero, or, more precisely, to residual heat generation.

3. Each concentration of uranium has its operating pressure in the pile, at which the reactor develops a maximum power.

Each power has correspondingly an operating pressure at which the specific thermal power is greatest.

4. As the reactor power rises, the specific thermal power of the uranium increases within the pressure limits considered.

The influence of the circulation of the active mixture may be seen from a comparison of Figs. 2 and 3 which characterizes the work of the same reactor, but with differing circulation rate. Less circulation is due to a reduction of the diameter of the up-pipe to 500 mm, and the down-pipe to 400 mm. The head remains unchanged.

A comparison of the figures shows the following:

1. A reduction in circulation results in a reduction

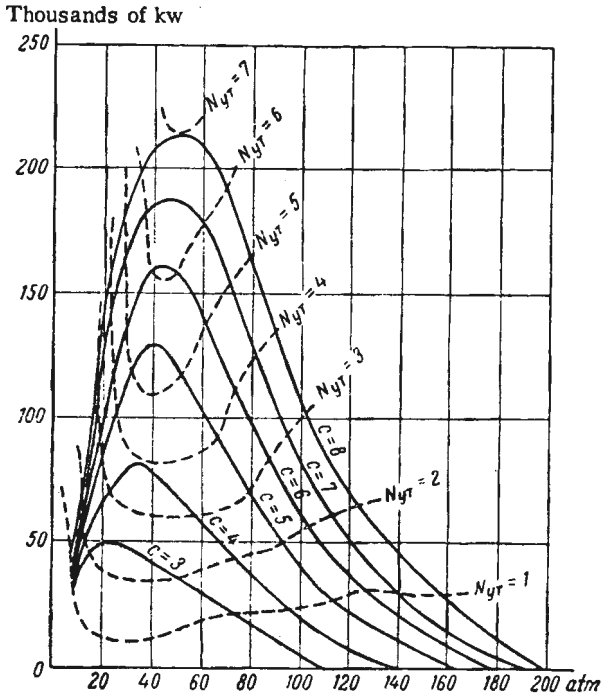


Figure 2. Dependence of heat-generating capacity of reactor on internal pressure with various concentrations of fissionable substance in working medium ($N_{sp} = N_{yT}$)

in the power of the reactor, for the same fuel concentration.

2. When the reactor operates at high power, for example at 150,000 kw, higher $N_{sp.}$ heat is obtained by greater circulation (Fig. 2). When the reactor operates at lower power, for example 50,000 kw, higher $N_{sp.}$ heat is obtained at a lower circulation rate (Fig. 3).

3. Given less circulation (Fig. 3) the peaks of equal concentration curves are situated in the region of higher operating pressures in the reactor than in the case of large circulations. Lower operating pressures correspond to large circulations.

Calculations show that when organized circulation in the reactor completely ceases it reduces the reactor power by approximately one-half.

Thus, for a constant uranium concentration in the core each pressure corresponds to a definite reactor power. Therefore, if we connect the reactor with a large enough heat exchanger, in which all of the steam formed in the pile condenses, then the pressure in the system will remain constant as will also the power developed by the reactor. This shows that in the case of constant heat consumption in the external circuit the boiling reactor is stable in operation. In the literature, this property is sometimes called self-regulation, which, in our opinion, is not an apt name. The property of stability is also found in nonboiling homogeneous reactors.

A change in the power output will result in a change in the reactor pressure and power. This is the essence of the property of self-regulation peculiar to the boiling reactor.

This may be illustrated. Let us assume that the power and pressure in the reactor correspond to point A in Fig. 3. If the consumption of electric power is constant, then the reactor pressure due to the property of stability, will be constant. But if the consumption of electric power begins to increase, then the turbine consumption of steam, under the influence of the turbine control system will also begin to grow. This will first cause a reduction in the secondary steam pressure in the heat exchanger followed by a reduction of steam pressure in the reactor. The operation point will move along the curve from point A towards the peak and will take up position B. Reactor power will grow, the pressure will cease to fall, and the system will revert to equilibrium, which corresponds to a new load on the turbine. When the consumption of electric power falls, the operation point will move in the opposite direction.

The position of the operation point on the curve must be selected in such a way that, when the maximum tolerance overloading of the turbine is reached, the reactor power should correspond to the maximum of the curve at point B. In this case, the nuclear power plant will truly have the properties of stability and self-regulation.

Calculations show that with variation in turbine power the pressure in the reactor reaches a new value with a time constant of about 0.5 min.

Control devices in a boiling homogeneous reactor are needed to vary the reactor power while maintaining the operating pressure or to vary pressure while maintaining the power at a constant level. A convenient control device may be a choke in the downpipe of the circulation circuit. The reactor is started by gradually introducing a concentrated suspension of uranium oxide into the core, in which a moderator is circulating.

Circulation of the moderator when the reactor is

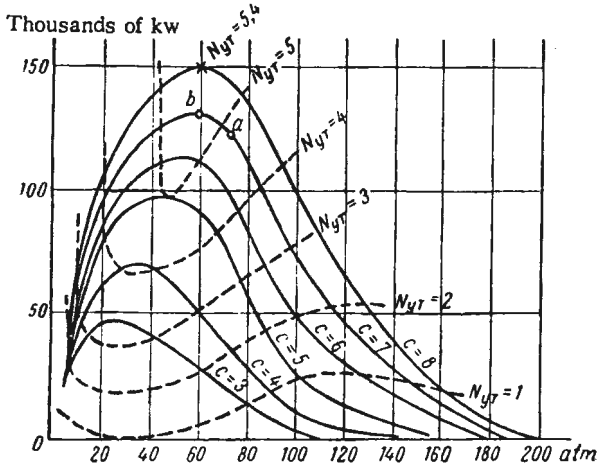


Figure 3. Dependence of heat-generating capacity of reactor on internal pressure with various concentrations of fissionable substance in working medium and with lower circulation than in the case shown in Fig. 2 ($N_{sp} = N_{yT}$)

put into operation is achieved with the aid of an auxiliary electric boiler.

In order to shut down the reactor, the active mixture is drained into tanks of a subcritical volume. In emergency cases, the reactor may be scrammed by safety-rods.

In examining reactor shut-down problems, it was feared that the suspension of uranium oxide would divide into layers, and both in the lower part of the reactor and in the circulation pipes there would be formed local centres of reactivity. But experience has shown that moderator boiling, caused by residual heat generation, proves sufficient to maintain the uranium oxide powder in a state of suspension. If in the course of time, lamination of the suspension does set in, and precipitation of the powder results in a slow rise in reactivity, then the enhanced steam production will very quickly reduce the concentration of the suspension in the reaction zone formed. As concerns circulation pipes of sizes given above, calculations showed that precipitation of fuel in the pipes cannot create conditions that might develop a chain reaction. Likewise, a sudden rise in the uranium concentration in the reactor, for example, as a result of an accident, will not cause an explosion. From Fig. 2 it may be seen that a uniform increase in the concentration (throughout the whole volume of the apparatus), even by as much as 20%, results in an increase in reactor power of only 30–40%. Thus, neither the precipitation of the fissionable material powder during shutting down of the reactor, nor a sharp rise in uranium concentration presents a serious hazard.

The possibility of the precipitation of thorium powder in the reflector is worthy of consideration. In this case the reactivity of the reactor rises noticeably. Thus, for example, in the case of 90% precipitation, the reactor power rises 1.5 times. Complete precipitation is much more dangerous; however, it is only slightly probable. When the reactor is being shut down, residual heat generation in the reflector will be very small, and therefore for certain cases, steam must be introduced into the reflector from an external source.

7. REACTOR PARAMETERS

Boiling heavy water reactors are stable at arbitrary fuel concentrations in the sense that when a portion of the water is replaced by steam, the reactivity always falls.

Light water reactors have a stability boundary at a concentration of about 20 kg of pure fuel per ton of water. At higher concentrations, these reactors are stable just as are heavy water reactors. When the concentrations are lower and a portion of the water in the core is replaced by steam, a decrease in moderation influences the pile reactivity to a lesser degree than the accompanying reduction of absorption, as a result of which the reactivity of the pile rises. Such reactors require constant power regulation. Light water reactors require considerable enrichment of the uranium but are smaller in size than heavy-water reactors.

The reactor geometry may vary, depending upon design requirements. It should be noted that for boiling reactors, higher performance characteristics in the power per unit weight of uranium are obtained in the case of a cylindrical form reactor with slightly increased radius in comparison with the demand for a minimum critical volume.

A pure-fuel reactor may operate for a practically unlimited time if fuel is added and if fission fragment removal is carried out. The active mixture of an enriched reactor must periodically be drained off and replaced by a new one.

It should be noted that for the construction of a low-power boiling homogeneous reactor, the following quantities of materials are required per 1000 kw of electricity produced.

Materials	*Quantity in kg*
Fissionable material	0.3–0.7
Thorium	0–160 (breeder reactor)
Heavy water	200–300

In the construction of a reactor of greater power, expenditure of materials per 1000 kw is lower.

Experiments and calculations show that boiling homogeneous nuclear power reactors are not inferior to the best modern heterogeneous power reactors.

Aqueous Homogeneous Power Reactors

By R. B. Briggs and J. A. Swartout,* USA

GENERAL FEATURES

The attractiveness of aqueous homogeneous reactors for producing nuclear power is attributable to two features. They are the fluid state of the fuel and the unique quality of heavy water as a moderator. Advantages deriving from these features include:

1. Aqueous homogeneous reactors can be designed for high power density and low fuel inventory. The critical mass of fuel is low, and the rate that heat can be removed from a reactor may be limited only by the rate that the fuel can be circulated and the temperature rise which is permitted. Heat is removed from the fuel in equipment which is external to the reactor. The equipment can be designed for maximum performance without regard for neutron cross section of materials or for the many other factors that must be considered when heat is extracted from the fuel in the reactor core.

2. Reactor fuels can be purified continuously to remove fission products and radiation damage products. Fuel preparation, handling, and reprocessing are simplified and high burn-up of fuel is achieved.

3. The reactors possess a very high degree of nuclear stability. Gross changes in operating conditions can be achieved by adjusting the fuel concentration. The strong negative power coefficient provides regulation and prevents harmful power excursions. Control rods or plates are unnecessary.

4. Simple mechanical design results from the use of a fluid fuel and moderator and the elimination of control elements. The reactor vessel can be a tank containing only those parts required to achieve the desired flow distribution and, in some designs, to separate the reactor into regions containing fissionable or fertile materials.

5. Absence of structural materials in the reactor core, use of heavy water as the moderator and continuous removal of fission product poisons make it possible to minimize the parasitic absorption of neutrons.

The use of fluid fuels and an aqueous moderator also results in undesirable characteristics which include:

1. High pressures are associated with the high temperatures which are required for producing power efficiently. The vapor pressure of water is 1200 psi at 572°F (300°C) and 2400 psi at 662°F (350°C). Circulating fuel reactors are operated with added pressure to prevent cavitation in the pumps and to minimize the volume of gases produced by decomposition of the moderator.

2. Large volumes of very highly radioactive materials are present in a fluid state at high temperature and high pressure. Provision must be made for containing those materials if they leak from the reactor vessels or piping. Pumps, piping and heat exchangers become radioactive from contact with the fuel. Their replacement or repair can be difficult and expensive.

3. Uranium, plutonium and thorium compounds, water and materials of construction form complex physical and chemical systems. Limitations on operating conditions are imposed by physical and chemical characteristics of the fuels and by corrosion and erosion effects.

4. Deuterium and oxygen are produced by radiolytic decomposition of the moderator. They must be recombined safely.

Although difficult problems remain to be solved, the aqueous homogeneous reactors appear to be among the most promising for producing economical nuclear power.

REACTOR TYPES

Aqueous homogeneous reactors are classified as "one-region" or "two-region" according to the distribution of fissile and fertile materials in the reactor. They are "circulating fuel" or boiling, depending upon the method of removing heat. In a one-region reactor the fissile and fertile materials are uniformly mixed. They are separated into a core of fissile material and a blanket of predominantly fertile material in a two-region reactor. A circulating fuel reactor is one in which the fluid fuel is circulated through the reactor and external heat exchange equipment by natural or forced convection. Heat is removed from a boiling reactor by vaporizing coolant in the reactor and condensing the separated vapor in external heat exchangers.

In the United States most of the engineering effort has been devoted to the development of the two-region circulating fuel reactors. Their advantages of good neutron economy in small reactors with low inventories of critical materials and of greater versatility in permitting the use of different fuel sys-

* Oak Ridge National Laboratory. Work by members of the Homogeneous Reactor Project, Oak Ridge National Laboratory.

tems in core and blanket have led to this emphasis. The technology developed for the two-region reactors is equally applicable to the one-region reactors.

Experience with the Homogeneous Reactor Experiment[1,2] has demonstrated that there are few uncertainties in the nuclear and process design of circulating solution reactors. The amount of heat that can be removed seems to depend upon how rapidly the fuel solution can be pumped through the core. Experience with boiling homogeneous reactors is limited to a few experiments with the Los Alamos Water Boiler. Until recently there has been little basis for evaluating the stability or the heat producing capability of boiling reactors.

A correlation of data from a variety of experiments now suggests that heat can be removed from a boiling liquid in an amount that varies directly with the cross sectional area of the container, some function of the pressure and the ratio $[f/(1-f)]$ where f is the vapor fraction. It is estimated that a cylindrical reactor, 10 ft diameter by 10 ft high and operated at a power of 440 megawatts at 2000 psi would contain 30 volume per cent vapor. Baffles to direct the circulation and separators to remove the vapor from the liquid appear to be necessary to achieve this high power output with moderate vapor fraction. Additional data are being obtained to test the accuracy of the correlation.

Experiments have shown that a rapid change in heat generation in a boiling liquid is followed almost immediately by a corresponding change in the rate of vaporization.[3] The fluid in the experiments was heated by its resistance to a flow of electrical current. The delay between an increase in current flow and an increase in vaporization rate was the time required to increase the superheat in the liquid by a very small amount. It is inferred from the experiments that boiling reactors would be self-limiting. If the reactivity were increased rapidly by the introduction of fuel or by a fluctuation in the rate of vapor disengagement, the generation of new bubbles would prevent the accumulation of enough excess reactivity to produce harmful power and pressure fluctuations.

The effect of coupling the nuclear, hydrodynamic and mechanical systems requires additional study before large boiling reactors are built. Slow, undamped oscillations have been produced in experiments with internal resistance heating. More must be learned about what conditions are necessary for the production and prevention of oscillations. A better understanding of the behavior of boiling reactors under varying load conditions is important in evaluating the feasibility of large power reactors.

ENGINEERING DESIGN OF A NUCLEAR POWER STATION

Basis for Design

A nuclear power station capable of producing 100 megawatts of electricity is a convenient size to be the basis for the design of a two-region, circulating fuel reactor. One conception of such a station is shown in Fig. 1. The station contains one reactor to generate steam and one turbogenerator to produce electricity. The turbine would be almost as large as can be constructed today for use with low-pressure steam. Plants having greater electrical generating capabilities would contain several reactors and turbogenerators. Major cost reductions achieved in the larger plants would result from the larger scale of construction and the more effective design and use of administrative, maintenance and processing facilities.

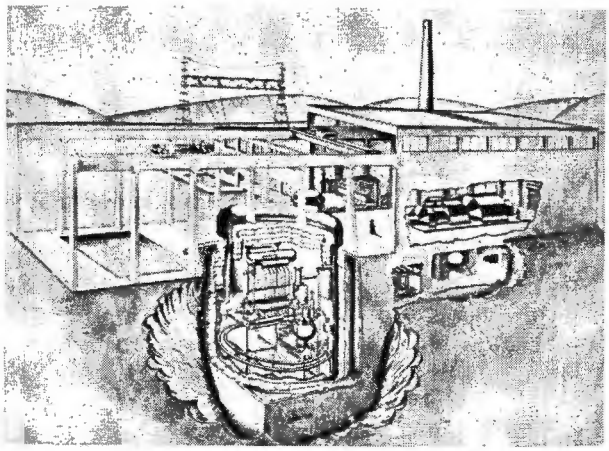


Figure 1. Homogeneous reactor power station

Basic operating conditions for a station containing a power breeder reactor using the thorium-U^{233} cycle are contained in Tables Ia and Ib. They are conditions which were found to result in minimum power costs in design studies based upon present technology and cost information.

A dilute solution of uranyl sulfate in heavy water is selected as the fuel for the core system.[4] Tests in the absence of pile radiations indicate that most of the equipment can be constructed of type 347 stainless steel. Low corrosion rates are found with fluid velocities as high as 35 ft/sec and temperatures at least as high as 572°F in uranyl sulfate solutions which contain less than 10 grams of uranium per kilogram of water. Parts of the equipment that will operate at higher velocities or that will require special corrosion protection can be made of titanium. Zircaloy, like titanium, has superior corrosion resistance in uranyl sulfate solutions.

The tank which separates the core from the blanket should be made of Zircaloy to minimize the neutron loss.

A thorium oxide slurry in heavy water is presently the most promising fuel for the blanket.[5] Type 347 stainless steel appears to be a satisfactory material of construction for the blanket equipment if the velocity is kept below 25 ft/sec. Titanium has shown good resistance to erosion by the slurries. Pump parts and other pieces of equipment in regions of

very high velocity or high local turbulence can be made of titanium.

Our experience with the handling of slurries is much more limited than our experience with solution fuels. It appears that much of the equipment which has been designed for use with solutions will prove satisfactory for slurries if minor changes are made to prevent the accumulation of solids and the plugging of small pipes.

The nuclear power station is separated into three plants for process design purposes. They are the reactor plant, where the steam is generated; the generating plant, where the electricity is produced; and the chemical plant, where the fuels are processed. Only the first two are discussed in this paper. Until there are simple and inexpensive chemical processing plants, one centrally located plant will be required to process the fuels for several 100-megawatt power stations. Only simple separations like the removal of gaseous fission products or the separation of corrosion product and fission product precipitates from solution fuels are likely to be undertaken in the reactor plant.[6]

Design of the Reactor Plant

The High-Pressure Circulating System

The reactor plant is subdivided into three systems or groups of equipment according to function. They are the high-pressure circulating system, the recombiner system and the low-pressure storage and waste disposal system. Each system has separate sets of equipment for handling core and blanket fluids in two-region reactors.

A flow diagram for the high-pressure circulating system is presented in Fig. 2. The uranyl sulfate solution containing 8.6 grams of total uranium per kilogram of heavy water is pumped through the core circuit. A thorium oxide slurry containing 1130 grams of thorium per kilogram of heavy water or 1000 grams per liter of slurry is circulated through the blanket equipment. Gas is removed from the fuels in the separators and sent to recombiner systems where heavy water is regenerated and returned to the high-pressure circulating systems. Heat is extracted in the exchangers to form steam which is used in the generating plant and in some reactor auxiliaries. Pressurizers are provided for both the core and the blanket and are interconnected to aid in preventing the pressure differences between the two systems from becoming large enough to damage the zirconium tank.

Table Ia. Design Data for Two-Region Power Breeder Station

Power plant	
Net electric output, megawatts	100
Gross electric output, megawatts	106
Net station efficiency, per cent	23
Steam temperature, °F (°C)	435 (225)
Steam pressure, psi absolute	365

Reactor Vessel

Conceptions for two-region reactor vessels are shown in Fig. 3 and 4. The core tanks are 5 ft in. diameter, ½ in. thick and are made of Zircaloy, an alloy of zirconium. Each pressure vessel has an inside diameter of 10 ft 4 in. and a wall thickness of 5 in. Four-inch-thick thermal shields are provided to reduce the heat production and thermal stress in the pressure vessel walls.

The two configurations differ in the designs of

Table Ib. Design Data for Two-Region Power Breeder Station

	Reactor plant		
	Core	*Blanket*	*Total*
Heat output, megawatts	360	80	440
Operating temperature, °F (°C)			
Inlet	482 (250)	473 (245)	
Outlet	572 (300)	600 (315)	
Operating pressure, psia	2000	2000	
Fuel circulation rate, gal/min	26,000	4400	
Core diameter, ft (cm)	5.0 (152)		
Blanket thickness, ft (cm)		2.25 (68)	
Reactor volume, ft^3	67	410	477
liters	1900	11,500	13,400
Volume of reactor and auxiliaries, ft^3	355	495	850
liters	10,000	14,000	24,000
Fuel	UO_2SO_4-D_2O	ThO_2-D_2O	
Fuel composition, gm/kg D_2O			
$U^{233} + U^{235} + Pa^{233}$	2.8	4.7	
Total U	8.6	3.4	
Th^{232}		1130	
Inventory, kg			
$U^{233} + U^{236} + Pa^{233}$	30	60	90
Th^{232}		15,000	
D_2O			26,000
Excess U^{233} produced, gm/day		60	60

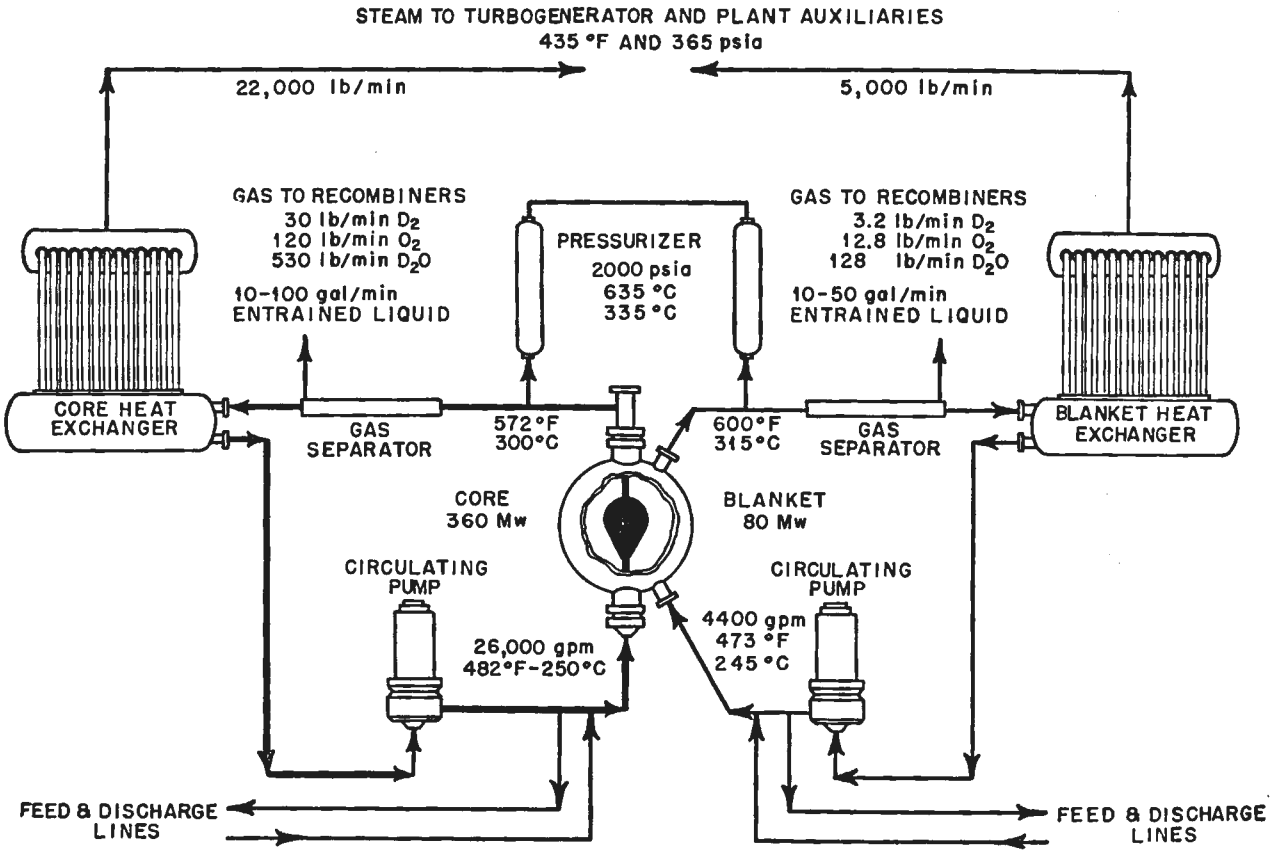


Figure 2. Flow diagram for high pressure circulating system

the cores. In the first, the fuel flows in through the bottom of the core and out through the top. Diffuser screens provide a smooth expansion of the flow. There is essentially no separation at the wall and no recirculation of fluid in large eddies. Calculations based upon the experimentally determined flow distribution and the calculated neutron flux distribution indicate that the temperature will be nearly constant in any horizontal plane through the core.

In the second configuration the fluid is introduced as a jet which passes through the center of the core, reverses direction and flows out through an annulus around the inlet pipe. There is a very high degree of mixing. The average temperature outside of the central jet and the average fractional volume of gas in the core will approach the temperature and the volume fraction in the exit stream.

The concentric inlet and outlet design has an advantage in mechanical simplicity. There are fewer problems arising from the difference in coefficients of expansion of Zircaloy and steel. Neither type appears to have an important advantage in nuclear or hydrodynamic design.

Construction of either of the two-region reactor vessels is considered to be feasible. A Zircaloy tank which is 32 in. in diameter and 5/16 in. thick has been fabricated for the Homogeneous Reactor Test.[7] Making a 5 ft diameter tank of ½-in.-thick plate should present few more problems. The pressure vessel probably would be constructed of carbon steel and clad with stainless steel. Carbon steel is stronger, easier to fabricate and less expensive than stainless steel. It has a higher thermal conductivity and lower thermal expansion coefficient so there are fewer thermal stress problems. The major question in the use of carbon steel is the effect of pile radiations. The temperature at which a carbon steel is transformed from a brittle material to a ductile material is raised by fast neutron irradiation. Whether the radiations would cause the vessel to be brittle in the operating temperature range is unknown. Presumably, a vessel of stainless steel would be less susceptible to embrittlement by reactor radiations. It is doubtful whether a satisfactory pressure vessel can be constructed of stainless steel in the near future except at a prohibitive cost.

Gas Separator

It is estimated that 30 lb/min of deuterium and 120 lb/min of oxygen will be produced in the core of the reactor by radiolytic decomposition of the moderator. This corresponds to the production of 160 ft^3/min of gas saturated with water vapor at 572°F and 2000 psi and to 2 volume per cent of gas in the fluid leaving the reactor core. Because there is much less fission heat, it is estimated that the gas production in the blanket will be one-tenth that in the core.

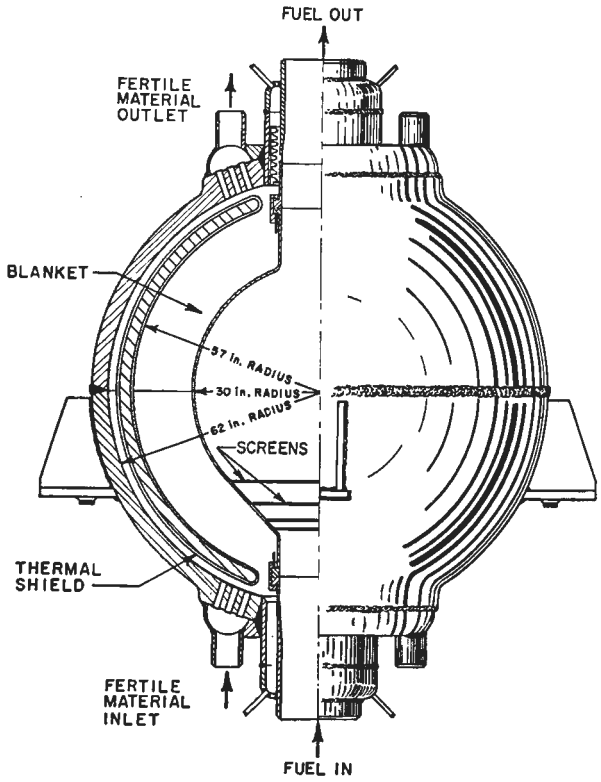


Figure 3. Two-region reactor with "straight-through" core

A device for removing these gases that has been proven successful by experimentation is the axial flow gas separator shown in Fig. 5. Vanes at the inlet impart a swirling motion to the fluid. The resulting centrifugal forces separate the liquid and gas causing the gas to collect in the void space along the axis. The void is stabilized on the hubs at the two ends and the gases are removed through a pipe which is attached to an opening in one hub. Vanes are provided at the discharge to recover some of the rotational energy. For adequate gas separation and vortex stability a conservative value of 4 or greater is specified for the Froude number. [$Fr = V_o^2/Rg$, where V_o is the tangential velocity, R is the inside radius of the pipe, and g is the gravitational acceleration.]

A separator designed for the full 26,000 gal/min flow in the core circuit is 24 in. diameter inside and 15 ft long between hubs. A similar separator designed for the blanket circulating system is about 9 in. diameter and 7 ft between hubs. Results of experiments with 14 in. diameter separators indicate that the volume of liquid entrained in the gas stream from the largest separator can be kept below 100 gal/min and probably below 50 gal/min.

Heat Exchanger

Several designs have been proposed for heat exchangers for cooling the core and blanket fluids. A typical design for a natural recirculation, steam generator to extract the entire 360 megawatts of heat from the core solution is shown in Fig. 6. The solution is pumped through 7800, ⅜ in. outside diameter, stainless steel tubes at a velocity near 18 ft/sec. Each tube has an effective length of 24 ft and wall thickness of 0.048 in. The tubes are welded into 5 ft diameter by 8 in. thick tube sheets made of stainless steel or carbon steel clad with stainless steel. The tube sheets are welded into the shell and to the heads where the process piping is attached. Steam is generated in water circulating outside of the tubes, separated from the water in the elevated steam drum and withdrawn to the turbogenerator.

It is estimated that the heat exchanger will cool 26,000 gal/min of core solution from 572°F to 482°F while producing 22,000 lb/min of steam at 435°F and 365 psi absolute. A similar exchanger containing 2000 tubes should cool the blanket fluid from 600°F to 473°F while producing 5000 lb/min of steam.

The single, large heat exchanger probably can be fabricated and at an initial cost which would be less than the cost of several smaller units of equal total capacity. Whether it is more economical to use one large exchanger depends on its reliability. The difficulty of insuring perfect tube joints and the cost of the exchanger increase with decreasing tube diameter and increasing number of tubes. However, the amount of heat transfer surface required and the inventory of fuel decrease with decreasing tube diameter. The selection of ⅜ in. diameter tubes as the optimum size was based on the results of a study of the effect of tube size on inventory and fabrication costs.

While heat exchanger designs similar to that described above have received the most attention, there are other attractive possibilities. In one type the feedwater is pumped into the shell at one end and steam is withdrawn from the shell at the other end. Many baffles are provided to direct the flow. This type of exchanger occupies a minimum volume in the shield and makes it relatively easy to generate steam at a constant pressure over a wide range of load conditions. In another type of exchanger the fuel solution flows outside of the tubes and steam is generated in the tubes. Such an exchanger can be designed so that the radiation level is low where the tubes are welded to the tube sheet, making the welds more accessible for repairs. Considerable ingenuity is required to keep the fuel inventory small.

Circulating Pumps

Developing satisfactory circulating pumps is one of the major problems of aqueous homogeneous reactors. The pumps must be completely leak-tight. They must operate for long periods without maintenance. The internals must be easy to remove when replacement or repair becomes necessary.

Figure 7 shows the design features of a canned rotor pump. Similar pumps have been built to circulate 4000 gal/min of water against a 250 ft of

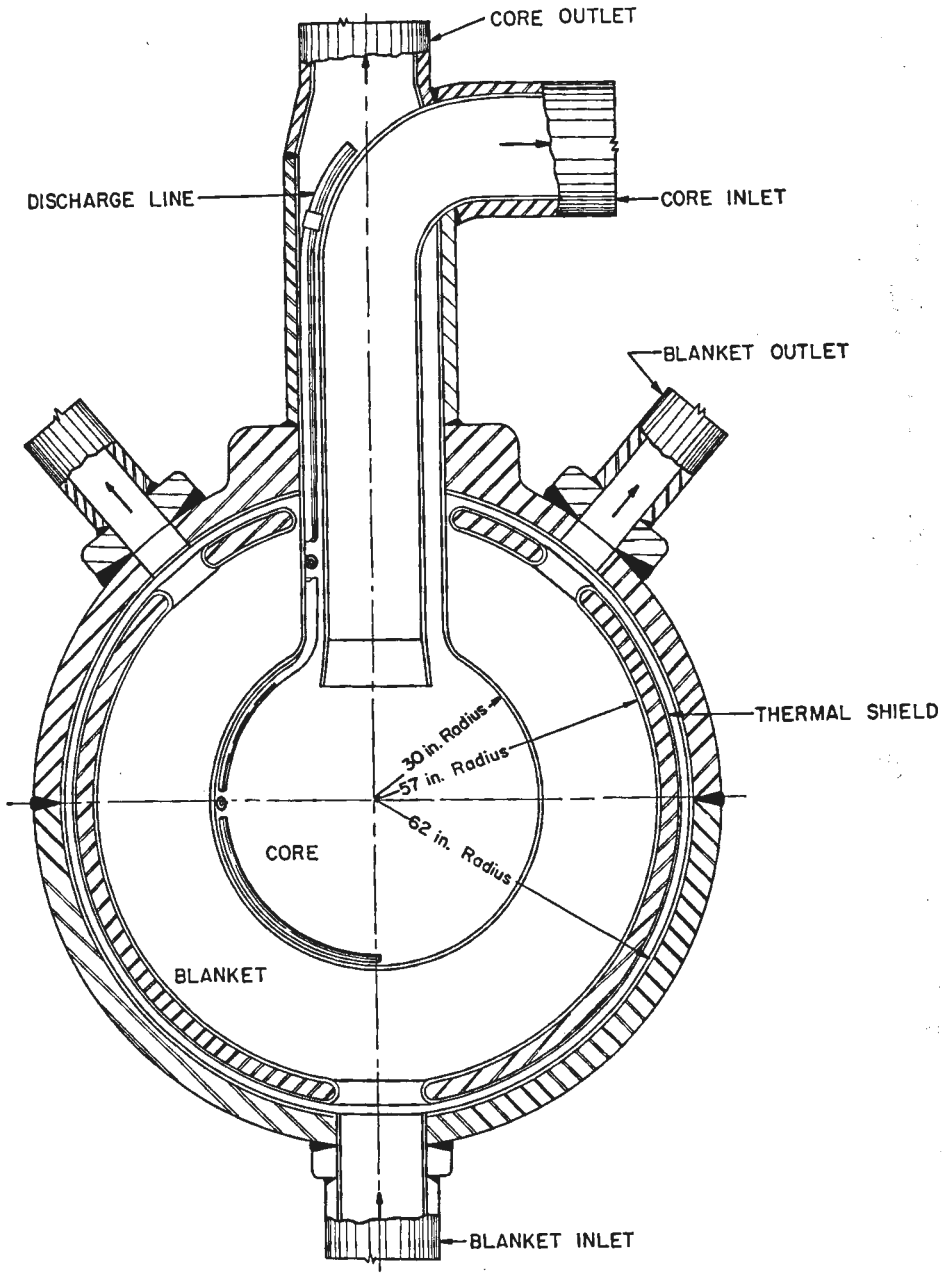


Figure 4. Two-region reactor with "concentric inlet and outlet" core

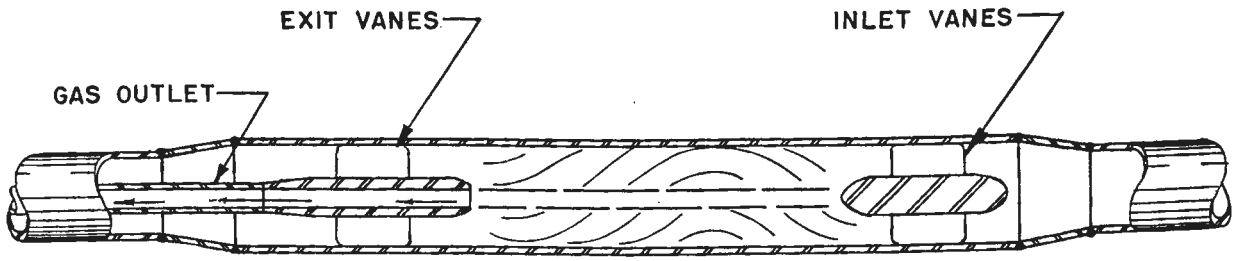


Figure 5. Axial flow gas separator

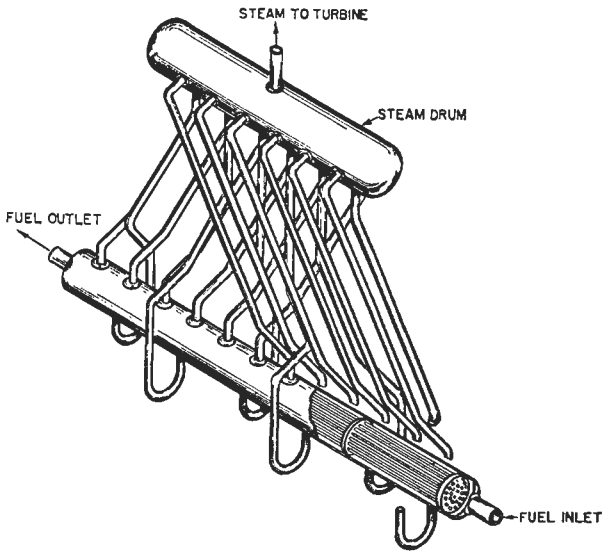


Figure 6. Natural recirculation heat exchanger

head at 2000 psi. Others are being designed for pumping 16,000 gal/min under similar head and pressure conditions. There is little doubt that a pump can be built to pump the full 26,000 gal/min core flow. Again the size and number of pumps may be determined on the basis of reliability and replacement cost.

Stainless steel is a satisfactory material for most of the pump parts which will be in contact with the fuels. Titanium should be used for the impeller, for the labyrinth seals and for reinforcing parts of the casing. The bearings can be hydrostatic bearings or bearings with Stellite running on Graphitar. Maintenance is simplified by making all of the internals removable through a flanged opening at the top of the pump. Condensate injected into the rotor cavity will keep the fuels out of the bearings and reduce the radiation level inside of the motor.

Pressurizers

In a circulating fuel reactor the reactor and high-pressure systems are kept at a pressure which is greater than the vapor pressure of the circulating liquid. This is done to control the vapor volume in the reactor and to prevent cavitation in the pumps. A common method of providing the excess pressure is to boil some of the fuels in vessels which are attached to the main circulating loops. The heat input is controlled so that the vapor pressure of the boiling liquid is the desired system pressure. Noncondensable gases are detected by measuring the temperature-pressure relationship in the pressurizers. They are bled from the system before an explosive mixture forms. The heat requirements may vary from 100 to 1000 kilowatts per pressurizer depending upon the design.

The heat load can be reduced by pressurizing with a non-condensable gas. It is necessary to catalyze the recombination of deuterium and oxygen that collect in the pressurizers to prevent the accumulation of explosive mixtures. Disposing of the gases when the reactor contents are discharged could prove troublesome.

The pressurizers also act as surge chambers to supply or receive liquid from the circulating systems during the sudden changes in temperature and density that occur as the result of sudden changes in reactivity. The volume of the vapor space, the distance from the reactor vessel to the vapor space and the size of the connecting pipes are specified to limit the pressures during power surges in the reactor. For this reactor the core pressurizer should have a vapor volume of about 35 ft^3. It should be located within 10 ft of the core outlet and the connecting pipe should have an inside diameter of about 20 inches.

Piping and Valves

All piping in the high-pressure systems will be type 347 stainless steel in sizes as large as 20 in. inside diameter and designed for service at 2000 psi and 600°F. Workmanship must be of the highest quality. Each weld must be X-rayed and tested for leaks at pressure and temperature. Although seamless piping is preferable, the use of welded piping probably can be permitted for sizes larger than 10 in. in diameter if the piping is tested properly. Larger seamless piping is difficult to obtain and is very expensive.

Providing joints in the piping such that heat exchangers and other equipment can be replaced presents many problems. A welded connection is preferable but there are as yet no welding machines that can be installed and operated remotely and that will make welds of the required quality. At present it appears that it will be necessary to have flanged joints. Ring joint flanges have been used successfully in sizes up to 24 in. diameter in the experimental homogeneous reactors or in test facilities.

In designing the reactor the use of large valves which must seal against 2000 psi pressure is avoided. The largest valves are those which control the discharging of the reactor contents to storage tanks. They will have ports as large as 6 in. in diameter. All valves will be operated through bellows seals to prevent the leakage of radioactive materials.

Recombiner Systems

Two types of systems have been designed for recombining the deuterium and oxygen which are produced in aqueous homogeneous reactors. One requires a flame recombiner in which a combustible mixture of the gases is burned. In the other the gases are diluted below the combustible limit and are recombined on a platinum or a palladium catalyst. The hazards of handling explosive mixtures containing radioactive contaminants have made it seem prudent to limit the flame recombiners to operation at low pressure. Catalytic recombiners have been

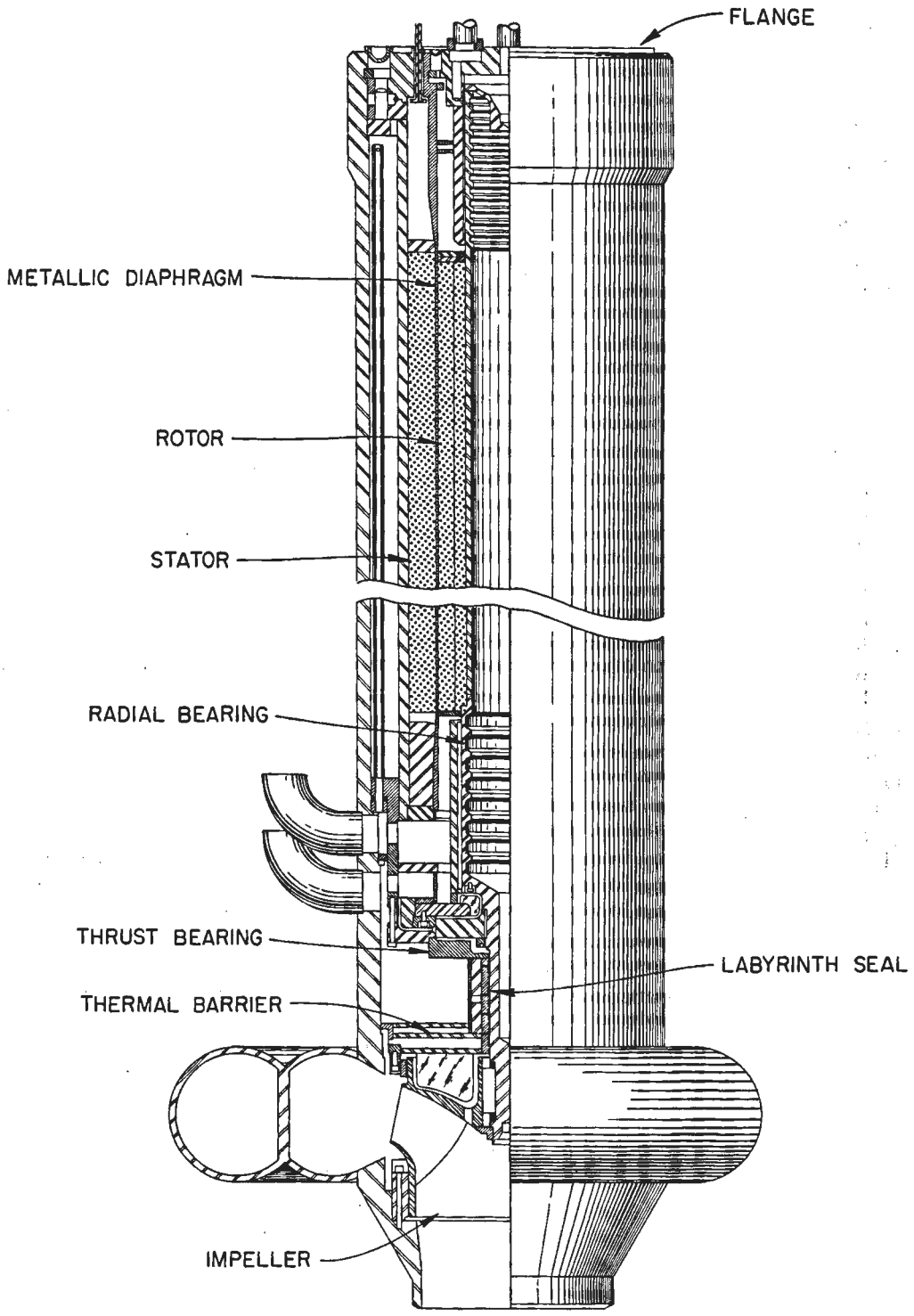


Figure 7. A circulating fuel pump

designed for operation at both low and high pressures.

A flow diagram for a low-pressure flame recombiner system for burning the gases generated in the reactor core is presented in Fig. 8. It is essentially the same as the system used for the Homogeneous Reactor Experiment. Gases and entrained liquid for the reactor pass through a heat exchanger where they are cooled from 572°F to about 212°F. The heat is exchanged with liquid being returned to the reactor. The gases and liquid then pass through a throttling valve and into an evaporator where the liquid is separated and the gases are diluted below the explosive limit, about 15 mole per cent of combustibles, with steam. The steam is condensed in a precondenser and the gases are fed to the burner where they are recombined. Finally, the gases are

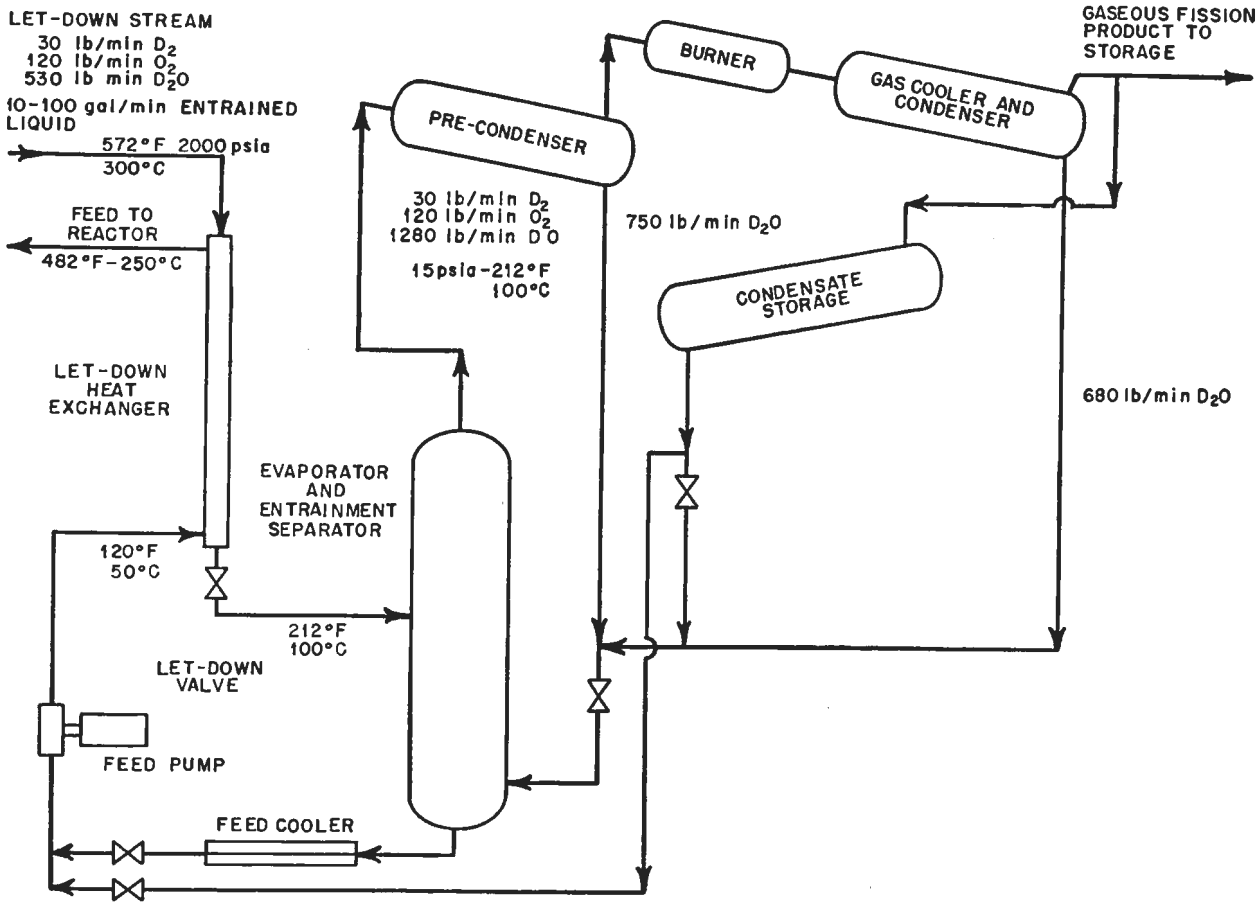


Figure 8. Flow diagram for low pressure flame recombiner system

cooled and the heavy water is condensed and flows to the condensate storage or to the evaporator. Liquid from the evaporator is pumped to the reactor to maintain a constant fuel volume in the circulating system. The concentration of uranium in the reactor is controlled by controlling the distribution of condensate between the evaporator and the storage tank. Gaseous fission products appear as non-condensables in the gas cooler and condenser and are discharged to storage.

This system wastes heat. Although dilution of the gases by steam in the evaporator may not be absolutely necessary it is considered to be desirable to minimize the volume of explosive gas in the equipment. Heat can be conserved during steady operation by using part of the heat of combustion to regenerate the steam diluent. Also, the losses can be reduced by using some of the heat to warm the feedwater for the steam boilers in the high-pressure systems.

The low-pressure flame recombiner system can be converted into a low-pressure catalytic recombiner system by eliminating the precondenser and substituting a catalyst bed for the burner. Platinized alumina in the form of ⅛ in. in diameter by ⅛ in. long pellets containing 0.03 to 0.3 weight per cent of platinum is an effective catalyst. Complete recombination can be obtained in 3 ft^3 of bed volume contained simply as shown in Fig. 9.

If a blower were provided to recirculate water vapor or a non-condensable gas, the evaporator would have only limited application in the recombiner system. By proper design of the gas cooler and condenser most of the heat of combustion could be used for generating steam for the turbogenerator.

A flow diagram for a high-pressure catalytic recombiner system is shown in Fig. 10. Gases from the separator are discharged into a fuel reservoir where they are mixed with a recirculated diluent. The diluent contains about 60 volume per cent heavy water vapor and 40 per cent non-condensable gas at 572°F and 2000 psi. Helium, oxygen or other gases can be used as the non-condensable gas. Explosions are prevented by keeping the deuterium content of the saturated gas below 3 mole per cent and 6 mole per cent respectively for oxygen and helium diluents.

The design of the high-pressure recombiner is similar to that of the low-pressure unit shown in Fig. 9. The vessels have smaller diameters and the walls are thicker. There is no change in the type or volume of catalyst. Following the recombiner is a condenser where the excess heavy water is

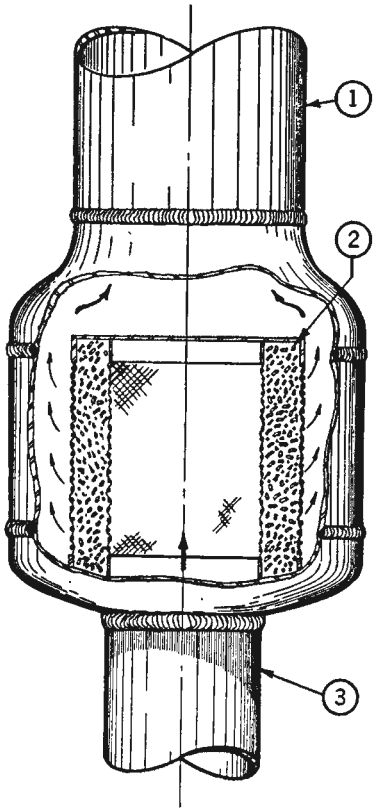


Figure 9. Catalytic recombiner. 1. 24" diameter outlet. 2. Catalyst bed, 24" diameter, 20" high, 4" thick. 3. 16" diameter inlet

removed and sent to the fuel reservoir or to a condensate tank. During normal reactor operation the fuel concentration and operating temperature would be adjusted by changing the distribution of water between the fuel reservoir and the condensate tank. In this flow diagram a blower is indicated for recirculating the gas. Designs have been made for blowers which are similar to the main circulating pumps but smaller. There is a possibility of eliminating the blower by designing the recombiner system for natural convection recirculation.

Part of the gas can be circulated through adsorbers for removing fission products. While this would lower the fission product level in the reactor, it is not required to obtain a low xenon poison level. The natural distribution of xenon between the liquid in the reactor core and the gas in the recombiner system and the decay of xenon in the gas would result in a lowering of the xenon poison level to 0.01 or less.

By properly designing and locating the gas separator and the piping to the fuel reservoir, the recombiner systems could also serve as the reactor pressurizers. The pressure would be controlled by varying the cooling in the condenser and by adding or removing gas.

Low-Pressure Storage and Waste Disposal System

The reactor plant requires equipment for storing fuel when the reactor is being serviced, for charging fuel into the reactor, for removing fuel to be sent to the processing plant and for disposing of the gaseous fission products. Storage tanks are sized so that a chain reaction cannot occur in them. Usually, it is convenient to make them long and to limit their diameter to 2 or 2.5 ft. An evaporator is included in the core storage tank system to separate most of the heavy water from the uranium and fission products before they are sent to the processing plant and for starting the reactor as will be described later. The same evaporator can be part of a low-pressure flame recombiner system. Cooling is provided for removing the fission product decay heat. There is a small recombiner for recombining the radiolytic gases that are generated in the stored fuel.

Low head, large volume, centrifugal pumps can be used for charging heavy water into the reactor core and slurry into the blanket in starting the reactor. High head pumps are required to feed the operating reactor. Positive displacement, diaphragm-type pumps were used successfully in the Homogeneous Reactor Experiment and are specified for the Homogeneous Reactor Test. Similar pumps should be satisfactory if the decomposition gases are recombined in high-pressure recombiners. Other types of pumps may be required for returning the liquid from low-pressure recombiners.

Fission product gases which are removed from the recombiner systems must be disposed of without creating a biological hazard. It is believed that this can be done for at least the first power reactors by storing the radioactive gases until all have decayed except for the 10-yr krypton-85. The krypton would be diluted with air and discharged through a tall stack. Gases to be stored are first circulated through cold traps where all but traces of heavy water are recovered. Then they are adsorbed on activated charcoal or on other adsorbers in pipes or vessels which are cooled to remove the decay heat. Assuming that the gases are retained for about 1 year, krypton-85 would be discharged from the stack at a rate of about 600 curies per day.

Reactor Assembly

An assembly of the reactor and auxiliary equipment is shown as part of the drawing of the power station in Fig. 1. The reactor equipment is shown as being housed in a tank which is 60 ft in diameter by 120 ft high and designed to contain the fuels and vapor if a leak develops. There is access to the pumps, valves and other critical pieces of equipment through openings in the dome and in the biological shielding. The entire dome would be removed to replace the reactor vessel or heat exchangers. In replacing large pieces of equipment, it probably would be expedient to fill the tank with water for shielding. A pool is shown adjacent to the reactor for storing contaminated equipment while it is being repaired. The economic feasibility of aqueous homogeneous reactors may depend upon the development

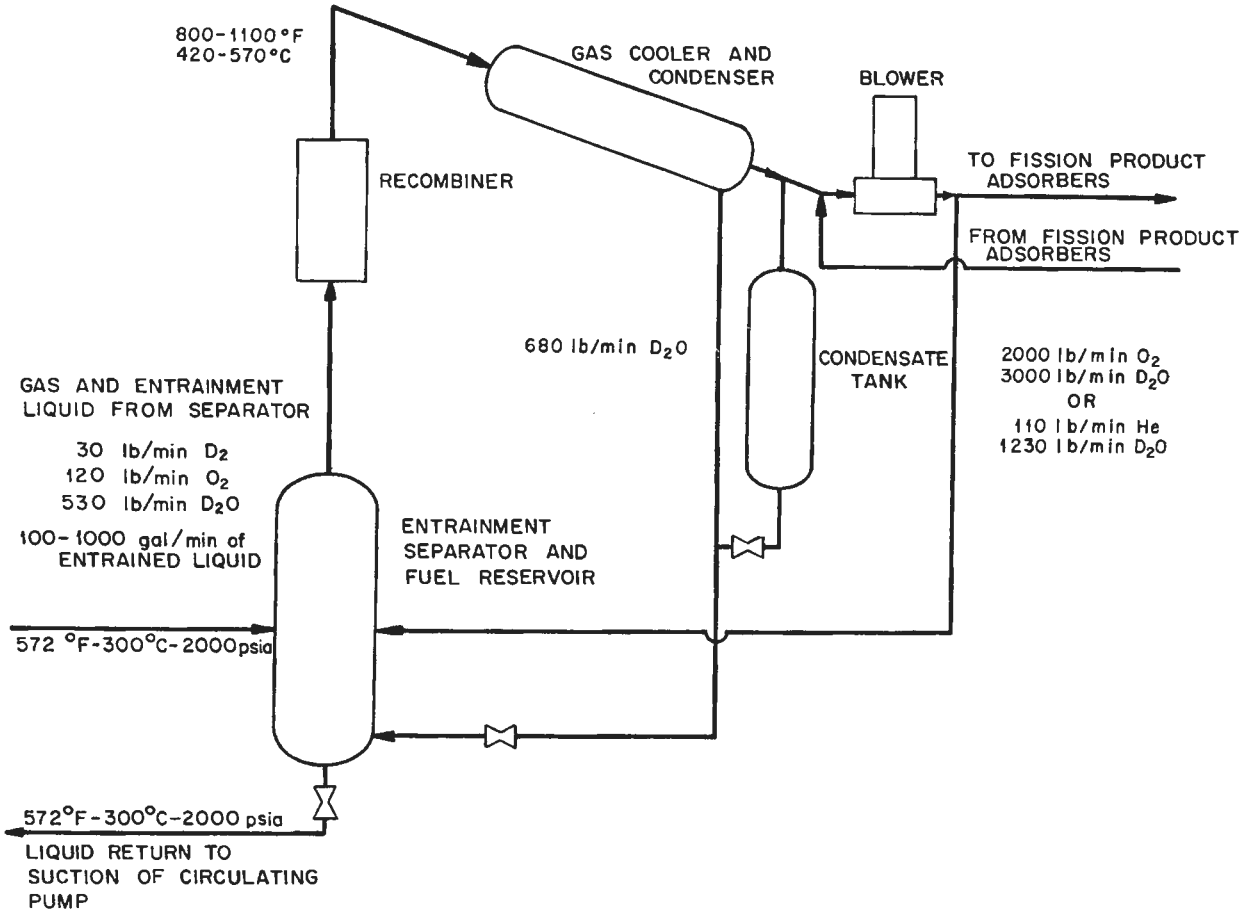


Figure 10. Flow diagram for high pressure catalytic recombiner system

of adequate equipment and methods for servicing the highly radioactive reactor components.

REACTOR OPERATION AND CONTROL

Aqueous homogeneous reactors rely for control completely on variable fuel concentration and temperature coefficient of reactivity. Shim control to put the reactor into operation, to change the operating temperature and to compensate for fission product poisons is accomplished by adjusting the core fuel concentration. Dependence is placed upon the negative temperature coefficient of reactivity to meet the regulation and safety requirements of the reactor.

The reactor will be started by an orderly charging of the fuel into the core and blanket systems. The slurry will be pumped directly from the storage tanks into the blanket circulating system. At the same time, heavy water will be evaporated from the core solution in the tanks and pumped into the core circulating system. When both the core and blanket systems are filled, they will be pressurized and the circulation will be started. At this point it will be desirable, although not absolutely necessary, to heat the circulating liquids to about 400°F by injecting steam into the heat exchangers. Then concentrated fuel will be added at a controlled rate. When the critical concentration is reached, the chain reaction will begin and the reactor temperature will rise with continued addition of fuel. When the temperature approaches the specified operating temperature, steam will be extracted from the heat exchangers to start the turbine. Final adjustments will be made to the concentration with the turbogenerator plant operating at full output. Excess fluid resulting from the expansion of the liquid in the reactor will be discharged to the storage tanks.

The process will be reversed in a normal shutdown of the reactor. Fuel from the core will be discharged to the dump tanks where the heavy water will be distilled and returned to the reactor. In this way the operating temperature can be reduced slowly until the reactor is cool. Depending upon the reason for the shutdown, the reactor may be started again by injecting the concentrated fuel that will have accumulated in the storage tanks. In an emergency the reactor fluids can be discharged rapidly into the storage tanks.

The temperature coefficient for the reactor will be about $-2 \times 10^{-3}\ \Delta k/k$ per °F. As there appears to be no good reason to require that the short time variations in power about the average be kept below several per cent, the regulation is achieved with ease. There seems to be little question that this temperature coefficient is sufficiently large to limit any fore-

seeable power excursion. However, the reactor must be designed and operated so that no reactivity changes can occur which are so rapid and of such magnitude that destructive pressures can be generated by the heated fluid as it is ejected from the core during the accompanying power surge.

The core tank as designed should be able to withstand an internal pressure of 400 psi greater than the external pressure without being damaged. With the pressurizer arrangement which was discussed, it is estimated that an instantaneous introduction of 1.7 per cent in reactivity above prompt critical is required to obtain this pressure rise. This corresponds to reactivity rise rates of about 1 per cent per sec for the reactor under start-up conditions and more than 2 per cent per sec for operation at powers above 1000 kilowatts. Rise rates as large as these could only be attained as a result of several simultaneous errors on the part of the operators and the failure of several safety circuits in the control system.

DESIGN OF THE TURBOGENERATOR PLANT

Equipment in the electrical generating plant is similar to that found in a conventional power plant. The major difference is in the steam turbine. Turbines for modern generating stations are designed to use superheated steam at temperatures above 1050°F and pressures above 1000 psi to obtain high thermal efficiencies. The turbine for this nuclear power plant would have to be designed for relatively low-pressure saturated steam. It probably would include the intermediate and low-pressure stages of the conventional turbine modified in design to obtain better utilization of wet steam. Although the mechanical efficiency of the turbogenerator would be high, the lower steam temperature would result in lower thermal efficiencies.

A flow diagram for the turbogenerator is shown in Fig. 11. Steam from the reactor plant is expanded from dry steam at 435°F and 365 psi absolute to steam containing about 12 per cent moisture at 30 psi absolute. The wet steam is dried in a moisture separator and is then expanded again to 12 per cent moisture and condensed at 92°F and 1.5 in. of mercury absolute. Before the feedwater is returned to the reactor plant it is heated to 315°F in three stages using steam extracted from the turbine. The gross efficiency of the turbogenerator plant is estimated to be near 27 per cent. Station electrical loads and heat losses in the reactor plant reduce the net station efficiency to about 23 per cent.

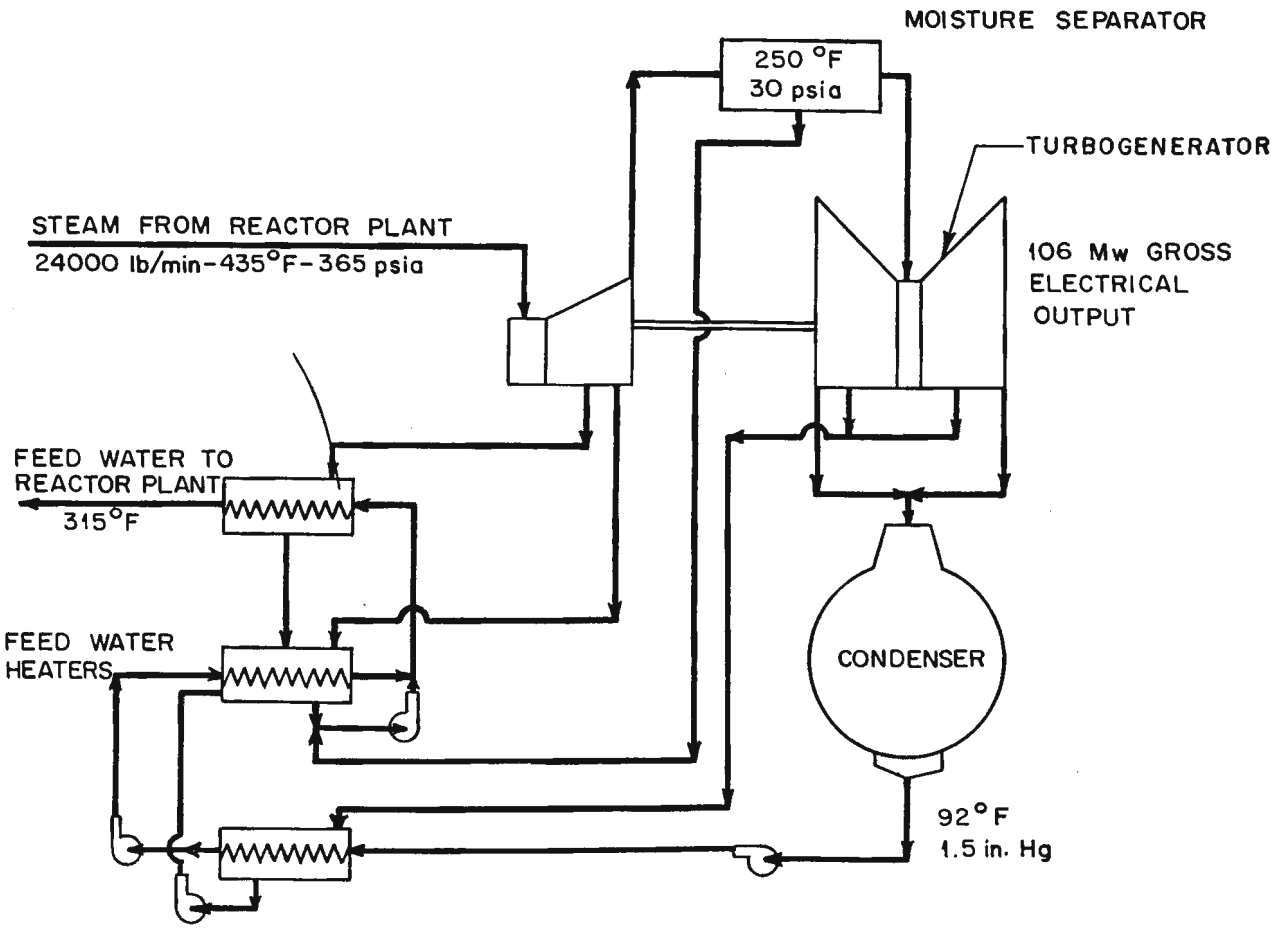


Figure 11. Flow diagram for turbogenerator

PROSPECTS FOR COMPETITIVE NUCLEAR POWER

The ultimate goal of the aqueous homogeneous reactor program is the development of reactor, fuel and processing systems which will make it possible to utilize nuclear fuels for producing low cost power. Much progress is being made in obtaining basic information about the fuel systems, the corrosion and erosion of materials, the processing of fuels and the design and construction of reactor equipment. The Homogeneous Reactor Experiment was built and operated successfully. Now it is being replaced by a slightly larger reactor, the Homogeneous Reactor Test.[7] There is confidence that the equipment can be made for a large power plant.

Whether power produced in large aqueous homogeneous reactor stations would be competitive in cost with power produced by conventional, fossil fueled plants depends upon the cost and the reliability of the nuclear power station and the fuel costs. During the past two years several designs have been made for large power stations containing aqueous homogeneous reactors. It has been estimated that a station capable of producing 100 megawatts of electricity and constructed during the next few years would cost between $20 million and $25 million. There would be an additional investment in inventory of thorium oxide, fissionable material and heavy water. This compares with a total cost of about $15 million for a comparable coal-fired plant. The cost of nuclear power stations should approach the cost of conventional stations when there is more experience in the design, construction and operation of the reactor plants.

For the present it must be assumed that nuclear power stations would operate as reliably as conventional stations. The electrical generating equipment represents such a small departure from the conventional plant that it is certain to operate satisfactorily. There is no substitute for experience in establishing the reliability of the reactor plant. Important information is being obtained in the testing of fuels, materials and reactor components. Experience with the Homogeneous Reactor Test will be invaluable. However, the final measure of reliability must be obtained from larger reactor installations.

Assuming the reliability of the reactor plants, nuclear power will be competitive power if the cost of nuclear fuel is significantly less than the cost of coal. Nuclear calculations indicate that the thorium fueled, two-region reactor described here would produce more U^{233} than it consumed. The doubling time should be about 1500 days of full power operation. The cost of the thorium feed would be negligible and the inventory charge on the fissionable material would be less than the value of the excess U^{233} produced. It is clear then that the fuel cost would be low if the fuel processing were cheap. The cost of fuel processing in plants which handle the fuels from five or more 100 megawatt power stations should contribute less than 1 mill/kwh to the total power cost. This cost would vary almost inversely with the plant size for smaller plants. Builders of the smaller power stations would have the choice of shipping their fuels to large, centrally located processing plants, where shipping is feasible, or accepting the higher cost of processing at the power plant site.

The prospects for producing low-cost power with aqueous homogeneous reactors are encouraging. Information being obtained from a development program now well under way should tell whether their promise is to be fulfilled within the next few years.

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Complete Automation of the Operation of Nuclear Reactors

By Jacky Weill,* France

PRINCIPLES

The use of a nuclear reactor for power production requires that automatic control devices be set up which are sufficiently positive and elastic in their operation to permit the operation of this method of energy production on an industrial scale.

Accordingly, the reactor is to be provided with controls capable of ensuring, either a constant power supply, or a power output which adapts itself to load requirements. In other words, the reactor can be compared with the steam turbines of the thermal plants of the conventional type in which the burners are controlled by the power required from the ac generators. Similarly, start up also has to be fully automatic, while every conceivable safety device is to be provided for the protection of the whole facility against any possible accident. Nevertheless, in view of the very nature of this new source of energy, a distinction must be made between two possible modes of operation of an atomic power plant (A) and (B) below.

A. The Atomic Power Plant Is Integrated in a Current Supply Network

It is to be expected, where an atomic power plant is integrated in such a network, that the electricity derived from atomic energy will not be the only power supply, but that hydroelectric or thermal plants also will feed energy into the system. Under such circumstances, it would appear desirable, according to the views currently held, to allow the nuclear power plants to keep running under full power, even during the "dead" hours, when the power requirements are well under the maximum capacity of the network, thus relieving the thermal and hydro-electric facilities at such a time. This will make it possible to effect a saving of coal or water for the latter, while, at the same time, the current initial cost of installation of an atomic plant requires, for a satisfactory write-off, that it be used nearly full time.

Other considerations than cost figures favor such a solution. Xenon poisoning, for instance, may rule out another start up, in some types of reactors until 30 hours have elapsed following a shutdown. On the other hand, where the breeding gain is favorable to the production of plutonium, it will be desirable to continue with this production as closely to full time as possible.

With proper regard for these considerations, an atomic power plant, fundamentally, will be, either operating at a high power rating, or else shut down, this shutdown being due to some failure or the need for servicing.

Thus, a device whereby the power output of the reactor can be made dependent on the power requirements of the ac generator will not be of use and it will be enough, for the case under review, to provide automatic start up, safety and automatic control devices.

B. The Atomic Power Plant Feeds an Independent Network

Such plants will probably be less numerous than those mentioned above, and their purpose will be either to supply a propulsion engine or a small power network isolated from the main ones. This is based on the assumption that the plant is to be installed right where the energy is to be used. Uranium consumption, under those conditions, may well be a non-negligible cost item. Fuel economies will then be in order, and the power of the nuclear reactor will depend on the power demand of the system.

We propose, in this paper, to outline a means of controlling reactor power in this fashion. It should be noted, however, that, in practice, the ratio of the maximum and minimum powers used will be less than five to one, so that reactor shutdown due to xenon poisoning will be avoided.

AUTOMATIC CONTROL DEVICE FOR CONSTANT POWER LEVEL REACTORS

Since power output is to be kept constant, it is necessary to provide a power measuring device such as an ionization chamber followed by a dc amplifier. The voltage delivered by the amplifier which is proportional to reactor power output, is compared to a reference voltage which represents the desired power. The difference in these voltages is the error signal, which is applied to the drive amplifiers of the reactor control rods which move the rods until the error signal cancels out. Thus, any power excursion is overcome by this control device (Fig. 1).

Good results have been obtained with this exceed-

Original language: French.

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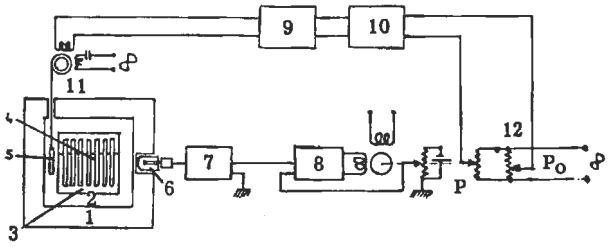

Figure 1. Error signal (P_0–P) control. 1. Concrete. 2. Reflector. 3. Moderator. 4. Uranium. 5. Shim plate. 6. Ionization chamber. 7. Direct current amplifier. 8. Power recorder. 9. Magnetic amplifier. 10. Electronic amplifier. 11. Motor. 12. Comparator

ingly simple control device, which is already in use on a number of reactors.[1] Figure 8 shows the stability which it provides. In some other control devices, the rods are moved at a variable speed, which is a function of the error signal. Control devices which use constant rod speed are, however, adequate for controlling a reactor used solely as a source of energy.

The only other problem to be solved, in an installation of this type, is that of providing automatic start up. A device which could be called a "control system director" for the start up rods (Fig. 2) might prove adequate. It would control the rod position at start-up so that reactor power output would increase according to an established scheme, determined by the reactor parameters. The automatic control device will cut in as soon as the desired steady power output is reached. This type of extremely simple and effective instrumentation would be positive in operation and easy to operate satisfactorily for a great variety of installations. On the other hand, it would be definitely inadequate if frequent rapid variations in reactivity were to take place during reactor operations. Such fluctuations could be caused, for example, by variations in the pressure of the circulating fluid between the reactor and the heat exchangers in some installations. The type of master control described above will not function satisfactorily for reactivity excursions exceeding 100 pcm/sec. This is due to the fact that

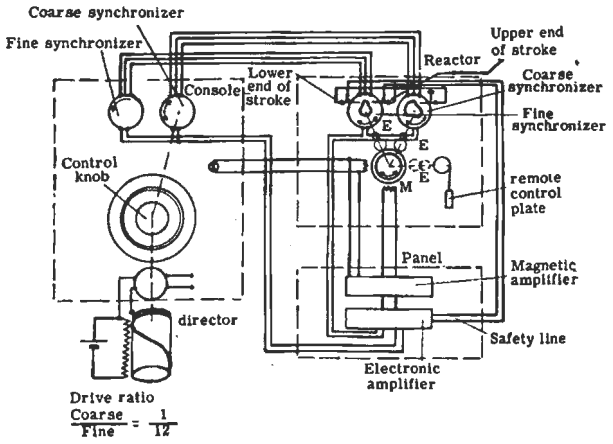

Figure 2. Director control

overshooting phenomena may appear, when this type of control device is used, either with changes in reactivity, or with variations in P_0, the reference power level. This phenomenon is depicted in Fig. 3, which shows a curve B, plotted for a 100 kw variation in the power requirements of the Saclay reactor, and curve A obtained under the same circumstances by means of an analog computer. An analogous curve will be obtained for a 100 pcm variation in the reactivity. Attention should be drawn to the fact that a factor of two power excursion, for instance from 1000 to 2000 kw, will cause an overshoot of approximately 1000 kw, followed by oscillations which may last for several minutes. Such effects cannot be tolerated, so the type of control described above will be suitable only in systems which are characterized by stable operating conditions. For non-stable operation, it would be necessary to make the control system more

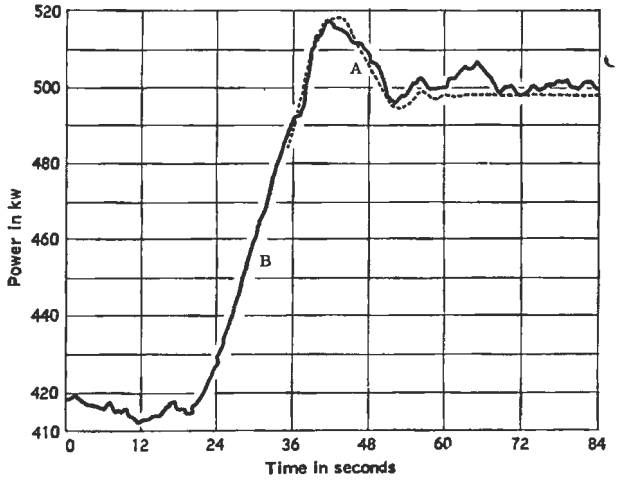

Figure 3. $u = (\Delta P_0 - \Delta P)$ control. Change in power requirements. A: curve obtained with analog computer. B: curve obtained with the Saclay reactor

complicated; in particular, it would be necessary to use an additional parameter in the error signal. This parameter can conveniently be the reactivity. Under such conditions, the control device can be identical to that described below, which is used for the automatic control of variable power reactors.

AUTOMATIC CONTROL DEVICE FOR VARIABLE POWER REACTORS

This device, while it offers the same characteristics as that described above, will make it possible to vary the power during operation, while providing appropriate safety devices for such operation. We already have described, in another article,[2] the principle of this control which makes use of a power error signal, and a reactivity correction signal, u, which is injected into the control rod drive mechanism, and has the form:

$$u = A(P_0 - P) - BR$$

in which P_0 is the required power, P is the power

output of the reactor at the time considered, A and B are two injection coefficients, and R is a signal which corresponds to the reactivity or, more accurately, to the period of the reactor at the time considered. This signal will be a function of the position of the shim rods, the minimum speed of which shall be chosen so as always to make it possible to cancel out rapidly the signal u. This device can be made as shown on the schematic diagram of Fig. 4.

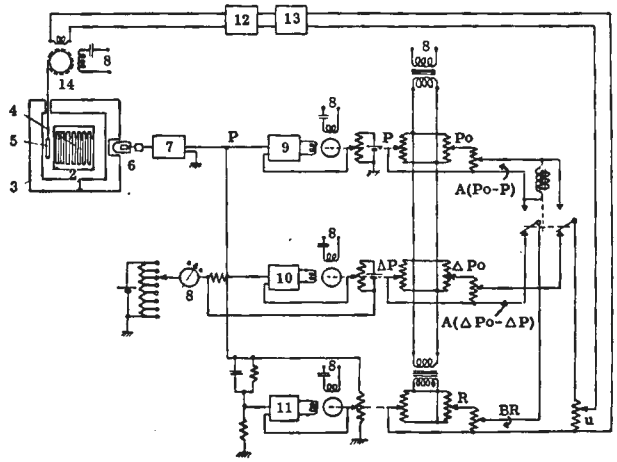

Figure 4. Schematic of automatic control device. 1. Concrete. 2. Reflector. 3. Moderator. 4. Uranium. 5. Regulating rod. 6. Ionization chamber. 7. Direct current amplifier. 8. Potentiametric switch. 9. Potentiometric power recorder. 10. "Vernier" potentiometric power recorder. 11. Potentiometric reactivity recorder. 12. Magnetic amplifier. 13. Electronic preamplifier. 14. Two phase engine

It should be noted that the method suggested resorts to two types of power error signals. In order to achieve a high degree of precision as well as considerable stability, a fine power measuring device is used. This instrument makes a coarse measurement of P by bands, each of which amounts to a twentieth of the rated power of the reactor, and provides a "vernier" reading ΔP, which gives the accurate indication within the relevant band. This measurement, which is described elsewhere[3] is fully automatic since passage from one band to the next is automatic (Fig. 4).

Only signal P is used for the first phase of the operation, since the correction signal is proportional to $P_0' - P$. Once stage P is reached, signal $P_0' - P$ which is controlled by a reversing relay, gives way to correction signal $\Delta P_0 - \Delta P$. Notice that with this method, there is a saturation value for the signals, to be designed by $A_P\ (P_0' - P)_{max}$ and $A\,\Delta_P\ (\Delta P_0 - \Delta P)_{max}$. In view of this, since u tends toward permanent canceling out, we shall distinguish, in the first phase of the power variation, a maximum reactivity: $R_{P\,max} = (A_P/B)\ (P_0 - P)_{max}$ and, similarly, in the second phase, a maximum reactivity: $R_{\nabla P\,max} = (A/B)\ \Delta P\ (\Delta P_0 - \Delta P)_{max}$.

This shows that throughout the first phase, which may extend over a wide range of power levels, reactivity $R_{P\,max}$ will not be exceeded, and that it will be easy, by a suitable adjustment of factor (A/B), to achieve power variation for any given constant value of the reactivity sought. Similarly, it will be possible to choose, for the second phase, a value $R_{\Delta P\,max}$ which will ensure, if the reactivity remains constant, proper control of the reactor close to power $P_0 = P_0' - \Delta P_0$, chosen as the level at which regulation is desired.

It is obvious that cancelling out of signal u which takes place when $P = P_0$ or $\Delta P = \Delta P_0$ (the reactivity value R is zero at the same time) causes the reactor to operate in such a way that the power level P_0, at which regulation is to take place, is achieved just as the excess reactivity drops to zero. Thus, there is no overshooting of P_0 and clearly, no oscillations. We show, in Figs. 5a and 5b, the positive and negative power excursion obtained inside a $(\Delta P_0 - \Delta P)$ band and, in Fig. 6, a more extensive excursion obtained by successively bringing into play signals $(P_0' - P)$ and $(\Delta P_0 - \Delta P)$ (see details on Fig. 7).

This automatic control device provides all the necessary safety features, namely:

a) It makes it possible to keep reactor power output constant at any desired level.

b) It makes it possible to vary power at will. This variation, brought about by a change in P_0, takes place automatically at a predetermined maximum reactivity.

c) It also makes it possible to control power by means of P_0, the power required by the energy consumption demanded on the shaft of the turbine driven by steam produced in the reactor.

d) It makes it possible, during these operations, for the reactor power output not to exceed a given limit P_{max}. This is achieved by setting a limiting value $P_{0\,max}$ on P_0, the required value.

e) It makes it possible to limit the value of reactivity to P_{max} at any time during a power excursion.

f) It provides for automatic start up from zero power to power P_0, as required, either by the operator or by the turbine.

This type of control has been the object of a stability study. Figure 8 shows, for comparative purposes, the stability performances obtained with the Saclay reactor under $(P_0 - P)$ control (curve A); under $[A\ (P_0 - P) - BR]$ control (curve B); and under manual control (curve C). This last named curve shows the best performance which can be achieved under manual control, this calling for about one adjustment a second of the shim plates.

TEMPERATURE AND POISONING EFFECTS

The control system described does not allow fully for the temperature and xenon effects, for these generally cause a greater negative reactivity than can be compensated for by a single shim plate. Thus, the position of the compensating rods, which are designed to counteract such effects, will have to be made dependent on the over-all configuration. This can be done very simply by the use of contacts cor-

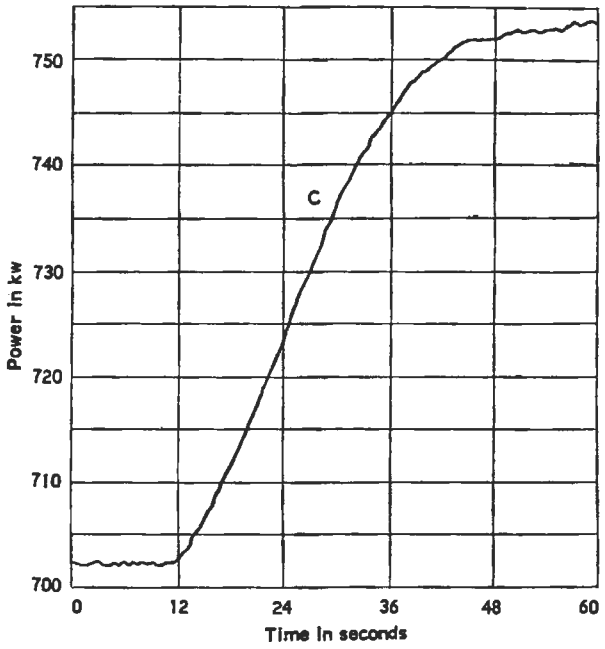

Figure 5a. $u = (\Delta P_0 - \Delta P) - BR$ control. Positive power requirement

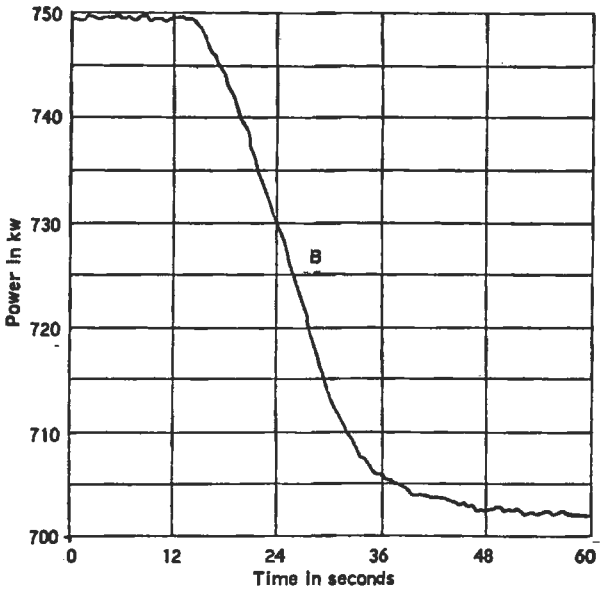

Figure 5b. $u = (\Delta P_0 - \Delta P) - BR$ control. Negative power requirement

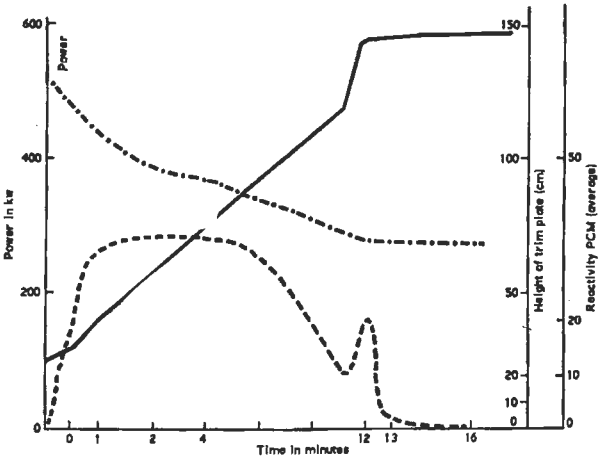

Figure 6. Power excursion for automatic passage from 100 kw to 600 kw. The corresponding reactivity excursion and height of trim plate also are shown. (Experiment carried out on P2)

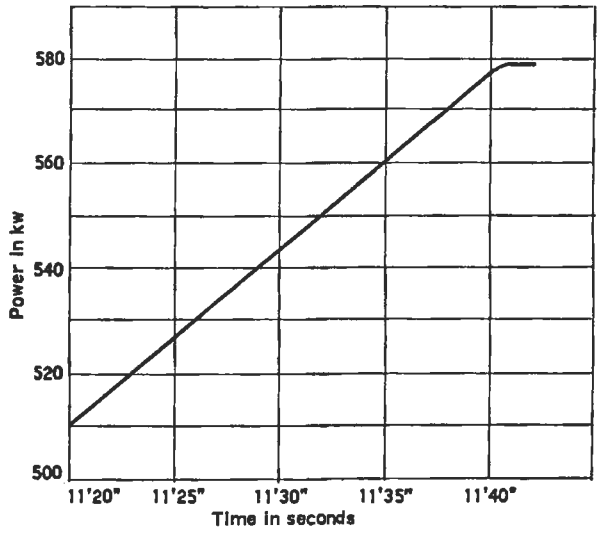

Figure 7. $u = (\Delta P_0 - \Delta P) - BR$ control. Details on Fig. 6

responding to a well defined stroke of the control plate. A contact placed at the top of this stroke will determine the upward stroke of a compensating rod, a contact at the bottom of the stroke shall set the limit of downward motion, and a halfway contact shall cause this upward or downward motion to stop. It will be necessary to choose, for the ascending speeds of the compensating rods, smaller values than for the shim rods, so that the latter can compensate both for the effects of positive or negative changes in the reactivity. As soon as a compensating rod reaches the end of its stroke, contact will cause another compensating rod to operate, and so forth until they are all used. This device, while retaining the features of automatic control which have been described above, will be usable regardless of the initial or final temperature and poisoning conditions.

OPERATING SAFETY

The device described is completed by safeties which shut down the reactor in case power P_0 and/or reactivity R_{max} are exceeded, as well as if a limiting temperature, typical of the reactor, is overshot. Experience shows that the device could ensure satisfactory control for several months without any breakdown. However, it will be necessary to provide two

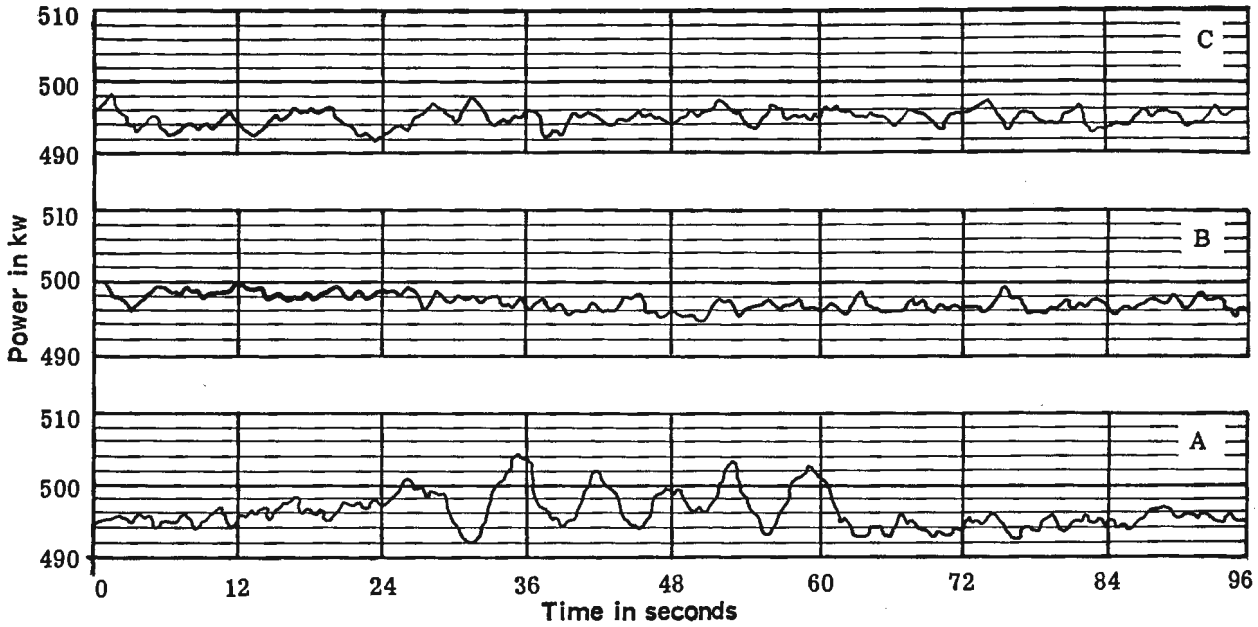

Figure 8. Stability achieved with various types of control. A. Error signal (P_0 — P) control. B. Error signal A (ΔP_0 — ΔP) — BR control. C. Manual control (touching the controls every second approximately)

complete control devices: should one fail, the other will automatically take over its function, this change being brought about by the fact that error signals u and P_0-P are of opposite signs.

This type of safety provides, in particular, for control rod jamming or undesirable displacement of same.

NOTE: Experimental results obtained with an analog computer will be submitted later, to the extent that these results can be obtained before August 1, 1955. In particular, some outlines of the control achieved by means of an error signal which includes the temperature will be shown.

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The Dounreay Fast Reactor Project

By J. W. Kendall and T. M. Fry,* UK

The fundamental reason for developing the fast reactor for the generation of industrial power was that it seemed capable of much more efficient utilization of uranium fuel than a system based on thermal fission. The energy content of U^{238} is little different from that of U^{235}, but when natural uranium is fed into a thermal system all but a few per cent, depleted in its content of U^{235}, must be thrown on to the ash heap. The need to conserve uranium may not be great at present but ultimately the possibility, afforded by the fast system, of using almost all of the uranium should be most attractive.

A fast neutron chain reaction cannot be sustained in uranium of natural enrichment. The core of the reactor must be charged with a much higher proportion of fissile material, either U^{235} or Pu^{239}, than is present in natural uranium. The cost of fissile material is measured in thousands of pounds per kilogram so that the initial charge represents a substantial capital investment. If it is to be used as a "catalyst" for deriving power from U^{238}, each kilogram of fissile material must yield hundreds of kilowatts of electricity. This sets one objective for development; we cannot consider the fast reactor as an economic source of power until ratings of this order have been demonstrated. Moreover, this rating must be based on the total fissile investment in the whole system, including processing plants, and not just that in the reactor.

The second target for development is outlet temperature. The heat produced in a power reactor must be at a sufficiently high temperature, so that it can be converted to power with useful efficiency.

Ideally one would like to feed natural uranium into the reactor and remove the fission products continuously. In practice it will be necessary to remove the fuel periodically for purification and re-fabrication, an expensive process which also increases the investment of fissile material needed to operate the system. This sets the third criterion of successful operation of a fast reactor; the cost of processing the fuel must not be excessive. If the process is expensive or takes a long time, the fuel must produce a lot of heat before processing is necessary. It is difficult to give an exact figure, but, with the processes at present envisaged, a burn-up of the order of 10% of the fissile material is probably needed. At the ratings required, this would take a few months irradiation in the reactor.

Finally, if the reactor is to be sited in the United Kingdom, it must be safe.

At Dounreay, on the north coast of Scotland, the United Kingdom Atomic Energy Authority is building a fast reactor. The project is essentially experimental, for no engineering experience is available on which to design a core with the rating, outlet temperature and burn-up required. The best way to get this knowledge seems to be to build a plant having the most flexible design that can be achieved. As failures are replaced by better components it is hoped to demonstrate that a fast reactor can be a sound economic proposition for the production of industrial power.

At this point mention must be made of the achievement in the United States of the production of electrical power from a fast reactor, the EBR, as long ago as December 1951. While this has shown that many of the technical problems can be overcome, a much bigger output must be attained if the power is to be produced at an economic cost. While not going the whole way to that suitable for a full-scale generating station, a core size has been selected for the Dounreay reactor, about 2 feet in diameter and 2 feet long, from which it should be possible to extrapolate to the larger reactor.

The plant will consist of a pot through which enough coolant can be pumped to remove more than 60 megawatts of heat. In this pot, tube plates and other structures can be placed to support the blanket and core of the reactor. If experience shows that the first designs are unsatisfactory, the contents of the pot can be removed and a fresh start made.

If high heat output is to be obtained, a coolant of high specific heat and high thermal conductivity is needed because of the small core size, while a low vapour pressure is a great advantage. These considerations led to the choice of liquid sodium with possibly some addition of potassium to lower the freezing point. Small scale experimental experience of these coolants was available when this choice was made.

The reactor is going to cost some millions of pounds, and its core will contain a substantial quantity of fissile material. It therefore represents an investment which must not be lost as the result of an accident. Because of the persistence of heat re-

*U.K.A.E.A., Risley and Harwell.

lease after reactor shut-down, it is essential that a supply of coolant to the core shall be guaranteed at all times. The core of the fast reactor is so highly rated that it would quickly rise to melting point if the coolant were removed. The design of the cooling circuits has therefore been prepared with this in mind.

The circuit will be made from stainless steel throughout. This is a good material for high working temperatures and also helps in getting a clean oxide-free circuit. The design and method of fabrication has been taken over from that of a highly radioactive chemical plant. Here a stainless-steel plant, containing nitric acid and other corrosive chemicals, had to be built to such a standard that leaks would not occur. By the exclusive use of butt-welding and radiographic inspection of almost every inch of weld, this requirement was met. The pot and the pipe-work forming the primary circuit of the fast reactor can be made equally reliable. All valves are excluded from the design and there will be no connected dump-tanks into which the coolant might accidentally be transferred.

Twenty-four parallel loops, each comprising two separate heat-exchanger sections and an electromagnetic pump, will complete the primary circuit. The lower part of each loop will be jacketed with an outer tube through which the secondary coolant will be pumped. Above this will be a short flattened section against which are pressed the electro-magnetic windings. At the top will be another jacketed heat-exchange section, providing thermo-syphonic rejection of shut-down heat to air in the event of total failure of electrical supplies. Twenty of the loops feed the core from a common header-space at the top of the pot, while four more provide coolant for the blanket.

Adhering to the principle that the removal of heat must be maintained, multiple secondary and subsequent heat-rejection circuits will be provided. Even the power supplies will be separated. Electrical power for the pumps will be provided by Diesel generators, each of which will be connected to only two of the primary pumps and their subsidiary circuits. Thus the failure of one generator will only cut out one-tenth of the pumps. The flow through the core will fall by a rather larger fraction because some coolant will flow in the wrong direction through the ineffective pumps. Even so, the reduction in flow will not be by a dangerously large amount.

Sodium or sodium-potassium alloy will be used for the secondary coolant. On grounds of safety it is argued that the secondary coolant must be chemically inert to the primary coolant, which rules out water. The secondary coolant will carry the heat to a boiler-house for steam raising. As there will be no radioactivity involved, a reaction between sodium and water in this part of the plant would not involve widespread hazard. Waste heat will be disposed of by a sea-water cooled condenser.

The secondary coolant must not become radioactive, so a neutron shield is needed between the pot and the heat exchangers. This shield can also perform another function. Measurements of fission rate must be obtained even when the reactor is shut down. In fact, it is especially important to make measurements when the reactor is not generating appreciable quantities of heat, for it is then that reactor accidents are likely to happen. A detector of thermal neutrons, placed in a mass of moderating material outside the pot, should combine sensitivity with accessibility.

The neutron shield consists of graphite stacked around the pot to a thickness of 4 feet. Most of the graphite will contain enough boron to absorb the neutrons when they have slowed down. In a belt at the level of the core, pure graphite will be used for all but the outermost layer. In this region a high density of thermal neutrons will build up, making it possible to determine the fission rate down to very low levels. To ensure that this rate can never fall below the minimum observable, a neutron source will be placed in the reactor.

The advantage, which it is hoped will be gained by the use of borated graphite for absorbing neutrons, is that very much less long-lived activity will be induced than if steel were used. This should simplify the problems of dismantling the neutron shield if ever it should be necessary. The graphite will be kept hot to reduce radiation damage.

The graphite and the pot will be supported on a steel structure standing on a concrete base, which forms part of the biological shielding. This will consist of a concrete bowl of outside diameter almost 90 feet, 45 feet high, with a minimum thickness of 5 feet. These dimensions are determined by the size of the heat exchangers and the activation of the coolant. The roof slab, which has a central hole 20 feet in diameter, will be supported on a ring of vertical columns. These introduce complications because they will stand in the space to be occupied by heat-exchange equipment, but the slab is too large to be supported round its rim without introducing a prohibitive concentration of reinforcement near the centre. The heat exchangers will hang from the roof slab.

Pure sodium is a better heat transfer medium than the alloy and is much cheaper, but, if it is to be used, the entire primary circuit must be kept above the freezing point of sodium so that blockages cannot occur. The whole of the vault will therefore be made into a hot-box, with air heaters. The concrete and the columns supporting the roof will be lagged and air cooled. In the early stages of operating sodium-potassium alloy will be used so that the vault will be accessible when the activity of the coolant has died away after shut-down.

This is a reactor which may require alteration and development before its operation is entirely satisfactory. For this reason there is to be a large hole in

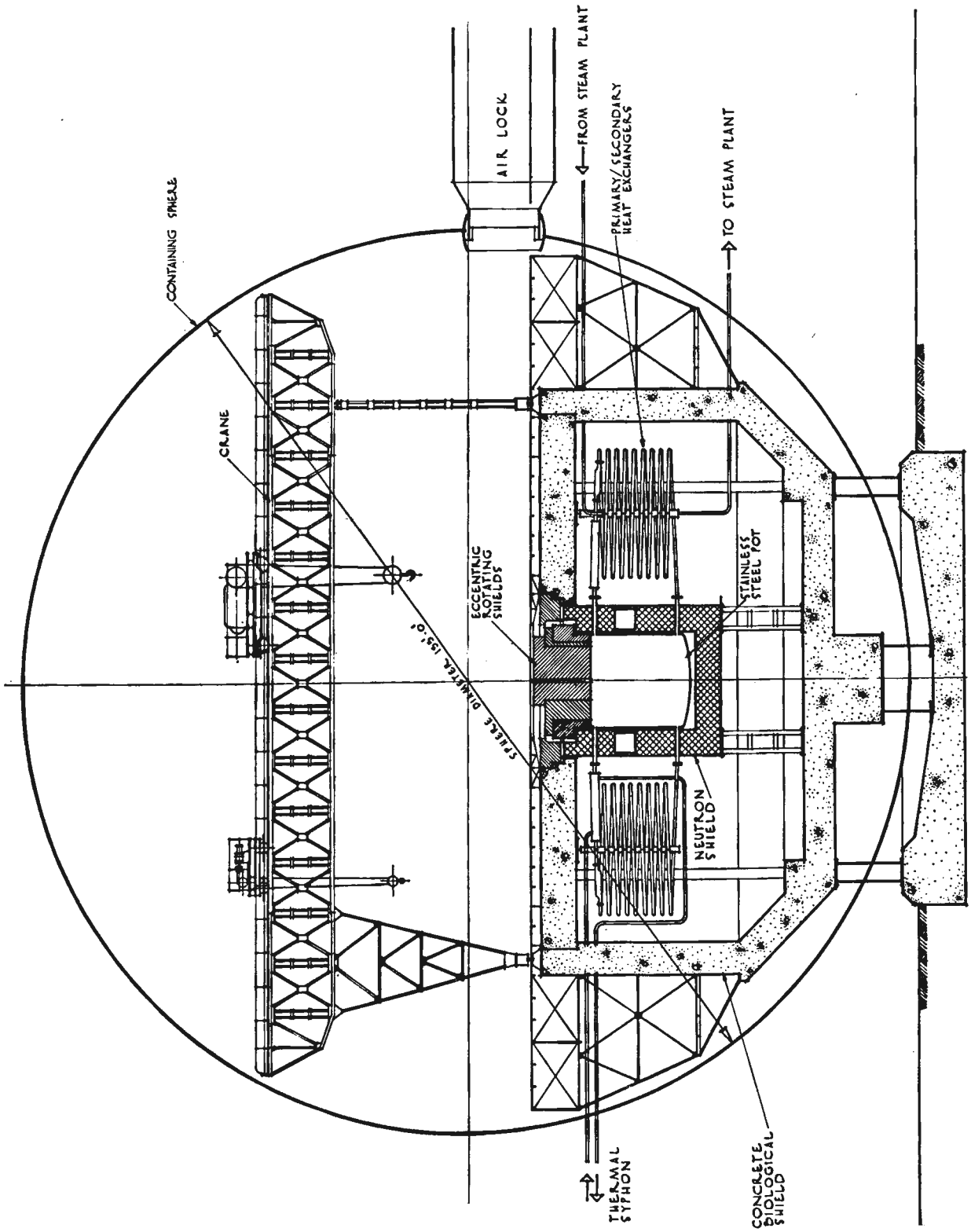

Figure 1. Arrangement of reactor

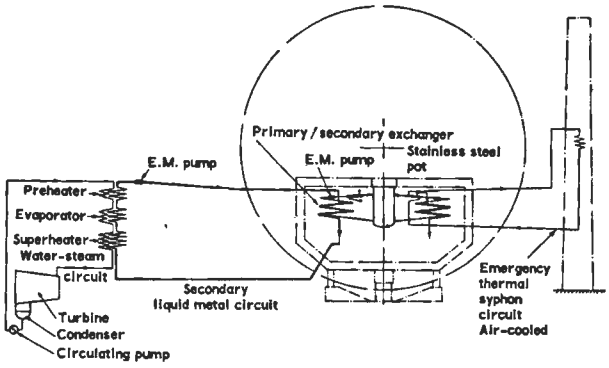

Figure 2. Diagrammatic arrangement of liquid metal and steam circuits for fast reactor

the roof slab, closed with removable shielding. It should be possible to remove the equipment from the vault through this hole. In the centre, the top of the pot will be closed by a thick plug of shielding in the form of eccentric cylinders with liquid lute seals, which can be rotated to align the charge machine over the inside of the pot. During operation the plug will be sealed mechanically against the coolant pressure with an intervening gas blanket.

Following the example of the Knolls Atomic Power Laboratory at West Milton, the Dounreay reactor will be enclosed in a steel sphere 135 feet in diameter and about 1 inch thick.

A theoretical examination of the problem of the safety of fast reactors has been made and the magnitude of the explosion that would occur, if the reactivity were rapidly increased beyond prompt critical, has been estimated. It is doubtful whether a pressure vessel of practicable dimensions could contain such an explosion. It is confidently believed, however, that the danger can be eliminated by suitable design of the control system and of the core itself.

Although an explosion will not occur, one cannot be quite sure that there will be no sort of plant failure which could result in fission products escaping. With something like a hundred million curies in the core, an escape of more than a few per cent would be a very substantial hazard over a wide area. The sphere has therefore been designed to contain these fission products in the very improbable event of their escape from the core.

The liquid metal coolant brings with it a risk of fire. To reduce this risk no significant quantity of water will be allowed inside the sphere, but there will be about 13 tons of oxygen in the air. The vessel is not designed to contain the pressure that would develop if all this oxygen were to combine instantaneously with sodium.

Sodium is a poor fuel. Like other fuels it reacts with air in the vapour phase. Its high boiling point, 883°C, and high latent heat of vaporisation, 1 kilocalorie per gram, make it slow to burn in air. The heat of combustion is only 2 kilocalories per gram, so that half of this must be fed back to vaporise a further supply of fuel for the fire. The high thermal conductivity tends to conduct heat away from the burning surface, which is quickly covered with an insulating layer of oxide. A sodium fire will therefore not be self-sustaining at a high rate of combustion. The heat necessary for vaporisation must be derived from some other source. In this case the only source available would be the radioactive decay of fission products in the reactor fuel.

Conditions may be envisaged in which it has ceased to be possible to extract heat from inside the sphere. The temperature of the reactor rises. The neutron reactor closes down, but the emission of heat from the decay of fission products continues. In the worst case all this heat might go to evaporating sodium. The burning of the sodium vapour produces twice the amount of heat absorbed in vaporisation. All this heat might be transferred to the air. This, in turn, would transfer heat, by natural convection, to the steel pressure vessel, which would be cooled by the outer air. On these assumptions the air would reach a maximum temperature of 340°C and a maximum pressure of 16 psi some fifteen minutes after the fire started. Making some additional allowance for barometric variation and so on, the design figure for the maximum internal working pressure of the sphere has been taken to be 18 psi.

The internal pressure is not, however, the only criterion to be considered in designing the sphere. After the reaction had gone to completion and all the oxygen in the sphere had been consumed, the temperature would eventually fall, leaving a partial vacuum inside. The removal of the oxygen would leave an external pressure of 3 psi. The pressure at which it would begin to buckle depends on the uniformity of the steel shell. The most pessimistic estimate is that, at 5 psi external pressure, a shallow dimple some 18 feet in diameter would appear at the top. This would not raise stresses in the steel greater than 1 ton/in^2. The most unfavourable combination of circumstances could not impose a greater pressure on it than this.

Because reliance is placed on heat transfer from

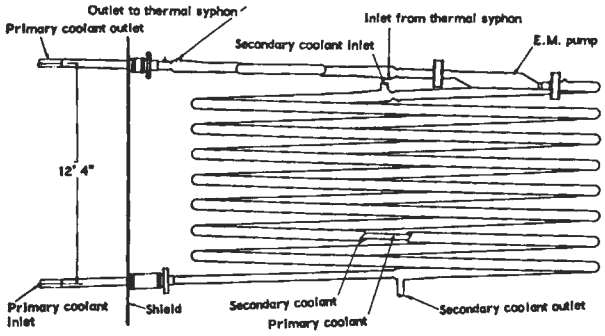

Figure 3. Primary-secondary exchanger

the outside to dispose of heat in the event of an accident, no lagging will be provided on the outside of the sphere and water will be piped to the top so that it can be pumped over the outer surface to assist the removal of heat if need be.

Prudence dictates that, in the design of such a novel and highly rated reactor, extreme precautions be taken. It is hoped that, when experience in fast reactors has been gained, they will provide a safe and enduring source of power for mankind.

Review of Fast Power Reactors

By W. H. Zinn,* USA

A nuclear power industry will be a major contributor to world energy resources if the supply of fissionable materials proves sufficient to give the industry a long life. In the numerous papers of the Geneva Conference on the subject of the search and discovery of uranium and thorium ores, there is vivid evidence that this fact is well understood. The evident success of the search for uranium and thorium in many parts of the world gives assurance that a nuclear power industry can, indeed, look forward to a supply of raw material adequate for a considerable time.

How long the uranium and thorium supplies will support a nuclear power industry depends not only on the amount available from the earth's surface but also upon the efficiency of its utilization. To increase the utilization of the raw material always must be a prime objective of the nuclear energy technologist, just as it has been a traditional objective of the developers of coal-fired electrical generating plants. It may be acceptable in the early development of the industry to place as the first goal reactor power plants which are competitive in an economic sense with the coal-fired plants. Success in this inevitably will bring a demand for an increased utilization of the nuclear raw material. It appears that the goal of economically competitive nuclear power plants may be realized in the near future in a number of countries, including the USA. It is equally clear that in the matter of the utilization of raw material nuclear technology is in a most embryonic state.

The first measurements of the number of neutrons emitted per fission, made in 1939, raised the hope that there would be more neutrons available than just those required to maintain a chain-reacting system. Some years later, about 1944, measurements of ν, α, and η† for uranium had been made with considerable precision. At that time, the question of breeding fissionable material from U^{238} and thorium began to receive serious attention.

If it is accepted that the maximum conversion ratio for a reactor is just $\eta - 1$, it is seen that the quantity of fissionable fuel assigned to a reactor can be increased by conversion at most by a factor of $1/(2 - \eta)$. It is here assumed that η is the value appropriate for the operating reactor, that is, the temperature and hence the neutron spectrum are to be characteristic of the operating reactor. Generally for thermal reactors η so defined will be less than 2.

As long ago as 1945, the values of η had been established as follows for a neutron velocity of 2200 m/sec:

$$\eta\ (U^{235}) = 2.11;\ \eta\ (U^{233}) = 2.34;$$
$$\eta\ (Pu^{239}) = 1.94.$$

Of the two regenerative fissionable isotopes, U^{233} and Pu^{239}, it appeared from this evidence that only the conversion of thorium to U^{233} had any sizable chance of contributing to the supply of fissionable fuel. The 1955 values of η are, for a neutron velocity of 2200 m/sec:

$$\eta\ (U^{235}) = 2.08;\ \eta\ (U^{233}) = 2.31;$$
$$\eta\ (Pu^{239}) = 2.03.$$

The modern values do not appreciably change the situation as it was believed to be ten years ago, although the improvement for the plutonium value is noted.

Because U^{238} seems to be more than one-half of the raw material resource available to the world, it is an important matter to devise reactor systems which potentially correct the unfavorable situation indicated by the above thermal neutron values for η. There were early indications that the value of ν did not vary remarkably with the energy of the neutrons. This indicated that an improvement in the value of η with neutron energy should be sought in an improvement in the value of α. In the USA, a number of laboratories, particularly the Los Alamos Scientific Laboratory, Knolls Atomic Power Laboratory and Argonne National Laboratory, have made measurements which give an indication of the variation in α with neutron energy. These measurements are extraordinarily difficult and no complete table of the variation of α with energy for any of the fissionable isotopes exists. Theoretical analysis of the question has also been undertaken and the variation of α with energy has been predicted and, indeed, the not unexpected prediction was made that with increasing neutron energy a range of smaller values of α is entered. In order to capitalize on this situation, it is necessary to devise reactors which operate on neutron energies other

*Argonne National Laboratory.

†By definition: ν = number of neutrons emitted per fission, α = ratio of cross section for radiative capture to cross section of fission, and η = number of neutrons emitted per neutron absorbed = $\nu/(1 + \alpha)$.

than thermal, and there is some indication that the most favorable situation would be achieved if the average energy of the neutrons in the reactor is kept as high as the non-moderating materials composing the reactor naturally permit.

THE PLUTONIUM-FUELED REACTOR

In the following discussion, only Pu^{239} and U^{235} will be considered as primary fissionable material, since it is believed that the breeding of U^{233} is a feasible accomplishment with thermal neutrons, although it is not at all excluded that it could be a successful enterprise with neutrons of non-thermal energies. Since α is a function of neutron energy, care must be taken to specify the energy involved. Experiments made by Knolls Atomic Power Laboratory have shown that a value of α for Pu^{239} of 0.63 is to be expected for neutrons of median energy of 30 electron volts. For neutrons of median energy of 15 kev, a value of 0.42 is to be expected. These intermediate energy neutron values of α do not indicate any improvement in the value of η over the value for thermal neutrons. Measurements on the Experimental Breeder Reactor by Argonne National Laboratory show that α for higher energy neutrons has a more favorable value. Unfortunately, the spectrum at no place in the reactor is mono-energetic and, therefore, in quoting the values of α, the appropriate spectrum also should be illustrated. Examples are given by Lichtenberger.[1] For the purposes of this discussion, it is sufficient to say that for Pu^{239} values of α as small as 0.06 and for U^{235} values as small as 0.11 have been obtained. If we accept that ν for Pu^{239} for fast neutrons is 2.88 and for U^{235} is 2.46, the corresponding maximum values of η are $\eta(U^{235}) = 2.33$; $\eta(Pu^{239}) = 2.7$. This leaves open the question of the actual value of ν for high energy neutrons, but it is quite clear that it is not less than the value cited.

It is the fact that for fast neutrons such favorable values of η can be derived which have been the incentive for the development of breeder reactors operating on unmoderated neutrons.

To develop a reactor for the large usage of U^{238}, a fuel cycle in which plutonium is fed into the reactor as the fissionable isotope must be realized. It is estimated by Spinrad, Carter and Eggler[2] that a Pu^{239}-U^{238} fuel operating thermal reactor cannot have an η greater than 1.84. This is an estimate which does not include losses due to parasitic capture in structural materials of neutrons in the reactor and does not take into account any losses in the chemical or metallurgical processing. For such a thermal reactor system, the maximum utilization of U^{238} will be about 5 per cent. The values of η for Pu^{239} which are indicated for a fast reactor seem to give, theoretically at least, a very adequate margin for the loss of neutrons in the reactor and for losses in chemical and metallurgical processing and give a real possibility of obtaining a breeding ratio greater than one for a practical fuel cycle.

BREEDING RATIO

The breeding ratio (B.R.) is

$$\frac{\text{Atoms of fissionable isotope generated}}{\text{Atoms of fissionable isotope consumed}}$$

or

$$\text{B.R.} = \frac{\nu - 1 - \alpha - A - L + F(\nu' - 1)}{1 + \alpha} \qquad (1)$$

in which A = loss of neutrons per fission of primary fissionable isotope to structural and coolant materials, L = leakage of neutrons from the reactor per fission of primary fissionable isotope, F = number of U^{238} atoms fissioned per fission of primary fissionable isotope, and ν' = neutrons per fission for U^{238}.

In order to make an estimate of the losses to be expected and thus of the breeding ratio of a reactor system, it is necessary to establish the composition of the reactor and the operating cycle.

In the USA, two reactors have been operated as fast reactors: one, constructed by Los Alamos Scientific Laboratory, was fueled with plutonium; the other, constructed by Argonne National Laboratory, is fueled with U^{235} enriched to at least 90 per cent. Because of a strong desire to limit the quantity of fissionable material committed to these reactors, the fissionable material is used in as undiluted form as possible. This has the consequence of giving a neutron energy spectrum whose median energy is as high as it is possible to conceive in an operating unit. Power reactors of a practical kind operating with unmoderated neutrons will have a much greater dilution of the fissionable material. This will result in a degradation of the median energy of the neutrons. The effect of dilution has been studied and some results are treated by Okrent, Avery and Hummel.[3] Since it is more realistic to speculate about dilute reactors, some estimates for a dilute reactor of the various factors which enter into the breeding ratio formula will be made in the following. Inevitably in these estimates it will not be possible to use constants which apply to the neutron energy appropriate for the dilute reactor. This situation must be accepted as indicative of the state of the art. The ingredients of the reactor will be assumed to be stainless steel, sodium, U^{238}, and either U^{235} or Pu^{239} as the primary fissionable isotope. For the dilute breeder, Table I sets forth the volume ratios of these materials and the corresponding atomic ratios.

It is assumed that the dilution of the primary fissionable material is by U^{238} and that dilution by stainless steel and sodium is only in the amount required as structure and to provide the necessary cooling. This gives an arrangement known as the internal breeder and implies that some of the generated plu-

Table I. Composition of Dilute Fast Breeder

Primary fissionable material	Pu^{239}
Volume per cent fuel alloy	32
Ratio of Pu^{239} atoms/U^{233} atoms in fuel alloy	0.1
Volume per cent of stainless steel in core	0.14
Ratio of Pu^{239} atoms/stainless steel atoms in core	0.118
Volume per cent of sodium in core	54
Ratio of Pu^{239} atoms/Na atms in core	0.108
Volume per cent of stainless steel in blanket	10
Volume per cent of sodium in blanket	20
Core stainless steel captures/Pu atom fissioned	0.091
Core sodium captures/Pu atom fissioned	0.005
Stainless steel and sodium neutron captures in the blanket are estimated to be 25% of corresponding core captures	0.024

tonium will be made in the core and the remainder in a surrounding U^{238} blanket. It is realized that the diluting material in the core very well could be an inert substance and that all of the generated fissionable material could be made in the blanket.

Choice of values to be substituted in the formula for the breeding ratio is fundamentally based upon an estimate of the neutron spectrum to be expected in a dilute reactor. The example of Okrent, Avery and Hummel[3] is followed. In the absence of specific experimental results, the thermal value for the number of neutrons per fission is accepted.

The most important estimate to make is the value of α. The calculated spectrum for the dilute reactor is approximated by the spectrum in the Experimental Breeder Reactor in the outermost part of the NaK-cooled blanket. At that place, the measured value of $\alpha(Pu^{239})$ is 0.19.

The values of the capture cross sections of the structural materials of the reactor are questions of the greatest importance. Two approaches to evaluating such cross sections have been used. Various isotopes have been activated by fast neutrons and the activation cross sections determined. There is the hazard that such a determination gives too small a value, since the activation cross section may be only a partial cross section for the element. In a second type of experiment, the effect on the reactivity of a reactor or critical assembly of an absorbing element placed at the center has been determined. These experiments are difficult when performed in fast neutron assemblies because scattering, and particularly inelastic scattering, as well as moderation affect the reactivity. In Fig. 1, capture cross sections, determined by these various methods for materials of possible interest to core construction are exhibited. A very sizable spread in the values for any element is observed. In the present analysis, it is assumed that a value of 20 millibarns is a reasonable capture cross section for the ingredients of stainless steel.

The capture cross section of sodium for fast neutrons is very small. All measurements agree on this point, although it would be of great interest to have a well-established value. For the estimate of the loss of neutrons to the sodium of the dilute reactor, a value of 1 millibarn is used.

The leakage of neutrons from the reactor must be kept small if a useful breeding ratio is to be obtained.

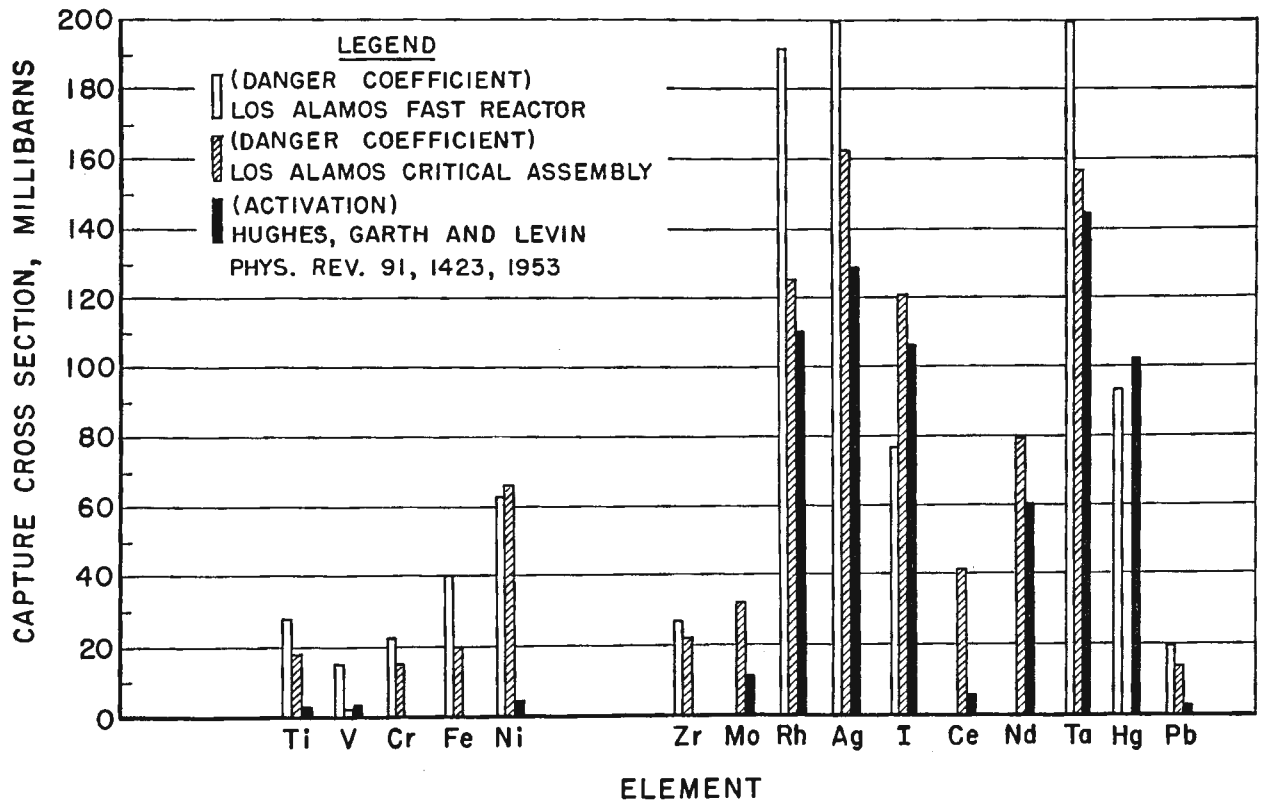

Figure 1. Fast neutron capture cross sections of selected elements

Presumably a sufficiently thick blanket will reduce the loss of leakage neutrons to a negligible quantity. Unfortunately, the necessity for conducting the sodium coolant into and out of the reactor gives the possibility of the loss of neutrons by streaming through the coolant channels. The fact that sodium has a low density and only a modest scattering cross section makes the prevention of escape of neutrons through the coolant channels a difficult matter. It is assumed here that 3 per cent of all neutrons generated escape the reactor.

The fast fission of U^{238} contributes a very sizable bonus of neutrons in the fast reactor. The magnitude of this bonus will depend upon the high energy component of the reactor spectrum and upon how much U^{238} is in the core where the fast component mostly will exist. For the Experimental Breeder Reactor, the number of U^{238} fissions per U^{235} fissioned in the core was found to be equal to 0.174. The Experimental Breeder Reactor, of course, has very little U^{238} at the place in the reactor where the fast component of the spectrum is greatest. In the dilute reactor under consideration, the fast fission bonus will be greater because of the large amount of U^{238} in the core. The factor F in the breeding ratio formula has been estimated to be 0.3.

No specific measurement of the number of neutrons emitted per fission of U^{238} when the incident neutron is fast is available. Arbitrarily, the same value as for Pu^{239} thermal fission is assumed. Table II summarizes the above factors for both a dilute breeder and a converter. The principal differences are that a smaller value of ν must be accepted and that a larger value of α is almost certainly required when the fuel is U^{235}. Parasitic absorption and leakage are assumed not to have changed. The fast fission bonus is smaller because there will be less U^{238} in the core. The breeding ratios found are sufficiently greater than one so that there is little doubt that both the breeder reactor and the converter reactor can be operated with a net gain in fissionable material.

Table II. Values of Factors Determining the Breeding Ratio

	PU^{239} *fuel*	U^{235} *fuel*
ν	2.88	2.46
α	0.19	0.23
A	0.12	0.12
L	0.086	0.074
F	0.3	0.24
ν'	2.88	2.88
B.R.	1.7	1.2

The operation illustrated with a net gain in fissionable fuel does not include losses due to fission product absorption, losses to the control system, or losses of fissionable fuel in chemical processing and in the fabrication of fuel elements. The frequency of carrying out these processing operations becomes a matter of first importance.

RADIATION DAMAGE

It is likely that radiation damage will be the effect which limits the exposure of fuel elements. Corrosion of structural metals in Na is so small that no difficulty is anticipated in holding fuel elements at temperature for any required time. Typical corrosion rates are given in Table III.

Table III. Typical Corrosion Rates for Sodium. Oxygen content less than 0.01 weight per cent. Temperature 500°C

Material	*Weight change, mg/cm²/month*
Iron	−0.2
Stainless steel	0.01
Nickel	0.01
Zirconium	0.2
Molybdenum	Negligible—oxygen sensitive
Thorium	4.4—oxygen sensitive
Uranium	−0.1 — very oxygen sensitive

Loss of reactivity may be a problem, even with internal breeding, but the advantage of obtaining high burnup is so overwhelming in the over-all evaluation of the cycle that it is unlikely that loss of reactivity will be permitted to curtail fuel element exposure. It is assumed that structures will be developed which will permit 2 per cent of the fuel alloy to be fissioned before radiation damage forces the removal of the fuel for reprocessing.

SPECIFIC POWER

In order to assess the effect of cycling of fuel, a goal must be set for the power density in the core. It is postulated that an average power density of 1000 kw/liter can be achieved. This is much greater than the power densities in thermal reactors, but is a necessary requirement if the fast reactor system is to be a source of supply of fissionable material. Power densities for thermal power reactors are in the range of 20 to 70 kw/liter. High power density is a unique feature of the fast reactor system and calls for heat transfer performance not usually encountered in industry.

For the fuel alloy itself, the power density on the average is 1000/0.32=3170 kw/liter. This high value only can be achieved if the fuel alloy is spread out into a large area. This, in turn, means that the physical thickness of the heat producing elements will be small. For the plutonium-fuel reactor, the ratio of plutonium atoms to U^{238} atoms in the alloy is 0.1. The average specific power, therefore, is 1670 kw/kg.

ADDITIONAL LOSSES AND PLANT PERFORMANCE

The frequency of reprocessing of the fuel is determined by the allowable burnup. Accepting that 2 per cent of the atoms of the fuel alloy can be fissioned before radiation damage forces removal, the loss in breeding ratio due to fission product absorption can be estimated. In a single fuel cycle, 20 per cent of

the plutonium is fissioned. During the cycle, the fission product concentration corresponds to the conversion to fission products of 10 per cent of the plutonium. For each plutonium atom fissioned, two fission product atoms are created. Thus (Number of fission product atoms/Number of plutonium atoms)=0.2. Inspection of Fig. 1 indicates that an absorption cross section for fission products of 0.2 barn may be reasonable, and the reduction in breeding ratio due to fission product absorption becomes 0.018.

The loss in breeding ratio to the control system cannot be estimated without a detailed design. In the event that control is by the motion of fuel or the motion of the U^{238} blanket, the loss should be negligibly small. If B^{10} is used as an absorbing control, its use should be limited to shutdown.

A specific power of 1670 kw/kg and a 2 per cent burnup lead to a residence time for the fuel in the reactor of 110 days. On the average, the whole core charge will be reprocessed after 110 days' exposure. If it is assumed that in the reprocessing and refabrication of the fuel elements a total loss of 0.5 per cent of the fuel alloy is sustained per cycle, the effect on the breeding ratio can be estimated. The ratio of Pu lost to Pu fissioned per cycle is 0.025. Five cycles are required to fission all of the plutonium, so the fraction of plutonium lost becomes 0.125. The reduction in breeding ratio due to these losses is 0.105. The final breeding ratio for the reactor and reprocessing cycle, including losses to fission products, now becomes 1.58.

The interest rate is the quantity which best measures the breeding performance of a reactor system. It is the extra material generated per unit of fissionable material invested in the plant per unit time. The interest rate is given by

$$\frac{\text{Breeding gain} \times \text{specific power}}{H \times 10^6} \text{ per day}$$

where H is the ratio of fissionable fuel invested in the whole plant to the amount actually in the reactor. With a 110-day operating cycle, it should be possible to maintain H at a value no greater than two.

The interest rate for the dilute breeder thus becomes

$$\frac{0.58 \times 1670}{2 \times 10^6} \text{ per day}$$

or

$$4.84 \times 10^{-4} \text{ per day}$$

The investment of fissionable material in the plant will be doubled in $(1/4.84 \times 10^{-4})$ days or 5.6 years.

HEAT TRANSFER

The feasibility of obtaining the performance indicated is largely dependent upon the feasibility of obtaining the high heat transfer rates required by the high power density.

For illustration, it is assumed that the fuel alloy is in the form of small diameter rods or wires and that these are closely packed to give the volume ratio cited above. Sodium as coolant is pumped over the surface of the rods parallel to their length. The rods have a diameter of 3.6 mm and are clad in stainless steel of 0.2 mm thickness. The coolant channel of an array of rods packed as indicated has an effective diameter of 3.8 mm. It should be clear that the choice of rods as the fuel element shape is not meant to indicate an optimum arrangement. An actual reactor might very well use thin plates or concentric hollow tubes.

The power density in the fuel alloy is 3.17 kw/cm³. The average heat flux for the surface of the fuel alloy is 68 cal/sec/cm². The thermal conductivity of the fuel alloy is a matter of very considerable concern. The physical properties of plutonium alloys are not well determined and particularly the thermal conductivity when up to 2 per cent of the material has been replaced by fission products is not known. A coefficient of thermal conductivity of 0.06 cal/sec/cm °C is assumed. The temperature difference between the center of the fuel rod and the surface is 100°C. The temperature drop in the stainless steel jacket is 30°C.

Fortunately, sodium permits film coefficients which make the high power densities practical. Figure 2 contrasts with the heat transfer coefficients, h, for sodium and water for flow in a tube having the effective diameter of the reactor coolant channel. It was the superiority shown by sodium which led to its choice in 1945 as a material for the study for the cooling of high power density reactors. A great deal of development work in heat transfer since that time has verified the fine performance of sodium in this respect. If a coolant velocity of 10 meters/sec prevails, Fig. 2 indicates an h of 5 cal/sec/cm² °C. Consequently, the film temperature drop is 13.6°C. Total temperature difference from the center of the fuel

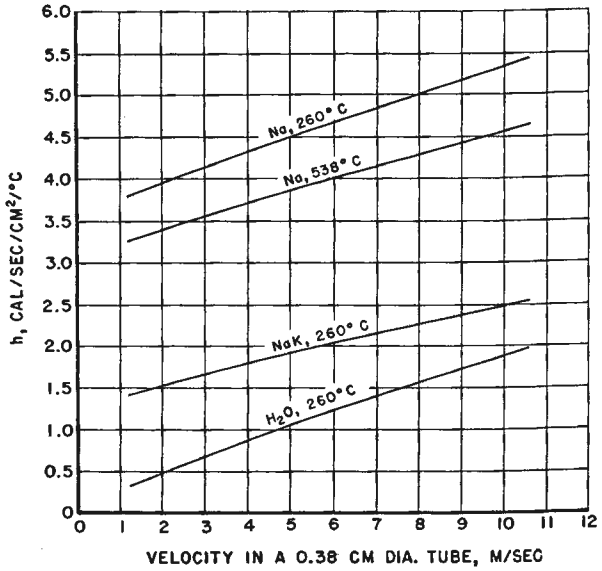

Figure 2. Comparison of heat transfer coefficients for Na and water

rod to the fluid is 143.6°C. This is based on the average power density in the reactor. Actually, the neutron flux will rise appreciably at the center of the reactor unless strenuous efforts are taken in design to make the power generation uniform. Also, allowance must be made for the fact that in any particular coolant channel, the conditions may not be precisely those specified due to variations in dimensions and physical properties. Such "hot channel" factors as well as the inability to flatten the neutron flux completely makes it necessary to allow for a maximum temperature difference which is perhaps 50 per cent higher than the one calculated. What must be accepted as a temperature limitation in the fuel rod depends on the properties of the material of construction of the fuel rod. Where metallic material is used and under the circumstances of high burnup, the wisdom of counting on a central temperature for the fuel which exceeds the alpha-beta transition temperature in the uranium is doubted. These considerations lead to a maximum temperature of the sodium leaving the reactor of about 415°C. This is a temperature for which the corrosion of stainless steel is minor and yet it permits generation of high pressure steam.

While the above analysis illustrates that with the type of subdivision of the fuel alloy indicated and with the sodium as coolant the postulated power density is manageable, it is conceded that to a large extent the practicability of such power densities will be determined by the feasibility of neutron flux flattening. Various schemes for doing this, such as including a blanket portion at the center of the reactor and the variation of fissionable material concentration, have been investigated. If flux flattening is not possible, the power density will have to be reduced perhaps by a factor as large as 2 with a corresponding loss in over-all performance.

REPROCESSING OF FUEL

The fast breeder reactor potentially makes its greatest contribution to the fissionable material economy when plutonium is cycled through it as the fuel. This requires the development of techniques for the removal of accumulated fission products from the fuel alloy and the refabrication of this material into fuel elements.

Decontamination of the fuel can be done by any of a number of processes. Solvent extraction is well demonstrated and the estimates of cost do not prohibit its use. Where the fuel alloy is to be recycled to the reactor, it is, of course, not necessary to separate plutonium from uranium. The fact that the recycled material will contain plutonium and therefore is intensely α radioactive makes the problem of fabricating the fuel elements one of great difficulty. The fabrication techniques must be of the simplest kind in order that they will be adaptable to the protective measures which must be taken when dealing with plutonium. The fact that such protective measures must be taken suggests that it may not be necessary to provide in the decontamination of the recycled material for the very high degree of decontamination customarily given reactor materials. This suggests that, for at least some of the cycles of the reprocessing system, decontamination might be by means other than aqueous solution chemistry. Especially pyrochemical processing, in which the fuel alloy is kept in its metallic state, may bring important savings in cost, mainly because the volume of material to be handled will be very small and, consequently, the shielded equipment also will be small. In any case, the development of fuel elements meeting both the heat transfer requirements of the fast reactor and the requirements of fabrication in the presence of plutonium presents a challenge to the uranium metallurgist and the mechanical engineer. The techniques under development for fuel element construction with α-active material will, of course, have application to heterogeneous thermal reactors, since for these it is inevitable that the recycling of α-active fuel will be demanded. One of the great virtues of the fast reactor is that for the jacket and other structural parts of the fuel element, conventional materials can be used for which there is the greatest amount of well-developed technology. Both in cost and feasibility, the elimination of special reactor structural materials is of great help in attacking the problem of the recycle of plutonium.

CONTROL

An unmoderated reactor will have a short neutron lifetime of about 10^{-7} second. This is much less than the lifetime for any thermal reactor. In contrast to many thermal reactors, however, the excess reactivity which must be made available to a fast reactor in order to permit operation is rather small. The fact that the neutron lifetime is short becomes a matter of consequence only if the area of prompt neutron multiplication is entered. For a reactor in which the total amount of available excess reactivity is small, the probability of accidentally entering the region of prompt multiplication also is small. It has been estimated that a fast neutron breeder reactor could be operated with an amount of available excess reactivity between 1 and 1.5 per cent. This is to be contrasted with water-moderated thermal reactors where 9 to 10 per cent excess reactivity is generally considered necessary in order to manage the reactor. The small amount of excess reactivity required by the fast reactor is due to the fact that xenon poisoning is not expected to be observable. Temperature coefficients, while negative, are small and the effect of burnup where the internal breeding arrangement is used also is not large. To provide inherent limitations on power excursions, the fast reactor will respond in a sensitive manner to the thermal expansion of the fuel and, where the core is diluted with a large amount of U^{238}, will have the advantage of increased non-fission capture of neutrons due to the Doppler

effect. These reactors lose reactivity upon lowering the density of the coolant and, therefore, are afforded a certain amount of protection by the boiling of the sodium. This, of course, would not be effective in the event of a short period excursion.

FAST AND THERMAL REACTOR SYSTEMS COMBINED

Thermal reactor systems fueled with natural or slightly enriched uranium will require additions of fissionable fuel because the conversion ratio probably is less than 1. In the first instance, this will be done with U^{235} which is separated from freshly mined uranium. In the further development of the atomic energy power industry, the recycling of plutonium to reactors will be accomplished. It then would appear better to supply plutonium makeup for the thermal reactor system from a fast reactor system with a net production of plutonium rather than to separate U^{235} from newly mined uranium. Economic analyses based on the fuel value only of plutonium indicate that with a fairly high interest rate the income to the fast reactor system of the sale of plutonium is a sizable item. It would appear to be adequate to pay for the added expense involved in making fuel elements of very extended cooling surface. By making use of fast breeder reactors in this manner, the thermal reactors and the breeder reactors operating together will be able to consume a sizable fraction of the mined U^{238}. This situation, of course, will not come about unless power production by fast and thermal systems are in balance. It appears that the development of thermal reactors for power production is proceeding on a scale considerably more rapid than that of the admittedly more difficult fast reactor development. This indicates that there will be a considerable time during which the usage of U^{238} will be very poor. Thermal reactors with long burnup, however, can be a source of plutonium for the fast reactor. A number of thermal converters are under design whose fuel elements call for a type of construction which is not likely to be accomplished with alpha-active material. The plutonium remaining in the discarded fuel elements from such reactors should be a convenient and relatively inexpensive source of the hundreds of kilograms of plutonium required by the fast reactor.

It is suggested that the reactor discussed here must be only one of several approaches which will lead to the sound development of a system which can bring about high usage. The fast reactor system apparently has an adequately high neutron and fissionable material economy and the very demanding heat transfer requirements lie within the capability of the liquid metal coolants. The system requires demonstration on an adequately large scale.

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3. Okrent, D., Avery, R, and Hummel, H. H., P/609, *A Survey of the Theoretical and Experimental Aspects of Fast Reactor Physics,* Vol. 5, Session 22A, these Proceedings.

Record of Proceedings of Session 12A

SATURDAY AFTERNOON, 13 AUGUST 1955

Chairman: Mr. D. I. Blokhintsev (USSR)
Vice Chairman: Mr. L. de Heem (Belgium)
Scientific Secretaries: Messrs. N. A. Dobrotine and F. de Hoffmann

PROGRAMME

P/495	Heavy water reactors for industrial power including boiling reactors	H. P. Iskenderian *et al.*
P/624	A boiling homogeneous nuclear reactor for power	A. I. Alichanow *et al.*
P/496	Aqueous homogeneous power reactors	R. B. Briggs, J. A. Swartout
	Discussion	
P/335	Complete automation of the operation of nuclear reactors	J. Weill
	Discussion	
P/405	The Dounreay fast reactor project	J. W. Kendall, T. M. Fry
P/814	Review of fast power reactors	W. H. Zinn
	Discussion	

The Chairman: I call upon Mr. Zinn to present his paper "Heavy-Water Reactors for Industrial Power Including Boiling Reactors."

Mr. Zinn (USA) presented paper P/495.

The Chairman: The next paper, "Homogeneous Power-Producing Water Boiler," will be read by Mr. Suvorov.

Mr. Suvorov (USSR) presented paper P/624.

The Chairman: The next paper will be read by Mr. Swartout.

Mr. Swartout (USA) presented paper P/496, "Aqueous Homogeneous Power Reactors."

DISCUSSION OF PAPERS P/495, P/624, P/496

Mr. Hahn (Austria): Mr. Zinn, I wonder if it is possible to shut down the reactor frequently for examination, to take it out of service every evening and to start up again the following morning. If so, are there any disadvantages or difficulties in this procedure?

Mr. Zinn (USA): I would answer Professor Hahn's question in this way: That of course a reactor can be shut down and started frequently—it does not take a great deal of time, say half an hour, to start a reactor—but it is not a good thing to do, especially if you would shut it down so that the temperature and pressure would fall; to do this frequently to a large apparatus is not wise. If you lower the power to where temperature and pressure are maintained, the only concern would be with the fact that in lowering the power the temperature of the fuel elements would drop a little. As some of the exhibits show, this sometimes brings about unfortunate results. For a power reactor you would not frequently shut it down.

Mr. Kruzhilin (USSR): I should like to ask Mr. Zinn to comment on the need to build gas-tight buildings or housings for reactors. These structures are provided for in the majority of the power plant schemes described by representatives of the United States and the United Kingdom. I shall clarify my question. Protection against radioactivity is the first and most important of our problems. Hence, the argument based on possible damage by earthquake which has been put forward here is entirely valid, but it is not clear how far this argument is important for all countries and all regions. A second possible argument based on the danger of rupture of the piping and tanks of the primary circuit seems less valid in view of the present standards of manufacture and inspection of components of this kind. In exactly the same way, ensuring the reliability of the cooling system does not, in my view, present any insurmountable difficulty; for example, the pumping units can be fed separately.

Mr. Zinn (USA): I agree with the statements made by the speaker that equipment with a high performance rating can certainly be built. I refer him, however, to a statement made by Mr. McCullough two days ago in discussing reactor safety in which it was suggested that in the United States, at least, there is a tendency now to substitute a gas-tight hermetically-sealed building for the exclusion

distance which previously had been required. Whether or not this is necessary is a moot question. In the United States quite a number of reactors with power ratings of all ranges have been in operation for many years and no incident in connection with safety has occurred. Nevertheless, it is thought that the expense in providing a gas-tight building is not as severe. At least during the stages when reactors are still of new types and new power ratings it is not considered to be a wrong thing to do to provide this extra protection.

Mr. GOODLET (UK): Mr. Zinn has told us that the working heat flux is about 400,000 BTU's per hour per square foot. Would he like to tell us what the estimated burn-out heat flux is too?

A second question relates to the method of steam separation. Observing that the steam speed through the water surface is about 3½ feet per second, which is rather high according to boiler practice, would he care to comment on the wetness of the water steam mixture leaving the vessel of his 1000 megawatt reactor and also on the means used to dry it before reaching the turbine?

Mr. ZINN (USA): With respect to the first question, I believe that the slide showed—and I checked it after I saw the question—a maximum heat flux of 400,000 BTU's per hour per square foot. I must go by recollection. I believe that for 600 lb and velocities of the kinds mentioned, the burn-out flux is in the neighborhood of 1 million BTU's per square foot per hour so one has a factor of safety of about 2½.

On the matter of steam, if the velocity of the steam was increased as it was in this design, the steam will come out of the vessel carrying more water. It is expected, of course, that steam separators, as they are called, will be introduced between the reactor vessel and the turbine. The turbine itself will still have wet steam in it and must be provided with collection rings—something I do not understand—which makes it possible to have the steam pass down through successive stages of the turbine. I believe the problem has been looked at by competent steam engineers, and they seem to be willing to accept the situation.

The CHAIRMAN: Would Mr. Goodlet please put his question?

Mr. GOODLET (UK): I should like to ask Mr. Suvorov about the steam decontamination equipment. The paper says that the equipment purifies the steam by 10^{10} times and I should like to hear a more detailed explanation of how this apparatus is constructed.

Mr. SUVOROV (USSR): The steam decontamination equipment was tested under laboratory conditions and it was found that a domed column with 10 plates was sufficient to decontaminate steam to the degree indicated in the paper.

Mr. LEWIS (Canada): My question is addressed to Mr. Swartout. I notice that one of the slides which reproduces Table I in the paper estimated an inventory of 90 kilograms of fissile material. Presumably this includes the material in the processing plant. If so, would he confirm that the yield of 60 grams per day excess U^{235} would amount to 90 kg in about 5 years of operation so that the operation could support an annual increase of generating capacity of 20 per cent of the installations made 5 years previously? I bring this up because a somewhat longer time was suggested yesterday.

Mr. SWARTOUT (USA): I will confirm that.

Mr. DANCKWERTS (UK): I should like to ask whether any thought has been given to the possibility of using the heat of recombination of oxygen and hydrogen to superheat the steam from the aqueous homogeneous reactor.

Mr. SWARTOUT (USA): We have given it consideration. It turns out that it is not quite adequate. A little additional superheat would probably be required.

Mr. WEINBERG (USA): The stability considerations in Mr. Suvorov's paper as far as I can understand them are essentially static stability considerations. In any system such as a reactor of this sort, it would seem that perhaps one has to look into the dynamic aspects of the stability. I ask Mr. Suvorov whether he has looked into or considered the dynamic stability of his reactor.

Mr. SUVOROV (USSR): Quite careful consideration has been given to the transition régime from one power to another.

With regard to dynamic stability, this problem is being studied; the results are promising.

Mr. WEINBERG (USA): I am not quite sure what is meant by transition régime. What I have in mind is the fact that there could, of course, be time delays in the effects which result from the energy input into the system. I wanted to know whether these matters have been given some thought.

Mr. WELTON (USA): Essentially my first question is identical with the question asked by Mr. Weinberg. My second question has to do with how your estimates were obtained for the bubble escape times which you would need for estimating the liquid density as a function of power level and pressure.

Mr. SUVOROV (USSR): That was done on laboratory equipment in experiments in which the dependence of the density of the medium on the pressure, on the quantity of steam in the reactor, and on the circulation level was measured.

The CHAIRMAN: Mr. Lane has something to say on the same paper.

Mr. LANE (USA): My question has already been answered. However, I do have a question concerning the written version of Mr. Suvorov's paper with respect to the fissionable material inventory given. I should like to know whether this includes the amount of U^{233} in the blanket of the breeder type reactor he

mentioned; and, if so, what is the concentration of the U^{233} in the blanket.

Mr. SUVOROV (USSR): I did not quite understand your question. Would you please repeat it?

Mr. LANE (USA): On the last page of your report, you mentioned a figure of 3/10 to 7/10 of a kilogram of fissile material as inventory per megawatt of electricity. This seems somewhat low. I just wondered if this included the buildup of fissionable material in the blanket or just merely that in the fuel solution.

Mr. SUVOROV (USSR): Yes, the figure includes all the fissionable material in the system, but I cannot cite the exact quantitative data from memory.

The CHAIRMAN: Then perhaps Mr. Suvorov will be able to give the figure later. I call on Mr. McCullough to speak on the first paper.

Mr. MCCULLOUGH (USA): I should like to take the opportunity to amplify Mr. Zinn's statement about the necessity for pressure containers around reactors. I tried to make it clear the other day that we do feel that we have taken all reasonable precautions in our control system in the inherent design of the stability of our reactors and also in the provision for emergency cooling in case the normal methods of cooling fail. However, we do feel that after you analyze all of these factors you find that there is some finite possibility—I mean only a possibility with a very low probability—that the fission products could escape from the reactor. As Mr. Zinn said, the cost of the pressure container is low enough that we feel that this is worth while. We are most anxious to find ways of designing reactors so that they have sufficient inherent stability and so that no nuclear runaway could happen. In addition to that, we would like to have systems of cooling which are so automatic that the melt-down due to afterheat could not occur. If our Soviet friends feel that they have such designs, we would be most anxious to hear why they do not think a pressure vessel is necessary.

Mr. KAY (UK): I have two questions addressed to Mr. Zinn. Would he tell us what is the method of production of heavy water that he has in mind which would give this figure of $28 per pound? Secondly, would Mr. Zinn care to comment on the problem of thermal distortion of the fuel elements in his proposed 1000-Mw reactor?

Mr. ZINN (USA): The figure of $28 a pound which I mentioned is the one which the United States has announced as its price for heavy water. I am not familiar with the heavy water production business in our country and I do not wish to comment on it.

The second question had to do with a thermal distortion. I assume that the question refers to thermal cycling. There, again, there are some obvious alloys of uranium which inhibit thermal cycling very much, and we would propose to use some of those.

Mr. ERIKSEN (Norway): I should like to ask Dr. Zinn whether the calculations presented in his paper have been tested by exponential experiments and, if so, whether they will be published.

Mr. ZINN (USA): Exponential and critical experiments for heavy-water natural-uranium lattices have been done of a variety of concentrations, and so on. I believe that some of them are on the Conference program in a paper to be given by someone from the North American Company and including work from the Argonne Laboratory.

The second question, as to whether they will be published, is answered by the fact that some of them are on the program. If they are not all there, I hope that in due course the others will be published.

Mr. VENDRYES (France): I should like to ask Mr. Zinn a question about the 1000-Mw natural-uranium, boiling heavy-water reactor. I should like to know: (1) what procedure he has in mind for detecting can bursts; (2) how much time he thinks there would be to detect the burst before its effects became serious; (3) whether the normal activity of the heavy water is not a hindrance in measuring the activity of the fission products.

Mr. ZINN (USA): I have been asked to make the answer short. Where you have all of your fuel elements in one vessel, obviously the detection of ruptured fuel elements is somewhat difficult. Of course, you look for radioactivity in the steam. You do not expect to find it very often because it is necessary to put the fuel elements through rigorous tests before they are put in the reactor. The activity of the steam does not interfere very much if you use detectors which are energy-sensitive, and if you set them to pick up a particular set of lines which are characteristic of fission products or the material which may come out of the fuel element.

The CHAIRMAN: I call on Mr. Weill.

Mr. WEILL (France) presented paper P/335.

DISCUSSION OF PAPER P/335

The CHAIRMAN: We will now proceed to discuss the paper we have just heard; I call on Mr. Harrer.

Mr. HARRER (USA): In our work in the United States on control of a reactor, we usually limit our control to the power range. This range I define as that where the fuel begins to heat the reactor moderator. The range is about 100—that is, about 1 per cent to 100 per cent. That is a wider range than Mr. Weill showed in his slides. At the same time, Mr. Weill indicated that a very wide range could be covered. My question is this: When he considers the poor signal that will come from chambers at low levels perhaps and at some intermediate levels, how wide a range, specifically, does he believe to be possible and useful?

Mr. WEILL (France): We believe that with an

arrangement of this kind we can cover a range of four decades without changing the ionization chamber or its sensitivity. At the Saclay pile, however, we did not consider it advisable to do this. It is a heavy-water pile and the number of decades it covers is relatively small, in view of its large residual power due to the large source of photoneutrons.

Mr. HARRER (USA): Mr. Weill has described a system of control in which the neutron power is continuously adjusted by the rod. Our work has progressed on the line of keeping the reactor power within a reasonable band for a power reactor; that is, one which is to deliver power to a power system. For example, we consider plus or minus 5 per cent as a reasonable band. Inside of this band we take advantage of any negative temperature effect on reactivity to assist in close regulation; we control the reactor with rods only when the power is outside of the designated band. Has the author any comments on the use of temperature in reactor control in this way?

Mr. WEILL (France): The temperature effect did not enter into this particular work. I just mentioned in closing that this temperature effect had been and was at present being studied in certain experiments, the results of which show that control by temperature effect is possible; this control, however, is confined to power reactors.

The Saclay reactor was a research reactor.

I do not know whether I have answered the question properly.

Mr. FRY (UK) presented paper P/405.

The CHAIRMAN: We will now hear the next paper. I call on Mr. Zinn.

Mr. ZINN (USA) presented paper P/814.

DISCUSSION OF PAPERS P/405, P/814

The CHAIRMAN: I give the floor to Mr. Kozachkovsky for a question on Mr. Zinn's paper.

Mr. KOZACHKOVSKY (USSR): Owing perhaps to bad interpretation, I did not understand whether there was any experimental basis for the conclusions referred to in the paper. Does Mr. Zinn not consider it possible to achieve in practice a burn-up of over 2 per cent, say 5 per cent or perhaps even 10 per cent?

Mr. ZINN (USA): I said 2 per cent. That is 2 per cent of all of the atoms of the fuel alloy. That means 20 per cent of the fissionable material. Of course, we realize the great advantage of raising this number. I did qualify my remarks by saying that I hoped that my estimates were conservative and, as we develop fuel elements for this type of reactor, I hope that this number can go to the more optimistic numbers the speaker just mentioned.

Mr. SHEPHERD (UK): I should like to ask Mr. Zinn if the value of ν for U^{238} was an experimental value and, secondly, does he consider it fair to use $1 + \alpha$ in the denominator in his equation since Pu^{240} will have fairly favorable fission properties in a fast reactor? Finally, I think the assumed value of F is rather low. The experimental value we got at ZEPHYR was 0.4.

Mr. ZINN (USA): The value of 0.4 for F of ZEPHYR is perhaps understandable since it is an undiluted or pretty well undiluted core, as I understand it, and there would be a greater component of fast neutrons. Of course, I would be willing to accept our number as conservative. On the matter of the value of ν for U^{238}, I used the same value there that one would find for the other fissionable isotopes for thermal neutrons since I have no measurement for fast neutrons of this number.

Session 13A

POWER REACTORS, PROTOTYPES

LIST OF PAPERS

Description of the Pressurized Water Reactor (PWR) Power Plant at Shippingport, Pa.

PART A. "NUCLEAR POWER GENERATION," Prepared by J. W. Simpson,* and M. Shaw,† USA

PART B. "TURBINE GENERATOR PLANT," Prepared by R. B. Donworth, and W. J. Lyman‡

PART C. "CORE DESIGN," Prepared by I. H. Mandil,† and N. J. Palladino*

The Shippingport Power Plant is sponsored by the United States Atomic Energy Commission to further the peacetime use of atomic energy in the field of electric power generation. That a plant to produce electric power from atomic fuel is technologically feasible is, of course, an established fact. However, much knowledge must be gained before the combined technical, practical, and economic aspects of such a plant can be properly evaluated. The design, construction, and operation of this full-scale nuclear-power plant will provide much knowledge in the fields of reactor physics, fuel element and core technology, and component and system design. By actually designing, manufacturing, operating, and maintaining the full-scale components required for such a power plant, future plants of a similar type can be developed and evaluated with greater accuracy. The plant organization will serve as a pattern for the establishment of operation forces for nuclear power stations, and the facility will also provide training for personnel for future plants.

The plant shown in Fig. 1 will be built on a site of over 400 acres located on the Ohio River in Western Pennsylvania, USA, about twenty-five miles airline distance west of Pittsburgh. Pittsburgh is located in a densely populated metropolitan area in which there is much heavy industry. The immediate area surrounding the atomic power plant site is, however, thinly populated and contains only one major industrial plant. An electric utility company is participating with the United States Government and its prime contractor in the development and construction of the atomic power plant. After completion, the utility company will operate and maintain the entire plant as a part of its integrated electric service system.

The reactor plant proper, the PWR, utilizes a pressurized light water-cooled and moderated heterogeneous thermal-type reactor employing a combination of highly enriched and natural uranium fuel assemblies to produce heat sufficient for a net plant capability of at least 60,000 kilowatts of electric power.

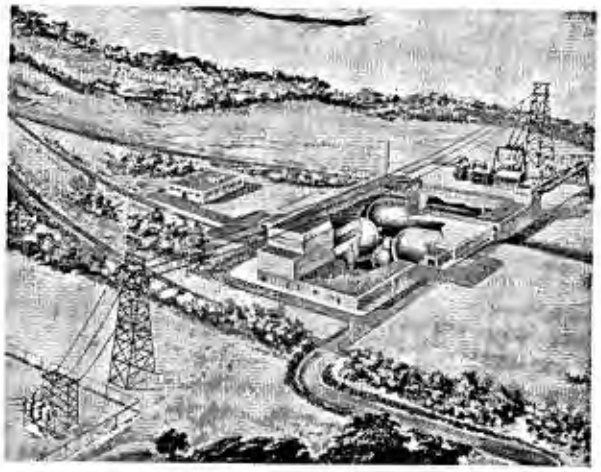

Figure 1. Artist's conception of the Shippingport atomic power plant (PWR)

PART A. NUCLEAR POWER GENERATION

II. General Description

Basic Plant Cycle

The basic schematic diagram of the reactor and steam plant is shown in Fig. 2. The nuclear reactor is the heat source and produces a minimum full power rating of 790×10^6 BTU/hr. This heat is produced in a nuclear core, which is a right circular cylinder consisting of assemblies of enriched uranium in clad plates and natural uranium in tubes. These assemblies are supported in a bottom plate and top grid, which, in turn, are supported from a ledge of the reactor pressure vessel. This reactor vessel is a cylinder with a hemispherical bottom and a removable hemispherical top closure. Coolant enters through four nozzles at the bottom of the reactor vessel and leaves through four nozzles at the top.

The reactor plant consists of a single reactor heat source with four main coolant loops. Three of these loops are required for producing the 60,000-kw minimum design power. Each loop consists of a single-stage centrifugal canned-motor pump, a heat-exchanger section of a steam generator, 16-inch gate-type isolation valves, and the necessary 18-inch outside diameter interconnecting piping.

* Westinghouse Electric Corporation, Atomic Power Division.

† Reactor Development Division, US Atomic Energy Commission.

‡ Duquesne Light Company.

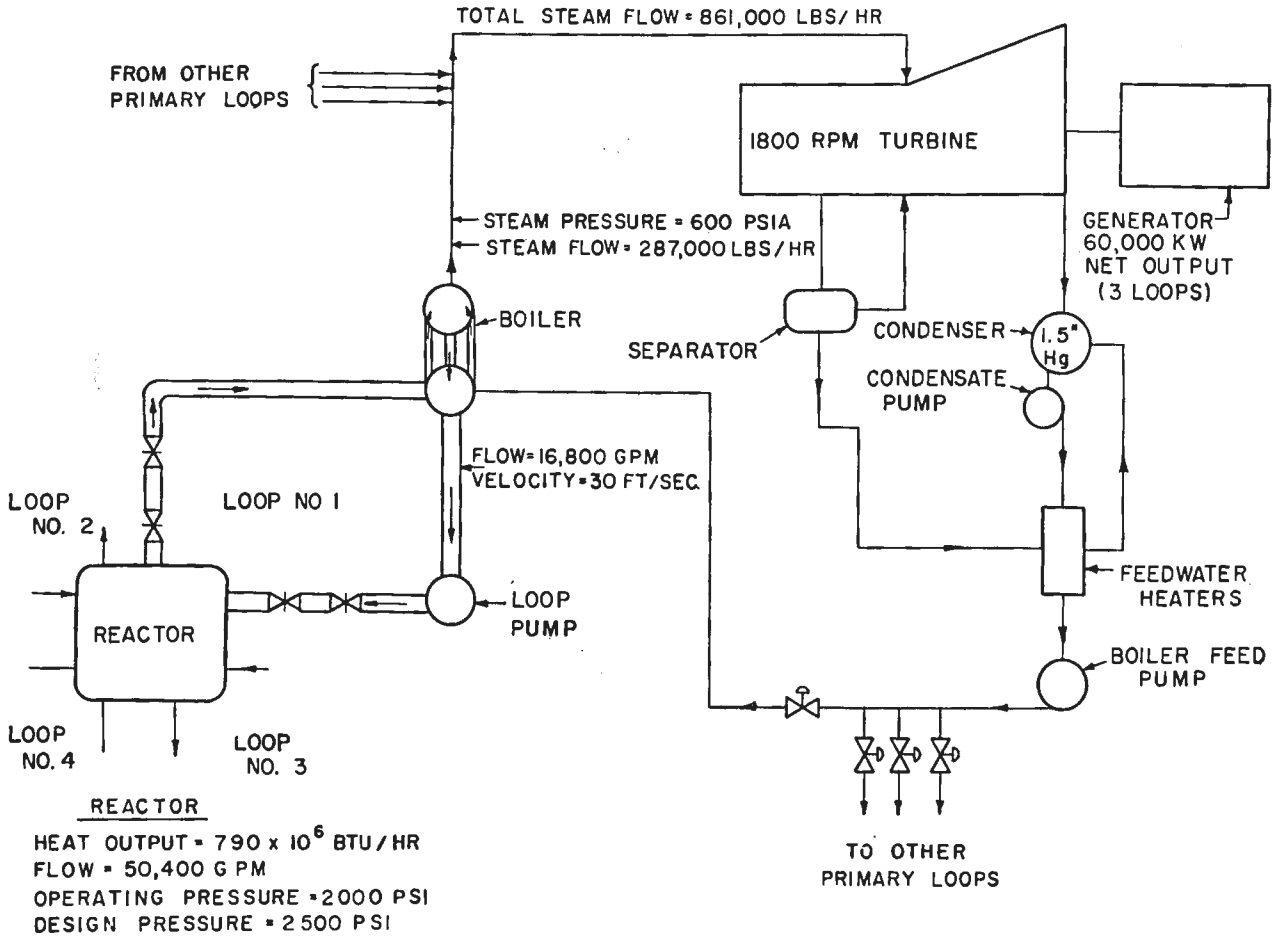

Figure 2. Plant schematic diagram

High purity light water serves as both coolant and moderator in this plant. This water is under a pressure of 2000 psi. The flow through the nuclear core is 45,000 gpm for three loops. At full power the inlet water temperature to the reactor is 508°F and the outlet temperature is 542°F. The water velocity in the 18-inch pipes is approximately 30 ft/sec, with a velocity of between 10 and 20 ft/sec in various parts of the nuclear core. The total pressure drop around the main coolant loop is 105 psi, and this drop is divided roughly equally between core, steam generators, and piping.

The coolant enters the bottom of the reactor vessel where 90% of the water flows upward between the fuel plates and rods, with the remainder bypassing the core in order to adequately cool the walls of the reactor vessel and the thermal shield. After having absorbed heat as it goes through the core, the water leaves the top of the reactor vessel through the outlet nozzles. It then passes through two 16-inch isolation valves in series and goes through the heat-exchanger section of the steam generator. The water then flows through the canned-motor pump and back through the inlet isolation valve to the bottom of the reactor vessel, completing the cycle.

Isolation valves are located immediately adjacent to the reactor inlet and outlet nozzles of each of the four loops. These valves permit isolation of any loop of the reactor plant, to provide maximum protection to the reactor and so that maintenance can be performed while the remainder of the loops are in operation. Adequate shielding is provided to permit this.

The main coolant flows through the inside of many hundreds of small stainless-steel tubes in the heat-exchanger section of the steam generators. These heat-exchanger tubes are surrounded by the water of the secondary system, which is heated by the primary coolant in the tubes. Wet steam is formed, which passes upward through the risers and enters the steam-separator portion of the steam generator. Here the moisture is removed and returned to the heat-exchanger section through the downcomers. The dry and saturated steam at 600 psia at full power leaves the top of the steam separator and goes to the steam turbine.

Considerations in Selection of Major Plant Parameters

The selection of the primary and secondary system temperatures and pressures is a compromise of many conflicting factors. To achieve highest thermal efficiency, steam-system pressure and temperature should be as high as possible. This gives the most efficient utilization of nuclear fuel and also, in the pressure range under consideration, the lowest cost for steam-plant equipment per kilowatt. In general, the cost of

the primary system components increases with pressure, and 2500 psi represents about the highest practical pressure with present-day technology. Above this pressure the cost of components rises quite rapidly. The nominal operating pressure consistent with a 2500-psi design pressure is 2000 psi, for which the saturation temperature is 636°F. The maximum metal-surface temperature for the fuel selected is, therefore, a few degrees below this value. The cost of a core usually increases with increase in metal-surface temperature.

The primary coolant temperature and flow are determined by a proper balance between such factors as cost of core as influenced by core surface area, pumping-power costs, and steam-generator costs. Increasing the flow of coolant for a given inlet temperature and power causes a decrease in core cost. The higher the inlet temperature, however, the higher the core cost. Increasing the flow, on the other hand, gives higher pumping and steam-generator costs. Increasing the inlet temperature gives lower steam-generator costs. By a series of design approximations, the most economical over-all design parameters for the given set of conditions was obtained.

In general, cost of equipment can be minimized by making the individual units larger and thus reducing the number required. There is, however, a competing factor in the increased mechanical complexity and fabrication difficulty associated with increased component size. The four loops selected give units such as primary coolant pumps, valves, and piping that are about as large as is practical with present technology.

The use of four loops gives the plant considerable operating flexibility because any one or even two loops can be shut down and even repaired without shutting down the remaining loops. The shielding between loops in the plant container makes this possible from a radiation standpoint.

One advantage of the design chosen is flexibility. This means that at any time, even after operation has begun, the core can be rearranged and the number of control rods changed to take advantage of knowledge gained in the critical experiment program. It also may be desirable to rearrange the fuel subassemblies at various times during the life of the core to compensate for reactivity changes or to equalize fuel burnup.

Great improvements probably can be made in the nuclear cores for this plant. To take advantage of this, the reactor vessel and the primary coolant system have been arranged to accommodate cores considerably larger than the first and with flow and pressure drop requirements covering a broad range. It is expected that the first core will produce substantially more power than the 60,000 kilowatts minimum plant rating. It is also expected that sufficient information will be obtained from the operation and test of the first core so that subsequent cores may produce sufficient heat energy to be compatible with the 100,000-kilowatt rating of the turbine-generator.

Due to lack of basic knowledge on pressurized water reactor plants, a considerable number of rather costly safety features, such as the plant container, have been included. The experience gained in operation of this plant will enable such features to be evaluated and therefore possibly removed from future plants.

III. Major Components

Physical Layout of Plant Components

To restrict the spread of radioactivity in the event of a dual casualty (i.e., rupture of the primary coolant system and subsequent melting of the nuclear core with attendant release of fission products) the nuclear part of the plant is completely inside a steel container, Fig. 3. This container is sized to contain the pressure created by a rupture of the primary coolant system of the most adverse size, including the effect of the stored energy in the water and the metal as well as any conceivable energy release due to a zirconium-water reaction.

The plant container is divided into four units connected by large tubular ducts. The reactor vessel is located in the spherical section shown in Fig. 4, and two of the main coolant loops are in each of the adjacent cylindrical sections. The fourth cylindrical section is connected into the other three and contains the pressurizing and other auxiliary systems.

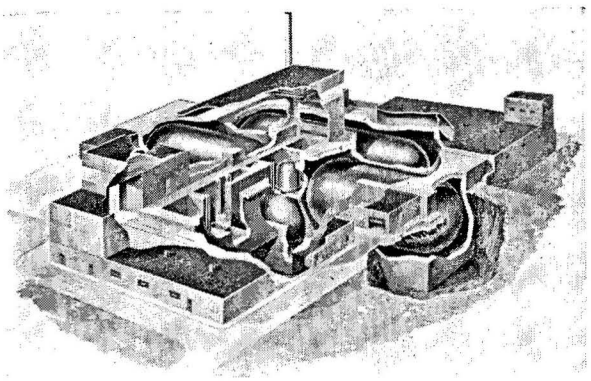

Figure 3. Cut-away view of plant container

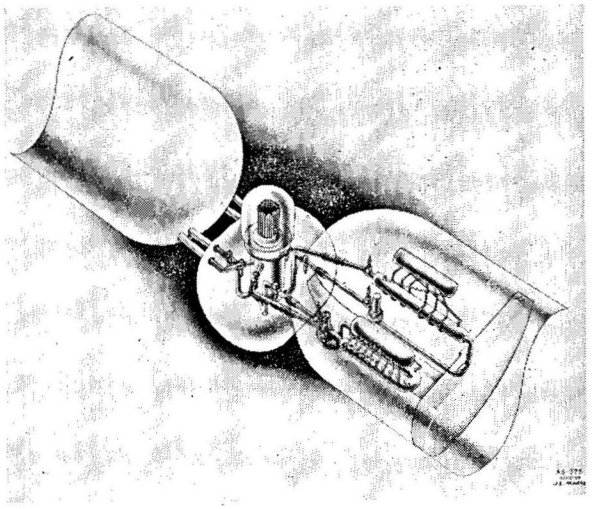

Figure 4. View of plant container showing location of equipment in two loops

The reactor pressure vessel is supported at a point just above the outlet nozzle level by the primary shield tank to permit radial thermal expansion of the four main coolant loops. The steam generators are supported by conventional constant-load spring hangers and the main coolant pumps are keyed into a sliding-base-type foundation. Piping is thus free to expand in any direction.

Reactor Vessel

The reactor vessel shown in Fig. 5 has an overall height of 33 feet, with a cylindrical section with an internal diameter of about 9 feet and a nominal wall thickness of 8½ inches. The vessel has a bottom hemispherical head and a standard bolted and flanged top closure modified to use a gasket or a seal weld for the final sealing. The total estimated dry weight of the reactor vessel is 250 tons.

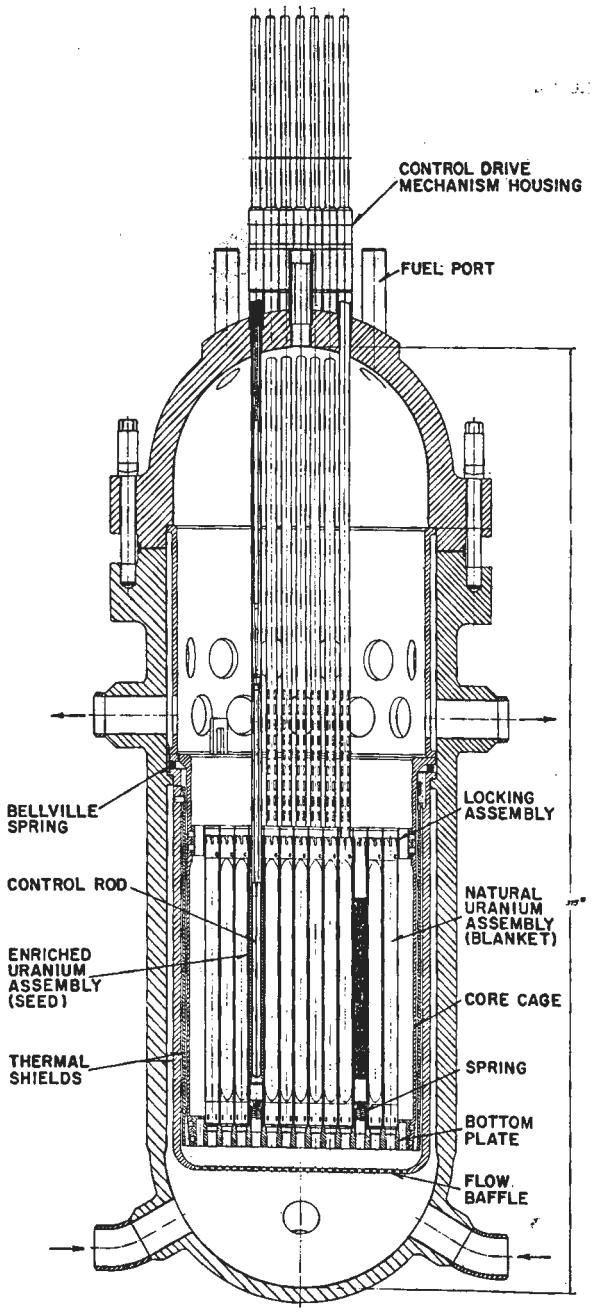

Figure 5. Longitudinal section of reactor vessel

The vessel is formed of carbon steel plates and forgings with a 0.25-inch stainless-steel cladding. The flanges for the closure will be 9 feet internal diameter and 23 inches square in cross-section, with each flange weighing approximately 75,000 pounds. The hemispherical closure head will be forged and have a final thickness of approximately 10 inches, with adequate reinforcement for the 24 control-rod penetrations and the 9 fuel-port tubes. The closure head will be held to the vessel by forty-two 6-inch diameter studs. The bottom head will be 6 inches thick.

The cylindrical portion of the vessel is made in two courses; each course consists of two semi-circular plates welded together. The cylindrical body of the vessel will require three circumferential welds and two longitudinal welds.

The four outlet nozzles are located 60 and 120 degrees apart, just below the vessel flange; the four inlet nozzles are located 90 degrees apart in the bottom head. The nozzles are fabricated from 6-inch clad plate rolled into a cylinder.

The vessel is supported by the neutron shield tank at a point just below the vessel flange. This permits the vessel to grow due to thermal expansion with a minimum increase in main-loop piping stresses. The entire vessel is thermally insulated with approximately 4 inches of glass wool insulation packed to a density of approximately 9 pounds per cubic foot.

The core assemblies are supported in a stainless-steel cage consisting of a barrel, a bottom plate, and a support grid. The cage is nearly 8 feet in outside diameter and slightly over 13 feet high. It is supported in the reactor vessel by a flange just below the outlet ends of the assemblies. The top support grid has slots for latching the assemblies in place. The core cage is clamped in place by a barrel extending from beneath the head to the top flange of the cage. A Belleville spring below the cage flange takes up machining tolerances and differential expansion during operation.

Main Coolant Pumps

A pump, Fig. 6, in each loop circulates coolant. The pump utilizes a single-winding motor of 1200-kw capacity, which can be connected for two speeds. Power supply is 2300 volt, 60 cycles per second, 3 phase for both speeds. The motor, weighing about 20,000 pounds, is mounted in the cast stainless-steel volute with a seal-welded joint. The single-inlet centrifugal impeller runs in a double cutoff single-outlet volute. The motor body is made from forged, or rolled and welded stainless steel. Two types of bearings are being developed. One consists of carbon graphite bearings on hardened stainless-steel journals and thrust runner. The thrust bearing is a pivoted shoe Kingsbury type, and the other is the hydrodynamic type.

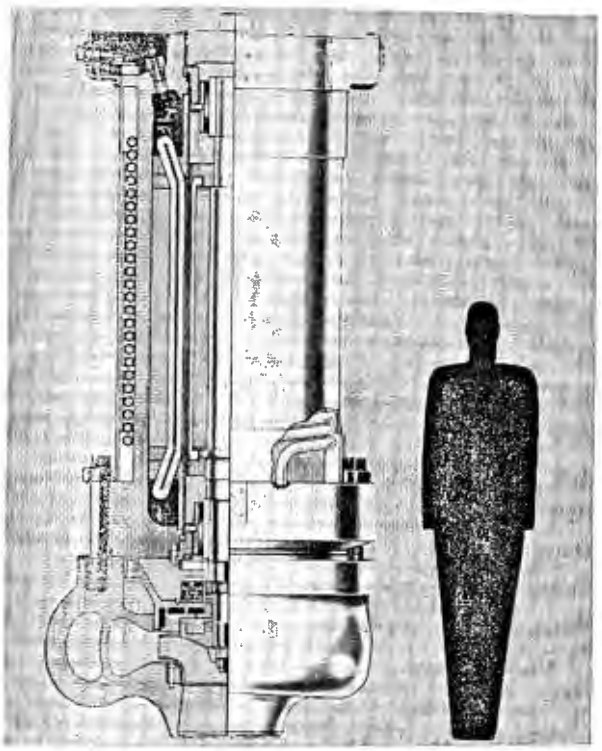

Figure 6. Main coolant pump

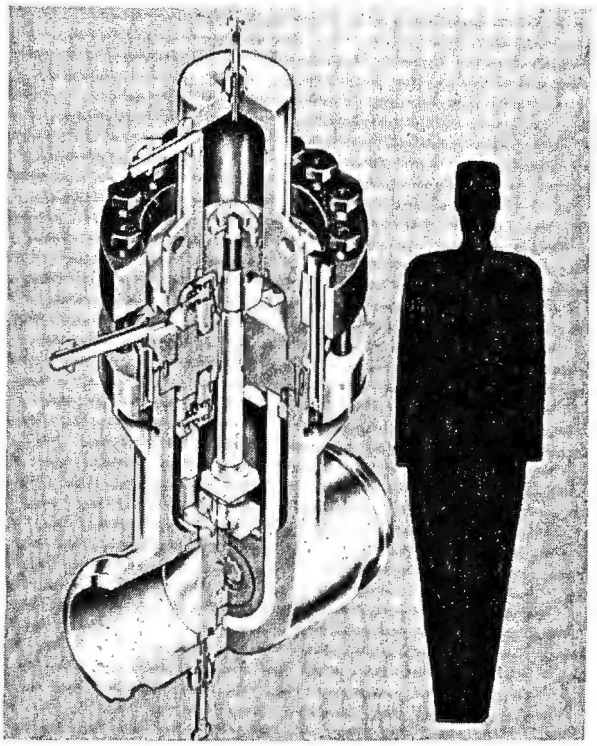

Figure 7. Main stop valve

Both rotor and stator are laminated. Rotor and stator cans are of Inconel and are 0.020-inch thick. Internal cooling and lubricating water is circulated by the thrust runner, which is designed to serve as an auxiliary impeller. Pump capacity may vary with impeller design as required by the type of reactor core used. Two alternate design conditions have been established: The first is 10,000 gpm at 145 psi head with water at 501°F, and the second; 16,700 gpm at 105 psi head with water at 508°F. Suction pressure varies from 1850 to 2150 psi, and design pressure is 2500 psi.

Main Stop Valves

Two hydraulically operated main stop valves are required in each loop, or eight for the plant. One of these valves, shown in Fig. 7, is a parallel-disc gate valve, hermetically sealed and provided with an integral cylinder and piston type of operation. Latches hold it in the closed position. The nominal size of the valve opening is 16 inches, with ports tapered to an 18-inch pipe size.

The valve is designed for normal operation at differential pressures of 600 to 1000 psi across the piston, but can withstand stresses resulting from emergency application of a 3000 psi differential pressure. This valve will shut off against a maximum differential gate pressure of 600 psi, although normal shutoff pressure is a maximum of 210 psia. Stem back seats are provided in the open and closed positions. Position indicators show when the valve is fully open or fully closed.

Eight manual valves serve as backups for the hydraulically-operated stop valves. This is a parallel-disc gate valve with a packed stem seal. These valves normally will be open. A cap is installed over the stem and seal-welded to the body; when the cap is removed, a motor-driven operator can be attached to operate the valves. These valves are closed during maintenance or repair operations on a primary loop.

Steam Generators

The steam generators are each rated at 263×10^6 BTU/hr and provide 600 psia full-power steam pressure; this pressure rises to 885 psia at no load. The secondary-side design pressure is 975 psia. The primary-side operating pressure is 2000 psi, with a design pressure of 2500 psi. Six and a half million pounds, or approximately one million gallons per hour of primary cooling water passes through the tubes, entering at a temperature of 542°F and leaving at 508°F. The allowable pressure drop through the boiler is 35 psi and the feedwater temperature is 315°F. Two of these units, shown in Fig. 8, are entirely of stainless-steel construction and are of a straight tube, fixed-tube sheet design. They are 36 feet long and 43 inches in diameter. There are 2096 tubes, each ½-inch outside diameter by 0.042-inch wall thickness on a ¾-inch triangular pitch to provide a total of 8400 square feet of heat-transfer surface. Steam separators are constructed of carbon steel and are of conventional design. The drums are 26½ feet long and 5 feet in diameter, with twelve 8-inch risers and six 8-inch downcomers. The other two units, shown in Fig. 9, are of a return bend or U-tube type. The overall length is 28 feet, and the diameter is 39 inches. They are fabricated from carbon steel with stainless-steel cladding in areas where

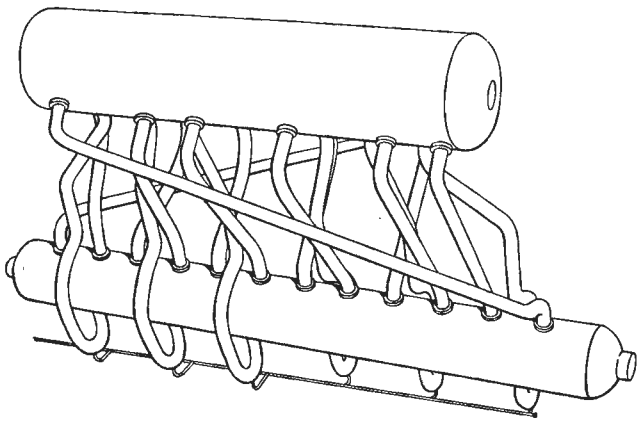

Figure 8. Steam generator with straight tube heat exchanger

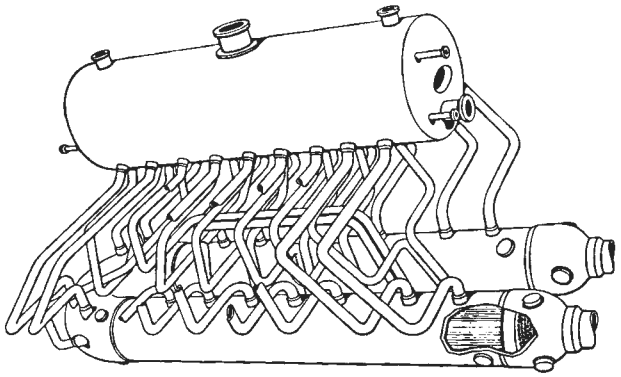

Figure 9. Steam generator with U-tube heat exchanger

contact is made with the primary coolant. Tubing is made of stainless steel, there being nine hundred and twenty-one ¾-inch outside diameter, and 0.060-inch wall tubes on 1-inch triangular pitch, providing 9050 square feet of surface. The carbon-steel separator utilized a 5-foot diameter drum, 23½ feet long, with eighteen 5-inch risers and fourteen 4-inch downcomers. Steam will be dry and saturated with less than ¼% moisture.

Steam generators of the two different types described above are used so that experience can be gained in the design, manufacture, and operation of both types to give more knowledge on which to base the design of future plants.

Pressurizing Tank

A cutaway perspective of the pressurizing tank is shown in Fig. 10. The tank is 18 feet high and 5 feet in inside diameter, with a total volume of 300 cubic feet, of which about 150 cubic feet is the steam dome volume with maximum water level. Normal surges will be as great as plus or minus 10 cubic feet under design plant-load fluctuations. Under these design conditions primary system pressures will be held within limits of 1850 to 2185 psia. A 6-inch surge line in the bottom head connects this tank with the primary coolant system.

The heater section contains 200 heater wells, into which are inserted 500 kilowatts of replaceable electric heaters. These are arranged electrically into three groups for operational convenience and are controlled to maintain a saturation temperature of 636°F (corresponding to a pressure of 2000 psia), and to form a steam bubble in the top of the tank. A spray nozzle at the top sprays colder (500°F) water into the steam during positive surges to help limit maximum pressures. Located in the center of the tank is a standpipe for use as a reference leg for water-level measurement. A low-level alarm warns the operator to add water to the system before the level drops to a point where negative surges could uncover heater wells.

IV. Core Design

Mechanical Design

The active portion of the nuclear core is in the form of a right circular cylinder 6 feet in mean diameter and 6 feet high. To minimize the amount of enriched U^{235} used, the core consists of some highly enriched (seed) and some natural blanket assemblies. The highly enriched, or seed assemblies are located in a square annular region about 6 inches thick. The area inside and outside of the annulus is filled with natural uranium oxide subassemblies. A cross-section of the core is shown in Fig. 11.

Each seed subassembly consists of several plates welded together to form a box. Four of these box-type units are welded together, with separation maintained by spacers, to form a central cruciform-shaped area. In each of these areas is located a cruciform-shaped hafnium control rod.

The uranium oxide, or blanket assemblies use a rod as the basic element, see Fig. 12. These rods are zircaloy-2 tubing with a 0.413-inch outside diameter and a 0.028-inch wall. The tubes are filled with UO_2 pellets and have zircaloy-2 end plugs welded to each end to form fuel rods. These rods are assembled into bundles of 100 rods each; the assembly is made by mechanically fastening together a stack of seven bundles, each about 10 inches long, for a total of about 6 feet. The individual rods are welded to plates at each end. Holes are drilled in the plates between the rods to permit the flow of coolant water.

The seed and blanket assemblies have the same cross-section shape and are each fitted with transition sections at the bottom and top, which terminate in a circular tube area. These circular ends fit into holes drilled in a support plate at the bottom. A special locking collar is attached to each of the assemblies at the top, and projections from this collar are used to lock the assemblies securely into the top grid support. Springs at the bottom of each assembly insure a tight mechanical fit and thus prevent any vibration or loosening of the assemblies under variations in operating temperature.

The bottom of the blanket assemblies are provided with the necessary orifices to properly distribute the coolant flow. In addition, a perforated flow baffle just below this plate insures uniform distribution of coolant and adequate mixing of the water that enters the vessel through the inlet nozzles.

Provisions are made for removing any individual seed or blanket assembly. To accomplish this, the cap

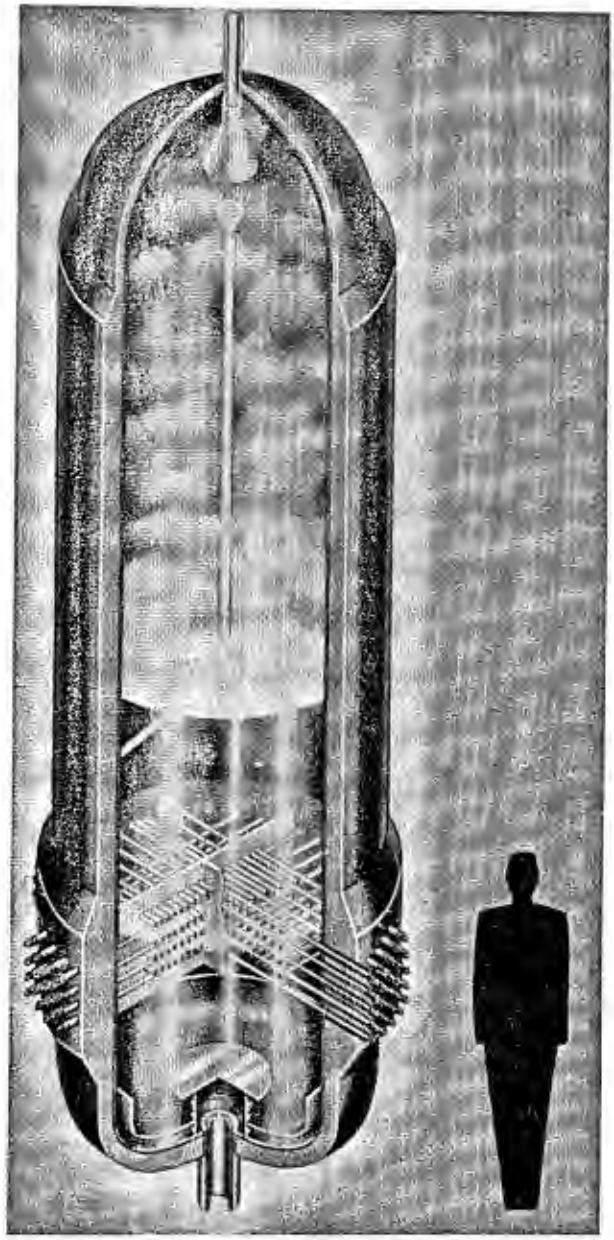

Figure 10. Cut-away view of pressurizing tank

on the fuel port is removed and an assembly removal mechanism is inserted.

The removal head can then be rotated as desired and indexed for the removal of any predetermined assembly. The first step is to unlock the assembly. After this is accomplished, the assembly is remotely secured to the head of the removal mechanism and withdrawn from the core. The area above the pressure-vessel head is flooded with water during the removal operation, and the assembly is transferred to the fuel canal with this water acting as shielding. Provisions are also available for removing the reactor vessel head and then removing the entire core as a unit, or any assemblies as desired.

A major problem in the operation of a nuclear-power plant is the difficulty of determining exactly what is taking place within the reactor vessel with respect to coolant flow, boiling of the coolant, temperature of the fuel elements, or failure of these elements. Suitable instruments for making these determinations are therefore provided.

Physics Design

The seed assemblies contain a total of 52 kilograms of enriched U^{235}, and the blanket contains twelve tons of uranium in the form of UO_2. At the start of reactor life approximately 60 per cent of the power will be produced in blanket assemblies and 40 per cent in the seed assemblies. The power distribution is a function of reactor life, inasmuch as the reactivity of both the seed and blanket vary with reactor life. The percentage of power produced in the blanket is expected to increase with time. Obviously, power distribution is also a function of the reactor temperature and the position of the reactor control rods.

In the PWR reactor most of the power comes from the fissioning of U^{235}. However, approximately 8 per cent comes from the fast fissioning of U^{238}, and a substantial fraction comes from the fissioning of plutonium. This latter effect increases with the lifetime of the reactor and is expected to reach a substantial fraction after 3000 Mw days/ton average burnup in the blanket assemblies. At the start of life, approximately 0.8 atoms of plutonium are formed for each atom of U^{235} that is fissioned.

Typical radial and axial flux distributions are shown in Figs. 13 and 14. In the seed the radial peak to average ratio is 1.15 to 1, and the axial peak to average ratio is 1.9 to 1. In the blanket the radial peak to average ratio is 1.85 to 1, and the axial peak to average ratio is 2.1 to 1. The hot spot occurs in a radial direction at the inner edge of the seed assemblies and in the axial direction at a point considerably below the center. Note also that large flux gradients occur at the interfaces between the seed and blanket sections of the core. Even such a small volume of water in the core as occupies the control rod holes when they are withdrawn causes peaking of the thermal flux. Care must be taken to reduce these peaks, if they occur at high points in the flux distribution, to prevent accentuating thermal hot spots in the core. During the operation of the reactor when the transient xenon is a maximum, the flux distribution may be more unfavorable.

The PWR reactor has a negative temperature coefficient of $2 \times 10^{-4}\ \Delta k$ /degree F, or possibly higher. This temperature coefficient is of sufficient magnitude to materially reduce the seriousness of the control problem. The reactivity will vary throughout the life of the reactor core, tending to increase during the early part of the reactor life and then decrease. The highly enriched assemblies must be replaced at least once to get maximum life out of the uranium-oxide blanket assemblies.

Obviously, sufficient control rod area must be provided to permit the reactor to be subcritical with a reasonable margin in its most reactive condition. Inasmuch as each additional control rod imposes a severe penalty in the mechanical design, it has been nec-

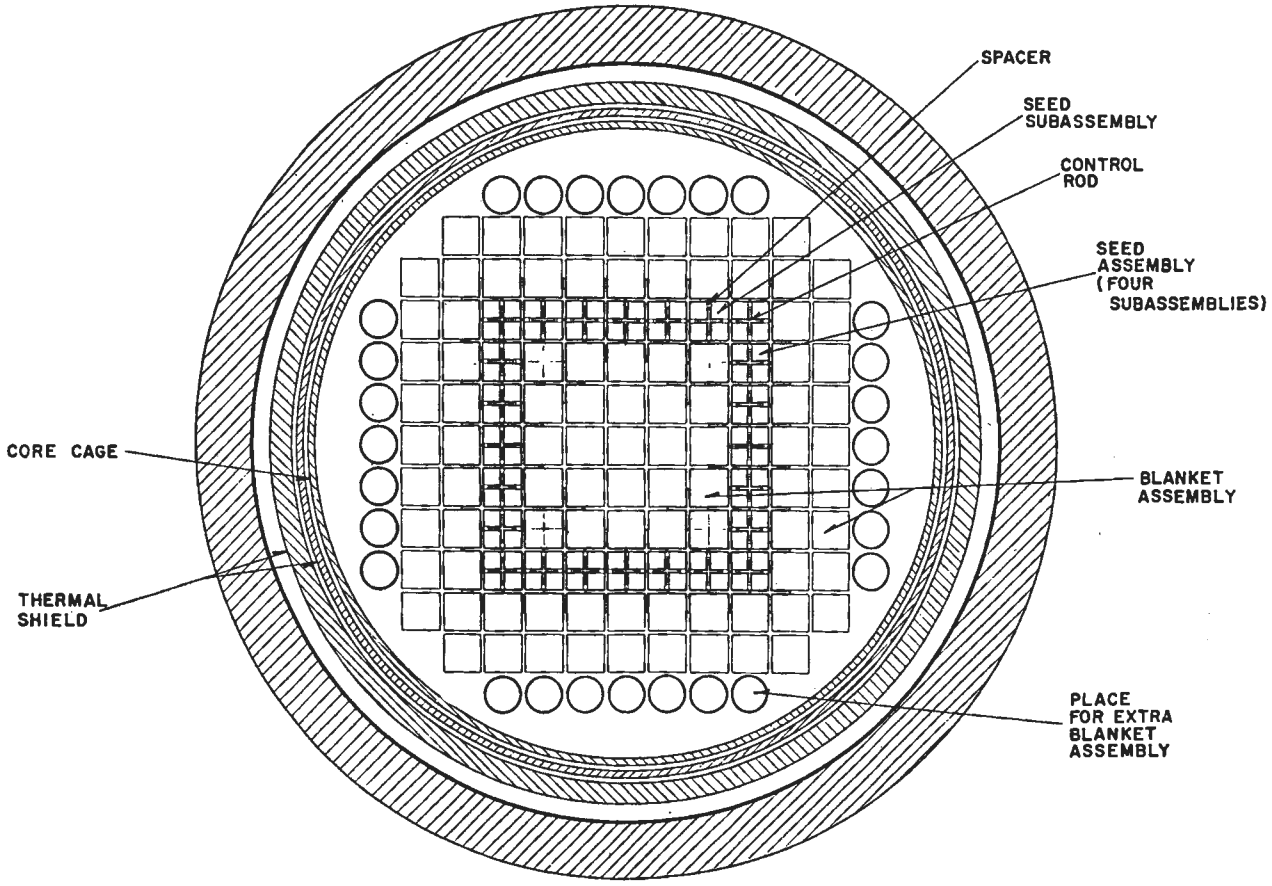

Figure 11. Cross-section view of reactor vessel and core

essary to keep the number of rods to an absolute minimum. To increase the loading of the highly enriched assemblies and thus give them increased life, burnable poisons have been added to these fuel plates. This poison is removed by capture of neutrons during the life of the reactor. Sufficient control must be provided to compensate for the equilibrium xenon poisoning. The control drive mechanism functional requirements include the following: (*a*) number of rods—24, with provisions for 4 more; (*b*) rod speeds—20 inches per minute; (*c*) minimum scram rate—200 inches per minute; (*d*) permissible delay after scram signal—0.1 second; (*e*) fail safe on loss of power; and (*f*) accuracy of position indicator—1.5 inches.

Thermal and Hydraulic Design

The basic criteria for the thermal design of the PWR are as follows:

a. Nucleate boiling must not occur at the hot spots during steady state operation.

b. Bulk boiling must not occur in water leaving the hot channel during loss of coolant flow transient.

c. Fuel-element internal and surface temperature must be compatible with the properties of the selected fuel element material.

To prevent nucleate boiling, the maximum coolant temperatures must be below 636°F, which is the saturation temperature for the 2000 psi operating pressure.

To calculate the maximum metal-surface temperature for a given assumed set of parameters, the core hot channel factors must be known. These factors represent such items as the peak-to-average flux ratio and the maximum permissible deviation from the average with respect to coolant channel width, fuel-element thickness, eccentricity of fuel within an element, and variation in fuel loading within the element.

In the seed assemblies, the peak power density is 277 watts per cubic centimeter, with an average power density of 81 watts per cubic centimeter. In the blanket the peak power density is 120 watts per cubic centimeter, and the average power density is 32 watts per cubic centimeter. In the seed assemblies the maximum heat flux is 382,000 BTU/hr/ft^2, and the average heat flux is 112,000 BTU/hr/ft^2. In the blanket the maximum heat flux is 240,000 BTU/hr/ft^2, and the average is 65,000 BTU/hr/ft^2.

Thirty-three per cent of the flow through the fuel elements of the nuclear core flows through the seed assemblies, which, however, represent only about twenty per cent of the area. The water velocity in the seed assemblies is 20 ft/sec, and that in the blanket assemblies is only 9.8 ft/sec.

V. Materials

General

A great many new and unusual materials are employed in the design of nuclear reactors. The need for their use comes from the requirement that structural

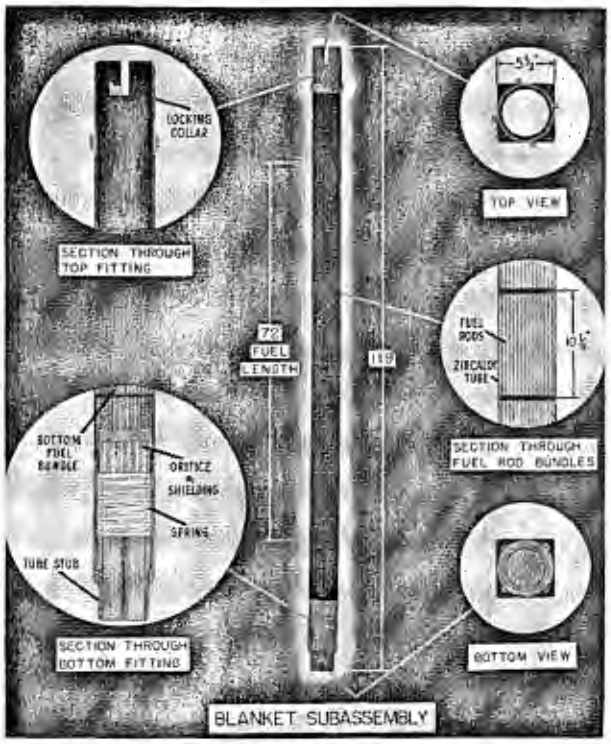

Figure 12. Blanket UO_2 fuel assembly

materials, heat-transfer fluids, shielding materials, and nuclear fuel and moderating materials must have certain nuclear properties. Numerous materials that, in the past, have been rare and little known from the standpoint of engineering properties are now being used in varying degrees.

Water normally is considered a rather non-corrosive agent. However, at high temperatures, particularly when only minimum amounts of corrosion can be tolerated, this is far from true. One of the most difficult problems has been the development of suitable materials and materials combinations for all of the applications involved.

The basic requirement is for a material that will simply withstand the corrosive effect of the primary coolant water for long periods of time and that has sufficient strength for the application and can be welded.

Fuel

The highly enriched assemblies consist of zirconium-uranium alloy plates clad with Zircaloy-2. The zirconium-uranium alloy fuel sections of these plates are relatively corrosion-resistant and are also metallurgically compatible with the Zircaloy-2 cladding. This increases the probability of a good bond between the center section and the clad. No imperfections in clad surfaces are expected. However, if there should be, this would cause only minor corrosion of the center section, would not cause any mechanical deformation of the plates and would release only negligible quantities of fissionable material into the coolant stream. The plate design was chosen to give high heat-transfer area, which was required because of the high power density in the seed assemblies.

For the blanket assemblies, a material that permits the highest possible multiplication constant is desirable. This is because the ratio of power produced in the blanket to power produced in the seed varies approximately as $k/(1-k)$. Pure natural uranium would, therefore, from a nuclear sense be the most desirable material. Such natural uranium is, however, extremely unsatisfactory both from a corrosion and a radiation damage viewpoint. There are two possible solutions. The uranium can be clad sufficiently well that the coolant water never reaches the uranium surface, and with sufficient strength to prevent any dimensional changes due to radiation damage. However, it is highly unlikely that cladding can be sufficiently perfect to never expose the uranium to the coolant water. Also, whether or not radiation damage can be prevented by mechanical restraint is not known. The other possible solution is to add something to the uranium to improve its characteristics, which can be done by making a uranium alloy, such as uranium-molybdenum, uranium-silicon, or uranium-niobium, or by utilizing the uranium in the form of uranium-oxide.

All alloying elements that improve the radiation-damage resistance or the corrosion-resistance properties of uranium alloys tend to give poor neutron economy, inasmuch as the alloying elements have relatively high cross-sections for capture of neutrons. Even if neutron economy is ignored, none of the alloys are suitable from both a corrosion-resistant and radiation-damage viewpoint. This means that either the life is seriously limited by radiation damage or by corrosion, or both. The radiation may cause a rupture of the cladding and subsequent failure of one element due to corrosion, or the corrosion may be initiated by a cladding defect. In either event, after several days of exposure to the hot water all of the uranium alloy and the contained fission products in the affected fuel element might be released to the coolant stream. Uranium alloys are also, in general, more expensive than the uranium-oxide fuel elements and also give less power production in the blanket assemblies than uranium oxide.

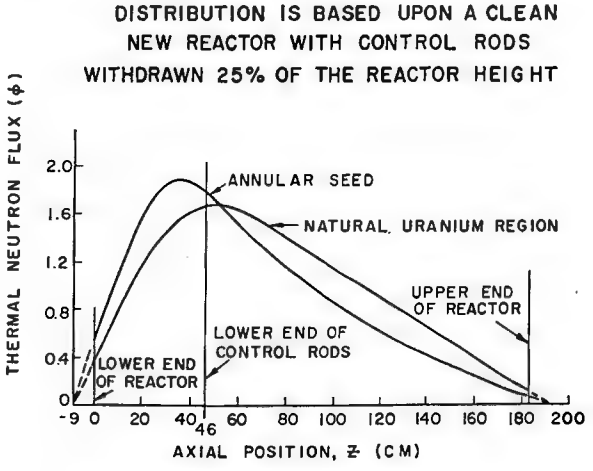

Figure 13. Axial distribution of thermal neutron flux

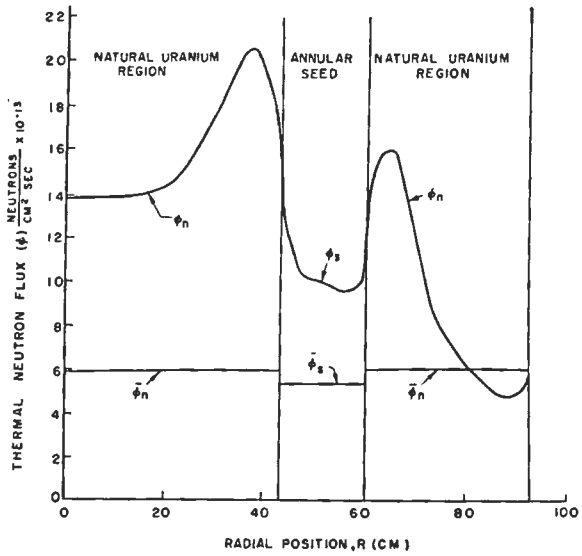

Figure 14. Radial distribution of thermal neutron flux

Uranium oxide is an excellent material for the blanket fuel elements, because it is completely inert in high-temperature water and is also satisfactory from a radiation-damage viewpoint for a relatively high burnup. The uranium oxide can be readily cold compacted into pellet form and then sintered to increase the density to in excess of ninety per cent of theoretical density. By mass-production techniques the dimensions can be ground to extreme accuracy and the uranium oxide loaded into Zircaloy tubing. The center temperatures of the uranium-oxide fuel elements may be as high as 2200°F, but this appears to cause no difficulty because it is still approximately 3000°F below the melting temperature of uranium oxide. The Zircaloy-2 tubing is ideal from a performance viewpoint. Its inherently high cost is a disadvantage; however, this disadvantage is more than compensated for by the decreased cross-section for capture of neutrons of this material.

Control Rods

Control rods can be made of hafnium, boron steel, or stainless-steel-clad cadmium-silver alloy. The PWR control rods are made of crystal-bar hafnium and are homogeneous rods. The hafnium has ideal mechanical properties and is also extremely corrosion-resistant.

Other Materials

For application to components such as mechanisms, valves, and pumps, it is necessary to have materials that not only will withstand the corrosive effects of the high-temperature water but also have good wear characteristics when in rubbing contact with other pieces of metal, without any lubrication other than the high-temperature water itself. Inasmuch as most of the materials that have the best corrosion-resistance have inherently poor wear-resistance and vice versa, a thorough investigation program was required to determine the combination of metals that presented the best compromise for the various applications. First, a large number of tests were made for short periods by means of simple shapes; later, actual full-size components were made and tested under actual service wearing conditions in the high-temperature water.

In addition to the applications mentioned, materials suitable for such uses as springs had to be selected. Springs must be made of a material that will withstand the corrosion of the coolant and whatever radiation is involved at the point of application, and will not lose temper due to prolonged exposure to the high-temperature water. Springs under stress also are subject to stress corrosion.

Another little-known problem faced was that of contact corrosion. Contact corrosion is the tendency of two metals, when in contact or near contact, to form a corrosion film between them when they are together for long periods of time. The resultant corrosion film, in effect, causes a welding of the two pieces, so that motion cannot be accomplished when required.

The basic material for the static parts of the reactor plant, such as heat exchanger vessels, exchanger tubes and pipes is AISI Type 304 (18-8 chrome-nickel steel with very low carbon content). Stressed parts, such as springs, are made of Inconel or Inconel-X. Rubbing parts, such as those in the control drive mechanisms, are made of hardened stainless steel, chrome-plated stainless steel, and the various grades of Stellite. Magnetic parts, such as magnetic slugs, for control rod position indicators are made of martensitic stainless steels.

VI. Primary Auxiliary Systems

A number of auxiliary systems are connected into the primary coolant circuit to insure proper operation of the plant. Some of these, such as the pressurizing system and the purification system, are in use continuously, while others are required for intermittent operation. In addition, there are the usual power plant auxiliary systems such as cooling water, compressed air, and electrical.

Coolant Charging System

The coolant charging system is used for filling the primary plant prior to operation and for maintaining the proper fluid level in the pressurizer. One 100 gpm low-head pump is used for filling and two 25 gpm high-pressure pumps are required for the intermittent makeup function. The charging system also contains facilities for charging fresh resin into the purification demineralizer and back-flushing the primary loops. No purification equipment is provided in this system as the water received from the boiler makeup water equipment meets primary water specifications.

The charging and discharging systems will be utilized for the injection and drain of the decontamination solutions. In the PWR plant the surfaces in contact with reactor coolant can possibly become radioactively contaminated. Primarily this would be due to the deposition of fission products which could result if many defects should occur in the fuel elements, and

to a lesser extent due to the deposition of stainless-steel corrosion products that have been activated by passage through the flux field of the core. If plant surfaces should become sufficiently activated as to unduly restrict access for maintenance then cleaning could be accomplished by the circulation of suitable decontaminants throughout the isolated portions of the plant. Possible reagents being considered for PWR are acids, complexing agents, and oxidizing agents. These include citric acid, periodic acid, versene, polyphosphates, hydrogen peroxide or combinations thereof. The use of oxygenated water has also been investigated and results are encouraging.

Pressurizing System

Changes in the core average temperature due to power excursions and changes in reactivity, and subsequent correction by control-rod movement make the system coolant volume a variable. These volume changes would cause wide variations in coolant-system pressure if the plant were operated as a solid system. The function of the pressurizer is to regulate the system pressure within a lower limit set by the reactor hot-spot temperature, and an upper limit determined by the safety and relief-valve settings. Relief valves are expected to function during abnormal positive surges only. The pressurizing system, Fig. 15, maintains a 2000-psi saturation steam head in a separately heated pressurizing vessel. This vessel has a volume of approximately 300 cubic feet, and contains about 100 cubic feet of water. The size is a function of the total volume of coolant and the transport time of water in loops, as well as the rate of change of power. The heat to the pressurizing vessel is supplied by 500 kw of electric immersion heaters, which are sufficient to raise the temperature of the water in the pressurized tank consistent with a 200°F/hr heating rate for the entire plant. A continuous spray into the top of the pressurizer is provided by the pressure drop across the reactor. This is used to reduce the steam bubble during surges and assist in keeping the corrosion inhibitor in solution during long periods of steady-state operation. The system is designed to accommodate surges associated with power excursions in the range

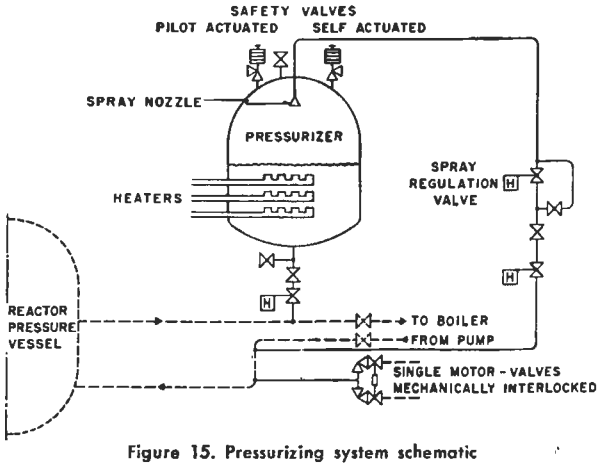

Figure 15. Pressurizing system schematic

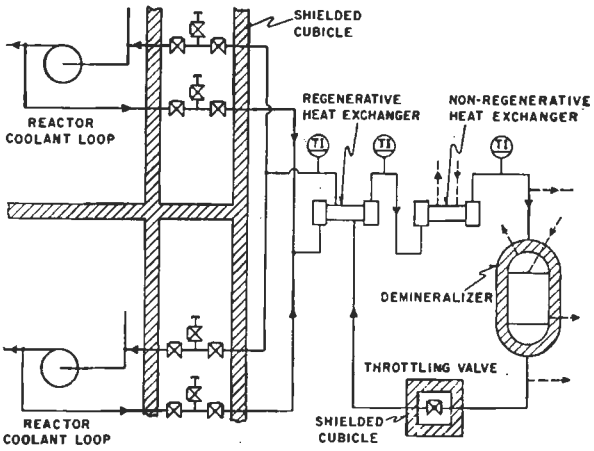

Figure 16. Purification system schematic

of 20,000 kw to full power as follows: (*a*) plus 15,000 kw or minus 12,000 kw as a step change; (*b*) plus or minus 15,000 kw at 3000 kw per second, and (*c*) plus or minus 20,000 kw at 417 kw per second. During these excursions the pressure varies between 1850 psia and 2185 psia. Below the 20,000 kw range the changes in power can be accommodated at a slower rate.

Purification System

A fraction of the coolant that passes through the nuclear core must be purified to limit the activity buildup of the long-lived impurities in the water. This is done by passing a certain quantity of the coolant through a bypass demineralizer, which removes soluble and insoluble matter. The reactor coolant-purification system, see Fig. 16, consists of two parallel loops, each of which provides purification for two of the four reactor coolant loops. The interconnecting lines between two reactor coolant loops and the common purification loop are valved and manifolded. The driving force for the system is provided by the available head of the reactor coolant pump. A controlled flow of 40 gpm per loop passes through the tube side of a regenerative heat exchanger, the tube side of a non-regenerative heat exchanger, a mixed bed demineralizer, the shell side of the regenerative heat exchanger, and back to the coolant loop. The following are the design characteristics of the system: (*a*) corrosion rate of 10 mg/dm^2/month for stainless steel and 100 grams/day for core fuel material, (*b*) flow rate for each demineralizer of 20,000 pounds/hour, and (*c*) resin of the mixed bed type, with a capacity of 12 gram-equivalents/cubic foot, and a temperature limit of 130°F max, with an area loading of 5 to 7 gpm/square foot.

Sampling System

The sampling system removes homogeneous samples from the purification system before and after each of the two demineralizers to provide information on conductivity, pH, impurities, and dissolved gases. The pH and conductivity measurements provide a means for determining the alkalinity or acidity of the primary coolant. These measurements also provide an

indication as to when the demineralizer resins are spent and require replacement. The suspended solids are collected and measured and show the effectiveness of the demineralizers. Samples are withdrawn for dissolved gas analysis on hydrogen, oxygen, and fission gas products.

The effluent from a sample room is discharged into the waste disposal system.

Valve Operating System

The primary plant fluid systems contain a number of hydraulic cylinder operated valves that must be operable during periods when direct access to these valves is impossible. Typical of these are the main-loop isolation valves, which must be capable of rapid closure in the event of a primary water leak in a main loop.

A group of high-pressure air bottles and water flasks located outside of the shielded spaces serve as the source of valve-operating pressure and fluid, Fig. 17. The valve-operating water is directed to the cylinder valves by hydraulic pilot valves inside the shielded spaces, and is vented from the cylinder valves into the primary system. Many of the vital system components are located outside the shield, and thus are readily accessible for maintenance.

Coolant Discharge and Vent Systems

The coolant discharge and vent system, Fig. 18, is designed to collect radioactive waste material from various sections of the primary plant and to convey these materials to the waste-disposal influent tanks. The discharge system contains: a 3600-gallon flash tank, which is used to depressurize and cool these materials before they are discharged from the vapor containers; and a 7000-gallon blow-off tank, which is designed to contain all of the steam or reactor coolant discharged through various primary plant relief valves. The discharge and vent system must be operated in order to drain a loop, to maintain the fluid level in the pressurizer within the design range, and to bleed off reactor coolant during plant warm-up.

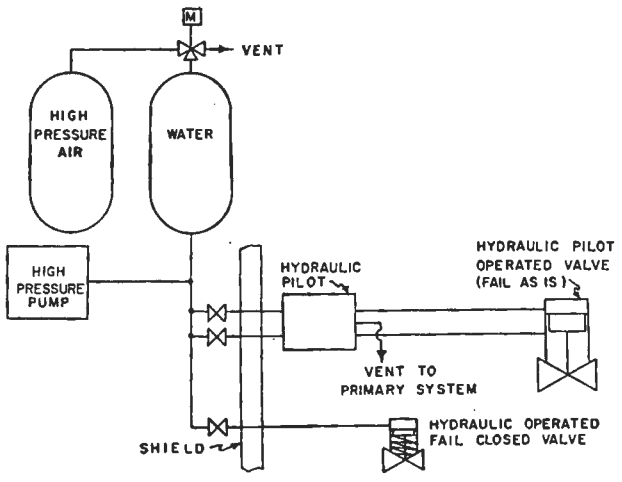

Figure 17. Valve operating system schematic

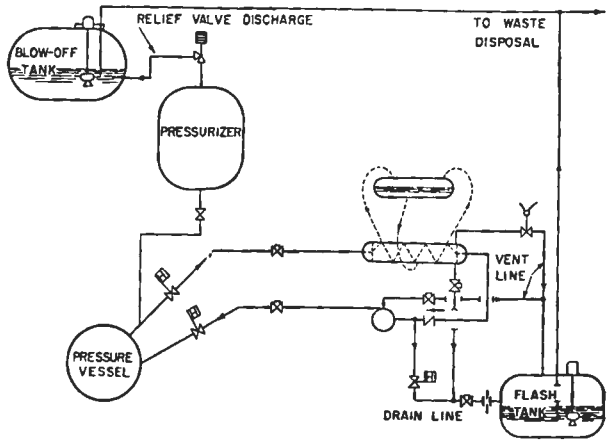

Figure 18. Coolant discharge system schematic

Waste Disposal System

The activity in the ion-exchange resin from the purification system and in the decontamination fluids and primary system effluent will be too high to permit dumping in the river. The volumes involved make packaging and subsequent disposal at sea infeasible. The waste disposal system, Fig. 19, now being developed will consist of two-stage evaporation, to reduce bulk, and subsequent underground storage of high-activity evaporator bottoms at the site. Spent resin from the demineralizers will be transferred by flushing, in the form of a slurry, directly to underground storage. Provision for collection and storage of radioactive gases is being made. Dilution and discharge through a vent stack will be used when meteorological conditions permit. The evaporator feed will be from 9000 to 21,000 cubic feet per month depending primarily on decontamination procedure. Evaporator vapor will be condensed and then diluted with condenser cooling water before release to the river. The permissible activity released to the river is being taken at ten per cent of the standard tolerance. A study is under way to determine the most economical method for disposing of combustible radioactive waste. The methods being considered are an incinerator at the site, or baling and shipping for disposal at sea. High and low activity laboratory wastes will be separated at the source and processed through the evaporator or diluted and dumped as dictated by the activity.

VII. Reactor Control

The reactor plant will be controlled to maintain a constant average temperature of the primary coolant as shown in Fig. 20. Therefore, the control method is one of average-temperature error detection and correction. On-off control of rods is initiated at plus or minus 3°F error, with a plus or minus 2°F deadband. Because of the slow response of the temperature-sensing elements and of heat transfer around the loops, the corrective action of the control would produce overshoot and oscillation of the system even if a considerably wider range of temperature variations were permitted. For this reason a damping effect is pro-

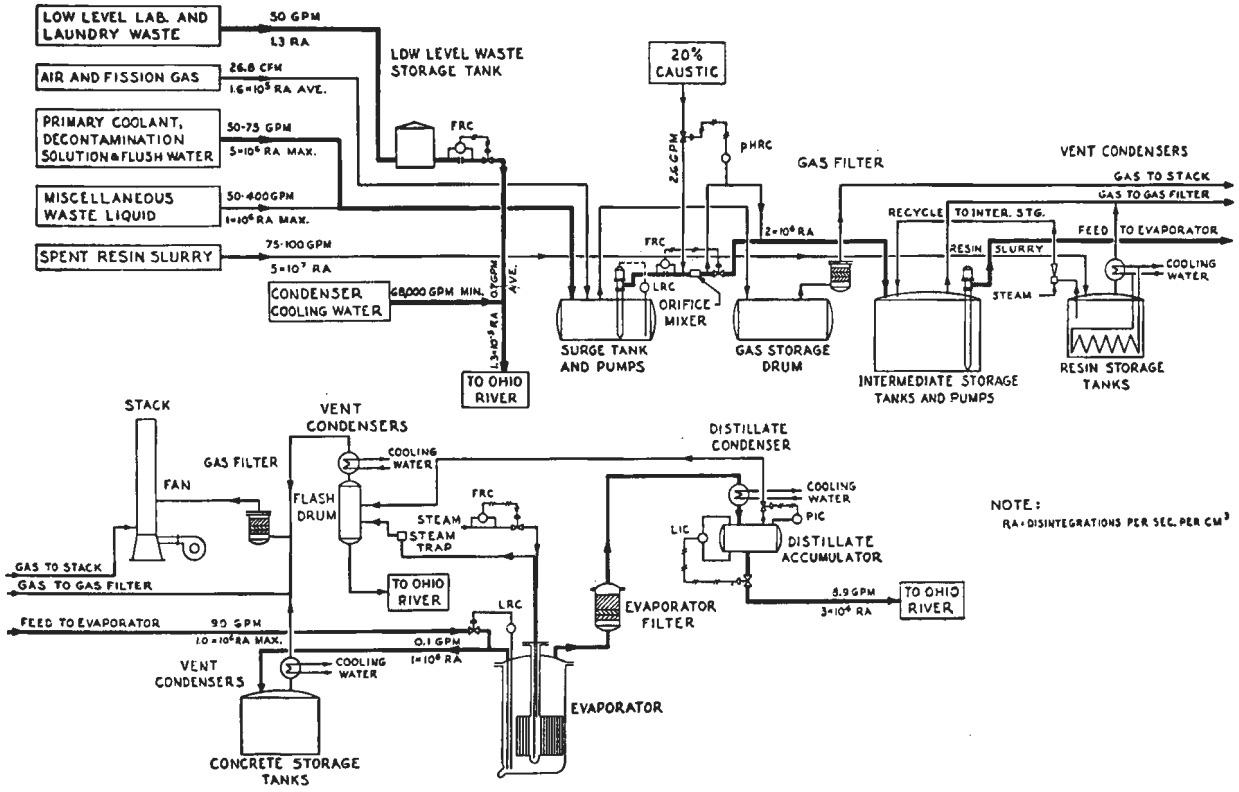

Figure 19. Waste disposal system schematic

vided by incorporating the rate of nuclear level change into the control signal.

The normal mode of operation in the power range and during low-power standby is by automatic control of rods, as described. Manual control can be employed by the operator if he desires. No provision is made for automatic startup except that automatic limitation is placed on the period obtainable by manual control during startup. It is also planned to have the operator manually adjust rods to warm up the plant at the design rate of 200°F per hour.

Since the plant is designed for constant average-temperature operation, the negative temperature coefficient of reactivity will adjust reactor power for short-time load transients. However, with no external control system, the average temperature of the plant would oscillate because of xenon variations with a period of several hours. The period of oscillation would be a function of load-change cycles. The external control loop removes these oscillations.

As designed, the external temperature-control loop permits the negative temperature coefficient of reactivity to adjust power without motion of the rods for load changes of the order of 10 to 20 per cent of full-power rating. This is shown here by a simulator curve, Fig. 21, for a step reduction in power from 100 to 84 per cent. No rod motion occurred, although the temperature control loop was connected. For larger load changes, the external loop assists the temperature coefficient by introducing rod motion to make a more rapid adjustment of reactor power.

Temperature measurements for the control system are made by resistance thermometers in the loops. Temperatures are measured at the boiler exits and at the reactor exit and averaged. The thermometers have a response of 3 to 5 seconds.

Nuclear Control

The functions of the nuclear detectors are to provide (1) level and period information to guide the operator during startup, (2) period control signals for startup protection, and (3) a rate of change control signal during power operation. Complete nuclear level coverage from source to 150 per cent of full power is provided by a BF_3 detector for the lower range in startup, and an ion chamber compensated for gamma

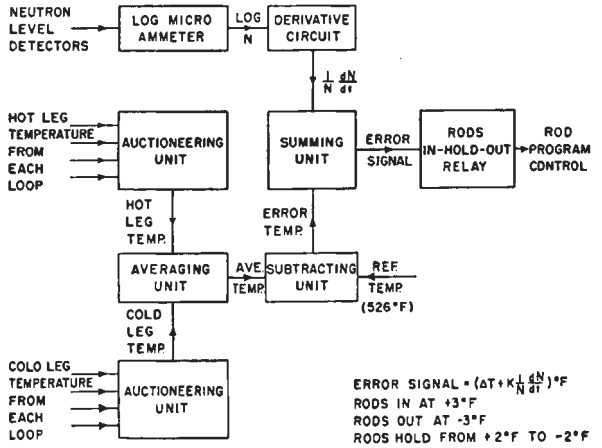

Figure 20. Reactor power and temperature control block diagram

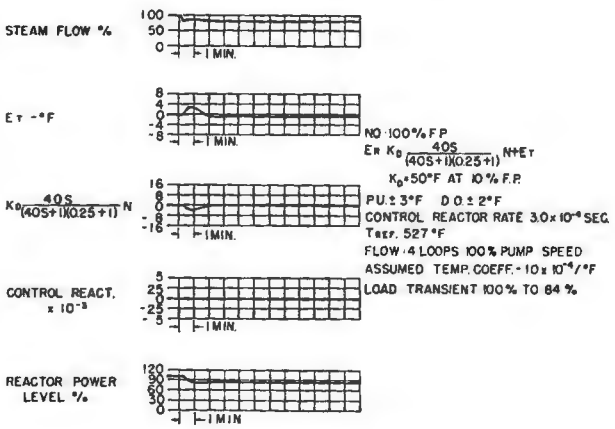

Figure 21. Simulator curves showing effect of load transient

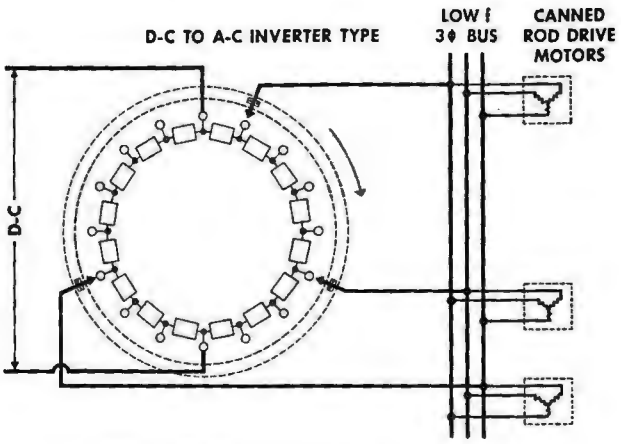

Figure 23. Control rod drive motor supply

radiation for the upper range, including power operation. Three channels, each having the two ranges of detectors, are provided for reliability. All detectors are placed next to the pressure vessel wall on the outside. No in-pile detectors are required.

Control Mechanisms

One of the mechanisms assembled on the reactor vessel head is shown in Fig. 22. These mechanisms are approximately 11 feet high and 6 inches in maximum diameter. The rod is attached to a lead screw, which is engaged by the four rollers of motor-driven roller nut. The roller cage and the motor rotor are carried on two water-lubricated ball bearings. The water-cooled motor stator is outside the pressure-containing cylinder, and hence is isolated from the hot water around the rotor parts. Three-phase, low-frequency, alternating-current voltage applied to the stator winding causes the salient-pole rotor and roller nut to rotate. See Fig. 23. The lead screw and control rod are thus raised or lowered at speeds proportional to the applied variable frequency. The low-frequency alternating current is provided by driving three equally spaced brushes over a commutator connected to stepped resistances arranged in a closed path similar to a motor armature winding. Direct-current power is applied across the diameter of the resistance circle. Upon loss of power, the rod weight causes the rotor to rotate and thus insert the rod into the core. The other cross-section view in Fig. 23 shows the rod position indicator. Solenoid coils outside the pressure walls indicate by a change of inductance the position of a magnetic slug attached to the rod. This change of inductance is sensed in the indicator circuit and results in a continuous indication of rod position.

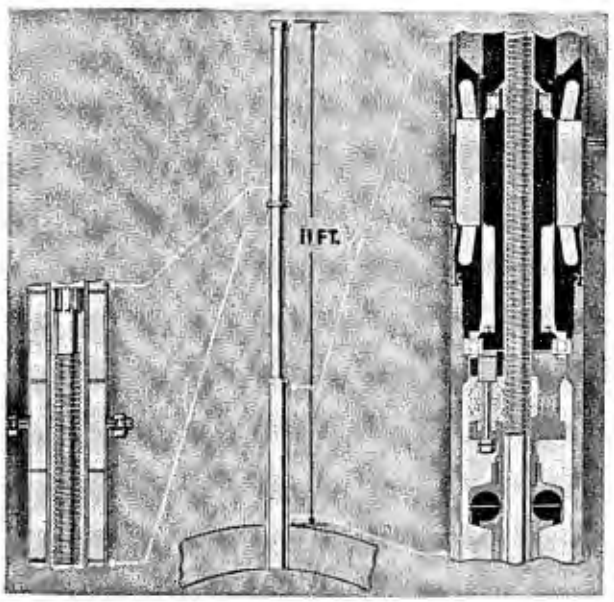

Figure 22. Control drive mechanism

The rod motors are programmed in groups of two or more moving together at any one time. Each group or pair of rods move a limited distance before motion passes to the next group, so that the tips of all rods are maintained in approximately the same plane.

Emergency Shutdown System

The function of the emergency shutdown system is to prevent the nuclear source from destroying or damaging itself. To perform this function the system must eliminate the possibility of any condition wherein the reactor generates more power than the load can absorb and the plant can safely handle. The action of the shutdown system is to reduce the source power level in anticipation of a damaging energy-conversion process.

The paramount requirement on a nuclear-reactor shutdown system is reliability. A system is reliable if it provides action when, and only when, such action is warranted. Accordingly, the objective in designing the PWR emergency shutdown system is to provide a system that protects the reactor against all damaging conditions, while in no way affecting normal plant operation, by utilizing the scramming function of the control drive mechanism.

If the emergency shutdown system is to be reliable, it must receive accurate information concerning the status of the plant and correctly interpret, transmit, and respond to this information. Inasmuch as all the conditions that endanger the reactor are manifested

by excessive core temperatures, the shutdown system signals must relate to temperature directly or to plant variables that can be correlated to it. The limitations imposed by presently available temperature detectors make the direct core temperature signals inadequate for shutdown. Shutdown signals must therefore be obtained from the behavior of plant variables that determine core temperatures.

The time required to detect, transmit, and actuate a shutdown signal is of the utmost importance because it determines the maximum permissible ratio of steady-state plant power output to maximum safe transient output. In PWR, the steady-state power limits have been extended very close to the transient limits, and therefore, the need exists for a very fast responding shutdown system.

The most reliable components and systems are not in general the fastest responding, and conversely the fastest responding systems have a tendency to respond to noise signals. The requirement that the shut down system incorporate fail-safe features imposes further conflicting requirements on the system components.

Regarding the character of the accidents that impose requirements on the shutdown system, it should be pointed out that once protection against a particular accident has been judged necessary, the probable frequency of occurrence of the accident has little or no effect on the design of the shutdown system.

Accidents that endanger the reactor core can be lumped into five groups, namely: (*a*) loss-of-flow accidents, (*b*) start-up accidents, (*c*) accidents involving the operation of the plant above safe maximum-power levels, (*d*) cold-water accidents, and (*e*) accidents involving the loss of primary coolant pressurization.

Loss-of-flow accidents in PWR fall into two categories: (*a*) those that require very fast action from the shutdown system, such as the loss of all pumping power, and (*b*) those for which an attempt must be made to keep the plant in operation by reducing generated power and load.

Start-up accidents can occur if the reactor period is not limited to a safe minimum value.

As pointed out previously, safe maximum level in PWR is a function of coolant flow rates or, equivalently, of the number of pumps in operation at any instant. The safe maximum power level is therefore threatened in any accidents involving loss of flow. However, even with undisturbed flow conditions, there exists the danger of the power level being raised inadvertently or accidentally beyond safe limits.

Cold-water accidents can occur if the water of a cold loop (or loops) is admitted to the reactor. The severity of the accident is a function of the ensuing reduction in coolant temperature and of the magnitude of the negative temperature coefficient of reactivity.

Leaks in the primary coolant system or failure in the pressurization system can cause the pressure to drop to values that result in intolerable boiling at operating temperatures.

VIII. Plant Container and Site Facilities

Description of Plant Container

The nuclear part of the power station is housed in four steel plant containers which, in turn, are in concrete compartments partially below ground level.

The reactor vessel is in a spherical shell 38 feet in diameter with an 18-foot cylindrical dome on the top to accommodate the extra height of the control drive mechanisms. Located on each side of the sphere is a 50-foot diameter cylinder, 97 feet long, in which are located two of the primary coolant loops and their associated equipment. An additional auxiliary container is located between the nuclear equipment containers mentioned above and the turbine generator. This container is a cylinder 50 feet in diameter and 147 feet long. The coolant pressurizing tank and other minor auxiliary equipment are located in it.

All of these containers are interconnected by several eight- and twelve-foot diameter ducts. The total gross volume is 600,000 cubic feet, and the net free volume is 473,000 cubic feet.

The plant container air-cooling system is designed to control the air temperature within the plant container to 122°F maximum and to provide means for filtering all air exhausted from the plant container. This system utilizes two 18,000 cfm fans to bring air into the plant container and two identical fans to exhaust air to atmosphere. The system contains no mechanical refrigeration equipment. Quick-closing air valves are provided in the inlet and exhaust air lines to close if required.

Access to the vapor container is permitted and interlocking access doors are provided so that the integrity of the container is not jeopardized by conventional entrance doors.

Design Considerations

The problem of containment is derived from safeguard considerations for protection of personnel against radioactive fallout outside of the 400-acre Shippingport exclusion area. The container is designed to prevent escape of any radioactive gases or vapor containing fission products that may be accidentally released as a result of any major plant casualty. The fission products normally remain in the fuel elements. They will only be released in any major quantities if a significantly large number of fuel elements fail during operation or if a large part of the core should melt. Both cases must be accompanied by a major primary coolant system boundary rupture; and in addition, the second case must be caused by a combination of high residual heat and loss of coolant. Both cases are considered as highly unlikely, but until more definite information is obtained it is an eventuality against which some protection is desirable.

The container is designed to handle the release of the energy in the approximately 20,000 gallons of primary coolant water at 535°F average temperature and 2000 psi pressure, the heat energy stored in the metal of the plant, and the energy of the worst conceivable

chemical reaction between the zirconium and water.

The pressure build-up in the container will vary in both magnitude and time, depending on the size of the break in the coolant system. The additional transfer of heat energy to the water for an intermediate-sized break makes this the worst case. The maximum pressure reached for the design volume is slightly over 50 psi.

If a rupture of the system occurs, a few seconds thereafter a spray will be started, and this will cause the pressure to decrease more rapidly after the first few seconds. This reduction of pressure makes it possible to take care of any likely chemical reaction between the zirconium and water, as this component of the total pressure comes into play only after an appreciable delay. With an adequate spray the total pressure will be no higher than the original peak.

Shield

The radiation level permitted throughout the plant varies depending on the amount of access required for operation or maintenance. In operating spaces the radiation level is held to less than 2 mr per hour. In spaces where only intermittent access is required the radiation is maintained at less than 5 mr per hour. Areas, however, which must be occupied only very infrequently and for short periods of time may have a radiation level as high as 50 mr per hour.

The reactor vessel is surrounded by an annular neutron shield tank providing a three-foot thick water layer. Other internal shields, such as provided between isolable loops, are concrete. In general, the external shield consists of the concrete structure surrounding the plant container. The thickness of the concrete required for structural purposes in supporting the load of earth on the outside is sufficient for biological shielding except in a very few places.

Radiation-monitoring equipment is provided as required throughout the plant area to determine the existing radiation levels and to warn the operating personnel of any unusual conditions.

Site Facilities

The fuel-handling building, which is the largest building on the site, is 44 feet wide, 182 feet long, and has an over-all height of 60 feet. Figure 24 shows this building and the fuel-handling canal and cranes. A 100-ton crane is provided for removing the reactor-vessel closure and the core as a complete unit if desired. Removal equipment is also provided for removing any fuel assembly or group of assemblies or for rearranging them either through the fuel ports in the closure head, or with the head removed.

The fuel assemblies will be removed from the reactor vessel under water and moved to the appropriate section of the canal for storage, further disassembly, repair or examination.

PART B. TURBINE-GENERATOR PLANT

IX. Design of Turbine-Generator Plant

The non-nuclear portion of the Shippingport power station is unconventional in many respects: It utilizes saturated steam at low pressure, and this pressure is not constant but rises substantially as the load is re-

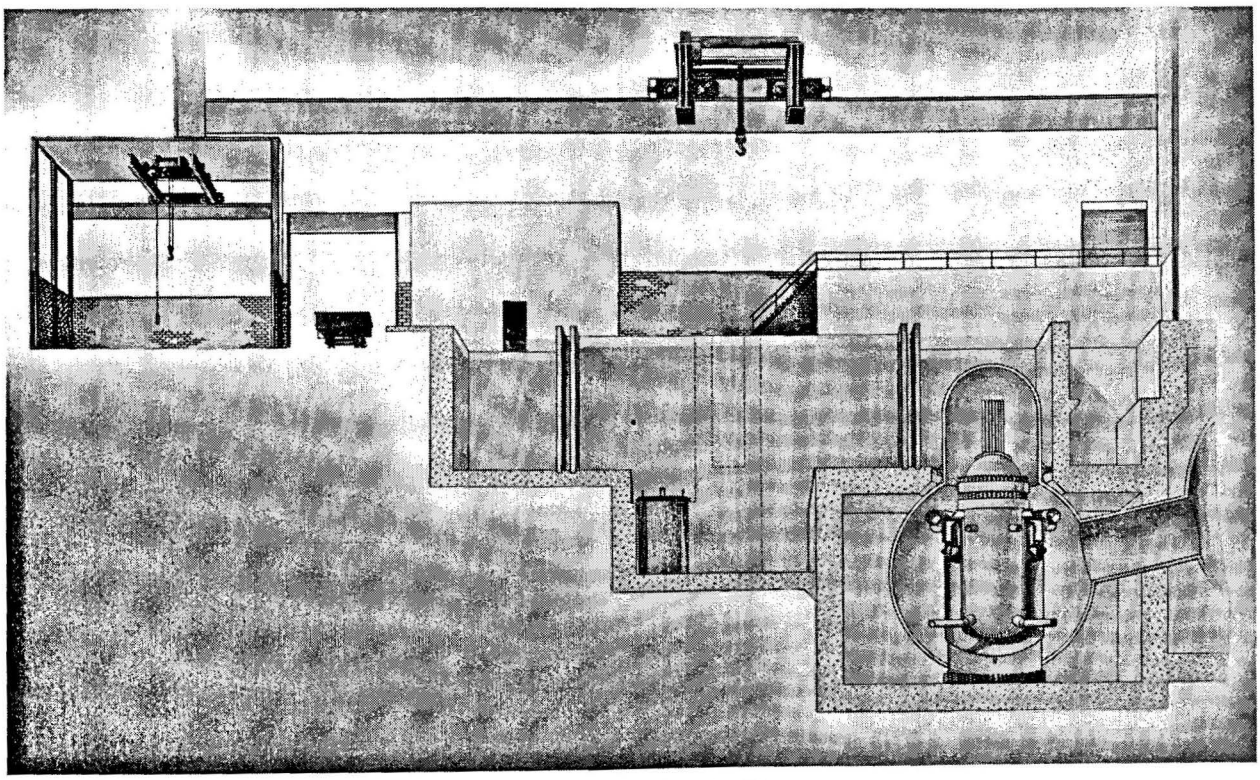

Figure 24. Fuel handling facilities

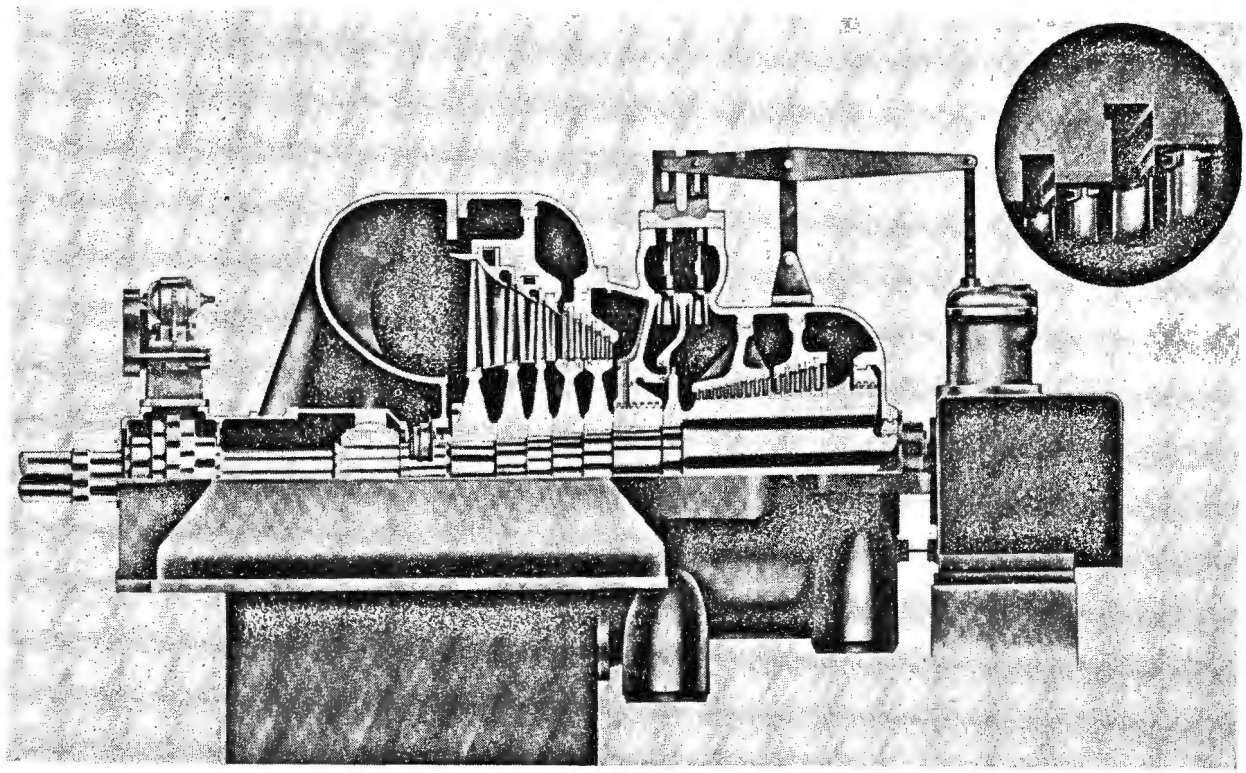

Figure 25. Turbine cross section

duced; the station service power supply is, due to the requirements of the nuclear portion, much greater than is customary for a plant of this kilowatt rating, and a small separate diesel-driven generator is also provided for extreme emergencies; the control, though essentially through a standard turbine governor, must recognize that the heat source is a reactor instead of a boiler fired by combustible fuel; and of course there is no conventional boiler with its fuel- and ash-handling equipment, fuel storage, dust collector, air heater, stack for combustion gases, and the like.

The turbine-generator is rated at 100,000 kw maximum capability and is a single-cylinder, 1800 rpm unit with direct-connected exciter. The turbine has three points of extraction of steam for feed-water heating. As shown in Fig. 25, the steam is admitted at some distance from the thrust-bearing end of the rotor, flows toward the thrust-bearing, and passes to an external moisture separator. It then passes back into the turbine at a point near the original point of entrance, from which it flows in a direction away from the thrust-bearing to the exhaust at the coupling end.

The cylinder barrel, between blade rows, will be lined with stainless steel laid in by electric welding, and all blades will be Stellite faced on the leading edges wherever the moisture content of the steam exceeds 6 per cent and the blade tip speed is 900 feet per second or higher. These precautions are expected to minimize the effects of the wet steam on the life of the turbine parts.

The full-load throttle steam pressure is 545 psig, and the corresponding feed water temperature is 342°F. The moisture content at the second extraction point, where the moisture separator is located, will be about 11.6 per cent, and this will be reduced to close to 1 per cent for the steam returned to the turbine. Under these same full-load conditions the exhaust moisture will be on the order of 13.2 per cent. This information, together with performance heat rates, is shown in Table IX-1 for two loads, and the turbine steam cycle diagram is shown in Fig. 26.

Table IX-1

Gross generator load, mw	79.3	100
Turbine heat rate, gross, BTU/kwh	11,272	11,385
Throttle pressure, psia	620	560
Throttle flow, lb/hr	987,350	1,286,900
Boiler feed temperature, °F	325	342
Condenser vacuum, in Hg	1.5	1.5

As load is reduced slowly under normal conditions of extraction and exhaust back-pressure, the steam flow, the heat transmitted in the boiler heat exchangers, and the temperature difference from coolant to steam will all become less. Except during transients, the mean temperature of the reactor primary coolant water will remain constant, and hence the boiling water temperature and pressure will rise. This rise is approximately linear and reaches a maximum of 870 psig at zero steam flow.

The thermal efficiency of a simple condensing cycle utilizing steam at 545 psig saturated and exhausting at 1½ inches of mercury absolute is 26.5 per cent. This is raised to 29.5 per cent by converting to the regenerative cycle of the type described. It is interesting to note that, since the thermal losses of the re-

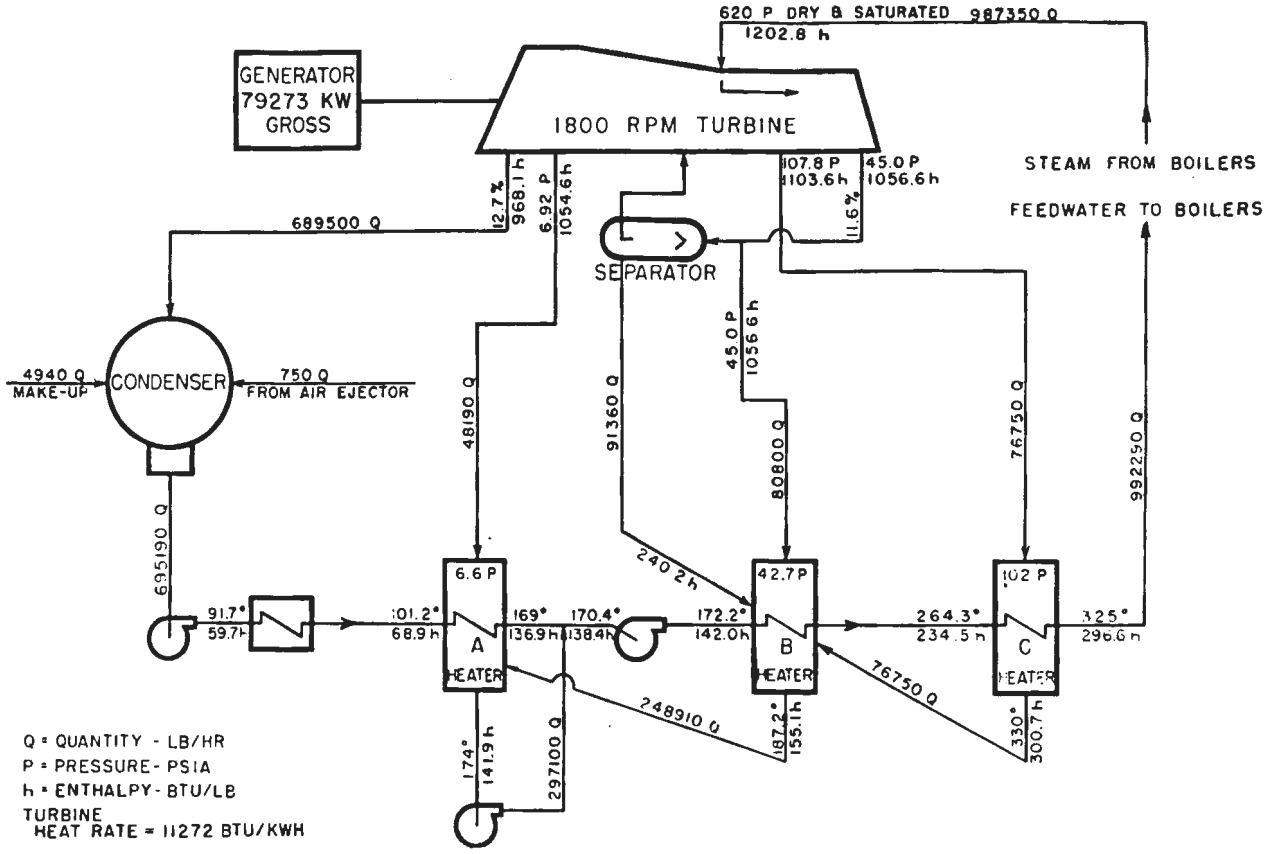

Figure 26. Heat balance diagram

actor and its appurtenances are very low, the turbine cycle efficiency after allowance for auxiliary requirements may be considered for practical purposes to be the thermal efficiency of the overall plant.

The station service power requirements are estimated to be 6000 kilowatts for the nuclear portion of the station, and 3500 kilowatts for the turbine-generator portion. The principal items in the former are the four primary coolant pump drives totaling 4800 kilowatts. In the turbine-generator portion the largest auxiliaries are the main condenser circulating water pumps, of which there are two at 900 horsepower each. These pumps operate at a substantially higher head than is usual for such pumps, due to the characteristics of the terrain and the plant elevation required by the occasional floods of the Ohio River at this point.

The turbine generator will be an outdoor type located on a deck, below which the condenser and auxiliary equipment will be housed. A semi-gantry crane will operate above the deck for handling of turbine generator equipment for maintenance purposes. The main and station service transformers will be located at ground level along the river side of the turbine structure. In order to protect the turbine generator and personnel in this area from the effects of a possible transformer oil fire, a heat-resisting barrier will be placed along the turbine side of the transformer area.

The station service busses to supply power for auxiliaries in both the nuclear and turbine generator portions of the station will be arranged in four sections, two served from separate secondaries of a transformer connected to the main generator leads, and two served from separate secondaries of a transformer connected to the 138 kv transmission bus. The four primary coolant pumps will then be served one from each of these four sections, and supply to other auxiliaries will be divided among them in such a way as to provide maximum security of service. A diesel generator of approximately 750 kilowatt capacity will be provided to operate certain essential components of the plant in the remote possibility of total loss of other power sources. These components are: coolant pumps for reactor decay heat, emergency lighting, storage battery charging, and turbine generator turning-gear operation.

In addition to the systems described above, there are many close connections between the turbine generator and reactor portions of the power station. In general, the supplies of water for charging, cooling, and other purposes originate in the turbine generator portion, also communications arrangements involving the telephone and loud-speaker systems, and sources for electric power, and main and emergency lighting.

A single control room will be provided in which will be centered the control of the reactor and its appurtenances, the turbine generator station service, and the outgoing power circuits. The principal electric circuits and piping arrangements for steam and water

will be pictorially shown on the control boards, with equipment, processes, switches, and valves shown, and suitable indicating lights placed thereon. This is considered to be a safety precaution to aid in minimizing operating errors, and it also presents at a glance the operating setup of the station for the guidance of the operators.

X. Electric Utility Company System

The Duquesne Light Company system is a metropolitan type system serving an area of 816 square miles, which includes the City of Pittsburgh. The system capacity before the Shippingport Power Station is installed will be 1,207,000 kilowatts. The bulk transmission system consists of 138 kv and 69 kv transmission lines with lower voltage distribution at 22 kv, 11 kv, and 4 kv. Large industrial customers are connected to the 69 kv transmission system, the smaller industrial customers being connected to the lower voltage distribution systems.

The system normally operates as part of a large interconnection with 32 million kilowatts of connected capacity extending through the entire east-central section of the United States. In order to provide adequate capacity reserves, each system must have a minimum reserve capacity equivalent to 10 per cent of its total installed capacity. The Duquesne Light Company system is tied to the interconnected systems through two tie lines, one at 69 kv and one at 132 kv, which have a combined capability of 130 megawatts.

The operating reserve capacity is based on replacing the output of the largest operating unit which during the initial operation of the Shippingport Station will be 150 megawatts. Of this total, 100 megawatts will be required in operating reserve capacity on the Duquesne Light Company system, and during emergencies 50 megawatts will be supplied by the interconnection. The system, on the other hand, is prepared to supply 50 megawatts to the interconnection during emergencies on connected systems. Since the capability of the Shippingport Station will be lower than the operating reserve capacity, the loss of this station due to a forced shutdown would not present a serious problem to the system.

The system load characteristics are of interest as they have influenced the design of the Shippingport Station. The usual daily load cycle occurs with the system maximum load usually occurring during the day as a result of the large industrial load. The weekday minimum night loads are about 50 per cent of the maximum day load, with the minimum week-end loads being about 42 per cent of the maximum. The heavy industrial load, which accounts for about 55 per cent of the system output, includes several continuous strip rolling mills and many large electric furnaces. These result in a load that is very erratic and has a variable demand. To meet the requirements of these loads the system output must be rapidly changed through a range of 60 to 80 megawatts at frequent intervals. A typical daily load cycle illustrating this is shown in Fig. 27.

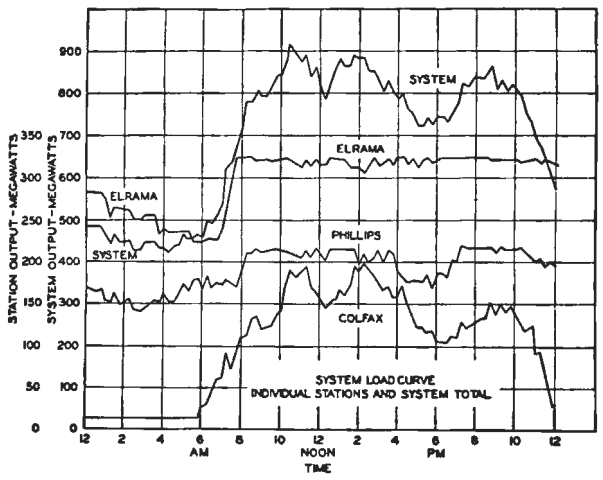

Figure 27. Typical daily curve

The several stations and the units within a station are assigned system load in accordance with the incremental cost of fuel. Adherence to this loading schedule, which produces the lowest fuel cost per kilowatt hour of output, results in the less efficient units or stations regulating the load changes during the maximum load periods, with the more efficient base load units aiding in load regulation during the minimum load periods. It is necessary, therefore, that all operating units be capable of sharing the regulation of system load. This requirement becomes more essential when consideration is given to the future expansion of the system. As newer and more efficient units are added, the older units initially operated as base load units shift to peak load operation, and are required to share more fully in the regulation of system load.

The actual change of station output to adjust to system load is accomplished by the automatic load control equipment. The tie line interchanges are telemetered to the System Operator's Office, compared to the scheduled interchange and the deviation between the actual and scheduled loads causes the turbo-generator unit outputs to increase or decrease until the tie line loads are equal to the scheduled loads. The load control equipment also allocates the load changes to follow the incremental fuel cost loading schedule. In Fig. 28 is shown the automatically regulated output of a peak load station during part of a typical day.

Emergency conditions usually involve the loss of generating capacity, the loss of one or more tie lines, or the loss of a block of system load. These conditions result in sudden changes of output which may equal the load of the largest unit and are often accompanied by system speed variation, causing rapid governor action. All units operating on the system must be capable of assisting in the adjustment of these load changes without becoming involved in difficulties.

The addition of a generating unit to a system such as the Shippingport Station involves several considerations. In general the unit must share with all other units the demands for power and reactive kilovolt amperes in proportion to the unit capacity. Also the energy sources, reactors or conventional boilers, must

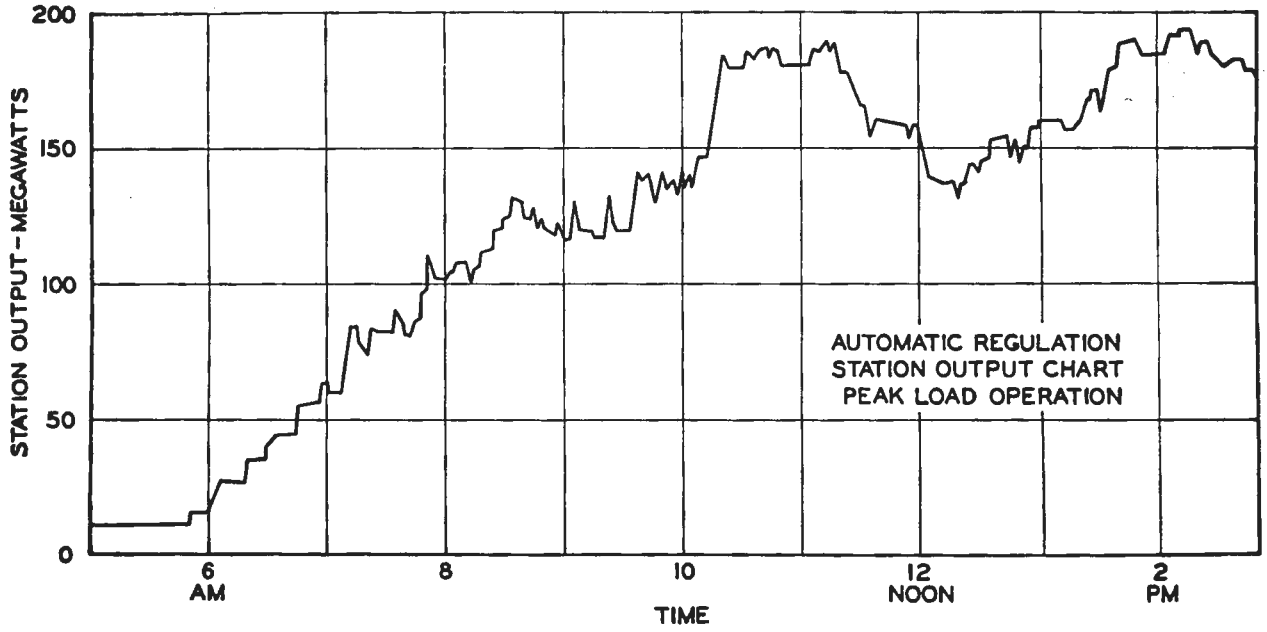

Figure 28. Output of peak load station under automatic regulation

not restrict the load adjustments within the specified limits. Otherwise additional capacity must be installed and operated to cover the deficiencies.

The design specifications for the Shippingport Station required the definition of the concepts outlined above in terms of significant numbers. The parameters were defined to cover the following basic requirements:

1. The over-all performance with respect to load changing ability should equal that of the average high pressure steam station using conventional boilers.
2. The load response characteristics of the nuclear portion of the station should provide for stable operation through system emergencies of normal magnitude.
3. The range of load changes and the rate of response to load changes should permit use of the nuclear station to assist other stations in regulating the system output.

The conditions stated for the emergency limits are as follows:

1. Loss of the largest operating unit with the system operating isolated from the interconnection. This is the Phillips No. 4 unit which is about 15 per cent of the expected maximum system output. Shippingport must increase its output 15 per cent, or 15 megawatts, for this condition.
2. Loss of a fully loaded tie line or one of several large sub-station loads. This emergency involves the loss of 100 to 125 megawatts, and Shippingport must reduce its output about 12 per cent, or 12 megawatts.

The conditions for normal load change limits are as follows:

1. A change of 10 per cent of the system load regulated on 50 per cent of the operating capacity requires a 20 per cent load change of the Shippingport output. This is about 20 megawatts.
2. A load change due to automatic load control will be limited to an increase or decrease of 20 megawatts and will occur at an average rate of 24 megawatts per minute.
3. A load change due to manual control will be limited to an increase or decrease of 15 megawatts for a block change. Further load changes will be limited to an average rate of 3 megawatts per minute.

The manual load changes may occur at a rate of 3 megawatts per second. The load changes due to automatic load control will not exceed a rate of 0.4 megawatts per second or 24 megawatts per minute. The difference between the response rates of manual and automatic control allows a higher load range for the automatic control.

XI. Operation of Shippingport Power Station

The Shippingport Power Station is being designed to meet all conditions outlined previously. The load changing characteristics of the overall station will be comparable to those of conventional station equipment. The reactor proper, having a negative temperature coefficient of reactivity, will be essentially self-regulating, and will possess a stability during load changes in excess of that found in conventional coal firing equipment.

The station will have a range of automatic operation extending from 20 per cent to 100 per cent full load. When operating within these limits, it will be capable of following daily system load changes at an overall average rate of 3 megawatts per minute, and of accepting load swings of 20 megawatts maximum at a rate of 24 megawatts per minute. It also will be capable of handling a proportional part of the step-change load swings imposed on the various stations by casualties on the system.

The start-up and shut-down requirements will compare favorably to those of conventional station equip-

ment. The time required for the station to reach the operating range from the cold shut-down condition will be approximately 3½ hours. Following an overnight shut-down, 1¾ hours will be required to reach this point. The time necessary to perform these operations on conventional equipment of this size is 5½ hours and 2½ hours, respectively. The required time for shut-down of this and conventional station equipment is essentially the same; however, provision must be made in this station to remove the heat which is generated by the reactor for a period following shutdown.

These requirements and the load-changing characteristics outlined above indicate that the reactor plant, although differing in many respects from conventional equipment, has no characteristic which would limit its ability to serve as either a base or peak load station on the system.

The presence of radioactivity requires that shielding be placed around the reactor plant equipment. This shielding and the presence of radioactive contamination after shutdown will complicate the maintenance of this portion of the station. The shielding is so designed that minor maintenance can be performed on certain portions of the plant with the remainder operating. For those items requiring plant shutdown, adequate time will be available during the light week-end load periods to carry out the necessary decontamination procedures, perform minor maintenance, and return the plant to service.

The test program planned for the first few years of operation may prevent maximum utilization of the station due to the reduced outputs necessary for the conduct of certain tests and inspections. This program is designed to determine the characteristics of the entire station, to provide detailed information on the core and associated equipment, and to determine operating costs.

XII. Manpower Requirements

A preliminary schedule of manpower requirements for the Shippingport Nuclear Station has been prepared. A comparison of manpower requirements at the Shippingport Nuclear Power Station has been made with a conventional station having a 100 megawatt single boiler, single turbine plant placed in service by the Duquesne Light Company on 3 April 1952.

	Conventional station	*Shippingport Nuclear Station*
Supervision	11	11
Operation	27	48
Plant protection	4	10
Clerical	1	7
Maintenance	8	14
House and yard laborers	5	12
Coal and ash laborers	6	—
Test engineers	1	9
Chemists (all grades)	1	9
Cooks and kitchen helpers	—	5
Telephone operators	—	5
Total	66	130

If this plant were a normal operating plant without the unusual necessity for tests, investigations, studies, and security, the plant organization could be reduced to 81 by the elimination of

Plant protection	5
Clerical	6
Chemists—including the chief chemist	9
Maintenance	6
Test engineers	6
House and yard laborers	7
Cooks and kitchen helpers	5
Telephone operators	5
Total	49

The manpower estimates for the Shippingport Station are conservative in that allowance has been made for the lack of operating experience with a nuclear station of this size. It is believed that the preliminary schedule provides an adequate operating force, and in addition provides constant supervision by specialists trained in nuclear matters to guide the operating forces. It is within reason to expect that operating experience will develop procedures in operation and maintenance so that personnel requirements for a nuclear station may be equal to or perhaps even less than requirements of a conventional coal burning station.

XIII. Training Program

In order to train personnel for the Shippingport Power Station in the wholly new task of operating a reactor, a training program has been planned which will start approximately two years before the initial operation of the plant. This program includes class work and practical training. The class work will include the following: nuclear physics, reactor technology, radio chemistry, reactor chemistry control, health physics and security control.

The remainder of the training program is designed to provide practical experience in these subjects along with training in actual reactor and conventional plant operation.

The training schedule assumes that some of the personnel will have had some of the training needed beyond specific qualifications, and the training schedule is made to provide additional necessary training. Actually when the personnel has been selected, some will have had at least a part of the training provided in the schedule in which case the training would not be repeated.

The training schedule is designed to give each employee all of the training which is needed for each job, the minimum training proposed is a course in health physics and a course in security control. In each case the course is to be designed to cover the particular needs.

PART C. CORE DESIGN

XIV. General

A most important item in the design of power producing nuclear reactors is the design of the reactor core itself. A successful nuclear power plant can only

be obtained if the core is able to withstand the thermal, hydraulic and mechanical conditions imposed on it not only under steady state conditions but under transient and foreseeable casualty conditions as well. Some of the major engineering problems faced in the design of cores for pressurized water reactors are covered below. The basic criteria used in the thermal, hydraulic and mechanical design of the core both from steady state and transient considerations are reviewed and their interplay with core physics considerations are discussed. For clarity of presentation the section below has been divided into two main sections; namely, Mechanical Considerations, and Thermal and Hydraulic Considerations. It is desired to emphasize, however, that in actual practice the design of a core cannot be divided arbitrarily into a number of self-contained parts and that a satisfactory core design is only obtained by judicious balance of the numerous and somewhat divergent requirements involved.

XV. Mechanical Considerations

The mechanical design of cores for pressurized water reactors lies in a new engineering field about which there is little practical experience. Since reactor cores do not lend themselves to the type of servicing which can be performed on conventional heat transfer equipment, they must by necessity be simple, rugged and able to take appreciable misuse. Unless great care is taken in the initial reactor core concepts, there is good likelihood of ending with a complicated core design having close tolerances, numerous precision-built small parts, and requiring an excessive amount of machining. Such cores would not be likely to withstand mechanically the long lives now indicated possible by such nuclear physics developments as burnable poison control.

Numerous factors affect the mechanical design of reactor cores. While the engineering considerations associated with conventional heat transfer equipment are also generally applicable to reactor cores, there are a number of major criteria listed below which particularly affect the mechanical design.

1. The fuel element material, the fuel element geometry, and the manner in which the fuel elements are to be grouped.
2. Flow conditions.
3. The type of mechanical control that is to be used, its associated drives, and the servicing requirements of the mechanisms.
4. The method of refueling, whether or not the refueling of the core is to be performed as a cartridge or in a number of separate pieces, and the extent to which the reactor vessel must be opened for the refueling process.
5. The extent and type of core instrumentation that is required.

These considerations, in turn, affect the design of the reactor vessel in which the core is to be installed. In addition to the above factors the designer must also face the problem of providing sufficient flexibility in the core design as it progresses to be able to incorporate necessary changes indicated from physics, mechanical, thermal, or hydraulic tests.

The effect that the above considerations have on various parts of the reactor, i.e., fuel element arrangement, core structure, control elements and associated drive mechanisms, core instrumentation, reactor vessel and its closure, and finally on the refueling equipment, is described below.

Fuel Element Arrangement

The core, or active portion of the reactor, usually consists of uranium-bearing fuel elements with flow passages around these elements for circulation of the coolant. The choice of fuel element material is based mainly on nuclear and metallurgical considerations. However, mechanical as well as nuclear, thermal and hydraulic considerations have to be taken into account in determining the shape of the fuel elements and how close they are to each other. A water-to-metal ratio that may be desired by the physicists may not result in an adequately rugged core for the intended service and consequently some compromise must be reached in the design. In determining the shape of the fuel elements and their spacing, one important factor among many that must be taken into account, is whether the deformation or rupture of the fuel element shape chosen is liable to clog the coolant passages around it and as a result cause failure, due to insufficient cooling, of adjacent fuel elements which in turn would cause others to fail. This type of casualty is sometimes referred to as "progressive failure." The adequacy of the fuel element design to be used cannot be proven by analytical methods alone. Therefore, representative fuel element bundles are tested in test loops both out-of-pile and in-pile under expected service conditions of flow, temperature, pressure, and neutron flux to prove the adequacy of the design. In the in-pile tests mechanical defects are purposely introduced in one or more of the fuel elements in the bundle to determine whether a progressive type of failure can occur. These tests are somewhat costly and time consuming but it is considered that they are fully justified and in the long run reduce over-all costs since they do provide maximum assurance of having a satisfactory fuel element arrangement. An example of failure of a natural-uranium fuel rod after corrosion testing in hot water with an intentionally induced cladding defect is shown in Fig. 29.

In the final selection of the fuel element shape every effort is made to use shapes which can be easily manufactured with a minimum of machining, which can be readily clad for retention of radioactive fission products, and which lend themselves to satisfactory grouping. Plates, rods, and ribbons are some fuel element shapes which have shown promise. Typical fuel element shapes are shown in Fig. 30. For example, depending on the transformation temperature of the fuel element materials, plate thicknesses of 30 to 100 mils with corresponding coolant channels are in the normal range of interest. As indicated in a previous section of this paper, the fuel elements for the PWR

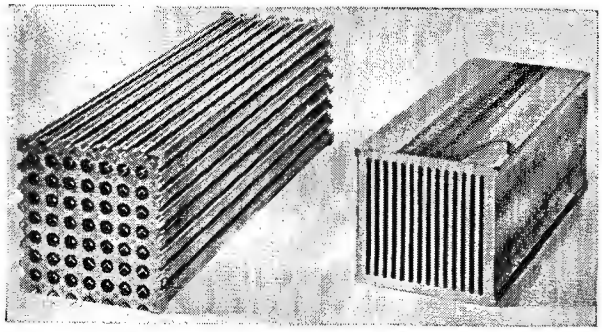

Figure 29. Failure of natural uranium fuel rod with intentionally defective cladding after testing in hot water

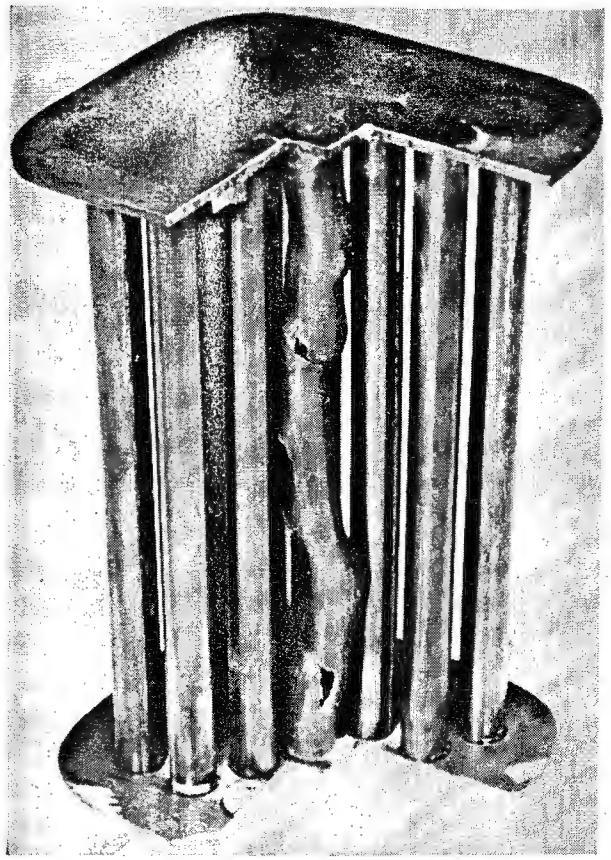

Figure 30. Typical fuel element shapes

seed are 0.080 in. thick plates with 0.080 in. coolant channels, and for the blanket are 0.413 in. OD rods on 0.468 in. centers.

A grouping of fuel elements with their coolant passages is usually referred to as a subassembly. Various geometries can be used for the cross-section of the subassemblies, hexagonal or rectangular patterns usually providing satisfactory arrangements. From the standpoint of ruggedness, subassemblies are desired wherein all the fuel elements are welded in the subassembly with the necessary space between them to form the coolant passages. In such a design there is the problem of welding various geometries and getting a good weld penetration without impurity pickup in the clad material and without having the heat-effected zone extend into the uranium-bearing material. Inasmuch as some of the impurity pickup may come from the air, the welding in some instances may have to be performed while shielded from the atmosphere.

Subassemblies can also be held together by mechanical means. This method, however, can result in requiring extensive machining of the fuel elements to receive the mechanical fastenings. In addition, close tolerances are necessary so that the pieces fit tight and are not subject to vibration, fretting corrosion, and crevice corrosion. A further consideration in the design of these subassemblies is the desirability of replacing one or more of the fuel elements if they fail prematurely in service so that the whole subassembly in which the failed fuel element is located does not have to be scrapped.

The division of the core into a number of subassemblies facilitates core fabrication and handling. Furthermore, the use of orifices in the subassemblies provides a convenient way for controlling the coolant flow to various portions of the core.

Interspersed among the fuel elements are located a number of control elements for controlling the chain reaction in the reactor. These control elements contain materials of high neutron absorbing properties such as cadmium, boron or hafnium. The shape and number of these control elements and their required location, as determined from nuclear considerations, play an important part in establishing the over-all arrangement of the fuel elements in the core. They also can affect the over-all core diameter since additional space may be required for inserting the desired number of control elements.

Reactor Core Structure

The methods that are to be used for refueling the core and/or for the replacement of a defective fuel element after the core has been placed in service largely dictate the mechanical means by which the fuel element subassemblies are to be supported. If the core is to be refueled in pieces, the subassemblies are usually supported in a stainless steel cage consisting of a barrel which sits on a ledge in the pressure vessel, a bottom plate which receives the ends of the subassemblies and a top support grid which contains means for latching the subassemblies in place. The core cage can be clamped in place by a barrel extending from beneath the head of the pressure vessel to the top flange of the cage. A Belleville spring can be used in this arrangement to take out machining tolerances and differential expansions of various materials used in the core during operation. Above the cage a structure has to be supplied for guiding the control element drive shafts and shrouding these control elements from the hydraulic cross flow forces arising when the elements are partially or wholly withdrawn from the core.

If the core is to be refueled as a cartridge the construction can be somewhat different. In such a design the subassemblies can be fastened directly to a bottom

plate and to a top support plate without an intervening barrel to tie the two plates together. The plates must have the appropriate flow passages for entrance and exit of the coolant. Above the top support plate an upper barrel assembly must be provided which contains the control element guides and cross flow shrouds. Such a construction provides a rigid structure in which alignments can be made and readily held within acceptable limits during actual operation of the core. This cartridge refueling concept, however, becomes rather impractical for large cores due to the large shield requirements, extremely large weight and size of the refueling equipment and the resultant transportation problem involved. Furthermore, this type of core construction does not lend itself readily to servicing, in that a failed fuel element cannot be removed without removing the entire core from the reactor vessel. In the PWR, the core is so arranged that any subassembly can be removed through small openings in the pressure vessel head as described later in this paper. In addition, the plant is so engineered, that if desired, the core can be removed as a cartridge and moved into a servicing area, the whole removal and transportation operation being performed under water. An isometric view of the PWR core showing the structural components is presented in Fig. 31.

An important factor which affects the over-all design of the reactor structure is whether the core is to have a one-pass or multi-pass flow. From a thermal standpoint a two-pass core, for example, has advantages over a one-pass core in that for a given heat transfer area the desired power can be obtained with a smaller flow, of the order of 60% of that required for a one-pass core. However, the baffling necessitated by the two-pass design usually results in appreciably increasing the complexity of the core so that a balance must be made between mechanical complexity and thermal performance. The importance of reducing mechanical complexities in the core cannot be overemphasized; from the nuclear and metallurgical standpoints, means are being developed for increasing the life of cores and unless simple and mechanically rugged core designs can be obtained, mechanical, rather than physics considerations will limit the life of cores. With these criteria in mind, the design of the first PWR core uses a one-pass flow arrangement and is being directed towards the use of an all-welded construction with practically no mechanical fasteners except for the individual subassembly latching devices. As experience is gained in the various technological fields involved it is expected that different types of cores of improved designs will subsequently be installed in PWR.

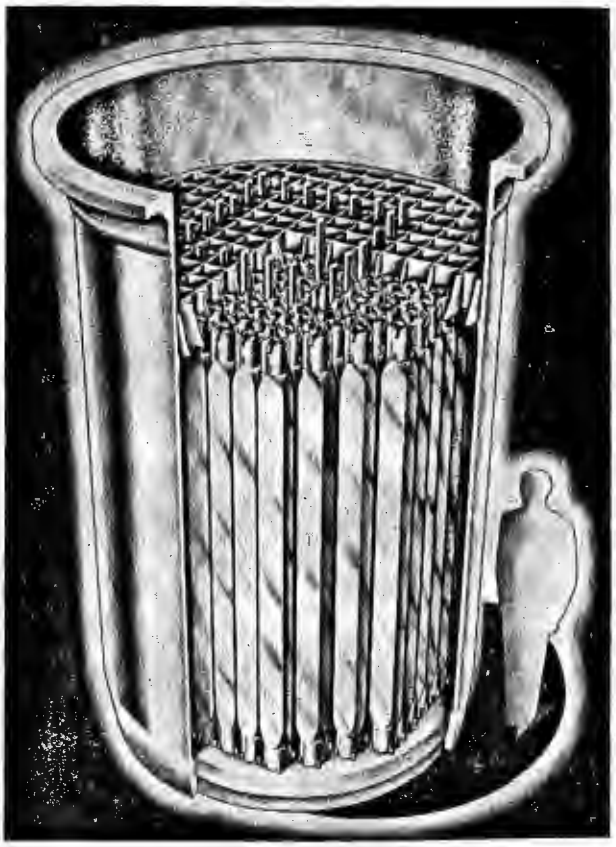

Figure 31. Perspective cut-away view of PWR core

In the design of the structural supports of the core, attention must also be given to the problem of cooling them. The gamma rays emitted from the core in passing through the structural supports cause internal heat generation which without proper cooling would create severe thermal stresses with resultant thermal distortions. Such distortions could lead to binding and malfunctioning of the control elements as well as obstruct the removal of individual subassemblies from the core. The heat generation in these structural members is obtained from analytical studies backed by nuclear critical experiments of representative reactor mockups.

Control Elements and Associated Drive Mechanisms

Water-cooled reactors are usually controlled by moving control elements in and out of the core. The amount of control required for a given core depends largely on the uranium loading of that core. With regard to determining the shape, number and location of the control elements, numerous conflicting factors come into play. The final design usually ends up in a compromise of these factors. Some of the factors are:

1. The control element must have adequate structural strength and must not distort under the thermal gradients it will experience.

2. The space left in the core when the control element is raised must be of such thickness as to prevent excessive peaking of the neutron flux in that area. This is sometimes achieved by having a non-moderating material of low neutron cross section material attached to the end of the control element to fill the space vacated as the control element is raised. These extensions are sometimes referred to as "control element tails." The objection to them is that they require larger reactor vessels, additional baffling and

guides and thus tend to complicate the over-all reactor design.

3. The number of control elements must be such that they can be installed in the core and that there is sufficient space in the reactor vessel head to provide the necessary penetrations for driving them. From a maintenance standpoint, it is desirable to be able to remove a control element through its penetration in the reactor vessel head so that the size, number and/or location of these elements in the core is limited by the holes that can be provided in the reactor vessel head.

4. From safety considerations, a larger number than the minimum number of control elements required for nuclear considerations must be provided to allow for the malfunctioning of one or more of these elements. For example, the reactor should be capable of shutdown with at least one and preferably more elements stuck in their least effective positions.

5. The distance between the centers of adjacent penetrations in the reactor vessel head must be such as to provide adequate space for the control element drives.

6. The shape of the control element must be such that it can be fitted either within the subassembly or between adjacent subassemblies. If the subassembly design does not lend itself to any reasonable control element shape, it becomes necessary to modify or even drastically change the subassembly configuration.

7. The control elements should preferably operate in some sort of closed channels so that the flow of coolant to these control elements can be properly controlled. These channels could also provide the surfaces for guiding the control elements in the core.

The control elements are usually made the same length as the core; at the top, each element is fitted with a shaft which extends through the reactor vessel to a drive mechanism for moving the element in and out of the core. This shaft operates in linear bearings contained in the core structure guide tubes described previously.

The problems associated with the development of control element drive mechanisms in the final analysis reduce to one of reliability. Mechanisms based on a wide variety of principles could be used. However, most of these would require the use of linear shaft seals with associated auxiliaries to prevent leakage of coolant from the reactor. In order to avoid these rather complex shaft seal systems which are of questionable reliability and require appreciable maintenance, effort has been extended in the development of mechanisms which can be placed in completely welded housings open to the interior of the reactor vessel. These housings are welded to the openings in the reactor vessel head, and care must be taken to insure their proper alignment. Of the mechanisms that have been developed for this application, one type which has been successful involves a canned-rotor variable-reluctance motor driving a lead screw through a ball nut or collapsible-roller-nut arms. The lead screw is directly or indirectly connected through a latch mechanism to the control element. The mechanisms provide for the desired different rates of insertion and withdrawal of the control elements and for spring driven scrams of the elements into the core in the case of emergency conditions, such as the loss of all electrical power. In case of any mechanism malfunction, the individual unit can be readily removed as a cartridge by opening a seal weld on the top of its housing. A new mechanism can then be inserted. By providing individual and self-contained drives for each control element maximum reliability and flexibility is obtained in the control of the core. The PWR core is provided with individual drives for each control element, and these drives can be readily removed for servicing if necessary.

The fact that control element drive mechanisms have to operate in hot water without any additional lubrication creates difficult material and mechanism design problems. Organic lubricants are not considered acceptable because they tend to sludge under irradiation and may deposit on fuel element heat transfer surfaces. The materials used in the mechanism must have high corrosion and wear resistance and materials in contact with each other must not be such that they enhance crevice corrosion. For this reason extensive tests in autoclaves at design temperature and the proper coolant chemistry must be made of bearings and all other mechanism components. Final autoclave tests under simulated operating conditions must be also performed on the completed mechanisms. It is only through thorough and extensive component and prototype testing programs that the suitability and reliability of a given mechanism design for reactor use can be determined. This procedure has been followed in the design of the PWR control drive mechanisms.

Core Instrumentation

In order to further the technology of water-cooled reactors, it is important to be able to measure certain core parameters while the core is actually in operation and check them against theoretical calculations. Fuel element temperatures, flow through individual subassemblies, flux distribution and detection, and location of failed fuel elements are among the more important measurements desired, especially for a developmental core such as the first PWR core.

The measurement of fuel element temperature can be performed by embedding small size thermocouples into selected fuel elements both in the seed and blanket portions of the core. The flow through each subassembly can be measured by the installation of dynamic and static pressure taps at the inlet and a dynamic tap at the outlet of the subassembly. While the installation of these thermocouples and pressure taps are appreciable problems in themselves, a more difficult design and arrangement problem exists in the grouping of all these leads from the core and providing for their passage through the reactor vessel head without impairing the leakproofness of the system. Furthermore, any failure of this instrumentation must not present any hazard to the core. One method which

has been used successfully for passage of leads through the reactor vessel is to braze the individual leads into plugs which in turn are welded to instrument ports in the reactor vessel head.

The problem of sensing fuel element failure is also a difficult one. A possible method is to provide a sampling tube for each subassembly; these sampling tubes are manifolded into headers at right angles to each other in such manner that a change in the radioactivity in two perpendicular headers will indicate the subassembly in which a failure has taken place. This works in about the same manner as one would use in determining the location of a point on a rectangular coordinate grid system. The problems of incorporating the sampling tubes into the core structure, manifolding them, and passing them through the pressure vessel wall are similar to those associated with the flow and temperature leads.

Neutron flux instrumentation can be accommodated by attaching to the pressure vessel head stainless steel thimbles extending down into the core. The neutron flux instrumentation can in this manner be installed or removed without depressurizing the system. Appropriate cooling, however, must be provided to protect the instrumentation.

Every effort must be made to prove the reliability of the instrumentation described above prior to its incorporation into a given core. To this end numerous component tests and full scale tests at design pressures, temperature and flow are performed out-of-pile. In addition, wherever possible, special in-pile tests are also run. Such in-pile tests particularly apply to temperature, flux, and fuel element failure instrumentation.

Reactor Vessel and Closure

In a pressurized-water reactor, the core must be housed in a reactor vessel suitable for the operating pressure and temperature and designed to withstand the nuclear radiation from the core. Corrosion resistance to the coolant can be provided by using a stainless steel cladding on the inside of the vessel. A typical material selection is a base material of SA-302 or SA-212 clad with AISI-304 or 347 stainless steel. Reactor vessel materials such as SA-212 for instance have received extensive in-pile irradiation tests to determine whether the physical properties are impaired to any significant extent. The in-pile test data obtained indicates these materials to be satisfactory for this service.

Reactor vessels usually consist of a cylindrical body with a hemispherical bottom and a removable hemispherical head. In the design of the reactor vessel, the following are some of the more important considerations:

1. The location of the inlet and outlet nozzles and the permissible minimum angles between nozzles on the same plane. In this connection whether the inlet and outlet nozzles are on the same plane or the inlet nozzles are at the bottom of the vessel is as much a function of the plant arrangement as the pressure vessel design.

2. The number, thickness, and arrangement of the thermal shields to protect the reactor vessel walls from nuclear radiation, and thus reduce the heating taking place in the vessel wall. These shields must be made of appropriate thickness so that they can be adequately cooled to avoid distortion. For ease of manufacture and installation the number of shields should be kept to a minimum. Maximum flexibility is obtained by making these shields removable after the reactor vessel head and core are removed.

3. The penetration requirements and closure method for the reactor vessel head. As indicated previously the major difficulty in designing the vessel head is providing the necessary relatively large and numerous head penetrations for control element mechanisms and subassembly and control element removal purposes. A preliminary evaluation of the design is made by the use of three-dimensional photoelasticity technique. Since the Poisson's ratio for the plastic model is different than for steel, a final stress evaluation is made using a ¼ scale steel model. However, in order to determine the effects of thermal cycling on the reactor head, on the alignment of control drive mechanism housings, and on the closure seal weld, a full-scale prototype of the head and closure must be built and thermally cycled. This full-scale mockup is also used for making, cutting, and remaking the closure seal weld under the space limitations that will exist in the actual plant to insure the adequacy of the design, for checking the performance of any automatic equipment used, such as for example weld cutting machines, and for training operators in the procedures to be followed in the installation and removal of the reactor vessel head.

4. The heating and cooling rates to which the vessel will be subjected. Rates of the order of 200-250°F/hr are desired.

5. The over-all height of the vessel and the method of mounting it in the system.

In the PWR reactor vessel the head opening is 109 in. ID, so that the reactor head design with its numerous openings is a major engineering undertaking. The test procedures described above are being followed in connection with the PWR reactor vessel to prove the adequacy of the design. Of a large number of vessel closures that are reviewed, two were selected for detailed evaluation. These are (1) a bolted closure using 42 six-inch studs as used in bolting large steam turbine casings, and (2) a conical-segment closure which is an adaptation of the more conventional shear ring design using individual conical segments in place of the shear ring.

In both closure designs a seal weld is used to prevent leakage. In the conical-segment closure, the seal weld is at the top and consists of a thin metallic membrane welded to the head and vessel and backed up by a carbon steel support ring and a pair of bolted retaining rings. In the bolted-closure design, the seal weld is at the periphery of the closure and consists of

a flexible torispherical ring welded to the head and vessel. Of these designs, the bolted closure has been selected as the reference design for PWR because of its relative simplicity both in design and fabrication and because it permits greater flexibility in the core design.

Refueling

The concept of being able to remove subassemblies, control elements and control drive mechanisms through relatively small openings in the reactor vessel head has already been discussed. This concept is especially important for a developmental reactor such as the PWR. Apart from increasing its over-all reliability and ease of servicing, it also provides means whereby different fuel element subassemblies can be readily installed and tested in the core. One alternative to this concept is to remove the reactor vessel head each time a subassembly has to be removed. This would entail a long reactor shutdown, since it would involve the breaking of numerous electrical and cooling-water connections, the removal of an appreciable number of large bolts, the cutting of a 109 in. diameter seal weld and finally the reassembly of all these parts.

To perform removal operations through ports in the reactor vessel head, the reactor is shut down and the system depressurized and cooled to about 200°F; to perform these removal operations with the system at temperature and pressure is not considered desirable since it would involve complicated mechanisms and pressure interlocks and furthermore, it is questionable whether any significant saving in time would be obtained.

The removal of control drive mechanisms is relatively straightforward. The seal weld at the top of the housing in which the mechanism is located is cut, the drive mechanism is detached from the control element, and the mechanism is removed as a cartridge. The removal of a control element can be performed in a similar manner except that in this instance, because of its radioactivity, it is either removed into a shielded and cooled coffin for transfer to a storage or inspection area, or it is directly transferred under water to the desired area. The procedure for removal of fuel element subassemblies, however, is not as easy. Possible removal schemes, core arrangement and design, and the reactor vessel design must all be coordinated together from the initial concept of the reactor if a satisfactory removal procedure is to be obtained. The size and number of removal ports are dictated by the areas of the core that can be handled by the refueling tool and by the size and number of penetrations that can be made in the reactor vessel head without impairing its integrity. In order to minimize the number of ports, wherever practical, control drive housing ports are also used as ports for the removal of individual subassemblies. A brief description of one possible scheme for subassembly removal is given below.

The primary coolant system is first cooled and depressurized. Then the individual welded port is opened which covers the region of the core where the subassembly to be removed is located. A long tool is then inserted into the reactor through the port to pick up the desired subassembly. The tool is offset from the tool post and is indexed to pick up any subassembly within its design radius. The head of the tool is then remotely made to engage the subassembly, unlatch it, withdraw it straight up above the level of the other subassemblies and translate it to the center of the refueling port in the reactor vessel head, from which position it can be withdrawn from the vessel. The subassembly can then be handled either in a coffin or under water as mentioned above for the control element. The length of tool required, coupled with motions necessary to unlatch, withdraw from the core, translate and withdraw from the vessel, present many problems involving forces, moments and deflections on gear trains, arms and power transmitting shafts. To study these problems on the PWR refueling equipment, extensive mockup testing is being performed.

XVI. Thermal and Hydraulic Considerations

The first problem encountered in the thermal design of a reactor core is that of establishing the boiler steam conditions and the core heat load for the reactor plant.

The boiler steam conditions in the secondary system are closely related to the maximum operating temperature and pressure in the primary system. For a given set of steam conditions, the water temperature in the primary system must be sufficiently high to transfer the heat from the core fuel elements to the primary water and in turn to the boiler steam. In this connection, because of the size and complexity of high-pressure equipment and associated shielding in the primary system it is found that high steam pressures are not as significant in producing low cost power from pressurized water reactors, as in the case in conventional power plants. The steam pressure selected for PWR represents a balance between the desire for high steam performance and the cost of high-pressure primary system components.

Once the steam conditions have been set, the maximum steady state heat load requirements of the core can be established by conventional analyses.

General Criteria for Thermal and Hydraulic Design

In the thermal and hydraulic design of the reactor core, attention must be given not only to the steady state problems and operational transients but to foreseeable casualty conditions as well. The detailed considerations applying to these problems are discussed below. The steady state design of the core is based on limiting the fuel element surface temperature to the saturation temperature associated with the coolant operating pressure. By this means, boiling at steady state is avoided. It is considered that with the present state of knowledge, boiling should not be permitted during steady state operation; both local and bulk boiling present problems of pitting and crud deposi-

tions on fuel element surfaces; furthermore, bulk boiling introduces significant differences in pressure drop between the steam-containing hot channels and the cooler water-filled channels, leading to maldistributed flow, and possible burnout as well as control instability. It is expected that the extensive heat transfer programs presently under way will provide quantitative as well as qualitative answers to these questions by the end of the calendar year. The PWR core will take every advantage of new information thus obtained.

During a casualty such as loss-of-coolant-flow, however, boiling could be permitted so far as pitting is concerned because of the short time duration of the boiling. Reactor stability questions are not considered significant during such a transient if only local boiling occurs; flow, nuclear control stability, and burnout questions still remain, however, in the case of bulk boiling. In the final analysis the overriding question arising in a casualty such as the loss-of-coolant-flow is whether burnout will occur at the heat fluxes, flow and coolant conditions obtained during this casualty.

These considerations, plus the consideration of metallurgical properties of the fuel elements have led to three main criteria for the thermal and hydraulic design of the PWR core. These are as follows:

1. There shall be no local boiling at the hot spot of the core during steady state full power operation.
2. There shall be no bulk boiling in the water leaving the hottest channel during a loss-of-coolant-flow casualty to avoid any possibility of burnout.
3. The maximum internal fuel element temperature shall not exceed the permissible value based on the mechanical and metallurgical properties of the uranium-bearing material.

In applying these criteria to the design of a reactor, due attention must be given to the power distribution pattern and to the mechanical tolerances affecting heat flux, flow and pressure drop. These items are discussed in the ensuing paragraphs.

Steady State Thermal Design

In preparing the steady state thermal design of a core to meet the permissible maximum fuel element surface temperature, consideration must be given to the following factors:

1. Distribution of power density in the core obtained from neutron flux patterns.
2. Mechanical hot channel factors which express the influence of fabrication and operational tolerances on the heat transfer characteristics of the core.
3. The method of combining the variables listed in *a* and *b* above to establish core design parameters.
4. The number of coolant passes for which the core is to be designed.

The problems and methods of obtaining data for these items are discussed below. Careful integration of all these items is necessary in preparing the steady state design.

Distribution of Power Density in the Core

The power density distribution pattern in the core is established from neutron flux patterns obtained by nuclear analyses and tests. These neutron flux patterns are difficult to predict until the core is designed because of their complex relationships to the materials ratios in the core and reflector, and to the number, size and location of control elements. Preliminary analyses, nevertheless, must be made even during the initial design studies of the effect of core materials and rod configurations on the neutron flux patterns.

In a reactor containing both enriched and natural uranium, such as in the PWR core, the neutron flux pattern must be corrected to account for differences in the macroscopic fission cross sections of the two fuel materials in order to obtain the power density patterns. In such cases special attention must be given to the power density gradients between the seed and blanket so as to avoid unacceptable thermal stresses in fuel element subassemblies.

A typical plot of the radial power distribution in the PWR core is shown in Fig. 32. It can be noted that the power density is highest in the seed with a value of better than 200 watts/cm³. In the blanket the peak density is about 95 watts/cm³. These values reflect not only the gross neutron flux distribution, but also local neutron flux peaking due to water channels between subassemblies and channels vacated by control elements, as estimated from tests on critical mock-up assemblies. The radial peak-to-average power generation factors for PWR, as obtained from the curves, are 1.15 in the seed and 1.85 in the blanket.

In the axial direction the power density distribution for PWR is identical with the neutron flux distribution because the uranium enrichment is uniform along the axis. In Fig. 13 are shown typical axial

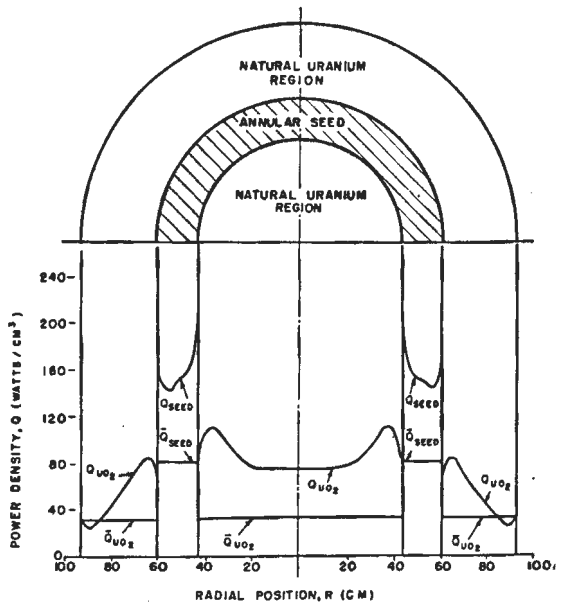

Figure 32. Radial distribution of power density for annular seed reactor with natural uranium in form of UO_2

power density distribution curves. As indicated in the figure, the peak power density is near the bottom ends of the control elements. These curves are normalized so that the average value is unity. Therefore, the peak-to-average ratios can be read directly from the curves as 1.90 for the seed and 1.68 for the blanket. In the blanket, an additional factor of 1.25 must be introduced to account for peaking near the ends of rod bundles, increasing the peak-to-average ratio from 1.68 to 2.10.

In the design of the core every effort is made to reduce neutron flux peaking by judicious selection of control element location, control element programming, and by the installation of low neutron absorbing spacers to alleviate local water-hole peaking. Critical experiments are used extensively for this purpose.

The power distribution factors must be coupled with mechanical hot channel factors arising out of fabrication and operational tolerances, in computing the maximum fuel element surface temperature.

Mechanical Hot Channel Factors

Fuel element fabrication tolerances when expressed as variations in the channel and fuel element dimensions imply respective variations in the heat transfer characteristics of the core. In addition, operational tolerances arising out of fuel element warpage and maldistributions in flow, imply further variations in heat transfer characteristics. Certain fuel element geometries such as, for example, rods in cross flow encounter uneven cooling, and this factor must also be included as part of the mechanical hot channel factors.

Extensive evaluation of pilot fabrication runs on fuel elements are required for determination of mechanical hot channel factors arising out of manufacturing tolerances. The warpage of fuel elements during operation can, in some cases, be estimated by analytical means and in other cases require experimental determinations. To determine factors for maldistribution of flow, a complete flow mock-up of the core and core entrance conditions is necessary. Analytical methods offer no assurance on the adequacy of flow distribution. Complete airflow mock-ups of cores in wood, plastic and aluminum have served as a practical method to establish good flow distribution and to determine the associated variations; testing is based on the use of air in the incompressible range.

The cooling characteristics around various fuel element geometries are best determined experimentally. They have been explored successfully by use of air flow around naphthalene models and by use of water flow around betanaphthol models employing the mass-transfer analogy to get the cooling characteristics.

All of these non-nuclear factors are grouped together as mechanical hot channel factors. There are three basic types of mechanical hot channel factors:

1. A factor, $F_{\Delta T}$, describing the peak-to-average temperature rise in the coolant.

2. A factor, F_{θ}, defining the peak-to-average temperature drop from the fuel element surface to the bulk coolant stream.

3. A factor, F_q, defining the peak-to-average heat flux in the core.

The preliminary estimates made for the PWR core prior to completion of conformatory tests are as follows:

Reactor region	$F_{\Delta T}$	F_{θ}	F_q
Highly enriched seed	1.40	2.03	1.31
UO_2 natural uranium	1.25	1.69	1.06

These values will have to be reviewed periodically as additional test data become available. The components of the mechanical hot channel factors used in the design of the PWR core are listed in Tables XVI-1 and XVI-2. The factors are obtained as the product of their respective components.

Table XVI-1. Components of Seed-Region Mechanical Hot Channel Factors

Component	$F_{\Delta T}$	F_{θ}	F_q
Flow distribution	1.07	1.06	1.00
Fuel plate warpage	1.04	1.03	1.00
Variations in thickness of uranium-bearing material	1.04	1.12	1.12
Variations in U^{235} concentration	1.015	1.025	1.025
Eccentricity of uranium-bearing material in the fuel plate	1.074	1.148	1.148
Variations in coolant flow channel thickness	1.114	1.13	1.00
Heat transfer correlation	1.00	1.25	1.00
Product	1.40	2.03	1.31

Table XVI-2. Components of Blanket Region Mechanical Hot Channel Factors

Component	$F_{\Delta T}$	F_{θ}	F_q
Flow distribution	1.07	1.06	1.00
Fuel rod bowing	1.02	1.00	1.00
Variations in fuel diameter	1.00	1.00	1.00
Variation in uranium concentration	1.03	1.05	1.05
Eccentricity of UO_2 in the fuel rod	1.00	1.01	1.01
Variations in coolant flow channel spacing	1.11	1.20	1.00
Heat transfer correlation	1.00	1.25	1.00
Product	1.25	1.69	1.06

Establishment of Core Parameters

Having selected the nuclear and mechanical hot channel factors, the steady state design can proceed by a parametric study of various core, fuel element, and power plant configurations to determine the optimum combination of core temperatures, heat exchanger temperatures, flows, pressure drops and pumping power to produce minimum cost per kilowatt-hour based on the permissible maximum fuel element surface temperature in the core. This fuel element surface temperature is related to the temperature of the coolant entering the core, the peak coolant temperature rise in the core, and the peak film temperature drop. These various values are in turn related to the core, fuel element and coolant channel dimensions, and to the coolant properties, to the core

heat output, to the number of coolant passes through the core and to the flow and pressure drop in the subassemblies. As a consequence, the process of steady state thermal design of the core proceeds as an iteration process whereby the core dimensions are varied in accordance with plant requirements and values for maximum fuel element surface temperature are computed for comparison with the limiting value. The variables entering into this process can be combined in a number of useful arrangements to facilitate the study.

Two important practical limitations enter into the exploration of the core parameters.

First is that of coolant velocity. While a high velocity is desired for good heat transfer purposes, the erosion problems and pumping power encountered above about 35 ft/sec make it impractical to consider velocities much above this value.

Second is that of internal fuel element characteristics. These characteristics involve: (*a*) the phase-transformation temperature of the uranium-bearing materials, and (*b*) the permissible fuel element thermal stresses. Acceptable values for these items are dependent on the particular fuel element materials and geometries used.

Number of Coolant Passes

In establishing the core parameters for a given application detailed consideration must be given to the number of coolant passes for which the core is to be designed. Generally, the question revolves about the use of one or two passes.

From a thermal standpoint a two-pass core usually offers two significant gains:

1. The water velocity through the core can be made as high as permissible by the considerations mentioned above without having a high total flow through the core, since only half the total cross section of the core is used for all flow during each pass.

2. The water passing through a hot channel in the first pass can be mixed with cooler water from the other channels of the first pass. Thus, the temperature of water entering the second pass hot channel is appreciably lower than if there had been no intermixing of the water between the passes — a condition representative of one-pass flow core.

In the design of a two-pass core a basic question to be decided is whether both passes are to involve upflow or whether down-flow is to be used in one of the passes. The resolution of this question involves consideration of the natural upward convection tendencies of heated water, and is especially significant in a loss-of-coolant-flow casualty. Down-flow tends to oppose this natural upward convection tendency of the water and could lead to earlier burnout by causing more rapid reduction in flow through the hot channel. Down-flow has the advantage, however, of permitting mechanical simplification of certain two-pass flow arrangements. So far, down-flow over fuel elements has been avoided because of its possible shortcomings during loss-of-coolant-flow casualties. As indicated previously, however, extensive heat transfer programs presently under way should give sufficient data to resolve this important question.

Transient Problems

An important factor in the over-all design of the core is determining its adequacy under operational transients as well as under foreseeable casualty conditions. Under operational transients must be evaluated the ability of the core to accommodate expected rates of load changes either up or down between 10% and 100% load. These evaluations involve changes in equilibrium xenon level in the core, changes in the average reactor temperature level with associated reactivity changes, changes in boiler steam conditions, changes in flow through the core, and time lags brought about by heat capacities in the core and system components.

Under foreseeable casualties, one casualty which greatly affects the design of the core and its protective devices is the loss of flow when the reactor is operating at substantial power. In this connection an important consideration is to design the core so that an appreciable portion of the coolant flow can be lost without resulting in a reactor scram.

If, for example, the flow of water is interrupted in a pressurized-water reactor, the core heats up as the flow slows down. If no corrective action is taken the water in the hot channel starts to boil, at first locally at the hot spot, then more extensively in the coolant of the hot channel. If the reactor is not shut down soon enough, there is finally formed over the fuel element in the hot channel a blanket of steam which reduces the heat transfer rate and causes rapid increases in fuel element temperature leading to local melting or burnout. One consideration is to obtain a design whereby there is adequate time for the sensing and scram circuitry to act. The times available are usually short being of the order of 1 second for various conditions. If conservatively too short a time is specified, it means that the accident must be sensed and corrected fast. This in turn means very fast response circuitry, and/or the scramming of the reactor as soon as a slight reduction in flow occurs; both of these are undesirable. On the other hand, if too much time is allowed before insertion of the control elements, some of the fuel elements may burn out.

In treating this problem, extensive data are required concerning the burnout characteristics of the selected fuel element geometries under a wide variety of heat fluxes, flows and subcooling or quality of steam at the point of burnout. In addition, because of the more severe loss of flow in the hot channel after steaming begins as compared with loss of flow in the average channel, detailed information is required on the pressure drops of two-phase water and steam systems. With these data, the equations describing the transient can be set up and solved with much tedious calculation. The problem becomes even more difficult, if during the transient, the nuclear constants vary sufficiently to affect the power output of the core during

the times under consideration. Such is usually the case with pressurized-water reactors utilizing strong negative temperature-power coefficients for control stability. In this work, analogue simulators are useful for simplified preliminary exploration and digital computors for more accurate treatment of the problem.

The results of preliminary analyses and tests for the complete loss-of-coolant-flow casualty on PWR are presented in Fig. 33. This figure shows permissible reactor power as a function of per cent flow based on steady state approximations. Four curves are shown: (*a*) power vs flow at which burnout might be expected based on preliminary test results from electrically-heated fuel elements; (*b*) power vs flow at which a steam quality of 40% is obtained at the exit of the hot channel; (*c*) power vs flow at which zero quality, zero subcooling is obtained at the exit from the hot channel, the target criterion for the PWR design; and (*d*) a cross plot of the expected power coastdown during the loss-of-coolant flow brought about by the negative temperature coefficient.

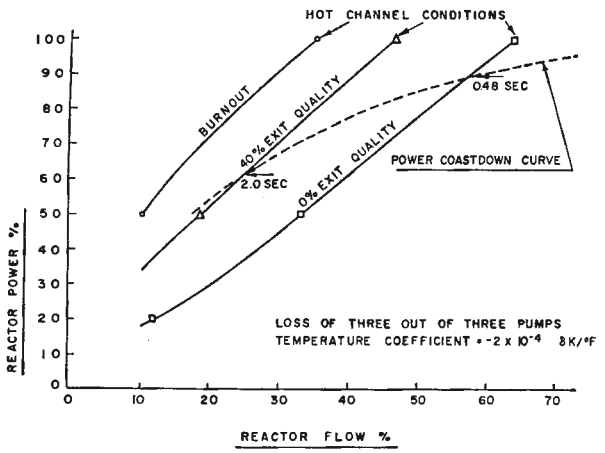

Figure 33. PWR hot channel conditions during loss-of-flow casualty

Indicated also, are times after loss of flow at which the various conditions would be achieved. From this curve it can be seen that if loss of pump power were to be followed by initiation of scram in 0.48 seconds, the basic design criterion of no bulk boiling out of the hot channel would be met. If scram were initiated at 2 seconds after loss of flow, the coolant leaving the hot channel would have a steam quality of about 40%, still expected but not yet proven to be below the expected burnout condition. Because of uncertainties in the burnout characteristics of the fuel elements and the power excursion of the reactor these curves are not yet considered to be quantitatively reliable, and require further confirmation. These curves do show, nevertheless, that if continuing tests confirm the extrapolation of present data, the seriousness of a loss-of-coolant-flow casualty on PWR may be mitigated.

Hydraulic Problems

Many of the hydraulic problems encountered in the design of the core have been implied throughout the description of the mechanical and thermal problems. Chief among these problems are those of flow distribution under a wide variety of operating conditions. The flow to the reactor vessel enters through a number of discrete nozzles with a jet action which, if permitted to feed the subassemblies directly, could cause severe maldistribution of flow through the core and impose large forces on the bottom of the core.

As pointed out in the discussion on mechanical hot channel factors, analytical methods are inadequate for defining the action to be taken in obtaining good flow distribution. Recourse again must be taken to testing. Considerable imagination must be exercised in establishing adequate tests without incurring excessive costs and unnecessary experimental difficulties. The use of air in the incompressible region for flow distribution tests on mock-ups of the core and reactor vessel permit great flexibility in testing and introduces only rather minor errors in the results. Correlation bench-tests are also necessary to establish the applicability of air to water distribution tests. From such mock-ups the proposed baffle designs for distributing flow can be tested and the final arrangement selected. The air flow core mock-ups also permit determination of hydraulic forces imposed on control elements by the flow passing outward from the core to the exit nozzles at the top of the core and furthermore, allow the testing of the suitability of shroud configurations for overcoming these forces.

Other hydraulic problems are encountered in establishing pressure losses at the entrances and exits of the subassemblies, pressure drops through the fuel element coolant channels and around the supporting structure, as well as establishing the required flow around specific points of high heat production in non-uranium bearing components such as around heavy structural segments and the tips of control elements.

In these cases as before, experimental techniques are found most applicable for determination of quantitative values.

Mock-ups and Tests

Because of the severe space and thermal limitations associated with the design of a compact power producing core, many of the mechanical and thermal features are new in concept and require mocking-up and testing prior to final fabrication. Some of these special tests have already been mentioned in describing the reactor mechanical and thermal design considerations. For example, as described above, the adequacy of the reactor vessel head and the handling tool are best established by use of model and full size mock-up tests. In order to fully indicate the magnitude of this work, listed below are some of the tests required to establish important design information to confirm the adequacy of reactor components.

1. Vibrational characteristics of the fuel elements and control elements.
2. Thermal distortion characteristics of control elements.
3. Stress concentrations in supporting structures.

4. Wear tests on reactor materials in hot water to establish reliable materials combinations for control element drive mechanisms under all types of possible operation and water chemistry.

5. Life tests on control-element drive mechanisms under all types of possible operating and water chemistry conditions.

6. Fatigue limits of new reactor materials at operating temperatures and pressures.

7. Strength and ductility tests on formed and welded components of fuel elements and subassemblies.

8. Crevice and fretting corrosion of mechanical fasteners.

9. The integrity of bolt-locking devices under the hydraulic vibratory loads.

10. The warping tendencies of fuel elements under severe temperature variations.

11. In-pile tests to determine the adequacy of fuel element shape and fuel element grouping under irradiation.

12. The effectiveness of shrouds on protecting control elements from hydraulic forces.

13. The fuel-element and water-flow-channel tolerances that can be maintained during fabrication for use in computation of reactor performance.

14. Flow distribution through the core, to thermal shields and to control elements.

15. The testing of core instrumentation mechanical, thermal and hydraulic features.

16. The buckling loads of control-element drive shafts under various conditions of guiding and alignment.

17. The adequacy of buffer mechanisms operating in hot water to snub the control elements under conditions of emergency insertion.

18. The form and adequacy of seal welds on the reactor head closure.

19. The piping stresses and moments on the reactor vessel.

20. The performance and adequacy of the closure means.

21. The procedures to be followed and special equipment to be used in correcting faults in the event that parts malfunction during operation.

22. The tool torques required to unlatch parts after exposure to long periods of times in hot water.

23. Deflections under load of the core-supporting structure and reactor-vessel head to determine their effect on control element channel alignments.

24. Natural convection flow characteristics through the core to determine the adequacy of core cooling when no power is available.

25. Burnout characteristics of various fuel-element geometries over the range of operating temperatures, pressures, flows, heat fluxes, sub-cooling and steam quality.

26. Pressure drop across subassemblies involving both single-phase and two-phase flow.

27. Tests to confirm the applicability of heat transfer correlations over the contemplated operating range of temperature and pressure.

28. Tests to determine uneven cooling characteristics involved with flow around different fuel element geometries.

In order to obtain certain of these data, a complete mock-up of the core serves best particularly in connection with the problems involving alignments, accessibility and interrelation of hydraulic and mechanical problems. Wherever possible, hydraulic forces, flow distributions, and uneven cooling around fuel element geometries are determined by use of air in the incompressible range. A complete airflow mock-up of the core in wood, plastic and aluminum has served a useful function not only in this connection, but also in demonstrating the mechanical feasibility of the design. In other cases, individual components or features of the components are tested separately. For example, burnout data has been obtained principally by the use of electrically heated fuel elements at operating pressure. To confirm these data, in-pile burnout experiments have been performed in a specially constructed high pressure loop in the AEC Materials Testing Reactor. It is desired to emphasize that at this stage of reactor development, such extensive testing is mandatory if successful reactors are to be obtained.

ACKNOWLEDGEMENT

The development and design work upon which this paper is based has been accomplished by many engineers and scientists from the various organizations participating in this project.

Preliminary Study of an Experimental Pressurized Heavy Water Reactor

By Odd Dahl,* Norway

It is to be expected that a country must deal with the problems and possibilities of nuclear energy according to conditions existing in that country.

Thus in Norway, our approach is influenced by the fact that we have energy reserves in undeveloped water power to cover expanding demand for at least 25 years and comparatively large, but not very rich deposits of uranium-containing shale. We operate a large merchant marine fleet, but we have no national sources of fossil fuel to speak of. Heavy water in quantity is produced in Norway, providing an incentive and a special opportunity to investigate heavy water reactors.

Although we say that water power electricity is cheap and plentiful, there is already a definite shortage in certain industrial districts. In such districts we feel that small unit nuclear power plants for electricity generation, or for the production of industrial steam, may be wanted rather soon.

Although we have to buy all of the fuel used in ships today, we may provide our own nuclear fuel in due time. Thus it is clearly of economic importance to study the prospects for ship propulsion by nuclear energy.

In this situation, we have tried to evaluate what we should do now and in the near future. We may forget about large nuclear power installations for a long time, but we believe that we should try to develop installations of 20–100 megawatt thermal output for selected industrial uses. As an output within this range is required for ships, such a program serves a dual purpose for much of the work involved.

As we acquired confidence in the field of reactor physics and engineering through the operations at the Kjeller Reactor Establishment, we gradually began to make plans for a new reactor installation, which would be a realistic step towards small unit power plants.

The economic and technological resources in a small country are small, and it can be argued that we try to go too far too soon. Many of us realize, however, that nationally speaking, we have repeatedly found ourselves in a position without sufficient knowledge to take full advantage of advanced technology. Norway, by the way, is very much an industrial country. Most of us believe nuclear energy is here to stay, and we want to develop knowledge inside our country in order to be able to judge progress intelligently, and to have bargaining power in the future.

I think we can all agree that the fuel elements represent the real problems in the design and construction of any reactor. I believe that nuclear fuel must to a large extent become an international commodity. Accepting that, one can go very far with reactor design and construction on the basis of just good engineering and physics, if we tolerate a certain conservatism and do not try to take full advantage of the very high concentration of energy in nuclear fuel.

Thus, both for shore and ship installations, an oversize reactor operating very conservative turbine machinery may well present an over-all highly efficient installation. The quest for maximum plant efficiency may not appear so pressing over a prolonged development period, considering the advantages of concentrated fuel. For plants intended to supply industry with low pressure steam, I see no pressing need for applying highly advanced reactor technology at all.

In Norway or in The Netherlands we have no opportunity to produce our own fuel elements now, but we have acquired a working knowledge of fuel metallurgy and chemistry, and we have for quite some time been accumulating a back-log of knowledge in neutron and reactor physics. We have also continuously been producing design studies for heavy water reactors, supported by experimentation in heat transfer and corrosion.

Last year we decided to co-ordinate our findings and to work out a specific proposal for an experimental power reactor to be realized as soon as possible. We had come to the conclusion that we could not have a good solution to problems without experimenting under conditions of high neutron flux at elevated temperature. Such conditions we could only obtain in a reactor operating at elevated temperature.

In establishing the basic specifications for a reactor study project, we had to accept certain limitations dictated by the state of our knowledge and resources. Thus we assumed that only aluminium and steel are available as construction materials, that we could have a limited quantity of heavy water, and no enriched fuel.

For a worth-while project we decided the reactor must be pressurized and that we should not be satisfied with less than 10 megawatt thermal output.

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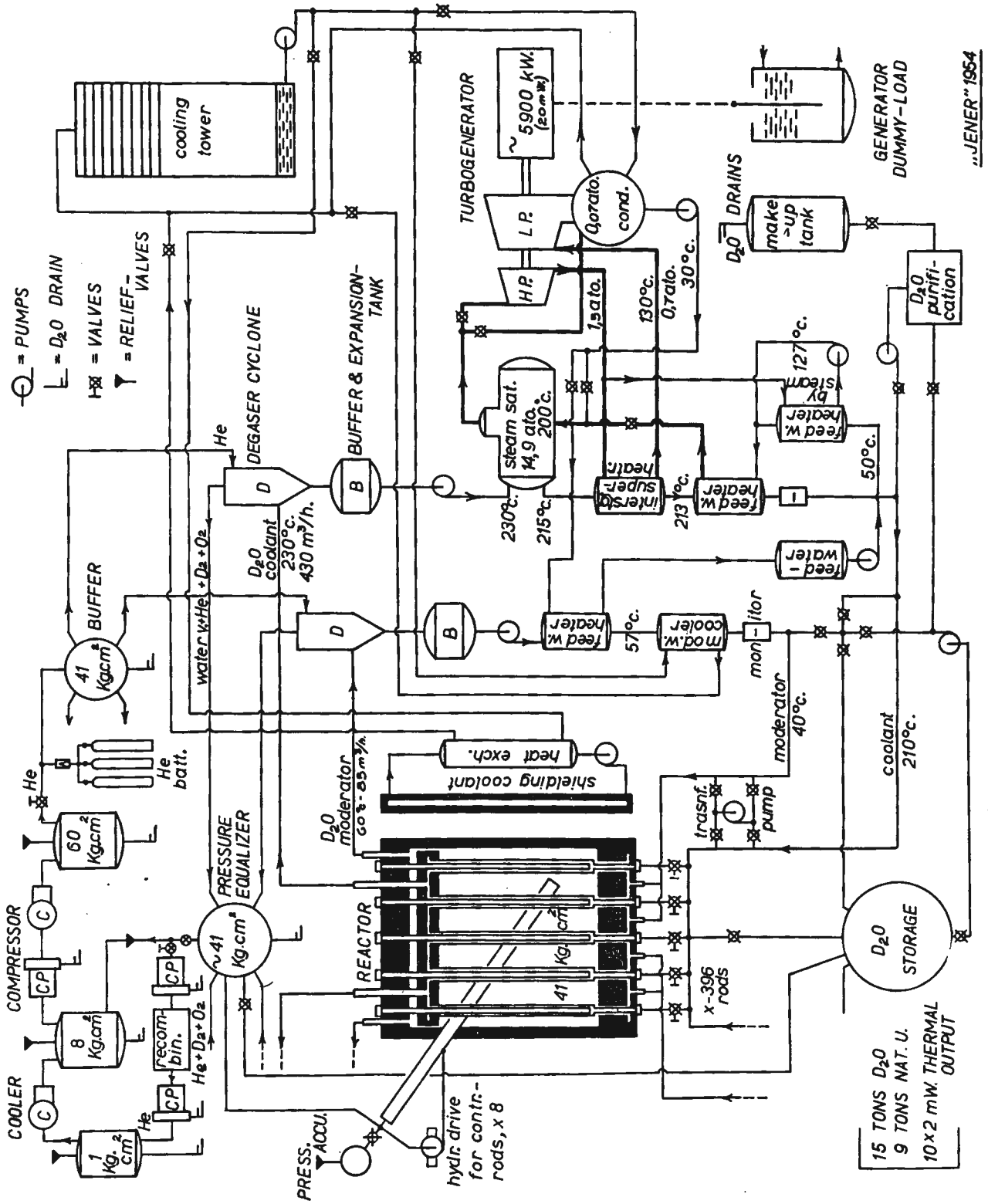


Figure 1. Flow sheet proposed for a pressurized, heavy water moderated and cooled reactor

As we had to contain the reactor in a thick walled pressure vessel, we could not make use of an external reflector.

An internal reflector of canned graphite or graphite in water or gas, or plain water, separated from the moderator by a thin walled Al tank, was considered, but our best engineering and physics judgement told us to leave the concept of a reflector, to use as much heavy water as we could reasonably have, and to distribute the fuel evenly throughout the complete volume.

We could count on 15 tons of water, and the physicists could show that this much water with about 9 tons of natural uranium distributed on an optimized 125 mm hexagonal lattice in rods containing a considerable quantity of neutron absorbing structural material would be critical, provided that the bulk of the moderator was maintained at about 50°C. Coolant water could be passed around the rods through tubes, and the temperature of this limited amount of water had little effect on reactivity.

Engineering studies were worked out on these assumptions in co-operation with the physicists and the chemists, and we can now write the following design and operations specifications for an experimental power reactor, using heavy water under pressure for moderator and coolant, and burning natural uranium, without reflector.

DESIGN SPECIFICATIONS

Free inside tank diameter	2700 mm
Free vertical height inside tank	2450 mm
Free diameter of a central column for rod experimentation	100 mm
Diameter of natural uranium fuel elements	25 mm
Thickness of aluminium canning	1.5 mm
Radial dimension of coolant annulus	3.5 mm
Thickness of aluminium coolant tubes	1.5 mm
Thickness of ceramics heat insulation outside coolant tubes, about	7 mm
Total length of uranium column	2450 mm
Lattice pitch, hexagonal	125 mm
Number of rods	396
Uranium weight	9000 kg
Heavy water inside tank, about	15,000 kg
Number of control rods	1
Number of shim rods	3
Number of cut off rods	4
Total cadmium surface for all rods	20,000 cm^2
Maximum design heat output	20,000 kw
Nominal tank pressure	41 kg/cm^2

OPERATIONS SPECIFICATION FOR 10,000 kw HEAT OUTPUT

Operating pressure in reactor tank	41 kg/cm^2
Coolant water inlet temperature	210°C
Coolant water outlet temperature	230°C
Total coolant flow	435 m^3/hr
Moderator water inlet temperature	40°C
Moderator water outlet temperature	60°C
Total moderator flow	70 m^3/hr
Power production in a central rod	56 kw
Maximum heat flux through fuel surface of a central rod	46 w/cm^2
Coolant velocity through a central rod	211 cm/sec
Power production in an outside rod	7.25 kw
Maximum heat flux through fuel surface of an outside rod	6 w/cm^2
Coolant velocity through an outside rod	27.5 cm/sec
Generator output, approximately	2500 kw
Expected excess reactivity in cold reactor	5%
Expected thermal neutron flux, peak	8×10^{12} $n/cm^2/sec$
Temperature difference, coolant-canning for a central rod	9.5°C
Temperature drop through canning for a central rod	3.5°C
Temperature difference, canning-uranium surface for a central rod	37°C
Temperature difference uranium surface-centre	110°C
Maximum temperature uranium centre	380°C

An outer rod has about 13% of the flux of a central rod.

GENERAL REMARKS

The reactor installation was to be planned so that one could make drastic changes in type and arrangement of fuel, dictated by possible developments before the reactor could be ready for operation.

One loop should be provided, permitting development work on advanced-design rods, and reactor control should include facilities for the development of possible new ideas for control.

The contemplated installation could be completed in three years, and serve our purposes for about three more years. We see no effective way of incorporating physics and isotope production facilities into an experimental power reactor.

A large scale technological experimental reactor must be overdesigned, and will contain many complications as compared to a prototype reactor. Our main concern is the fuel rods, but I do not think we will be able to produce a very good first design to be tested in the reactor. However, in order to study the behaviour of rods in an operating reactor, one must have a design which will permit control of conditions affecting rod performance. Coolant flows and representative temperatures must be known, and beginning rod failures must be detected at once. Frequent removals of the fuel rods for inspection are to be expected and the process of inserting or extracting rods should therefore be quick and as direct as the radiation hazard permits.

The control installation becomes elaborate because of the heavy instrumentation and the many inter-locks and safety devices required to deal with possible situations arising from the experimental nature of operations. Radiation monitoring should be given much attention as it is important for the planning of future

permanent installations to know much about radiation leakage through ducts and the shielding of service openings.

It must also be possible to extract components and give service under hot conditions to a higher degree than one would reasonably require in a permanent plant.

DESCRIPTION OF THE STUDY-REACTOR

The reactor tank is completely filled with moderator water. It has a flat detachable top and bottom.

The fuel rods are fitted pressure tight through the top lid and guided at the free ends in fittings passing through the bottom lid. Coolant water to each individual rod enters through the bottom fittings, passes through a jacket surrounding the canned fuel, and is collected from all rods in a space between ceramic layers fitted to the under side of the top lid. From this space the coolant water is passed on to the main heat exchanger and steam generator.

The hot coolant water is insulated from the colder moderator water by ceramics around the coolant jackets, and by the ceramic layers mentioned.

There is an axial free space in the reactor, and steel tubes, fitted in the centre of the lids, connect this axial space to the outside in such a way that loop experiments may be performed inside the reactor.

Reactivity control is achieved by narrow canned cadmium plates which may be moved hydraulically into or out of the lattice system, at an angle from above sufficient to have gravity fall-in in an emergency failure of the hydraulic system.

There are eight such stations, grouped in pairs high up on the circumference of the tank wall.

The tank hangs in a pit. The pit is closed by steel radiation shields flush with the floor in the reactor hall. All auxiliary equipment is located underground. We believe that an installation under the floor is cheaper and perhaps safer than a similar installation mounted mainly above floor level in a concrete block.

ROD DESIGN

The fuel element is to be a metallic uranium slug, 100 mm long and 25 mm in diameter. If it should be necessary to decide today about the properties and quality of the uranium, we would propose:

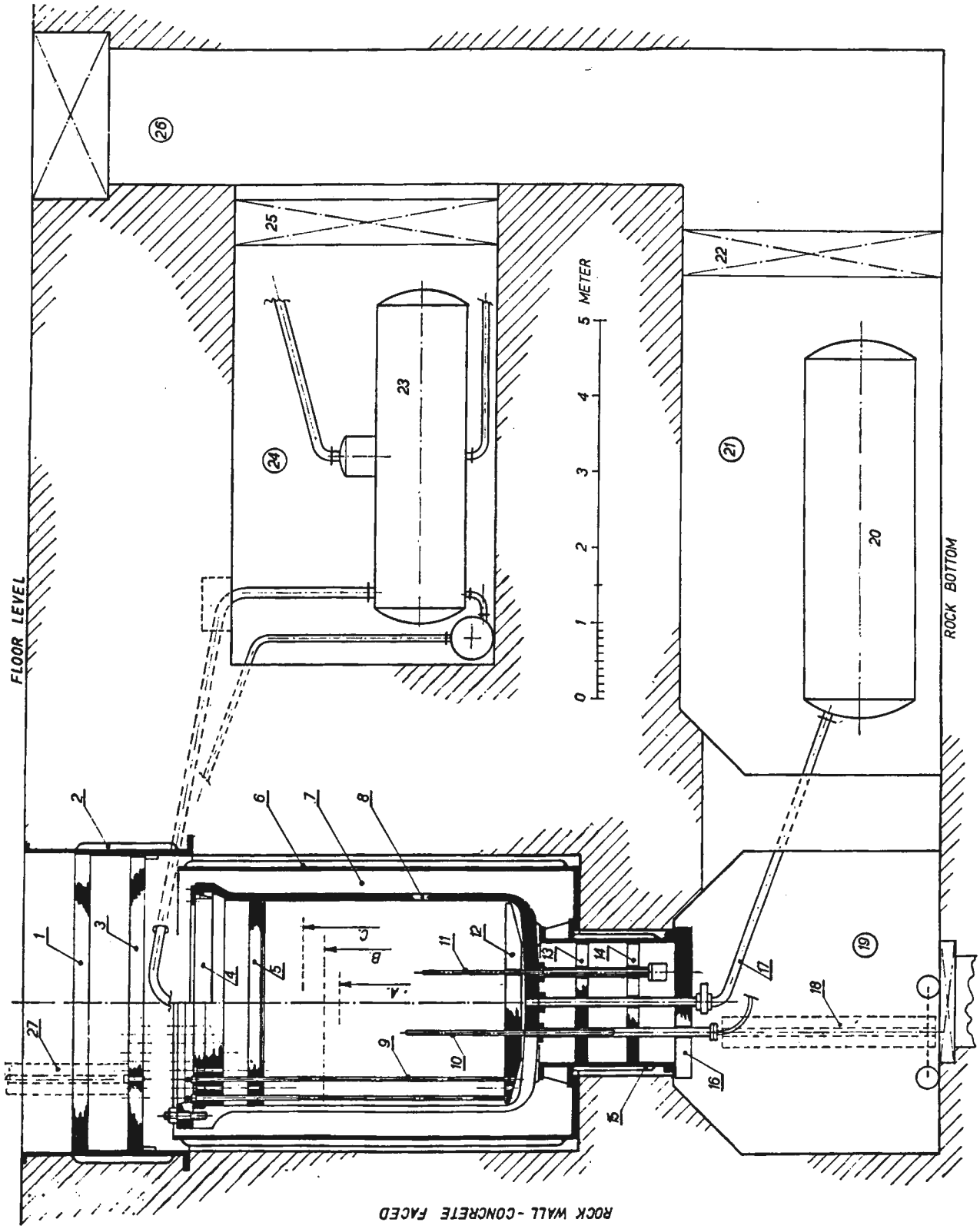
The metal cast in high purity graphite molds, remelted in vacuum to remove the hydrogen, cold rolled, heat treated in vacuum at 6000°C, and quenched from the gamma phase or high alpha phase.

We assume an Al alloy is canning material. It is known that diffusion may occur between U and Al at elevated temperatures, which in the long run may cause puncture of the canning. Until further research tells us otherwise, we plan to retard this process by separating the two surfaces by thin layers of U and Al oxides.

A good contact between U and Al is desirable for good heat transfer. If we start out with a stressed contact between the surfaces, we expect to have no contact after a few heating and cooling cycles because of the widely different physical properties of the two metals. We will therefore start out with a comparatively loose fit in the canning and depend upon the operating pressure in the reactor of 40 kg/cm^2 to provide a firm contact fit, but also assist heat transfer by a He atmosphere inside the canning.

The effect of "creep" in a long continuous canning due to the mechanical properties of U and the difference in thermal expansion between U and Al is con-

Figure 2 *(at right)*. Basic lay-out for a 10–20 megawatt pressurized D_2O boiling reactor installation. The complete installation is below floor level of room blasted in rock. Access to room through about 20 meter long horizontal tunnel. Control room located at entrance to tunnel. 1. Final radiation shield af 200 mm steel. 2. Water cooled cylindrical support for radiation shields. 3. Intermediate radiation shield with holes to pass the rods. The holes are closed by steel plugs. The shield remains in place and supports the coffin when extracting rods. 4. Reactor tank top lid with holes for seating the rods, 400 mm steel. Unit weight about 18 tons. The 30 kg/cm^2 pressure gasketing material on the circumference may be renewed without removing the lid. Two clamping down methods are indicated: (a) Clamping ring with bolts to take up the total pressure load. (b) Split ring in groove to take up the full pressure load on shear, with gas tight seal on top. 5. 180 mm steel in two layers to absorb gamma heat, which would cause severe heat stroins if absorbed in the top lid. 6. Steel cylinder with jacket to contain boron water. Cooling air is passed between cylinder and concrete face. 7. Heat insulating material, such as rock-wool or MgO, contained in the cylinder. 8. Reactor tank of steel at about 30 kg/cm^2 pressure, resting on bottom support. Unit weight about 37 tons. 9. 373 individual natural uranium rods on a 125 mm lattice system. Total weight of U about 8200 kg. Canning, 2 mm Al. 10. Emergency control rods. Sealed off cadmium lined thin-walled steel tube with positive buoyancy. 11. Hydraulically operated shim and control rods containing enriched uranium. (A total amount of about 25 stations for emergency, shim and control is probably required for the safe control of the about 9% excess reactivity in cold reactor.) 12. 180 mm steel disc to absorb gamma heat, and fitted with holes to guide the free ends of the rods. 13. 150 mm steel radiation shield. 14. 150 mm steel radiation shield. 15. Water cooled cylindrical support for reactor tank and radiation shields. 16. Final bottom radiation shield, 200 mm steel. 17. Heavy water drain tube to two storage tanks. 18. Coffin to facilitate the removal of hot control elements. 19. Room accessible when reactor is not operating. During operation, radiation and temperature will be at a too high level for occupancy. 20. Two storage tanks for heavy water. 21. Room for tanks and auxiliary equipment. 22. Brick wall erected after the installation is completed. 23. Main heat exchanger in two units, D_2O steam raising H_2O steam. Steam outlets in tank lid, 110 mm diameter × 2. 24. Room for heat exchangers and auxiliary equipment. 25. Brick wall erected after the installation is completed. 26. Shaft giving access to installations, of cross section to permit passage of tanks. Shaft is closed off at floor level with shield against secondary radiation. 27. Coffin for the removal of hot rods, handled by 50 ton traversing crane. A. Indicating the D_2O level at room temperature. B. Indicating the top level of the uranium in rods. C. Indicating the D_2O level at 230°C if boiling is prevented


FLOOR LEVEL
ROCK BOTTOM
ROCK WALL - CONCRETE FACED
0 1 2 3 4 5 METER
A
B
C
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sidered to be very serious. The accumulated effect may readily pull a fuel rod apart.

Short section fuel elements in the rods may tend to decrease the effect.

The full rod length canning which is supported by extruded fins inside the Al coolant tube to form one rigid unit, terminates in a screw-fitting with coolant outlet ports near the top and a slide-fit connection to the coolant inlet tube at the free end, which allows for longitudinal thermal expansion.

The coolant tube is surrounded by sections of ceramic material for heat insulation between coolant and moderator water. As insulating material, oxides of Be, Mg, Al, Si and Zr, produced with 60–70% pores have been under consideration. We have performed some experiments and find that it is very difficult to find a material which has both good physics and mechanical properties and is able to withstand temperature and pressure cycling, and radiation.

An outstanding requirement for good rod construction is that the insulation must not break off and cause the rod to stick. In designing fittings for such a rod it is of equal importance to remember that the rods must not stick due to swelling or corrosion, as one stuck or broken rod constitutes a serious problem.

We have produced a great variety of design studies for a rod, none of them are very satisfactory. The problem is accentuated by the necessity for thermocouples inside the rod and a device for canning-leak detection.

EXTRACTION AND UTILIZATION OF HEAT ENERGY FROM THE STUDY-REACTOR

The complex of problems may be studied under the three headings: (1) heat extraction from the rods, (2) heat extraction from the tank unit, and (3) utilization of the heat.

A fair amount of theoretical and experimental work on heat extraction from rods has been done as indicated. The salient resulting figures are listed under specification. We are reasonably convinced that we may extract 10 megawatts of heat from our study-reactor without great difficulties, but have no clear concept as to how long we could continue to do so.

Not all the heat developed in the reactor can be carried off by the coolant water. There will be an unavoidable interchange between moderator and coolant water. The moderator and the tank will absorb radiation energy, and there will be heat-losses by conduction from coolant to moderator and to construction material.

Assuming the reactor operating at 10 megawatts thermal output from the rods, our findings show that about 1.5 megawatts are lost to the moderator volume and tank structure, and must be removed at moderator temperature.

How best to utilize the heat from a reactor to run a turbo generator may be regarded as conventional engineering, but in the present case one would also like to provide the turbine with the 1.5 megawatts of low temperature heat energy. Our conclusion is that the best power cycle for the study-reactor should include feed water pre-heating by moderator water and by steam extracted at an intermediate stage during expansion in the turbine, and make use of interstage super heating only.

Over-all efficiency studies indicate that, out of 10 megawatts total heat output, close to 2.5 megawatts of electricity may be taken off the generator.

There are unconventional circuits and equipment which must be incorporated in the flow sheet for a reactor installation. Thus, during operation the heavy water will partly decompose into a mixture of D_2 and O_2 which must be recombined in order to preserve the water and to reduce the danger from explosive mixtures. For the last reason, a blanketing gas of helium or carbon dioxide should also be maintained over all free surfaces in the D_2O circuit. A purification unit is required in this circuit to free the water from admixed impurities, mainly in the form of aluminium and iron oxides.

It is also necessary to raise the pressure in the tank above the steam equilibrium pressure to prevent surface boiling on the rods, and expansion volumes are needed outside the tank as the tank is completely filled with water at the starting temperature.

For good heat exchange the heavy water should be freed from absorbed gases in de-gasser units before it is passed through the exchangers.

Finally, a storage tank for the entire quantity of D_2O is required.

The proposal for an experimental power reactor as outlined may appear fair enough, but a closer study reveals technical problems in addition to those already mentioned for which it is rather difficult to find good engineering solutions. Most of these are associated with the requirement for keeping the moderator water and the coolant water separated at about 170°C difference in temperature.

This gives rise to severe thermal stresses throughout the reactor unit. The heat insulation problems are also aggravated because the nuclear physics conditions must be fulfilled.

Although we feel that we could cope with the difficulties, we now feel that a project as outlined would be too much of a dead-end venture as a design contribution, realizing that only a slight enrichment of the fuel will permit operation at uniform temperature and thus circumvent the present difficulties.

Recent events, culminating in this conference, made it reasonable to assume that we could have some enriched fuel for a reactor. With this freedom of conception we set about to see how our reactor would behave with the rods stripped down to the bare canning in a hot moderator, which brought us on to the boiling reactor principle.

For some time we had been aware of the possibilities connected with the boiling reactor principle, and encouraging reports from USA made us want to investigate how our study-reactor would behave if we

permitted it to boil with stripped rods. We found that stripped rods in a moderator boiling at about 230°C left us with about the same neutron situation as we had for the cold moderator, hot coolant-flow version.

I understand that it is difficult to make accurate reactivity calculations on an engineered design for a reactor, and our physicists are not satisfied that our boiling reactor would remain critical at high temperature with natural uranium only. If we saw no way of obtaining some enriched fuel, we probably would have regarded a boiling reactor project as too hazardous. As it is we expect to have some such fuel, so that we will be able to bring the reactor to criticality at the temperature we want.

In the study of heat transfer in a boiling reactor we find that, if we can extract 10 megawatts by water at 230°C from the cold moderator, hot-coolant reactor, we might extract 15 megawatts as 230°C D_2O steam from a boiling reactor, assuming similar geometry and fuel canning.

If a boiler engineer had a free hand in proposing a boiling reactor design, I believe his first concern would be to ensure a positive up-draft of water throughout the volume occupied by the rods in order to break up opposing turbulent currents which may cause rods to overheat. He would probably find a solution by using much water in the tank, by using directing tubes or funnels, or by circulating the moderator water violently. We can visualize a beautiful boiling reactor with full command of degree of enrichment, type and shape of fuel elements, but we intend to proceed under the restrictions already mentioned, that we must rely upon natural fuel and use heavy water, of which we can have about 15 tons.

Looking at a revised project, we are satisfied that we can build a 10-megawatt pressurized boiling reactor, with good hope to extend the operating level to 20 megawatts after some experimentation on the first installation.

The reactor would produce 10-20 tons of 230°C D_2O steam per hour. Circumstances are such that we may install a plant on an industrial site and sell steam on the site if and when we can deliver, but with full opportunity to run the reactor for our own purpose.

A boiling reactor design programme is now under way to the following leading specifications:

Free inside tank diameter	2700 mm
Operating pressure, about	30 kg/cm²
Length of uranium column	2400 mm
Heavy water inside tank, about	14,000 kg
Uranium weight, about	8200 kg
Fuel element diameter	27 mm
Central bore in fuel element	10 mm
Length of fuel element	100 mm
Number of fuel rods on 125 mm lattice	373

A satisfactory design depends very much upon how well we can deal with the hydro-thermodynamic condition in the given tank, as the moderator volume must in the main be filled with rods, and we hardly have enough heavy water to establish a clear up and down stream condition in the tank. We find that good experiments in the laboratory to study the condition amounts to a large undertaking, and we will rather regard the running reactor as our real experiment.

The fuel rods may be very simple for a boiling reactor. We intend to sheathe the fuel elements in aluminium alloy tubes, support them in the tank lid and guide them in the bottom lid, and watch their performance. We expect to find more advanced solutions for placing a charge of fuel in the reactor, and for this purpose the lid of the tank may be removed in hot condition and replaced with one adapted for other shapes of fuel.

It has given us some concern, how best to site a pressurized reactor, as we must assume that the reactor tank may blow up. As we want to take all reasonable precautions to deal with such a situation, we find that it becomes very expensive to put up a building on free land.

The best solution for us appears to be to blast space for the installation into a granite mountain side, where we could contain an explosion.

ACKNOWLEDGEMENT

This review is based upon the findings by the joint Dutch-Norwegian staff at the Kjeller Reactor Establishment and a group in the Chr. Michelsens Institute in Bergen. Quite a few persons should be named as important contributors to the studies, if any at all. Under the circumstances I find it prudent to mention none, and express my thanks to colleagues in the Establishment and at my Institute for the opportunity to present our material.

The Engineering Design of a Prototype Boiling Water Reactor Power Plant

By J. M. Harrer, A. S. Jameson and J. M. West,* USA

INTRODUCTION

The Argonne National Laboratory is designing and building the Experimental Boiling Water Reactor (EBWR) as part of the United States Atomic Energy Commission's five-year program for development of power reactors. The planned power of EBWR is 20,000 kw of heat. It will employ natural circulation of water on a scale which can be extrapolated to full-scale power plants for central station use.

The construction advantages of the direct boiling cycle reactor for power generation may be seen from Fig. 1 upon comparing the boiling and the sub-cooled, water-moderated type of reactor plant. In order to have a turbine pressure of 600 psig (42 atm), the sub-cooled reactor pressure and all of the primary system piping, pumping and heat exchange equipment must operate at about 2000 psig (138 atm), whereas in the direct boiling cycle, plant equipment operates directly at the lower value of 600 psig (42 atm). Further, the feedpump is smaller and less expensive, and a heat exchanger is not required.

BACKGROUND DATA

The design of the EBWR is based on limited experience and data from two sources. The first is the Borax reactor: a small, fully enriched core, having aluminum-clad, aluminum-U^{235} alloy fuel plates, which was operated at the Reactor Testing Station in Idaho; the second is a series of boiling density tests with electrically heated simulated fuel channels set up in the Laboratory.

The Borax fuel plates were spaced evenly in vertical channels to allow natural circulation of the water coolant. A steady, stable output was obtained from the reactor at power densities as high as 38 kw/liter (1080 kw/ft^3) of coolant at an operating pressure of 300 psig (21 atm). The inherent safety resulting from water expulsion and the control by steam void formation at significant power levels were both ably demonstrated by Borax and used as starting points for design of the EBWR core.

A fully enriched U^{235} reactor core is not suitable as a central station power source because: (1) the cost of operation when using U^{235} would not be competitive with coal- or oil-fired plants, and (2) a large, fully enriched core would have a reactivity increase when water was expelled and, therefore, would be comparatively unsafe. To fulfill its role as a full-scale plant prototype, EBWR must have a core of natural uranium as would full-scale direct boiling nuclear reactor power plants.

REACTOR DESIGN

A partially enriched reactor core can either gain or lose reactivity as a result of water expulsion. The water-to-uranium ratio and the core size are the determining factors. Critical-mass studies show that to be certain of a negative void coefficient, the water-to-uranium volume ratio must not exceed about 2:1 for large plants. For a smaller reactor, such as EBWR, this ratio can be increased. The ratio is 2.5:1 for EBWR. Adding the enriched elements increases this to 3.6:1.

The selected core volume is based on a nominal power density of 25 kw/liter (706 kw/ft^3) of coolant. For a total heat output of 20,000 kw, the required core size is 4 ft (1.22 meter) in diameter and 4 ft (1.22 meter) in length. In line with the experimental nature of EBWR, provision is made to vary the diameter up to 5 ft (1.52 meter), although the 4-ft (1.22 meter) core diameter is expected to be ample.

The enrichment required for criticality will vary with the diameter. To provide for this, two types of fuel elements are used: type A contains natural uranium; type B contains U^{235} diluted with zirconium. Type B will have the same heat capacity and fission rate as the natural uranium elements. The total U^{235} content of the core can be varied if a sufficient number of type A and type B elements are on hand. The 4-ft (1.22 meter) reference core requires 77 natural assemblies, about 4536 kg (5 short ton) of natural uranium, and 35 enriched assemblies, about 19 kg (42 lb) of U^{235}.

Both elements are structurally similar and are completely interchangeable in the core (Fig. 2). Each fuel plate, including the Zircaloy-II cladding, is 0.28 in. (0.71 cm) thick. Six fuel plates are edge welded to Zircaloy-II side plates in such a manner as to leave coolant channels 0.37 in. (0.94 cm) wide.

*Argonne National Laboratory.

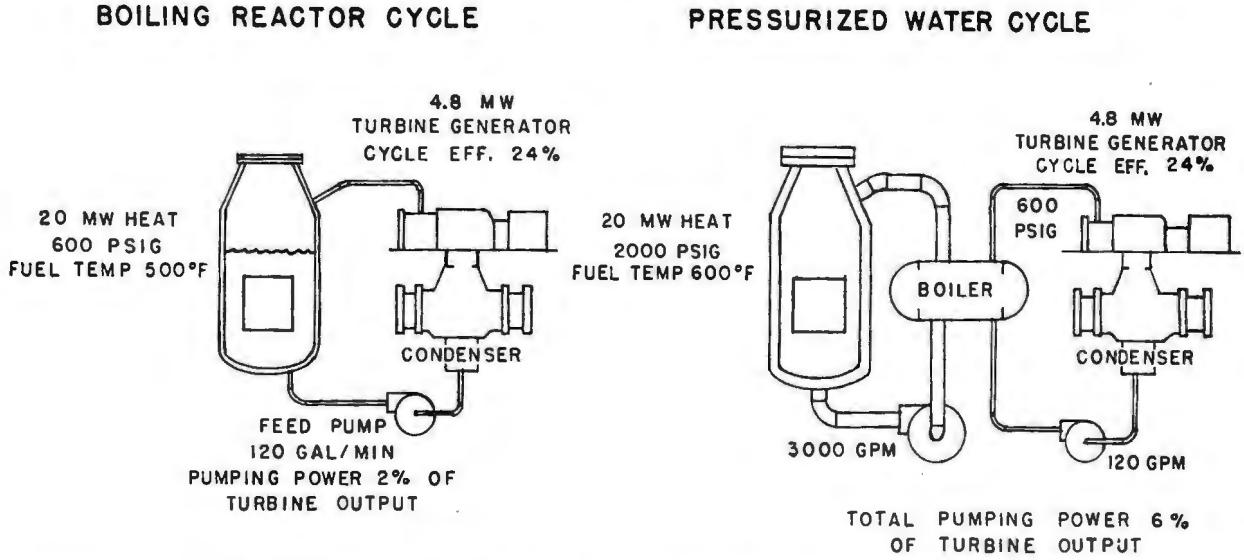


Figure 1. Comparison of the flow cycle of a boiling reactor with a pressurized water reactor

Perforations in the side plate make this support less rigid and thus allow for fuel plate expansion. A hollow socket centers the fuel element holes at the bottom of core structure. Cooling water enters the element at this point. Leaf springs at the top hold the adjacent elements apart in the core.

The over-all conversion ratio for the core is about 0.7. The specific conversion ratio for the natural uranium elements alone is approximately unity. The enriched elements are replaced when 40% of the fuel is burned out. The average heat output of the enriched elements is 37% of the total reactor heat.

With a constant ratio of natural to enriched elements a slightly negative void coefficient exists at room temperature, but it becomes more strongly negative at the 488°F (252°C) operating temperature, as shown in Fig. 3. The slope of the reactivity versus void curve decreases as the enriched core diameter is increased. For the 4-ft (1.22 meter) core, 3.5% reactivity is lost between 68°F (20°C) and 488°F (252°C). Of this, a 1% loss results from variations in cross section and a 2.5% loss results from the 20% change in the density of water. A further decrease of 2.2% results when the voids change from zero to 17.5% at 488°F (252°C) to obtain 20,000 kw output.

The actual loading of EBWR will be determined experimentally. A small negative void or density coefficient of reactivity near 68°F (20°C) is desired so that too much reactivity is not tied up in voids at the operating power. The void coefficient of reactivity increases substantially as the temperature is raised. Safer operation is obtained by pre-heating reactor water to 325°F (163°C) with an external heat source. This makes the inherent shutdown mechanism of void formation more effective.

A horizontal cross section through the core is shown in Fig. 4. Fuel assemblies are tightly packed to avoid excessive hot spots. A nominal 4-in. (10.16 cm) square space is assigned to each element. Nine fuel assemblies are contained in the 12-in. (30.5 cm) square cells between the control rod guides, which

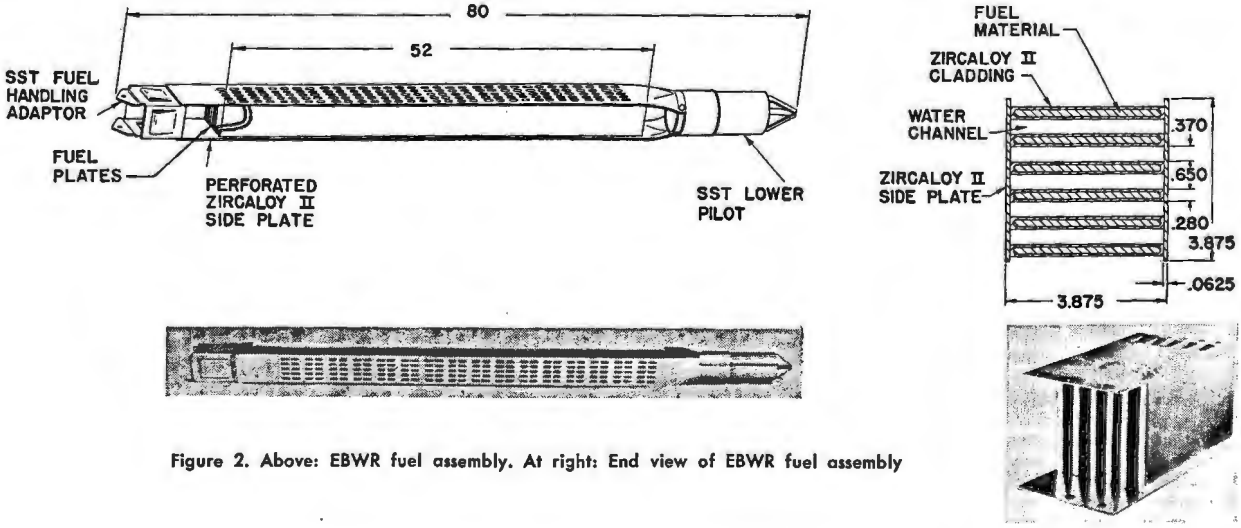


Figure 2. Above: EBWR fuel assembly. At right: End view of EBWR fuel assembly

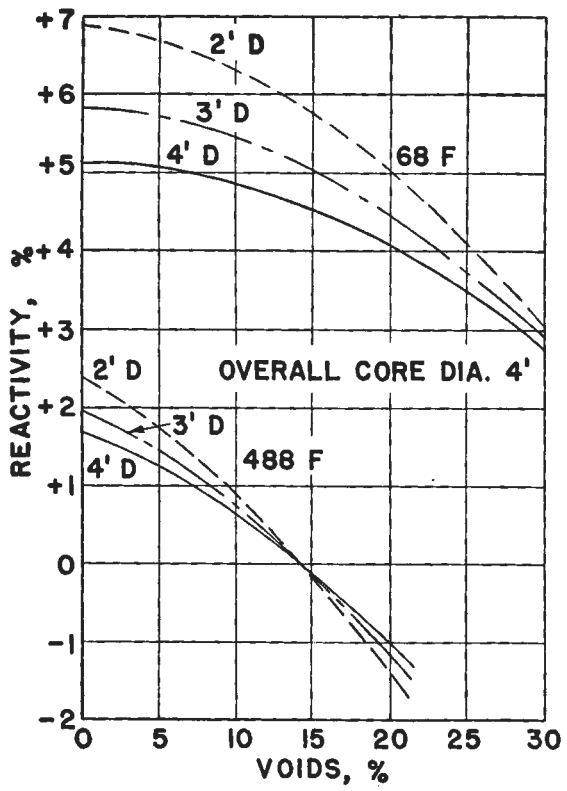


Figure 3. Reactivity vs void for various diameters of enriched zone

are made of Zircaloy-II. They provide ½-in. (1.27 cm) spaces for the insertion of hafnium cruciform control rods, ⅛ in. (0.32 cm) thick.

Nine rods are used, giving 12.5% of reactivity control. The rods are driven from mechanisms located under the core. As shown in Fig. 5, the hafnium rod section is attached to a zirconium cruciform follower of equal size to reduce the hot spots which would result from water-filled control rod channels. When the hafnium sections are fully above the core the follower occupies 75% of the core height.

The core is centered in a pressure vessel with a 7-ft (2.14 meter) inside diameter. The 1½-ft (45.7 cm) wide annulus remaining with a 4-ft (1.22 meter) core acts as a downcomer for natural circulation of water. The support structure and general arrangement are shown on Fig. 6. Feed-water is injected at about 110°F (43°C) through a ring above the downcomer. This helps quench steam bubbles entrained with water at this point. The downcomer water is sub-cooled about 11°F (6°C) as it enters the fuel assemblies through the hollow sockets in the base of the fuel elements. Boiling begins about 33% above the bottom of the fuel element.

The average steam content of the fuel channels proper at 20,000 kw is 20%. The average voids for the entire moderator amount to 17.5%. The change in reactivity attributed directly to steam voids is referred to as per cent reactivity excess, i.e., the value above zero void criticality at the operating water and steam temperature. The variation in power output with per cent reactivity excess is shown in Fig. 7. An increase in power can only be accomplished by moving control rods in such a manner as to increase the reactivity excess, which, in turn, is offset by the increase in steam voids when more power is generated. For every rod setting, a fixed power, dependent upon poison buildup and fuel depletion, results.

The average heat flux at the fuel surfaces is 40,000 Btu/hr/ft² (3 gm-cal/cm²/sec). Hot spots at control rod channels or at new fully enriched fuel elements may reach a flux as high as 150,000 Btu/hr/ft² (11 gm-cal/cm²/sec). For the conditions of flow and density of EBWR, boiling tests using electrically heated simulated fuel channels indicated that the burnout heat flux is 600,000 Btu/hr/ft² (45 gm-cal/cm²/sec) or more.

An increase in heat output can be obtained by forced circulation. In line with the experimental nature of the plant, forced-circulation pipes are installed through the shield for future use, as shown in Fig. 6. Water would enter through 6-in. (15.2 cm) diameter lines below the core and leave through the 12-in. (30.4 cm) diameter lines at the side of the reactor vessel. A shroud would also be placed around the core and bottom grid to direct the water upward through the core.

A cylindrical section of 1-in. (2.54 cm) thick stainless steel is used as a thermal shield for the pressure vessel. It is located in the region around the core to reduce thermal stress by absorbing gamma radiation. A lighter shield is installed above the water line to prevent thermal shock to the pressure vessel due to accidental introduction of cold water.

The steam dome is 7 ft (2.12 meter) high and the steam velocity in the dome is about ½ ft/sec (15.2 cm/sec). This low velocity arrangement provides a very good steam-water separation mechanism. The reduction in radioactivity between steam and water is expected to be of the order of 10^4.

Protection is provided against exposure of the core as the result of an accidental loss of water. If not immersed, the core may melt due to residual heat in the fuel. A ring above the core provides a spray of water which will cool the elements. The water is passed over boric acid crystals, so that even if the control rods are not inserted, the nuclear reaction will not start as the water enters the core.

Shutdown in the case of control rod failure when the reactor is at full pressure is also provided. Water containing boric acid stored in a tank at 1600 psi (109 atm) can be injected into reactor water below the core to obtain a quick shutdown effect.

The control rods are operated through labyrinth seals on the lower ends of thimbles attached beneath the pressure vessel. The seal operates on the pressure breakdown principle. To prevent water leakage an atmospheric gland seal is added. Leakage between the seals is collected and re-injected to the water system. About 2% of the feedwater is intro-

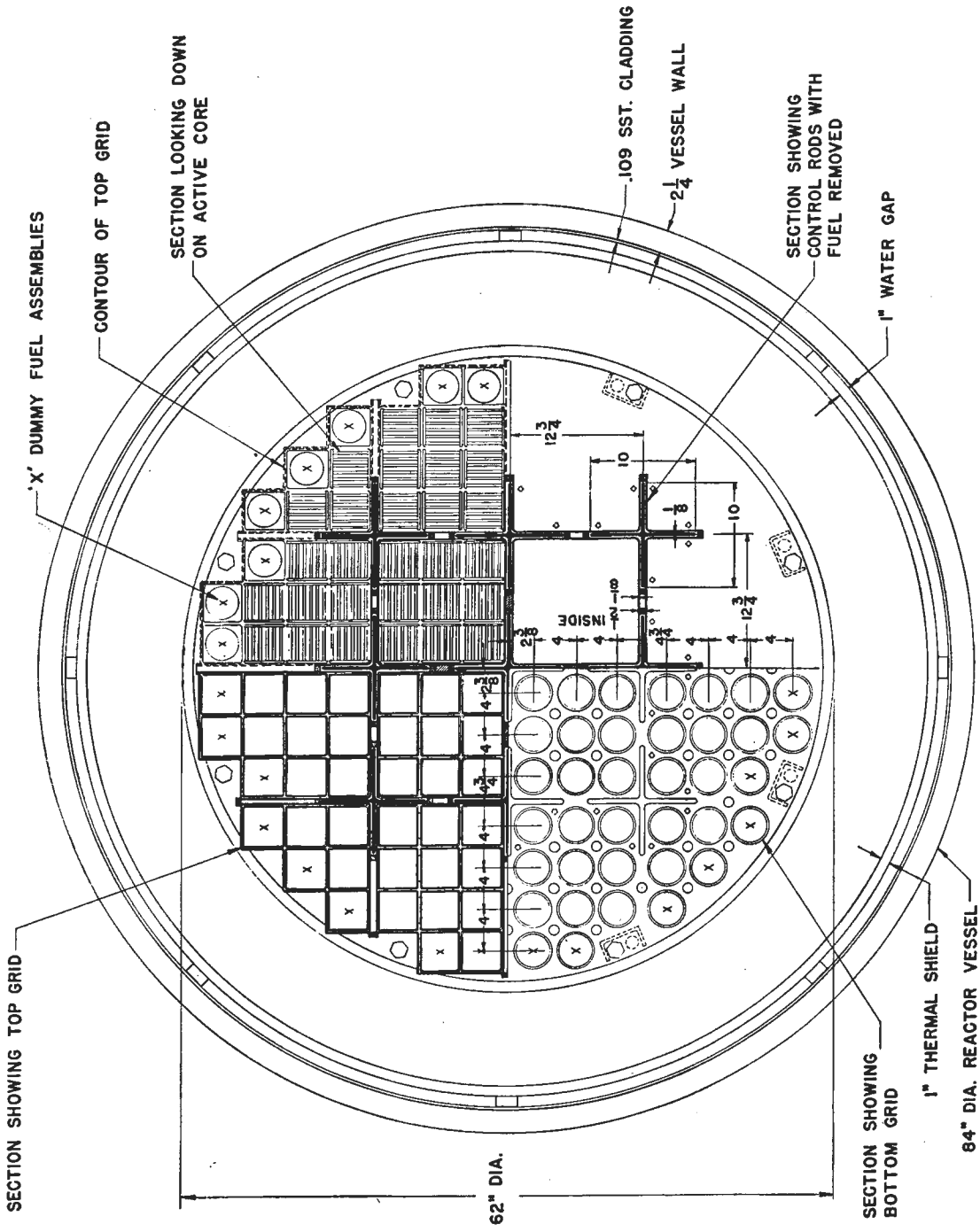


Figure 4. Horizontal sections of EBWR

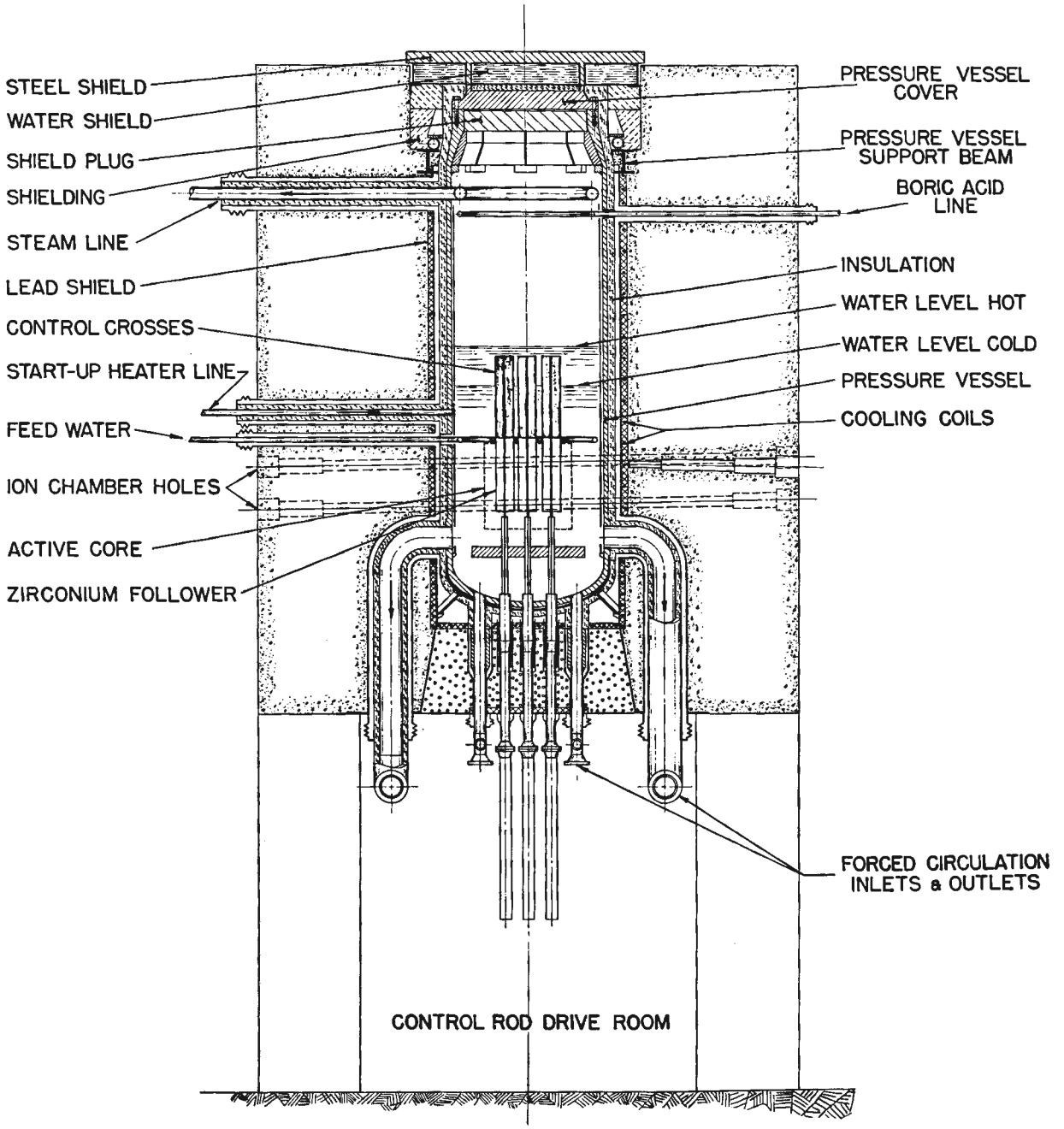


Figure 5. Vertical section of EBWR

duced at the lower end of the thimbles to cool both the seals and the thimbles. Adequate shielding is placed around the thimbles to allow maintenance and inspection of the drive mechanisms.

The drive mechanisms are screw jacks on which the nut travels. A 1-in. (2.54 cm) diameter drive rod is held to the nut by an electromagnetically operated latch. Release of the rod and rapid insertion for reactor shutdown is designed to take place in ½ sec. Insertion of the rods is assisted by springs which develop about 1000 lb of force. The springs are effective for about 4 in. (10 cm). Their principal function is to get the rod into motion quickly. Both gravity and the force from the pressure in the reactor vessel assist in driving the control rods into the reactor. An external dash pot decelerates the rod in the last 5 in. (12.7 cm) of insertion.

The reactor shield proper consists of 3 in. (7.6 cm) of water-cooled lead and 7½ ft (2.28 meter) of concrete. Steam and water lines penetrate this shield. Bending the lines prevents the escape of radiation. The shield concrete adjacent to the lines will be either air cooled or water cooled.

The top opening of the reactor vessel is large enough to allow removal of the entire core structure along with the bottom grid so that a replacement core

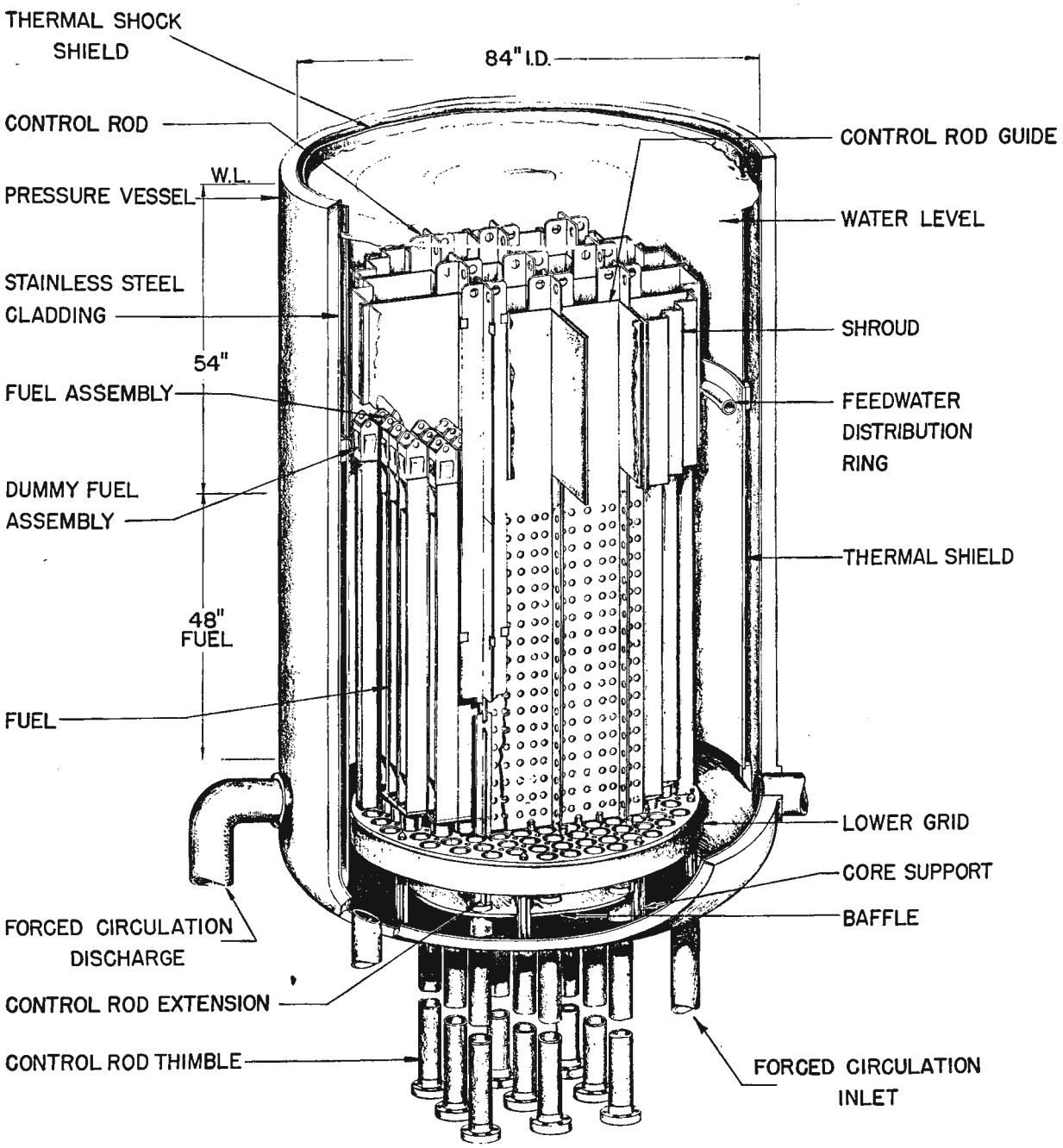


Figure 6. Perspective of EBWR

designed for heavy water operation can be inserted. Possible conversion to D_2O makes designing for a very low net leakage from the primary system equipment mandatory. The method of controlling net leakage is discussed later.

Normal access to the core elements for removal and installation is through a rotating shield plug which contains a smaller eccentric plug. This combination is located at the top of the pressure vessel. The center plug of the concentric pair contains two holes: one is for handling and one is for viewing the handling operation. By rotating the two plugs together and then one within the other the access holes can be centered over any fuel element. Filling the steam dome with cold water provides shielding for viewing and handling. A coffin will be used to transfer spent fuel to a water-filled storage well near the reactor.

Control rods are designed for installation and removal from the top. Sockets above each rod thimble assist in centering the rods and also prevent the rods from falling below the core.

Should an explosion occur in the reactor vessel, the fragments would be directed downward by virtue

of the design of the structure around the reactor. Heavy members are provided above and lighter sections below the reactor location. The portion of the building below the main floor is lined with a 2-ft (61 cm) thick concrete wall inside of the building shell to protect the shell from fragments and to provide the strength necessary for the deep excavation.

STEAM PLANT DESIGN

The arrangement and selection of equipment and construction materials were influenced by a desire to minimize the quantity of corrosion products which could enter the reactor and the possible steam-entrained radioactivity which would result. A schematic diagram of the major equipment in the steam and feedwater systems is shown in Fig. 8.

Makeup feedwater is demineralized in a mixed bed ion exchanger and stored in a 200-gallon (750 liter) tank. Since net leakage of water will be held to a minimum, the introduction of water to the system will not be more than 10 gal/min (0.63 liter/sec). This makeup water enters at the condenser hotwell, where it is degassed. In this way the oxygen concentration in the feedwater is held below 0.005 cm^3/liter.

The feedwater pump is of chrome steel construction. All the feedwater system piping is stainless steel. The air ejector condensers are cooled by the feedwater. A minimum circulation of 75 gal/min (5 liters/sec) of feedwater is required to cool the condensers. A recirculation line (not shown) conducts the feedwater back to the main condenser when the water flow to the reactor is below the minimum requirement for cooling the air ejector condensers.

The feedwater is passed through a cotton fiber filter before it enters the reactor tank. The flow of feedwater is controlled by an electrically operated valve. The level of water in the reactor is measured by a differential pressure type level instrument.

The deviation of reactor water level from a preset level causes the adjustment of the valve opening. The water level is controlled within about 1 in. (2.54 cm). The design of the feedwater control system is not critical; operation will be similar to that in coal-fired plants. The important consideration is not to change the water flow rapidly because this could result in a rapid positive change in reactivity. A 3-phase ac electrical motor valve drive is particularly well suited to limiting the maximum speed of the valve in opening or closing.

Level measurement in addition to the control instrument includes a pair of floats to show if the water is at a high or low limit. To facilitate maintenance, a 6-in. diameter stand pipe is installed outside of the reactor shield and attached to the reactor vessel at the steam dome and below the water level. To this is attached the solid type floats which are sealed into pressure tight housings and operate switches through bellows. In addition a sight glass is installed to cover about 5 ft (152 cm) of water level. Viewing of the glass is by closed-circuit television.

The water in the reactor is recirculated at a low rate through mixed-bed ion exchangers designed to maintain the dissolved solids concentration at not greater than 1 ppm.

Steam leaving the reactor enters a steam dryer, from which condensed water can flow back to the reactor. The separation of moisture at this point reduces carryover of radioactivity to the steam system.

The steam dryer serves an additional function. A cooling coil, which is normally dry, is included. This coil connects to a 15,000-gallon (57,000 liter) overhead storage tank. If the main steam valve is closed by a motor drive, the steam from the reactor is confined to the reactor vessel-steam dryer circuit. The line from the cold water storage tank can then be opened and water circulated through the coils which condense reactor steam. The heated water returns to the overhead storage tank in a natural circulation process. Water condensed from reactor steam flows back to the reactor vessel. Thus shutdown heat can be removed from the core with no power available to run pumps. This method of emergency cooling is effective for about twelve hours without boiling the 15,000 gallons (57,000 liters) of stored water. This gives sufficient time to set up other cooling provisions.

The turbo-generator is protected in the usual way by a quick-acting trip valve in the steam line. This valve is automatically shut in about ½ sec if the turbine overspeeds or if the generator load is lost. This valve is tied into a steam bypass system for the turbine, discussed later.

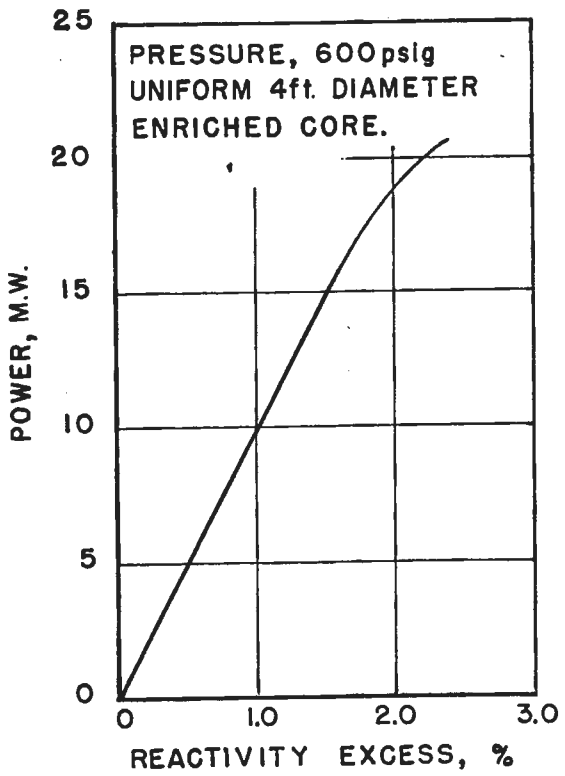
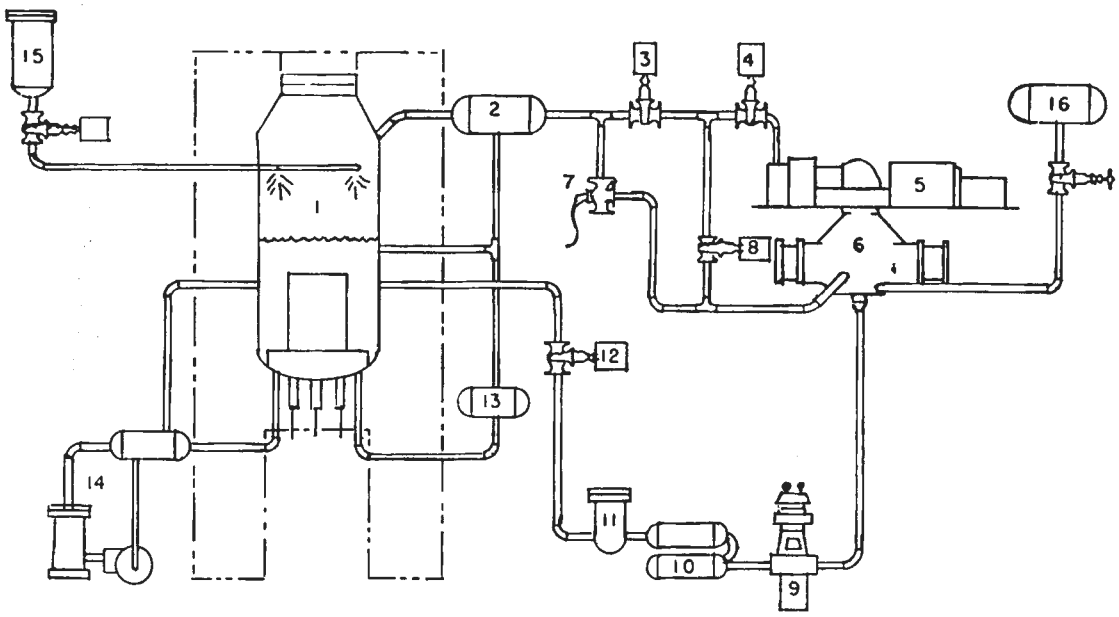


Figure 7. Thermal power vs reactivity % excess



1 REACTOR
2 STEAM DRYER-EMERGENCY COOLER
3 MAIN STEAM ISOLATION VALVE
4 TURBINE TRIP THROTTLE VALVE
5 TURBINE GENERATOR
6 CONDENSER
7 RELIEF AND SAFETY VALVES
8 STEAM BY-PASS VALVE
9 REACTOR FEED PUMPS
10 AIR EJECTOR COOLERS
11 FEEDWATER FILTER
12 FEEDWATER REGULATOR
13 START-UP HEATER
14 REACTOR ION EXCHANGERS
15 BORIC ACID INJECTORS
16 DEMINERALIZED WATER STORAGE TANK

Figure 8. Schematic diagram of steam and feedwater system

The turbo-generator, of conventional construction, is rated at 5000 kw. The throttle pressure is 560 psig (39 atm). The exhaust is 2½ in. Hg abs (0.1 atm). The steaming rate is about 12.5 lb/kwh (5.7 kg/kwh) for these conditions. A light-gauge, welded metal housing encloses all flanged and bolted joints to completely seal the unit. At the horizontal housing joint a recess is cut into the flange face. This is evacuated to a fluid recovery system to prevent pressure buildup in the welded cover.

The main condenser is a 5300-ft² (500 m²), single-pass unit. It has double tube sheets, providing an evacuation space between them to trap any leakage. The tubes are of aluminum in place of conventional copper to reduce potential radioactivity in feedwater. Steam bypassing the turbine is introduced below condenser tubes to assure good deaeration at low turbine loads.

Circulating water for the main condenser cooling is passed over a cooling tower at a rated maximum of 14,000 gal/min (882 liters/sec). The cooling tower basin is above the condenser so that natural circulation of water to cool the condenser will take place if pumping power is lost.

Steam piping and valves are of carbon steel. The insulation used is 85% magnesia block. All joints are welded except where provisions for sectionalizing and cleaning are necessary.

Leakage of steam and condensate is minimized at the valve and turbine shaft seals by a double buffer arrangement using 2-lb steam at the first buffer and dry air at the second stage. Between the steam and dry air stages the fluid is drawn off to a recovery system at a vacuum of 4 to 6 in. (10 cm) of water. Dry air for the seals is obtained from this recovery system. A diagram of this system including a typical seal is shown in Fig. 9. Recovered fluid is returned to the feedwater system at the condenser. This minimum loss of fluid provides for the future use of D_2O.

REACTOR AND STEAM PLANT CONTROL

As indicated in connection with Fig. 7, the power output is changed by moving rods to new settings. The rods remain stationary at the new setting, the power being nearly proportional to the reactivity excess. For EBWR, automatic control or adjustment of reactor power to match load changes is not provided. The control of essential parameters is handled in the steam system. Steam from the reactor goes either directly to the condenser (called steam bypass) or to the turbine. The principal control parameter is steam pressure. The pressure will rise if the reactor power is higher than the combined turbine and steam bypass load.

The bypass valve is rated at 82,500 lb/hr (10.4 kg/sec) at 600 psia (42 atm) upstream pressure, which allows a 130% steam bypass load. This valve is hydraulically operated and is positioned by a hydraulic control system. The steam pressure, P, at the reactor is compared to a mechanically set reference pressure,

P_0. The difference $(P - P_0)$ is applied to the bypass valve so that the amount it is open depends on the value $(P - P_0)$. A desired pressure value for P can be obtained by adjusting P_0. In this way the steam load is matched to the reactor power output.

When the turbine is to be operated the turbine trip valve is opened. Steam to the turbine is then controlled by turbine admission valves. These in turn are positioned by the standard speed control for the turbine through the turbine shaft governor. A cam attached to the operating lever on the admission valves is calibrated to give a signal proportional to turbine steam flow.

When the turbine is running the reactor output must be equal to the turbine steam load plus the steam bypass to the condenser. A hydraulic link is made between the turbine admission valve and the steam bypass valve. This makes the sum of the openings of these valves equal to the set pressure difference $(P - P_0)$.

In operation it is planned to allow a nominal 5% of the total reactor steam output to bypass directly to the condenser. Variations in load are, therefore, not reflected to the reactor. The reactor power is independent of the load as long as the 5% margin is maintained. When this control system is bypassing less than 5% of the steam, the reactor power is raised by resetting the rods until that value is restored.

A sudden loss of generator load is also handled by this automatic control system. A sudden loss of load is detected by circuits interlocked to the generator breaker which trip-close the turbine trip valve. This is a protective measure against overspeed and takes place in ½ sec. A second link is inserted in the pressure control system for the bypass valve which detects the open or shut position of the turbine trip valve. The link causes the full $(P - P_0)$ signal to be applied to the bypass valve and it then opens to the position demanded by this difference. Full reactor load is then passed to the condenser. The time constants of this control system are such that this interchange of steam load can take place with less than a 5% pressure excursion in the reactor vessel.

Inaccuracies in the valve positioning which result from the control system design tolerances can cause the pressure P to run at a value slightly above or below the desired value. This is adjusted to the exact value wanted by changing P_0. P_0 is a motorized setting which the operator in the control room can change. Under steady conditions of steam flow, the control system is designed to hold the pressure to ± ½% by the control system.

Experience with the Borax reactor has shown that a boiling reactor is very sensitive to steam flow changes: the power rises as steam flow is reduced and falls as steam flow is increased. The boisterous boiling process in the reactor vessel is especially disturbed by large pressure reductions. Short period fluctuations or chugging through large power changes may result. Therefore, all steam valves in this control system are designed to close and main-

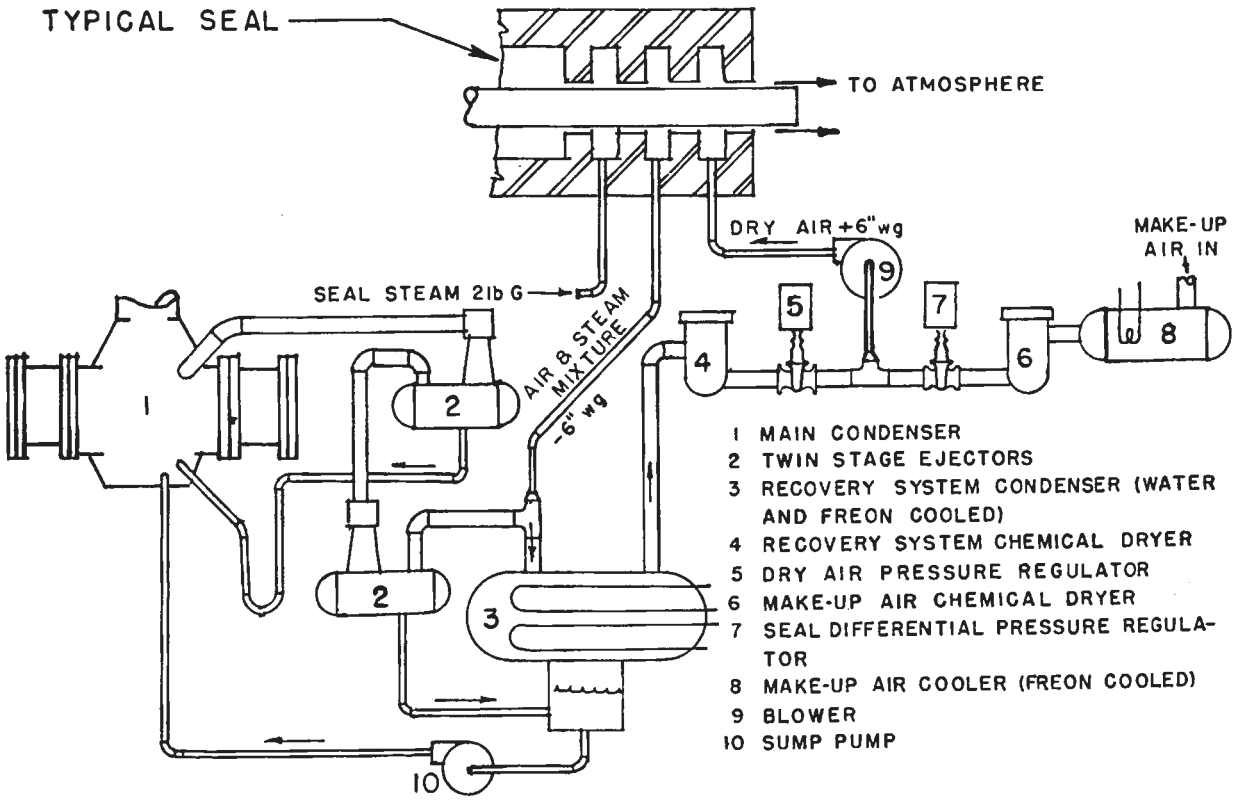


Figure 9. Schematic diagram of air drying and fluid recovery system

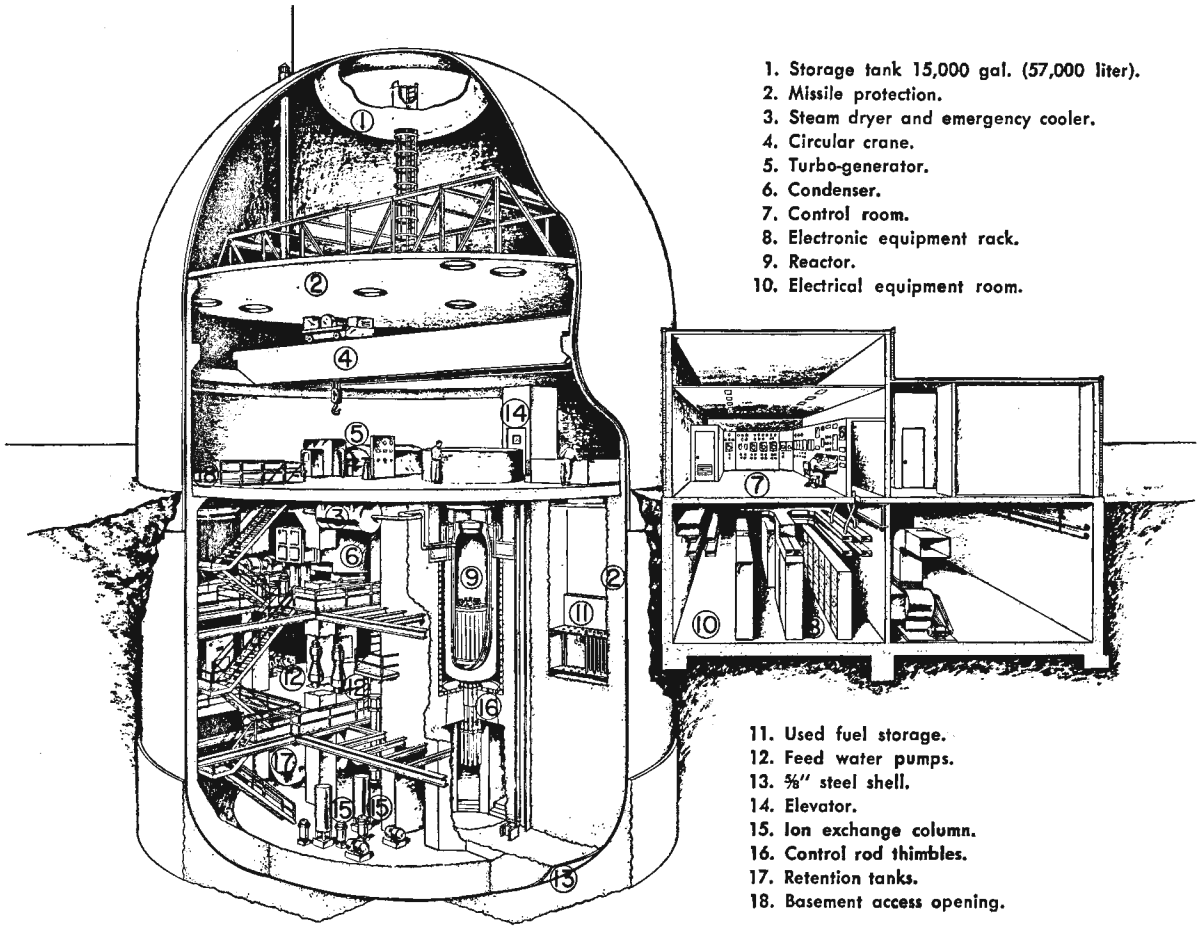


Figure 10. EBWR plant perspective

tain reactor pressure if hydraulic power is lost. When this power is lost the bypass valve is held against a stop by a spring. The stop is motorized. The electric power for this motor can be obtained from several sources. This places control of the reactor pressure under the operator's supervision.

Normal reactor shutdown, with all steam and feedwater controls functioning, is simply a matter of inserting the control rods. Steam is bypassed to the condenser which is cooled by the circulating water from the cooling tower. If the pumps fail in the circulating water system, natural circulation is provided through a bypass valve around the cooling tower. The cooling tower basin is located a few feet above the main condenser, as stated before.

Reactor power is sensed by neutron-detecting ion chambers and displayed at the control console from which the nine rod drive motors can be controlled. The rod positions are displayed by dials linked to the drives by synchro receiver-transmitter pairs. Damage to the reactor core is prevented by a shutdown interlock circuit containing (1) three over-flux circuits, (2) a reactor flux period circuit, to be used during startup, (3) a temperature element located in the reactor water and set to prevent operation below 325°F (163°C), and (4) a rate of pressure change instrument set to function at a change of 20 psi/sec. A series of procedural interlocks will prevent improper startup or operation of control rods and plant equipment.

An important interlock touched upon earlier is reactor water temperature. The interlock is a gas pressure-operated switch. It fails in a safe direction, i.e., if the sensing element fails it indicates low water temperature. The startup temperature is set at 325°F (163°C).

Plant and reactor control is centralized in the control room outside of the steam plant building shell. The connections between the plant equipment and control room are made through about 1200 electrical wires. To maintain gas tightness these wires are sealed where they leave the shell. Valves are motorized. Measurements of flow, pressure and temperature are transmitted electrically. Approximately 50 recording or indicating instruments are used for neutron flux, gamma radiation and plant system measurements. Some important points at which continuous radiation monitoring is provided are: the condenser hotwell, the reactor water ion exchangers, air exhaust ducts and the main floor work area. Holes 6 in. (15 cm) in diameter run through the shield, tangential to the reactor vessel, to admit neu-

tron-detecting ion chambers and counters for reactor flux measurements.

Switchgear for the main generator is located outdoors and operated from the main control room. All electronic equipment is located in the basement directly below the main control room.

EBWR PLANT BUILDING

Components of the power plant system which might contain radioactivity are contained in the gas-tight building shown in Fig. 10. The building shell is of ⅝-in. (1.59 cm) steel in the form of a cylinder 80 ft (24.5 meter) in diameter, 119 ft (36.4 meter) high, designed for 15 psig (2 atm) internal pressure. If, as a result of a steam system rupture, all the water flashed to steam, less than 9 psig (1.6 atm) and 180°F (82°C) would develop in the building. A spray system, fed at 1000 gal/min (63 liters/sec) from an overhead 15,000-gallon (57,000 liters) tank at the top of the building can condense the vapor and reduce temperature to nearly normal in 15 minutes. A continuous supply of 100 gal/min (6.3 liters/sec) is available through the tank from a water supply source outside of the building. Any small leaks of radioactive gas which develop in the shell during a catastrophe would be limited in duration to the pressure reduction period.

For access to the building a personnel air lock is located between the plant area and the service building. Mechanical interlocks prevent opening both doors of the air lock simultaneously.

Excess heat in the plant building is removed by air conditioners to hold the air change requirements to a minimum. Only 3000 ft³/min (1415 liters/sec) of air is changed and the inlet and exhaust ducts have quick-closing dampers to prevent egress of contaminated air in an emergency. The dampers are released through radiation monitor circuits set to operate if radioactivity is in the exhaust air.

A circular crane is used for handling equipment. An access hatch is used for removing equipment from the basement. A securely gasketed freight door is provided for removal of equipment such as ion exchangers, vessels, and large equipment sections needing shop repairs or routine replacement. This door is opened only if the reactor is not operating.

ECONOMIC CONSIDERATIONS

The small size and experimental nature of EBWR makes the cost of power generated at the outgoing bus bar higher than it would be in a plant designed only for the routine production of power. Firm bid prices have been obtained on all of the EBWR plant except the reactor core. The cost of construction is about $3,000,000. To this, $800,000 must be added for the fabrication cost of the core. The total of $4,000,000 of capital investment together with current uranium costs form the basis for cost computations. These uranium costs are $40/kg of natural uranium and $15 to $30 per gram of 85-90% U^{235}.

In Table I, the cost of operation has been broken down for a 20-year operating period. A 4% annual rental charge was applied to uranium cost. A plant factor of 80% was used. Two cases are compared.

Table I. Cost of Electrical Power Generated by EBWR

Cost base	*Case 1*		*Case 2*	
Output of generator (net), kw	4500		9000	
Operating period, yr	20		20	
Cost				
Plant less core	$3,200,000		$4,000,000	
Reactor core	800,000		800,000	
Net electrical output kwh/yr	31.5×10^6		63×10^6	
Natural uranium discharged/yr, kg	335		670	
Kw days exposure per ton (908 kg)	10^7		10^7	
Operating cost	*Millions of dollars*	*Mills/kwh*	*Millions of dollars*	*Mills/kwh*
Capital	0.600	19.1	0.720	11.4
Fuel rental	0.019–0.030	0.6–0.9	0.019–0.030	0.3–0.5
Natural U used	0.013	0.4	0.027	0.4
Burnout of enriched elements	0.040–0.080	1.3–2.5	0.080–0.159	1.3–2.5
Fabrication				
Natural elements	0.150	0.5	0.030	0.5
Enriched elements	0.045	1.4	0.090	1.4
Separation				
Enriched (40% burnup)	0.028	0.9	0.056	0.9
Operation				
Labor and overhead	0.266	8.5	0.266	4.2
Total	1.026–1.077	32.7–34.2	1.287–1.377	20.4–21.8

Case 1 has a net output of 4500 kw. The resulting power cost is between 32.7 and 34.2 mills/kwh depending on the price of the highly enriched uranium.

If irradiation levels higher or lower than the 10^7 kilowatt days/ton used in Case 1 are assumed, the cost of power will be reduced or increased, respectively. If the irradiated fuel were chemically processed to recover the plutonium and depleted uranium, the value of the recovered products should, at least, compensate for the cost of chemical separations. Thus, the penalty associated with short exposures is due primarily to fabrication cost.

The heat output of EBWR, designated as 20,000 kw, is conservative. It is likely that the power can be increased to 40,000 kw. If a second generator is added and the output is doubled, the estimated additional cost is $800,000. The effect of doubling the plant output is shown by Case 2. The power cost which results is 20.4 to 21.8 mills/kwh.

This last unit cost can be compared with the cost of power from small coal plants. The average 10,000 kw plant in the United States of America produces electricity at between 10 and 12 mills/kwh.

The difference of 10 to 12 mills by which the EBWR cost exceeds the coal plant is not unexpected. Nuclear plants of this size are not competitive unless located where the coal cost is relatively high. Furthermore, if the experimental features of the plant and the containment features of the building were omitted, the capital investment would be 30% less. Applying this saving to Case 2, the cost is reduced about 6 mills/kwh. The result is between 14.4 and 15.8 mills/kwh. It is generally accepted that the cost of electric power from a reactor plant will decrease more sharply as the plant size is increased than does the cost of electrical power from a coal-fired plant. The above figure of about 15 mills/kwh indicates that large electrical capacity boiling reactor power plants have a high probability of producing power at competitive costs.

The power plant described herein is practical as a power source where cost is high and where the quantity of electricity needed is small. These conditions are precisely those which exist in parts of the world where industrial development is in its early stages. It is suggested, therefore, that this boiling reactor power plant or its heavy water version immediately be applied to meet the electrical power needs in such areas. It should be noted in particular that in connection with fuel element fabrication quite large savings could be achieved if the demand from a number of reactors were to exist simultaneously. The costs exhibited for this plant necessarily include engineering and design costs which do not have to be repeated. These considerations, together with the expected early realization of the operation of this reactor and therefore accumulation of operating experience, further emphasize the suitability of this plant for application where the power demand is in the five to twenty megawatt range.

APPENDIX 1. EBWR DATA

General

Heat output	20,000 kw
Gross electric output	4800 kw
Operating pressure	600 psig (42 atm)
Operating temperature	488°F (252°C)

Fuel assemblies

Number in core, natural uranium	77
Number in core, enriched U^{235}	35
Number of plates per assembly	6
Dimensions of plates	0.28 × 3.75 × 48 in. (0.71 × 9.5 × 122 cm)
Cladding for plates	Zircaloy-II

Total fuel in core

4535 kg of natural uranium (5 short tons)
19 kg of enriched U^{235} (42 lb)

Nuclear data

Average thermal flux in fuel	1.6×10^{13} n/cm²/sec
Ratio of flux in enriched elements to natural elements	1.153
Power from enriched elements	37%
Power from U^{235} fissions in natural elements	56%
Power from U^{238} fissions in natural elements	7%
Conversion ratio	0.7
Reactivity	
Temperature 68°F (20°C) to 488°F (252°C)	3.5%
Xe + Sm (equilibrium values)	3.36%
17.5% steam voids	2.2%
Xenon everride (½ hour)	0.23%
Total reactivity to control	9.29%
Strength of control rods	12.5%
Neutron lifetime	6×10^{-5} sec

Heat transfer and fluid flow

Average power density in core coolant	25 kw/liter (706 kw/ft³)
Steam flow (at 20,000 kw)	60,000 lb/hr (7.56 kg/sec)
Average steam voids in heated channels	20%

Average steam voids in total moderator	17.5%
Average exit steam quality	2.56%
Water recirculation per lb of steam	29 gal/min (1.83 liters/sec)
Subcooling at channel inlet	11°F (6°C)
Average length of core boiling	32 in.
Average heat flux	40,000 Btu/hr/ft² (3 gm-cal/cm²/sec)
Maximum heat flux	150,000 Btu/hr/ft² (11 gm-cal/cm²/sec)
Average temperature of fuel at centerline	518°F (270°C)
Maximum temperature of fuel at centerline	561°F (294°C)
Average surface temperature of fuel	495°F (257°C)
Maximum surface temperature of fuel	508°F (265°C)

Control rods

Total number	9
Spacing	12¾ in. (32.4 cm)
Length of cruciform control section	48 in. (122 cm)
Composition	100% hafnium
Total travel	48 in. (122 cm)
Penetration of absorber into core	45 in. (115 cm)
Insertion time full travel	0.5 sec
Maximum withdrawal rate	0.4 in./sec (1 cm/sec) 0.01% reactivity/sec

Pressure vessel

Diameter	84 in. ID (2.14 meters)
Height (inside)	23 ft (7 meters)
Design pressure	800 psig (55 atm)
Thickness of cylindrical portion	2⅜ in. (6 cm)
Working pressure	600 psig (42 atm)
Material	SA 212 Gr. B
Clad material	Type 304 stainless steel
Bolted top closure diameter (inside)	64 in. (1.62 meters)
Thermal shield thickness	1 in. (2.54 cm) stainless steel
Relief valve setting	700 psig (48.7 atm)
Total weight	~60 tons (54,430 kg)
Total weight of contained water	10 tons (9072 kg)

Shielding

Thermal	1 in. (2.54 cm) stainless steel inside the pressure vessel plus pressure vessel itself
Biological	3 in. (7.6 cm) of lead plus 7½ ft (2.28 meters) of concrete on sides
	4 ft (1.22 meters) of heavy concrete plus 3 in. (7.6 cm) of lead on the bottom

Power plant

Condenser pressure	2½ in. Hg (0.1 atm)
Feedwater flow rate	120 gal/min (7.58 liters/sec)
Feedwater temperature	110°F (43°C)
Gross electricity generated	4800 kw
Net electricity	4500 kw
Heat sink	Cooling tower

The Homogeneous Reactor Test

By S. E. Beall and J. A. Swartout,* USA

The Homogeneous Reactor Test is the second of the experimental, aqueous homogeneous reactors to be constructed under the program of the United States Atomic Energy Commission for the evaluation of nuclear power reactors. Its predecessor, the Homogeneous Reactor Experiment, was constructed in 1951 and dismantled early in 1954 after two years of successful operation.[1] The Homogeneous Reactor Test is now being assembled in the building formerly occupied by the Homogeneous Reactor Experiment at the Oak Ridge National Laboratory. It represents an advance over the latter in power, physical size, and quality of construction and is, in nearly all respects, a prototype of a full-scale nuclear power product.

Based upon technical factors and economic analyses of the various embodiments of the aqueous homogeneous reactor principle, the two-region, forced circulation type, operating on the thorium-U^{233} breeding cycle, possesses the potential for lowest cost power for the plant sizes now contemplated. Therefore, the research program has concentrated upon the resolution of the technical uncertainties associated with this type and the demonstration that construction and operation of a full-size power unit will be practical. The approach has been three-fold: (1) construction and testing of experimental reactors of the two-region type, (2) laboratory research[2] and engineering component development,[3,4] and (3) design and analysis of large power installations.[5]

As a result of the operation of the Homogeneous Reactor Experiment and of the extensive experimental program conducted with it, several uncertainties were resolved regarding the nuclear and chemical behavior of aqueous homogeneous reactors at the high temperatures, and correspondingly high pressures, required for power generation. Included were demonstrations of (1) a remarkable degree of inherent nuclear stability, a result of the very large negative temperature coefficient of reactivity, (2) the elimination of the need for mechanical control rods as a consequence of this inherent stability, (3) flexibility and simplicity of fuel handling, (4) stability of the fuel, (5) the ability to attain and maintain leak-tightness in a small high-pressure reactor system, (6) the safe handling of the hydrogen and oxygen produced by radiation decomposition of the water, and (7) the direct dependence of reactor power upon turbine demand.

The objectives of the Homogeneous Reactor Test are (1) to demonstrate that a homogeneous reactor of moderate size can be operated with the continuity required of a power plant, (2) to establish the reliability of engineering materials and components of a size which can be adapted to full-scale power plants, (3) to evaluate equipment modifications which will lead to simplifications and economy, (4) to test simplified maintenance procedures and, in particular, maintenance underwater, and (5) to develop and test methods for the continuous removal of fission and corrosion contaminants.

Construction is scheduled for completion early in 1956. Following the usual mechanical and leak testing, the reactor will be put into operation with only heavy water in the blanket system; the fuel will be a dilute solution of uranyl sulfate (about 90 per cent U^{235}) in heavy water. Operation will continue with this combination for a minimum of six months to establish the reliability of operation and to test the effectiveness of continuous chemical extraction processes. For the latter purpose, a chemical pilot-plant is attached directly to the reactor. During this period the power output of the reactor will not exceed 10,000 kw (heat). As a second phase of the reactor test program, the blanket will consist of a suspension of thorium oxide in heavy water. Dependent upon the results of operation with U^{235} under these conditions, U^{233} may be substituted as the fuel for a final evaluation of the U^{233}-Th cycle.

In the description which follows, emphasis will be placed on the system as it will operate with the D_2O blanket.

REACTOR SPECIFICATIONS

The bases for the design of the reactor, which are summarized in Table I, were selected early in 1954 and were intended to take fullest advantage of the progress in chemistry, materials and component development, and the experience with the first reactor experiment. In order that the objective of a significant test of the engineering feasibility of a large power station be satisfied, it was necessary that the physical size of the reactor and its auxiliaries be increased appreciably beyond that of the first experiment. To hold the cost within reasonable bounds for an experiment, it was decided to limit the power output and

*Oak Ridge National Laboratory. Work by members of the Homogeneous Reactor Project at the Oak Ridge National Laboratory.

Table I. Design Bases

Power, heat	5000 kw to 10,000 kw
Temperature, core outlet	300°C
Pressure	>500 lb/in.² in excess of vapor pressure
Core diameter	32 inches
Core solution	UO_2SO_4 in D_2O
Blanket	D_2O or ThO_2 in D_2O
Fuel circulation rate	400 gpm
Core flow pattern	Straight-through
Core material	Zircaloy-2
System material	Type 347 stainless steel
Radiolytic gas removal	Pipe line separator
Radiolytic gas recombination	Low pressure—platinized alumina catalyst
Fission product gas disposal	Decay on activated carbon
Control	
Normal	Variable solution concentration
Safety	Temperature coefficient

thus the expense for heat removal equipment. The size of the reactor core represents a compromise between two objectives, attainment of the high specific power required for economy in a large plant and evaluation of the fabrication and durability of a zirconium alloy core tank. The power output was therefore set at 5000 kw (heat) with the possibility of a maximum of 10,000 kw and the core diameter at 32 inches. Although these together result in a low specific power of 17 kw per liter in the core at 5000 kw, this was considered acceptable since operability at a relatively high specific power of 30 kw per liter had been demonstrated in the first experiment. Another factor affecting the selection of core diameter was the opinion of fabricators that current technology would be exceeded for a zirconium vessel larger than 36 inches.

The increase in fuel temperature from 250°C in the first experiment to 300°C was based upon the favorable corrosion resistance of both stainless steel and Zircaloy-2 to dilute uranyl sulfate and the higher temperature at which the two-liquid phase region appears in the more dilute fuel.[2]

The design pressure of 2000 lb/in². resulted from the necessity for an overpressure on the system to prevent boiling and to reduce the volume of gas in the core. A maximum pressure of 750 lb/in². was thus provided in excess of the vapor pressure of water of about 1250 lb/in². at 300°C.

The thickness of the blanket of 14 inches between the core and pressure vessel was selected partially to limit neutron leakage, when thorium would be present, to an extent that a high conversion ratio or perhaps even breeding might be demonstrated and partially to permit use of a simple pressure vessel of dimensions within the means of standard fabrication techniques.

The fuel circulation rate of 400 gpm was based upon heat removal requirements and the availability of suitable pumps. Pumps capable of this capacity had been successful under comparable conditions.

The vortex type flow through the core of the first experiment was abandoned in favor of a straight-through flow because hydrodynamic experiments demonstrated the pressure drop of the former to be excessive for larger cores. As a result the extraction of radiolytic gas directly from the core was excluded and external gas separation in the exit pipe from the core was specified.

As in the first experiment, the separated radiolytic gas was to be recombined at low pressure, but by means of a platinized alumina catalyst instead of combustion in a flame-type recombiner.

To dispose of the fission product gases which are stripped from the fuel by the radiolytic gas, adsorption and subsequent decay on beds of activated carbon were specified. Experience with this method of disposal was completely satisfactory in the first reactor experiment.

Type 347 stainless steel was designated as the material of construction except for the Zircaloy-2 core vessel and titanium at points of high turbulence such as the pump impellers and the gas separator. Previous corrosion and welding experience with this grade of stainless steel has been excellent. The Homogeneous Reactor Experiment also had been constructed of this material.

Finally, the design bases required that control of the reactor depend entirely on solution concentration for shimming and on temperature coefficient for safety. No control rods or other mechanical control devices were to be designed into the reactor.

From these general bases, the design parameters of Table II were established.

REACTOR FLOW SHEET

The flow diagram for the reactor is illustrated in Fig. 1. Since the fuel and blanket systems are identical—the equipment differs only in that larger sizes

Table II. HRT Design Parameters, HRT-1 (D_2O Blanket)

	Core	*Blanket*
Power, heat, kw	5000	220
Pressure, lb/in.²	2000	2000
Vessel, inside diameter, inches	32	60
Thickness	⅜	4.4
Material	Zircaloy-2	Stainless steel Clad carbon steel
Volume, liters	290	1550
Specific power, kw/liter	17	0.14
Solution	UO_2SO_4-D_2O	D_2O
Uranium concentration, gm U^{235}/kg D_2O	9.6	0
Circulation rate, gpm	400 at 256°C	230 at 278°C
Inlet temperature	256°C (493°F)	278°C (532°F)
Outlet temperature	300°C (572°F)	282°C (540°F)
Volume of gas generated, ft³/second at STP	0.96	0.013

are necessary to accommodate the greater volume of blanket fluid—only the fuel system will be described.

The High-Pressure Circulating System

The fuel solution is pumped into the bottom of the reactor core at 256°C and, at the 5 Mw operating power, is heated to 300°C as it proceeds upward to the outlet pipe. At the top of the outlet pipe is attached the "pressurizer" which is electrically heated to a maximum temperature of 335°C to produce 2000 lb/in^2. steam. The fuel flows past the pressurizer to the gas separator where directional vanes cause the fluid to rotate sufficiently to separate the radiolytic gas (D_2 and O_2) and fission product gas. The separated gas forms a vortex along the axis of the pipe and is bled to the low-pressure system. The reactor solution continues from the gas separator to the U-tube primary heat exchanger where it is cooled from 300°C to 256°C by transferring heat to the boiler feed water surrounding the tube bundle. The 244°C, 520 lb/in^2. steam produced on the shell side of the heat exchanger is bled partially to the 312 kva turbine-generator remaining from the Homogeneous Reactor Experiment and partially to an air-cooled steam condenser.

The uranyl sulfate solution flows next to the intake of the 400 gallon per minute canned rotor circulating pump and thence is pumped to the core for reheating.

The blanket fluid follows an identical cycle at a flow rate of 230 gpm.

The Low-Pressure System

The gases and some entrained liquid removed by the gas separator are transferred to the low-pressure system through a "let-down heat exchanger," a jacketed pipe which cools the gas liquid mixture to 90°C. A valve downstream of the heat exchanger throttles the gas-liquid stream to atmospheric pressure. The mixture then discharges into the "dump" tanks which have sufficient capacity to hold all the reactor liquid. An evaporator built into the dump tanks provides continuous mixing and, more important, steam for dilution of the deuterium and oxygen below the explosive limits. The gas and steam mixture flows upward to the catalytic recombiner in which the deuterium and oxygen react on a bed of platinized alumina pellets to form water vapor. The heavy water is condensed by the shell and tube condenser following the recombiner and normally flows back to the dump tanks. However, the water may be diverted to weighed storage tanks in case it is desired to change the concentration of the reactor fluid. Water which is returned to the dump tanks is mixed with the excess fuel solution stored there (approximately 25 gallons), and then fed to the intake of a sealed diaphragm injection pump which returns the liquid to the high-pressure circulating system at a rate of about 1 gpm, thus constantly replacing the liquid removed via the gas separator.

The small volume of intensely radioactive fission gas remaining after condensation of the reformed heavy water is dried in cold traps at −23°C and sent to beds of activated carbon. Here the fission gases are absorbed for decay. Gas leaving the bed is diluted with 1400 cfm of air and discharged to the atmosphere from a 100-foot stack.

All of the equipment described on these flowsheets is located within the primary shield pit. The relative positions of the various items are shown in Fig. 2 on the following page.

SHIELDING AND CONTAINMENT

The design of the shield was influenced by several factors including a requirement for accessibility and flexibility because of the experimental nature of the installation, provision for complete containment of the contents of the reactor should a leak develop or should the pressure vessel or heat exchangers rupture, efficient utilization of the space within an existing structure and capability of flooding with water for maintenance or replacement operations.

The reactor shield pit occupies the center high bay area of the building. The tank is constructed of ¾-inch welded steel plate, so reinforced that an internal pressure of 30 lb/in^2. will be contained, corresponding to the instantaneous adiabatic release of the entire contents of the reactor system. It is 54 feet long, 30½ feet wide, and 25 feet deep. The chemical processing cells, each 12 feet wide by 25 feet long, are designed similarly. The upper surface of the roof shield slabs is at ground level.

The roof is of high density concrete, five feet in total thickness; it consists of two layers of removable slabs with a completely welded steel sheet sandwiched between the layers and extending across the top of the pit to form a gas-tight lid. The roof slabs are anchored to the girders and supporting columns by means of a slot and key arrangement. The vertical columns are imbedded in a concrete pad which is 3 feet thick and is heavily reinforced with steel.

The wall between the reactor pit and the control room area is constructed of two ½-inch steel plates spaced 5½ feet apart, with the intervening space filled with a shielding material such as high density barytes gravel and water. The use of a fluid shield between the reactor and control room areas allows flexibility in the locations of service piping and instrument or electrical conduit. All lines leaving the reactor tank are welded into the shield wall; conduits are connected into junction boxes inside the pit with gas-tight seals on the individual wires.

Nowhere outside the shield will the radiation dosage from the reactor exceed 10 mrep/hr when the reactor is at 10 Mw. For the purpose of decreasing the neutron activation of equipment inside the tank, the reactor vessel is surrounded by a shield for thermal neutrons consisting of 1 to 2 feet of a mixture of barytes sand, Colemanite (containing boron) and

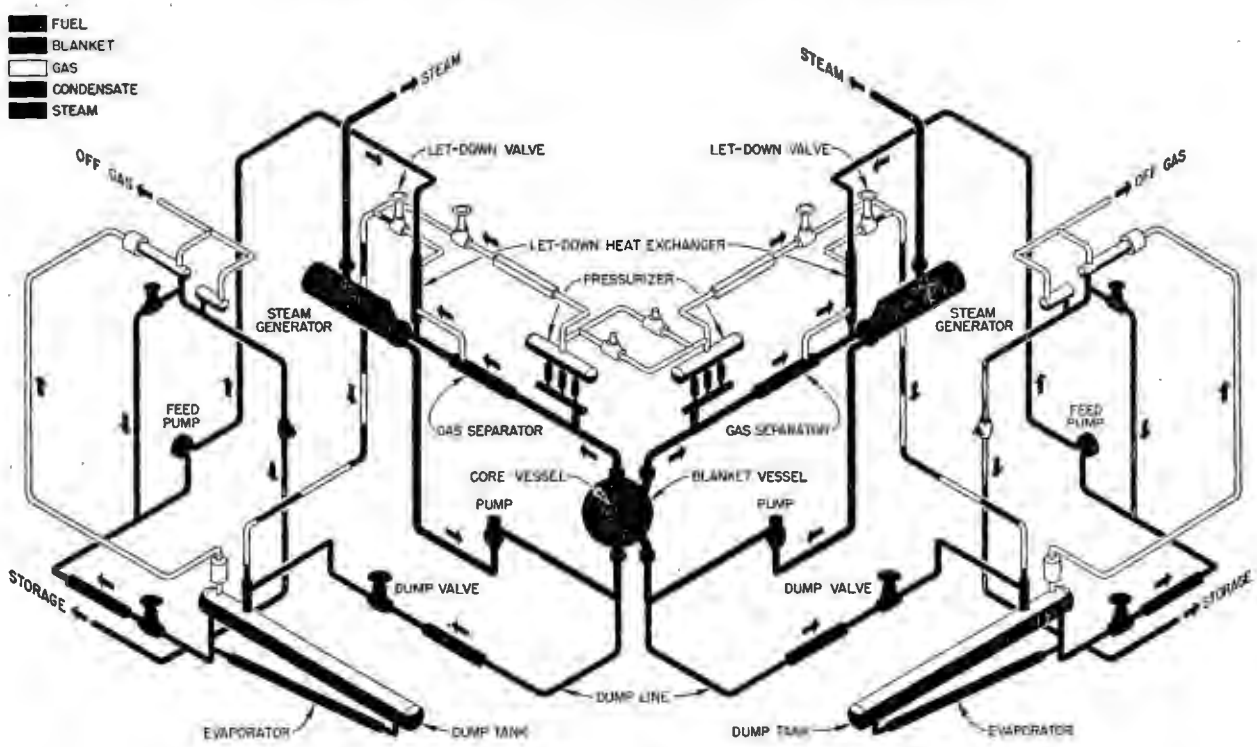


Figure 1. Reactor flow sheet—schematic

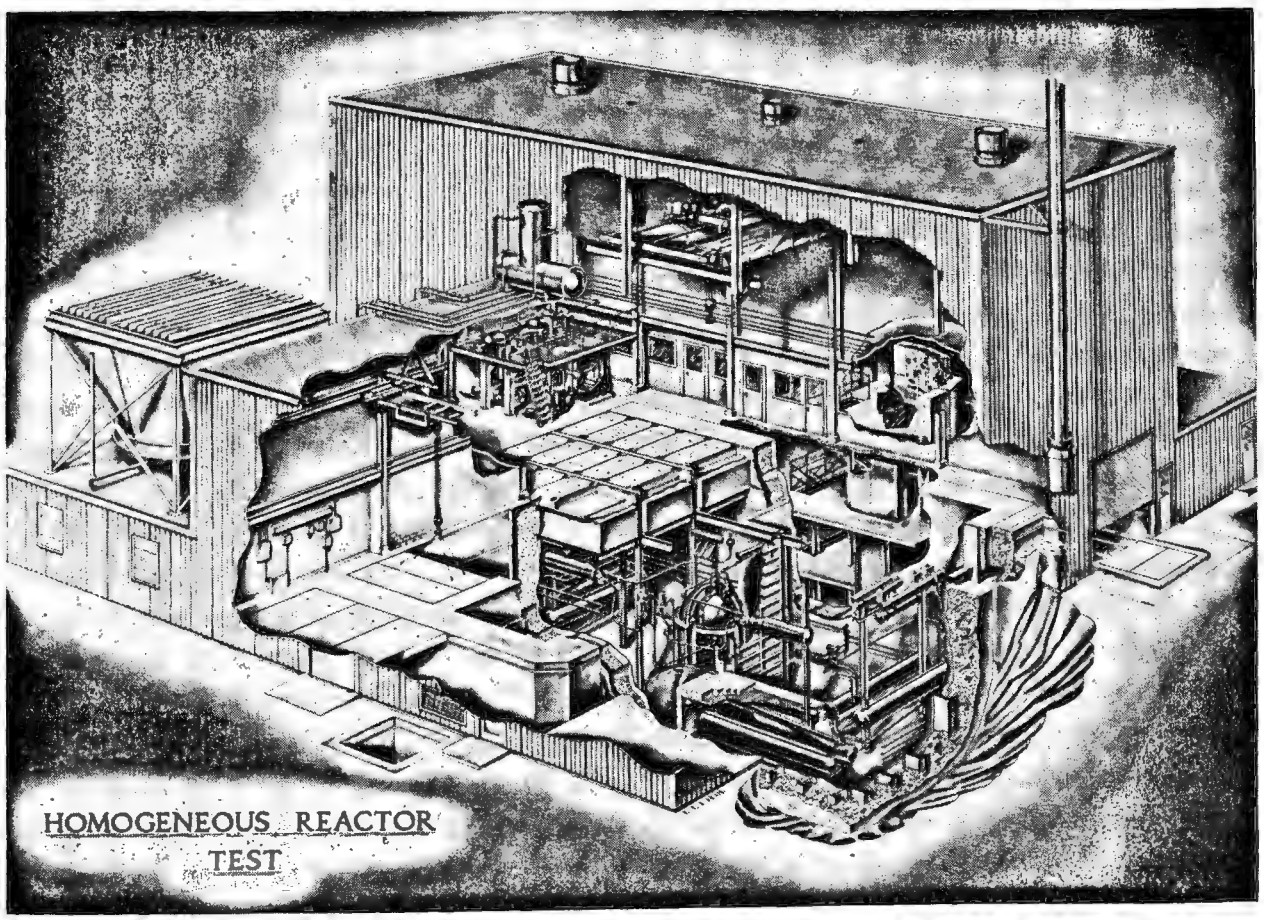


Figure 2. Artist's concept of homogeneous reactor test

water having a bulk density of 3 gm/cm³. This is contained in a steel tank shaped like an igloo.

Underwater Maintenance

The shield tanks and all the equipment contained therein have been designed for flooding with water to provide a flexible shield to protect workers during maintenance operations. Special long-handled tools will permit a worker to stand at the ground level while tightening bolts or making adjustments on equipment inside the flooded pit.

The components considered most likely to fail (e.g., circulating pumps, pressurizer heaters, valves) have been located and designed for remote repair in this fashion.

When it becomes necessary to open the system piping to remove equipment, the reactor fuel will be drained and flushed to the outer storage tanks. Next, the pipeline will be filled with heavy water, evaporated from the fuel solution. Freezer jackets have been provided around the lines adjacent to the disconnect points so that a heavy water ice plug can be frozen in the line. This will prevent the leakage of shield water into the reactor piping during disassembly. However, should light water leak into the piping, provision has been made to remove it by drainage.

Equipment Within Shield-Container

The arrangement of reactor components within the main reactor pit is shown in more detail in Figs. 3 and 4. The reactor is near the center of the pit, enclosed by the 2-foot thick thermal shield. To the left of the reactor is all the fuel system equipment, and to the right is the blanket equipment. The rows of supporting columns indicate the location of possible future dividing walls. Space between the columns has, in most cases, been kept free of equipment to permit installation of shielding for the isolation of groups of equipment in case repairs cannot be made while the pit is filled with water.

The pressurizer is positioned directly above the reactor. The gas separator is a part of the S-shaped outlet line to the heat exchanger. The main heat exchanger (steam generator) and circulating pump are located as close to the reactor as thermal stress considerations and accessibility for repairs would allow, in order to keep the hold-up of fuel solution in the connecting lines to a minimum. The area to the left of the reactor has been reserved for future equipment modifications or additions.

To the far left is grouped all of the low-pressure equipment including dump tanks, recombiner, condensate tank and cold traps. These components are assembled on a rigid structural steel frame.

Insofar as possible, valves are situated close to the control room wall so that air lines and leak detector lines can be kept short. The arrangement of the valves is such that all flanges can be easily disconnected from above.

CONTROL OF THE REACTOR

As mentioned previously, because of the inherent stability of homogeneous aqueous reactors, the reactor is designed with no control or safety rods. The function of the control rod is assumed by other mechanical devices such as pumps, evaporators, and valves. These control the concentration of fuel in the core region, and thus the average temperature of the reactor, while the power level is controlled by the rate of heat removal from the reactor. Protection from large increases in reactivity, which is normally a safety rod function, is afforded solely by virtue of the large negative temperature coefficient of reactivity, $-2 \times 10^3\ \Delta k/k$- °C at 280° C. Compensation for reactivity increases is accomplished by the thermal expansion of the fuel solution, some of which is driven out of the core via the relief pipe to the pressurizer whose vapor space serves as a surge volume. The basis for relying completely on the negative temperature coefficient stems entirely from experiments performed on the Homogeneous Reactor Experiment. It is therefore of interest to describe some of the experience with this earlier reactor.

Reactivity Experiments in the Homogeneous Reactor Experiment

In the Homogeneous Reactor Experiment, a series of kinetic experiments was performed in which the power responses to reactivity increases were observed. First, the entire range of normally available reactivity increases—fuel concentration, rod withdrawal, and reflector level—was tested with initial power levels as low as 10 watts and reactivity rates up to 0.05 per cent per second. Then, in order to provide more drastic tests, the main fuel circulating pump was stopped, the reactor was maintained at a low power and at high temperature, but the heat exchanger was cooled about 100°C. When the pump was re-started, the cold fuel from the heat exchanger was rapidly injected into the core, producing a rate of reactivity increase of as much as 0.8 per cent per second. The results of two experiments in which only the initial powers differed are shown in Fig. 5; the power increased on a period as short as 35 milliseconds reaching a peak of 10 Mw in 1 second, and then approached the equilibrium power demand within 0.2 seconds after the peak. The calculated pressure rise associated with the peak power of 10 Mw was only 5 lb/in².

Kinetics of the Homogeneous Reactor Test

Although no pressure surges were detected even in the most severe experiment performed in the Homogeneous Reactor Experiment, calculations for certain conceivable Homogeneous Reactor Test accidents indicated that the pressure rise might be sufficient to cause concern; the system was, therefore, designed to eliminate such possibilities.

When large excess reactivities must be overcome by large increases in the fuel solution temperature,

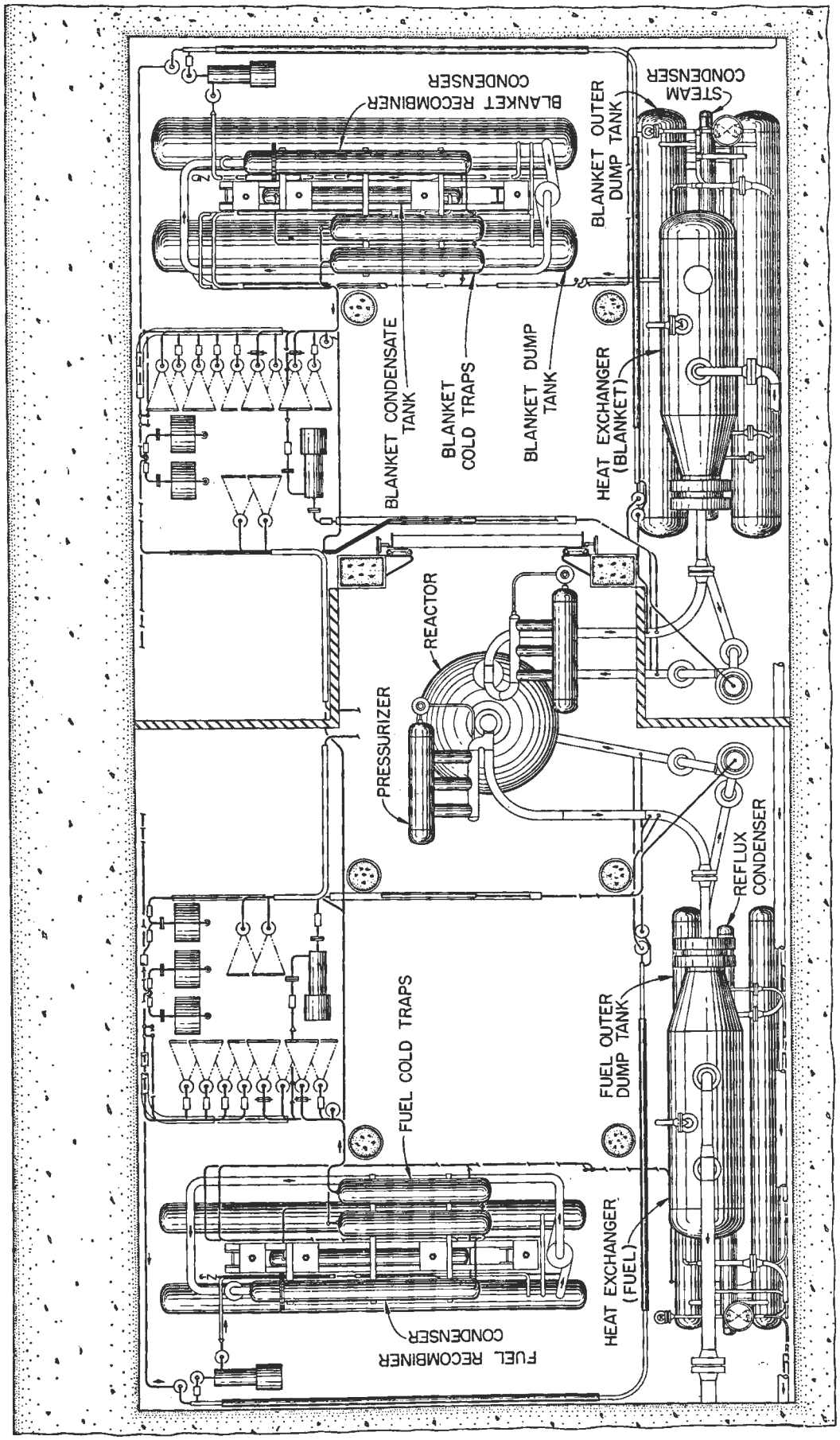


Figure 3. Reactor component arrangement—plan

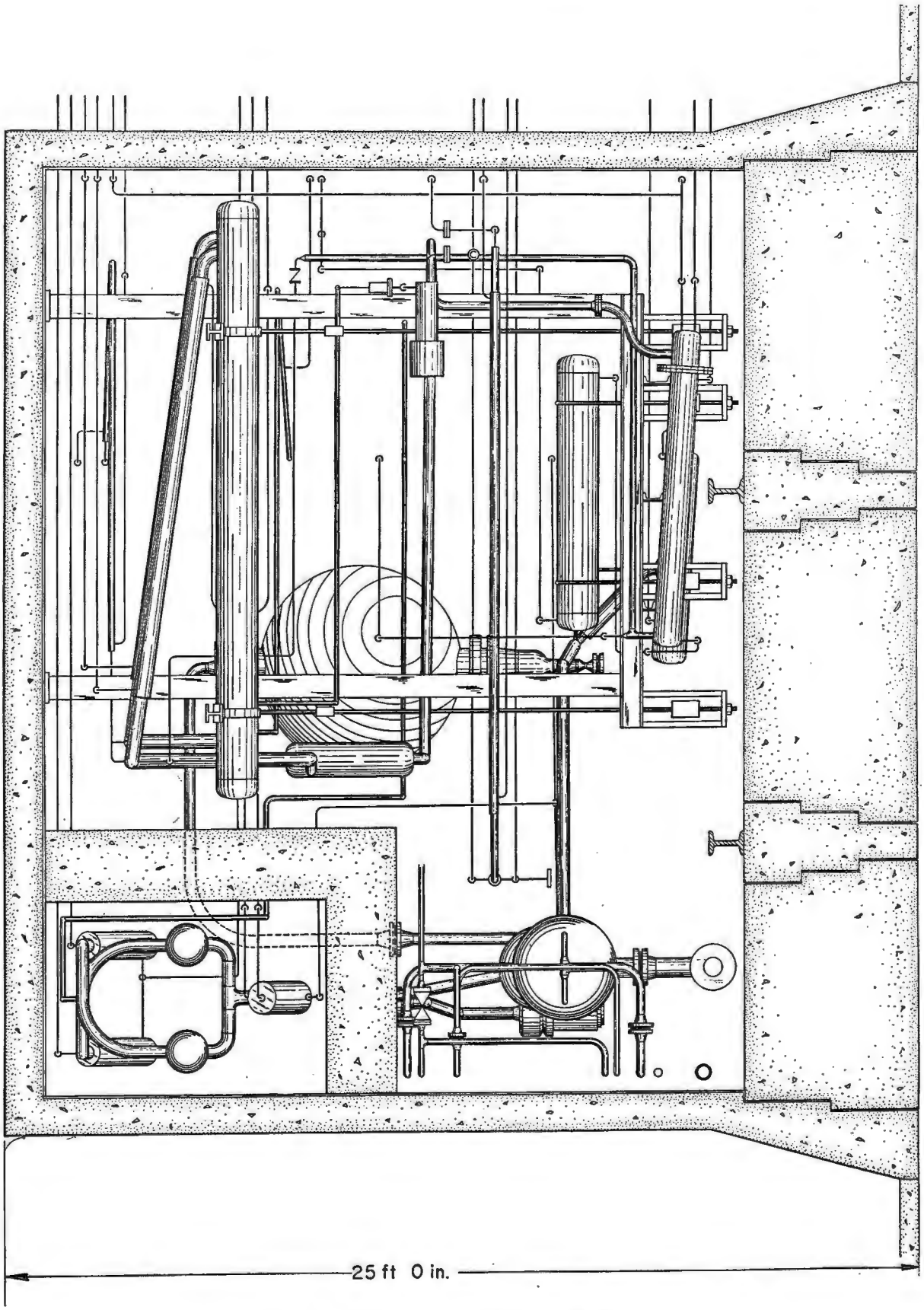


Figure 4. Reactor component arrangement—elevation

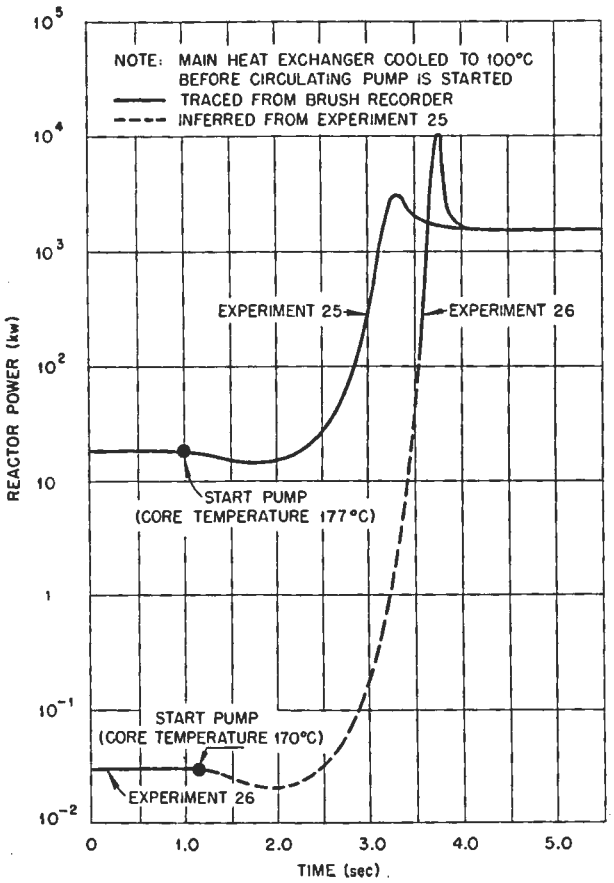


Figure 5. Power response of HRE during reactivity increase of 0.8%/sec

the resulting expansion of the solution is sufficient to momentarily increase the pressure in the core vessel by significant amounts. For example, if the main steam line from the primary heat exchanger ruptured, the sudden reduction in steam pressure to essentially atmospheric pressure would cool the fuel solution with an equivalent reactivity addition rate of about 1.6 per cent per second. If no preventive action were provided, the resulting pressure rise at the core would be 100 to 800 lb/in^2., depending on the power level of the reactor at the time of the accident. The maximum pressure rise of 800 lb/in^2. would result if the reactor power level were as low as 0.5 kw and would endanger the Zircaloy-2 core vessel. Rupture of the core tank would be a great inconvenience, to make an understatement, but would not be hazardous. It is estimated that rupture of the pressure vessel would require a rate of reactivity addition considerably in excess of 5 per cent per second.

In comparison, the most serious accident which could result from a departure from normal operating procedures would produce a negligible pressure rise. If, during start-up, the fuel were concentrated and pumped into the core cold, reactivity could be added at a rate of 0.66 per cent per second. In this case, a peak power of 1 to 10 Mw would result, with a rise in fluid temperature of 10 to 25°C and an increase in core pressure of 5 to 10 lb/in^2.

The control and instrumentation systems for the reactor are so designed that operations or sequences of operations which might result in excessive rates of reactivity are excluded. For instance, protection against the effect of the steam line rupture is provided by a device which stops the fuel circulation pump when the steam pressure is reduced at a rate greater than 20 lb/in^2. per second. In addition, automatic pressure relief is provided to prevent an excessive differential across the core vessel. (See Appendix.)

REACTOR OPERATION

Start-Up

The start-up procedure for this reactor is extremely simple. Initially all reactor fluids will be contained in the storage tanks. The operator begins by evaporating the water from the fuel solution in the dump tanks. As the heavy water condensate is collected, it is pumped into the high-pressure circulating systems by the high-pressure injection pumps. The fuel and blanket systems are filled simultaneously. When the level indicators on the pressurizers show "full," the pressurizer heaters are turned on and the pressure is raised to 2000 psig over a period of 3 to 4 hours. During this time the circulating pumps are started and building steam is admitted to the shell sides of the heat exchangers.

As the pure condensate circulates through the heat exchangers, it is heated until a temperature of 200°C is reached. At this point, the steam is cut off and the addition of the concentrated fuel from the fuel dump tanks is begun by opening a valve to the injection pump intake. As this concentrated fuel is added to the high-pressure circulating system at a rate of 3.2 kg solution per minute, the reactor core approaches the chain reacting state at a rate of 0.008% δk/second. When the concentration of the fuel being circulated exceeds 5.3 gm/kg, the reactor becomes supercritical and begins generating heat. The addition of concentrated fuel is continued beyond this point and the reactor temperature continues to increase as more fuel is added, until after about 80 minutes, it begins to approach the equilibrium value (252°C) corresponding to uniform mixing of all the fuel charge and D_2O. The variation of temperature with time over this period is shown in Fig. 6. If no steam is being extracted, the reactor idles at this temperature at a power sufficient to balance heat losses from the system.

When the operator wishes to raise the temperature of the reactor, he does so by closing a valve, diverting to the condensate storage tanks the heavy water which would normally return from the recombiner to the circulating system. This slowly increases the fuel concentration and therefore the operating temperature. The manner in which fuel temperature is related to concentration is plotted in Fig. 7. Concentrating is stopped when the fuel temperature

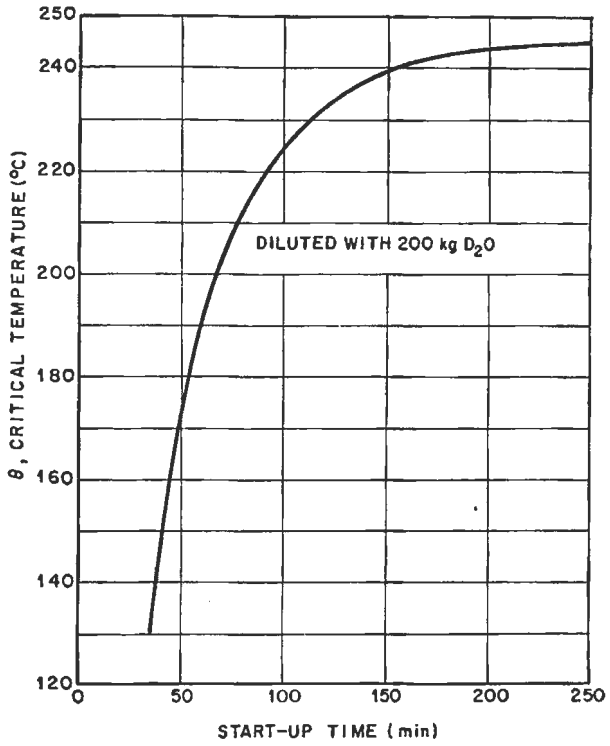


Figure 6. Start-up of HRT

reaches 280°C, equivalent to a fuel concentration of 9.6 gm U^{235}/kg D_2O. The reactor is now ready for power withdrawal. The total time consumed by these operations will be approximately 8 to 9 hours which coincides with the permissible rate of temperature increase (30°C/hour) set by the thermal stress limitations of the more massive components.

Power Removal

As a result of the relatively large temperature coefficient and low heat capacity, this reactor cannot generate more heat than is being removed except for very short periods of time. Heat removal is accomplished simply by opening the steam throttling valves. This causes the fuel leaving the core to be cooled as it passes through the heat exchanger. Since any reduction in core temperature causes an increase in reactivity as a result of the negative temperature coefficient, the cold fluid entering the core must re-heat until it overcomes the excess reactivity. Thus, in passing through the core, the fuel solution is heated at a rate corresponding to that rate at which steam is being withdrawn. At the 5 Mw operating level, the entrance temperature of the core is 252°C and the exit temperature is 300°C. The average core temperature will always be 280°C for a fuel concentration of 9.6 gm U^{235}/kg D_2O, although the inlet and outlet temperature will vary in proportion to the power extraction rate. At any time the operator may increase the average temperature by removing water from the circulating fuel, or change the power extraction by adjusting the throttling characteristics of the steam removal valve. When steam is being bled to the turbogenerator, operation is even simpler. In this case, it is necessary only to adjust the position of the turbine governor. The servo mechanism of the turbine governor will control whatever power level the operator has selected. The expected performance of this system is best illustrated by the actual experience recorded in the Homogeneous Reactor Experiment. Figure 8 shows how the reactor responded to sudden changes in heat extraction rates. Starting at a power level of 800 kw, the steam removal valve was suddenly closed. It is seen that the power decayed slowly for approximately 5 minutes, reaching a level of 225 kw. At that time, the load was transferred to the turbo-generator and during the next 30 minutes the power level was increased to 1000 kw with no underride or override.

Shut-Down

Shut-down of the reactor normally is accomplished by essentially the reverse of the start-up procedure. After closing the steam withdrawal valve, the operator begins pumping pure condensate into the high-pressure system. The fuel solution displaced by the incoming condensate is collected in the fuel dump tanks. As this operation proceeds the circulating fuel becomes more and more dilute and, after 2 hours, has a sufficiently low uranium content to be subcritical at 20°C. By that time, almost all of the uranium has been collected in the dump tanks and the reactor is ready for another start-up.

Shut-downs are considered abnormal if it is necessary to stop the reactor by emptying the high-pres-

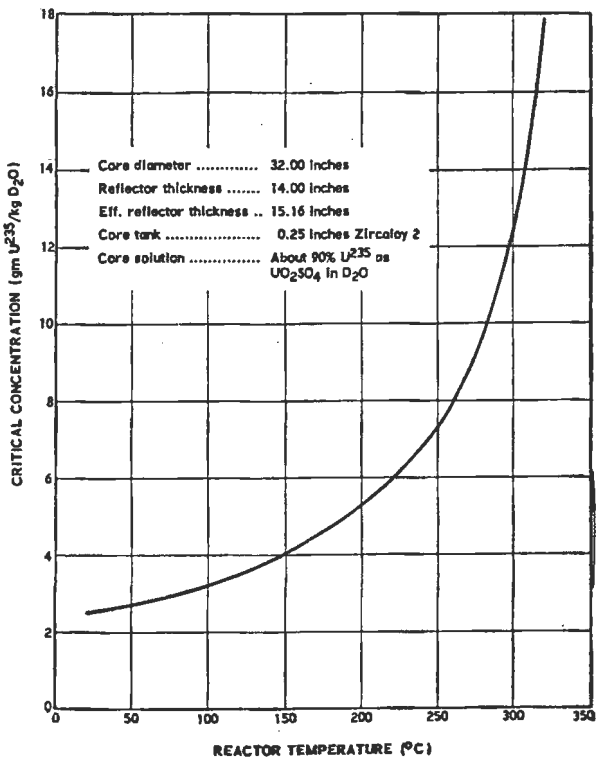


Figure 7. Critical concentration of U^{235} in core of HRT with D_2O blanket

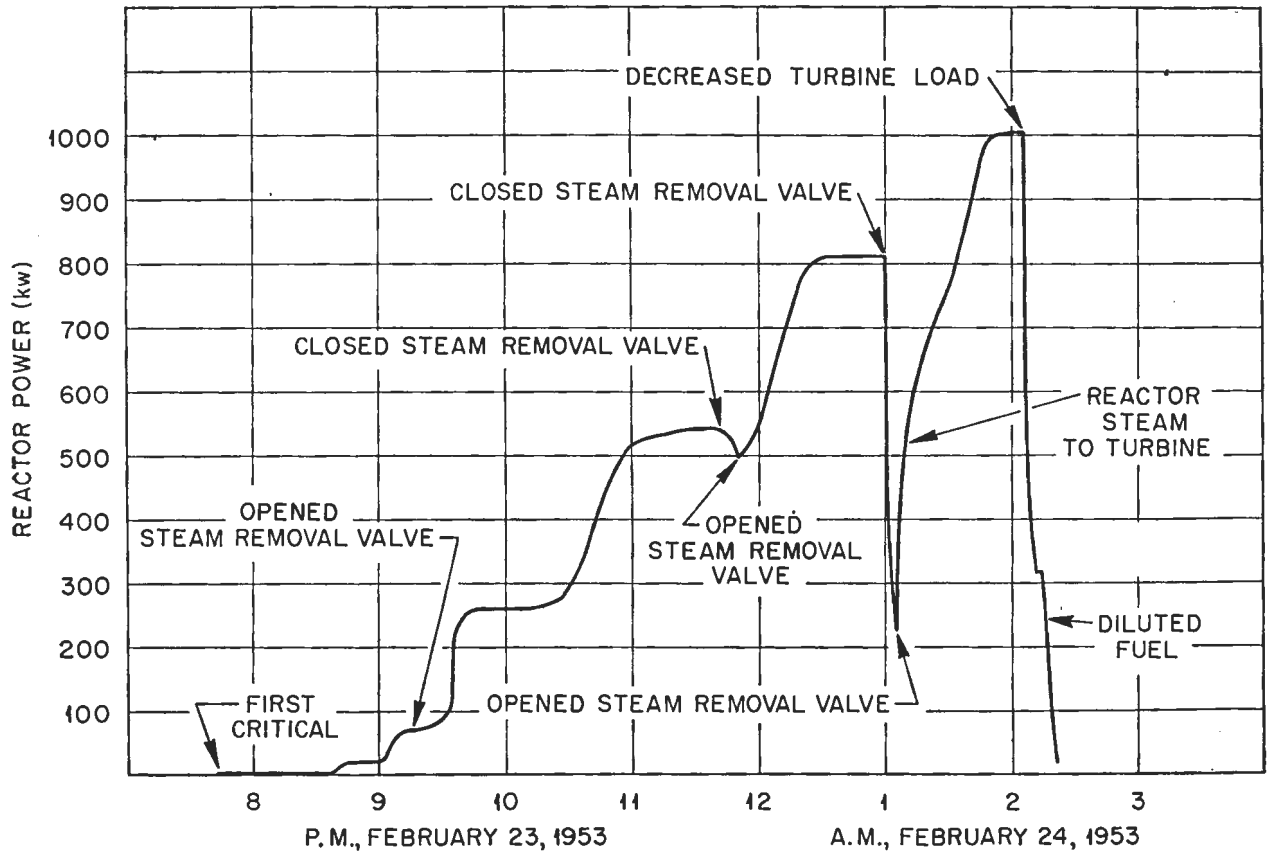


Figure 8. Initial operation of the Homogeneous Reactor Experiment at full power

sure systems. Dump valves are provided for this purpose but are not expected to be opened unless leakage of the fuel solution is detected. In such a circumstance the reactor contents can be discharged in approximately 10 minutes.

APPENDIX

High-Pressure Component Details

The Reactor Core

The reactor core is shown assembled with the blanket pressure vessel and the blast shield in Fig. 9. The core is a 32-inch diameter pear-shaped vessel fabricated of Zircaloy-2, a zirconium-tin alloy with a very low neutron absorption cross-section. The inlet or diffuser section is formed of ⅜-inch plate into two cones, the first with a 30° angle, and the second with a 90° angle. Nine perforated plates ⅛-inch thick are arranged in the conical section to distribute the fluid entering from the 3½-inch inlet pipe. The top half of the tank is a hemisphere with a 5/16-inch wall.

The results of numerous hydrostatic tests indicate that the Zircaloy-2 core tank will withstand an internal pressure greater than 900 lb/in². and an external pressure of over 400 lb/in². During operation of the reactor, pressure balancing instrumentation, relief valves, and a rupture disc will protect the thin core against pressure differences greater than 200 lb/in².

Pressure Vessel

Surrounding the reactor core is the blanket pressure vessel (60-inch inside diameter) which contains the 2000 lb/in². operating pressure. It is fabricated from two hemispheres of 4-inch thick carbon steel (ASTM A212-52at), clad with a 0.4-inch layer of type 347 stainless steel. This thickness of carbon steel is sufficient to limit the combined thermal and pressure stresses to 18,000 lb/in². at 10 Mw. Flanged nozzles with oval-ring gaskets are provided for the 3½-inch blanket solution inlet and exit lines. Flanges are also provided for the core inlet and exit lines. The bottom (inlet) flange is seal welded, but the top flange is removable.

Core-Pressure Vessel Assembly

Provision for differential expansion between the two vessels is provided by the 2-ply Zallea formed bellows at the upper end. The bellows is designed for continuous service at a 200 lb/in². differential pressure at 300°C and a total flexure of ⅜-inch. A similar bellows has withstood 1000 cycles under these conditions without failure. The normal differential pressure is less than 50 lb/in².

Blast Shield

The bombardment of carbon steels by fast neutrons has been reported to result in measurable embrittlement, in samples irradiated at 21 to 60°C. The

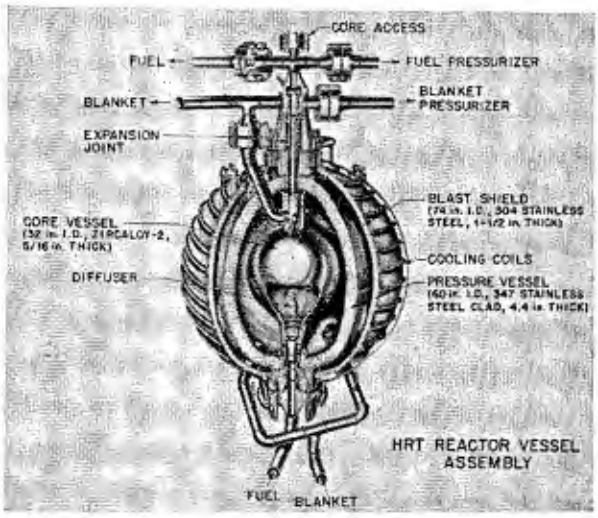


Figure 9. HRT reactor vessel assembly

damage is reduced, apparently by a healing effect, in the temperature range (260 to 300°C) of the Homogeneous Reactor Test. There is a remote possibility that such damage would make the pressure vessel susceptible to brittle-fracture. The possible damage to the gas-tight shield from such a violent accident would be sufficiently serious that provision for positive protection was considered necessary.

Protection has been provided by enclosing the pressure vessel in a 1½-inch thick stainless steel sphere, spaced 3 inches from the heavier vessel. This blast shield is designed to retard the very rapid expansion of the hot liquid from the pressure vessel, and to stop fragments of the pressure vessel which might break the shield seal or damage other equipment. It is not welded to the nozzles projecting from the pressure vessel, although clearances are minimized so that steam would be vented slowly in the shield.

Pressurizers

The pressurizer shown in Fig. 10 is designed to boil a small volume of fuel or blanket liquid at a temperature approximately 35°C greater than the maximum core temperature. By this means, steam at pressures as high as 2000 psi is accumulated in the vapor space at the top of the pressurizer vessel. The entire high-pressure circulating system communicates with this space and is thus subjected to whatever pressure exists there.

The heat required for the higher pressurizer temperature is supplied by an electric heater assembly clamped on the three 2-inch IPS pipe legs. Protection against overheating and damage to the pipes is provided by casting the Calrod heaters in a low-melting (450°C) aluminum alloy and by thermostat circuit breakers in each heater.

The pressurizer design is influenced by the requirement of balanced pressures between the core and blanket. During normal operation, the differential pressure between the two pressurizers is controlled by instruments to ± 20 lb/in². by varying the heater power input. In order to protect the relatively thin-walled core from damage by excessive pressure in case of instrument failure, the vapor regions of the core and blanket pressurizers are interconnected with 2 relief valves arranged to open at a pressure rise of 200 lb/in². in either direction. A 2-way rupture disc is also provided in case the relief valves fail.

The pressure balancing system is taxed most severely during a discharge of the fuel and blanket solutions to the dump tanks. In this situation it maintains a minimum pressure difference across the core wall by throttling the core and blanket dump valves.

Gas Separators

The gas separators, as illustrated in Fig. 11, provide an efficient device for separating undissolved gas from a flowing liquid stream. Each separator is capable of removing 2.4 cubic feet per minute with an efficiency of 98% and with maximum entrainment of 1 gallon per minute of liquid. However, only 0.9 cubic feet per minute must be removed during normal Homogeneous Reactor Test operation at 5 Mw, 300°C, and 2000 lb/in².

Let-Down Heat Exchanger

The let-down heat exchanger is designed to cool the mixture of gas, steam and liquid from the gas separator to a temperature of approximately 90°C so that the throttling valve at the discharge end of the heat exchanger will not be corroded excessively by high temperature fluid.

The mixture flows through a 35-foot long annulus formed by assembling ⅞-inch outside diameter tubing inside a 1-inch Schedule 80 pipe. Cooling is accomplished by the counter-current flow through the inner tube of solution from the dump tanks as it is fed back into the circulating system by the high-pressure feed pump. Additional cooling capacity is provided by an 8-foot long water jacket on the 1-inch pipe near the down-stream end of the heat exchanger.

Main Heat Exchangers

The design of the primary heat exchangers is shown in Fig. 12. Hot solution enters the upper side of the divided channel at 300°C, 2000 psi pressure and is circulated through 250 ⅜-inch outside diameter, 0.065-inch wall thickness U-tubes. The total heat transfer area is about 500 ft². Heat is removed from the hot solution to the feed water on the shell side at a rate of 5000 kw to produce approximately 1.62×10^4 lb/hour of steam at 520 lb/in². and 244°C Considerable attention has been given to the problem of obtaining the very highest quality tubing for these units to reduce the likelihood of tube failures and consequent leakage of reactor solution into the steam. The tubing was inspected by ultrasonic and magnetic flaw detectors capable of detecting imperfections as small as 0.002 in diameter. Following the bending

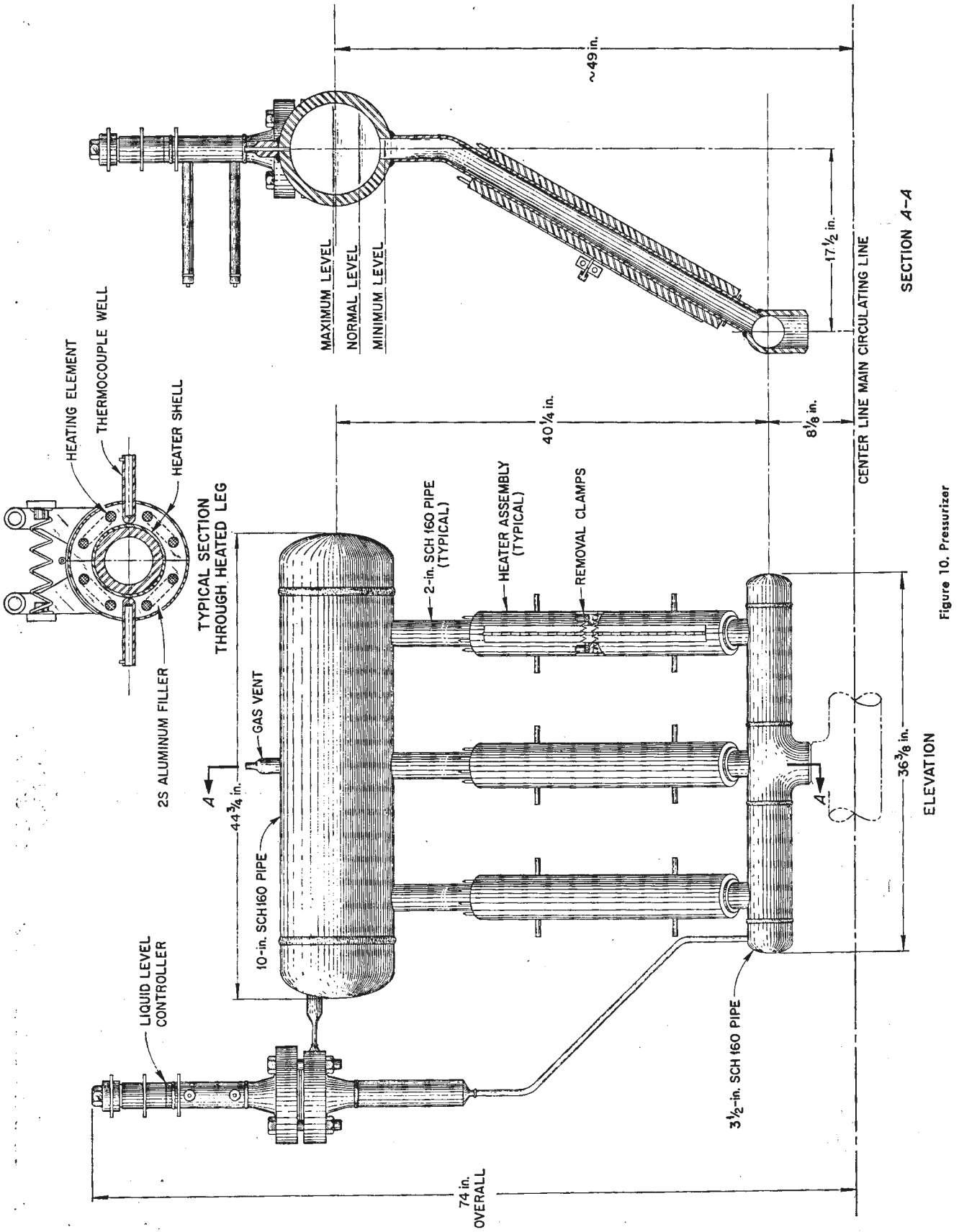


Figure 10. Pressurizer

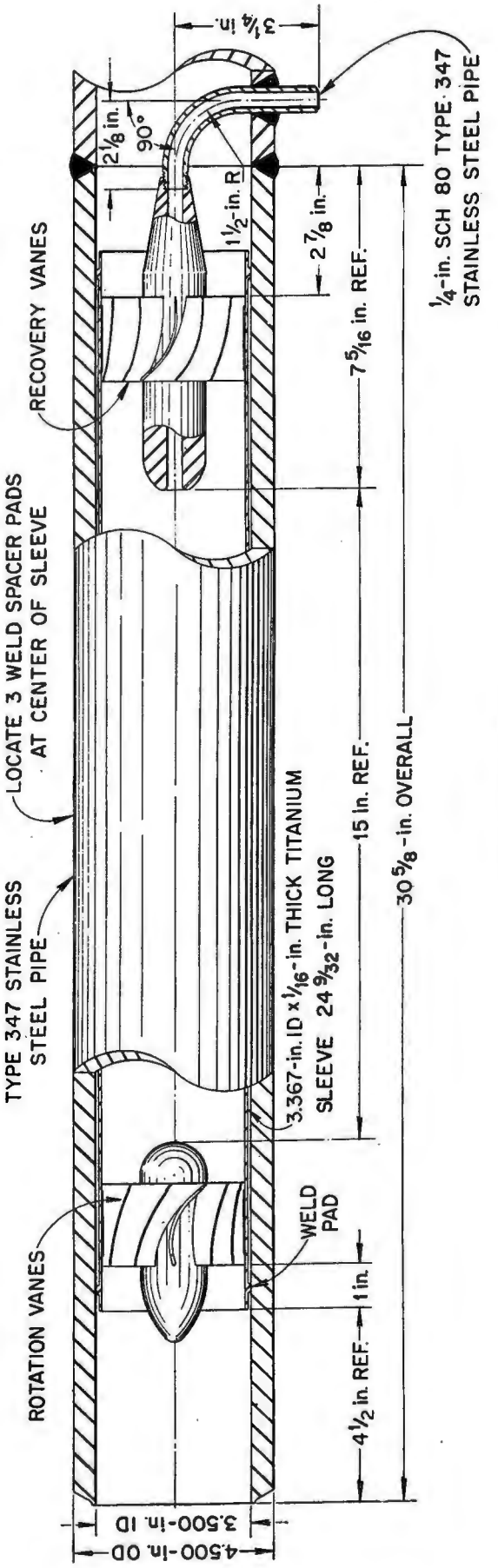


Figure 11. Gas separator

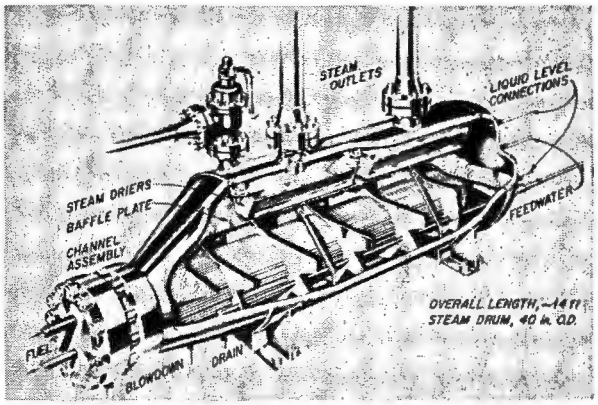


Figure 12. HRT 5 Mw steam generator

and annealing operations, each tube was inspected for surface defects with a liquid penetrant. Finally, each tube was subjected to a 4000 lb/in². hydrostatic test. After passing all of these tests, the tubes were rolled into the heat exchanger tube sheet, and then welded by an inert gas tungsten-arc process.

Main Circulating Pumps

The circulating pumps chosen for the Homogeneous Reactor Test are modifications of Westinghouse Electric Corporation Model 150C, which are totally enclosed, centrifugal pumps (Fig. 13). Both the rotor and stator are enclosed in stainless steel cans which are leak-tight and which permit the rotor to operate in the fluid being pumped. The impeller is attached directly to the rotor. The 400-gpm fuel circulation pump and the 230-gpm blanket pump are designed with the same motor, the only difference being the size and design of the impeller. High velocity areas in both pumps are lined with titanium.

Feed Pump

Liquid is pumped continuously from the dump tanks to the high-pressure circulating system by means of diaphragm-type injection pumps. These pumps operate on a positive displacement principle and make use of a 0.030-inch thick stainless steel diaphragm sealed between two dish-shaped pressure heads. The reactor fluid being pumped is on one side of the diaphragm and the hydraulic fluid is on the other. At 78 strokes per minute, the output of one head is 0.9 gallon per minute. Double-ball check valves are used on both the suction and discharge lines. This complete assembly is located inside the reactor shield.

A single head assembly is illustrated in Fig. 14 with a portion of the actuating device. The complete actuator is a Scott and Williams Hydropulse pumping unit which consists of a variable speed rotary oil pump for supplying high-pressure oil to a slide valve which alternately pressurizes and relieves a rubber pulsator (also shown in Fig. 14). The pipe connecting the pulsator to the diaphragm head is filled with D_2O, acting as a hydraulic fluid. Contraction and

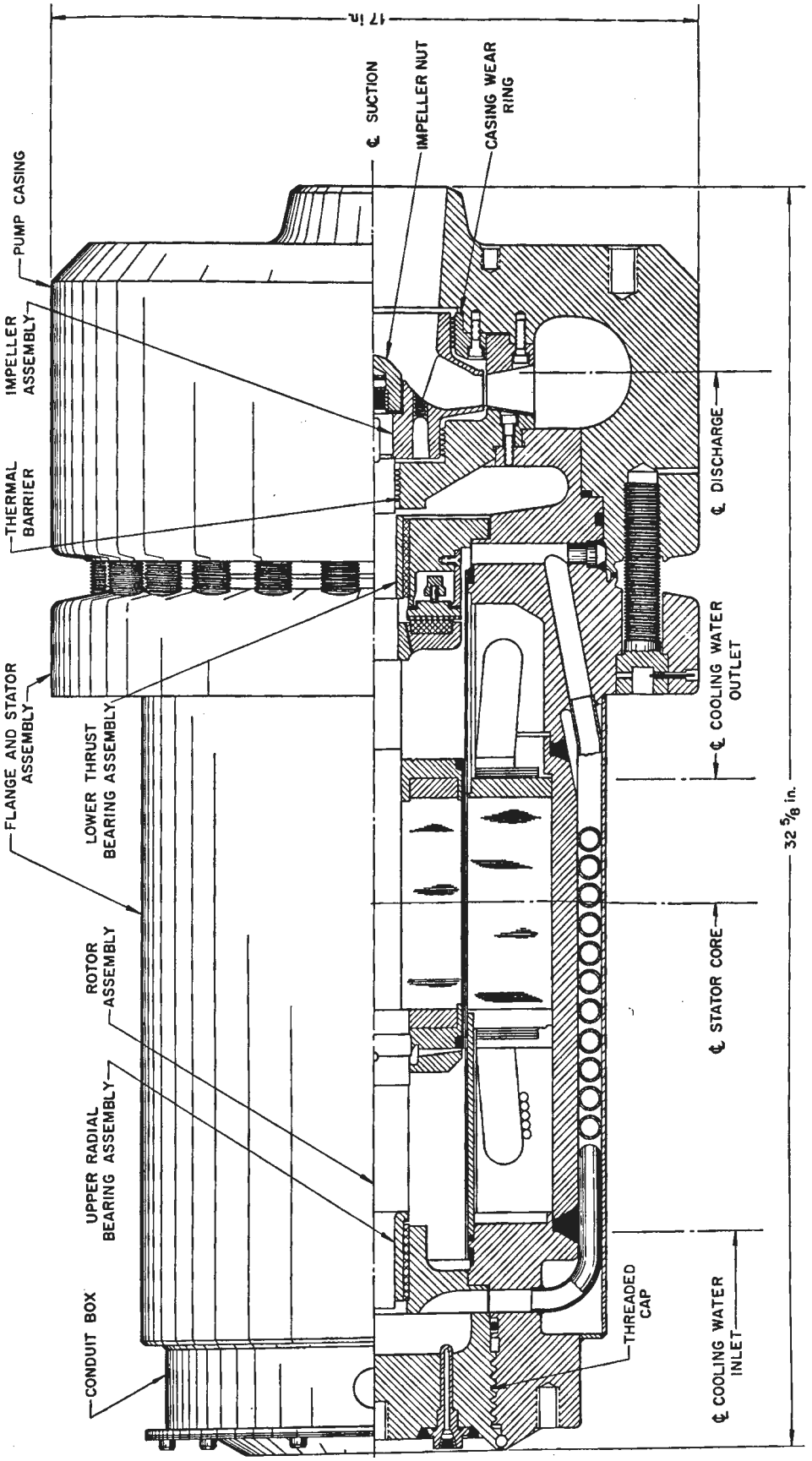


Figure 13. Circulating pump

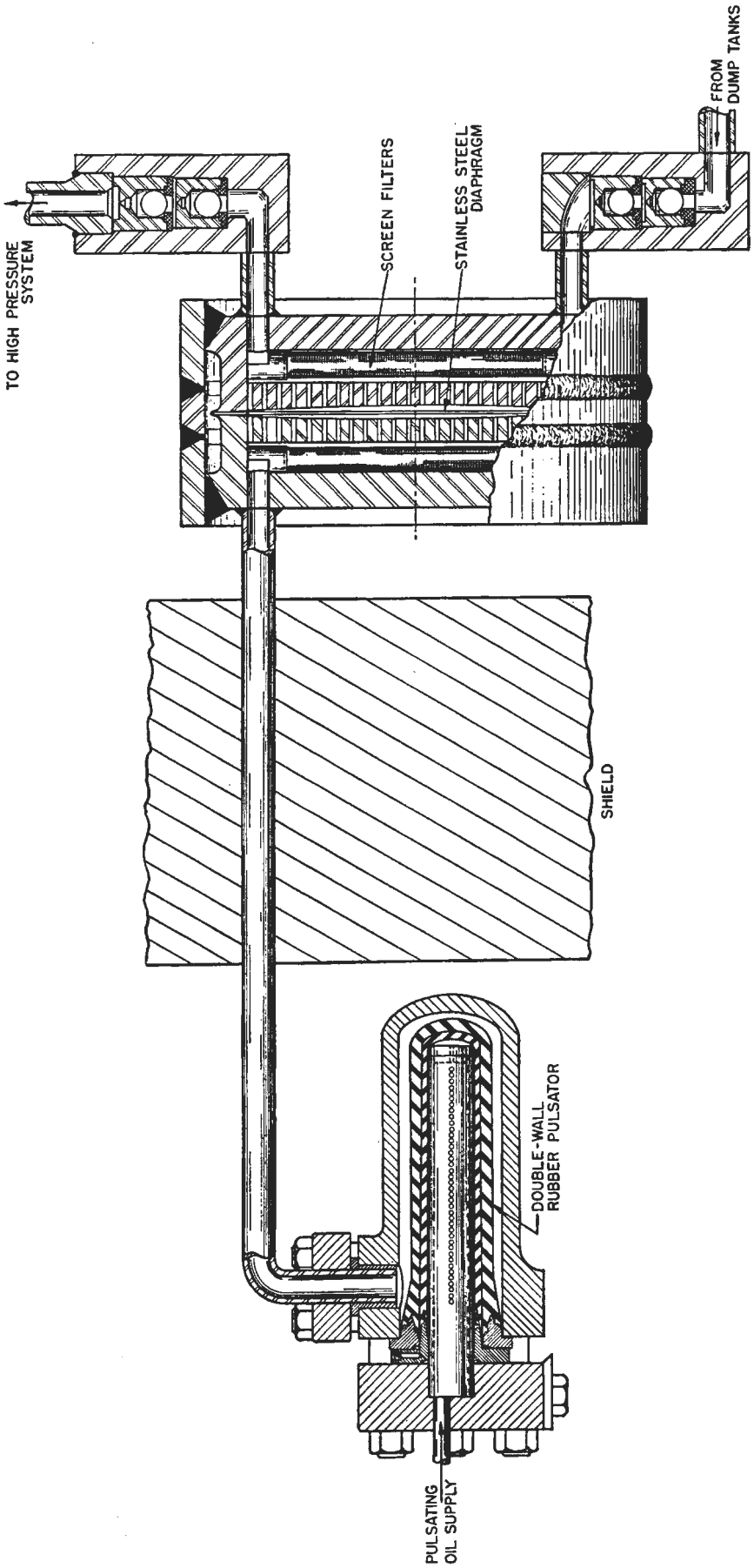


Figure 14. Injector feed pump

expansion of the rubber pulsator sleeves result in a backward-forward motion of the stainless steel diaphragm in the welded head, and a flow of reactor liquid past the discharge check valves.

Piping and Flanges

All of the process piping system is of welded construction, with flanges located to permit the removal of any piece of equipment. The piping in the high-pressure circulating system is either 3½ inch IPS, 0.469-inch wall thickness, or 4 inch IPS, 0.500-inch thickness. Only seamless pipe is permitted, and bends are used wherever possible to eliminate fittings and welds. The welds throughout the piping are made with exceptional care. Only the Heliarc process is allowed for the portion of a joint in contact with reactor liquid. All welds are inspected to rigid specifications by radiography and liquid penetrant techniques.

Flanges are the ring-joint type pictured in Fig. 15. A special feature of this design is the leakage detector provided by drilling into the bottom of the gasket groove and through the body of the gasket. This space and the transmission lines are filled with D_2O pressurized to 2500 lb/in^2. Any decrease in pressure indicating leakage past the sealing surface of the gasket activates an alarm so that an orderly reactor shut-down can be made. Flanges of this same design were used throughout the Homogeneous Reactor Experiment piping system and are specified for the Homogeneous Reactor Test because of their excellent performance.

Valves

All valves in contact with reactor fluids are of the bellows-sealed type and are actuated by a bellows air motor. The bellows seal on the valve shaft is backed up by an ordinary packing glad which is tight enough to prevent gross leakage if the bellows fails. As a consequence of the Homogeneous Reactor Experiment experience with valves which did not shut off completely, the seat design has been improved and a stronger air motor designed. Also, every valve is to be installed with flanges to permit replacement.

The high-pressure valve design is shown in Fig. 16 and the low-pressure valve in Fig. 17. The only valve larger than ½-inch IPS is the 1-inch blanket dump valve.

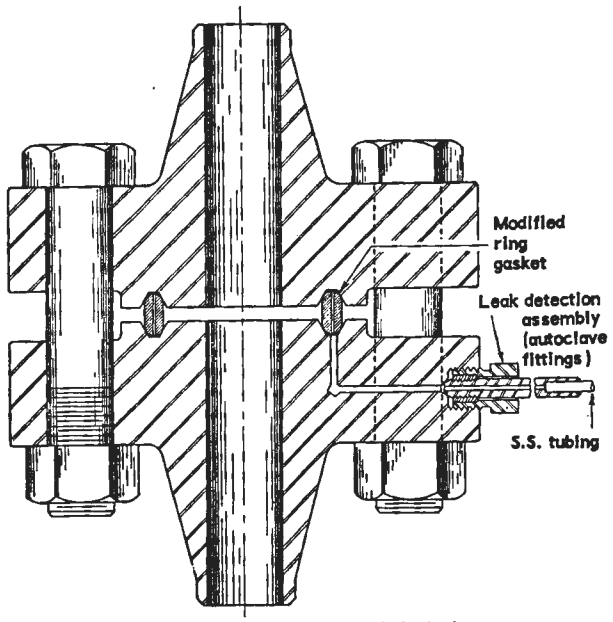


Figure 15. Ring joint flange with leak detector

LOW-PRESSURE COMPONENT DETAILS

Dump Tanks, Evaporator, and Entrainment Separator

The gas-liquid mixture from the let-down heat exchanger discharges into the entrainment separator above the dump tanks. The assembly of the dump tanks, evaporator, and entrainment separator is pictured in Fig. 18.

The dump tanks are of sufficient capacity and are horizontal to provide a receiver for all the liquid stored in the high-pressure system in which criticality cannot be attained. The two 16-foot long fuel tanks are 14-in. diameter; the blanket tanks are the same length but have a diameter of 24-in.

The evaporator consists of the two 4-inch steam jacketed pipes underneath. Its functions are to provide dilution steam to keep the deuterium-oxygen gases non-explosive, to concentrate the fuel for reactor start-up and for changing the operating temperature, and to circulate the contents of the tanks sufficiently for good mixing.

The entrainment separator is the vertical pipe section above the dump tanks. It contains baffles in the lower end for proper flow distribution. The upper section is packed with a 6-inch layer of coarse Yorkmesh gauze followed by a 6-inch depth of fine mesh. Entrainment less than 10 ppm has been measured in studies with the full-scale arrangement.

The storage tanks which are identical to the dump tanks are provided for isolated storage of the core and blanket liquids while repair work is performed on other parts of the system.

Catalytic Recombiners

Recombination of the radiolytic gases is accomplished in the catalytic bed shown in Fig. 19. The mixture of gas and steam rising from the dump tanks is contacted with the platinized alumina pellets contained by the perforated basket inside the recombiner shell. Thermocouples shown embedded in the bed will warn when the catalyst ceases to perform properly. The recombiner is provided with a high-pressure (200 lb/in^2.) steam coil to keep the bed dry when no recombination heat is available. The temperature rise expected during normal operation is 300°C.

Recombiner Condenser

The condenser shown in Fig. 19 directly downstream from the catalytic recombiner is designed to

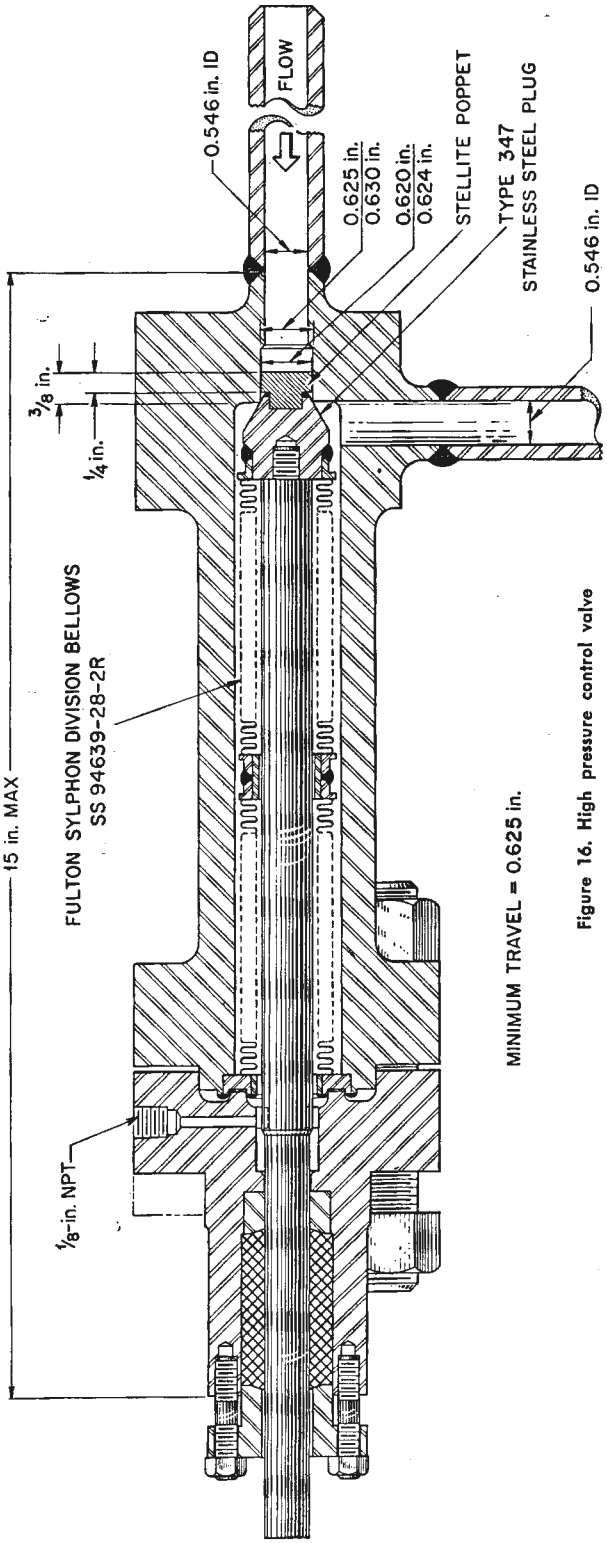


Figure 16. High pressure control valve

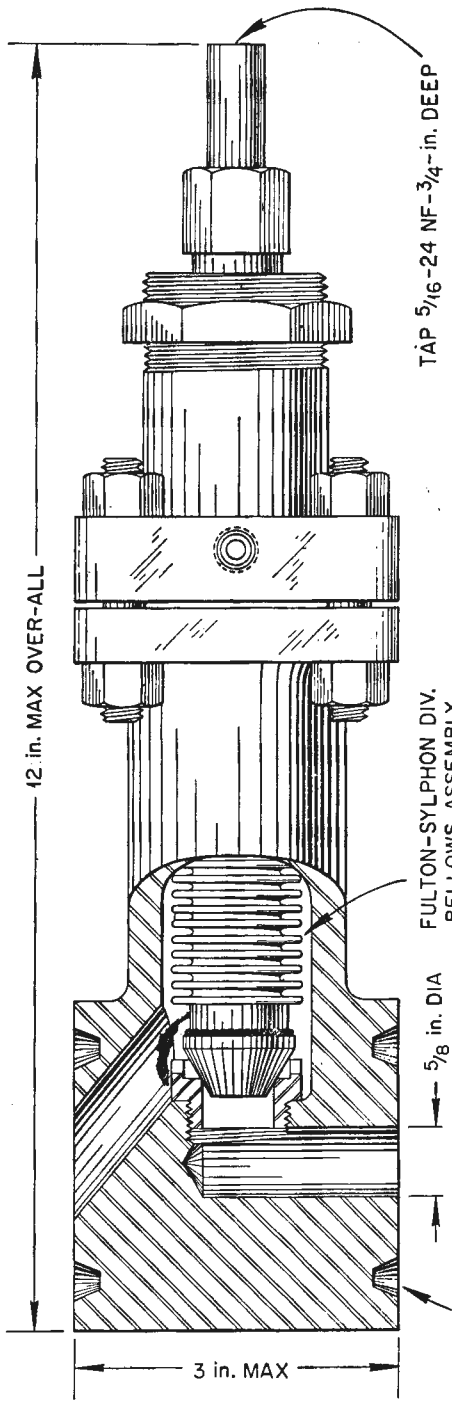


Figure 17. Low pressure control valve

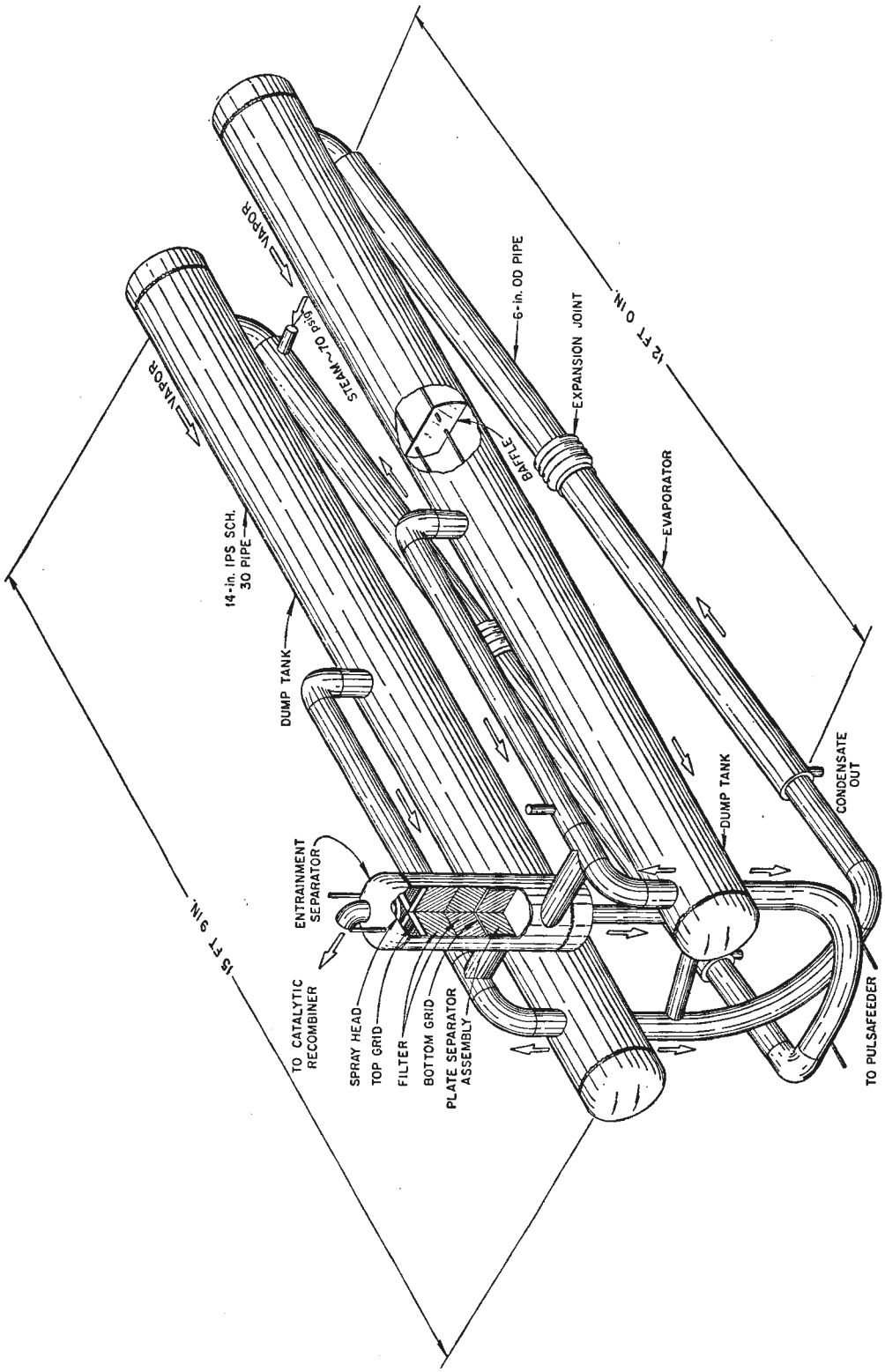


Figure 18. Dump tanks

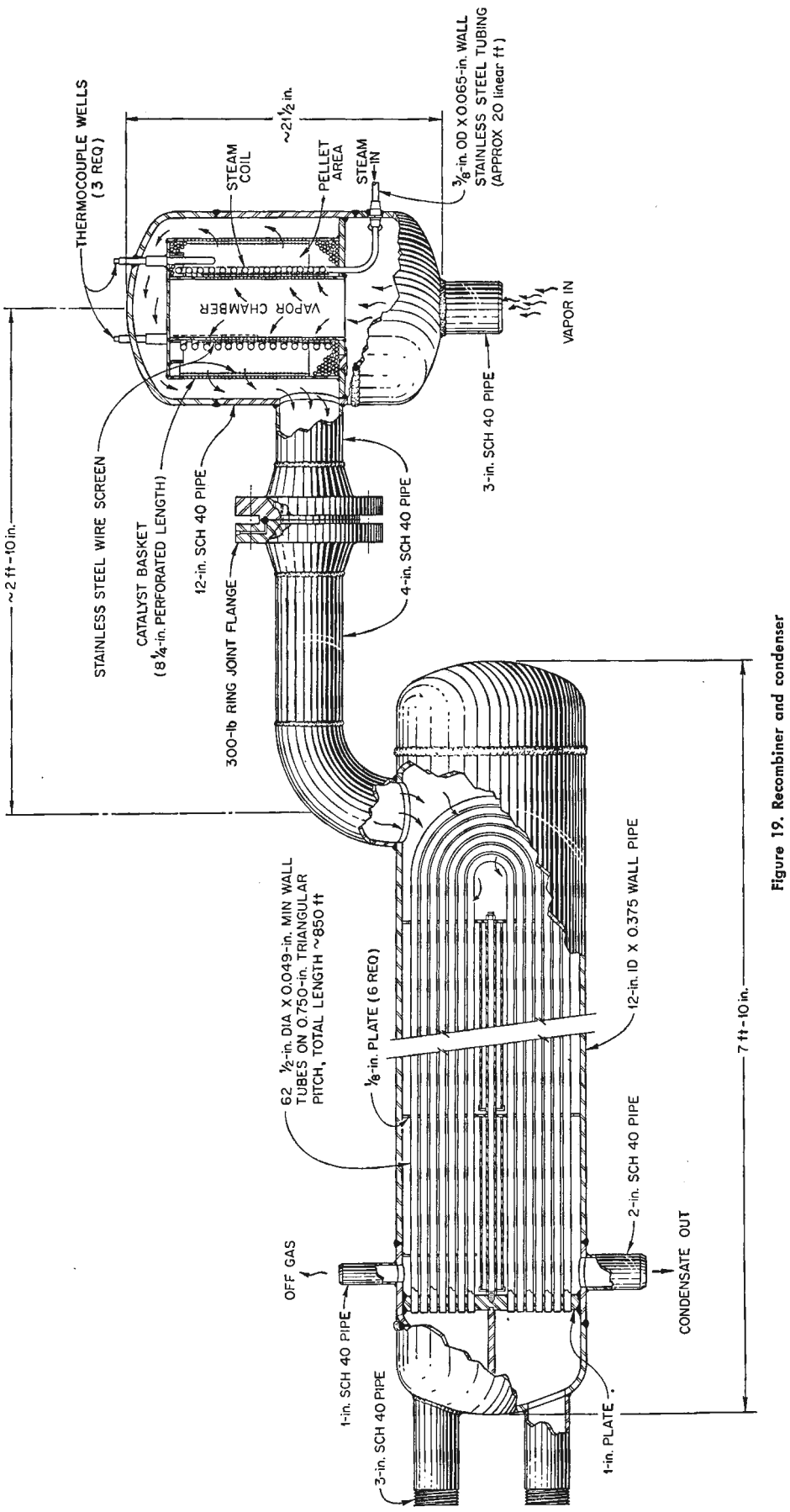


Figure 19. Recombiner and condenser

condense the D_2O vapor and cool the non-condensable gases released during a dump as well as during normal operation.

The design makes use of 62 U-tubes (½-in. diameter 18 gauge) with an average length of 13 feet, mounted inside a 12-inch diameter stainless steel shell. The U-tube arrangement minimizes thermal stresses and pressure drop. The non-condensable gases leaving the condenser are cooled sufficiently to impose only a small sensible heat load on the cold traps which follow the condenser.

The Charcoal Adsorbers

The activated carbon, on which the fission gases are adsorbed, is contained in an assembly of pipe lengths of diameter ½-inch to 6 inches, with a total length of 328 feet. The assembly is located in an underground pit with earth for shielding and is submerged in water for cooling. The gases enter the ½-inch pipe section, which is selected to give the necessary surface-to-volume ratio for heat removal. The total volume of carbon in a complete assembly is 14 cubic feet. Three such assemblies are provided.

The decay time expected with two of the three beds in parallel is 2200 hours for xenon and 180 hours for krypton, based on measurements made while the same beds were in use with the HRE. With 400 curies per second entering the beds from the reactor, these hold-up times are sufficient to reduce the activity discharged to a few curies per day at a 10 Mw reactor power, although the equilibrium activity of the beds will be about 2×10^6 curies.

Waste System

It is anticipated that, in the course of the experimental operation, small quantities of highly radioactive waste solutions containing enriched uranium will be accumulated. In order to recover the uranium and dispose of the waste safely, a system has been provided consisting of a small evaporator with a capacity of 1 gallon per minute, a 12,000 gallon storage tank, in addition to a 300,000 gallon storage pond.

In the event of a spill or leakage from the reactor, the radioactive solution is collected in sumps in the shield tank. From here it is jetted to the 12,000 gallon tank and then fed to the waste evaporator in 100 gallon batches. Condensate from the evaporator flows into the 300,000 gallon basin for dilution and decay. The basin also receives the water used for shielding during maintenance operations in the reactor tank. The effluent from the basin is continuously monitored and stopped if the radioactivity exceeds tolerance concentrations.

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Los Alamos Power Reactor Experiments

By Darol Froman, R. P. Hammond, and L. D. P. King,* USA

Two versions of relatively low power homogeneous reactors having the same basic fuel system characteristics are under construction and test. The objectives of this work are to achieve means of developing high pressure superheated steam appropriate for good turbine utilization, and to achieve simplicity by keeping the number of vessels, controls and pieces of special equipment to a minimum. These reactors, using enriched fuel, are considered to be prototypes of only relatively small nuclear power installations in which compactness, operating simplicity and low initial cost might outweigh the disadvantages of the use of enriched fuel and lack of breeding. The system is applicable to propulsion or stationary plants having electrical outputs up to 20,000 or 25,000 kilowatts.

The two reactor versions described here are alike in having the advantages common to all homogeneous reactors with respect to fuel handling and reprocessing. They both use phosphoric acid solutions of uranium, though at different acid concentrations, and in both the heat exchanger for removal of power is enclosed in the same pressure vessel as is the reacting fluid. In both reactors corrosion is prevented by using gold covering for the structural metals in contact with the solution and both are operated at sufficiently high temperatures to cause automatic recombination of the radiolytic gases. Thus, no special recombiner devices are necessary. Both reactors have high, negative temperature coefficients of reactivity and, consequently, can be thought of as thermostats supplying heat at essentially constant temperature as demanded by the power load. Heat is removed from both reactors by pumping water through coiled, gold clad, stainless steel tubing about 5 mm outside diameter. The water emerges as high pressure, fairly high temperature steam.

The two reactors differ mainly in that the first uses the more dilute acid, 50 per cent by weight, with consequent higher vapor pressures and the heat is transferred to the heat exchanger by circulation of the reactor fluid by means of an electrically driven, sealed rotor pump. In the second reactor the acid is about 95 per cent strength and the fluid circulates through the heat exchanger by the less efficient but simpler process of convection.

We will first examine some of the characteristics of the forced circulation reactor. Figure 1 shows the construction of the reactor. In the main, it consists of a long stainless steel pressure vessel about 38 cm inner diameter with 8 cm walls. The lower portion of the vessel serves as a reservoir for fuel solution and it is divided into two noncritical regions by a cylindrical shell containing boron. The soup, as the fuel solution is usually called, has a high coefficient of thermal expansion which we will discuss in more detail later. However, when cold the solution occupies only the noncritical lower region and a little of the proper reactor region, and the reactor can be maintained at room temperature in a nonreactive condition when the boron control rods are in the platinum clad thimbles shown in the diagram. On removal of some control rods, the reactor starts up and self heat-

*Los Alamos Scientific Laboratory.

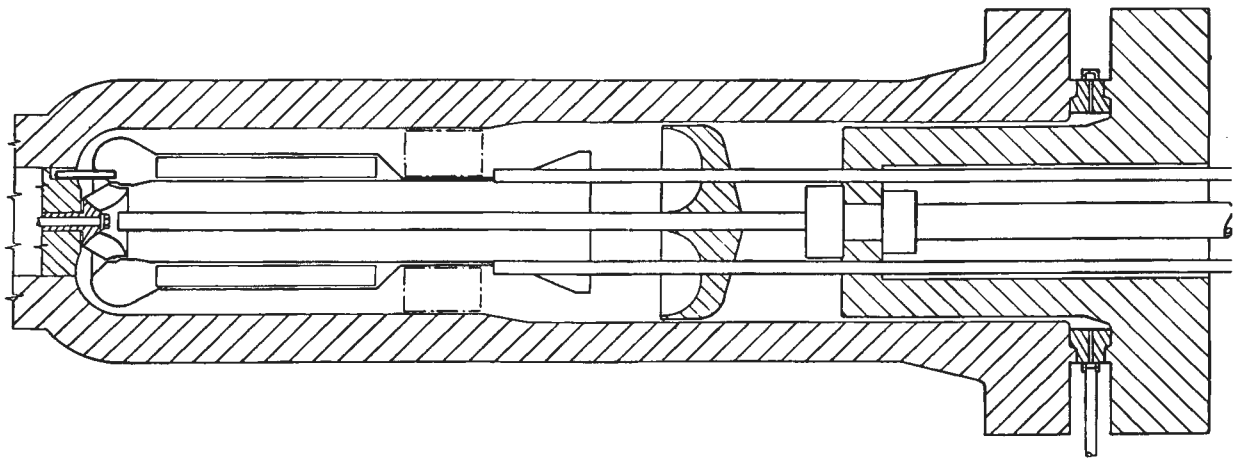


Figure 1. Diagram of forced circulation, 2 Mw homogeneous reactor

ing expands the liquid to fill the reacting region proper at about 430°C. Thus, the system can be closed and it is not necessary to add or remove soup in order to change from shut-down at room temperature to operating conditions. The circulating pump is used to force the fluid through the coiled heat exchanger at about 4500 liters per minute.

The fuel solution contains per liter 140 grams of uranium enriched to about 90 per cent in U^{235}. This requires a total of about 9 kg of uranium, 4.2 of which are in the reactor region. The power density in the 43 liters of the reactor proper when in operation at 2 megawatts is about 46 kilowatts per liter. The average thermal flux in the core is about 2×10^{13} neutrons per cm^2-second and the fast neutron flux is nearly 10^{14} neutrons per cm^2-second in the core.

Figure 2 shows the thermal expansion characteristics of the liquid in a closed container filled at room temperature to 59 per cent. Expansion of the liquid into the region above the top baffle and around the inverted "top hat" reactor lid reduces the criticality markedly because neutrons from the upper region are quite well screened from the reactor region proper. This effect accentuates the normal negative temperature coefficient of reactivity which would appear in an unbaffled, cylindrical container.

Figure 3 shows the reproduction constant, k, as a function of temperature as the fuel solution is heated from room temperature (reactor 59 per cent filled) to operating temperature and the solution is expanding to fill the vessel. This gives the values k would assume if control rods were not moved during heating and all the liquid was heated uniformly. A maximum temperature difference between the upper and lower regions of only 50°C is required at the peak of the curve in Fig. 3 to remove all excess k in the upper region. Thus, when the upper region of the reactor is made just critical by removal of control rods at room temperature, if the pump is idle, heating in this region will keep the reactivity very near unity and the entire solution will be slowly heated by convection currents.

Figure 4 shows the effect of sudden changes in power demand which might come about by sudden opening or closing of the steam throttle valve to a turbine. One curve shows the power surge (left hand ordinate) and the other shows fuel temperature fluctuations (right hand ordinate) when power demand is increased from 100 watts to 1000 kilowatts. Figure 5 shows the corresponding functions when the power demand is suddenly decreased from 2000 to 1000 kilowatts. It is easily seen that the reactor automatically adjusts itself to the power demand without making troublesome temperature and consequent pressure excursions.

The pressure exerted by the solution in this reactor is about 250 kilograms per cm^2. (The abstract is in error in stating a lower value.) This high pressure makes necessary the thick walls of the reactor vessel. The outside of the vessel is covered with a thin insulator to allow a controlled heat leak of about 50 kilowatts to the water bath in which the reactor is placed. The very thick steel walls introduce a definite power limitation of a few megawatts on the reactor because of gamma-ray heating in them. This heating produces thermal stresses comparable with the internal pressure stresses at an operating power of about 2 megawatts. Higher powers can be obtained with this type reactor if the pressure walls are internally cooled.

In this reactor the many internal, rather complicated shapes are covered by gold plating except for the heat exchanger tubing which is gold clad by means of a drawing technique, and the control rod thimbles which are clad with platinum because the neutron capture in gold would reduce the effectiveness of the rods. It is difficult to obtain gold plating completely free of pinholes and other techniques of covering may be necessary to obtain long, corrosion-free life of the reactor. The uranium is in solution in the form of $(UO_2)^{++}$ ion and an excess pressure of oxygen is maintained in the vessel to keep the uranium in solution and prevent excessive corrosion of any exposed stainless steel.

Since the above reactor was designed, it has been found possible to keep uranium in solution in the

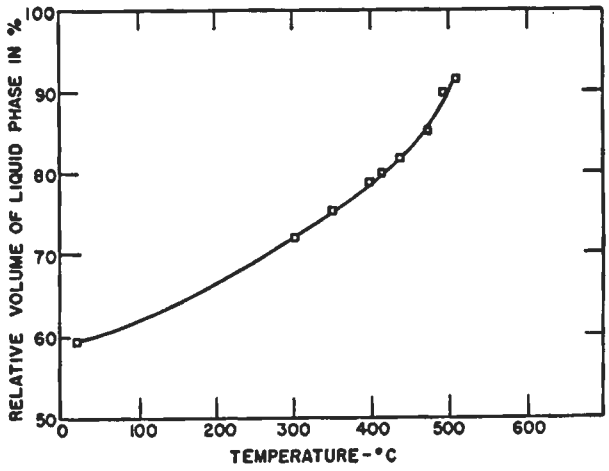


Figure 2. Thermal expansion characteristic of 50% phosphoric acid-uranium solution in closed vessel filled to 59% at room temperature

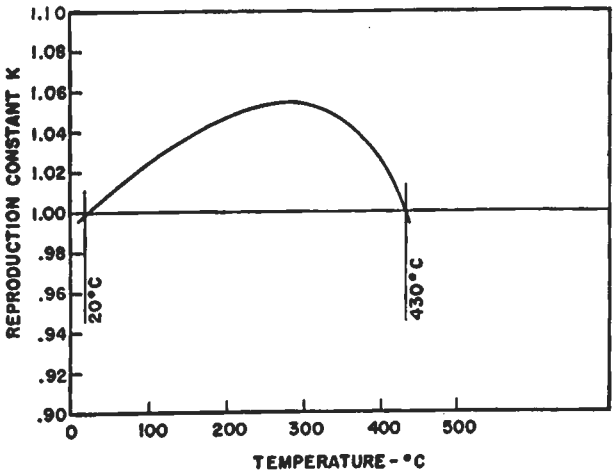


Figure 3. Reproduction constant, k, in reactor as a function of temperature of entire volume of solution

form of U^{++++} ion in high concentration phosphoric acid under a reducing atmosphere of hydrogen. The vapor pressures encountered with this fuel solution at operating temperatures of 350°C to 450°C are 40 or 50 kilograms per cm^2 rather than about 250 as observed with the weaker acid. This means that pressure vessels can be built with much thinner walls and the gamma-ray heating effect in the walls becomes relatively unimportant. Also in this reducing atmosphere, silver is perhaps more resistant to corrosion than is stainless steel in the oxidizing atmosphere. Consequently gold plated silver coatings and, of course, pure gold cladding, form good protective coats.

The advantages of the concentrated acid fuel are so clear that a long term test to study corrosion and other aging effects has been begun with this system. It appeared that the most meaningful test could be made with an operating reactor and the simplest possible system is being used. The complications of a fuel circulating pump are avoided by allowing the soup to move through the heat exchanger by convection. In this way, with our choice of the size of heat exchanger, about 1 megawatt of heat can be removed. This is sufficient to show the effects, if any, of radiation on corrosion, and the system even at this size may have some special, practical applications.

Figure 6 shows schematically the parts of this reactor. The fuel is introduced into the reactor, and

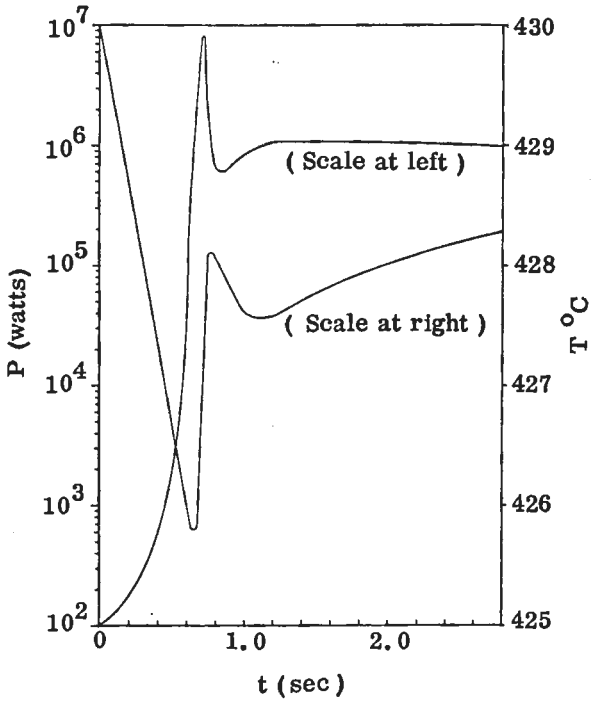


Figure 4. Instantaneous power development and temperature following an abrupt increase in heat removal rate from 10^2 to 10^6 watts

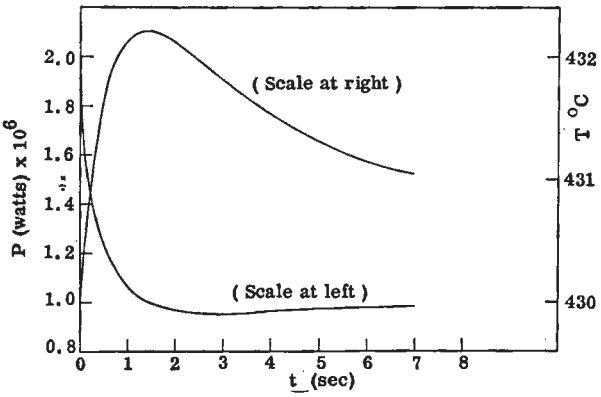


Figure 5. Instantaneous power development and temperature following an abrupt decrease in heat removal rate from 2 × 10^6 to 1 × 10^6 watts

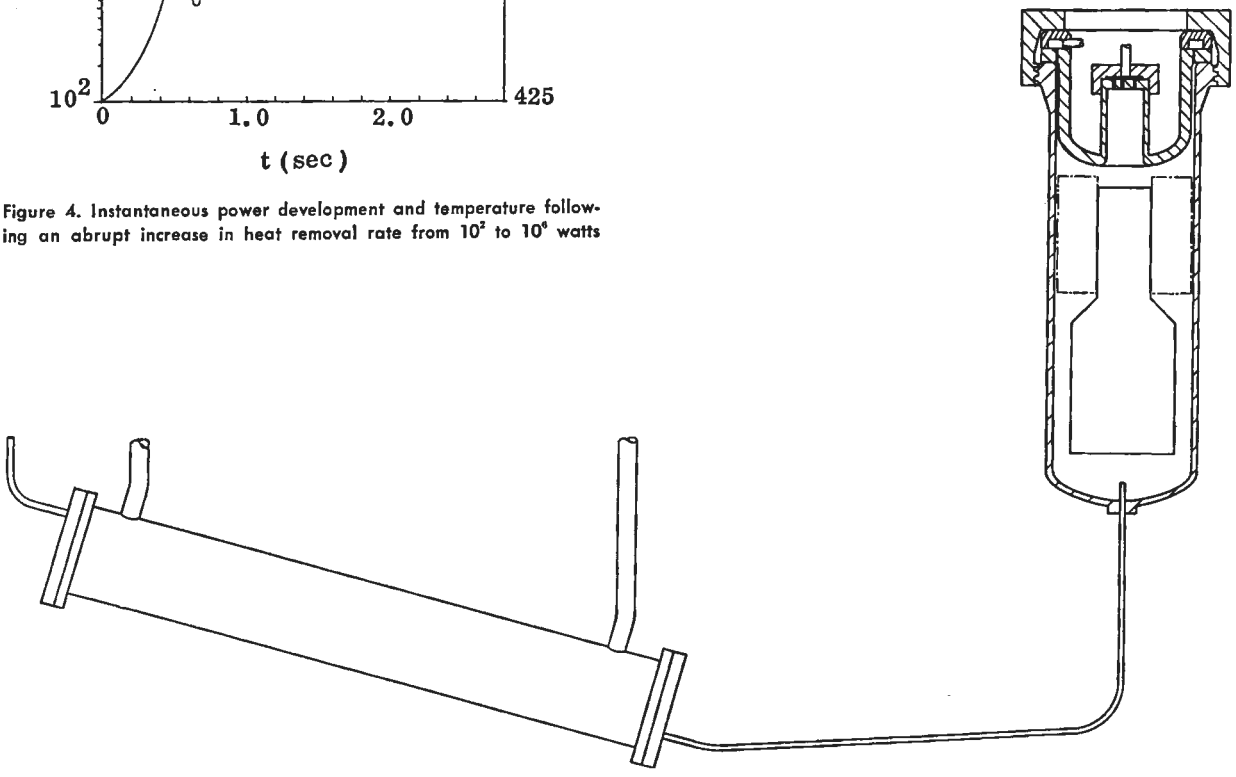


Figure 6. Diagram of convection cooled, 1 Mw homogeneous reactor

held there, by means of gas pressure above the solution in the storage tank which has a noncritical shape. If for some unlikely reason the reactor should tend to run away, rising temperatures would increase the vapor pressure in the reactor forcing solution out of it into the storage tank and reducing the criticality. This effect is in addition to the normal negative temperature coefficient of reactivity which is augmented also in this case by the geometry of the cover.

The reactor has a volume of 105 liters, the core region containing 63 liters. The fuel contains about 60 grams per liter of uranium, the power density is about 16 kilowatts per liter and the thermal flux is about 10^{13} neutrons per cm^2-second in the core.

It will be noticed that the reactor has no internal control rods but there is a graphite neutron reflector around it. A portion of this graphite reflector is moveable for control. It appears that even this is not absolutely essential. By introducing an appropriate amount of burnable poison, such as boron, into the solution and allowing slight changes in solution temperature, the burn out rate of fuel can be nearly compensated so that the reactor could operate for years without having to make external adjustments to the reactivity in any way. The automatic high temperature coefficient of reactivity will compensate for the slight mismatch of fuel and poison burn out rates by temperature changes of only a few degrees. This seems to be about the simplest possible reactor system. It does not require an operator and there are no electrical or mechanical servo mechanisms or operating devices which might go wrong and cause trouble. The reactor simply supplies the power load demanded up to its maximum capability with no attention except that every couple of years additional fuel must be added or the solution removed for reprocessing and replaced by new fuel solution.

Record of Proceedings of Session 13A

MONDAY MORNING, 15 AUGUST 1955

Chairman: Mr. E. P. Wigner (USA)

Vice-Chairman: Mr. M. S. Vallarta (Mexico)

Scientific Secretaries: Messrs. J. Goedkoop and M. Trocheris

PROGRAMME

P/815 Description of the pressurized water reactor (PWR) power plant at Shippingport, Pa ..... J. W. Simpson *et al.*
DISCUSSION

P/879 Preliminary study of experimental pressurized heavy water reactor ..... O. Dahl
DISCUSSION

P/497 The engineering design of a prototype boiling water reactor power plant ..... J. M. Harrer *et al.*
DISCUSSION

P/498 The homogeneous reactor test ..... S. E. Beall and J. A. Swartout
DISCUSSION

Mr. SIMPSON (USA) presented paper P/815.

DISCUSSION ON PAPER P/815

Mr. LEWIS (Canada): Why has the operating pressure been chosen as high as 2000 psi? What troubles would be anticipated if this were reduced, for example, to 1500 psi?

Mr. SIMPSON (USA): I assume that this question is based upon it operating at the same temperatures as the design. This gives an added margin for transient load swings which might take the pressure up higher. It gives better operation of the system for loss of coolant flow by accidents; that is, if the coolant flow should cease due to the loss of all pumps simultaneously, then, in the static case, you could operate a power plant nearer to the upper limit in heat flux without getting boiling under adverse operating conditions.

Mr. WARD (Canada): This question refers to assemblies used in the blanket. What is expected to limit the useful life of these elements? What is known about the pressure buildup due to irradiation and production of gaseous fission products, particularly with such oxide fuel elements?

Mr. SIMPSON (USA): The specification lifetime is 3000 megawatt days per ton average. Obviously, we expect to meet the design lifetime. We certainly hope that the fuel elements will last considerably longer than this. As for the gaseous pressure buildup due to fission products, we have studied this problem upon a calculation and analytical basis and find that the pressures created are so low as to cause no trouble with the thickness of zirconium plating used for the oxide elements.

Mr. GOODLET (UK): If boiling reactors are regarded as a possibility, is it really necessary to prevent surface boiling in a reactor in which the heat is removed by circulating the water, by applying an overpressure? Suppose a little surface boiling and recondensation does occur in such a system, will conditions be any different from those of an ordinary forced circulation boiler?

I should also like to point out a certain discrepancy between views in this paper and views in other papers we have heard. It is considered with the present state of knowledge that boiling should not be permitted during steady state operations. In paper P/815, it is said that "both local and bulk boiling present problems of pitting and crud depositions on fuel element surfaces." This of course would reflect on the design of boiling reactors.

Mr. SIMPSON (USA): I would say that it is felt by us to be quite probable that boiling, particularly local boiling, will not cause trouble. The design of the PWR is certainly ultra-conservative. This means that it can be extrapolated to a much higher performance rating than has been indicated.

Mr. BOGAARDT (Netherlands): If a stepwise decrease of the required steam output of the reactor occurs, could it be that a temporary complete shutdown of the reactor is imposed by the automatic control system; and, if so, could that system cope with subsequent pile startup?

Mr. SIMPSON (USA): I do not believe that the

system would shut down. However, if it did, the control system could certainly cope with the pile startup. There is sufficient provision for that.

Mr. DESCANS (France): I should like to know at what pressure the tank containing the reactor is tested. Paper P/815 mentions two pressures—an operating pressure of 1400 kg/cm² and a designed pressure, which I take to be the test pressure, of 1750 kg/cm². Is there a third pressure at which the tank is actually tested?

My second question is this: How is a tank in operation inspected for wear and for the determination of what fissuring and corrosion may have occurred? Is any provision made for this?

Mr. SIMPSON (USA): In the first part of the question, the normal operating pressure is 2000 lb/in² in the reactor vessel. This is maintained by making steam at 636°F, 2000 lb/in² and with pressurizing tanks. According to the American Society of Mechanical Engineering Boiler Code, the design pressure for a 2000 lb operating pressure is 2500 lb/in² to take into consideration transients which may occur in the operation. The reactor vessel is tested with a cold hydrostatic pressure of 3750 lb/in² which is 1½ times the design pressure.

I am afraid I did not understand the last part of the question. We have no remote control devices built in to do any maintenance on the pressure vessel. I am not sure what the reservoir is. If the reservoir is the pressuring tank, we also have no built-in remote control devices for doing maintenance there. We do not believe that the corrosion will be sufficient to require such devices for normal maintenance.

Mr. DALTON (Australia): I have four questions. First, what are the differential cooling arrangements for the seeded assemblies in the core?

Mr. SIMPSON (USA): The seeded assembly just gets a higher velocity of flow—20 feet per second as against 10 feet per second in the blanket areas. The blanket areas are orificed, and there is a higher heat flux in the seed area.

Mr. DALTON (Australia): Is that due to an arrangement of larger channels?

Mr. SIMPSON (USA): Due to an arrangement where?

Mr. DALTON (Australia): In the high flux area and the high enrichment area.

Mr. SIMPSON (USA): The orifices to cut down the flow are at the entrance of the blanket assemblies. This is just full flow permitted through the seed assemblies. There are more square feet of heat transfer area than in the case of the blanket due to the use of the plates rather than tubes.

Mr. DALTON (Australia): My second question is: What is the arrangement of the pellets in the fuel element and what is the over-all density of uranium in the fuel elements?

Mr. SIMPSON (USA): I do not believe I remember the over-all density. The pellets themselves are in excess of 90 per cent of theoretical uranium oxide density. The pellets are rather closely packed in the tube.

Mr. DALTON (Australia): The third question is: What is the initial excess reactivity in this reactor, and how much of that is taken up by your burnable poisons?

Mr. SIMPSON (USA): I believe that the answer to that question will be given in a paper that will be presented on the physics of the PWR reactor at a session on Friday.* I do not know the exact answer to that.

Mr. DALTON (Australia): The fourth question is: What do you estimate to be the cost per kilowatt installed, and what do you reckon to be the cost per kilowatt-hour of power generated?

Mr. SIMPSON (USA): Well, of course, one of the main reasons for building the PWR reactor was to determine what such reactors do cost, and to learn enough about the production of the equipment to get such an idea. So much depends upon the basis on which you are making your prediction—as to whether it includes factors that are put into the plant strictly for developmental purposes or factors, such as two different designs in steam generators, that are there to get more experience. All of these things influence the cost to a large measure. I therefore do not believe that I could give a meaningful answer as to the cost per installed kilowatt. It will be definitely higher in the PWR than a comparable conventional steam power plant today. Likewise, the cost in mills per kilowatt-hour of the PWR as it will be built and is being built will not in any sense be competitive with the cost of a coal-fired plant. We do, however, expect to learn a sufficient amount so that in an extrapolated plant from this we can build a pressurized water reactor that may come very close to, or be competitive with, the cost of coal in the future.

Mr. BOWDEN (UK): I want to ask whether consideration has been given to the economics of a separate fired superheater in the plant layout. That is my first question.

Secondly, could we be told the test code that is being used for the construction and testing of the pressure vessel?

Thirdly, could we have some details of the type of joint that is used in the main reactor pressure vessel?

Mr. SIMPSON (USA): The answer to the first question: The use of an oil-fired superheater was not considered in the layout. It was considered in the earlier design. The advantages appeared to be marginal. It could conceivably have advantages, but it was an additional complicating factor which did not add to the primary knowledge that the PWR plant was

* P/601, "Pressurized Water Reactor (PWR) Critical and Exponential Experiments," by S. Krasik and A. Radkowsky. Volume V, Session 21A, these Proceedings.

designed to give—that is, knowledge on the operation of the reactor power plant itself.

As to the welded joints in the reactor vessel, I assume you mean where the main 8½ inch plates are welded together to form the cylindrical section. I am afraid I could not go into enough detail to be very helpful on that. It is a case where the carbon steel is welded by rather conventional welding practices, and then the cladding on the inside is welded over, with different types of welding rod being used in the transition zone to give good metallurgical properties at the point and to reduce the micro-cracks that might be stress-raising points to cause fracture.

I did not get the middle question. I think it was: What is the code to which the pressure vessel was designed?

Mr. BOWDEN (UK): Yes.

Mr. SIMPSON (USA): The American Society of Mechanical Engineering Boiler Code.

Mr. BOWDEN (UK): And my last question related to the details of construction of the bolted joint on the dome of the pressure vessel, rather than to the welded joint.

Mr. SIMPSON (USA): That is a seal-welded joint.

The CHAIRMAN: We have time for one last question, and I must refer all other questions to personal discussions. We are running a bit behind time. The last question is by Sir John Cockcroft of the United Kingdom.

Sir John COCKCROFT (UK): How do you propose to inspect for corrosion of the pressure vessel after periods of operation?

Mr. SIMPSON (USA): We expect, of course, to inspect thoroughly during all stages of construction. Prior to the time that the thermal shields are put in we expect to inspect the carbon steel by all the techniques of X-raying, ultrasonics, and so forth. A considerable body of test information indicates that the cladding of the vessel, which is a stainless steel, does not corrode. The vessel is designed without taking into consideration the strength of the cladding in selecting the strength of the pressure vessel. The thermal shields can be removed from the vessel when a core is removed and, although difficult, it would be possible to look in and determine the condition of the surface.

In addition to that, by merely analyzing the water for the elements that are contained in the stainless steel, we can tell whether any unusual amount of corrosion is taking place. This obviously would not tell about a pitting in a point. This is a definite problem and it is one that does give us concern.

Mr. DAHL (Norway) presented paper P/879.

DISCUSSION ON PAPER P/879

Mr. DRALEY (USA): Are you doing anything to the water in the way of controls, or adding to the water to control the corrosion of the aluminum in your reactor?

Mr. DAHL (Norway): Our basic idea is to establish this experiment and to install full monitoring for the rods. We will then see how far and how fast to push these rods. We have not yet taken any decision as to how to treat this aluminum. In addition, we have made no decisions yet on how we will alloy the aluminum. As the rods will be the last units to be put into the reactor we feel that we have time to consider that question further. We are starting out with a relatively simple basic conception so that we may decide on the solution later. We have been conducting some corrosion experiments in our laboratory, and think we will be able to install aluminum rods if we watch their performance carefully.

Mr. HARRER (USA) presented Paper P/497.

DISCUSSION ON PAPER P/497

Mr. SCHMID (Switzerland): I should like to ask Mr. Harrer the following two questions: First, for what reasons do your control elements contain hafnium which is known as a strong resonance absorber; and secondly, do they contain hafnium exclusively or were thermal absorbers as boron or cadmium contained also?

Mr. HARRER (USA): The control element is hafnium. There is no additional absorber added to this element. I believe that the physics of this control are very good. I am not entirely familiar with the calculations which lead to the desirability of hafnium control rods. However, I understand from our physics people that hafnium is the most desirable type of element for the control rod for this type of reactor.

Mr. TOPSOE (Denmark): I should like to put a question on the stability of direct boiling reactors, both of the homogeneous type and of the heterogeneous type, to Mr. Harrer, realizing that it could be put to several other authors of papers presented last week including Mr. Suvorov and Mr. Swartout.*

If you consider a given time when the power level as a consequence of steam production in the reactor is higher than normal, you will increase the void space and, as we all know and have seen, the reactivity will thereby decrease. The steam production will consequently decrease and the reactivity will go up again. This could manage to cause a cycling effect whereby severe hydrodynamic problems would have to be overcome. For example, in the stream of a homogeneous reactor you might imagine a severe hammering effect, and with heterogeneous reactors you might imagine severe damage to core construction and fuel elements. I should like to ask whether such hydrodynamic cycling effects have been considered and what results have been derived.

Mr. HARRER (USA): The question of the effect of pressure changes on void content and then the void content on power, and again the power on void content is one of the factors which caused us considerable

*P/624 and P/496.

concern before we built a boiling reactor. It is now a matter of record that we have wide experience with the operation of boiling reactors. We have found that if we limit ourselves to the power delivery that we have called for here, we are well within the capability of the reactor to operate in a stable manner. It is further the opinion of the group on this and associated projects that nobody can really predict what a boiling reactor will do until that reactor or something quite similar to it has been built so that it can be operated directly in the field and experience gained. All computations made prior to our having run a boiling reactor indicated that the system could be stable. On the other hand, however, the computations were made in such a way that we were trying to convince ourselves that it would be stable. Therefore, one could easily pick holes in all sorts of computations or qualitative arguments. The only result that we can point to, I think, is the direct experiment. Having run such reactors, we are fully convinced and very confident that we can go on to better and bigger reactors.

Mr. VAUGHAN (UK) : The figure quoted for steam voidage in the reactor implies a very high circulation ratio, something in the order of 50:1 or more. However, you do not consider the separation of steam from this large water quantity to be a problem because the steam velocity is stated to be in the neighbourhood of ½ ft/sec. Do you think this would still be possible in a much larger reactor of, say ten times the output, or would it become a problem then?

Mr. HARRER (USA) : I am not particularly in a position to speculate on what would happen with a reactor ten times as large. I wish I could give you a quantitative answer to that. I would be very happy to do so. In our study reactors, those which we do not have under construction at the present time but which are paper designs, we have not anywhere experienced what we might call concern over this point. I am afraid that is as far as I can go in giving an answer since we do not have anyone with us who could expound on that point.

Mr. BOGAARDT (Netherlands) : I have wondered whether mixing of the injected boric acid solution really occurs fast enough in emergency cases. I presume that the solution has to travel downward through the reflector part before it could enter the active core.

Mr. HARRER (USA) : I am very happy to have this question asked because during the preparation of this paper for presentation it was not possible to show clearly that we were not going to inject the borated water from the top of the reactor. The figures became too complicated for me. However, we actually inject the borated water for shutdown purposes as close to the bottom of the core as we can. There are many openings, of course, in the pressure vessel in the practical plant. We will inject at the bottom and we will obtain quick action from the borated water.

I should like to add these remarks. I have somewhere in my paper a statement that it takes somewhere around one second or so, less than a second, to get the borated water effect. However, there are many other effects, of course, which shut down or tend to limit the maximum power of a boiling reactor. This is the feature which makes it so desirable. We believe that the experience with this reactor will show that it will not be necessary to be in an extreme hurry to get the borated water in there but that other self-limiting power effects will hold the reactor within limits that will not damage the core until this borated water can take its full effect.

Mr. MOORE (UK) : Low fuel element costs depend on long irradiation and high load factor. The appearance of the core of this reactor shows that clearances between fuel elements are rather small and the fuel elements themselves geometrically are rather long. In a reactor of this type in which the primary coolant is used as a working fluid, it seems to me that the emission of fission products from the fuel elements would contaminate the whole plant—turbines, condensers, pumps, etc. Would the author like to comment on the methods used for burst slug detection and any experience he has had with burst fuel elements.

Mr. HARRER (USA) : In the reactors which we have operated up to the present time, we have not had any burst slug problem. The problem which you are presenting of how to detect a burst slug in a bulk reactor where all of your elements are together is really one of major concern to us, too. The solution to such a problem that occurs immediately is to have many points of take-off in the reactor with pipes that lead out, at which points you detect any fission products that may be picked up. This, of course, makes the reactor somewhat complicated in construction so that you lose all of its desirable features. We have no really direct solution that would give us the location of a failure. We do think that we will settle for bulk monitoring. Just exactly how much value such a system would be to us we do not know. Our experience has been so favorable that at the present time we do not contemplate any really detailed monitoring of the core but rather we will watch the ion exchanger circuit, the steam circuit, the water circuit, the condensers of the turbine and the like, to try to determine if we have any rise in radioactivity there which would indicate a slug failure. The only plan that I know of at the present time is to remove elements and examine them. This, of course, is a tedious job. On the other hand, it holds down the cost of the initial installation. Possibly we will never experience the necessity for this type of examination. At any rate, we do not have a real solution to your problem. We really wish that we had one.

Mr. DUNWORTH (UK) : The speaker has suggested that this type of plant would be quite suitable for places where the cost of transporting conventional fuel is high. Unfortunately, it will be necessary in a plant of this sort to transport the irradiated fuel which has been in the nuclear reactor back to some central

point, and presumably these costs are not negligible. Has the speaker any estimates for this cost?

Mr. HARRER (USA): I did not really intend only transportation as being the fuel cost. I am talking about possibly the cost availability also—perhaps just the total cost of coal or oil as compared to the total cost of uranium and your transportation of fuel elements away from the site. There probably could be a simpler method of transportation worked out, because ostensibly the plant located in a remote area, where the cost of transportation would be high, would also offer considerable facilities or space for storage. Perhaps a simple method of handling the spent fuel could be worked out, but this was not quite what I intended when I suggested just high fuel cost. I included all fuel costs.

Mr. SWARTOUT (USA) presented paper P/498.

DISCUSSION ON PAPER P/498

Mr. BAXTER (Australia): Has the author of this paper any information on the stability of Zircaloy under high neutron irradiation for long periods? Is its ability to resist pressure and corrosion likely to be affected by such irradiation?

Mr. SWARTOUT (USA): In answer to your question, we have done a great deal of work studying the corrosion of zirconium stainless steel and titanium by our solutions. Unfortunately, practically all of this work to date has been done in the absence of irradiation. We have experiments in progress now in loops, one of which is demonstrated in the reactor exhibit. In addition, we feel that this will be one of the major objectives of the reactor we are now building—to find out what does happen to the reactor core under long-term exposure to actual conditions.

Mr. KRONBERGER (UK): I take it that the pressure is equalized by the common gas blanket of the reactor and breeder. What, then, is the function of the contrivance in the diagram marked "pressure equalizer"?

Mr. SWARTOUT (USA): We have connected a pressurizer to each of the systems. These are independently heated. In order to be sure that the differential between the two systems does not exceed the strength of the Zircaloy vessel, we have provided lines connecting the two. Actually, they are lines in parallel. We have pressure release valves set at about 50 psi on each, and also a rupture disc as a final safeguard. During normal operation, it will be possible to control the pressure within these differentials. The major difficulty will arise when we discharge rapidly the contents from the two systems because they are of quite different volumes. We will regulate this by a differential control on the valves, but in case these fail we have the last resort of the pressure release valve and a rupture disc.

Mr. LEWIS (Canada): I was surprised to hear the gastight blast shield described as an extra safety measure. I had supposed that this presented the most essential and most difficult engineering problem, needing a solution before any considerable scaling up of any fluid fuel reactor could be contemplated. Has any experience been obtained on a small scale of volatile and gaseous fission products which give a measure of the hazard to be expected from a large power reactor?*

Mr. SWARTOUT (USA): This question is raised, I think, by my statement about the blast shield around the reactor. We agree: a reactor of this type should be, without any question, enclosed in a gastight container. Our shield is gastight and capable of withstanding pressure. My statement regarding this blast shield was that this might be somewhat of an extreme. That we will find out. So far as experience goes, I might mention that in operation of our first homogeneous reactor we had no leak in the high-pressure system itself. We had a small leak in one of the auxiliaries and this happened to be a leak in the gas phase. Such a leak is very easily detected and very easy to find. In fact, once a reactor is operating it is its own best leak detector.

Mr. TARANGER (France): We have learned that part of the biological shield is made of sand and water so that it can be modified to facilitate experimentation. Does not the activity of the sand, especially because of the delayed neutrons, greatly complicate handling for this purpose?

Mr. SWARTOUT (USA): Actually, in answer to your question, there are two such shields. One surrounds the reactor itself. This is the one I mentioned in an igloo-shaped dome. This will certainly become quite radioactive. The one which I referred to which would provide flexibility for permitting later alterations in pipes is in the shield wall between the reactor pit and the control room. The neutron level at that point will be very low.

Mr. SPINKS (Canada): Have you developed any techniques for dealing with a leak which develops during operation without having to shut down the reactor?

Mr. SWARTOUT (USA): This depends upon where the leak is. The piece of equipment we are most seriously concerned about—and this is true of all those working with high-pressure water systems—is the heat exchanger. If we have a leak in the exchanger, what do we do? At the present time, for this reactor, all we can do is unbolt the heat exchanger, put in a new one, and work on the first one while it is submerged under water. It is not a very satisfactory answer for a full-scale power plant. We have an alternate design which has not yet been tested in which it may be possible to remove individual tube bundles. So far as other parts of the equipment are concerned, we

*Mr. Swartout has spoken as follows about the blast shield: "The blast shield is probably something of an extreme in safety measures. The entire reactor is enclosed in a completely gastight shield. In order to keep it gas tight we wanted to be sure that no accidents which might occur could rupture it. Therefore, we have placed a secondary line of defense in a 1½ inch stainless steel shield built around the entire vessel."

have considered the problem of breaking lines while the system is under water, not while the reactor is operating but when shut down, and to prevent dilution and mixing of light water with heavy water. We plan—and we do have located in what we think are critical points—freezers around the lines so that we can freeze water on each side of the broken joint. These have, at least in small sized pipes, worked very well.

Session 14A

POWER REACTORS, PROTOTYPES

LIST OF PAPERS

The Sodium Reactor Experiment

By W. E. Parkins,* USA

I. INTRODUCTION

The Sodium Reactor Experiment (SRE) is a reactor facility being constructed as part of the five-year program for the development of nuclear power. It is specifically intended for the exploration and improvement of technology associated with the sodium-cooled, graphite-moderated type of reactor. A more inclusive program of work is being carried out concurrently and will make use of the SRE as the major tool in developing this technology. While the SRE is primarily intended as an experimental facility, and not a true pilot plant, it incorporates many of the features believed to be desirable in a full-scale central station nuclear plant of this type.

The schedule being followed calls for completion of the SRE construction early in calendar year 1956. Various criticality and preoperational tests will follow and lead to full power operation at some later date. Costs are estimated to be approximately $3,500,000 for the reactor and its various auxiliaries, and approximately $500,000 for the building and facilities. Cost of development work required to prove out many of the components being incorporated in the SRE is separate. The location is approximately thirty airline miles northwest of downtown Los Angeles, in the Simi Hills. The site is situated at an elevation of 1850 feet and is four miles due south of the small town of Santa Susana, California.

The design and operation of the SRE are the responsibility of North American Aviation, Inc., under direct contract with the Atomic Energy Commission. This project, together with the experimental program for the development of sodium-graphite reactor technology, is being financed by the Atomic Energy Commission, with substantial support directly from North American Aviation, Inc. The design and location of the SRE have been approved by the Advisory Committee on Reactor Safeguards, and as of May, 1955, construction work is well under way (Fig. 1).

*North American Aviation, Inc., Nuclear Engineering and Manufacturing, Los Angeles, California. Work performed by W. E. Abbott, R. L. Ashley, R. O. Crosgrove, T. F. Edziak, D. T. Eggen, R. C. Gerber, W. J. Hallett, M. P. Heisler, E. Matlin, M. Mueller, H. E. Richter, W. Sanders, T. T. Shimazaki, S. Siegel, C. Starr, A. M. Stelle, C. A. Trilling, and other members of the staff of the Nuclear Engineering and Manufacturing Dept. at North American Aviation, Inc.

Figure 1. SRE construction in progress, May, 1955

II. GENERAL DESCRIPTION

The SRE is designed for the production of nominally 20,000 kw of heat. No provision is being made in the initial installation for a steam cycle and the production of electrical power. Steps are being taken, however, which will permit such an addition at a later date. The reactor is cooled by sodium, which circulates in a primary system and becomes radioactive. This primary sodium transfers its heat to a secondary, non-radioactive sodium system in intermediate heat exchangers. The secondary sodium rejects its heat to the atmosphere in air-cooled heat exchangers. Both the primary system and secondary system have two separate circulating loops; a main loop, capable of the transfer of 20,000 kw of heat, and an auxiliary loop capable of the transfer of 1000 kw of heat. It is intended that the auxiliary circuit, comprised of the auxiliary primary and auxiliary secondary loops, be operated simultaneously with the main circuit in order to assure heat removal capability in the event of some component failure in either circuit.

Views of the reactor building and the equipment arrangement are shown in Figs. 2 and 3. The reactor is located below grade, with the upper surface of its top shield at floor level in the reactor room. The two primary loops are also below floor level and are installed in separate concrete-walled galleries. Motors for the mechanical sodium pumps and for the control rod drives are located above floor level for easy maintenance. The secondary sodium lines extend from the intermediate heat exchangers to locations above ground level and outside the reactor building to where the air-cooled heat exchangers are located.

A 75-ton handling bridge is designed to move within the reactor room and be capable of supporting lead-shielded coffins used for the removal of radioactive elements from the reactor core. At one end of the reactor room special facilities are installed, again below floor level, for the cleaning and storing of these elements. For the purposes of the experimental program, especially that associated with the fuel development, a hot cell is installed below grade with access holes to receive elements from the handling coffin. Here the fuel elements will be disassembled, inspected, and certain measurements made.

In addition to the sodium coolant and graphite moderator, the SRE will initially use fuel elements fabricated from slightly enriched uranium, containing 2.80 atom per cent U^{235}. These elements, as well as all other elements penetrating the reactor core, are suspended from small plugs in the top shield. The general arrangement of the reactor is shown in Figs. 4, 5, and 6.

The entire core is contained in a stainless steel vessel, 19 feet deep and 11 feet in diameter. The graphite moderator is supported and located on a stainless steel grid plate near the bottom of this core tank. The graphite is in the form of cell-sized hexagonal prisms, placed on a triangular lattice 11 inches between centers. Each prism is 10 feet in height and is clad with thin zirconium sheet. The 10-foot height includes 6 feet for the moderator and an additional 2 feet at the bottom and at the top for reflector. The graphite assemblies making up the core region contain an axial zirconium tube, in which fuel elements are suspended.

Inlet lines from the main primary and auxiliary primary loops enter the core tank above the graphite assemblies and extend vertically downward in double-walled pipes, in order to discharge into a plenum between the bottom of the core tank and the grid plate. This sodium at a temperature of 500°F then passes up

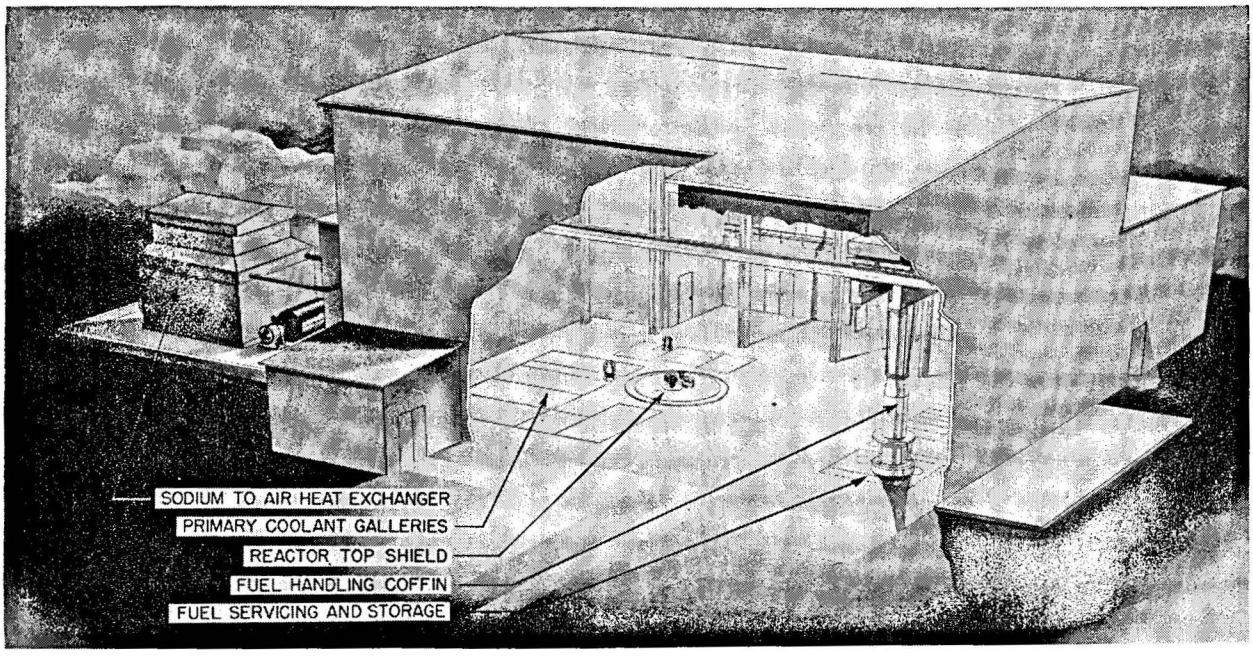


Figure 2. SRE reactor building

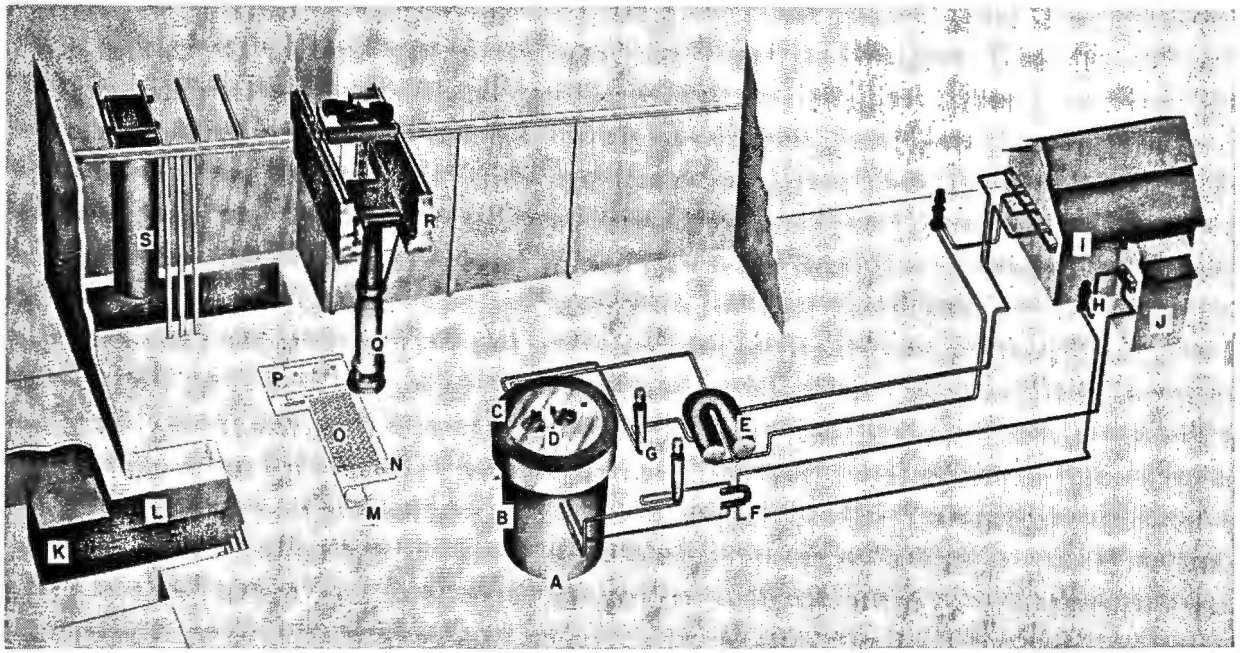


Figure 3. General arrangement of components. A: Reactor; B: Core tank; C: Ring shield; D: Rotatable face shield; E: Main sodium heat exchanger; F: Auxiliary sodium heat exchanger; G: Pump; H: Pump; I: Main sodium-air heat exchanger; J: Auxiliary sodium-air heat exchanger; K: Metallurgical hot cell; L: Primary hot cell; M: Shipping coffin cells; N: Moderator can storage cell; O: Fuel storage cell; P: Fuel cleaning cell; Q: Fuel handling coffin; R: Handling bridge; S: Moderator handling coffin

through the axial tubes, cooling the fuel elements and discharging into a pool 6 feet deep, having a mean temperature of 960°F. Separate outlet pipes for the two primary loops are also located in the core tank above the graphite assemblies.

Surrounding the core tank is a steel thermal shield 5½ inches thick. Immediately outside of the thermal shield is the outer tank intended as an emergency means of containing sodium in the event that a leak should develop in the core tank. The outer tank is surrounded by approximately a foot of thermal insulation. This, in turn, is contained in a tank called the cavity liner. The function of the liner is to serve as a form for the concrete foundation and to aid in the removal of heat from this region. Steel pipe is tack-welded to the outside surface of the cavity liner to provide a means of circulating a fluid for removing the small amount of heat developed in the concrete together with that conducted through the layer of thermal insulation. The fluid circulated is toluene, furnished from a special cooling system intended for many components around the reactor where there might be some conceivable possibility of contact between this fluid and the sodium coolant.

The concrete foundation extends up to the biological shield at floor level, where it is planned to use a special grade of concrete made with magnetite iron ore aggregate. As a closure for the reactor vessel there is a ring-shaped shield supported on a ledge in the cavity liner, and a circular (or rotatable) shield supported on steps on the inside of the ring shield. All of the small plugs permitting access for the core components are located within this rotatable top shield. Large diameter bellows provide a gas seal for the core tank atmosphere and for the atmosphere between the core tank and the outer tank. Separate bellows and diaphragms seal the galleries from the atmosphere between the outer tank and the cavity liner, while permitting thermal expansions of the various pipes penetrating this region.

III. FUEL ELEMENTS

The fuel elements planned for initial operation in the SRE are fabricated in the form of clusters of seven rods, as shown in Fig. 7. Each rod consists of a 6-foot high column of 6-inch uranium slugs in a thin-walled stainless steel jacket tube, thermally bonded by NaK alloy. The slugs are 0.750 inch in diameter, the jacket tube 0.010 inch in wall thickness, and the NaK annulus 0.010 inch in average thickness. The stainless steel jacket material is type 304 and is closed at each end by a welded stainless steel plug. The NaK bonding alloy extends a few inches above the slug column to a free surface above which is helium gas at atmospheric pressure (when at room temperature). This space serves as a volume for the thermal expansion of the NaK and for the accumulation of possible fission gases during irradiation. While sodium might be used as the bonding agent, some difficulty has been experienced in assuring a bond with no voids, since the sodium is a solid subject to fracture at room temperature and since it contracts appreciably upon solidification.

The seven rods forming each cluster are retained at their ends. Support is provided by the fixture at the top, while individual expansion of the rods is permitted by the fixture at the bottom end. Also at the bottom of the assembly is a locating guide and an

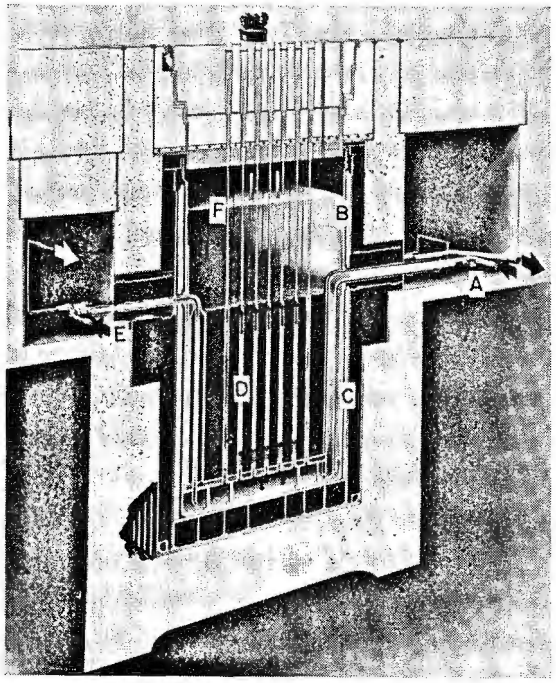


Figure 4. SRE reactor section. A: Main sodium inlet; B: Core tank; C: Thermal shield; D: Fuel element; E: Auxiliary sodium inlet line; F: Sodium level

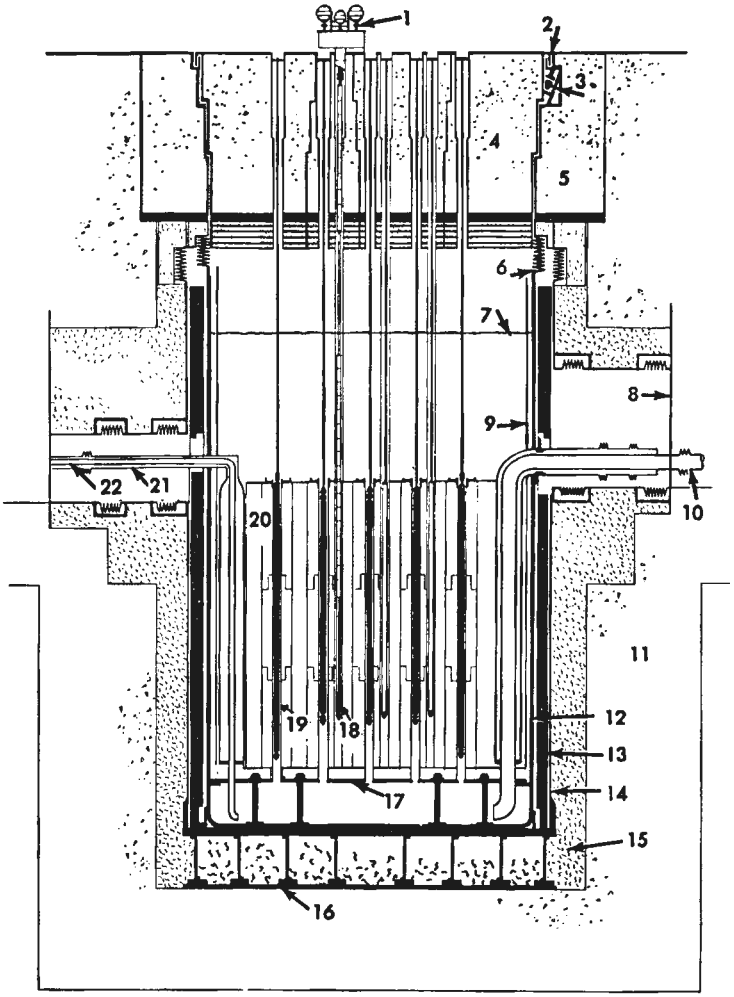
orifice plate for controlling the flow of the sodium, as required for each particular coolant tube. In order to prevent the rods from touching each other or the coolant tube, the six outside rods forming the cluster are spirally wrapped with an 0.092 inch diameter stainless steel wire with a pitch of approximately 10 inches. The direction of wrap is arranged such that adjacent rods are wrapped in counter directions. In order to determine the temperature of the exit sodium from any coolant tube, a thermocouple is inserted down the inside of the stainless steel tube supporting the entire cluster from its plug in the top shield. The junction of this thermocouple is located just above the fixture at the top of the fuel element cluster.

It is planned to operate the SRE so that a maximum uranium metal temperature of approximately 1200°F is attained in each fuel element. This temperature will be determined by the total power level of the reactor, the flow and temperature conditions in the sodium system, and by the selection of orifice plates for the fuel elements. During normal operation there will be a pressure drop of 2.5 psi across the central fuel element and 1.5 psi across its orifice plate, making a total of approximately 4 psi. The flow in this central tube will be 5 feet per second, corresponding to 17,500 pounds per hour of sodium. Flow in a tube requiring the least cooling (near the outside where the average thermal neutron flux is least) will be 3 feet per second. Nominal sodium outlet temperatures, with the 500°F inlet, are 908°F for the central tube and a maximum of 986°F for a tube in the lowest average neutron flux, giving a mixed mean in the sodium pool of 960°F. The maximum heat flux anticipated is about 340,000 BTU/hr-ft^2, which will be accompanied by a heat transfer coefficient of approximately 10,000 BTU/hr-ft^2-°F. Approximate values of the peak-to-average heat flux are 1.35 for any channel and 1.63 for the entire reactor. Under these conditions, the central fuel element will produce 650 kw, corresponding to 340 kw per kg of U^{235}.

To achieve the conditions just described, it is necessary that the flow through the coolant tube and the interstices of the fuel element be such that there is a considerable degree of mixing of the sodium as it moves through the inner passages just surrounding the central rod and through the outer passages surrounding the ring of six rods. Otherwise, the temperature would increase more rapidly in the inner passages, since the total heat flux to sodium cross section is greater there than in the outer passages. The effectiveness of this mixing has been measured by experiments circulating water past a fuel element mock-up and determining the concentration at various locations of a solution of manganous chloride, which was injected into the inner passages near the upstream end of the element. These experiments showed the need for wrapping the wire in counter directions on adjacent rods so that the fluid is most effectively forced outward from an inner passage and inward from an outer passage in alternate openings around the ring of six rods. The pitch of the rotation of the fluid as determined by these chemical experiments is slightly greater than the pitch of the wire wrap, but is sufficiently short to guarantee adequate sodium mixing in the coolant tube during SRE operation. These hydraulic experiments with the fuel element mock-up were also useful in establishing the pressure drops and vibrational conditions. No difficulty is expected from rod vibration even though there is no provision for securing any cluster of rods along its length.

It is estimated that sufficient reactivity for operation will be obtained with a loading of thirty-one fuel clusters of the type described, and employing the 2.8 atom per cent U^{235} enriched uranium. This will provide a core region 6 feet high and approximately 6 feet in diameter, wherein the neutron flux has been calculated to be that shown in Fig. 8. The average thermal neutron flux at the center of the core is estimated to be 2.5×10^{13} neutrons/cm^2-sec. The peak thermal flux in the fuel is estimated to be 1.7×10^{13} neutrons/cm^2-sec. Some of the calculated nuclear parameters for the SRE at design operating conditions are as follows: $\epsilon = 1.034$; $p = 0.799$; $f_{25} = 0.745$; $k_\infty = 1.28$; $L^2 = 137$ cm^2; $\tau = 370$ cm^2; and $B^2 = 4.84 \times 10^{-4}$ cm^{-2}.

With the design conditions of sodium cooling and neutron flux which have been described, the reactor power level with thirty-one fuel elements will be slightly less than the nominal value of 20,000 kw. To achieve this power with design conditions, a loading more nearly equal to thirty-seven fuel elements is necessary. The actual number of fuel elements used will depend upon the results of criticality measure-



1. Control element drive
2. Seal
3. Roller
4. Rotatable shield
5. Ring shield
6. Bellows
7. Sodium level
8. Diaphragm
9. Inner liner
10. Main sodium inlet line
11. Biological shield
12. Core tank
13. Thermal shield
14. Outer tank
15. Thermal insulation
16. Beoring plate
17. Grid plate
18. Control element
19. Fuel element
20. Moderotor element
21. Guard pipe
22. Auxiliary sodium inlet line

Figure 5. Reactor elevation

1. Ring shield
2. Rotatable shield
3. Main sodium inlet pipe
4. Control element
5. Fuel element
6. Side biological shield
7. Core tank
8. Thermal shield
9. Outer tank
10. Thermal insulation
11. Inner liner
12. Bellows
13. Auxiliary sodium inlet line

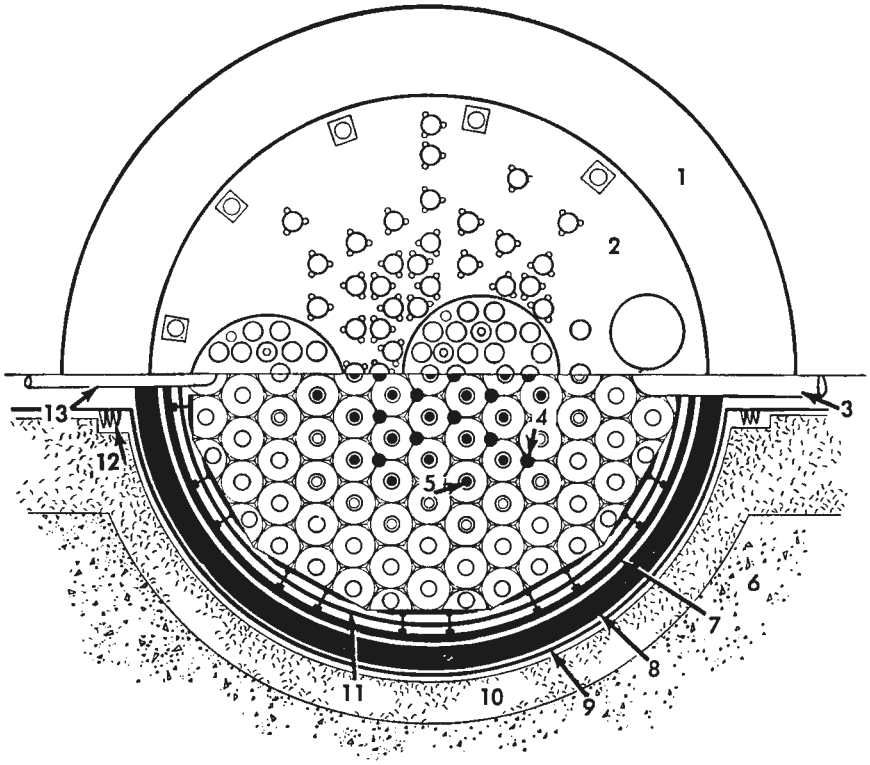


Figure 6. Reactor plan

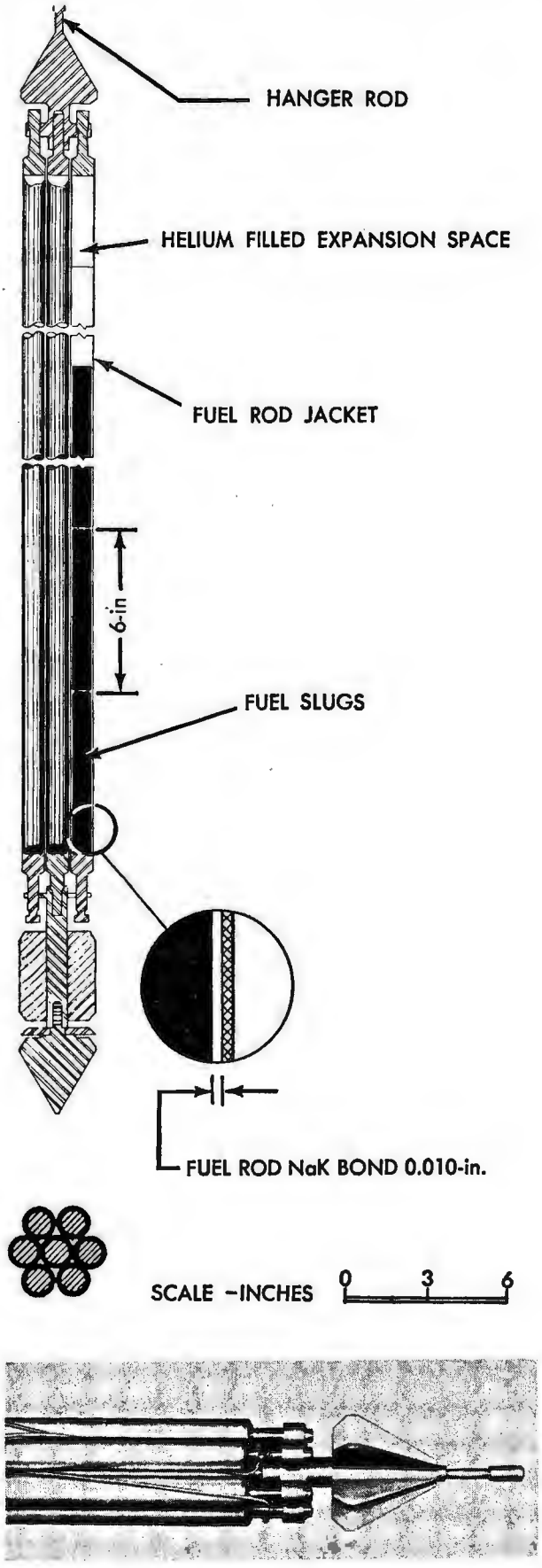


Figure 7. Seven-rod element

ments to be made on the assembled core and upon the requirements of the experimental program.

IV. MODERATOR AND REFLECTOR

The principal problem on the graphite moderator and reflector is that of preventing contact with the sodium coolant. Such contact results in the absorption of sodium into the pores of the graphite, causing excessive neutron losses. Since the amount of metal required for the cladding of the graphite in the SRE design configuration is large, it is impractical to employ stainless steel for this purpose even though stainless steel has been chosen as the fuel jacket material. The graphite is protected by zirconium sheet fabricated into individual can assemblies, shown in Fig. 9. As designed, 0.035-inch thick sheet will be used on the side panels of each hexagonal column of graphite, and 0.10-inch thick stock will be used for the bottom and top can heads. The distance across the flats of each can assembly is slightly less than the 11-inch center-to-center spacing of the triangular lattice. This reduction from 11 inches is sufficient to provide for an average gap between cans of approximately 0.170 inch during normal operation. Such a gap in the form of a thin, flat channel is necessary to permit some heat removal at the can wall by the sodium coolant. Sodium for this purpose enters the core tank in a separate pipe, branching from the main inlet sodium stream and accounting for a total flow of approximately 7 per cent of that in the main stream. This branch line discharges its sodium into a low pressure plenum above the grid, but below the graphite assemblies. This sodium is then free to seep upward through the passages between cans and through special channels provided for core elements other than the fuel elements. It removes heat from these core elements, from the cans, and from the tanks and shielding at the sides of the reactor. Average flow is to be adjusted so that this sodium exits at approximately 960°F.

Each graphite assembly is bolted by means of zirconium studs to a supporting pedestal at the base of the can and to a spacer plate at the top. Both the pedestal and spacer plate are fabricated from type 405 stainless steel. The pedestal at the bottom serves not only as a support for the can, but locates it laterally by fitting into a hole provided in the grid plate. The seal at the grid plate is sufficiently tight to prevent sodium leakage into the plenum above the grid. In addition, the pedestal has a circular channel along its axis which directs the sodium from the main plenum up through the coolant tube in the moderator can assembly. The spacer plate at the top serves as a lifting fixture for the entire assembly and as a means of lateral support. The spacer plates from adjacent cans nest together and are maintained in place by a clamping band around the outside. If it should be necessary at some later date to replace a graphite assembly, this arrangement makes it possible to remove the particular unit by lifting vertically without the complication of unfastening a mechanical connection.

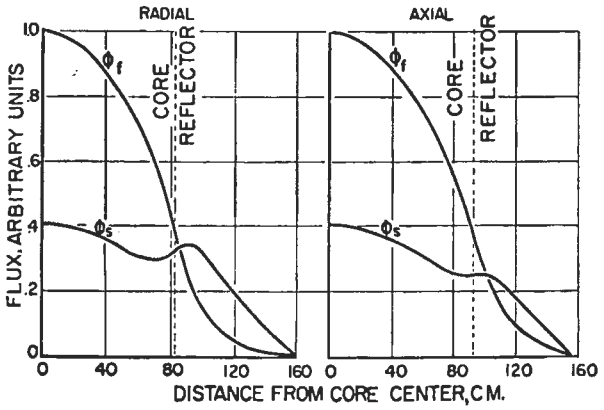


Figure 8. Calculated average neutron flux in SRE core

Another feature, which prevents the moderator assemblies from ever being appreciably displaced in the vertical direction during operation is a tube which extends down from the top shield around each fuel element hanger rod. The lower end of this tube would bear against the spacer plate if the moderator assembly were dislodged.

The zirconium canned graphite assemblies which make up the side reflector do not have axial channels. Each assembly in the core region is penetrated along its axis by a zirconium tube 0.035 inch thick and 2.80 inches in inner diameter, welded to both the bottom and top heads. Such a tube would usually contain a fuel element. At the time of reactor scram, the sodium temperature in the coolant tube would suddenly decrease, causing a contraction relative to the outside walls of the can and producing a deflection in both the bottom and top heads. The stresses resulting here and those from other static and transient operating conditions have been calculated to be safe from the known mechanical properties of zirconium.

One of the problems in canning the graphite is the question of gas pressure which might be produced as a result of long-time operation at elevated temperature in the radiation field. With the can design just described, the side panels are of thin material and sufficiently flexible so that it is only safe to operate with some external pressure collapsing these panels against the graphite. At the time of assembly, the top head is spaced very close to the top of the graphite column, but during operation as the temperature increases, the zirconium expands away from the graphite, necessarily leaving the head unsupported. This makes possible the flexibility required for the scram condition just mentioned, but at the temperatures involved makes it imperative that only a small pressure difference exists between the sodium on the outside of the can and the gas atmosphere on the inside. Since the outgassing properties of the graphite during operation are not fully known and since it is extremely difficult to predict the average temperature of the graphite which will determine the pressure buildup inside any sealed can assembly, provision is being made to control the pressure on the inside by means of a vent tube extending from the bottom of the assembly out through the top head and to the helium atmosphere above the free surface of sodium in the core tank pool. This vent tube is ¼ inch in outside diameter and is protected by a vertical stainless steel guard tube attached to the spacer plate. At the bottom of the vent tube and resting on the bottom can head is a small cup to accept any sodium which enters during the lifetime of the unit, as a result of diffusion of sodium vapor and pressure fluctuations. The diffusion of sodium by this means has been calculated and measured experimentally to be acceptably small. The vent tube arrangement, however, is a mechanical complication which it is hoped may be eliminated in future designs.

The graphite is machined from large logs approximately 13 inches in diameter, but each cell is vertically assembled in three pieces with plug connections. There are no special strength requirements on the graphite except insofar as it must support its own weight and resist lateral earthquake accelerations while supported at the top and bottom ends. Thermal stresses and radiation damage effects have been calculated from available information to be well within safe limits. To provide extra channels for the insertion of control elements, safety elements, and other special units, a number of can assemblies are shaped at the corners to provide circular passages extending down to the grid plate. Each circular channel is formed by 120 degree shaped corners in three adjacent assemblies, making a passage approximately 3¼ inches in diameter.

The amount of heat which flows to the outer walls of the can assemblies varies with position within the reactor. No special provision is necessary to orifice the sodium flow in the channels between cans according to the heat generation rate. If the temperature of the sodium in one channel is increasing in temperature more rapidly than the average in the other channels, the resultant decrease in density automatically increases the flow rate in that particular channel. As a result, this buoyancy effect tends to assure nearly uniform temperature of the sodium throughout any horizontal plane taken through the reactor. This feature is also important in the region outside of the side reflector where considerable heat is received from the core tank and the thermal shield. Calculations show it is necessary for the flat channels between cans to be wider than some amount of the order of $^1/_{16}$ inch in order to assure sufficiently uniformized temperatures. While these gaps will be maintained by the action of the pedestals and spacer plates, an additional precaution has been taken wherein dimples 0.045 inch in height are rolled into the zirconium panels before can fabrication. During operation, these dimples will prevent the closing off of any channel, as might be caused by misalignments.

V. CONTROL ELEMENTS

There are four control elements in the SRE core, located as shown in Fig. 10. Calculations indicate that this arrangement will permit a total of about 10 per

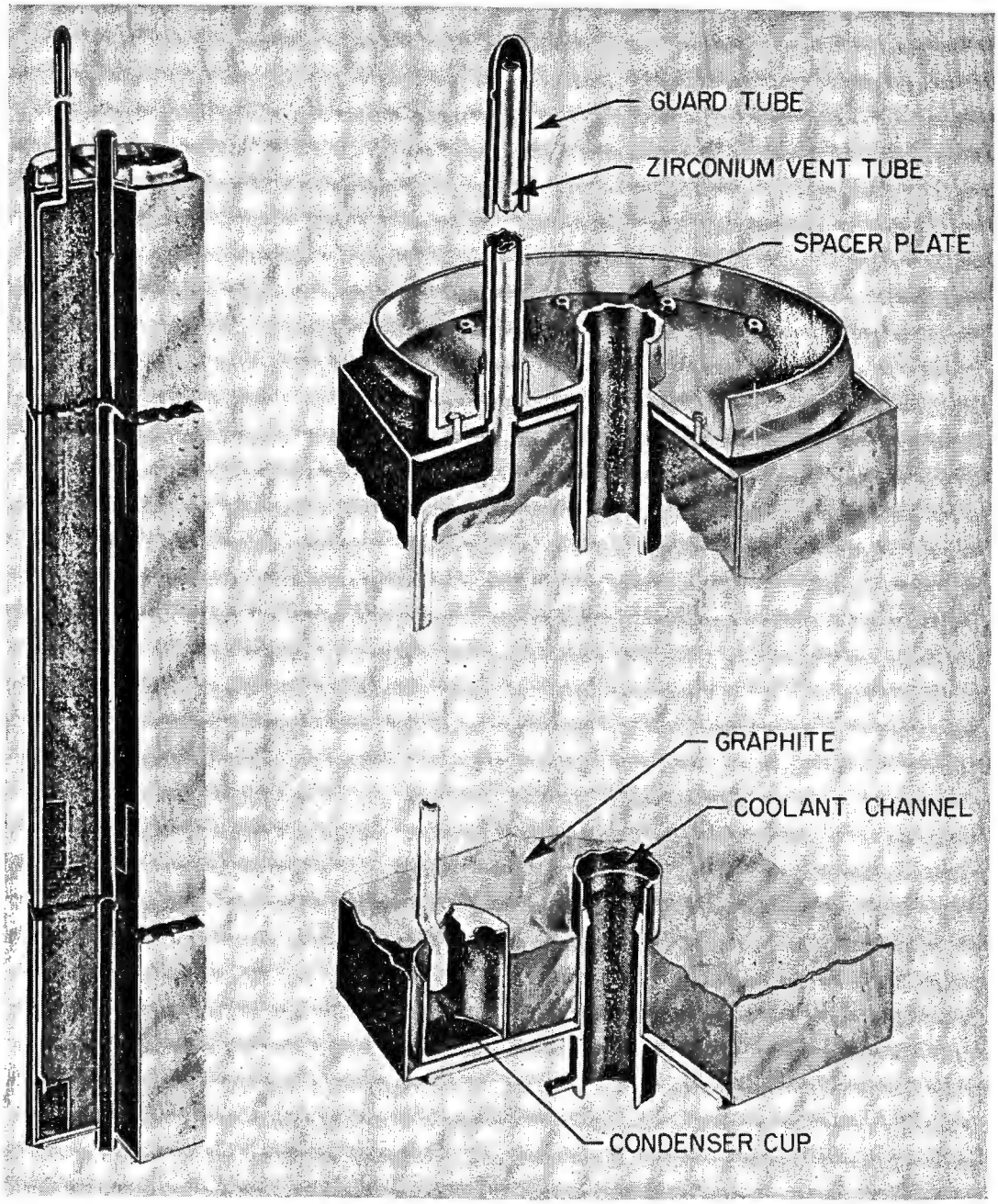


Figure 9. Zirconium-canned graphite moderator assembly

cent control in reactivity. Each control element is contained in a thimble assembly which extends from the top of the rotatable shield to a point just below the core, a total distance of 23½ feet. The thimble material is type 304 stainless steel and in the core region it has a wall thickness of 0.049 inch. This arrangement makes possible a self-contained control rod assembly in which no sodium or its vapor contact the moving mechanical parts. See Fig. 11.

The poison column is made up of a series of eighteen rings of a boron-nickel alloy suspended on a "pull-tube." Each ring is 2½ inches in outside diameter, ⅜ inch in annular thickness, and 4 inches long. The boron concentration in the alloy is approximately 2 weight per cent. Control rod motion is obtained by a ball nut screw arrangement wherein the pull-tube is attached to the nut and a motorized drive mechanism above the top shield turns the screw. The nut is prevented from rotating by guides which move in flutes machined into the inside surface of the heavy wall of the upper portion of the thimble. Suitable shielding is provided within the thimble in the neighborhood of the top shield. Two of the control rods have single motor drives to produce 0.24 foot per minute motion of the control rods for shim action only. The other two rods have dual speed drives for this same shim motion and for a regulating motion at a rate of 3 feet per minute. The latter is mechanically limited to a

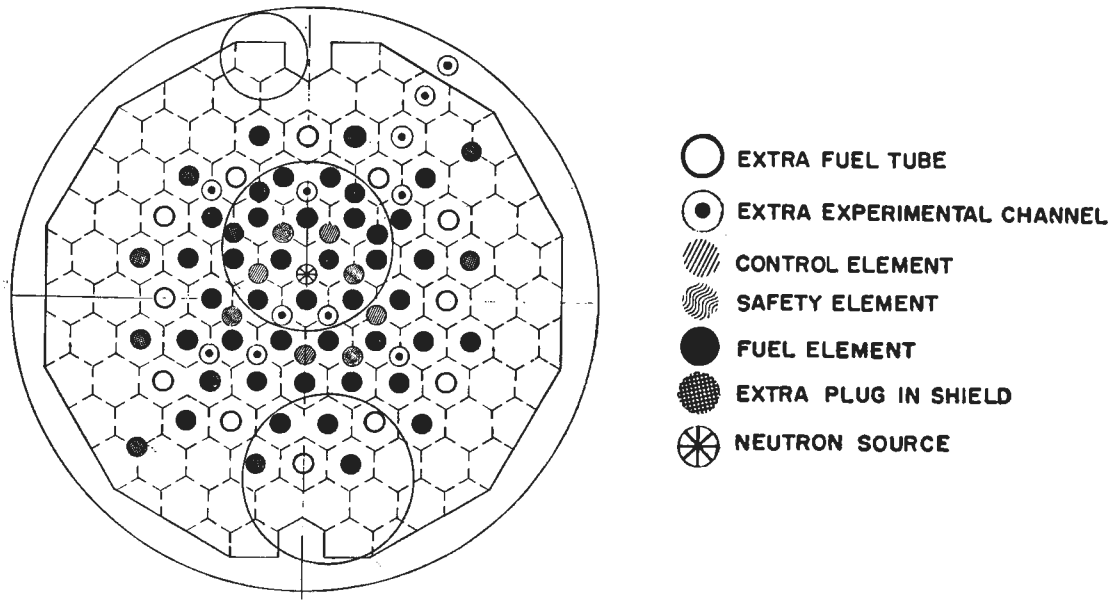


Figure 10. Loading face pattern

range of ±6 inches of travel. The motor drive units rest on the top shield and engage at a clutch several inches below the top surface of the shield. Limit switches and Selsyn motors indicate the control rod positions at all times.

One of the problems with the thimble arrangement is that of transferring the heat generated in the poison material to the sodium coolant on the outside of the thimble. To prevent excessive temperatures in the boron-nickel, helium gas at about 16 psig is used as an atmosphere within the thimble and relatively close clearances are maintained between the rings and the thimble. Experiments have been conducted to establish these conditions. With the maximum heat generation rate anticipated at 8 kw per foot, a maximum temperature of about 1300°F is expected in the boron-nickel. This is for an eccentric condition wherein the ring touches the thimble and the helium annulus has an average thickness of about 0.020 inch. In spite of variations in control rod position and in heat generation rates, the sodium flow on the outside of the thimble will, to a very large extent, automatically adjust itself as a result of density changes with temperature, so that the sodium temperature will increase at the same rate as in the other channels and exit into the pool at about 960°F.

VI. SAFETY ELEMENTS

The SRE core also uses four safety elements, located as shown in Fig. 10. The total reactivity control available in these elements is slightly less than 10 per cent. The design is similar to that of the control elements in that the entire assembly is contained in a thimble extending from the top of the rotatable shield to a location below the core. The thimble material is type 304 stainless steel and in the core region has a thickness of 0.035 inch. The safety element is illustrated in Fig. 12.

Rings of the same boron-nickel alloy as used in the control elements make up the poison column. Each ring is $2^{5}/_{16}$ inches in outside diameter, $^{3}/_{16}$ inch in annular thickness and 4 inches long. A total of 14 rings is used, assembled in series, on an internal tube. This tube may be raised by the action of a ball nut screw, but a latch mechanism is incorporated to release the rod at any time while it is being withdrawn or when it is in the fully cocked position. The latch is actuated by a "torque tube" flattened on two sides and extending down the center of the assembly from an actuating solenoid near the top to a position near the bottom of the thimble. The ball screw is driven by a motor located above the top shield as in the case of the control element. To reset the safety rod, the ball nut is driven downward by the action of the screw until engaged to the rod unit by the latch. The direction of the motor is then reversed for withdrawal. Shielding material is incorporated within the thimble in the region of the top shield. There are also electrical contacts arranged to determine the position of the rod as inserted, fully cocked or being withdrawn.

When the safety rods are released to scram the reactor, each latch is disengaged and the rod units fall freely within the thimble under the influence of gravity. In order to decelerate without the production of excessive stresses, the upper end of the tube on which the poison rings are assembled is attached to a piston within a 24-inch long cylinder. As the rod unit falls, the piston and cylinder move with the rod until it reaches a position 24 inches from the bottom of the thimble at which time the cylinder is arrested by a shoulder and the rod unit is then decelerated as the piston is forced to move through a helium atmosphere contained in the cylinder.

In the case of the safety elements, there is not the same problem of heat transfer that exists with the control elements since the introduction of the safety

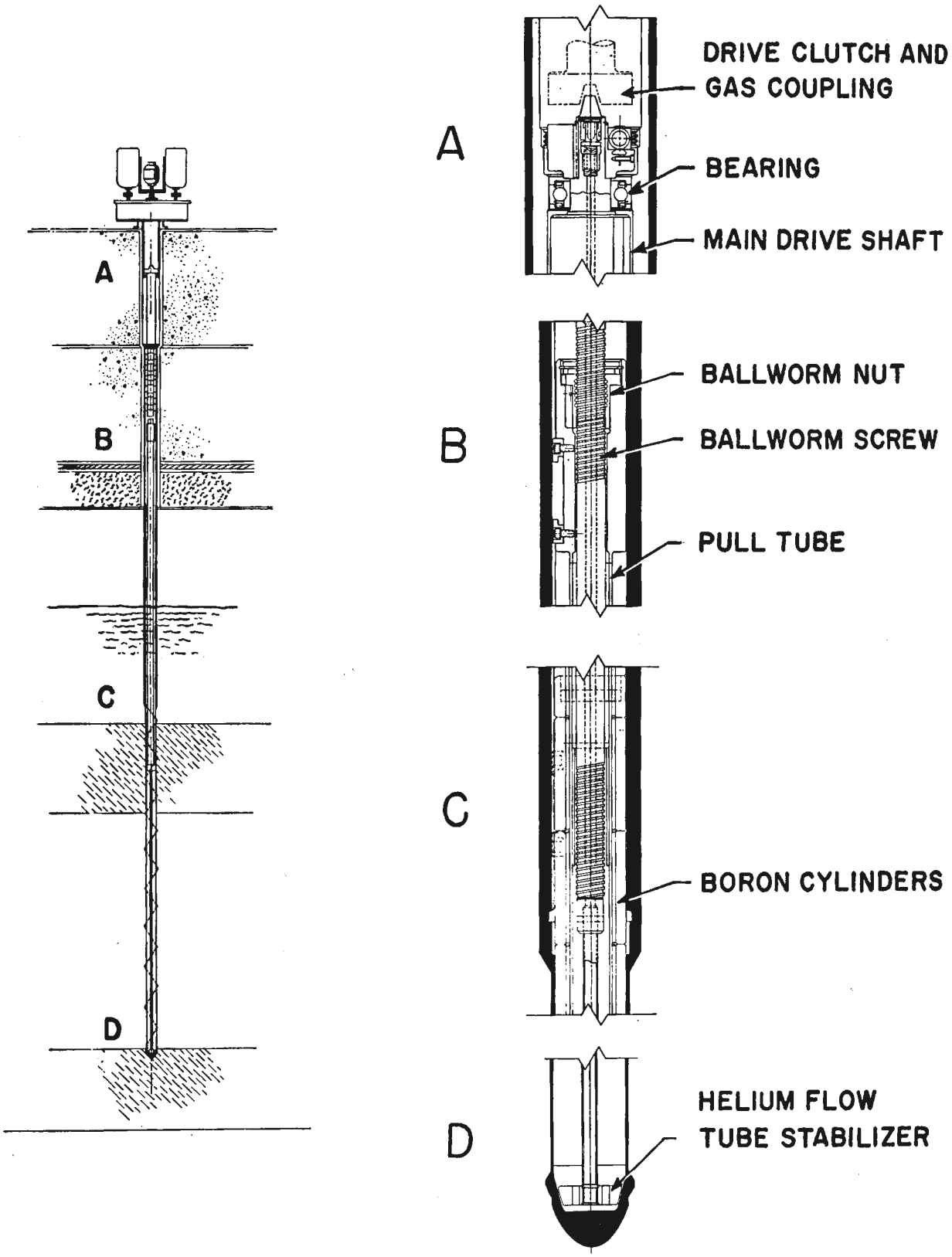


Figure 11. Control element

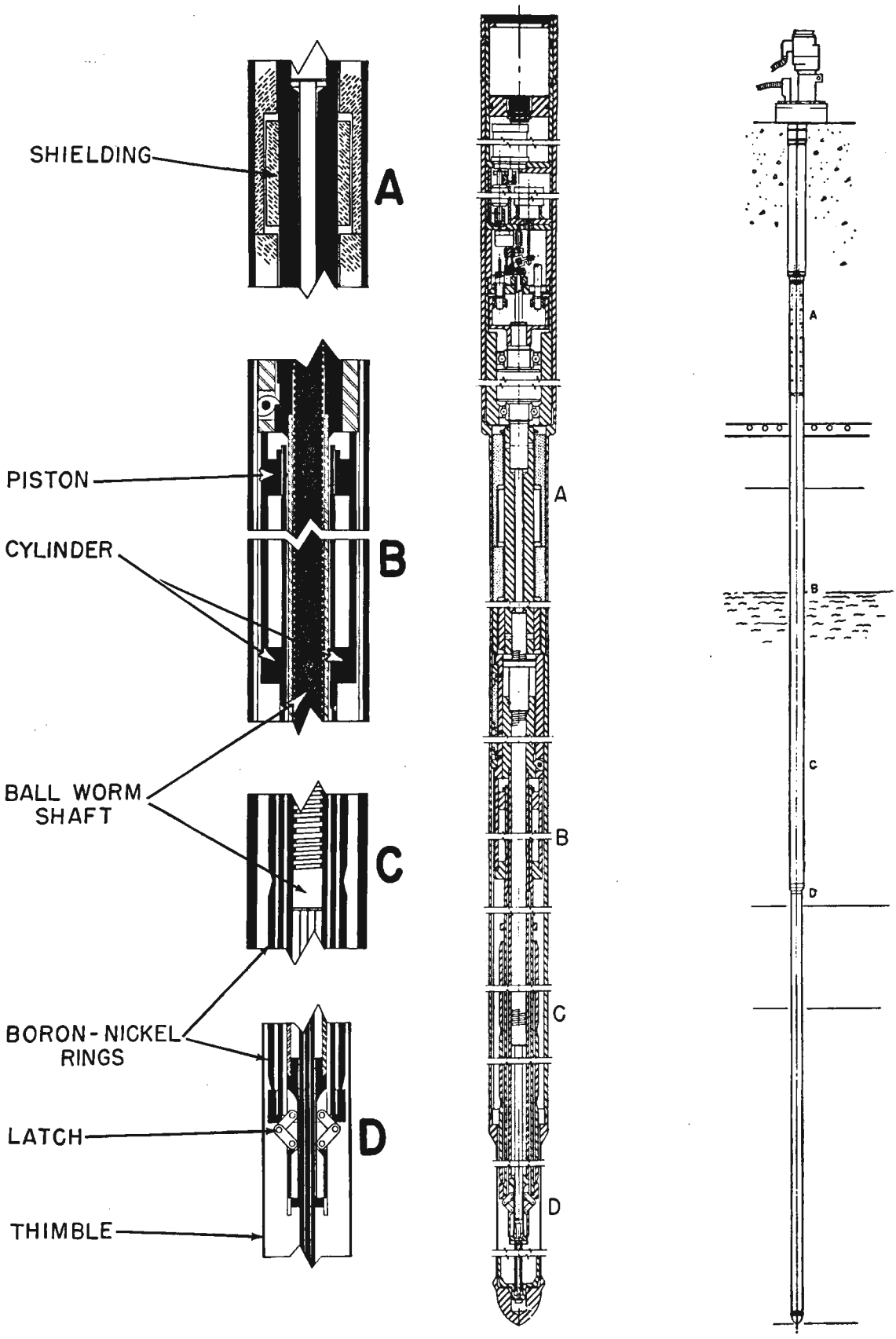


Figure 12. Safety element

rods into the core always results in shutting down the reactor. A much larger annulus, an average of 0.125 inch, exists between the boron-nickel rings and the stainless steel thimble. This has the added advantage of decreasing the possibility of binding due to some misalignment at the time the rods are dropped. An atmosphere of helium at about 16 psig is used within the safety element thimble.

VII. OTHER CORE ELEMENTS

The SRE core is designed so that there are a total of 43 axial tubes in the moderator assemblies capable of receiving fuel elements. There are also 17 channels at the corners of the moderator assemblies capable of receiving control or safety elements or other elements as desired. It is planned to load 31 fuel elements for the necessary criticality, although additional units may be loaded into the remaining 12 fuel tubes if required. This permits a certain degree of flexibility but necessitates the use of "dummy elements" for those tubes not containing fuel elements. The dummy elements serve to replace coolant with moderating material and to prevent the flow of unheated sodium up into the pool above the core. In the same way, those corner channels which do not contain control or safety elements, nine channels as planned, are to be filled with dummy elements or special purpose elements of some kind.

The dummy elements for the fuel tubes consist of graphite cylinders clad in thin zirconium sheet and suspended on hanger rods from plugs in the top shield much the same as the fuel elements themselves. The zirconium can for the dummy elements is 2½ inches in outside diameter, and extends nearly the full height of the core plus top and bottom reflectors. A vent tube is attached which extends up along the hanger rod to a location in the helium atmosphere above the sodium surface. One special feature required for these elements is a weighted plug which hangs from a flexible joint at the bottom of the zirconium can and rests on the upper lip of the pedestal used for support of the moderator assembly. Except for that sodium passing through a small orifice hole which is drilled up the length of the plug, the flow from the main inlet plenum is effectively stopped. The plug assembly is fabricated from type 304 stainless steel and is of sufficient length to seal against a pressure of 8 psi in the lower plenum without the requirement of any other mechanical connection.

The dummy elements for the corner channels are constructed similarly but require no plug at the bottom since the corner channels only open to the low pressure plenum above the grid, which receives sodium for cooling the moderator. Sodium flow in these channels will adjust according to the heating rate so the exit temperatures into the pool are approximately the same as from all other channels which open to this plenum.

One special type element to be used in a corner channel is the neutron source. As designed, this unit consists of a column of beryllium cylinders suspended on a ⅝ inch outside diameter tube extending down from a plug in the top shield. The beryllium cylinders will have the same diameter as the dummy elements and will extend vertically the full 6 feet of the core. A small cylinder of antimony ¼ inch in diameter, 1 inch high and enclosed in a protective graphite capsule, is suspended in a steel tube inserted down the inside of the tube which supports the beryllium cylinders. The graphite capsule with the antimony is sealed in the end of the inner tube and is located at the center of the core. This antimony will be activated before installation in the SRE by exposure in another reactor. With an expected activity of 18 curies, the antimony will produce approximately 5×10^7 neutrons per second by means of the (γ, n) reaction on beryllium. This arrangement permits replacement of the antimony if required, although it should be continually activated during operation of the SRE. While less beryllium could be used without appreciably decreasing the neutron production rate, the design just described has considerable advantage in simplicity.

VIII. REACTOR INSTRUMENTATION

In order to determine the temperatures in the sodium stream cooling the moderator, two of the corner channels will receive special elements containing thermocouples with their junctions at various elevations within the core. These elements are designed very similarly to the neutron source element just described. Each temperature measuring element consists of a column of beryllium cylinders suspended on a stainless steel tube extending down from a plug in the top shield. All of the thermocouples are contained on the inside of this supporting tube and do not contact the liquid sodium.

Aside from thermocouples in the special elements just mentioned, and those within the hanger rods of the fuel elements, no means is provided for measuring temperatures within the core tank. A number of thermocouples are installed to measure temperatures at various locations on the thermal shield, the outer tank, the foundation concrete, and the top shield.

There are some eighteen small plugs in the top shield which are not over corresponding fuel tubes or corner channels within the core. Two of these will be used for sodium level indicators which dip into the pool above the core. The principle of these indicators is a tubular type heater unit to which stainless steel sheathed thermocouples are attached at various heights. The sodium level can be approximately indicated by the recorded temperatures when this unit is being electrically heated.

To determine the neutron flux level, there are eight detecting chambers which can be lowered to positions at the cavity liner in the horizontal mid-plane of the reactor. Figure 13 shows the position of the tubes which permits the insertion of the chambers from wells at floor level in the reactor room. There are four tubes, capped at their lower ends, on each side of the

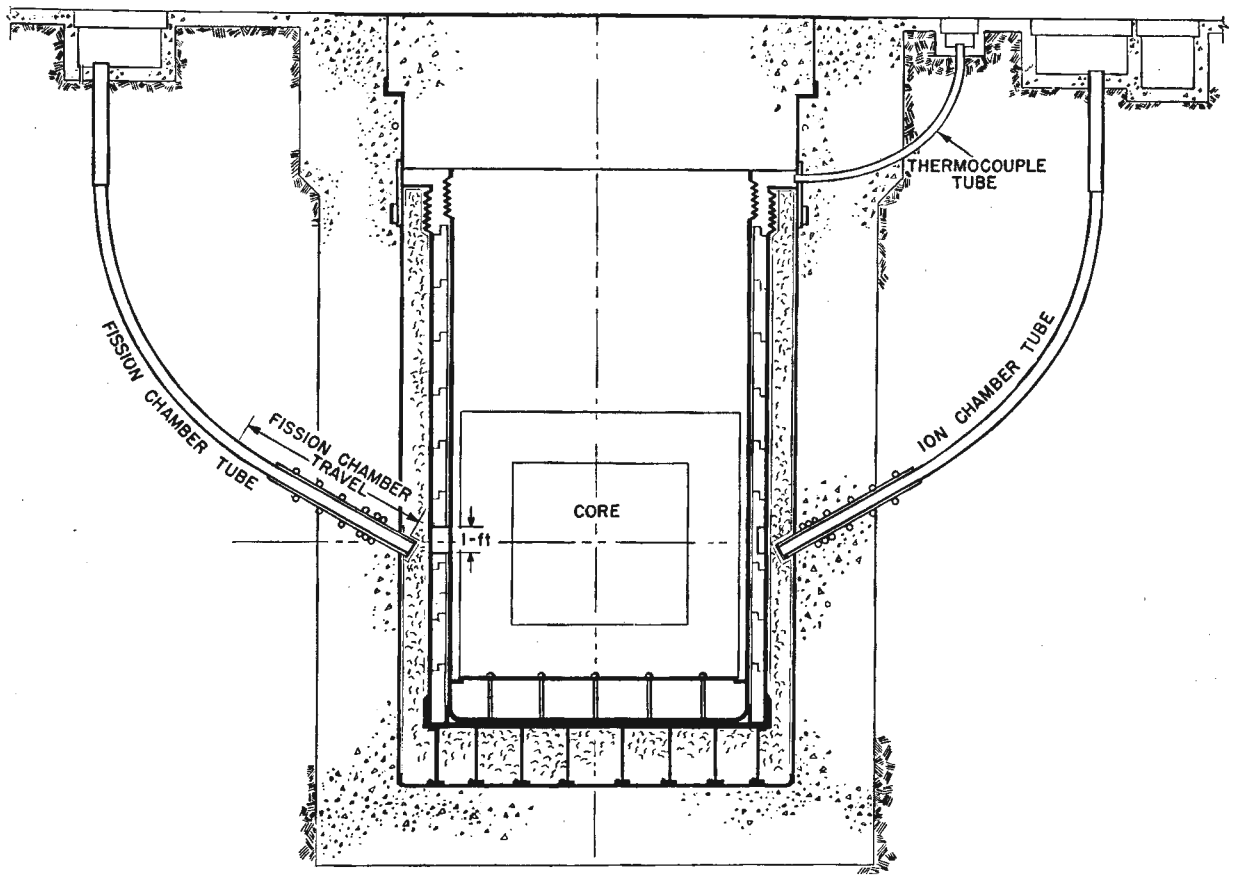


Figure 13. Location of fission and ionization chambers

reactor. They permit access for a total of two fission chambers and six ionization chambers. Before reactor start-up, the neutron source at the center of the core provides sufficient neutrons to obtain an indication from the fission chambers. The operator will use these chambers in coming up to a power of about 3 kw. Three of the ionization chambers, which are gamma compensated, are used for establishing flux level and period in the range from 3 kw to full power. The other three ionization chambers are uncompensated and are used for neutron level in the range from 30 kw to full power. One of these uncompensated ionization chambers provides a signal which automatically regulates a control rod to hold neutron flux level constant. Other chambers are connected to circuits which automatically release the safety rods in the event of excessively high neutron level or excessively short neutron period. In the power range above that intended for use of the fission chambers, these units will be raised from their locations near the cavity liner to positions where the neutron flux is sufficiently low to avoid damage. The tubes which receive the fission and ionization chambers are cooled at their lower extremities by attached piping through which the toluene coolant is circulated. This arrangement avoids the problem of operating the detecting chambers at the temperatures of the primary sodium.

IX. CORE TANK

The large tank which contains the SRE core is 1½ inch in wall thickness and is fabricated from type 304 stainless steel. The tank has no openings below the top of the graphite assemblies. Its discharge line for the auxiliary loop is located just above the graphite level and the auxiliary inlet is about 1 foot above the graphite. A rupture in this loop in the primary auxiliary gallery could not lower the sodium level in the reactor tank below the main discharge line. Openings for the main inlet and discharge, the moderator coolant inlet, the tank drain line, and the gas vent pipe for the core tank are all located approximately 1 foot above the top of the graphite. Ruptures in any of these lines would leave the graphite assemblies covered with sodium and the auxiliary primary loop operative. The inlet pipes for the main and auxiliary loops and for the moderator coolant are of a double wall construction to provide thermal insulation between the inlet pipes and the sodium pool. Helium gas is used to fill the annular space between the pipes.

At the top of the core tank is the large bellows to permit vertical expansion. This bellows is approximately 18 inches high and is fabricated from type 321 stainless steel, 0.10 inch in thickness. A welded flange near the bottom of the core tank supports the 1¼ inch thick type 304 stainless steel grid plate. Additional

support for the grid plate is provided by means of staybolts arranged in three circular patterns. The inner circle of bolts is welded to the bottom of the core tank to guarantee sufficient strength against deflection of the grid plate in the event of a pressure surge in the main plenum.

To minimize thermal stresses due to any possible mismatch in temperature of the sodium flowing upward to the isothermal pool, an inner liner has been added. This liner is fabricated from type 304 stainless steel, is ¼ inch in wall thickness, and assures a stagnant layer of sodium 2½ inches thick adjacent to the inner surface of the core tank. A flange welded to both the core tank and liner at a location just above the graphite assemblies provides support. This flange has small weep holes for sodium drainage. The liner extends from 24 inches above the grid plate to the top of the core tank.

To center and support the graphite assemblies in the horizontal plane, a ¼-inch thick band 6 inches high is located at the inner surface of the liner at the elevation of the spacer plates. This band is of type 405 stainless steel, the same material as the spacer plates. It is supported and centered within the liner by means of dowels welded to its outside surface to project radially into holes drilled in the liner. The spacer plates for the graphite assemblies are clamped into position by means of twelve straight bars of type 405 stainless steel mounted on the band. Mechanical means are provided to adjust these bars radially inward to bear against the entire cluster of spacer plates in order to center them and to provide proper clearance upon initial assembly. The type 405 stainless steel was chosen for the components in this region because of its lower coefficient of thermal expansion. This feature helps to minimize the sodium gap which appears between the zirconium cans as the temperature of the reactor is increased. The arrangement just described holds the moderator and reflector rigid and properly located beneath the top shield.

X. OUTER TANK AND THERMAL SHIELD

The outer tank is a flat bottom vessel fabricated from low alloy steel. It is 12½ feet in diameter and 19 feet high. The side wall is ¼ inch thick and the bottom is 2½ inches thick. The bottom serves to support the core tank as well as the thermal shield, which is located in the annular space between the core tank and the outer tank. At the top of the outer tank, there is a transition to a type 321 bellows similar to that used on the core tank. The bellows provides a gas seal for the helium atmosphere used in the annular space between the tanks. Pressure of this helium is nominally 3 psig, which is the same as the pressure in the helium atmosphere above the sodium surface in the core tank. The available space in the annular region between tanks is such that the sodium level cannot drop more than 3 feet in the event of a leak in the core tank. This safety feature, which assures sodium cooling within the reactor core, is the primary function of the outer tank.

The flat bottom of the outer tank is supported on a group of four concentric cylinders 21 inches high made from low alloy steel. These cylinders rest on circular bearing plates attached to the bottom of the cavity liner by means of anchor bolts extending into the concrete foundation below. Since these supporting cylinders are not welded at the bearing plates or where they contact the outer tank, there are essentially no thermal stresses as a result of the approximately 400°F temperature gradient established along these cylinders during normal operation. The three inner cylinders are ¼ inch in thickness whereas the outermost cylinder is ½ inch. This outermost cylinder also serves to maintain the outer tank centered relative to the concrete foundation. This is achieved by means of narrow flanges welded to the outside surface of the cylinder at its bottom and at its top. These flanges are machined to have radial notches which fit over cleats welded to the outermost bearing plate and the bottom of the outer tank. This arrangement constrains the outer tank to radial motion only and assures that the center line of the outer tank will remain fixed, in spite of any thermal cycling occurring during operation. The same principle is utilized in maintaining the core tank centered relative to the outer tank. To accomplish this, there is a skirt which is welded near the bottom of the core tank and which contains milled slots fitting over cleats welded to the top surface of the outer tank bottom.

The thermal shield is fabricated from low carbon steel in the form of a series of seven rings with interlocking joints. The over-all height of this thermal shield assembly is 19 feet and each ring is 5½ inches in thickness. The rings are not welded to each other and the bottom ring simply rests on the bottom of the outer tank while being centered by the cleats located there. Practically all of the heat developed in the thermal shield and tank structure is conducted back into the sodium seeping up along the inner surface of the core tank liner. The maximum heat flux produced in this way is estimated to be 3000 BTU/ft²-hr. Of this amount, only about 100 BTU/ft²-hr are estimated to be conducted through the thermal insulation to the cavity liner.

XI. INSULATION, CAVITY LINER, AND FOUNDATION

The region between the outer tank and the cavity liner, approximately 12 inches in annular thickness at the side, and 23 inches at the bottom, contains Superex block insulation. This is a commercial product consisting principally of calcined diatomacious silica and asbestos. In order to decrease the dusting during and subsequent to installation, the Superex blocks are coated with a solution of sodium silicate. In the annular region at the side, the insulation is mounted by wire supports fastened to studs welded to the inside of the cavity liner. An atmosphere of

about 3 psig of nitrogen will be maintained in this region during operation.

In order to heat the core tank at the time of sodium filling, tubular type heater units are installed on the underside of the outer tank at the supporting cylinder structure. In addition, at this time, it is planned to use tubular heaters suspended from special small plugs in the top shield. These heaters will extend down into the fuel tubes of the reactor core. No other heaters on the reactor will be required. Those at the base of the outer tank are considered necessary in order to reduce the thermal stresses which would occur with the use of the suspended heaters alone.

The cavity liner which is fabricated from low carbon steel is 1 inch thick at its bottom and ¼ inch thick at its side. There are twenty-eight standard 1-inch steel pipes attached to its outer surface for purposes of cooling. At the bottom, where there is more heat load due to conduction down the supporting cylinders, these pipes are patterned to assure adequate cooling and no excessive thermal stresses. As mentioned above, the cavity liner is secured to the concrete base by means of anchor bolts. The openings in the cavity liner through which the anchor bolts project are filled with weld material to assure that the cavity liner also be leak tight.

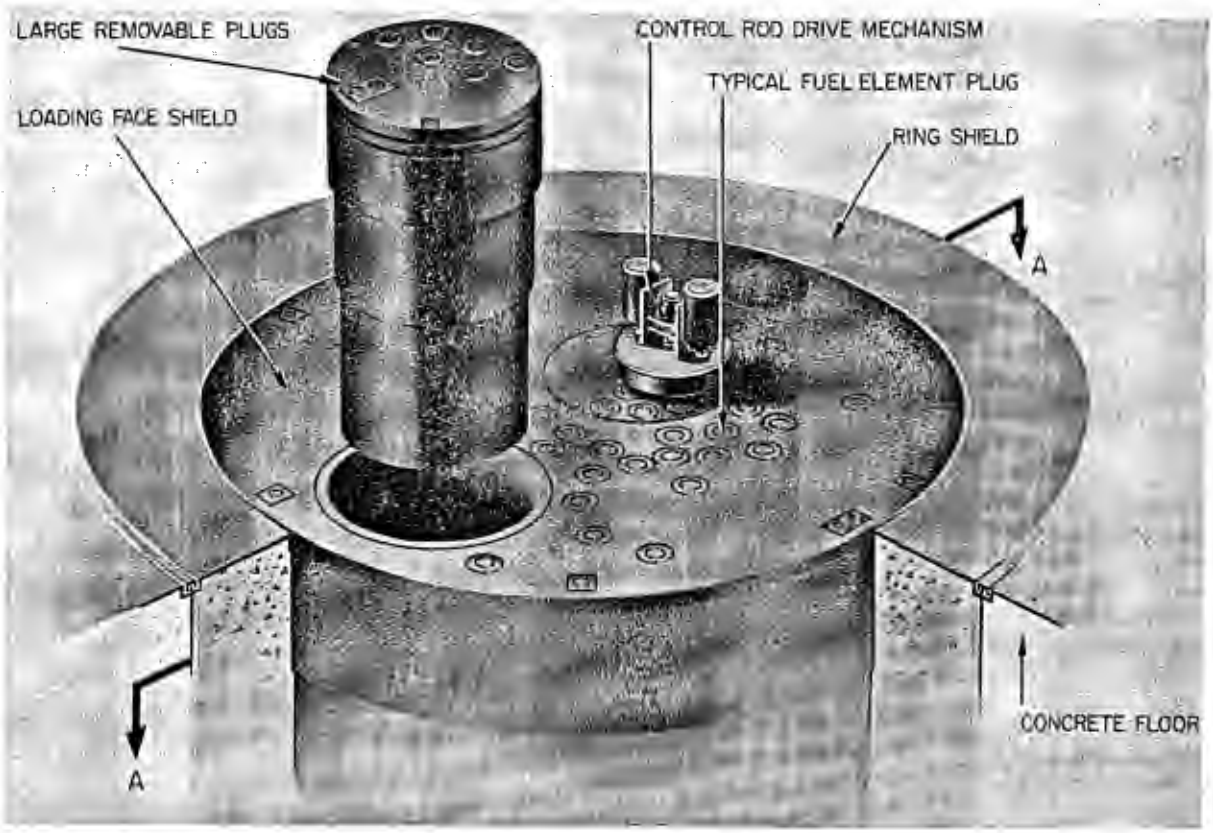
Under the cavity liner is a 4-foot thick reinforced concrete pad poured on a sandstone base. Around the side, and extending radially outward from the cavity liner, is a cylinder of reinforced concrete about 3 feet in thickness. The maximum estimated heat production in this concrete is 50 BTU/ft²-hr from nuclear radiation. This is easily removed at the cooled surface provided by the cavity liner without excessive temperatures in the concrete. Concrete is also poured around special extensions welded to the cavity liner in the region where the sodium pipes extend into the two primary galleries. Since the galleries are to operate with a nitrogen pressure of only a few inches of H_2O, the diaphragms sealing the insulation region from the galleries must be capable of withstanding a pressure differential of approximately 3 psi. These diaphragms are welded to the cavity liner and are arranged to have the necessary flexibility required for thermal expansions.

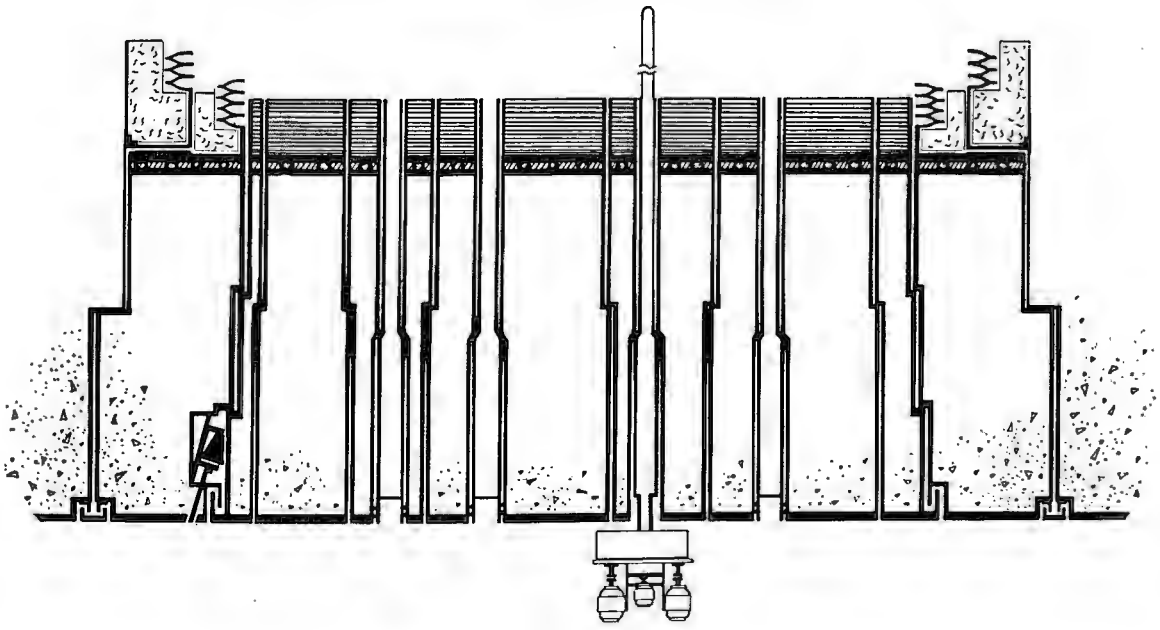
XII. TOP SHIELD

The concrete in the shielding blocks of the primary galleries, in the ring shield and in the rotatable shield is made using magnetite iron ore aggregate. This ore has a density of approximately 4.6 and is mixed in proportions leading to a final concrete density of about 3.6. All of the shields just mentioned use 6 feet of thickness of this dense concrete. The ring shield is 15 feet in outer diameter and 11½ feet in inner diameter at its top surface. It is encased in a type 405 stainless steel form and weighs approximately 60 tons. A gas seal between the ring shield and the surrounding foundation is made by the use of Cerrobend, a commercial low temperature melting alloy, cast into a suitably shaped trough at floor level.

The rotatable shield also uses type 405 stainless steel as a form and weighs a total of about 75 tons when all of its internal plugs are in place. The top stainless steel plate on the rotatable shield does not necessarily need to be welded to the many casings which extend vertically through the shield, but the plates at the sides and bottom are seal welded throughout since these surfaces come in contact with the helium and sodium vapor within the core tank. There is a total of eighty-one small plugs extending through this shield. In addition, there are three larger plugs, two about 40 inches in diameter and one 20 inches in diameter. These three are located so that there can be access for removal of any graphite assembly within the core tank if the shield is rotated to a proper position and one of these three plugs is removed. For this operation, it is necessary to melt the Cerrobend alloy in a tongue and groove seal at the outer edge of the rotatable shield. The shield must then be raised about ½ inch by means of a hoist on the handling bridge, and the shield rotated. Three built-in rollers serve to center the shield at this time. Certain of these features of the rotatable shield are shown in Fig. 14.

It is estimated that approximately 1000 BTU/ft²-hr of heat will be received by the underside of the top shield. About 85 per cent of this is due to thermal radiation from the hot sodium pool. The contribution from thermal radiation is reduced by the use of insulation in the form of a series of horizontal thin stainless steel plates suspended from the shield. There are thirteen such plates, 1/32 inch in thickness, separated by ¾ inch, and assembled in a number of sections. While it serves as a thermal radiation shield, this plate assembly is not gas-tight and will permit some sodium condensation on the seal plate at the bottom of the rotatable shield. The seal plate is type 405 stainless steel, 1 inch in thickness. On its upper surface it is in thermal contact with a 1¼ inch thick lead layer in which tubing is embedded for the circulation of toluene. Immediately above the lead layer is a 1-inch plate of low carbon steel acting as additional thermal shield for the nuclear radiation. The heavy concrete extends from this surface to the top of the rotatable shield but incorporates no other tubing for cooling. Normally, the bottom of the shield operates at a slightly higher temperature than the circulating toluene and may reach about 140°F. If, after a long period of operation, sufficient sodium condenses on the underside of the shield to short out thermally some of the stainless steel plates serving as thermal radiation shielding, it is possible to increase the temperature to the melting point of sodium by reducing the flow of toluene. At this time, the stresses in the casings for the plugs are at a maximum since the lower seal plate expands radially. As designed, this condition is tolerable, but has been made so by the choice of a low coefficient of expansion stainless steel, and by the use of relieving sleeves which are installed at the lower





SECTION A-A

Figure 14. SRE top shield

ends of the casings to prevent seizure of the casings by the concrete in this region.

All of the plugs within the rotatable shield which have been mentioned are stepped to prevent radiation streaming. The step also serves as a means of mechanical support. Sodium vapor is free to diffuse up the annulus between each plug and its casing but should condense in the low temperature region near the bottom of the shield. In all cases the final gas seal on the plugs is made by two "O" rings compressed between the plug and the casing near the top of the rotatable shield. As an extra precaution, a third gasket, depressed by a retaining ring is incorporated at the top lip of each plug, where it is available for maintenance without the necessity of lifting the plug. This retaining ring arrangement is used on all of the plugs in the rotatable shield and performs the additional function of positively locking the plugs in place.

XIII. HANDLING COFFIN

To remove any element from the reactor core it is necessary to use the shielded coffin designed for this purpose. It is carried on the 75 ton handling bridge and may be moved as required within the reactor room. To remove an element from the SRE, the coffin is located over the plug of the element in question. Electrical connectors and the retaining ring with its gasket would have first been removed. A pneumatic mechanism within the coffin then forces a cylinder vertically downward to make an "O" ring seal at the top of the plug casing. Following this, a large lead shield skirt is pneumatically lowered to the surface of the top shield. A gas lock at the lower end of the coffin, where the seal has been made to the casing, is then evacuated by means of a pump and purified helium admitted to the 3 psig pressure existing in the core tank atmosphere as well as in the atmosphere of the body of the coffin. This lock is then opened to the atmosphere within the coffin and a latch mechanism is lowered until it engages the top of the plug of the element to be removed. The direction of the motion is then reversed to raise the plug with its element up into the coffin. A separate mechanism then rotates this entire assembly which performed the lifting operation around to bring a new element into position over the center of the casing. This procedure is then reversed to lower the new element into place, disengage the latch, retract the lifting mechanism into the coffin, close the opening between the lock and coffin body, evacuate the lock, admit air into the lock, raise the shielding skirt, and break the seal made by the lock at the casing. The coffin is then free to transport the irradiated element to the cleaning or storage facilities at the other end of the reactor room.

The coffin just described is 35 feet in over-all height and weighs 50 tons. It has a maximum thickness of 9 inches of lead shielding near its base. This thickness is decreased to ½ inch near the top. No special provision is made for removing heat from the coffin except for a blower which circulates air between the coffin body and the lead shield. Mechanical motion within the coffin is obtained by means of externally mounted motors which rotate shafts sealed at the coffin wall with lubricated "O" rings.

It is planned at a future date to construct another handling coffin, specifically designed for replacement of the graphite assemblies. This coffin can be used on the same handling bridge since side rails are being provided for storage of either or both of the coffins. The storage area will also be used for coffin servicing and will permit the handling bridge to be used within the reactor room with only its general purpose hook attached. The coffin for handling the graphite assemblies will necessarily be larger in diameter and shorter in height but will require a maximum of only about 5 inches of lead.

XIV. CLEANING, STORAGE AND SHIPPING FACILITIES

As an element is removed from the sodium pool in the core tank a small amount of sodium will cling to its outer surface. A special facility is installed near the end of the reactor room for removing this sodium which normally will be radioactive. There are three cleaning cells, each consisting of a steel pipe installed below ground level and equipped with six water spray nozzles. The element to be cleaned is lowered from the handling coffin into a cleaning cell. The plug on the element makes a seal to the cleaning cell casing and also provides radiation shielding. A fine water spray is then used to remove the sodium. Experiments have shown this technique to be effective without leading to excessive heat generation. After the water spray the elements are permitted to soak in the cleaning cell filled with water for perhaps fifteen minutes. The water is then drained to the liquid waste disposal system and the cleaning cell dried by evacuation. Following this, helium is readmitted to the cell. The helium is necessary to provide a heat transfer agent for irradiated fuel elements and to prevent contamination of the coffin atmosphere when the element is removed to another location.

For storage of fuel and other irradiated elements, 96 storage tubes are available. Each tube is a 4-inch low carbon steel pipe, 25½ feet long and capped at its lower end. The tubes are installed vertically and are set in a 2-foot thick layer of dense concrete at their upper ends, which are open at reactor room floor level. The plugs attached to elements in storage serve as additional shielding in this region. It is necessary to use special shielding plugs in storage tubes not containing elements to prevent excessive levels of scattered radiation in the reactor room. In order to remove the after-glow heat generated by irradiated fuel elements, each tube has a cooling pipe welded to its outer surface. The cooling pipe extends from a toluene system manifold near floor level, down, around the bottom, and up the buried tube. This storage facility is necessary not only for the spent fuel, but also to provide a means of storing all elements normally

suspended in the SRE core at such time as it may be necessary to rotate the top shield.

The storage tubes just described may be used with all elements in the SRE core, but are not suitable for the canned graphite assemblies. Three larger, 20-inch diameter tubes are installed below floor level as a means of storing such graphite assemblies, if this should be necessary. Also, the three cleaning cells are designed so that they may be easily modified to permit their utilization in cleaning the sodium from these assemblies.

No provision for chemical processing of fuel is being made at the SRE. The fuel elements will be dissembled in the hot cell and then transferred into a specially designed shipping cask by means of the handling coffin. A recess in the floor of the reactor room is available for containing the fuel cask during the loading operation.

XV. SODIUM PIPING AND VALVES

A simplified flow diagram for the SRE cooling system is shown in Fig. 15. All piping is fabricated from type 304 stainless steel with the joints welded to 100 per cent radiographic inspection requirements. Consumable type backing rings are used wherever possible. Piping in the main circuit is schedule 40, 6-inch diameter and in the auxiliary circuit schedule 40, 2-inch diameter. Flexibility to accommodate thermal expansions is achieved by bends. The sodium velocity in the main and auxiliary circuits is nominally 13 feet per second and 5½ feet per second, respectively. Anticipated total pressure drops due to all components are approximately 17 psi and 7 psi in the main primary and auxiliary primary loops, and 36 psi and 17 psi in the main secondary and auxiliary secondary loops. In order to heat the piping and vessels in the cooling system for sodium filling, tubular type heaters and thermal insulation are strapped to all surfaces. Sodium leak detecting cable is attached to the underside of the piping and vessels. Instrumentation in the form of thermocouples, electromagnetic flow meters, and pressure indicators is installed to indicate the performance of the sodium system during operation. Level indicators are also used in the fill tanks and expansion tanks.

Gas vent pipes are connected to the sodium system to permit filling with sodium and to make possible regulation of the gas pressure over the free sodium surface in each expansion tank. The core tank serves as the expansion tank for the primary loops. As a result, the level of sodium above the reactor core is dependent upon the sodium temperature. Over the temperature range to be utilized during operation this sodium level will vary approximately ½ foot. The fill tank for the primary system, which is empty when the sodium is in the reactor, is connected to the core tank by an unobstructed vent line and acts as a ballast to prevent large pressure variations due to temperature or to sodium level changes. The main and auxiliary secondary loops each have an expansion tank in which the gas pressure is regulated to assure a higher sodium pressure in the secondary system than in the

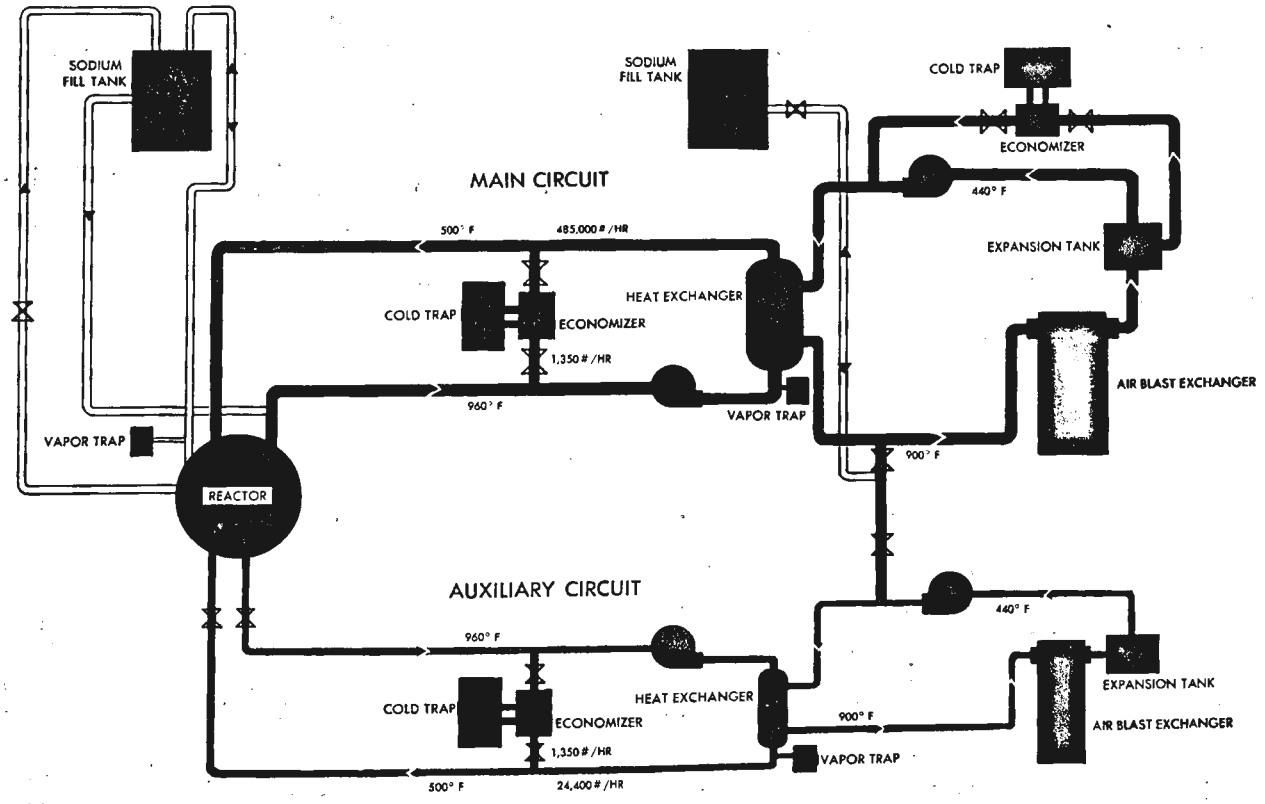


Figure 15. Sodium flow diagram for SRE cooling system

primary. Thus, if a leak should develop between the two systems at either of the intermediate heat exchangers, the flow of sodium would be toward the radioactive primary side. The gas vent lines contain vapor traps to condense any sodium vapor which might otherwise be transported into the gas vent system. In locations where it is possible that liquid sodium could be forced into such a line, a freeze trap is incorporated to solidify the sodium and block the line. These freeze traps have installed heaters which can be used to melt the sodium out at such time as this may be required. The freeze trap serves the function of preventing the sodium from solidifying in parts of the vent line which cannot be conveniently heated to reopen the line for gas flow.

At such time as it may be necessary to empty the cooling system of sodium, drainage will be to the primary and secondary fill tanks. Drain lines and electromagnetic pumps are provided for this purpose in both the primary and secondary systems. In the primary system the drain line reaches to the bottom of the core tank and in the secondary system to the outlets of the intermediate heat exchangers which represent the lowest points in the two secondary loops. Other connections to the sodium system are available for cleaning and flushing the piping, if that should prove necessary.

All valves in the primary and secondary sodium systems use frozen sodium stem seals. Bellows seal valves are restricted to use on drain lines and in the sodium service system. An example of the frozen sodium sealed valve is shown in Fig. 16. A gland circulating liquid toluene replaces the valve packing and causes any sodium in the annulus around the stem to solidify. The valve may be opened and closed by rotating the stem and shearing the frozen sodium. Helium gas is admitted to the region between the frozen seal and the top of the valve to avoid oxidation of the lip of the sodium annulus. There is an additional seal using conventional packing at the valve top to confine the helium gas. Cam lift plug valves are used as blocking valves at the inlet and discharge to the core tank in both the main and auxiliary primary loops. An angle valve is used to throttle the flow of sodium for moderator cooling in the branch line from the main inlet to the core tank. The plug valves in the primary system make it possible for either of the primary loops to be drained independently. Since the secondary loops are separate, they may be independently drained without the use of blocking valves.

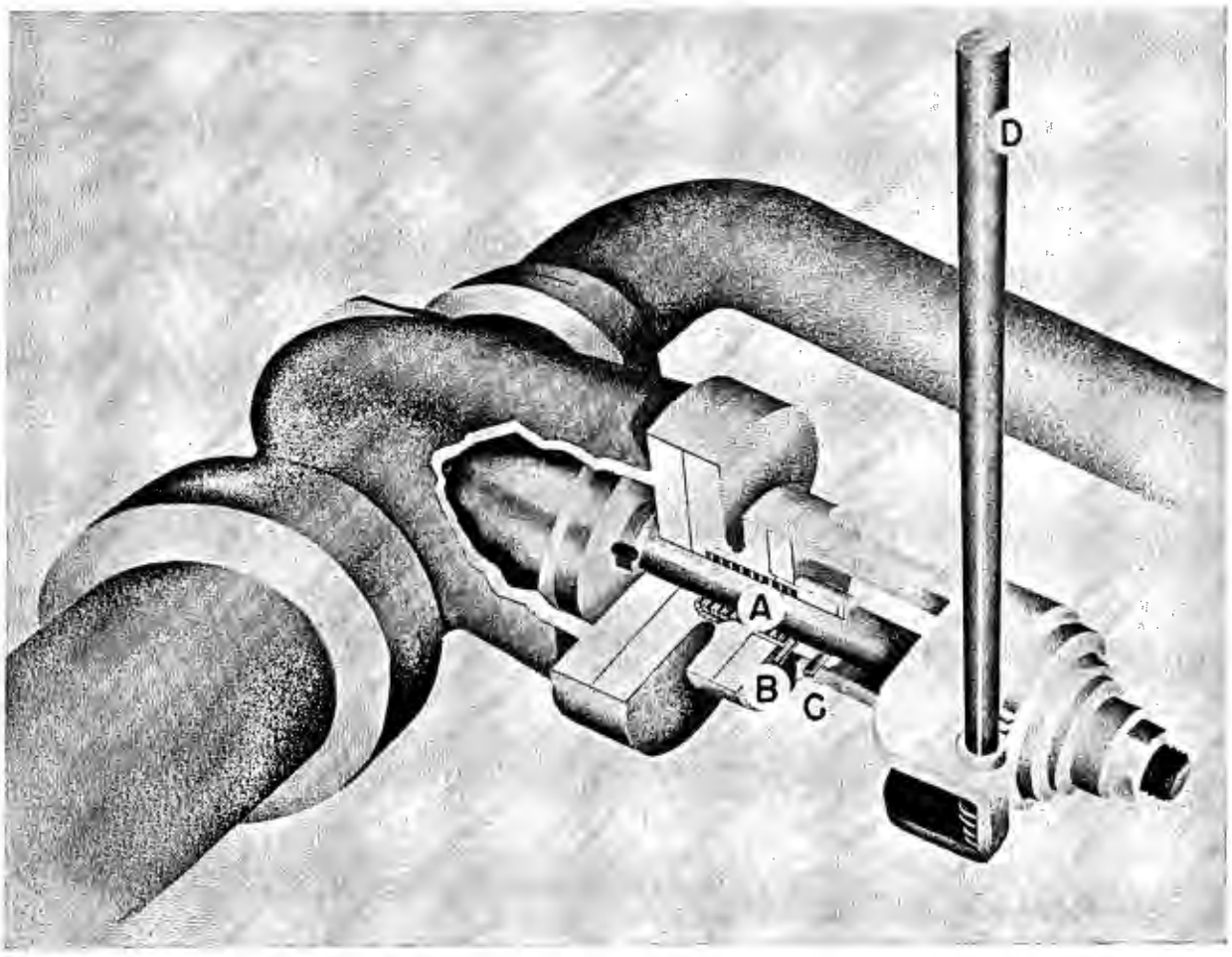


Figure 16. Arrangement for valve with frozen sodium seal

(A) Shaft seal, (B) Coolant line, (C) Gas line, (D) Valve operating shaft

XVI. SODIUM PUMPS

The sodium pumps are modified hot process pumps similar to the type used in refinery service. The principal modifications consist of vertical mounting and the addition of frozen sodium seals at the shaft and at the case. Each of the primary pumps also has its case and drive shaft extended sufficiently to pass up through the gallery shielding to floor level. This arrangement makes it possible to service either primary pump by withdrawing all of its internal parts through the case into the reactor room. Lead shielding is incorporated into this removable assembly just below the floor level.

The case also provides a container for a helium atmosphere maintained at 10 psig, to protect the frozen sodium seal from contact with oxygen. The helium is sealed from the atmosphere by an "O" ring and lubricated face seal at the shaft penetration.

An illustration of the frozen sodium seal pump is given in Fig. 17, and a cross section of the main secondary pump in Fig. 18. Since no shielding is required in the secondary system, the pumps used there do not need the extended case and drive shaft arrangement. The shaft seal is made at a cooled gland inserted in the space normally provided for the pump packing. Toluene coolant is circulated through this gland and through a cooling coil provided at the pump housing for the case seal. Toluene cooling is also circulated to a radial bearing located a short distance above the shaft seal.

Oil for lubrication is fed to this bearing and is continuously removed by being entrained in the helium atmosphere bleed line from a sump immediately below the bearing.

Experiments have been conducted using a prototype of the pump intended for use in the main sodium circuit. With a 50 hp drive, this pump is capable of delivering 1285 gallons a minute against a 130-foot sodium head, while at a speed of 1460 rpm. The shaft power dissipated in the freeze seal is less than 1 kw. Several kilowatts of heat removal are required at both the shaft seal and the case seal in order to maintain the sodium in a frozen condition. The pump is fabricated from type 304 stainless steel except for the shaft and impeller, which are type 316. The diameter of the shaft is 3 inches and that of the impeller is 13½ inches. The impeller is overhung a distance of 15 inches below the lower bearing. Pumps for the auxiliary circuit are similar in design, but of smaller capacity.

The impeller diameter for these pumps is 10 inches.

All of the pump motors are variable speed drive and are accessible during operation. The main primary motor is 25 hp, the main secondary motor 50 hp, and the two auxiliary motors are 2 hp each. During operation of the SRE it will be necessary to adjust the pump speeds as well as the cooling conditions in the air-cooled exchangers to remove the heat generated in the reactor core at required operating temperatures.

XVII. HEAT EXCHANGERS

The intermediate heat exchangers use a shell and tube counterflow arrangement with a "U" type design. This configuration conserves space and minimizes thermal stresses. Both intermediate exchangers use type 304 stainless steel throughout with the tubes being single wall, seamless, ¾-inch outside diameter, 0.058-inch thick wall. The main intermediate heat exchanger has 316 such tubes with a total heat exchange area of 1155 square feet. At the design conditions of 4 feet per second flow in the tubes, the heat transfer coefficient is approximately 985 BTU/hr-ft²-°F. In the auxiliary intermediate heat exchanger, only 38 tubes are used and with the design flow of 1.5 feet per second will have a heat transfer coefficient of approximately 800 BTU/hr-ft²-°F.

The air-cooled exchangers in the secondary system have "U" shaped tube bundles installed in protective housings above motor driven fans. The tubes are similar to those in the intermediate exchangers but have mounted type 410 stainless steel fins 2 inches in diameter, spaced eight per inch. All tubes are connected in parallel from headers at one side. The direction of airflow is concurrent to the sodium flow to reduce the danger of freezing in the tubes at low sodium flow rates. Mechanically operated louvers are installed above the tube bundles to assist in the control of the flow of air. Just below the tube bundles are mounted tubular electric heaters for preheating the system and for providing make-up heat during operation at low power. These heaters will maintain a temperature of 350°F in the tubes during shutdown of the reactor.

In the main air-cooled exchanger there are 204 "U" tubes 26½ feet in length. They provide 23,600 square feet of heat removal surface. At normal operating power with the estimated heat transfer coefficient of 8.2 BTU/hr-ft²-°F, the exhaust air temperature will be 277°F based on air in at 100°F. Below the tube bundles are mounted two 11-foot diameter fans each driven by a 50-hp variable speed motor. The auxiliary air-cooled heat exchanger is similar in design, but has only thirty "U" tubes 8 feet in length. It utilizes a 5-foot diameter fan driven by a 5-hp variable speed motor.

In the SRE the intermediate heat exchangers are mounted in the galleries at an elevation above the reactor core. The air-cooled exchangers are installed at a still higher elevation. As a result, even without forced circulation by the pumps, it is possible to obtain cooling by natural convection in both primary loops and in both secondary loops. This arrangement constitutes an excellent safety feature since it is possible to remove a certain amount of after-glow heat from the reactor core under emergency conditions. It is estimated that with normal operating temperatures, as much as 2000 kw of heat could be removed by natural convection circulation within the complete main circuit, including the cooling by air convection at the main secondary exchanger.

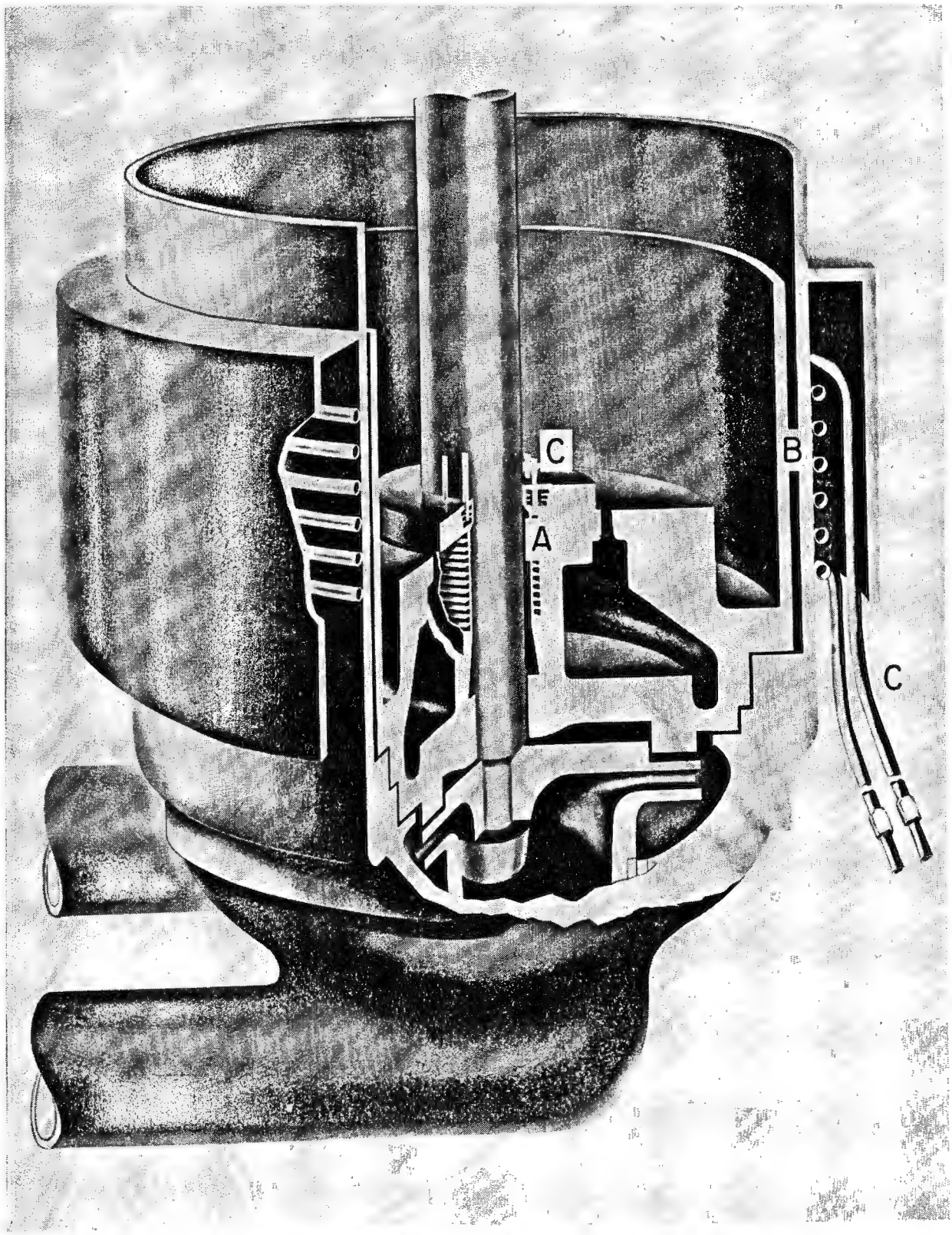


Figure 17. Arrangement for mechanical pump with frozen sodium seals
(A) shaft seals, (B) housing seal, (C) coolant lines

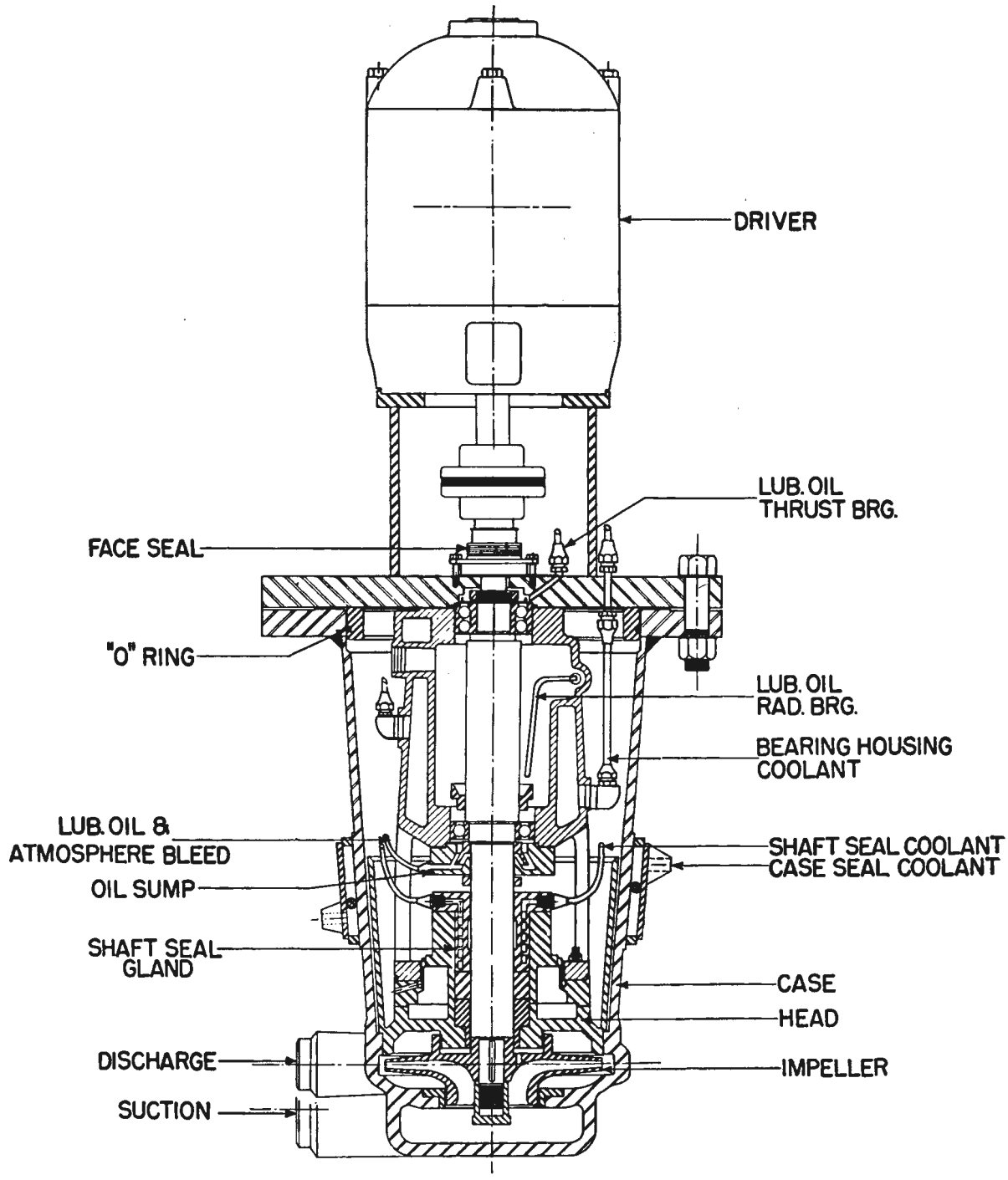


Figure 18. Cross section of main secondary sodium pump

XVIII. COLD TRAPS

Cold traps are incorporated in each of the four loops of the sodium cooling system in order to maintain the sodium oxide content at a sufficiently low value. A special design of circulating cold trap is being developed for use in the primary loops and the secondary main loop. This design is schematically shown in Fig. 19. The principal new feature is the toluene jacket and condensing loop to maintain the low temperature of the sodium in the trap at a nearly constant value in spite of variations in sodium temperature and pressure in the cooling system during operation. The lowest temperature of the sodium in the cold trap will be a few degrees above the 232°F boiling point of the toluene. The toluene which boils from the jacket surrounding the cold trap passes into a condenser and then as a liquid into the reservoir which controls the liquid level within the jacket. Separate cooling for the condenser is provided from the toluene system servicing the various SRE components. Pressure for circulating the sodium through the cold trap is automatically obtained by connecting it across the pump in any one loop. The sodium first passes through an economizer section, and then through the cold trap vessel which is filled with stainless steel mesh. This particular trap is designed for a flow of 1350 pounds per hour of sodium and is expected to maintain the sodium oxide content below 0.005 weight per cent.

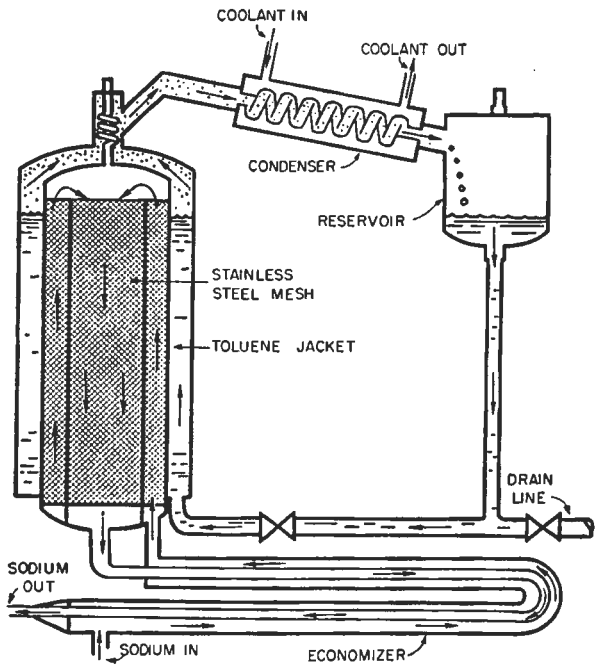


Figure 19. Sodium oxide cold trap utilizing boiling toluene jacket

Where the capacity of the cold trap is less important, it is planned to use diffusion cold traps. One will be attached at the base of the surge tank in the second auxiliary loop and one on the secondary fill tank. These units each consist simply of an 8-inch diameter capped pipe 15 inches long welded to the bottom of the tank. At the base of the diffusion trap a pipe for circulating toluene coolant is attached by welding. Sodium in the base of the diffusion trap may be frozen but the heat losses with this arrangement are negligibly small.

To determine the oxygen content in the sodium, a plugging meter is used in conjunction with each circulating cold trap. The plugging meter is connected in parallel with the cold trap. Its principle of operation depends on the closing of small orifice holes as the temperature of the sodium is lowered and the sodium oxide precipitates out. The sodium oxide content may be determined after calibration by the break in the sodium flow versus temperature curve. In order to speed the rate to which successive plugging meter determinations may be made, a special design is being developed in which the orifices are machined into the seat of a valve. The valve may be opened for a short period to encourage dissolution of the sodium oxide and prepare for another determination. The plugging meter also incorporates an economizer and a cooling line from the toluene system. An electric heater is used to vary the temperature.

XIX. SODIUM SERVICE SYSTEM

The sodium service system comprises equipment for melting, filtering and introducing sodium to the fill tanks as well as for flushing the primary system cold traps. Type 304 stainless steel is used throughout. When the sodium service system is not in use, the lines and vessels are emptied of sodium and a helium atmosphere maintained in the system. All vessels and sodium lines are provided with heaters and insulation in order to preheat to 350°F.

Sodium is first introduced from drums at two melt stations and flows through sodium filters at about 250°F. It may then be directed to either the primary or the secondary fill tank. To transfer the sodium from the primary fill tank into the primary system, it is necessary to use the sodium pumps, once prime has been established by gravity flow. This method is necessary since the primary fill tank communicates to reactor core tank which cannot be pressurized. In the secondary system filling of both the main and auxiliary loops is accomplished by pressurizing the fill tank.

Since the primary system becomes radioactive, special provision has been made for flushing the cold traps in both primary loops to a disposable cold trap in the sodium service system. An electromatic pump and valves are so arranged that hot sodium can be passed through either primary cold trap in order to carry the sodium oxide to the disposable unit. This disposable cold trap in the service system may also be used for removing the sodium oxide from sodium being stored in the primary fill tank, without requiring circulation into the primary system piping. There is no diffusion cold trap on the primary fill tank.

XX. TOLUENE SYSTEM

As a safety precaution all components in the SRE requiring special cooling are served by the system

circulating liquid toluene. This material does not chemically react with sodium, but can be pyrolitically decomposed at sufficiently high temperatures. A summary of the components to which toluene cooling is provided is given in Fig. 20. Toluene from a reservoir containing 500 gallons is circulated by two pumps operated in parallel. Throttle valves for each of the major components served are used to regulate the flow to obtain a temperature increase of about 30°F. The toluene is supplied from the reservoir at 95°F. The maximum estimated flow is approximately 360 gallons per minute, which with the temperature rise just indicated, provides for the heat removal of roughly 500 kw. This amount includes 150 kw allowed for the fuel storage tubes. Other principal heat loads are approximately as follows: top shield, 60 kw; cavity liner, 70 kw; main primary loop, 80 kw; auxiliary primary loop, 55 kw; main secondary loop, 45 kw; and auxiliary secondary loop, 40 kw. The return toluene is cooled in two evaporative cooler units operated in parallel and is then piped to the reservoir tank. Low carbon steel pipe and vessels are used throughout. Separate motors drive the two circulation pumps. In the event of power failure these motors will be automatically operated from the emergency electrical system. In order to reduce the load on this emergency system, a gasoline engine is also provided which may be started and coupled to either of the toluene pumps.

There are certain disadvantages to the use of toluene instead of water as a special coolant. Since its flash point is quite low, the use of toluene is accompanied by a fire hazard, and the vapor is also somewhat toxic. Other hydrocarbons have been considered as alternates for the toluene and may be substituted at some later date without requiring any system modification. Two alternates being considered are xylene and tetralin, of which the tetralin appears the more promising. In addition to a higher flash point and lower toxicity, tetralin would have the advantage of a substantially higher boiling point, while still being liquid at room temperature. Other characteristics of the liquids mentioned, including radiation decomposition rate, do not appear to offer special problems.

XXI. INERT GAS SYSTEM

An inert gas system provides helium and nitrogen at various locations around the SRE installation. Both gases are stored in banks of high pressure bottles in an area outside the reactor building. In the case of each gas, the pressure is reduced to 50 psi in standard regulators and is directed to a low pressure storage tank. In order to minimize the oxygen and water vapor content of the helium, this gas is passed through NaK bubblers prior to introduction into the low pressure tank. Distribution from these storage tanks is then made to various points within the reactor building where individual regulators further reduce the pressure to a value suitable for each component serviced. Over-pressure safety valves are provided at various points. The gases are non-circulating during operation except for the helium used to circulate lubricating oil in the sodium pumps.

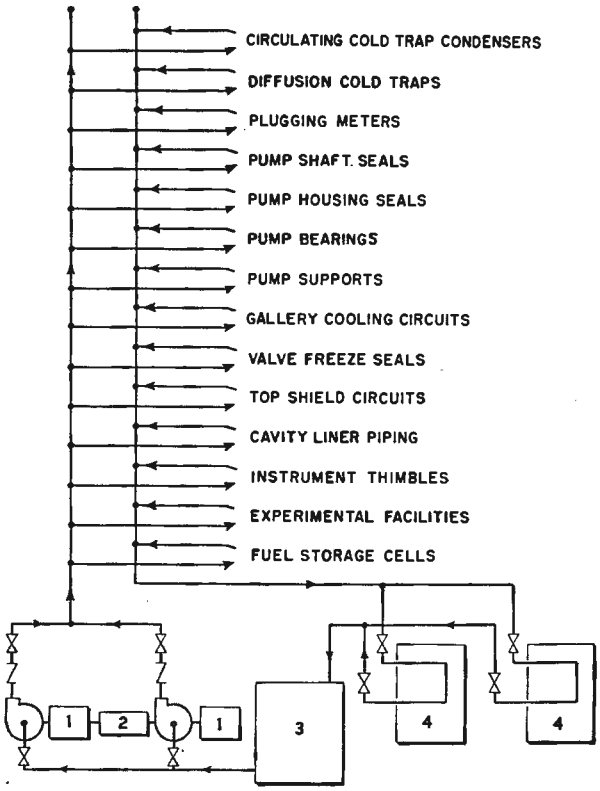


Figure 20. Components serviced by toluene coolant system; (1) motor; (2) emergency gasoline engine; (3) toluene reservoir; (4) evoporative cooler

The helium is used in numerous locations. Examples are the fuel handling coffin, cleaning and storage facilities, sodium pumps and valves, control and safety element thimbles, regions surrounding the core tank and within the double walled inlet pipes, as well as all components in the sodium system which require a free sodium surface during filling and draining or during normal operation. Examples of the latter are the fill tanks, expansion tanks, and cold traps.

The nitrogen distribution is somewhat simpler since this gas is used primarily only for the various galleries and the cavity containing the reactor insulation. To reduce the amount of nitrogen used while maintaining the gallery pressures at a very small value in spite of temperature fluctuations, the gallery atmospheres connect to a constant pressure tank located a short distance from the reactor building. These galleries may be isolated by valves in the piping, but normally will have a pressure of ¼ psig as automatically established by the movable top on the constant pressure tank.

Where helium or nitrogen are provided to regions which will contain sodium vapor only in the event of a leak in the sodium system, it is planned to sample the gas periodically. This method of sodium leak detection is especially useful for components in which electrical shorting detectors cannot be conveniently installed. Examples are the control and safety ele-

ment thimbles and the double walled sodium inlet pipes. The method of detection to be used involves bubbling of a gas sample through an indicator solution of thymol sulfonthalein. It is not difficult by this means to detect as little as 10^{-7} mol of sodium in the gas sample. Use of this technique does not require heated tubing for transporting the gas containing sodium vapor. Experiments have shown the vapor to be almost quantitatively transported through 30 feet of ¼-inch tubing at room temperature.

XXII. WASTE DISPOSAL SYSTEM

Provision is made for the disposal of both gaseous and liquid radioactive wastes. The gaseous waste problem arises from the possibility of ruptures in the fuel element cladding which might release gaseous fission products into the primary sodium system. As a result, each gas line which connects into the primary system and is required to vent helium for reasons of pressure control, is connected in such a way that if excessive radioactivity is detected, this gas will be pumped into radioactive storage tanks. This is accomplished by means of a radiation detector coupled to valves which automatically shunt the gas stream being vented into a compressor suction tank whenever an excessive level of radioactivity is detected. From the suction tank, one of two parallel compressors forces this gas into a storage tank where it can undergo radioactive decay. There are two shielded storage tanks of 5400 cubic feet at 100 psig capacity. If at some later time the activity level is determined to be sufficiently low, this gas may be bled from the storage tanks and discharged out the building vent line. This line emerges at the roof of the reactor building where the gas released is diluted by the forced circulation from the building ventilation system.

The arrangement just described is shown in simplified form in Fig. 21. As indicated there, the principal components which may require gas venting to the radioactive storage tanks are the core tank and primary fill tank, the gas lock on the fuel handling coffin, and the fuel cleaning cells.

The liquid waste disposal system is somewhat simpler, as shown schematically in Fig. 22. The only components producing radioactive liquid waste are the cleaning cells and the hot cell. Drains from these components will normally carry water with radioactive contaminants directly to a sump. From here a pump will force this liquid up into one of a series of ten 50-gallon hold-up tanks, as determined by valve settings.

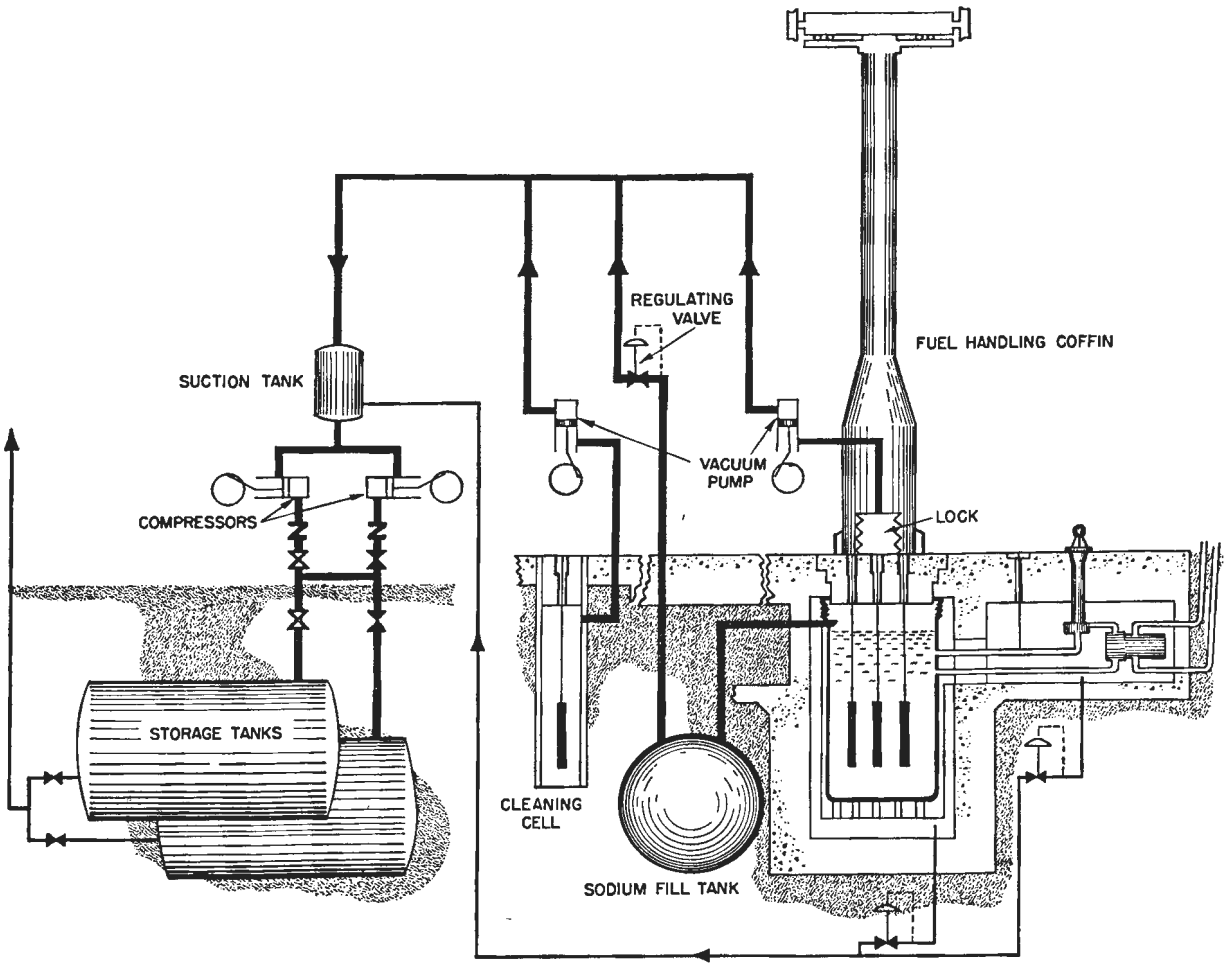


Figure 21. General arrangement for gaseous waste disposal system

Liquid in these tanks may be stored and sampled as required. Depending upon the radioactivity level after this period of hold-up, the waste is directed into either of two 5000 gallon storage tanks, or to the industrial sewage system in the case of very low radioactivity levels. No further provision is made for disposal, although if these large storage tanks should ever become full, it will be necessary to finally concentrate the waste and dispose of it by special means.

XXIII. EMERGENCY ELECTRICAL SYSTEM

In the event of failure of the electrical power provided to the reactor building, there are some SRE components which must be maintained operative on the emergency electrical system. The reactor itself will be automatically shut down, but instrumentation and heat removal equipment must continue to operate. This total emergency load approaches 50 kw, of which almost one-half is required for toluene pumping.

The continuous emergency power comes from a battery driven motor alternator. This set consists of a 100 hp synchronized motor and a 60 kw dc diverter pole generator. It will act as a battery charger and dc power source during normal operation and as a battery driven motor alternator during emergency operation. Because of the continuous operation of the motor generator set necessary to insure emergency power, two identical sets are provided with switching arranged so that either one can be idle for maintenance. The battery is rated at 120 volts and is capable of delivering 480 amperes for ½ hour and 280 amperes for an additional hour.

As a backup for the battery supply there is a delayed emergency power unit in the form of a Diesel engine driven alternator rated at 100 kw. This engine alternator will start automatically at loss of normal power. When up to proper speed and voltage, the output of the alternator will be automatically synchronized with the output of the motor generator and the two will be paralleled. The Diesel set will then take over the entire load and the motor generator will become a battery charger again. Since the toluene pump comprises the largest emergency load a separate gasoline engine is installed there, as described previously. This engine is of greatest value in case the Diesel alternator cannot be started, at which time complete dependence will be on the batteries which have limited life. When normal power returns, the electrical output of the emergency system is automatically synchronized and the load is switched.

XXIV. EXPERIMENTAL FACILITIES

The purpose of the construction and operation of the SRE is not only to carry through the engineering of an experimental sodium-graphite type reactor but to perform measurements and system modifications which will establish performance limits and design improvements. In the previous sections the design of the SRE for initial installation has been described with some discussion of the engineering choices which have been made. This description has been based on a set of normal operating conditions. It is realized that some experience will be required during the early stages of operation to even achieve these design conditions, but it is also hoped that later in the program it will be possible to exceed these conditions and better establish the limits of performance of the various components.

As one step in striving for high performance, an attempt will be made to operate at maximum sodium temperatures approaching 1200°F. As designed, the vessels and piping in the reactor and sodium system

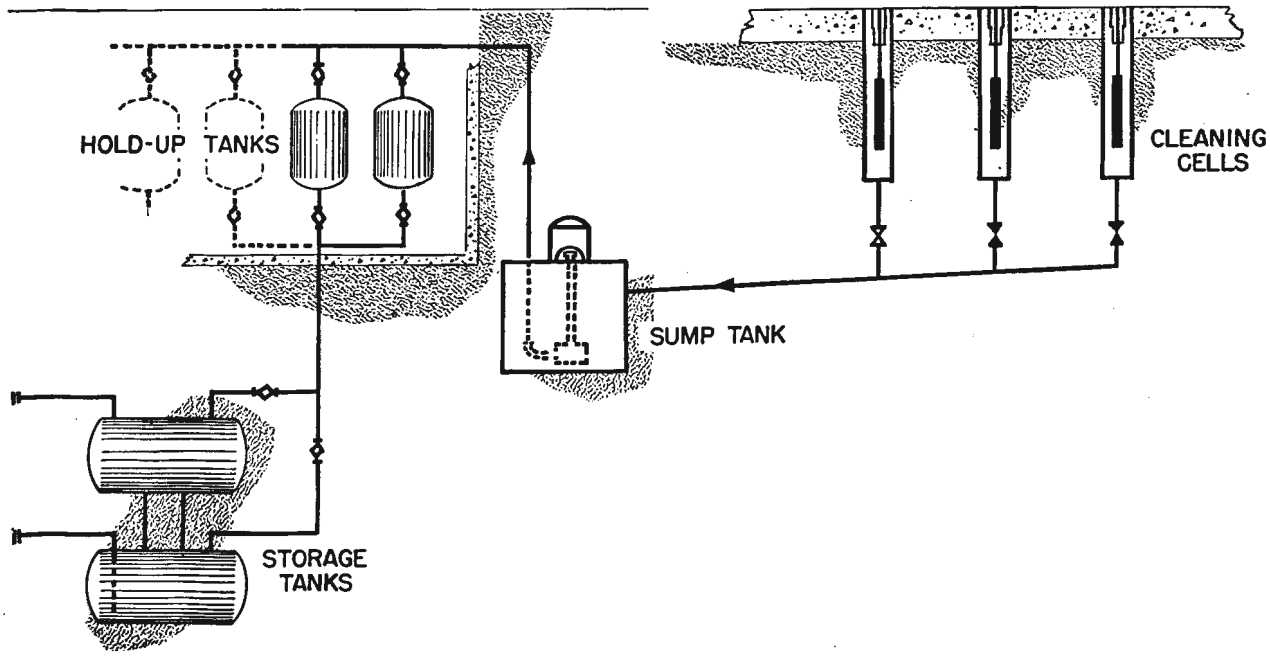


Figure 22. General arrangement for liquid waste disposal system

have been stressed for this temperature condition. To achieve such operation will probably require modifications in the fuel and in the zirconium canned graphite assemblies. Actually, one of the principal objectives of the experimental work is in connection with improvements in the reactor fuel, especially from the standpoints of specific power, operating temperature, and burn-up. Measurements will also be made on the graphite assemblies to determine their outgassing properties and mechanical integrity. In a similar manner, measurements will be made on many other components such as the control and safety elements and sodium pumps.

The hot cell, located below floor level at the end of the reactor room, provides a facility for examination of any radioactive components removed from the reactor. Its principal use will be in connection with fuel element studies, but measurements can also be made on other parts, such as graphite assemblies and control and safety elements. The hot cell is divided into two parts with internal dimensions 5 feet by 15 feet by 16 feet and 5 feet by 8 feet by 11 feet. There are three sets of manipulators and three viewing windows. Components may be inserted into the hot cell through one of three holes at the end of the cell nearest the reactor. Two of these holes are the same size as the small plugs in the reactor shield and can receive any of the elements normally suspended in the reactor core. The other hole is larger in diameter and can accept a graphite assembly. In either case, a handling coffin would be used to transport the radioactive component from the reactor and into the hot cell.

As described earlier, there are twelve fuel tubes and six corner channels in the reactor core which will probably contain dummy elements. It is possible to replace these elements with special units for experimental purposes. If it is necessary to avoid contact with the sodium, such a unit can use a stainless steel thimble attached to a shield plug. Special cooling within the thimble could be provided if it is necessary to perform the exposure at a lower temperature than that of the circulating sodium. The number of experimental units used at any one time will depend upon the reactivity requirements of the core as well as the location and type of experimental units desired. In addition to the channels just mentioned, there are three at center locations of the graphite assemblies at successively greater radii. The tubes associated with these locations do not connect into the lower plenum and hence receive only the moderator cooling sodium. They are intended for the measurement of the radiation levels at various heights and radii within the side neutron reflector, but are also available for other experimental purposes if desired.

Another experimental facility deserving mention is incorporated into the main primary loop. One-inch pipes are welded into the inlet and discharge lines of the main intermediate heat exchanger and carry sodium to two vertical 2½-inch diameter stand-pipes. These two sodium streams are then mixed and returned in a common pipe to the suction side of the primary pump. This arrangement permits insertion of samples from floor level in the reactor room down into the stand-pipes; one of which will normally contain flowing sodium at 500°F, the other at 960°F. During operation, a frozen sodium seal is made at the top of each stand-pipe. Above this is a shielding plug. It is planned to insert samples of different materials into this facility to study corrosion and mass transfer effects under actual operating conditions.

The Graphite-Moderated, Gas-Cooled Pile and Its Place in Power Production

Sir Christopher Hinton, UK

In every engineering development there is an element of chance which may sway the balance between the choice of one design or another. When I was asked to undertake responsibility for the production of fissile material in 1946, it was the intention that Great Britain should build reactors of the original Hanford type. This type of pile is graphite-moderated and water-cooled and it cannot be made inherently stable if operated solely on a natural uranium fuel. Because of this very large safety distances were considered essential. These safety distances are permissible in large continental areas, but in the thickly populated United Kingdom it was difficult to find suitable sites. Therefore it was thought that we would do better to choose a type of pile which was inherently stable and which did not demand isolation in the interests of safety. It was for this reason that our efforts were turned towards the graphite-moderated gas-cooled type of pile. It was realized that in order to bring the blower power for circulating the coolant down to the lowest practicable level it would be necessary to operate the pile under pressure and that this would introduce problems of the design of a pressure shell, of fuel element changing gear and of control rod operation. Although solution of these problems was certainly possible, it was clear that they would take time to solve. But in the course of these preliminary studies it became apparent that although a pressurized gas was necessary for really economic operation, it would be possible to operate at atmospheric pressure if sufficient flattening of the neutron distribution across the pile was done. This meant that an immediate and simplified solution of the problem of bulk plutonium production could be achieved by building graphite moderated piles cooled with air blown through them at atmospheric pressure. It was therefore largely force of geographical circumstances which led us to the graphite-moderated gas-cooled reactor. If Great Britain had not been such a densely-populated country and we had easily been able to find sites where safety distances could be met, we probably should have followed the American lead and built graphite-moderated water-cooled reactors.

THE WINDSCALE PILES

The first reactors built for plutonium production at Windscale have been described in earlier lectures elsewhere. Long sections and cross sections of one of the reactors at Windscale are shown in Figs. 1 and 2.

It will be seen that the mass of graphite moderator lies within a concrete biological shield designed to reduce the escape of neutrons and gamma rays to the point of biological safety. The biological shield is itself protected by a steel plate thermal shield and lagging which guard the concrete against excessive temperatures and radiation effects.

The general structure of the graphite core consists of layers of accurately machined graphite blocks perforated with horizontal channels. The graphite structure is retained in position by spring loaded structural steel members which hold the structure in compression while allowing for thermal and other movement. The charge platform is housed within the main biological shield wall because it would have been inconvenient to charge through the full wall thickness. Irradiation intensity in the charge hoist well is below tolerance when the pile is shut down but access is not possible when the pile is working.

On the discharge side a water duct is provided below the pile, and this passes through the main shield to connect the discharge space with the main storage pond. Irradiated cartridges, which are discharged from the channels in the graphite core fall into a hopper carried by a truck on the under-water railway. They are transferred to the pond in this for cooling prior to processing in the chemical separation plant.

Control of the pile is effected by means of 24 boron steel control rods clearly seen in Fig. 2. These rods are arranged horizontally at right-angles to the uranium channels and they are moved by electric motors operated from the pile control room. Emergency shutdown arrangements are provided by 16 boron steel shut-off rods suspended through the pile vault roof by means of electromagnets. In all emergencies the shut-off rods will fall into the graphite core within seconds, thus shutting down the pile.

Cooling air is supplied to the pile by a number of double entry radial flow centrifugal blowers running normally at 990 rpm. Each is driven by an 11 kv wound-rotor induction motor giving 2400-3000 hp. They are situated in two fan houses the walls of which contain inlet air filters, consisting of both wet and dry filters. The floors of the blower houses are of dust-proof concrete and the concrete air ducts are lined with Alclad sheets to prevent erosion of the concrete (see Fig. 3). Continuous monitoring of the air in the

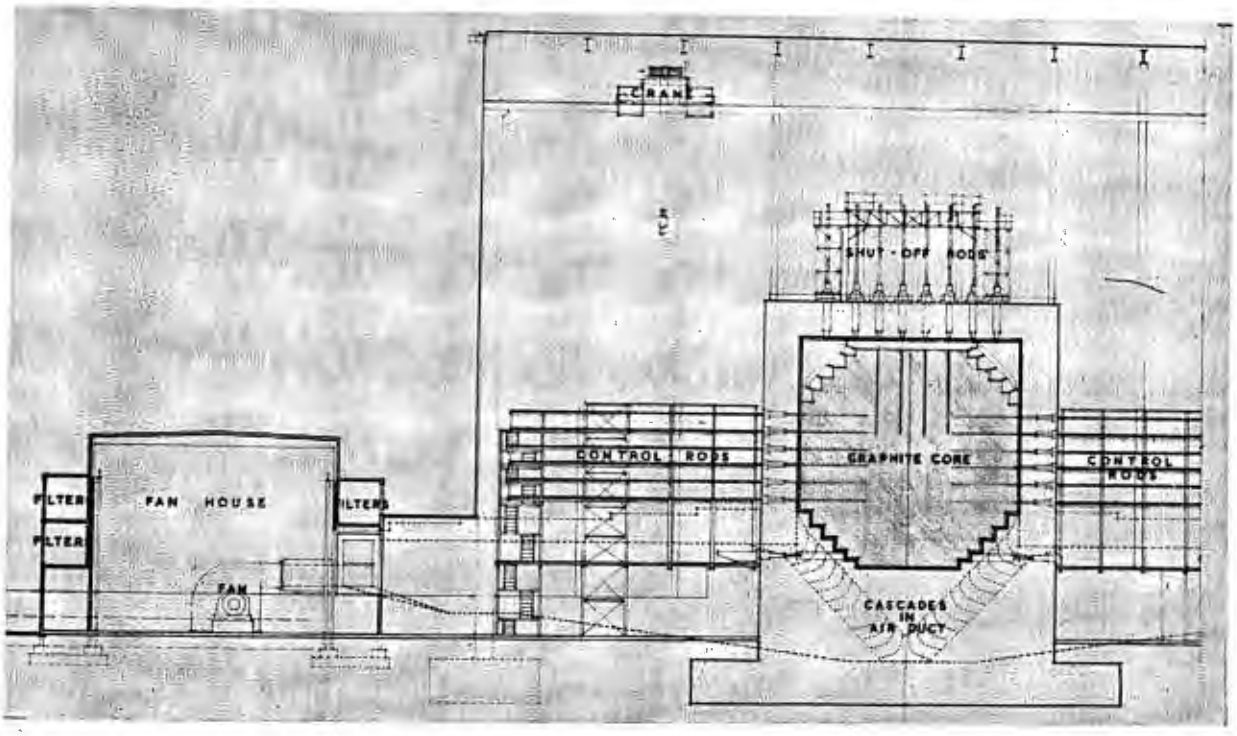


Figure 1. Longitudinal section

uranium channels is carried out in order to detect burst slugs and the cooling air is filtered once more in the filter gallery at the top of the 400 ft stack before it is exhausted to atmosphere. These filter units are removed by remote control after use as they may be radioactive and they are taken to the bottom of the chimney in a protective lead coffin by means of a 7½ ton hoist.

THE CALDER HALL PILES

In the Windscale piles all the heat is wasted. While they were being built study of the graphite-moderated pressurised gas-cooled pile from which power would be produced was continued, and in 1953 it was decided that it was reasonable and prudent to proceed with the construction of such piles at Calder Hall in Cumberland. In principle the pile is shown in Fig. 4. It will be seen that the core of the reactor is surrounded by an octagonal-shaped concrete biological shield several feet thick, and this shield is protected from thermal and radiation effects by a thermal shield made of 6 in. mild steel plate.

Induced draft air-cooling is supplied to the space between the thermal shield and the biological shield and the outlet air is exhausted 200 ft above ground level from the two stacks on the pile building.

The mass of graphite moderator, weighing over 1000 tons, is contained in a cylindrically shaped pressure vessel. The machined graphite blocks have vertical uranium charge and control rod channels, the adoption of vertical channels simplifying the graphite structure and making the design of the coolant circuit more convenient. The weight of the graphite is taken on a diagrid structure which consists of an I-section ring girder spanned with steel girders forming a rectangular lattice. Across the members of the diagrid lattice are bolted 4 in. thick steel base plates which have holes in them to coincide with the channels in the graphite. These base plates support the graphite through a number of races which allow for differential thermal expansion of the diagrid and graphite. The weight of graphite and diagrid is taken by brackets directly through the walls of the pressure vessel on to ten inverted A frames which rest on the thermal shield. These A frames are loosely bolted to the brackets on the pressure vessel and on to the thermal shield, the three contact surfaces of each A frame being radiused. In this manner, allowance is made for radial thermal expansion of the pressure vessel and diagrid, when the A frames roll slightly out of the vertical, and the bolts ensure that the rolling surfaces do not slip.

The pressure vessel is of 2 in. welded steel plate and is about 40 ft in diameter and approximately 60 ft high. The construction of the vessel provided a novel and interesting job. Fabrication was carried out on the site, all parts (except the inlet manifold) arriving piece small. Extensive site welding facilities were set up and the vessel was built up into five main sections, namely the bottom dome, the two parallel centre sections, the diagrid and finally the top dome. As each section was completed it was transported to the lifting area where a 100 ton crane was used to hoist the sections and to lower them through the pile vault roof, the final welds being completed *in situ*. Each butt weld was radiographed and all important fillet welds were examined by crack detection techniques. The task of stress relieving the completed ves-

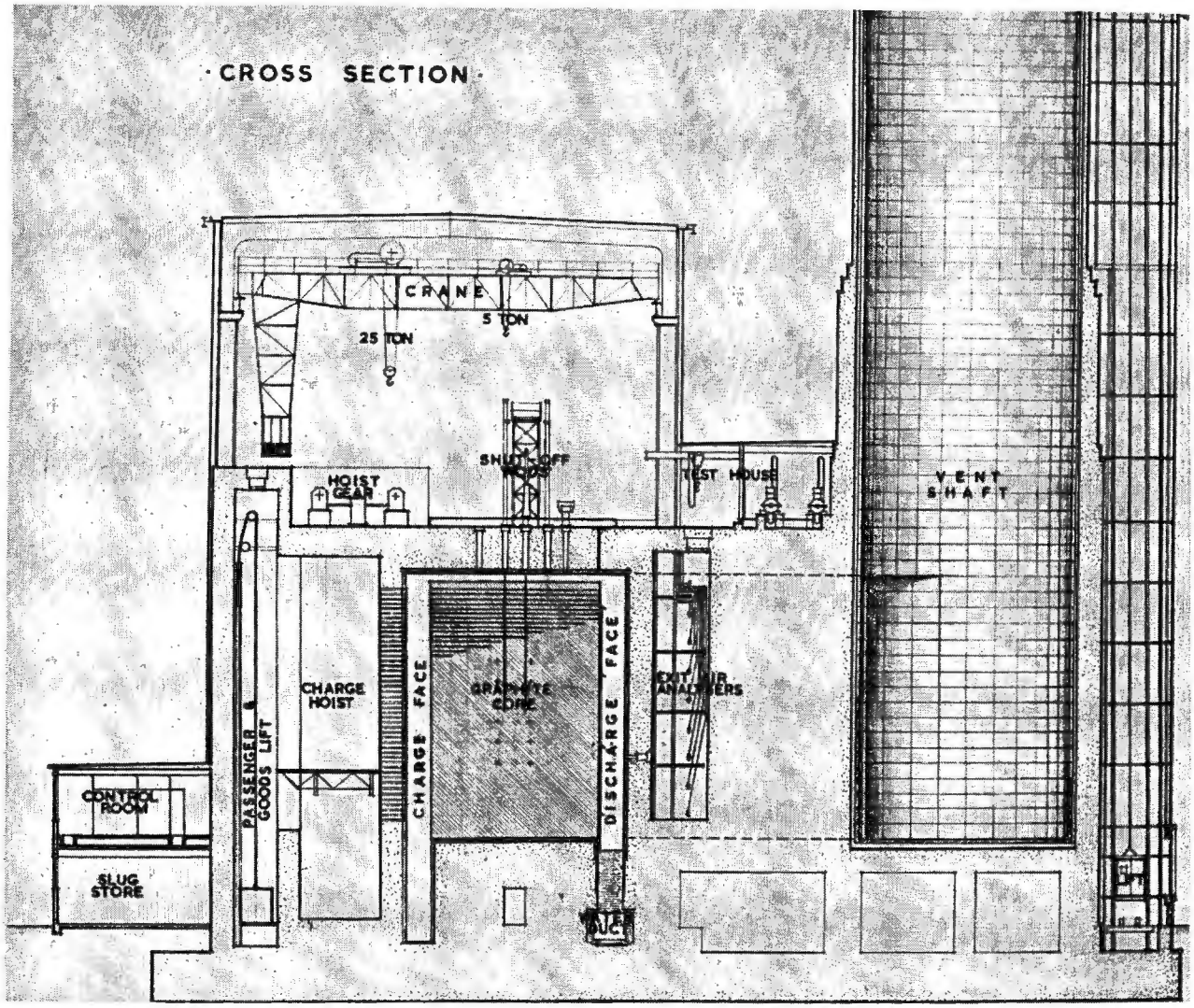


Figure 2. Cross section

sel was achieved by radiant heating which required a peak electrical load of about 1½ megawatts. The vessel temperature was brought up to over 550°C and held there for 8 hours before being allowed to cool slowly and evenly.

The vessel was then subjected to pressure test. The use of hydraulic testing techniques would have overloaded the structure and for this reason a pneumatic test was carried out. A constant check on the behaviour of the vessel by means of strain and other gauges was maintained throughout the test and formed an important part of it.

The final test was under vacuum at one-tenth of an atmosphere pressure and was done by means of the large rotary vacuum pump which is part of the operational equipment required for exhausting the vessel when charging or changing the CO_2 coolant.

The reactor charge and discharge arrangements are accommodated on the pile vault roof along with the pile control and shut-off gear. Emerging from the reactor top dome are the charging tubes which terminate at the upper surface of the pile vault roof. An overhead crane lowers the charge chute into the desired charging tube, and the charge machine supplies fuel elements from a magazine. Conversely the discharge machine removes spent fuel elements by withdrawing them up the charge chute and placing them in a basket where they are kept cool. The basket is lowered into a coffin at the bottom of the discharge well, the coffin lid is automatically positioned and the coffin is withdrawn from the discharge room to be transported to the cooling pile pond.

The control rods and shut-off rods are made of stainless steel tube lined with boron steel fillers and are suspended through the roof of the pile.

The hot CO_2 coolant is led away from the top of the pressure vessel by circular mild steel ducts of 4 ft 6 in. diameter to the four heat exchangers which are situated at the corners of the pile building. The cool CO_2 coolant from the base of each heat exchanger is boosted by four main blowers and returned to the inlet manifold at the base of the pressure vessel. Motorized isolating valves of the wedge gate type are included in each hot and cold duct and flow cascades are

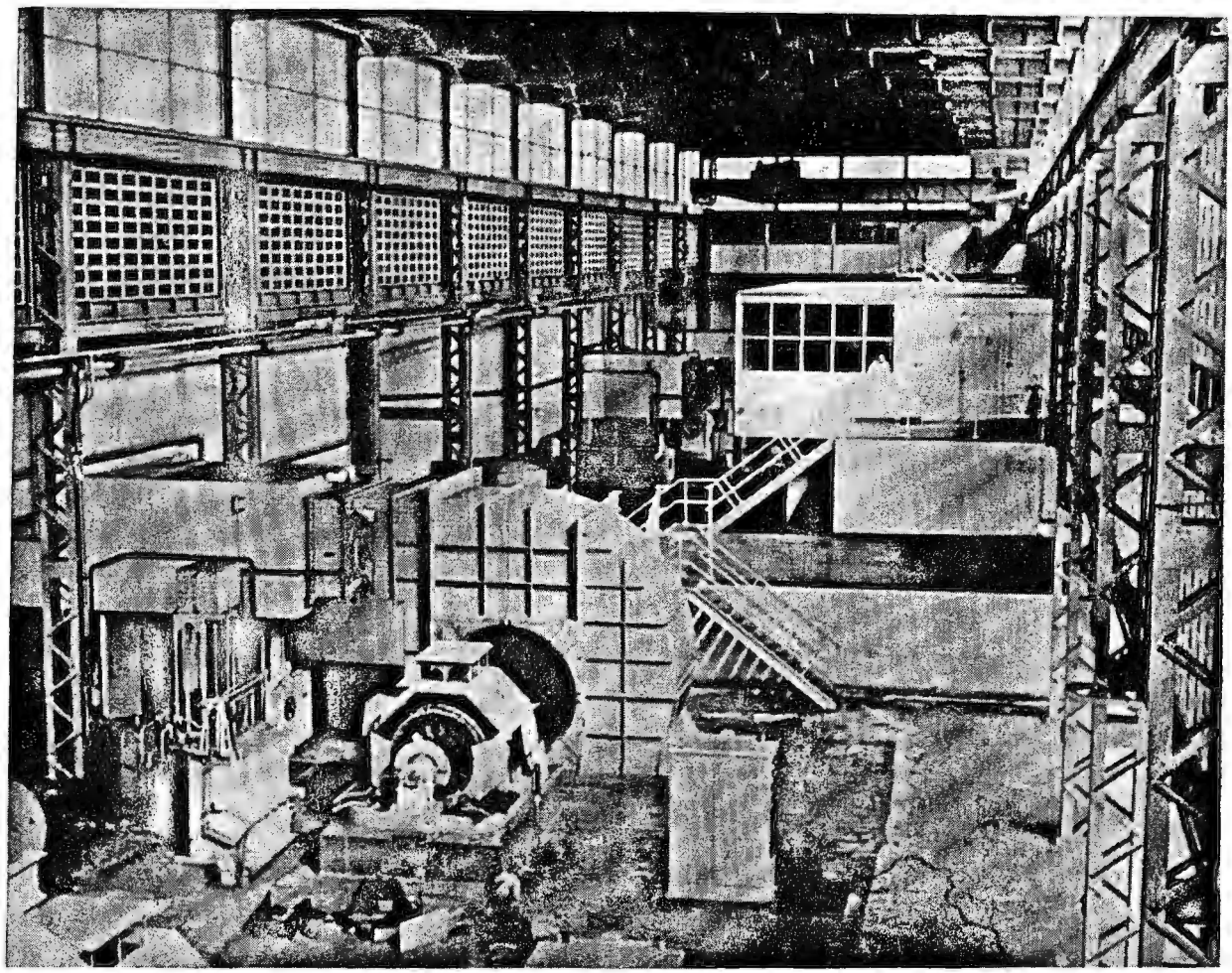


Figure 3. Blower house

fitted at the bends in the ducts. The problem of accommodating the large thermal expansions between the pressure vessel and the heat exchangers under various conditions of operation is solved by the use of "hinged bellows" in the ducts. The duct flexibility is obtained from the bellows, which are of welded steel plate, while the thrust due to the internal pressure of the gas is taken by the pin joint in the bellows assembly. The whole of the duct structures are carried on spring-type constant load supports.

The four CO_2 coolant blowers are of the single stage centrifugal type with an overhung impeller. The overhung impeller arrangement has the advantage of requiring only one gland but it has the additional advantage of facilitating an arrangement in which the gland and bearings can be so located that access to them can be obtained without exposure to radiation which is above tolerance level. The gland is of the face seal type.

It was decided that control of the mass flow of coolant could best be carried out by controlling the speed of rotation of the blowers in preference to varying the static pressure of the coolant system as a whole. Thus the dc motors, which develop some 2000 hp each, are supplied from motor generating sets in the blower house which permit the blower speed to be varied over the range of 10:1 on the Ward Leonard system. About 2% of the mass flow through each blower is taken from the downstream side of the blower, filtered and reintroduced to the system on the upstream side of the blower with a view to preventing the accumulation of graphite dust and iron oxide in the coolant system. A constant check is kept on the water content of the coolant which is kept down by alumina drying units which hold the dew point of the CO_2 below the necessary limit.

As in the Windscale piles continuous monitoring of the coolant gas stream for the detection of burst fuel elements is also carried out.

Over 20 tons of carbon dioxide are required to complete the filling of the reactor and coolant circuits. Charging is carried out by exhausting the system and then admitting CO_2 which is generated from eight standard converters charged with blocks of solid carbon dioxide. Purging by repeated charging and discharging is necessary in the first filling.

Each of the four heat exchangers is contained in a vertical steel pressure shell of $1^{5}/_{16}$ in. steel plate; they are about 18 ft in diameter and some 70 ft high. The 30,000 ft^2 of water heating surface comprising

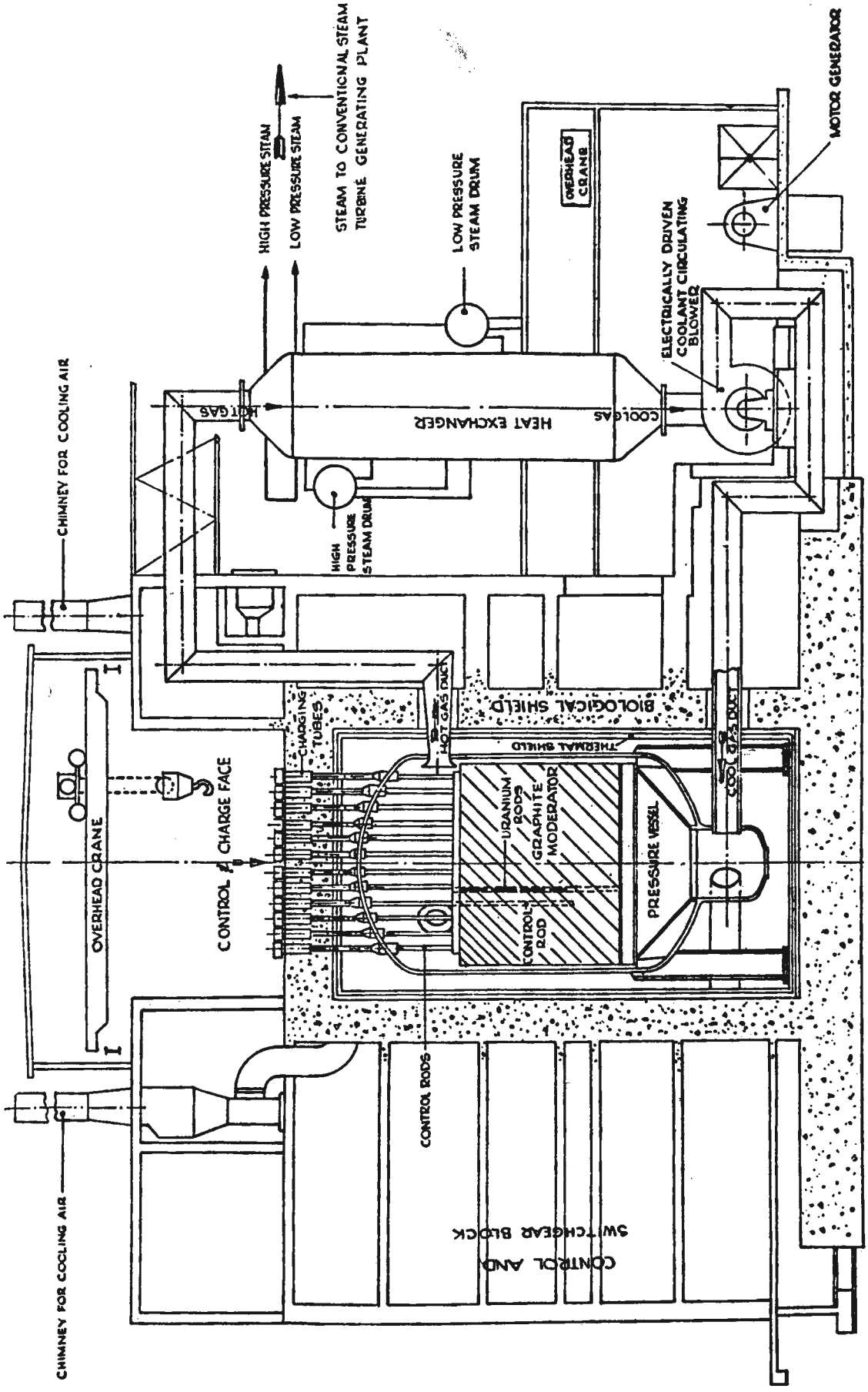


Figure 4. A gas-cooled power reactor

high pressure and low pressure economizer, evaporator and superheater sections is made of 2 in. over-all diameter mild steel tubing through which the water and the steam pass. The gas surface of the tubes is extended by the use of elliptical steel stud tube construction. Flow in the evaporator sections is by forced circulation from electrically driven pumps which maintain a circulation rate of about four times the steam output of the section. Each evaporator has a single steam drum, 4 ft in diameter and 18 ft long which contains conventional cyclones and scrubbers. Particular attention has been given to minimizing the possibility of steam or water leaking into the coolant circuit by keeping the number of tube joints contained within the pressure shell to a minimum. All tube joints sited within the first heat exchanger to be assembled were subjected to an abnormally high pressure test followed by a stringent vacuum test. The ends of each tube element pass separately through a sleeve in the pressure shell so that if leakage should occur at a joint, steam will escape to atmosphere. As in the case of the reactor pressure vessel the final assembly of the heat exchangers was done on site. Each heat exchanger arrived in nine main sections consisting of the dished ends, the six circular centre sections and the skirt. After the heat exchangers were assembled and inspected, the final welds were stress relieved and the heat exchangers were lifted onto their concrete plinths using a pair of 100 ton gin poles.

The dual pressure steam system which is used obviously adds complexity in the design and operation of the station and it was adopted only after careful consideration. The fixed point in the design of a graphite-moderated gas-cooled reactor is the temperature which can safely be achieved in the outgoing gas from the reacting core. This temperature depends on the design and materials of construction of the fuel elements, the possibility of chemical reaction between coolant and moderator, and on the size and design of the pressure vessel which encloses the reactor.

Having decided what coolant outlet temperature can safely be achieved, it is obviously possible to get the maximum output from the reactor for a given expenditure of power in the circulating blowers by making the inlet temperature of the gas entering the reactor as low as possible. This can clearly be done by adding cooling surface in the heat exchangers. But if an attempt is made to reduce the temperature of the coolant below a reasonable level the heat abstracted will be of too low a grade to be used in the thermodynamic cycle for power production; the bottom slice of heat removed from the cooling gas will have to be rejected to drain. Thus thermodynamic efficiency would be reduced by endeavouring to increase reactor rating. The inlet temperature of the coolant to the reacting core is therefore fixed by the minimum temperature acceptable in the thermodynamic cycle.

Reactor rating can therefore be kept up by generating steam at a low pressure. However, it would not be economical to raise all the steam at a low pressure since by doing this the capital cost of the turbo-alternator would be increased and the thermodynamic efficiency would be reduced. These considerations lead to the adoption of a dual pressure steam cycle.

It would be possible to consider a three pressure steam cycle using separately fired superheaters for the high pressure steam, but such additional complications do not at present seem to be justified.

The steam is taken to the generator house, which is situated between the pile buildings, where four turbo-alternator sets are installed. The turbines are two cylinder reaction machines of established design. The low pressure cylinder has twin exhausts which discharge into a surface condenser operating at 1¾ in. Hg absolute pressure i.e. at a temperature of about 38°C. The condensate is passed through a de-aerating plant and the steam cycle is completed by the high pressure and low pressure feed water pumps which return the water to the economizers. Each turbine is directly coupled to a 3000 rpm air-cooled alternator of standard design and having a full load capacity of 23 megawatts at 11 kv. Supplies for electrical ancillaries are tapped directly from the alternators and the remaining power is exported through standard transformers, switchgear and transmission equipment to the British Electricity Authority's grid distribution lines.

A separate "dump" condenser is provided in which up to the full-load steam production of the heat exchangers may be condensed without passing through the turbines. The condensate from the dump condenser is de-aerated and fed back into the heat exchangers. The dump condenser is essentially for use during start-up and shut-down of the plant, but if required it will permit the reactors to be run at full load in the event of turbine outage.

A closed-circuit cooling water system is employed for the condensers. Cold water at about 21°C is fed from the cold pond to the suction basin where four electrically driven pumps send the water to the condenser battery. The hot water at a design temperature of 30°C is fed to the sprays in two natural draught cooling towers from where it returns to the cold side pond. The cooling towers which are somewhat larger than normal, being 290 ft high and 220 ft in diameter at the ring beam, are each capable of cooling 3,000,000 gallons of water per hour by 10°C.

FUTURE DEVELOPMENTS FOR POWER PRODUCTION

It is shown in the White Paper published by the British Government in February 1955 that when a reasonable allowance is made for the value of the plutonium produced as a by-product in graphite-moderated gas-cooled reactors, electrical power can be generated from them at an all-in cost of 0.6d per unit sent out. This estimate of cost is based on the use of techniques which are at present known. With the development of better techniques, it should be possible to reduce the generating cost even further. The graphite-moderated gas-cooled reactor has tremendous advantages in that it is extremely safe in operation, it can employ natural uranium as a fuel and it does not

demand the use of any exotic materials in its construction. It is therefore worthwhile to consider what further developments can be made in this type of reactor which will enable it to produce electrical power at even lower costs.

One of the limiting factors in the design of reactors of the Calder Hall type is the size of pressure vessel which can reasonably be constructed. If the volume of the pressure vessel is increased then the total thermal capacity of the reactor can also be increased and the capital cost per kilowatt sent out is brought down. But to construct a larger pressure vessel to operate at a given working pressure the thickness of the shell must be increased and more complicated problems of supporting the graphite and transmitting the load through the shell walls are introduced. In the case of Calder Hall it was decided that the maximum thickness of plate which could be welded on site was 2 in., and so the size of a cylindrical pressure vessel which may withstand the necessary pressure was fixed immediately. If the working pressure had been reduced the diameter of the core and the output of the reactor could have been increased but the pumping power for circulating the coolant would have gone up. It is hoped that future development will include the welding of thicker gauge plate, possibly up to 3 in., enabling the use of larger pressure vessels to become practical. The thicker plate would alternatively allow working pressures to be lifted for a given pressure vessel size and this would result in a lower blower power being required to remove a given amount of heat. Another line of approach is to use a spherical pressure vessel which, with a given plate thickness, could be made nearly twice the diameter of a cylindrical container. Considered purely as a problem of pressure vessel design the sphere offers very great advantages but it gives rise to difficulties in the support of the graphite structure of the moderator within it. Also the costly biological shield is larger for a given total thermal capacity, but this difficulty could be alleviated by incorporating the biological shield in the pressure shell.

The search for a better reactor coolant is continuous, but as happens so often in engineering a compromise between various conflicting requirements must be accepted. Neutron economy is one of the main problems of reactor design and parasitic neutron losses to the coolant must be small, while due consideration must be given to the characteristics of the coolant as a moderator and under conditions of irradiation. The coolant must be compatible with the circuit components such as the can metal, the pressure vessel and the moderator while (since defects of canning cannot be absolutely eliminated) its chemical affinity for the fuel itself must not be overlooked. One of the most suitable coolants is helium which besides being chemically stable has a high specific heat and offers a negligible capture cross-section to thermal neutrons. The main objections to its use in Great Britain are the high cost and the fact that it is not an indigenous material.

Carbon dioxide on the other hand is cheap and readily available while its neutron capture cross-section is entirely satisfactory. In addition it is fairly stable chemically under Calder Hall pile operating conditions, where it does not react with the graphite moderator to any important extent.

Nitrogen is not attractive as it is a poor coolant, the circulation of which would absorb an unduly high proportion of the energy generated, while its neutron cross-section is high.

Hydrogen on the other hand offers interesting possibilities as a coolant. Firstly, it would act as a moderator resulting in a reduction of graphite and structural costs while a loss of coolant would reduce the pile reactivity, thus giving great stability. Secondly, it would only require about one-tenth of the blower power required for CO_2. It reacts with uranium at certain temperatures however and this might cause serious trouble, because it can pass through the can by diffusion even while the can remains sound. Diffusion of the hydrogen into the structural steel of the reactor may also be a cause of trouble.

The use of a slightly enriched uranium charge may also lead to economies. It may make it possible to use canning materials which have previously been rejected because of their high neutron cross section. The use of such materials may make it possible to use higher fuel element operating temperatures with safety and so to achieve higher thermodynamic efficiencies. A slightly enriched uranium charge towards the outside of the reactor core enables flattening of the neutron flux distribution across the core to be carried out, which in turn increases the total thermal capacity of the pile and hence the electrical output of the station.

In the pure engineering field, there can be no doubt that the design of heat exchangers can be considerably improved and cheapened. The problem of raising steam by transferring heat to the water from a perfectly pure and clean gas has never before had to be considered on a large scale. The stud tube extended surface heat exchanger which is used at Calder Hall had the tremendous advantage that this type of heat exchanger surface was already established industrially and was in mass production. It is not an ideal type of extended surface for the absorption of heat from a completely pure gas. Each one of the Calder Hall heat exchangers weighed some 200 tons and the problem of transporting the sections, assembling them on the site and lifting the complete heat interchanger into position caused many anxious problems. There is no doubt whatever that a cheaper, lighter and more compact heat exchanger could be developed to deal with the problem.

THE BRITISH WHITE PAPER PLAN

The first reactors which are at present being built may have shortcomings but they have the advantage that they are inherently safe and that they can generate electricity in large quantities to a guaranteed programme and at prices comparable with a modern conventional power station. They cannot stand on

their own and are competitive only when there is a market for the by-product plutonium; this market must and can be found by recycling or by use in advanced types of reactors, culminating in the Fast Reactor.

A careful and cautious estimate has been made of the value which can reasonably be accredited for the by-product and, as has been stated already, the cost of electricity from the initial stations has been calculated to be not more than 0.6d per unit sent out; reduction in capital and fuel costs should progressively improve the figure.

As already pointed out it is on the construction of reactors of the graphite-moderated gas-cooled type that the first stages of the British Programme for the development of industrial power are based.

The programme starts with the construction of two gas-colled graphite-moderated stations, each with two reactors, on which construction will begin about mid-1957 and they should come into operation in 1960–1961. The construction of two further stations would begin about 18 months later in which the reactors would be of a similar type to those in the earlier stations but they should show an improved performance particularly in heat rating. The total electrical output from these four stations, which should be in operation by 1963, should be between 400 and 800 megawatts. By 1960 four more stations should start construction followed by four more 18 months later, and when they are all in operation by about 1965 the eight stations in this group should supply well over 1000 megawatts. The first four of this group may well be developments of the gas-cooled graphite-moderated type while the last four may be of the liquid-cooled type which may then have been developed sufficiently to be economically satisfactory. The new stations will have a higher capital cost but lower running costs than the conventional station and they will be run as base load stations at a high load factor of possibly 80%.

It is reasonable to think of nuclear reactors in an ascending scale of specific heat rating of their fuel elements. At the bottom of this scale is the graphite-moderated gas-cooled reactor of which we have been talking. It is essentially a heavy, land-based reactor, using low specific ratings but economical because it does not need to make use of expensive and exotic materials of construction. It can, perhaps, be regarded as the slow-speed reciprocating engine of the reactor world, reliable and almost conventional in design.

At the other end of our reactor scale, we have the fast fission reactor, very much more highly rated and using expensive and rare materials of construction. High ratings are necessary in order to make this reactor economical because the fissionable material which it uses as a fuel is extremely expensive. It can perhaps be regarded as the gas turbine of the nuclear reactor world.

In between these two extremes are dozens of alternative types of reactor making use of fast neutron, intermediate neutron, or thermal neutron velocities; using no moderator or light water, heavy water, beryllium or graphite as moderators; using gases, water, hydrocarbons or liquid metals as coolants in any combination that can be imagined.

Which of these reactors will at future dates be found to be economical is more than can be safely predicted today. We in Great Britain are attacking the two ends of the scale; we have designed the graphite-moderated gas-cooled reactor which will soon be producing large quantities of industrial power at Calder Hall, and we are building the fast reactor at Dounreay. In between these two extremes we are applying our resources at those intermediate points which appear from time to time to offer the greatest promise.

The development of highly rated reactors is essential because such reactors are needed to burn the by-product plutonium produced in the reactors of conservative rating which are built as the first step in our industrial programme. It would, however, be a great mistake to assume that these conservatively designed reactors of the graphite-moderated gas-cooled type have only a limited future. With development, the reciprocating steam engine held its own over a period of nearly 200 years as an economical prime mover and the simplicity and reliability of the gas-cooled reactor will entitle it to a similar place in the history of nuclear power.

The Engineering Design of EBR-II, A Prototype Fast Neutron Reactor Power Plant

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The Argonne Experimental Breeder Reactor II (EBR-II) is one of the five prototype industrial power reactors selected by the United States Atomic Energy Commission for development and construction. The EBR-II is a plutonium-fueled, unmoderated, sodium-cooled reactor with a power rating of 60,000 kw (heat). The plant consists of the reactor and heat removal system, the steam-electric power plant, and an integral fuel reprocessing facility. The plant is now in the engineering design and development stage, with construction scheduled to begin in 1956, and operation to begin in 1958.

The primary purpose of the EBR-II is to demonstrate the engineering feasibility of a fast power reactor system which will be economically competitive with existing power sources (in the United States). The EBR-II itself need not (and is not expected to) produce electricity at a competitive price. It should, however, demonstrate the engineering feasibility of a reactor system which, when scaled up to central station size, will be competitive.

An economic analysis (described later) of the cost of power produced by a large central station fast power reactor indicates certain plant characteristics which are necessary or desirable. The most important considerations affecting the engineering design are: reactor performance, plant cost, and fuel cycle costs (processing and fabrication). The EBR-II design reflects the influence of these considerations.

Although the power capability of EBR-II is not of a magnitude comparable to central station power plants, a determined effort is being made to develop and employ components which will require little modification for use in such installations.

GENERAL PLANT DESIGN

The EBR-II Plant is shown in Fig. 1. It is divided into four major systems which may be defined as follows:

1. The Primary System: the reactor and the primary sodium cooling system.
2. The Secondary System: the intermediate sodium heat transfer system.
3. The Steam System: the steam-electric system.
4. The Fuel Process System: the fuel recovery and fabrication facilities.

The primary system is contained in a single vessel (referred to as the "primary tank"). All of the components in the primary system, including the reactor, the primary sodium pumps and piping, the heat exchanger, and the fuel transfer and storage system, are submerged in sodium, as shown in Fig. 2.

The reactor consists of an enriched core of uranium-plutonium alloy, in the approximate shape of a hollow cylinder, surrounded on all sides by a uranium breeding blanket. The average core power density is approximately 1000 kw/liter, and the average core heat flux is approximately 1×10^6 Btu/ft^2-hr. Reactor cooling is a critical problem, not only during operation, but also after shutdown. Considerable attention, therefore, is given to the primary cooling system.

The fuel and blanket material are contained in subassemblies of hexagonal cross section positioned vertically in a close packed array. The coolant enters the bottom of each subassembly, flows upward through the subassembly, and then passes from the reactor through the intermediate heat exchanger.

The reactor loading and unloading process is carried out with the subassemblies completely submerged in sodium. Shutdown cooling requirements of the fuel dictate this unloading procedure. The subassemblies are transferred to the submerged storage racks, where the fuel continues to cool by natural convection of the sodium.

The heat is transferred from the heat exchanger to the steam generator by the secondary sodium system. This system is non-radioactive and serves to isolate the radioactive sodium in the primary system from the steam generator, and also to isolate the reactor from the moderating effect of water. Conventional piping and pumping arrangements are employed in this system, which is accessible for normal maintenance.

The steam generator is a sodium-to-water/steam heat exchanger. Steam at 850°F and 1250 psi is supplied to a conventional turbine-generator of 20,000 kw capability.

The plant includes an integral fuel and fabrication facility to determine the feasibility and cost of a fuel cycle specifically designed to meet the needs of a fast power reactor. Low reprocessing costs and a short cooling time, to minimize fuel inventory, are desired. A direct metallurgical separations process has been selected for EBR-II fuel recovery

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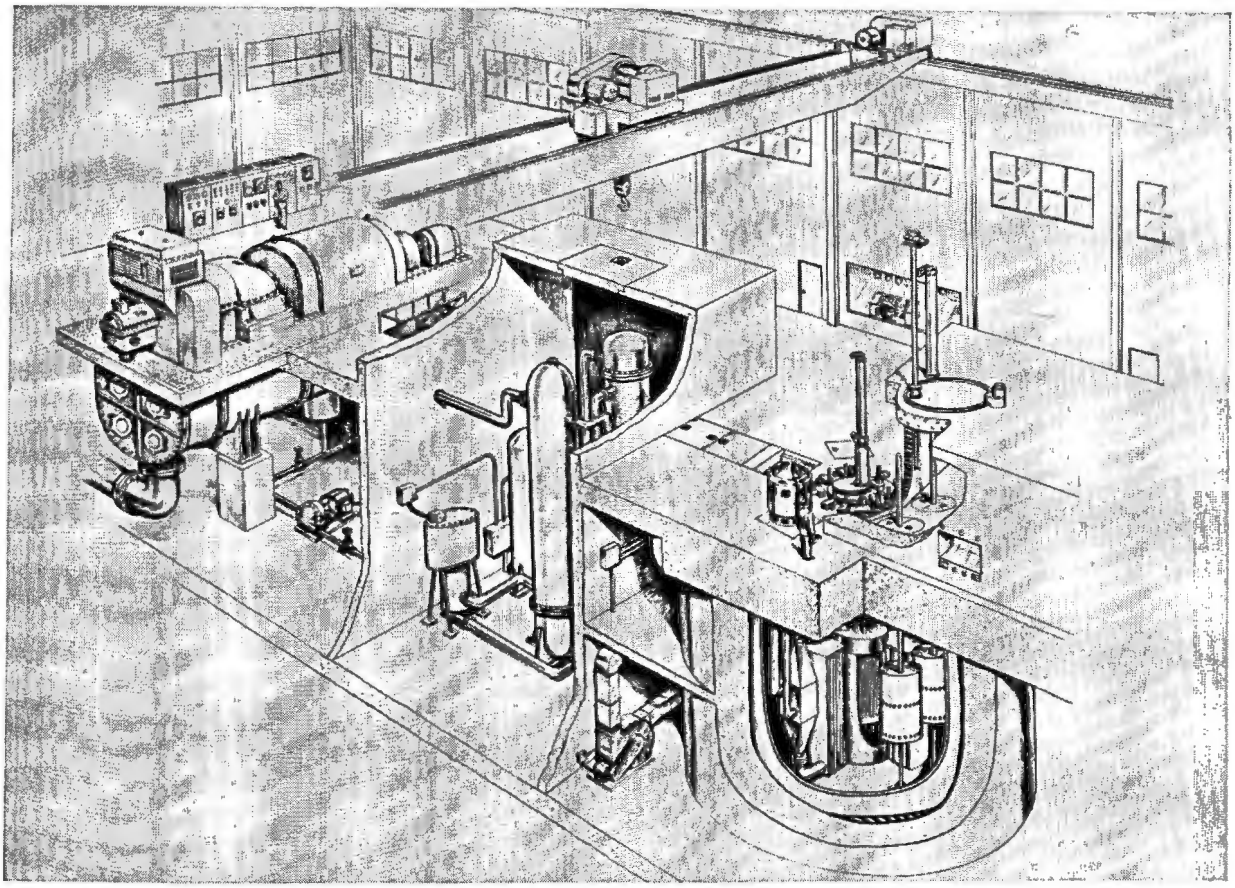


Figure 1. The EBR-II plant

which shows promise of accomplishment of both of these objectives.

The Reactor Assembly

The reactor assembly consists of the reactor proper (approximately cylindrical in shape), the containing tank, and the surrounding neutron shield. Approximately over-all diameter and height are 10 and 13 feet, respectively. A schematic plan view through the center of the assembly is shown in Fig. 3, and a schematic elevation of the assembly is shown in Fig. 4.

Reactor

The reactor is divided into four main zones, viz., Core, Central Blanket, Inner Blanket, and Outer Blanket, as indicated in Fig. 3. Each zone is comprised of a number of right hexagonal subassemblies containing the fuel or blanket elements. All subassemblies are of identical size. Their numerical distribution is as follows:

Central blanket	=	7
Core	=	42
Control	=	12
Inner blanket	=	66
Outer blanket	=	510
Total		637

Construction of the subassemblies and elements is described later.

An annular core with a central blanket of uranium has been incorporated for the following purposes: to flatten radial distribution of neutron flux and power generation within the core; to provide for experimental enlargement of the core, if desired, by substitution of core subassemblies for central blanket subassemblies; and to accommodate high neutron flux irradiation facilities for experimental purposes. There are thought to exist additional, less direct advantages of a central blanket, as well.

Division of the annular blanket surrounding the core into two separate zones, the inner blanket and the outer blanket, is necessitated by the wide variation in power generation across this region and the desire to achieve everywhere the highest possible fraction of unit volume devoted to blanket material (uranium). In the blanket area immediately adjacent to the core, the unit volume fraction of uranium permissible is relatively low, since the power density is high and cooling is difficult. In the area near the outer periphery, however, the uranium fraction may be high, because the power density is low and cooling is more easily effected. The division of the blanket into two zones of differing unit volume fraction of uranium represents a practical compromise between an infinite number of such zones and a single zone of constant uranium fraction.

Because power densities within the central and

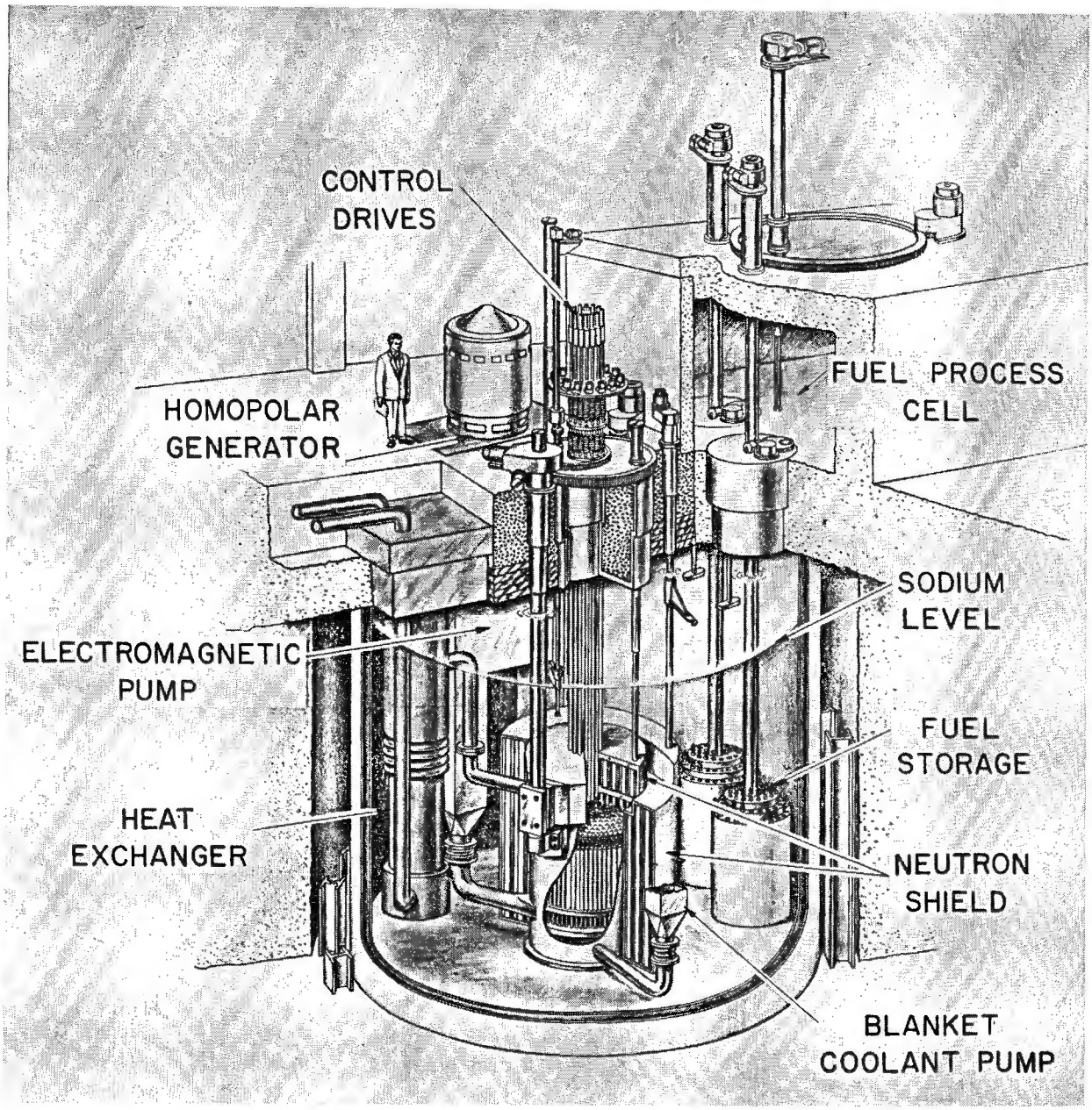


Figure 2. EBR-II primary system

inner blankets are similar in magnitude, identical subassemblies (both in composition and construction) are used in these zones.

A single subassembly size is employed throughout the entire reactor because: a maximum of flexibility in reactor configuration is obtained; the amount of parasitic reactor volume is reduced; and subassembly handling equipment and reactor tank structure are simplified.

Approximate composition (by volume per cent) of each type of subassembly is shown in Table I. The fuel alloy employed is composed of an alloy of uranium and plutonium. The blanket material is stabilized uranium.

Each subassembly consists of a hexagonally shaped container in which are supported a number of fuel elements (or blanket elements) of size and shape appropriate to the particular type of subassembly. An adaptor is provided at the upper end for accommodation of the gripper mechanism used in loading and unloading from the reactor. An extension at the lower end, through which the coolant enters,

Table I. Subassembly Composition, Volume Per Cent

Type of subassembly	*Portion of subassembly*	*Fuel alloy*	*Uranium*	*Stainless steel*	*Sodium*
	Upper blanket	0	30	15	55
Core	Core section	30	0	15	55
	Lower blanket	0	30	15	55
Central blanket	Entire	0	55	15	30
Inner blanket	Entire	0	55	15	30
Outer blanket	Entire	0	80	10	10
Control	Core section	20	0	20	60

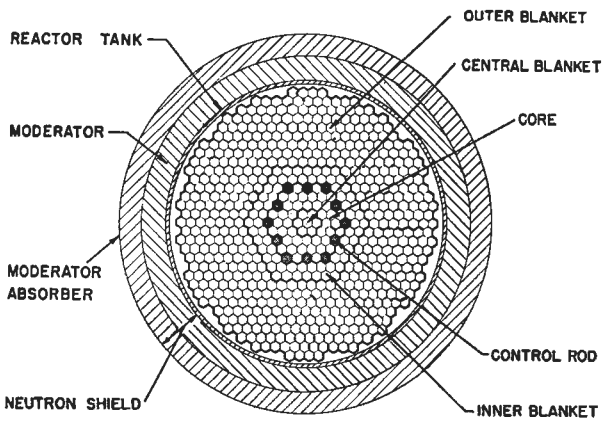


Figure 3. Schematic plan view of EBR-II reactor assembly

serves to hold the subassembly in an upright position when inserted into the support grid of the reactor tank. A brief description of element arrangement in each type of subassembly is given below.

The core subassembly (Fig. 5) is comprised of three sections: upper blanket, core, and lower blanket. The core section consists of ninety-one cylindrical fuel elements spaced on a triangular lattice by a single, helical rib on the outside of each element. The elements are supported within the subassembly by fastening their ends to support grids which are attached to the subassembly hexagonal tube. The fuel elements are "pin type," consisting of a right circular cylinder (pin) of fuel alloy fitted into a thin-walled, stainless steel tube. The coolant flows along the outside of the element tube.

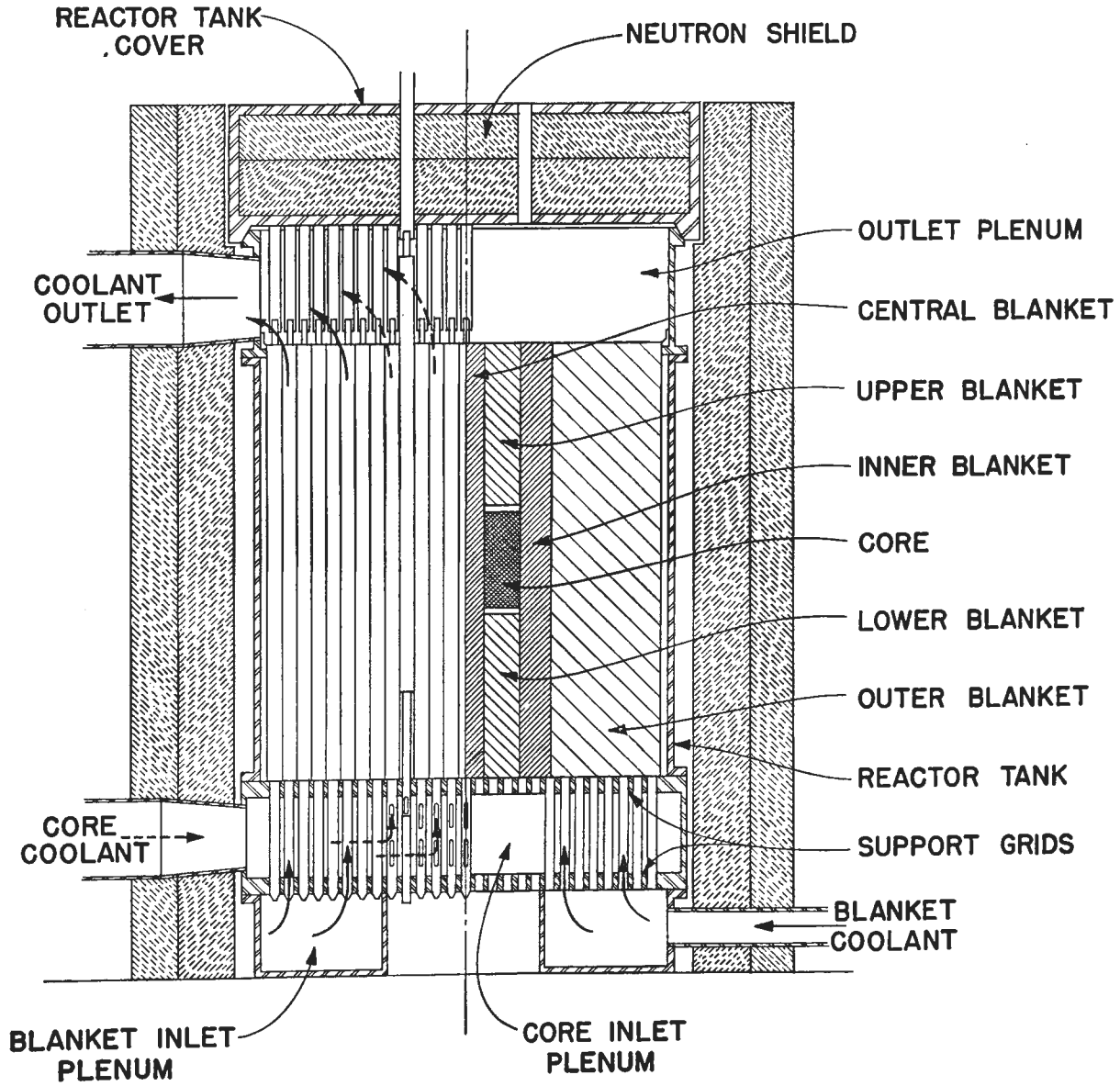
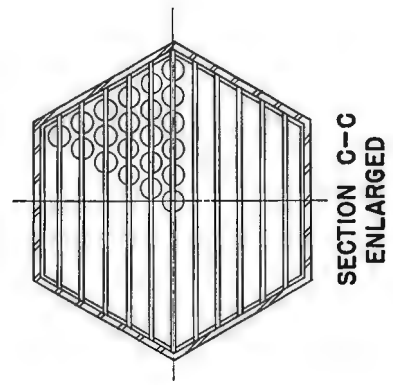
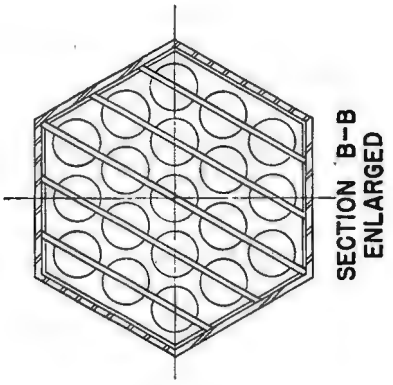
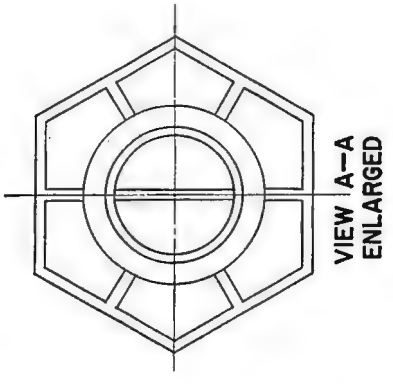


Figure 4. Schematic elevation of EBR-II reactor assembly







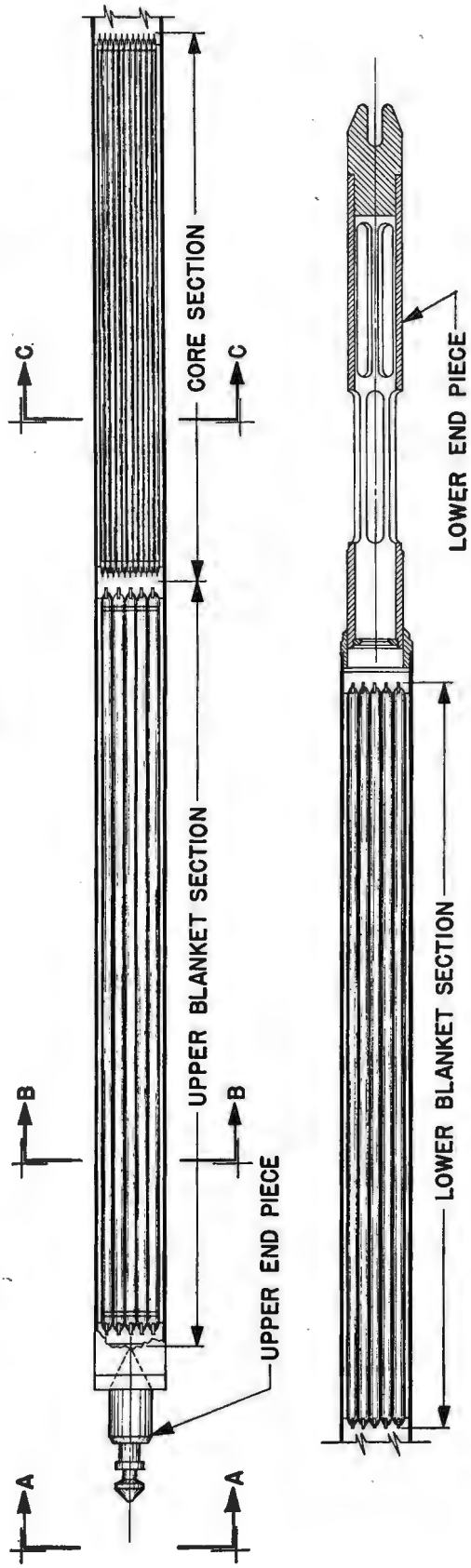


Figure 5. Core subassembly

Figure 6. Photograph of core subassembly

The upper and lower blanket sections are identical in construction and consist of nineteen pin type elements spaced and supported in a manner similar to that employed for the core section. The pin diameters and pitch, of course, are larger, and uranium is used in place of fuel alloy.

Figure 6 is a photograph of a core subassembly.

The central blanket (and inner blanket) subassembly (Fig. 7) is comprised of three identical sections. Each section consists of nineteen pin type blanket elements spaced and supported by the same means as employed within a blanket section of a core subassembly.

The outer blanket subassembly (Fig. 8) is a "radiator type" arrangement consisting of six right triangular prisms nested together. Seven tubes, through which the coolant flows, pass through the uranium and are attached to headers at each end.

The control subassembly (Fig. 9) consists of a modified, movable core subassembly contained in a guide thimble. The guide thimble is hexagonal in cross section and of external dimensions the same as for the other types of subassemblies. The control rod has four sections: upper blanket, void (sodium), core, and lower blanket. The upper and lower blanket elements are identical to those of a core subassembly except that the length of the lower blanket is decreased somewhat and the element spacing (pitch) in both sections is reduced. The core section is identical with that of a core subassembly except that the outermost row of fuel elements is omitted, reducing the number of elements to sixty-one. Control is effected by movement of the core section into and out of the core (with simultaneous substitution of void section for core section).

Reactor Tank

As indicated in Fig. 4, the reactor is contained in a tank which positions and supports the subassemblies, and directs the flow of coolant sodium. A movable cover at the top of the tank provides access to the reactor. Twelve vertical holes are provided in this cover through which the control rod drives extend to actuate the rods during normal reactor operation.

Each subassembly is located and supported at the bottom by a combination support grid and inlet coolant plenum. Each subassembly is positioned accurately at the top by a tubular guide which projects from the bottom of the cover and engages the subassembly top end piece when the cover is lowered into its normal operating position.

The coolant flow direction is upward through the reactor. Distribution of flow is accomplished by orificing of each subassembly, identical orificing being used in all subassemblies in a given row (except in the special case of control subassemblies). In order to accommodate the very large range of flow rates required, two parallel flow systems, each with its own pump, are used. One system supplies the central blanket, core, control subassemblies, and the first subassembly row of the inner blanket. The other supplies the second subassembly row of the inner blanket and all of the outer blanket. The two systems have separate inlet plenum chambers and a common outlet plenum chamber. The pressure drop between inlet and outlet plenums of the system supplying the core (referred to as the high pressure system) is less than 75 psi, and for the other system (the low pressure system) is less than 10 psi. Because the first subassembly row of the inner blanket is included in the high pressure system, the arrangement permits experimental enlarging or reshaping of the core by substitution of core type subassemblies within this row. The same may be done with the central blanket. Coolant flow paths and plenum arrangements are shown in Fig. 4. Approximate total flow rate and average coolant temperature rise are 11,000 gal/min and 130°F, respectively.

Neutron Shield

Neutron shielding is provided around the reactor tank as shown in Fig. 4. The primary purpose of this shielding is to protect components exterior to the reactor vessel from excessive activation and so to permit their repair and maintenance (after removal from the primary tank) under conditions of reasonable activity levels. Gamma shielding and additional neutron shielding are provided on the exterior of the primary tank in order to meet biological requirements. Both the radial shield and the top shield contain a combination of moderator and neutron absorber. The shield material is contained in stainless steel cans which are nested so as to permit passage of sodium between them for cooling purposes. In the radial shield, sodium from the primary system tank, flowing by natural convection, effects the cooling. In the top shield, sodium from the reactor outlet plenum is diverted to flow through the shield.

PRIMARY COOLANT SYSTEM

The primary coolant system includes the reactor, the intermediate heat exchanger and the circulating pumps. A significant feature of this design is the close-coupled arrangement of the components of the primary system in the primary tank in which they operate submerged in liquid sodium. This submerged primary system arrangement has been adopted for the following reasons:

1. The large bulk volume of sodium contributes significantly to the reliability and integrity of the primary cooling system. Since the entire system is flooded with coolant (to a level approximately 10 feet above the top of the reactor), loss of reactor coolant is virtually impossible. In addition, the system is ideally suited to natural convection cooling, providing very reliable shutdown cooling in the event of loss of forced convection.

2. The large bulk volume of sodium also provides thermal inertia to the primary system, preventing rapid changes in load demand from being reflected

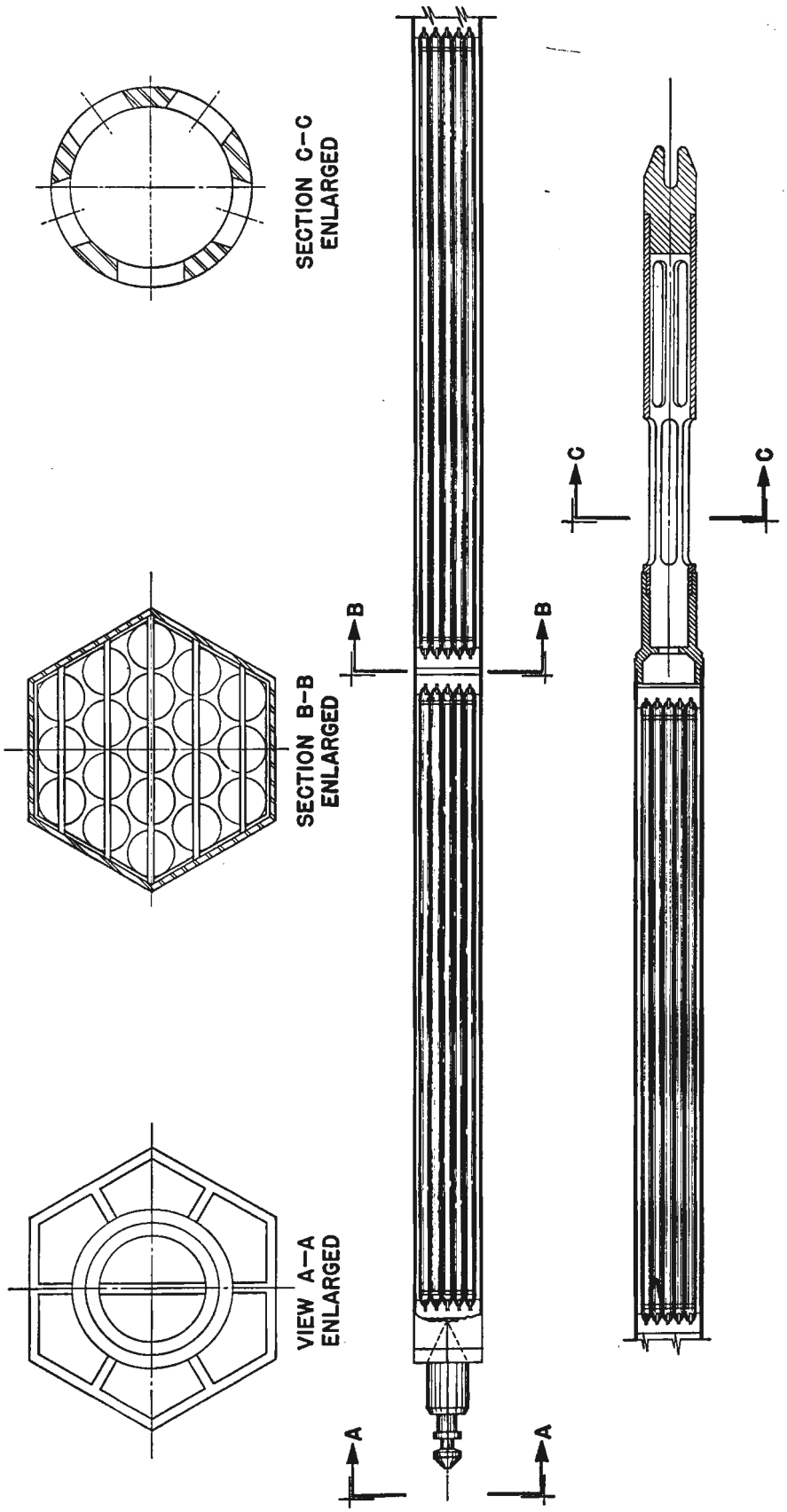


Figure 7. Central and inner blanket subassembly

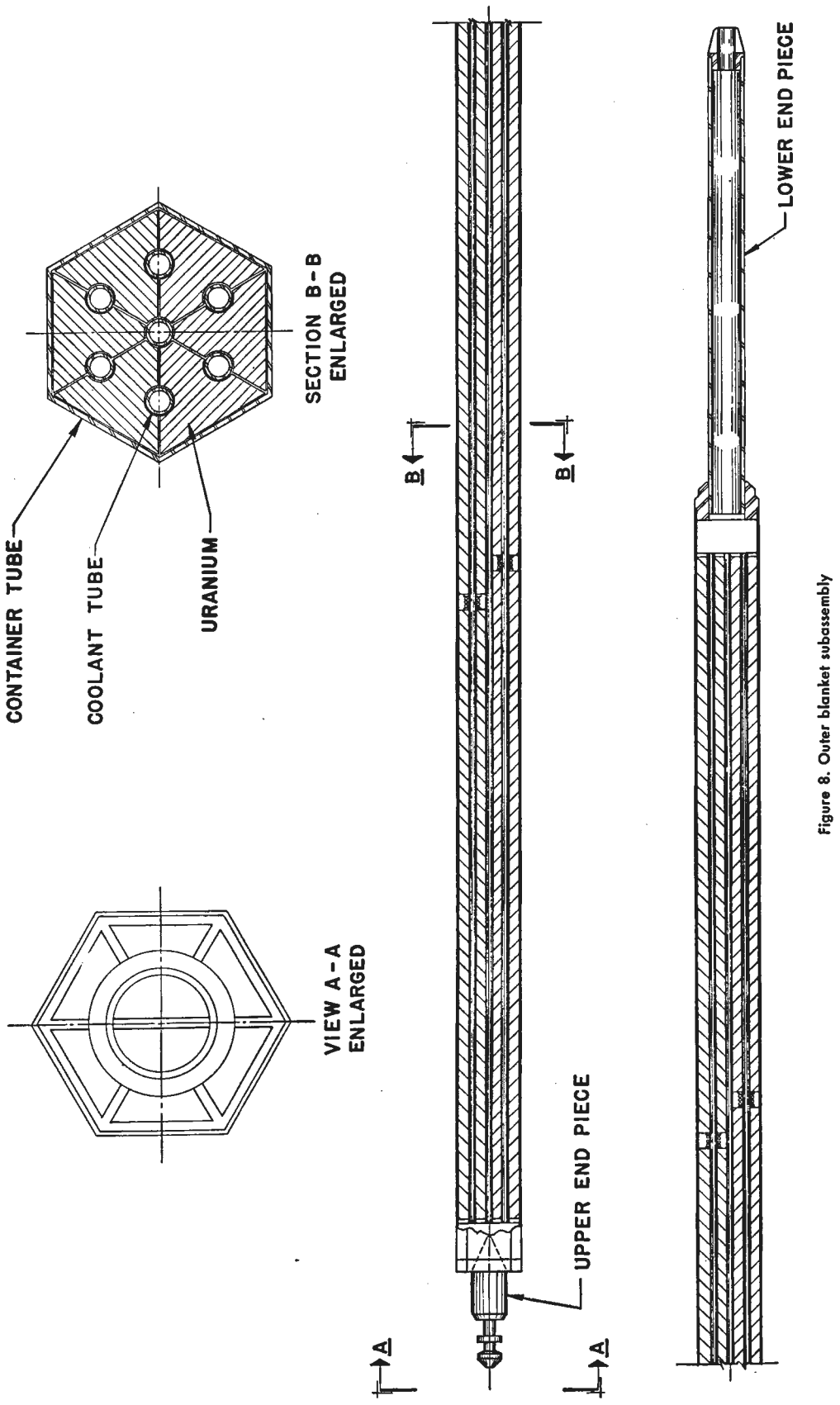


Figure 8. Outer blanket subassembly

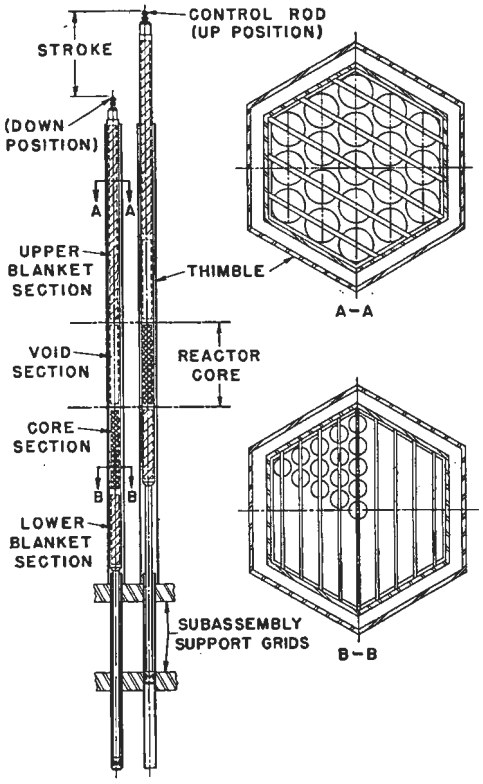


Figure 9. Control subassembly

as temperature changes in the coolant entering the reactor. The large heat capacity of the system also provides intrinsic emergency cooling, in conjunction with natural convection, in the event of failure of the heat removal system (secondary sodium system) simultaneous with the loss of forced convection. Such circumstances might arise in the event of a total power failure, in which case the bulk sodium heats very slowly and considerable time is available to initiate stand-by cooling procedures.

3. Since the reactor is to demonstrate the method of operation to be employed in a central station power plant, the replacement of fuel must be accomplished in a short time. Shortly after reactor shutdown, the heat generation in the fuel element is high, and reliable cooling must be provided. This is accomplished by unloading and transferring the fuel elements while they are submerged in sodium. They are cooled by natural convection of the sodium, and unloading can begin immediately after shutdown. The fuel elements are transferred to a fuel storage chamber within the primary tank where they continue to cool, by natural convection of the sodium, until removed for processing.

4. A maximum of integrity is provided with regard to containment of radioactive sodium. The entire radioactive system is confined within the primary tank. A very high degree of integrity can be constructed into this vessel, since it is of relatively simple design and contains no external connections below the liquid sodium level. As shown in Fig. 2, it is of double wall construction as further insurance against failure.

5. The need for high integrity of the primary sodium piping is eliminated. Small amounts of leakage are permissible, since the leakage is internal. A small amount of leakage actually does occur in the piping system at the connections between the pumps and reactor and between the reactor and heat exchanger. These connections are slip joints employed to permit the pumps and heat exchanger to be removed from the system without the necessity of cutting or disconnecting a tight piping system.

6. All of the radioactivity in the plant is confined in the primary tank, and therefore only the primary tank requires shielding. Shielded equipment cells and pipe galleries are eliminated.

7. Auxiliary heating of the primary system sodium (to prevent freezing) is simplified, since the entire system is heated as a unit. The individual components and pipes, etc., are in an "atmosphere" of sodium and the entire system is at the same temperature.

Coolant is pumped directly from the bulk sodium in the primary tank to the two reactor inlet plenum chambers. After flowing upward through the reactor into the common top plenum chamber, it passes through the intermediate heat exchanger and then returns to the primary tank. The flow through the reactor is monitored by electromagnetic flow meters provided with insulating jackets to permit submerged operation.

This design requires that the pump be located within the primary tank and that it be capable of operating submerged in sodium without any form of auxiliary cooling other than that provided by the passage of the liquid being pumped. A d-c conduction type electromagnetic pump is employed, with sufficient capacity to handle the pumping requirement of the main coolant system. The pump, shown in Fig. 10, has a capacity of 10,000 gallons per minute at a head of 75 psi. The pump duct is 6 inches by 18 inches in cross section and carries the liquid at a velocity of 31 ft per second. The current of approximately 250,000 amperes required to drive the pump is supplied by a homopolar generator located near the top of the primary tank shield. The very large current requirement makes it necessary to place the pump and generator as closely together as possible. Consequently, the pump is located directly below the generator near the level of the sodium surface in the primary tank. The current is carried from the generator downward through the shield in sodium-filled ducts which attach to the pump housing. The ducts are partly filled with copper bars to provide adequate gamma shielding. The pump, generator, and drive motor are connected so that they can be individually removed should servicing become necessary. In order to avoid the entrance of liquid containing entrained gas, the pump intake is located

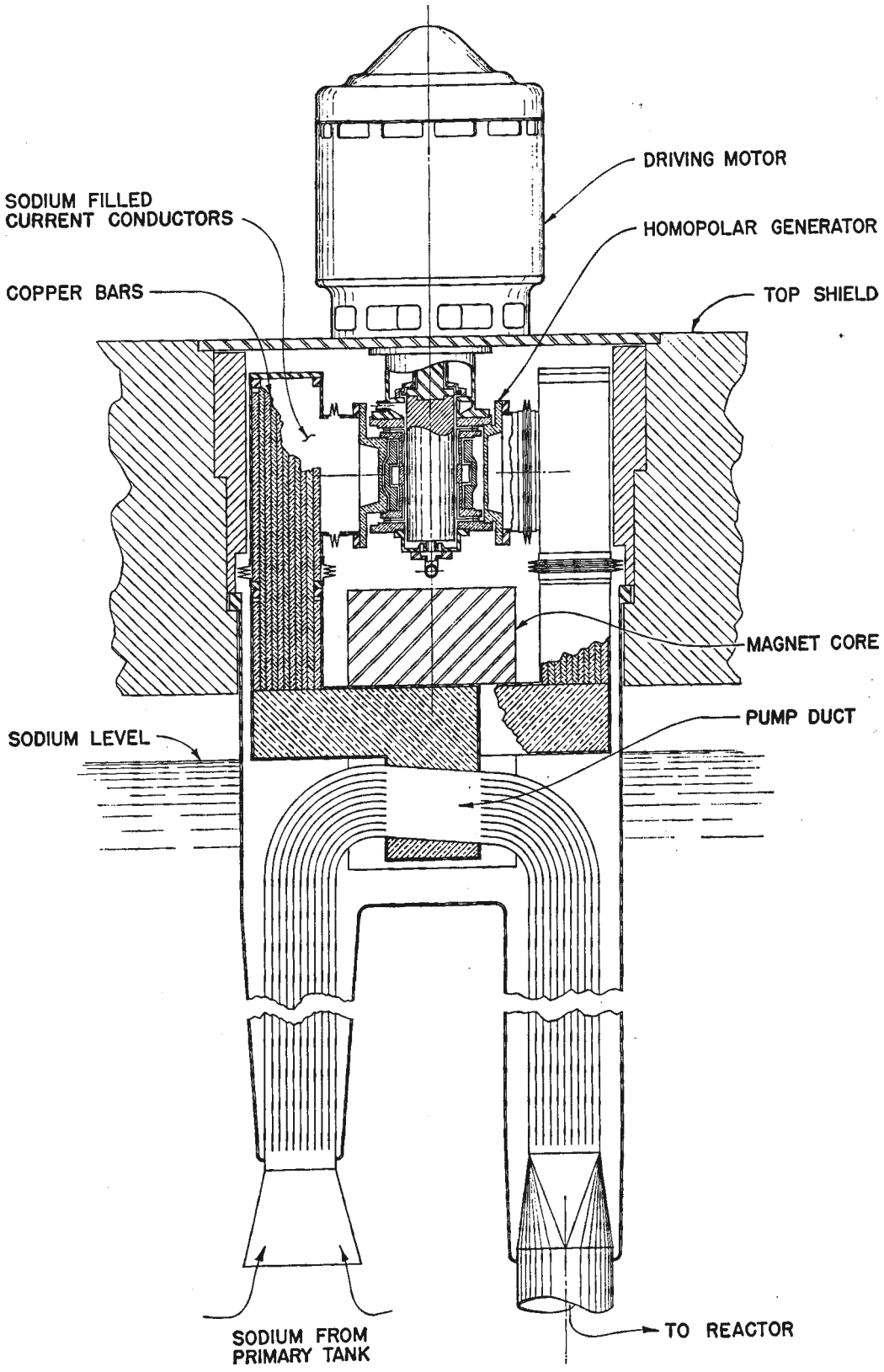


Figure 10. Primary system electromagnetic pump and generator

well below the surface of the sodium in the primary tank.

The heat exchanger receives the high temperature sodium from the reactor and discharges it directly into the main primary tank. The exchanger is a counterflow, shell and tube, single pass, sodium-to-sodium unit. The primary flow rate is 5.2×10^6 pounds per hour of sodium at an inlet temperature to the heat exchanger of 900°F and an outlet temperature of 770°F with a pressure drop of 5 psi. The secondary flow rate is 2.2×10^6 pounds per hour of sodium at an inlet temperature of 580°F and an outlet temperature of 880°F, also with a pressure drop of 5 psi.

Dump tanks are provided so that the sodium may be removed from the primary tank. Transfer is accomplished by pumps, rather than by gravity, to minimize the possibility of accidental loss of primary coolant.

Provisions are made for continuous oxide clean-up of the sodium system by a by-pass circuit through a cold trap filter. Oxide concentration is determined by plugging indicators and analysis of samples removed from the system.

FUEL HANDLING AND CONTROL DRIVE MECHANISMS

The fuel unloading and reloading procedure involves the removal and transfer of fuel subassemblies shortly after reactor shutdown. To provide reliable cooling of the fuel elements during this period, the entire operation is carried out with the fuel subassemblies submerged in sodium. Since an intermediate transfer operation is required, the adapter at the top end of each subassembly provides two shoulders (or collars) which can be grasped simultaneously by two mechanisms to effect the transfer.

The subassemblies are positioned in the reactor on an equilateral triangular pitch. The gripper mechanism, employed to remove a subassembly from the core, is positioned by a pair of rotating plugs located in the top shield. The gripper mechanism is located in the smaller plug, which, in turn, is eccentrically located in the larger plug. The center line of the large plug corresponds with the center line of the reactor. The gripper is positioned over any subassembly in the reactor by rotation of the two plugs. It is also necessary for the gripper mechanism to rotate about its own centerline. The subassemblies are supported only at the bottom during the unloading operation. To eliminate the possibility of disturbing adjacent subassemblies, a funnel-shaped "hold-down mechanism" which surrounds the fuel subassembly to be removed is employed and holds the surrounding units in their proper position. The hold-down mechanism is also mounted in the small plug, and is always positioned directly under the head of the gripper mechanism.

The equipment employed in fuel unloading and transfer is shown in Fig. 11. After disconnecting and raising the control drives, the reactor cover is opened, making the reactor accessible to the fuel handling equipment. The two plugs are rotated to position the gripper over the subassembly to be removed, and the gripper mechanism is rotated to the proper angular position to index with the selected subassembly. The hold-down mechanism is then lowered, and the gripper removes the subassembly and transports it to the transfer point by rotation of the plugs. The subassembly is then transferred to the storage rack.

Subassemblies are transferred from the storage rack to the reactor in the reverse manner of that described above.

After sufficient cooling, the subassembly is removed from the storage rack by a unit similar to the gripper mechanism located in the process cell. New subassemblies are delivered to the storage rack from the process cell in a reverse manner.

The control drive units are electro-mechanical devices imparting vertical linear movement to the control rods. The motion requirements are simply a slow up or down motion, and a fast down motion. A rack and pinion drive from a gear head motor imparts the slow motion. Rapid acceleration (in the downward direction) is accomplished by springs, and a hydraulic cylinder serves as a decelerating cushion. The spring unit is actuated by a latching mechanism which is carried in a "cocked position."

Figures 12 and 13 are photographs of an experimental unit incorporating fuel handling and control drive mechanisms. They are shown as set up prior to installation in sodium. The units are scaled down in size from those required in EBR-II.

SECONDARY SYSTEM AND STEAM PLANT

The heat generated in the reactor is transferred to the secondary sodium system in the intermediate heat exchanger. Next, the heat is transferred by the secondary system to the steam generator. Finally, the steam system transfers the heat to a conventional, condensing turbine-generator. A skeleton flow chart of the three heat transfer systems indicating principal components, temperatures, pressures, and flow quantities is shown in Fig. 14.

Sodium is employed in preference to sodium-potassium alloy as the working fluid of the secondary system, because of its superior heat transfer characteristics, and to eliminate the possibility of contaminating the primary system sodium with potassium in event of leakage within the heat exchanger. An alternating current electromagnetic pump is employed, with a capacity of approximately 5000 gallons per minute at a head of 25 psi. An expansion tank, with an inert gas blanket to maintain the desired system pressure, is provided at the inlet to the pump. Piping is 12-inch nominal size, of stainless steel type 304. External heating provisions are installed on the piping to assure maintenance of the sodium in the molten state under all conditions.

In the steam system, water enters the steam gen-

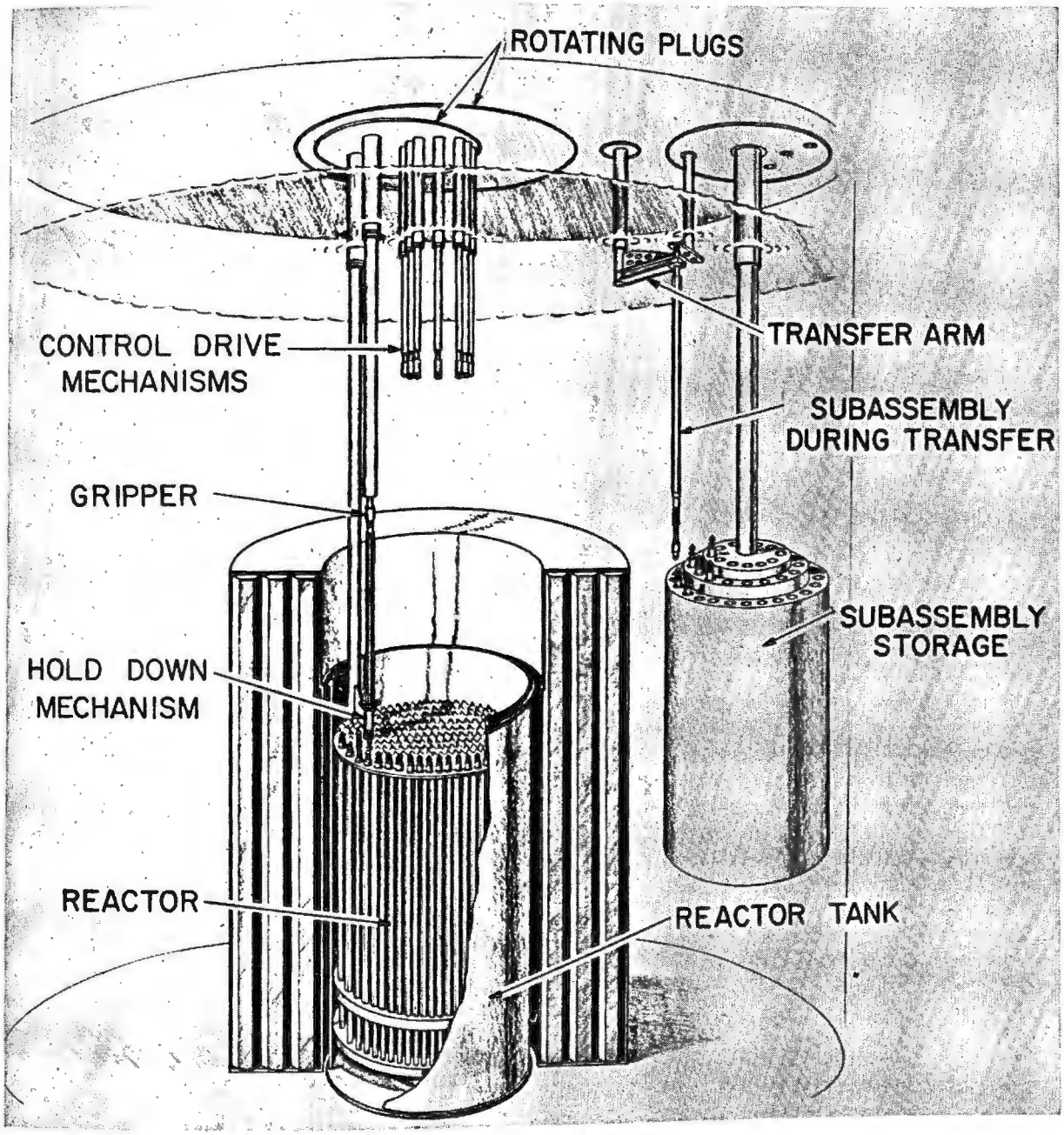


Figure 11. EBR-II fuel handling system

erator at 450°F and leaves as superheated steam at 850°F. The steam generator is "once-through," with integral preheating, evaporating, and superheating sections. Use of the relatively inexpensive once-through type is made especially attractive by the freedom from the possibility of burnout inherent in this application. Shell and tube construction is used, with the sodium on the shell side. The tubes are double-walled, with provision made for detection of leakage into the annular space between the tubes during operation. Steam conditions at the turbine throttle are 1250 psi and 850°F, with a maximum flow rate of approximately 200,000 lb/hr. The turbine employed is an extracting, condensing, single flow type unit of 20,000 kw maximum capability. Five steam extraction points are employed to supply steam to five stages of feedwater heating. Main steam piping is 8-inch nominal size and feedwater piping is 6-inch nominal size.

The secondary system sodium flow rate and the steam throttle pressure values employed were established, in part, on the basis of minimizing the thermal stress conditions in the evaporator section of the boiler. Within this section the heat transfer film coefficients on both sides of the tubes are very high, and the temperature difference across the tube wall represents an unusually large fraction of the total temperature difference between the two heat transfer fluids. This, of course, tends to produce abnormally large thermal stresses in the tubes. In order to minimize

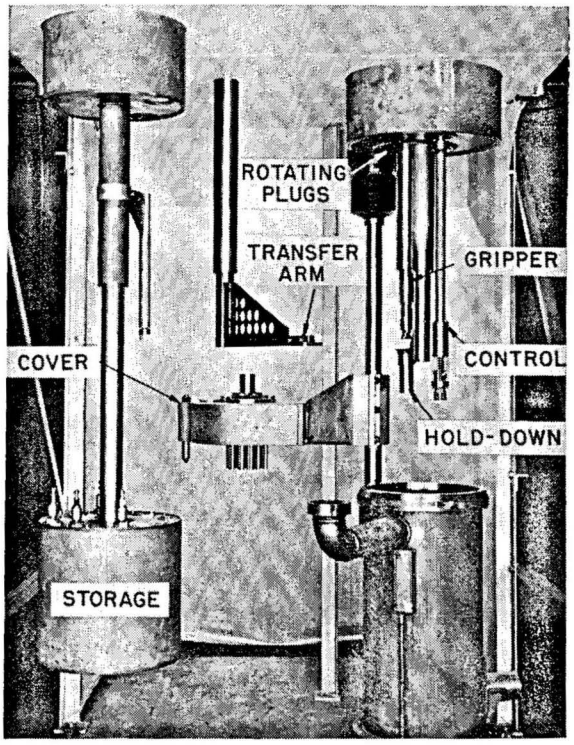


Figure 12. Mock-up of fuel handling system

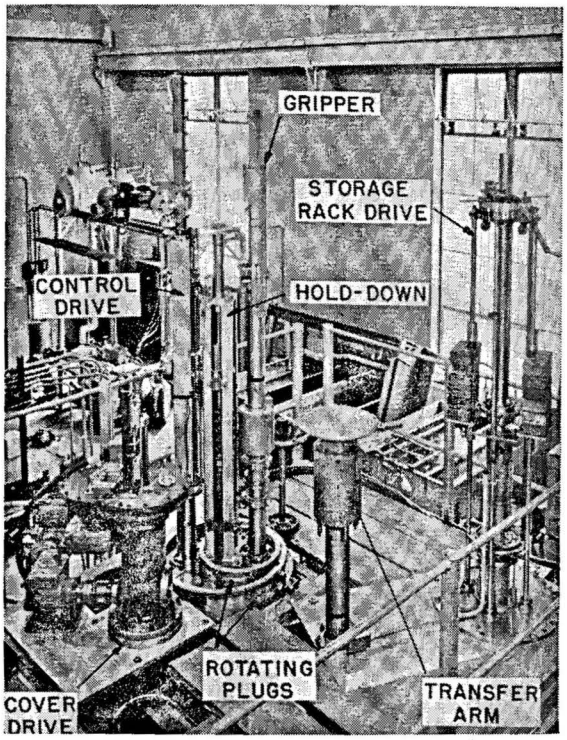


Figure 13. Mock-up of fuel handling system drives

this condition (through reduction of the total temperature difference), the flow rate of the secondary system sodium has been selected for maximum temperature rise through the heat exchanger consistent with allowable thermal stress in the cold end of that unit. Also, the steam saturation temperature has been established at a high level by selection of as high a throttle pressure as economically practical, consistent with maximum conventional pressures for standard turbine units utilizing 850°F steam. A temperature-enthalpy diagram for the water side of the boiler, with corresponding temperature distributions for secondary and primary system sodium superposed, is shown in Fig. 15.

FUEL PROCESS SYSTEM

It is desirable in a fast power reactor to recycle the fuel as rapidly and as economically as possible to minimize the costs of fuel inventory and of fuel processing. Because of the high specific activity of

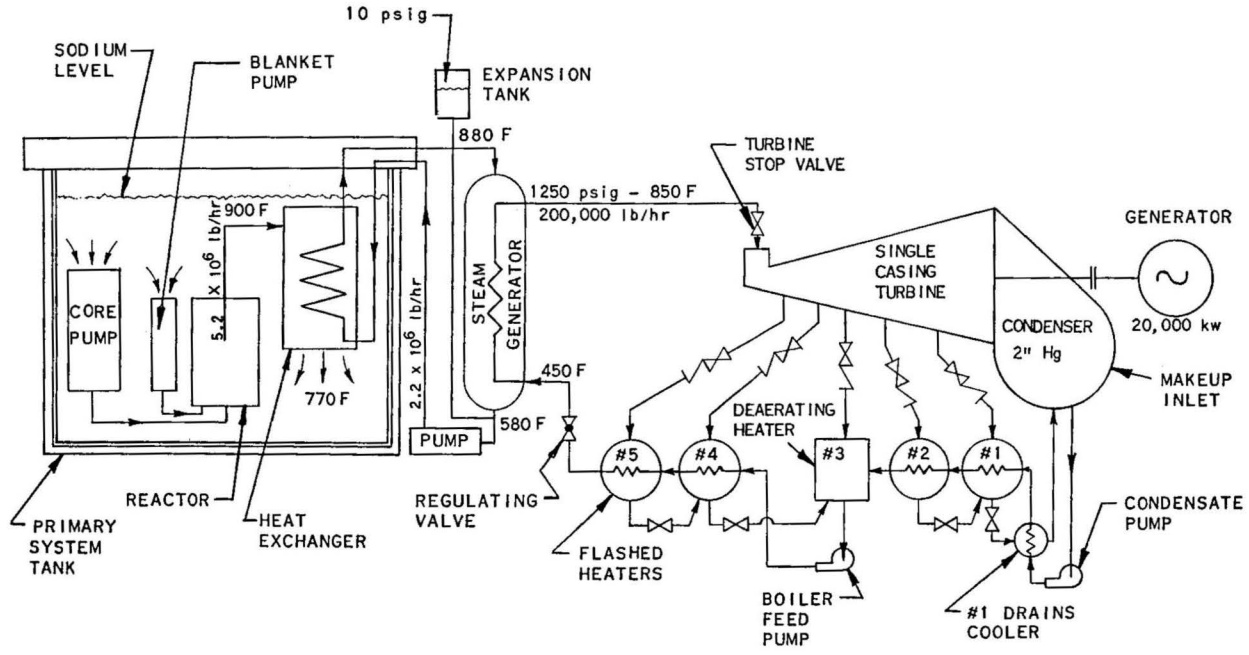


Figure 14. Skeleton flow chart

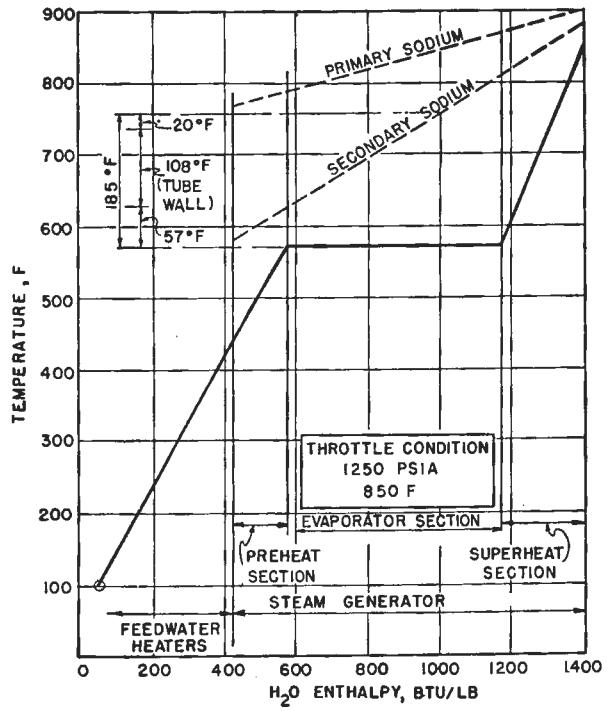


Figure 15. Temperature-enthalpy diagram, EBR-II steam generator

the fuel from the reactor, a long "cooling time" is required if the fuel is processed by present aqueous methods, and a large economic penalty results because of the large fuel inventory required for cooling.

The EBR-II plant employs "pyrochemical" processing which involves high temperature slagging in the molten metal phase. The costly conversion steps required in aqueous systems are avoided, and the size and complexity of the equipment are reduced considerably. However, the process is not as efficient, and complete decontamination is not obtained. The results to date indicate that in excess of 90 per cent of the fission product activity can be removed by the process. This is an acceptable decontamination for fast reactor operation, because it is relatively insensitive to fission product poisoning. Sufficient residual fission product activity remains in the fuel, however, to require shielded remote control fabrication of the alloy. After processing, therefore, it is necessary to reconstitute and fabricate the material, and assemble the fuel elements, completely by remote control methods. The fuel is fabricated into pins, loaded into the fuel element tubes, and assembled into the subassemblies. Obviously, as much preparatory work as possible is accomplished outside the fuel process cell. The structural materials are completely fabricated outside, and the remotely controlled assembly operation is limited to assembly of the fuel elements and the subassemblies.

Since the EBR-II is fueled with plutonium, semiremote operation and complete containment would be required even if complete fission product decontamination was obtained. Therefore, the requirement of complete remote control operation due to residual fission product activity does not represent a large additional burden.

ECONOMIC CONSIDERATIONS

The fast reactor is being developed as a power source for the production of electric power. To be successful, it must be competitive with existing plants (or other new power sources, including other reactor types). In addition to power, the fast reactor produces excess fissionable material (plutonium) which is also a salable product (in a nuclear power economy), and this income can be applied to reduce the cost of power.

The cost of electricity produced by a fast power breeder reactor is a function of the following: (1) annual capital charge (interest, depreciation, insurance, taxes, etc.); (2) fuel costs; (3) operating costs; (4) income other than sale of electricity.

These four elements of cost can be expressed as follows:

1. The cost of electricity due to the annual capital charge can be expressed as:

$$Ca = \frac{D \times A \times 1000}{8760\ P} \qquad (1)$$

where Ca = cost of electricity due to annual capital charge, mills/kwh; D = plant investment, $/kw of electrical capability; A = annual capital charge, fraction per year; P = plant factor, fraction (actual output/possible output); and 1000 = mills/dollar and 8760 = hours/year.

D is the most critical item in this expression, and a primary design objective is to achieve low plant costs. P should be large, which can be attained if operating reliability is achieved. A should be small, but it is controlled by economic structure, not engineering.

2. The fuel costs consist of the following three components:

Inventory + Consumption + Fuel cycle

and can be expressed as follows:

$$Cf = \frac{M \times F \times I \times 1000}{G \times E \times 8760 \times P} + 0 + \frac{R \times 1000}{H \times U \times E} \qquad (2)$$

where Cf = cost of electricity due to fuel costs, mills/kwh; G = reactor capability, kw of heat; M = fuel inventory, kg of plutonium (total system including cycle); F = fuel costs, $/kg of plutonium; E = thermal efficiency, fraction; I = inventory rate, fraction per year (interest on fuel investment); R = fuel cycle cost, $/kg of fuel processed and fabricated; H = "heat value," kwh/kg burned (approx. 24×10^6); and U = fuel utilization factor, kg burned/kg processed.

The second term is zero because this reactor is a breeder and produces more fuel than it consumes.

Later, a credit is taken for excess fuel production which is based on the breeding gain.

To minimize inventory M and processing costs R the EBR-II plant includes an integrated fuel processing and fabrication facility. Considerations of M, E, and U dictate a high specific power, high coolant temperatures, and long burn-up.

3. The operating cost can be expressed as:

$$Co = \frac{S \times 1000}{G \times E \times 8760 \times P} \qquad (3)$$

where Co = cost of electricity due to operating costs, mills/kwh; and S = operating cost—wages, materials, etc., $/year.

The operation of the EBR-II will permit a reasonable extrapolation of these costs to a full scale plant.

4. The Power Breeder Reactor derives a second income from the sale of excess fuel production (credit) which can be expressed as follows:

$$Cc = \frac{B \times F \times 1000}{H \times E} \qquad (4)$$

where Cc = credit for excess fuel, mills/kwh, and B = breeding gain, kg produced/kg burned, equals 1. Note that F has the same value as in item 2 above. It is assumed that the value of the fuel does not change (purchase and sale price are the same).

The total cost of power therefore is:

$$C = Ca + Co + Cf - Cc \qquad (5)$$

A hypothetical reactor has been selected to indicate the contribution of each of these costs to the total cost of power. The numbers selected are rather arbitrary, and their only purpose is to clarify the relationships and to indicate the distribution of costs. If:

D = $300 per kw of electrical capability
A = 0.01 annual capital charge (10% charge)
P = 0.8 plant factor (80%)
G = 500,000 kg—reactor capability (heat)
M = 500 kg—total fuel inventory
F (Case I) = $15,000 per kg—fuel cost
(Case II) = $30,000 per kg—fuel cost
E = 0.3 thermal efficiency (30%)
I = 0.05 inventory rate (5% charge)
R = $3,000 per kg—fuel processing and fabrication cost
U = 0.1 fuel consumed per cycle (10%)
B = 0.5 breeding gain

Table II summarizes the calculations using the two assumed fuel costs for plutonium.

Table II

	Case I		*Case II*	
Plutonium cost, $/kg		15,000		30,000
Ca		4.28		4.28
Cf	0.36		0.72	
	4.17	4.53	4.17	4.89
Cc, Pn credit		−1.04		−2.08
Operating cost		1.00		1.00
C, total power cost		8.77		8.09

The fast power breeder does not require low fuel cost. Considering the above two cases, the cost of power decreases as the price of fuel increases. As fuel varies from $15,000 to $30,000 per kilogram, the first term of Cf increases from 0.36 mill/kwh to 0.72 mill/kwh, while the fuel credit Cc increases from 1.04 mills/kwh to 2.08 mills/kwh. This rather unique condition prevails as long as the inventory charge on the fuel (in the plant) does not exceed the income from the fuel produced. In the above example this does not occur until the fuel inventory M = 1460 kg. This is obtained by equating the first term of Cf to Cc, which are the two relationships involving F.

It is not intended to advocate high fuel costs, but rather to indicate that the success of the fast power breeder is dependent upon engineering development rather than upon low fuel costs. It is obvious from the preceding example that the cost of power is due largely to the amortization of the plant investment, and the cost of processing and fabricating the fuel. The design of EBR-II is directed toward the reduction of these costs.

Operating Experience and Experimental Results Obtained from an NaK-Cooled Fast Reactor

By H. V. Lichtenberger, F. W. Thalgott, W. Y. Kato, and M. Novick,* USA

The Experimental Breeder Reactor (EBR) was designed and built to promote further understanding of the principles of the fast breeder reactor system. It is a liquid metal-cooled reactor operating on fast neutrons and is designed for purely experimental purposes. It is quite flexible, both as to loading and as to operating conditions. It was designed at a time when enriched fissionable material was not available in large amounts and, therefore, the design calls for a small critical mass. Some features of this reactor, particularly the coolant flow system, have been described previously[1] and will not be repeated here.

The reactor consists of three principal parts. The core of the reactor is an assemblage of small diameter cylindrical fuel elements. These contain U^{235} enriched to about 90 per cent. Surrounding the core there is an inner blanket composed of larger diameter rods, each containing natural uranium. The core and the inner blanket are cooled by sodium-potassium alloy. The coolant is contained by a double-walled tank which fits closely around the inner blanket. The third part of the reactor consists of an outer blanket constructed from natural uranium and having a high uranium density. The outer blanket is air-cooled and is movable. It also contains the control rods. By restricting all movable parts of the reactor to the air-cooled outer blanket, it was possible to avoid bearings and other moving machinery inside the tank containing liquid metal. At the time the reactor was designed, the techniques of operating moving parts in a liquid metal system were not as well developed as they are now, and the expedient of putting all moving parts in an air-cooled blanket was recognized as being suitable only for this experimental application. Ultimately, it turned out that the air-cooling of the blanket was the limitation on the operating power level of the reactor.

FUEL ELEMENTS

The enriched uranium in each fuel element is in the form of a cylinder 0.384 inch in diameter and 4¼ inches long. Two of these cylindrical slugs are contained in each fuel element. In order to provide the blanket above and below the reactor core, each fuel element contains, in addition to the enriched uranium slugs, slugs of natural uranium which are placed above and below the enriched uranium slugs. All of these slugs are contained in the bottom part of a stainless steel tube which has a length great enough to extend upward through the shield, which is placed in the reactor tank above the core and the inner blanket. On the part of the stainless steel tube which contains the fuel, there are two ribs. The rods are closely packed together in a hexagonal array as is indicated in Figs. 1 and 2. Rotation of each rod through 30 degrees after insertion in the reactor brings the ribs in contact, or nearly in contact, with neighboring rods and thus each fuel element is firmly held in position by its neighbors. The coolant flows vertically upward over the fuel elements in the core and makes its exit through the upper holding plate by means of a fluted piece which is inserted in the fuel rod at the point where it passes through the holding plate. Figure 2a is a photograph of a 4-inch thick, stainless steel holding plate. The 217 positions for the core fuel elements are shown. The recess surrounding these 217 positions accepts a stainless steel hexagonal pipe which separates the flow of of coolant in the core and inner blanket. Figure 2b shows the topmost holding plate of the reactor with some of the fuel elements in position. The plate shown in Fig. 2b lies above the level of sodium-potassium alloy in the reactor and normally is blanketed by argon gas. Fuel elements are removed from the reactor by opening a hole in the top lid of the reactor vessel and attaching a grapple to the top of the fuel element and then drawing the fuel element into a lead transfer coffin. The over-all length of a fuel element is 10 feet. The greater part of this length is composed of a stainless steel rod.

INNER BLANKET

The inner blanket consists of pieces of natural uranium 15/16 inch in diameter and 20¼ inches long. These are inserted in a stainless steel tube of 0.020-inch thickness. This tube is also provided with a long handle which projects through the upper shield

*Argonne National Laboratory. Including work by B. C. Cerutti, E. N. Pettitt, L. J. Koch, F. D. McGinnis, G. K. Whitham, R. A. Cameron, R. O. Haroldsen, W. H. Zinn, M. Levenson, E. Turk, C. E. Crouthamel, D. Krause, C. D. Curtis, S. H. Klein, D. Okrent, W. C. Redman, S. Untermyer, N. Krisberg, G. R. Ringo, S. G. Kaufmann, M. G. Inghram, D. C. Hess, C. M. Stevens, W. F. Murphy, S. H. Paine, C. Eggler, C. M. Huddleston, V. E. Krohn, G. H. Stonehocker, Argonne National Laboratory.

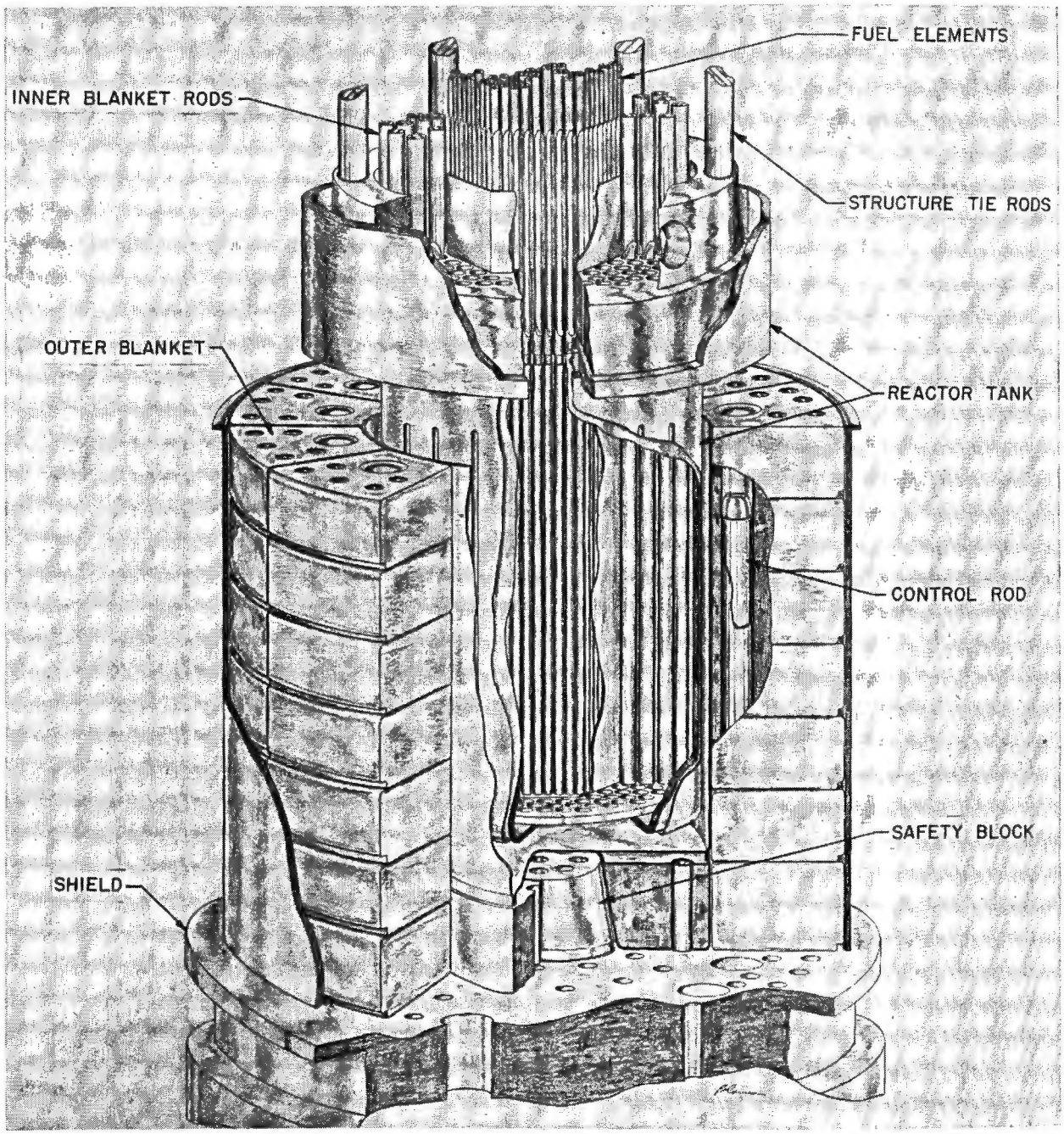


Figure 1. Cut-away showing EBR construction

of the reactor. The coolant enters the reactor in the upper shield above the natural uranium part of the blanket rods. The coolant flows downward over the blanket rods and is prevented from crossing into the reactor core by the hexagonal separator tube mentioned previously. At the bottom of the reactor there is a small plenum chamber where the coolant which has come down through the blanket is turned for flow upward through the core.

REACTOR VESSEL

The reactor vessel, or tank, is shown in both Figs. 1 and 2. It is double walled, as is the piping which leads from the reactor vessel through the reactor shield. The part of the reactor vessel surrounding the reactor core has an inside diameter of 15.87 inches and a length of 28 inches. Above this small diameter part, the reactor vessel increases in diameter and is filled with shielding material, mostly steel. The whole reactor vessel rests on the shoulder formed by the change in diameter; thus the reactor core itself projects below the point of support as a smooth cylinder. The stainless steel plate on which the reactor vessel stands is notched to provide cooling air, which first flows over the outside cover of the reactor tank and then through the notches, thus cooling the reactor sup-

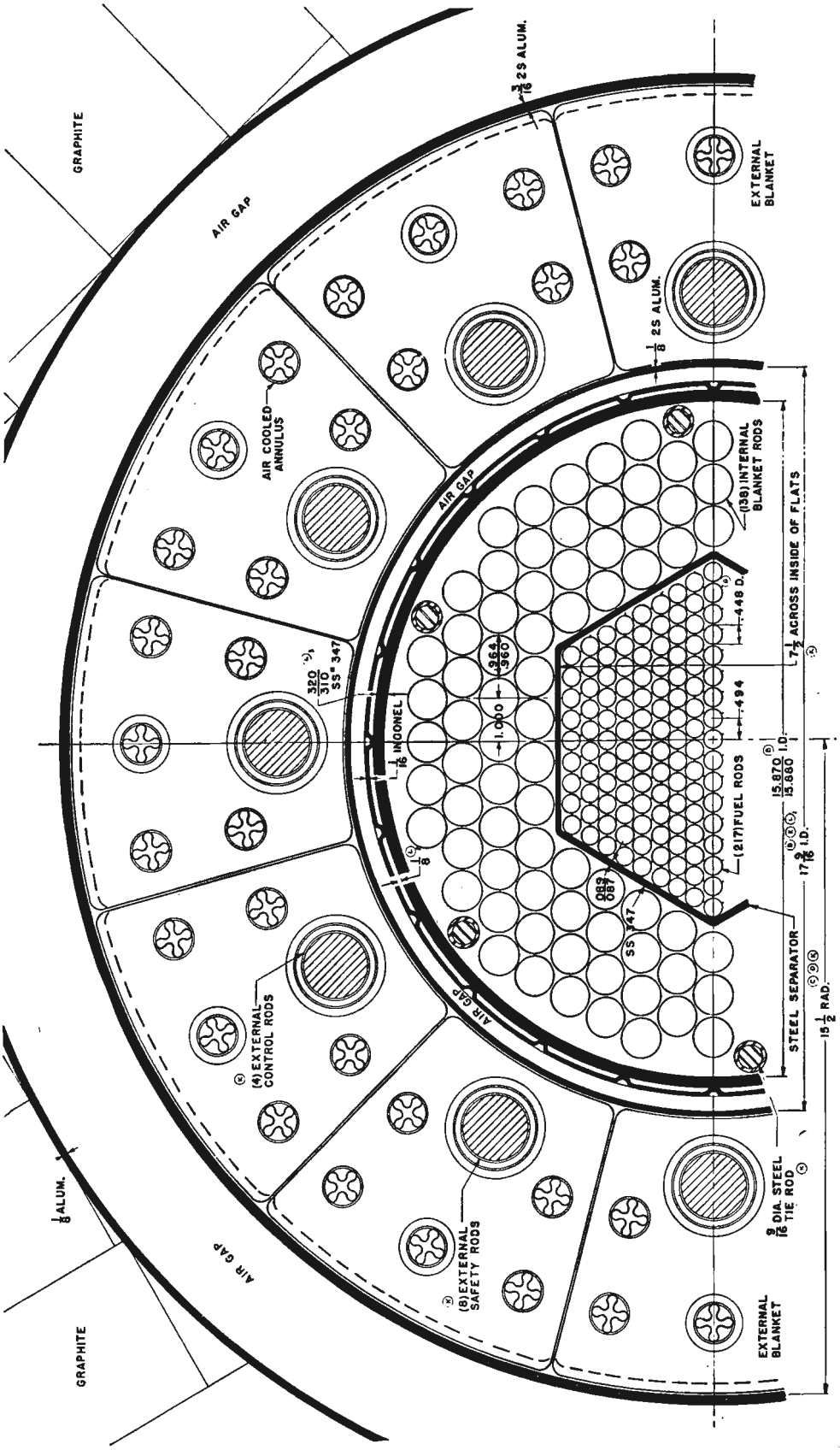


Figure 2. Horizontal cross section at midplane

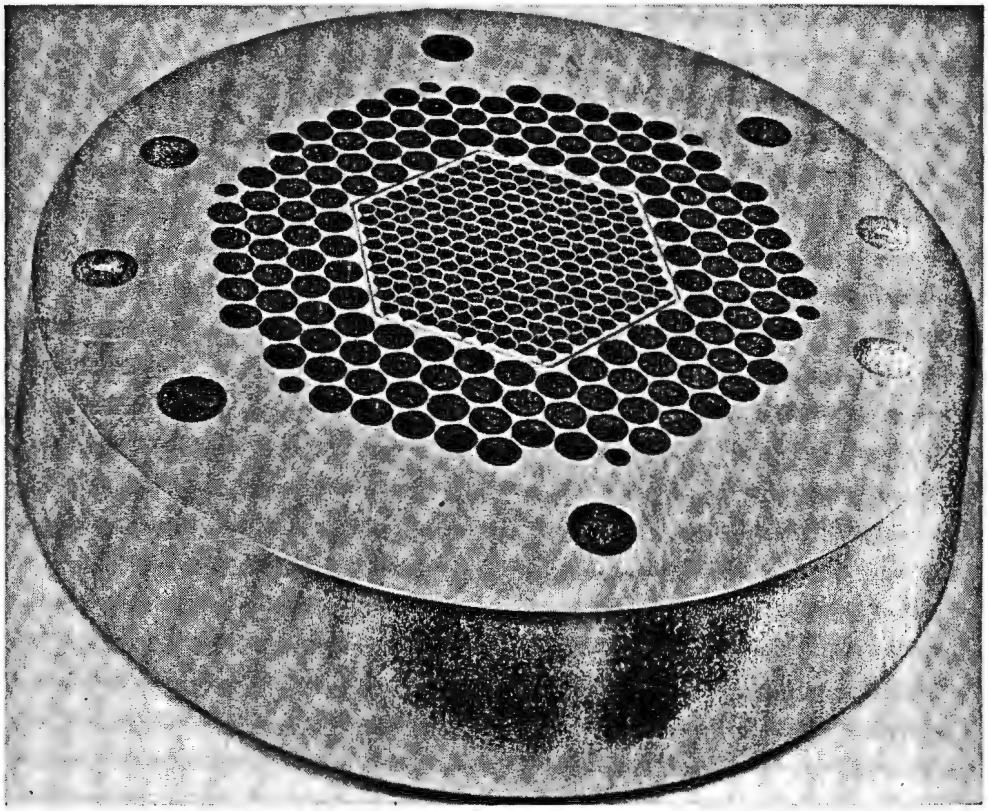


Figure 2a. Stainless steel holding plate for core fuel elements and inner blanket rods

port. It then is drawn through the outer blanket. Figure 2c shows the small diameter part of the reactor tank projecting through its support. The photograph was made with the outer blanket removed from the top of the lower shield. The small diameter part of the reactor tank consists of a stainless steel vessel of wall thickness 5/16 inch, made by deep drawing. It is surrounded by a second tank made of Inconel 1/16 inch thick. The second tank fits snugly on ribs which have been formed in the Inconel. The upper portion of the reactor vessel also is double walled. The double-walled construction serves a number of purposes. The gas space between the two walls provides some thermal insulation, it gives a method for testing for integrity of the inner vessel at any time, and finally, in the event that the inner vessel should develop a leak, the outer vessel would prevent any serious consequences.

OUTER BLANKET

The outer blanket consists of 84 keystone-shaped bricks of natural uranium, each weighing about 100 pounds. These bricks are jacketed in stainless steel of 0.020-inch thickness. The bricks fit together in a stable array because each is provided with a recess on top into which fit projections on the brick above it. Air-cooling of the bricks is obtained by drawing air through five holes which are fitted with sleeves in close contact with the uranium. In order to increase the area for heat transfer, each sleeve carries fins on its inside. Each brick also provides a passage for a control rod and air is similarly drawn through this passage. Figure 2d is a photograph of the outer blanket in which an inner aluminum cylinder is present but in which an outer aluminum cylinder has been removed. The purpose of the aluminum cylinders is to provide additional stability.

CONTROLS

There are 12 control rods, each 2 inches in diameter, made of natural uranium jacketed in stainless steel. These move vertically in the outer blanket bricks. Eight of these control rods normally are used as safety rods. Their time of travel out of the blanket is short. The remaining four normally are used for the running controls and can be positioned with considerable accuracy. The whole outer blanket is mounted on an elevator which is hydraulically driven. Figure 2e shows the main platform of this elevator which carries a shield section on which the outer blanket rests. Below is shown the drive motors which activate the 12 safety and control rods. The elevator can be driven upward at varying speeds. Its last 4½ inches of travel is mechanically controlled, permitting location of the outer blanket around the reactor core with a precision of 0.001 inch. For shutdown, the outer blanket, the shield plug on which it rests, and the elevator can be dropped quickly. The handling of the outer blanket for replacement of bricks is done by lowering the elevator and lifting the outer

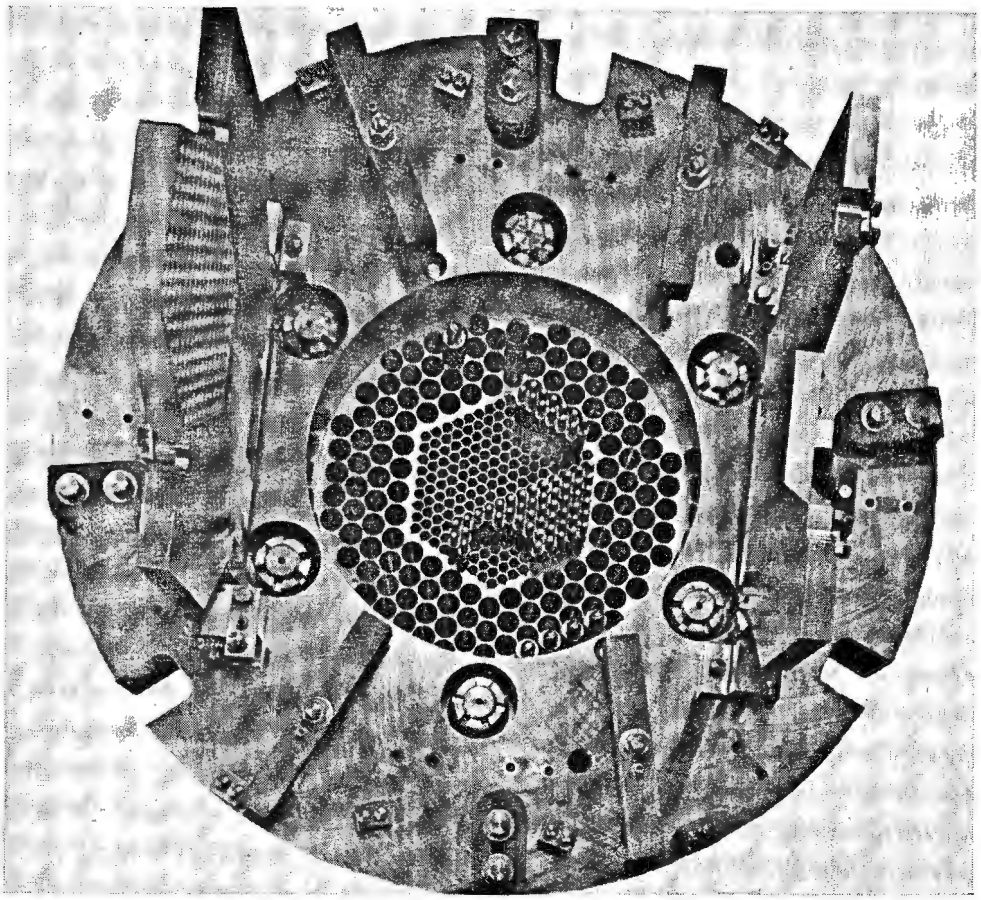


Figure 2b. Top of reactor showing some fuel elements and blanket rods in position

blanket from the shield plug with a handling dolly and transporting it into a neighboring shielded room. There it is disassembled with a remotely operated manipulator. Figure 2f is a photograph of the outer blanket being carried by the handling dolly into the disassembly room. In addition to the rods, a part of the outer blanket in the form of a cylindrical block is arranged so that it can be driven out of the bottom of the outer blanket with pneumatic force. The reactivity changes which are produced by these various controls are indicated in Table I.

Table I. Values of Control Components

	Inhours	*%k*
Four control rods	40	0.10
Eight shutdown rods	80	0.20
One bottom safety	27	0.07
4¼ inch elevator travel	350	0.89
Complete removal of outer blanket	3220	8.2

The values for the reactivity were arrived at by the usual technique of measuring periods and calculating inhour values and Δk. The data of Hughes, *et al.*[2] were used without any corrections for energy of the delayed groups, possible contribution of delays from fissions in fertile material, etc.

There thus exists a total of about 1% k in the operating controls. The fast-acting safety rods can remove 0.27% k in a time of 100 milliseconds. The slower acting external blanket removes 8.2% k in a time determined by the downward acceleration of a little less than "g."

The reactor requires 48.2 kg of U^{235} for criticality in this loading. In order to overcome the temperature coefficient, provide operating control, and provide some additional reactivity for experiments the reactor is loaded with about 52 kg of U^{235}. Over the range from 38°C to 200°C the average temperature coefficient is -3.5×10^{-5} $\Delta k/°C$.

Critical experiments performed with and without coolant indicate that the coolant adds an amount of reactivity approximately equal to that added by 2 kg of U^{235}. It is to be noted that loss of coolant in this reactor reduces reactivity and causes the reactor to shut down, thus contributing to the inherent safety of the reactor.

The reactor is equipped with the instrumentation indicated by the block diagram shown on Fig 3. Separate instruments are provided for start up, safety, and steady operation, having sensitivities and ranges commensurate with their uses. All safety circuits are duplicated in order to provide a high degree of re-

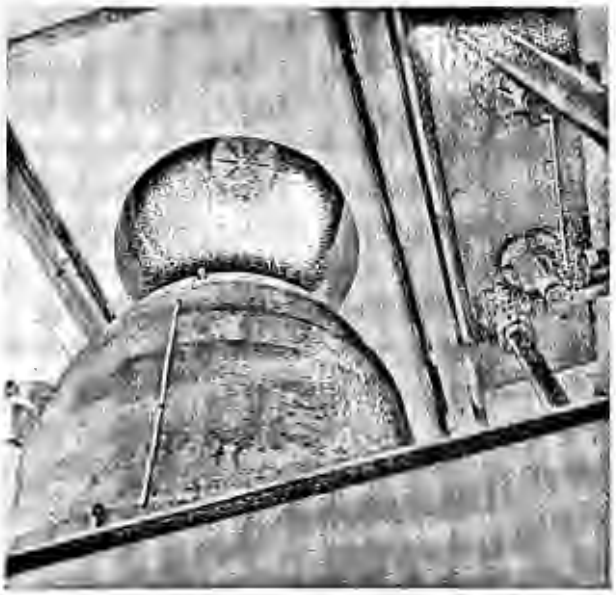


Figure 2c. Reactor vessel viewed from position of outer blanket elevator

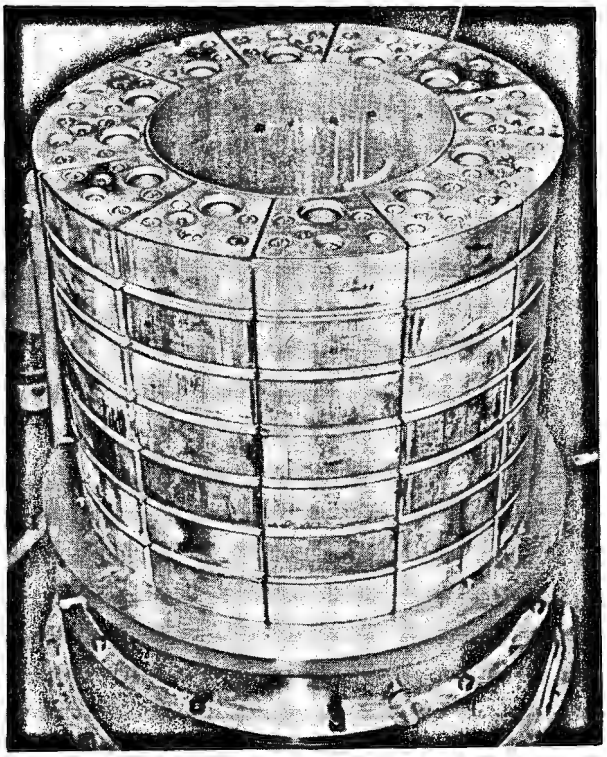


Figure 2d. Outer blanket of reactor

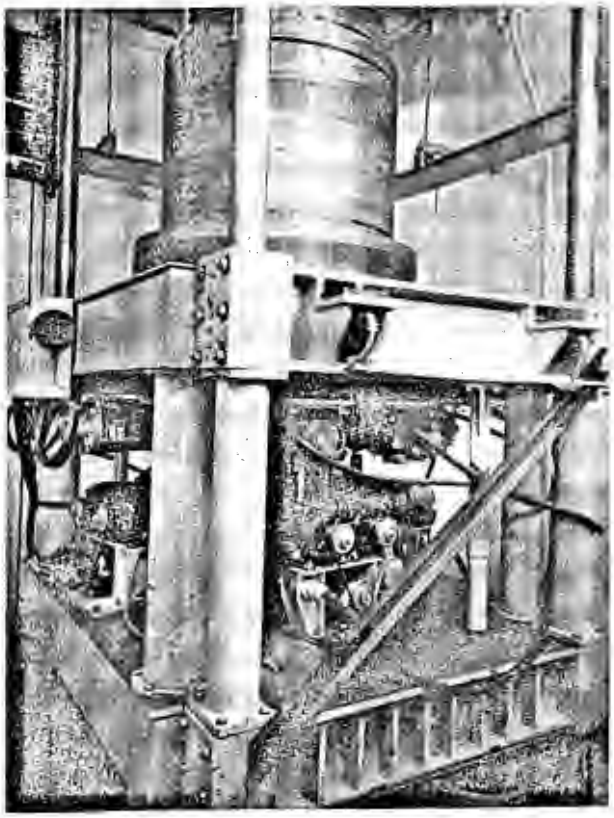


Figure 2e. Outer blanket elevator and control drive motors

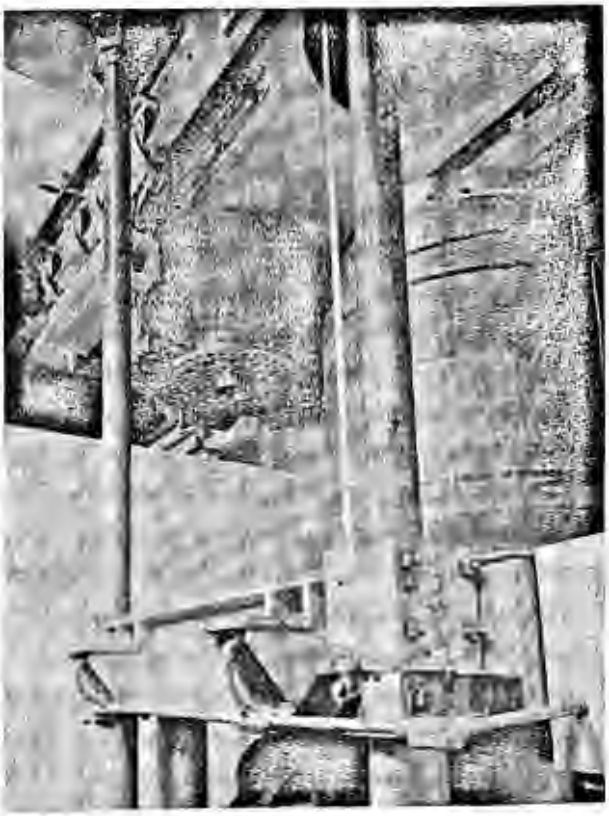


Figure 2f. Transporting outer blanket to disassembly room

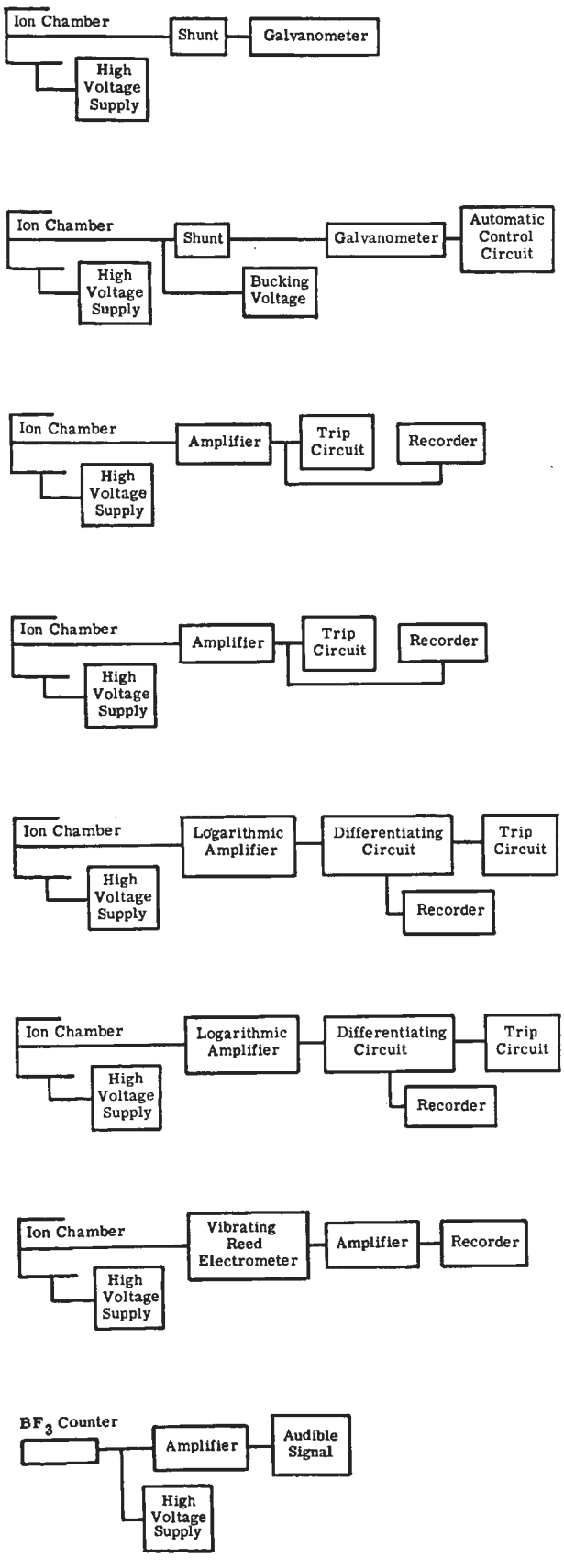


Figure 3. EBR nuclear instrumentation

liability. Ion chambers are located in openings in the reactor shielding at the center plane of the reactor and are either boron-coated or BF_3-filled. The very low neutron level at start up requires instruments of exceptionally long range, approximately 6½ decades.

The reactor operates at a maximum power of 1400 kw. Of this, 72% is generated in the core section of the reactor, 14% in the inner blanket section, and 14% in the air-cooled blanket section. An air flow of 6000 cubic feet per minute maintains the blanket bricks at a temperature of 200°C. The NaK flow through the reactor is normally 290 gallons per minute, giving a temperature rise of 88°C.

The coolant enters the reactor tank above the blanket section, is distributed in the header, flows through the top plate in the fluted section of the rods, and down between the blanket rods to the bottom of the reactor tank. Here it turns and flows up among the fuel rods, through the top fluted sections in the holding plates, and exits sidewards above the inlet.

The sodium-potassium coolant is supplied to the reactor from a high tank, flows through the reactor where it is heated, through a heat exchanger where it is cooled by the secondary NaK system and thence to a lower storage tank. Pumps return it to the high tank. In the secondary system the pump draws the NaK from a storage tank, circulates it through the primary heat exchanger and the steam generator, and returns it to the storage tank. Further details of this system are given elsewhere.[1,3]

Data of a particular day's running which are typical of normal operation are given in Table II. The

Table II. Heat Transfer Data for the EBR Core and Internal Blanket

1.	Reactor core and internal blanket	
	Temperature NaK in	228°C
	Temperature NaK out	316°C
	Flow rate	292 gpm
	Total power produced 1158 kw	3,950,000 BTU/hr
	Power produced in core	960 kw
	Power produced in internal blanket	196 kw
2.	Core	
	Total fuel rod surface area	15.68 ft²
	Average thermal flux, 15.7 cal/sec cm², 209,000 BTU/hr ft²	
	Temperature at center of fuel rod adjacent to center rod	357°C
	Temperature difference between rod center and surrounding NaK	77°C
	Average specific power	18.1 kw/kg
	Cross sectional area for coolant flow	0.1008 ft²
	Coolant velocity	6.5 ft/sec
3.	Inner blanket	
	Total surface area of rods	56.2 ft²
	Average thermal flux, 0.89 cal/sec cm², 11,900 BTU/hr ft²	
	Cross sectional area for coolant flow	0.368 ft²
	Coolant velocity	1.77 ft/sec
4.	Outer blanket	
	Air temperature in	3°C
	Air temperature out	91°C
	Air flow rate	5800 ft³/min
	Power produced in outer blanket	213 kw

high heat fluxes shown point out both the necessity for and the advantages of liquid metal cooling.

The NaK-to-NaK heat exchanger is of a conventional shell and tube design with all joints welded. Primary flow is through the tubes with secondary flow countercurrent through the shell.

The steam generator is divided into three components: the economizer, the boiler, and the superheater. Flow of NaK through these units is countercurrent to the flow of water and steam. Heat transfer tubes in each component are similar and consist of a composite assembly of inner nickel, intermediate copper, and outer nickel tubes. These tubes were assembled by mechanically drawing together and thermally diffusion bonding them for good heat transfer. Total wall thickness of the tube is 5/16 inch, of which 3/16 inch is nickel. An outer stainless steel tube makes up the shell of the heat exchanger, and a bellows is used to allow for differential thermal expansion. Thus each heat exchanger is of a single tube in a shell type where NaK flow is in the shell side and water or steam is in the tube.

In order to limit the quantity of water in the system and increase the heat transfer rate, a forced circulation, falling film type boiler is used. This utilizes the above described heat exchangers in a vertical position. A water film is established at the upper end of the internal tube on its inner surface by means of a baffle. It then runs to the bottom where excess water and generated steam are piped into a drum. Steam is led through a separator in the drum out to the superheater which consists of horizontal heat exchangers with the NaK in the shell side.

The economizer is also a horizontal unit and serves to heat the feed water from the de-aerating tank to steaming temperature before injection into the boiler drum.

Table III gives pertinent information concerning the heat exchanger and steam generator.

Figure 4 is a heat flow diagram giving measured values of flow rates, heat loads, heat transfer rates, temperatures, and other pertinent information for a typical run. Temperature differences throughout the plant are indicated on Fig. 5 which shows temperature-load distribution.

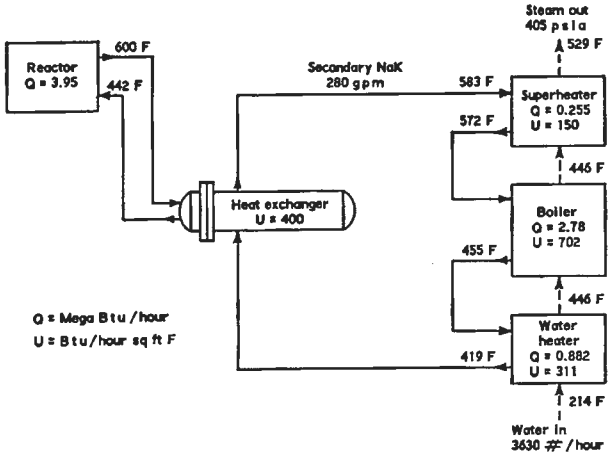


Figure 4. EBR heat flow diagram

Table III. Data on the EBR Heat Removal System

1. Intermediate heat exchanger	
Over-all length	14 ft 8 in.
Shell outside diameter	17 in.
Flow type	Double pass shell and tube
Number of tubes	102
Type of tubes	¾ in. OD, 16 gauge, hairpin
Tube material	"A" nickel
Outside area of tubes	495 ft²
2. Heat exchanger tube used in superheater, boiler, and economizer	
Effective length	9 ft 6¾ in.
Outside diameter	2⅝ in.
Inside diameter	2 in.
3. Superheater	
Number of heat exchangers	4
Arrangement	Series with countercurrent flow
Total inside area of tube	20 ft²
Shell size	5 in. IPS
Shell side cross sectional flow area	14.6 in²
NaK velocity	6.15 ft/sec
Steam velocity, average	58.8 ft/sec
4. Economizer	
Number of heat exchangers	9
Arrangement	Series with countercurrent flow
Total inside tube area	45 ft²
Shell size	5 in. IPS
Shell side cross sectional flow area	14.6 in²
NaK velocity	6.15 ft/sec
Water velocity in annulus around 1¾ in. OD baffle	3.43 ft/sec
5. Boiler	
Number of heat exchangers	18
Arrangement	Parallel with countercurrent flow
Total inside tube area	90 ft²
Shell side cross sectional flow area per tube	1.98 in²
Shell size	3 in. IPS
NaK velocity	2.51 ft/sec
Water flow rate	13 times steam rate

Values calculated for heat transfer coefficient in the economizer appear to be quite low. It turns out that the final four tubes are transferring heat to saturated water. This was determined by measuring local water temperatures in each tube. By dividing the economizer into two parts, which include the first five and the second four tubes, respectively, quite different values are obtained. The first has a value of 508 BTU/hr-ft² °F, and the second has a value of 204 BTU/hr-ft² °F. Temperature differentials in the second section are very low for boiling heat transfer, and the resulting gas blanket may be responsible for the poor values obtained.

Steam which is produced in the steam generator is used to drive a turbogenerator set and provide power for the building and area. Rather than try to control reactor power by load requirements, it is held at constant power and excess steam is generated. A back pressure regulating valve maintains constant steam main pressure by unloading the excess steam directly to the condenser. The efficiency of this plant is found to be about 17%. This low value may be

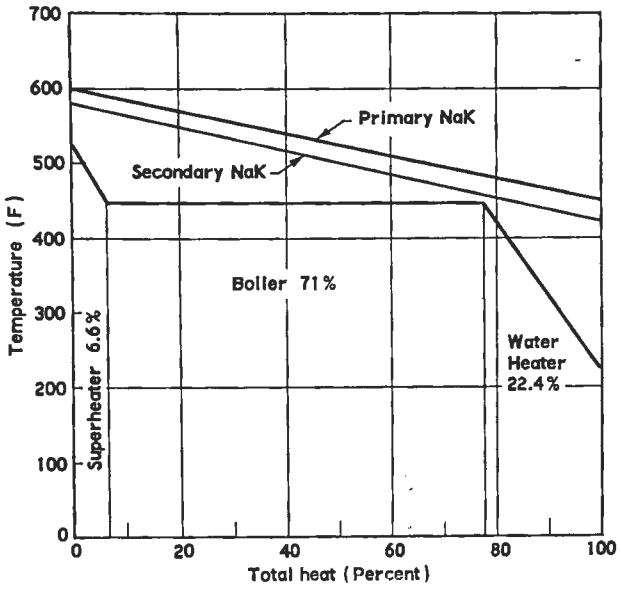


Figure 5. Temperature-load distribution of EBR

attributed to the small size of the plant and to the fact that it is operated at only part load.

The reactor system has proven to be very reliable and has given very good service during 3½ years of operation. It has been used for a large number of experiments and has generated 4×10^6 kwh of heat, a good part of which has been used in the generation of electricity for plant operation. This plant is inherently quite stable and is largely self-regulating so that very little effort is required by the operators in maintaining the system at proper operating conditions.

The neutron flux in the reactor has been measured by the use of foils of phosphorus and gold. The 170-minute activity of Si^{31} formed by the *n-p* reaction on P^{31} having a threshold of about 1 Mev was observed as a measure of the high energy neutron flux. The 2.7-day activity of gold was used to determine the flux of low energy neutrons. The absolute fluxes were estimated by comparison of the activity of phosphorus foils irradiated in the reactor with that of a phosphorus foil irradiated on a converter plate of U^{235} placed in a known thermal flux. The fluxes obtained by these measurements are given in Table IV.

Table IV. Neutron Fluxes in the EBR (All measurements taken on center line of reactor)

Position	*Neutrons/cm²/sec*
Center of core	1.1×10^{14}
Boundary between core and blanket	5.7×10^{13}
Reactor tank wall	2.9×10^{12}
Outer edge of external blanket	3.4×10^{11}
6 in. from edge of blanket in thermal column	1.8×10^{12}

These high fast neutron fluxes make this reactor a very valuable tool for the study of irradiation damage to various materials. Figure 6 gives the result of a typical set of such experiments. The effects of radiation on tensile strength and hardness of notched test specimens of type 347 stainless steel taken from the same heat used to fabricate the tank, the baffle and rods of EBR have been determined for irradiations up to 2.2×10^{20} nvt. It should be noted that this plot gives integrated flux on a logarithmic scale, i.e., these effects which required approximately one year of continuous full power operation for specimens in this position will require ten years to increase the hardness an additional 3 Rockwell A points or the tensile strength another 20,000 psi, provided saturation effects do not appear.

Operation of the Experimental Breeder Reactor at power for extended periods, with the attendant burnup of U^{235} and production and burnup of Pu^{239}, has permitted the determination of the actual conversion ratio achieved by the reactor and has provided a base point for extrapolation to predict the performance of future designs. The conversion ratio (C.R.) is defined as the ratio of the number of Pu^{239} atoms produced for each U^{235} atom destroyed:

$$\text{C.R.} = \frac{\text{Pu}^{239}\text{ production}}{\text{U}^{235}\text{ fission}} \cdot \frac{\text{U}^{235}\text{ fission} + \text{U}^{236}\text{ production}}{\text{U}^{235}\text{ fission}} \quad (1)$$

or

$$\text{C.R.} = \frac{N^{238}\ \overline{\sigma}_c^{238}}{N^{235}\ \overline{\sigma}_f^{235}} \cdot \frac{1}{1 + \overline{a}(235)} \quad (2)$$

where N^{238} and N^{235} represent the relative amounts of fertile and fuel material through the reactor. Two methods of determining the conversion ratio of the reactor have been employed, which, although differing in some details of the results, yielded conversion ratios of 1.00 ± 0.04 and 1.01 ± 0.05.

The basic method, the one originally planned for the measurement, consisted of radiochemically determining the total (time-integrated) production of Pu^{239} and fission of U^{235} from a liberal sampling of representative sections of core and blankets. A cou-

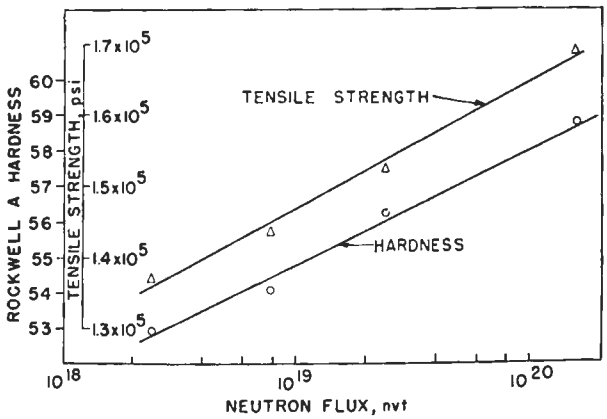


Figure 6. Hardness and tensile strength of type 347 stainless steel vs total flux

pling of these data with a determination of α, the ratio of radiative capture to fission, of U^{235}, properly averaged over the reactor, yielded a conversion ratio of 1.00 ± 0.04. Details of the radiochemical techniques employed in the determination are presented elsewhere.[4] A typical distribution resulting from the sampling, the distribution of plutonium production in the outer blanket bricks of EBR, is presented in Fig. 7. The contribution of core, inner blanket, and outer blanket to the conversion ratio determination are summarized in Table V.

Table V. Comparison between Radiochemical and Physical Determination of Conversion Ratio

	U^{235} atoms destroyed		Pu^{239} atoms produced	
	Radiochemical	*Physical*	*Radiochemical*	*Physical*
Core	0.909	0.908	0.004	0.004
Inner blanket	0.038	0.029	0.354	0.331
Outer blanket	0.050	0.060	0.595	0.639
Control rods	0.003	0.003	0.042	0.035
Total	1.00	1.00	1.00	1.01

$\bar{\alpha}_{core} = 0.138$ $\bar{\alpha}_{reactor} = 0.143$

A secondary method for determining the conversion ratio of the reactor, consisted of measuring, in a short irradiation, the distribution of the rates of U^{235} fission and U^{238} capture throughout the core and blanket.

The relative fission distribution throughout the reactor was determined both by foil irradiation and fission counter traverses. Thin foils of enriched U^{235} were irradiated at various positions in the reactor core and blanket. The integrated gamma-ray activity of the resultant fission products was measured with a NaI-scintillation counter to determine the fission rate. A series of vertical irradiation traverses made with the foils along two radial directions from the center of the reactor was normalized to a foil irradiated at the same time in the thermal column; thus a fission distribution rate of fast fissions per thermal column fission as a function of position was obtained. The fast fission distribution was also determined with the use of small fission counters which were lowered into empty thimbles in the reactor. These fission counters had dimensions of about 1 cm diameter and 4 cm length, with a central electrode coated with enriched U^{235}. Again the distribution of fast fissions per thermal column fission was obtained by exposing the fission counter in the thermal column, with subsequent normalizations accomplished using BF_3 ionization chambers to indicate the reactor power. Typical fission rate distributions, both axial and radial, are shown in Figs. 8 and 9.

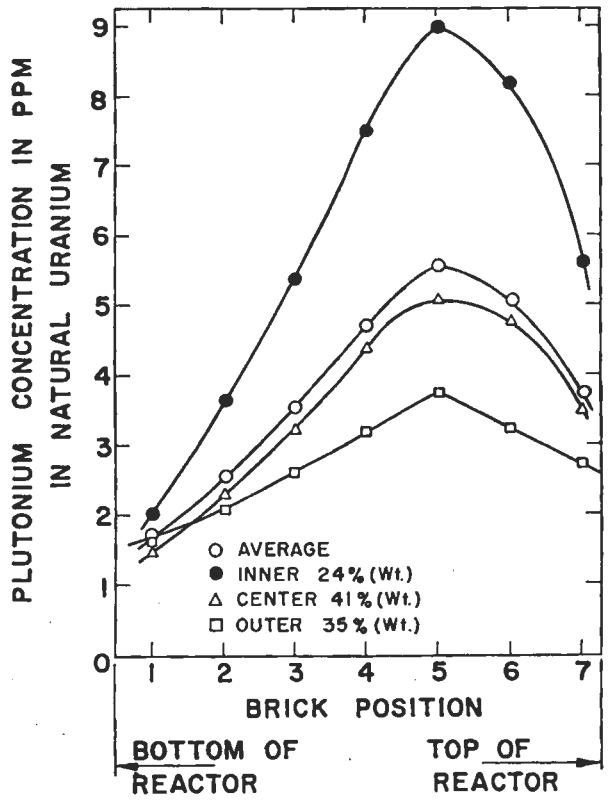


Figure 7. Plutonium distribution in the EBR outer blanket bricks

The capture distribution of U^{238} was determined by irradiating thin U^{238} foils at a number of positions distributed throughout the reactor. A series of vertical traverses was made with the foils sandwiched between simulated fuel or blanket elements in the core and blanket of the reactor, the normalization between the different irradiations being made with a foil irradiated in the thermal column of the reactor. The reactions which take place are:

$$n + U^{238} \longrightarrow U^{239} \xrightarrow[23.5\ \text{min}]{\beta,\ \gamma} Np^{239} \xrightarrow[2.33\ \text{days}]{\beta,\ \gamma} Pu^{239} \xrightarrow[2.3 \times 10^4\ \text{years}]{\alpha,\ \gamma}$$

The capture rate is obtained by observing the 100-kev gamma-ray emitted from the decay of Np^{239} with a conventional single channel pulse height analyzer as shown by Axtmann and Stutheit.[5] The analyzer was set so that just the top of the 100-kev photoelectric peak in the thin NaI crystal was observed. From these foils a distribution of fast captures per thermal column capture is obtained as a function of position throughout the reactor. An integration of the fast captures per thermal column capture over the entire reactor volume resulted in a total capture rate per thermal column capture for the reactor. The distribution of fast captures per thermal column capture in a vertical traverse is shown in Fig. 10. The capture distribution over the radial plane is shown in Fig. 11.

These rate distributions when coupled with a determination of α of U^{235} must be properly normalized to yield the conversion ratio. The proper normalization, i.e., the proper ratio of the effective capture cross

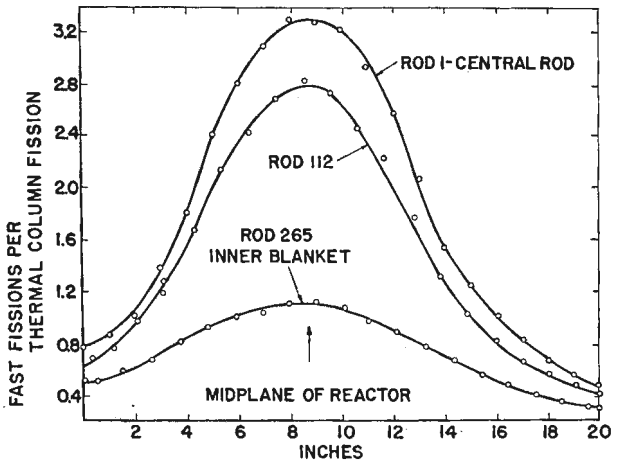


Figure 8. U^{235} fission distribution along vertical axis of EBR

section of U^{238} to the effective fission cross section of U^{235}, may be written as:

$$\frac{\overline{\sigma_c}^{238}}{\overline{\sigma_f}^{235}} = \frac{\sigma_c{}^{th}}{\sigma_f{}^{th}} \times \frac{C_c{}^f}{C_c{}^s\left(1 - \frac{1}{CR_c}\right)} \times \frac{C_f{}^s\left(1 - \frac{1}{CR_f}\right)}{C_f{}^f} \tag{3}$$

where $\sigma_c{}^{th} = 2.8$ barns (thermal capture cross section of U^{238}); $\sigma_f{}^{th} = 580$ barns (thermal fission cross section of U^{235}); $C_c{}^f =$ counting rate of depleted U^{238} foils irradiated in fast flux; $C_c{}^s =$ counting rate of depleted U^{238} foil irradiated in thermal column; $C_f{}^f =$ counting rate of enriched U^{235} foil irradiated in fast flux; $C_f{}^s =$ counting rate of enriched U^{235} foil irradiated in thermal column; $CR_f =$ cadmium ratio of U^{235} foil irradiated in thermal column, and $CR_c =$ cadmium ratio of U^{238} foil irradiated in thermal column. The conversion ratio is obtained by integrating the normalized capture and fission distribution over the reactor volume:

$$C.R. = \frac{\sigma_c{}^{th}\left(1 - \frac{1}{CR_f}\right)\int N^{28}\,(C_c{}^f / C_c{}^s)\,dV}{\overline{\sigma_f}^{th}\left(1 - \frac{1}{CR_c}\right)\int N^{25}\,(C_f{}^f / C_f{}^s)\,dV} \cdot \frac{1}{1 + \bar{\alpha}} \tag{4}$$

where N^{28} and N^{25} are the relative amounts of U^{238} and U^{235} in the entire reactor.

The contributions of core, inner blanket, and outer blanket, to the conversion ratio as determined physically are also summarized in Table V. For the movable outer blanket in its position of maximum effectiveness, the physically determined conversion ratio is 1.01 ± 0.05.

There is rather strong circumstantial evidence that a considerable number of neutrons are leaking out of the blanket of EBR and that a more efficient blanket would make possible a conversion ratio of approximately 1.3.

Multigroup calculations on EBR, done is spherical geometry, with a blanket of composition and thickness equal to the radial blanket of the EBR, followed by two feet of solid graphite reflector, indicate a leakage of 0.2 neutron per U^{235} atom destroyed in the core. In the actual EBR design the upper blanket is much inferior to the radial blanket, and neither upper nor lower blankets are backed by graphite. In addition 1/7 of the radial blanket bricks face beam holes rather than graphite reflector.

Furthermore, measurements in the thermal column, where the two-foot thick graphite reflector is reinforced by several additional feet of graphite, indicate the leakage rate is sufficiently high in this direction that even with a one-foot thick blanket and the extra thick graphite reflector 2 to 3% of the neutrons still leak from the blanket.

Insufficient data are available to calculate the leakage through the top and bottom portions of the blanket. However, examination of Figs. 7, 8, and 10 for axial distribution of both production of plutonium and fission of U^{235} indicates no net flow of neutrons across the central plane of the reactor. Under this assumption, if the effects of the unsymmetrical blanketing above and below the core are separated, one obtains a conversion ratio for the lower half of the reactor of 1.2 while that for the upper half is 0.8.

Since there is a considerable leakage even out of the lower half of the reactor, this suggests that sufficient blanket surrounding the entire reactor would achieve a conversion ratio higher than 1.2.

A neutron balance substantiates this hypothesis. The conversion ratio may be expressed as

$$\text{C.R.} = \frac{\nu - 1 - \alpha - A - L + F\,(\nu - 1)}{1 + \alpha}$$

where $\nu = 2.5$ neutrons per fission; $\infty = \sigma_c{}^{U^{235}}/\sigma_f{}^{U^{235}} = 0.143$; $A =$ neutrons absorbed in structure and

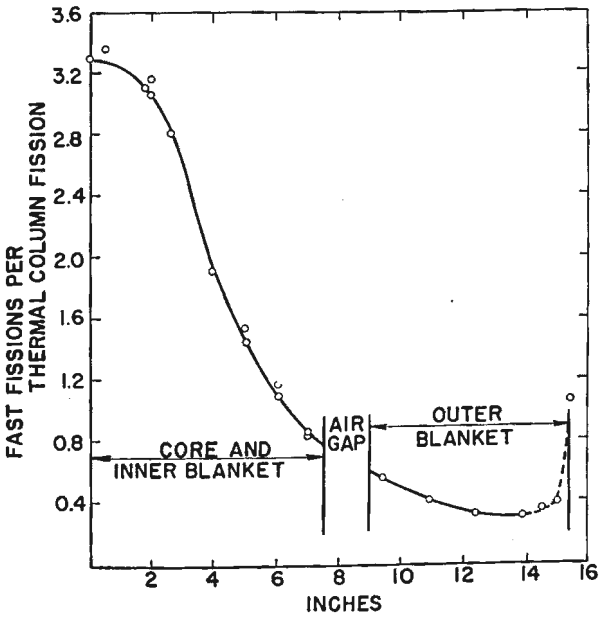


Figure 9. Radial U^{235} fission distribution along midplane of EBR

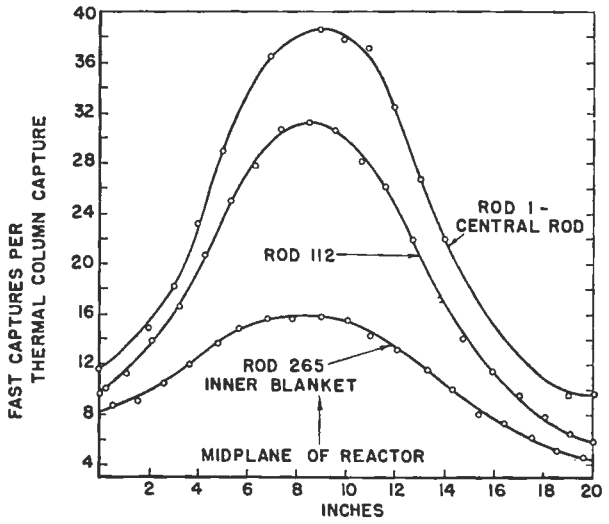


Figure 10. U^{238} capture distribution along vertical axis of EBR

coolant = 0.08, assuming an effective capture cross section of 40 millibarns for iron; L = neutrons leaking from blanket per U^{235} fission; F = U^{238} fissions per U^{235} fission = 0.174 from integration of U^{238} fission rate data; and C.R. = 1.01.

Solving for the leakage yields L = 0.36 neutron per U^{235} fission. If 90% of these leaking neutrons could be captured in U^{238}, the breeding ratio would become approximately 1.3.

Both methods for obtaining the conversion ratio of EBR make use of the value of α of U^{235} as a function of position in the reactor. Again, the irradiated fuel provided material which permitted a measurement to be made for the core, where α is important to the conversion ratio. In addition, irradiated blanket slugs provided material which permitted a determination of α of Pu^{239} in the blanket. Mass spectrometric analyses of the material gave the ratio of U^{236} to U^{235} in the core, and Pu^{240} to Pu^{239} in the blanket. These ratios coupled with determination of U^{235} fissions in the core material and estimates of Pu^{239} fissions based on the measured and normalized Pu^{239} fission rate distribution in the blanket material give the respective α's. The α of U^{235} as a function of radial position in the core is given in Table VI and plotted in Fig. 12; α of Pu^{239} as a function of radial position in the blanket is given in Table VII and plotted on Fig. 13. Details of the techniques employed are presented elsewhere.[6]

It is evident that, as one moves from the center of the core outward, with the attendant softening of the

Table VI. Radioactive Capture to Fission Ratio (α) of Uranium-235 in EBR

Distance from center (cm)	$\frac{U^{236}}{U^{235}} \times 10^5$	α
0	4.5 ± 0.5	0.11 ± 0.01
5.0	4.6 ± 0.7	0.13 ± 0.02
0.4	4.4 ± 0.2	0.17 ± 0.01

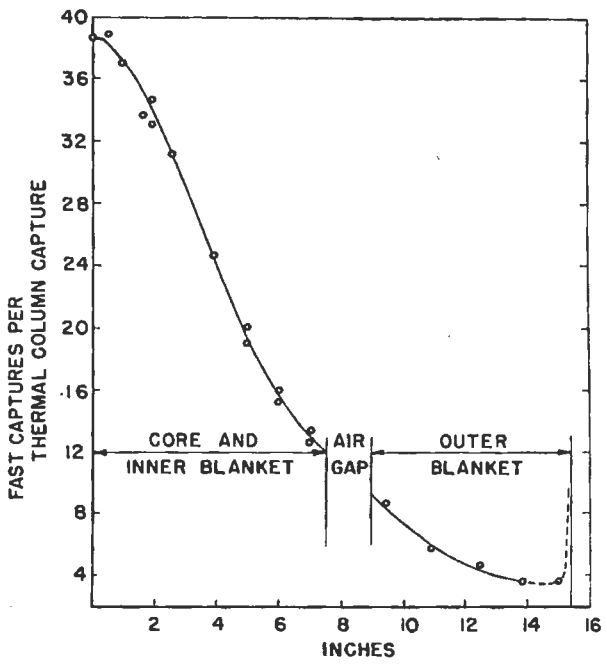


Figure 11. Radial U^{238} capture distribution along midphone of EBR

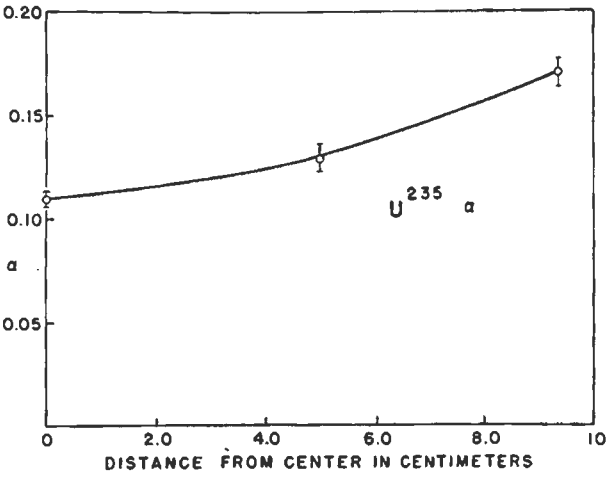


Figure 12. Radiative capture to fission ratio (α) of uranium-235 in EBR

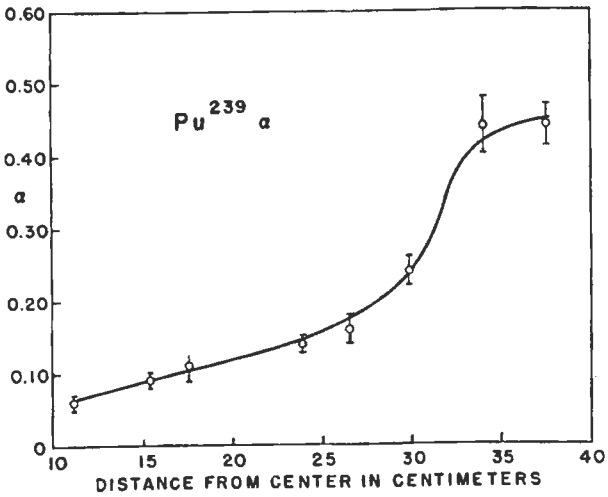


Figure 13. Radiative capture to fission (α) of plutonium-239 in EBR

Table VII. Radiative Capture to Fission Ratio (α) of Plutonium-239 in EBR

Distance from center (cm)	$\frac{Pu^{240}}{Pu^{239}} \times 10^5$	α
11.1	1.1 ± 0.2	0.06 ± 0.01
15.4	1.0 ± 0.1	0.09 ± 0.01
17.6	0.9 ± 0.2	0.11 ± 0.02
23.9	3.2 ± 0.2	0.14 ± 0.01
26.5	2.8 ± 0.3	0.16 ± 0.02
29.8	3.5 ± 0.4	0.24 ± 0.02
34.1	6.8 ± 0.7	0.44 ± 0.04
37.6	23.0 ± 1.6	0.44 ± 0.03

neutron spectrum, α of U^{236} increases, rising from 0.11 at the center of the core to 0.17 at the edge of the core. In the blanket the effect of the softening spectrum, particularly near the graphite reflector, is quite pronounced. α of Pu^{239} goes from 0.06 at the inner edge of the blanket to 0.44 at the outer face.

It is also evident that, if one compares a plutonium-fueled breeder reactor with a U^{235}-fuel converter reactor, assuming the same loss of neutrons by parasitic capture and leakage CL in both, the same fast fission bonus, $F(\nu^{28} - 1)$, and the same high energy spectrum of neutrons, the breeding ratio BR

$$BR = \frac{\nu - 1 - \alpha - CL + F(\nu^{28} - 1)}{1 + \alpha} \qquad (6)$$

of the plutonium-fueled reactor can be much greater than the conversion ratio of the uranium-fueled reactor, primarily due to the higher ν of Pu^{239} (2.9 as opposed to 2.48 for U^{235}), but also because α of plutonium can be lower.

In the light of this strong variation of α with energy spectrum of neutrons, as well as with the variation of other cross sections with neutron energy, a particular emphasis is placed on determination of energy spectra in the EBR. Unfortunately the energy range of interest, from a few kilovolts to a few thousand kilovolts, is one which is particularly hard to measure, the basic difficulty being lack of a suitable spectrometer for use in the energy range from 10 to 500 kev.

This problem has been attacked in two ways. First, measurements of the detailed neutron spectrum in a beam extracted from the reactor have been made with a cloud chamber and, for the higher energies, with proton recoil plates. Second, measurements of the fission ratios of various fissionable isotopes have been made, the ratios giving integral spectral information. This program of measurement is not complete and results obtained are tentative.

Beam measurements have been made at the center and 2.6 inches from the center of the core. The tube used to extract the beam from the reactor core was 11 feet 10 inches long followed by a two-foot steel liner in a lead plug seated in the top concrete shield. The collimation of the beam was thus determined by a steel lined hole 0.40 inch ID, 14 feet long, with a 2-inch gap in its wall 2 feet from the end. The alignment of the collimator was checked optically. Photographic films exposed to the beam gave circular spots of apparently uniform darkness. A major difficulty of beam measurements is the determination of correspondence of the spectrum in the core and in the beam. To establish this, the tube was filled to the vertical mid-plane of the core with a mixture of uranium, sodium-potassium, and iron within a few per cent of the actual core composition. The correspondence of extracted beam and core spectra was checked by obtaining U^{235} to U^{238} fission ratios with fission counters and the ratio of phosphorus activity to gold activity using foils in both the core and the beam. Results of these measurements are given in Table VIII. Within the accuracy of these tests, the spectrum in the beam was the same as that in the core.

Table VIII. Measurements of the Beam from the Central Rod Position in the EBR

Fission counters

Position	*Bias of U^{238} counter in volts*	*Bias of U^{236} counter in volts*	$\frac{U^{235}\ counter\ fissions}{U^{238}\ counter\ fissions}$
Core	150	200	9.0 ± 2.5
Beam	150	200	10.9 ± 2.5
Core	200	300	7.3 ± 2.0
Beam	200	300	8.6 ± 2.0

Foils

Position	$\frac{Phosphorus\ activity}{Gold\ activity}$
Core	1.98 ± 0.20
Beam	1.95 ± 0.20

The hydrogen-filled cloud chamber used to measure the neutron spectra of beams was 12 inches in diameter and 2 inches deep. Hydrogenous material was carefully excluded from the immediate vicinity of the cloud chamber in order to minimize scattering effects that could give rise to undesired tracks. This precaution was rewarded by an almost total absence of tracks not originating along the narrow and well-defined neutron beam.

The cloud chamber expanded at pre-set time intervals of about one-half minute and was photographed stereoscopically. An arrangement of mirrors allowed three views to be taken simultaneously by one camera, as illustrated schematically in Fig. 14, thus permitting, from the oblique views, a measurement of the dip angle of the track in the cloud chamber.

If a track lay within 5 degrees of the plane of the cloud chamber, the track was accepted, provided it

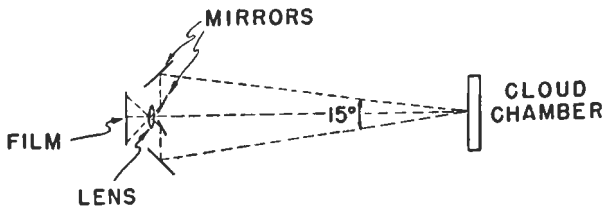


Figure 14. Stereoscopic arrangement for recording tracks

met certain other criteria; namely, it had to originate in the lower half of the cloud chamber and along the line of the incident neutron beam, and it had to lie within 30 degrees of the direction of the vertical beam. This restriction to recoil angles of less than 30 degrees was imposed for two reasons: because it was desired to minimize uncertainty in the cosine-squared angle correction factor, and because the stereoptical arrangement was most sensitive to dip angle in or near the vertical direction.

The lengths and recoil angles of tracks as measured from the central view were converted to neutron energy and corrected for recoil angle, cross section for (n,p) scattering, and the probability of track rejection due to escape from the illuminated volume.

Around 5000 acceptable tracks were found for the entire survey, including four cloud chamber pressures, 15, 30, 60, and 120 cm Hg, thus spanning the region from 100 kev to 1 Mev quite satisfactorily.

For higher energies the technique of observing recoil proton tracks in nuclear emulsions was used. These emulsions (Ilford C-2 Plates) were exposed in the beam just below the cloud chamber for a few hundred watt seconds of reactor operation. These were developed and measured by standard methods using tracks that lay within ± 7 degrees of the beam direction in the emulsion. The results gave a spectrum which was considered usable from 800 kev to 4 Mev.

The complete spectrum was then obtained by fitting the two observations in the region of overlap. The results are shown in Figs. 15 and 16.

Figure 17 shows the comparative spectra in the two locations normalized to the fission spectrum at 3 Mev. It is obvious that the spectrum softens as one moves outward from the center. It is also obvious that lower energy measurements must be made to complete the spectrum.

A check on the validity of these spectra is furnished by a comparison with experimental values of the calculated ratios of activation or fission for materials of fairly well-known cross section exposed to the measured spectra. Such a comparison is given in Table IX.

As previously mentioned integral spectral information can be obtained by comparing the ratios of activities or fission induced by the neutron spectrum in isotopes whose cross sections are known and whose cross sections vary differently over the energy range of interest. Threshold reactions are particularly useful for this purpose. A program of such measurements is being carried out in EBR. Since the ratio of the responses is the quantity needed, the measurements can be made either on an absolute basis, such as radiochemical determination of total fissions, or fission counting with counters of known efficiency containing known amounts of material, or they can be made on a relative basis, by normalizing in a thermal flux, knowing either the cross section in the thermal flux, or using known mixtures of a threshold fissioning material with U^{235} in a thermal flux. In the latter case comparison with the response of a U^{235} counter gives that portion of the fission in the fast flux due to the U^{235} diluent.

Miniature fission counters, mentioned before, were developed for these measurements in EBR. An example of this counter is shown in Fig. 18. The fission ratio of U^{238} to U^{235} along the vertical axis is shown in Fig. 19, and the radial distribution along the reactor mid-plane is shown in Fig. 20.

Fission ratios of various materials at the core center are shown in Table X.

Table X. Fission Ratios at the Center of the EBR Core

Materials	*Ratio*
U^{238}/U^{235}	0.115
Th^{232}/U^{235}	0.0305
Pu^{239}/U^{235}	1.42
U^{234}/U^{235}	0.625

Of some interest to reactor theorists are measurements of the relative worth, reactivity-wise, of fuel as a function of position in the reactor. This is plotted for a standard fuel element in Fig. 21. At a distance of 3.75 inches from the center of the core, where fuel is worth 74 inhours, an equal volume of normal uranium is worth 27 inhours and steel is worth 24 in-

Table IX. Comparison of Measured Reaction Rate Ratios and Those Calculated from the Spectrum

Ratio	*Hole No. 1 (Center)*		*Hole No. 118 (2.6 in. out)*	
	Calculation from spectrum of Fig. 15	*Experiment*	*Calculation from spectrum of Fig. 16*	*Experiment*
$\frac{U^{235}\text{ fission rate}}{U^{238}\text{ fission rate}}$	7.9 ± 0.9	8.7 ± 0.5	9.7 ± 1.1	9.5 ± 0.5
$\frac{\text{Au activation}}{\text{P activation}}$ in reactor	3.6 ± 0.4	3.7 ± 0.4	5.0 ± 0.7	4.2 ± 0.4
$\frac{\text{Au activation}}{\text{P activation}}$ in fission spectrum				

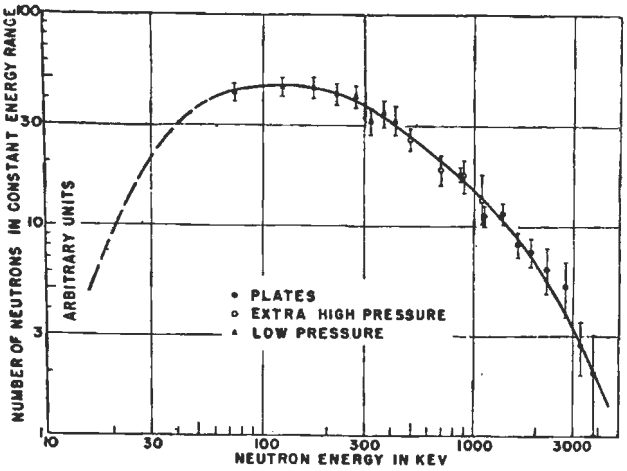


Figure 15. Neutron spectrum from hole #1 (center) of the core of the EBR

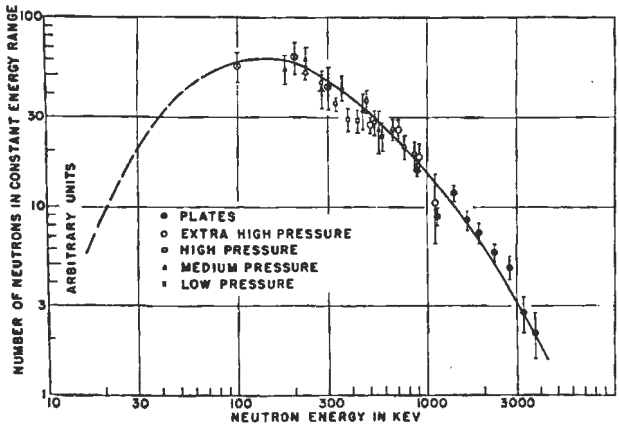


Figure 16. Neutron spectrum from hole #118 (2.6 in. from center) of the core of the EBR

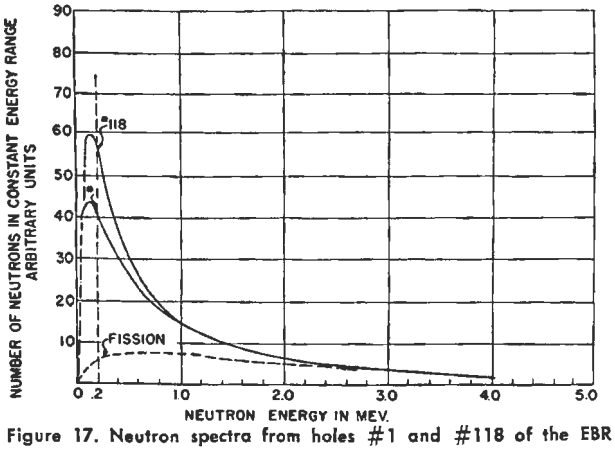


Figure 17. Neutron spectra from holes #1 and #118 of the EBR

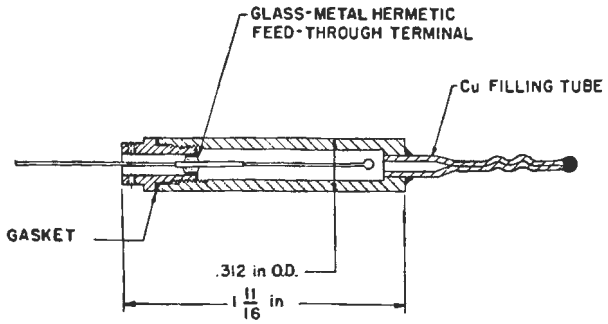


Figure 18. Fission counter used in EBR fission rate measurements

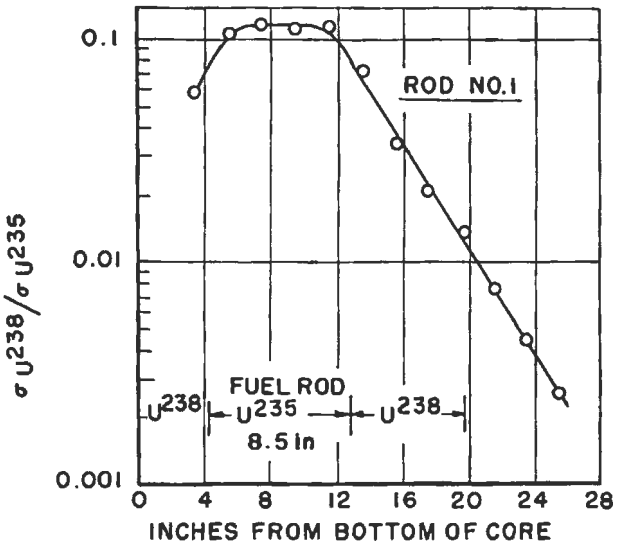


Figure 19. Vertical traverse of U^{238}/U^{235} fission ratio in center of EBR

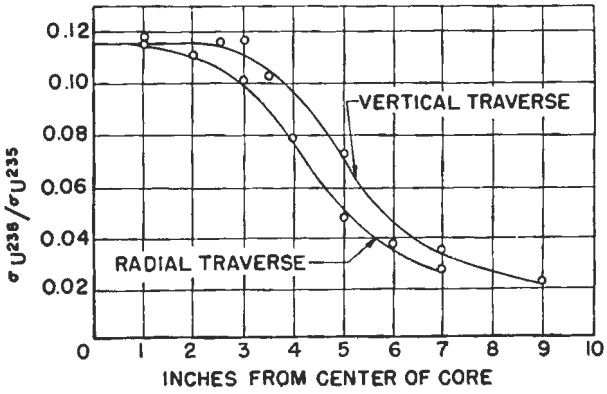


Figure 20. U^{238}/U^{235} fission ratio traverses in the EBR

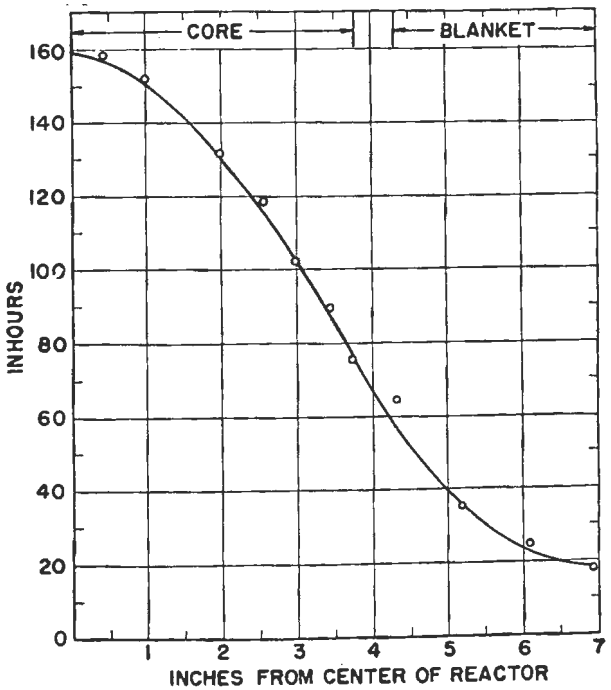


Figure 21. Inhour value of a fuel rod in EBR

hours. In all cases removal of the sample rod resulted in replacing the sample rod by the NaK coolant. It is also of interest to note that removing the entire external blanket and replacing it with a blanket of similar volume made of lead results in almost no change in reactivity.

Many of these measurements are yet in progress and new and better techniques for measuring are still being evolved. It is also planned to load this reactor with plutonium fuel in the near future and to repeat many of these measurements with that loading.

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Objectives and Summary of USAEC Civilian Power Reactor Program

By U. M. Staebler,* USA

Energy requirements of mankind are increasing rapidly as a result of increasing population as well as increasing use of energy per unit of population. An increasing per capita use of energy is a basic component for increased production and higher material living standards. It is only natural, therefore, that the first demonstration of an entirely new source of energy must be heralded as an important event in history. Such was the demonstration of the self sustaining fission chain reaction in December 1942, at the University of Chicago. Since that historic event an enormous amount of effort has been devoted to the development of technology which will permit practical application of energy from the fission process. In the United States this research and development effort has occupied the professional talents of an increasing number of scientists and engineers during the 12½ years since initiation of the first chain reaction.

The main responsibility for administration and direction of this effort has been assigned to the United States Atomic Energy Commission which was organized by the Atomic Energy Act of 1946. The Commission has supported the development of power reactor technology since its organization, building on technology and facilities provided by its predecessor, the Manhattan District of the Army Corps of Engineers. The Division of Reactor Development was organized within the Commission on 1 February 1949, in order to give increased attention to this important field. A plan to build four major reactors as part of the program to bring about practical utilization of power from reactors was announced early in 1949. Brief mention of the projects which finally evolved from this initial program is important to an understanding of our present program.

1. The Experimental Breeder Reactor produced over 100 kw of electrical power in December 1951, and thus was the first nuclear reactor to produce electrical power. This is another very important milestone in our power reactor program. This reactor is still in operation at the National Reactor Testing Station, Arco, Idaho, providing data for design of fast breeder power reactors.

2. The Materials Testing Reactor was placed in operation at the National Reactor Testing Station, Arco, Idaho in May 1952. It has been a valuable tool for testing effects of intense reactor irradiation on various materials.

3. The Homogeneous Reactor Experiment produced over 150 kw of electric power on 24 February 1953, at Oak Ridge, Tennessee. This reactor provided much valuable information on performance characteristics of a circulating fuel power reactor. It was shut down and dismantled early in 1954 to make room for a second homogeneous reactor experiment which will be described as part of our current civilian power reactor program.

4. The Submarine Thermal Reactor first generated substantial amounts of power on 31 May 1953, at the National Reactor Testing Station, Arco, Idaho. This reactor is still in operation and is the source of much of our technology for water moderated and cooled power reactors.

In addition to the reactor projects summarized above as major components of the Commission's original power reactor program, there has been an extensive program of research and development not related to the specific reactor projects. This program ranges from fundamental studies on such problems as radiation damage to engineering component tests. Various other reactors have played an important part in contributing to power reactor technology. These include simple research reactors used as neutron sources for experimental studies. Special mention is due the plutonium fueled, mercury cooled fast neutron reactor for its contribution to new fuel and coolant technology. This reactor was built by the Los Alamos Scientific Laboratory and operated for 3 years but has now been dismantled. Special mention also should be made of boiling reactor experiments conducted by the Argonne National Laboratory at the National Reactor Testing Station starting in the summer of 1953. These are important both for their contribution to evaluation of reactor safety and for their demonstration of the feasibility of boiling water power reactors.

The above is obviously a very brief summary of the background of technology leading to the Commission's current civilian power reactor program. The significance of this background is not in the number of reactors which have produced power or operated under conditions suitable for production of

*U.S. Atomic Energy Commission.

power, nor in the amount of power which has been produced. The significance lies rather in the wide range of technology which has been developed and utilized in these power reactors. The reactors include both homogeneous systems and heterogeneous systems. Water has been used as coolant under both boiling and non-boiling conditions. Several kinds of liquid metals have been used for heat transfer. Both fast neutrons and thermal neutrons have been utilized as the major cause of nuclear fissions. Thus, the framework within which our present civilian power reactor program was formulated was one of clear demonstration of technical feasibility of power production from a wide range of reactor concepts.

PROGRAM OBJECTIVES

The basic objective of the civilian power reactor program of the United States Atomic Energy Commission is to develop the technology of all elements of nuclear power systems to provide the foundation for a nuclear power industry. Since there will be no enduring nuclear power industry unless the technology permits production of power at costs comparable with other methods, this objective requires serious consideration of cost factors.

It will, of course, be very difficult to determine exactly when one has achieved economically competitive nuclear power. In a strict sense the nuclear power plant would be required to operate sufficiently long for normal amortization of the cost of the plant. Our objective is the development of power reactor technology to the point where nuclear power systems are economically feasible or competitive. In this sense an economically competitive nuclear power plant is one which, when built and operated under normal and appropriate industrial financing and operating practices, will produce power at the same or at a lower cost than the best conventional power plant which could have been built at the same location at the same time. Such a nuclear plant would not be subsidized directly or indirectly; it would pay actual costs for materials and services received, and it would receive a real and fair value for any materials produced.

Implementation of a program naturally depends upon certain principles of operation. In the case of the Commission's civilian power reactor program these principles may be stated as follows:

1. Developing economically competitive nuclear power for civilian use will be aided by maximum practical utilization of the financial incentive common to business ventures in order to stimulate ingenuity and imagination and the assumption of calculated technical and economic risks.

2. The Commission's role should be to develop advanced technology primarily at government expense and to stimulate outside groups to undertake developmental or demonstration power reactor projects primarily with non-government financing.

3. The scope of the program should give adequate recognition to a wide range of civilian power reactor sizes and applications for effective contributions to nuclear power both domestically and in other countries.

The reactor development program always has recognized the importance of designing, building, and operating reactors. Such specific reactor projects provide a focus on overall development problems in a reactor system which cannot be achieved in any other manner. Thus, our present civilian power reactor program centers around the building of selected experimental power reactors. These reactor projects continue to be supplemented by extensive research and development in areas not essential to completion of the projects but important to eventual performance of reactors. This general research and development effort also is the source of new reactor concepts which will justify project exploitation in the future.

Based on the objective and principles of operation summarized above one may define three phases of the civilian power reactor development program as follows:

1. The development of basic technology through laboratory work and experimental reactor projects.

2. The construction and operation of demonstration power reactors which, at least initially, also are largely experimental and probably not economically competitive but which are intended principally for commercial power production and could become economically competitive through advancing technology.

3. Industrial power reactors fully justified on an economic basis.

The civilian power reactor program in the United States is advancing rapidly into phase two as outlined above with continued extensive activity in phase one to maintain a broad base of technological development. Some plants now in advanced stages of planning could themselves become economically competitive nuclear power plants.

CIVILIAN POWER REACTOR PROGRAM OF FEBRUARY 1954

The United States Atomic Energy Commission's program for the development of nuclear power plant technology for civilian application was formalized in February 1954. Five reactor concepts were selected as both potentially capable of producing economically competitive nuclear power and sufficiently well advanced in technology at the time to make reactor construction technically feasible. These concepts are (*a*) pressurized water reactors, (*b*) boiling water reactors, (*c*) sodium graphite reactors, (*d*) fast breeder reactors, and (*e*) aqueous homogeneous reactors. It was noted at the time of formalizing the program that other concepts also should be capable of producing economically competitive nuclear power. The technology of some of these, such as the graphite moderated-water cooled, and the low pres-

sure water moderated and cooled was considered sufficiently well advanced that experimental reactors primarily at government expense were not justified. Emphasis in the program was placed in the building of experimental reactors. These reactor projects have been described many times including two reports of the Subcommittee on Research and Development of the Joint Committee on Atomic Energy of the Congress of the United States.*

The projects are summarized briefly in Table I.

It was considered desirable to have one reasonably large scale nuclear power plant in the program in order to obtain certain information on costs and acquire operating experience which could be obtained only with such a large scale plant. The pressurized water reactor project fulfills that need. The other projects are relatively smaller in size but chosen so as to duplicate operating conditions and engineering components which would be utilized in larger sized plants.

The program seeks to provide technology for utilization of both uranium and thorium as nuclear power source materials. The homogeneous reactor and sodium reactor experiments in particular will explore the utilization of thorium and U^{233}. The use of plutonium as a power fuel and the practical breeding of fuel in a fast neutron system will be further investigated with the Experimental Breeder Reactor No. 2.

The individual reactor projects are complementary in many ways. Development results for one project are in many cases of direct interest to other projects. For example, problems of fuel element development and primary coolant system design for heterogeneous water cooled reactors are very similar. Likewise, comparable problems for the liquid metal cooled systems are very similar except for the increased incentive for achieving very high specific powers in the fast breeder reactor design. The primary coolant pumps and other engineering components of the pressurized water system can be used directly or with slight modifications in the aqueous homogeneous system. In some cases similar problems are being approached with different solutions in different projects with resultant benefits of more exhaustive investigation of pertinent technology.

Several of the experimental reactors are being designed with considerable flexibility for investigation of variations in the initial design. For example, the experimental boiling water reactor experiment is planned for experimental operation with heavy water as well as ordinary water. The sodium reactor experiment has removable graphite moderator pieces so that in addition to later experiments with thorium in place of uranium, it may try other moderator materials. The homogeneous reactor experiment is designed having in mind blanket operation on either uranium or thorium as solution or slurry.

The program is intended to cover a wide range of technology so that future power plants can be of designs which will yield minimum cost nuclear power for the particular application desired.

ECONOMIC CONSIDERATIONS ON EXPERIMENTAL REACTOR PROJECTS

None of the experimental power reactors now in design or construction is expected to produce economically competitive nuclear power in the United States. Some of the projects, such as the SRE and the HRE No. 2, do not include facilities for converting the energy to full potential electrical capacity. In all cases the actual capacity of the reactor core for production of energy must be considered subject to errors of as much as a factor of two, or perhaps more, over the design value. The cost of a building and provisions for containment of radioactive materials in case of an accident is a large item of cost in these small capacity plants which would not increase substantially for larger plants. All of these factors make it difficult to predict the cost at which these plants may produce nuclear power. Some estimates of these cost factors are summarized in Table II.

Estimates such as those given in Table II are of little value in assessing the relative economic promise of the various systems because of differences in size and in the effect of experimental features of the particular reactor design. One example of how a real effort to cut costs can yield results is found in the second boiling reactor experiment which Argonne National Laboratory built at the National Reactor Testing Station. It was demonstrated that a plant

*Under titles of "Five Year Power Reactor Development Program Proposed by the Atomic Energy Commission", dated March 1954, and "Current Statement of the Atomic Energy Commission on the Five Year Reactor Development Program", dated 4 May 1955. Available at the United States Government Printing Office, Washington, D. C., U.S.A.

Table I. Summary of Reactor Projects in Civilian Power Reactor Program of US Atomic Energy Commission

Project	*Moderator*	*Coolant*	*Reactor power (Mw)*	*Elec. power (Mw)*	*Status*	*Scheduled completion*
Pressurized water reactor	Water	Water	270	60	Constr.	1957
Experimental boiling water reactor	Water	Water	20	5	Constr.	1956
Sodium reactor experiment	Graphite	Sodium	20	7.5*	Constr.	1956
Homogeneous reactor experiment No. 2	Heavy Water	Heavy Water	5–10	1–2*	Constr.	1956
Experimental breeder reactor No. 2	None	Sodium	62.5	15	Design	1958

*Full generating capacity not included in original project plans.

Table II. Cost Estimates for Experimental Reactors

Reactor	Est. cost incl. bldg. and turbo-generator, $ millions	Electrical kw (Design)	$/kw elec.	With possible improved performance $/kw
Pressurized water reactor	37.5	60,000	620	375
Experimental boiling water reactor	4.0	5000	800	450
Sodium reactor experiment	4.7*	7500*	630	450
Homogeneous reactor experiment No. 2	2.1*	2000*	1050	600
Experimental breeder reactor No. 2	12.5	17,000	740	450

*Cost estimates increased to provide full generating capacity although not included in original project.

with a capacity of a few thousand kw can be built for approximately $200 per installed kw.

EXPERIMENTS TO FULL-SCALE REACTORS

The Commission's civilian power reactor program of February 1954, was planned with the assumption that, as the goal of economically competitive nuclear power is approached, industry would accept an increasing share of the responsibility for this development and would invest money in power demonstration reactor plants. As this paper is written in May 1955, this assumption appears to have been justified. The Consolidated Edison Company of New York has requested a license to build a large-scale nuclear power plant. Proposals have been received from four other organizations to build large-scale nuclear power plants under the Commission's power demonstration reactor program.

The experimental reactor projects were selected to cover a wide range of technology. Design specifications were chosen to duplicate expected operating conditions and engineering components of large-scale plants. The transition from experimental reactors to power demonstration reactors appears well under way even before the experimental units have been completed. The importance of the experimental units to the larger plants and the success to which they are expected to pioneer the conditions of the larger plants may be shown by a table of comparative specifications. This is done in Table III.

While it cannot be said with certainty as this is written that all of the power reactors listed in Table III will be built, they represent plans and proposals now under serious consideration. These, plus the various experimental power reactors now in operation or under construction would represent a total nuclear electric power capacity in the United States considerably in excess of 700,000 kw in 1960.

Table III. Comparative Specifications of Experimental and Power Demonstration Reactors

Reactor	Reactor power, kw	Net elec. power, kw	Specific power, kw/kg-fm	Max. heat flux, BTU/ft²/hr	Outlet coolant temp, °F
Pressurized water reactor	270,000	60,000	1000	380,000	542
Consolidated Edison Co. of New York	500,000	140,000*	2170	500,000	500
Yankee Atomic Electric Co.‡	500,000	100,000	740	375,000	422
Experimental Boiling Water Reactor	20,000	5000	420	130,000	488 (steam)
Nuclear Power Group§	692,000	180,000	3000	425,000	480 (steam)
Sodium Reactor Experiment	20,000	7500†	350	420,000	960
Consumers Public Power District of Nebraska	250,000	75,000	565	390,000	925
Experimental Breeder Reactor No. 2	60,000	17,000	290	1,000,000	1000
Detroit Edison Co. and Associates¶	300,000	91,000	580	850,000	800
Homogeneous reactor experiment No. 2	5000 to 10,000	1000 to 2000†	1770 to 3540	—	572
Oak Ridge National Laboratory design for power breeder station	444,000	100,000	12,000	—	572

*Will be increased to 250,000 kw by superheating.

†Total electric generating capacity not included in plant design.

‡Composed of American Gas and Electric Service Corporation, New York; Bechtel Corp., San Francisco, Calif.; Commonwealth Edison Company, Chicago, Ill.; Pacific Gas and Electric Company, San Francisco, Calif.; and Union Electric Company, St. Louis, Missouri.

§Central Hudson Gas and Electric Corporation, Poughkeepsie, New York; Cincinnati Gas and Electric Corporation, Cincinnati, Ohio; Consumers Power Company, West Jackson, Michigan; Delaware Power and Light Company, Wilmington, Delaware; Detroit Edison Company, Detroit, Michigan; Long Island Lighting Company, Mineola, New York; Philadelphia Electric Company, Philadelphia, Pa.; Rochester Gas and Electric Corp., Rochester, New York; Toledo Edison Company, Toledo, Ohio.

¶Members of which are New England Power Company, Boston, Mass.; Connecticut Light and Power Company, Hartford, Conn.; Boston Edison Company, Boston, Mass.; Central Maine Power Company, Augusta, Maine; Hartford Electric Light Company, Hartford, Conn.; Connecticut Power Company, Hartford, Conn.; Western Massachusetts Electric Company, Greenfield, Mass.; Public Service Company of New Hampshire, Manchester, N. H.; Montaup Electric Company, Fall River, Mass.; New Bedford Gas and Edison Light Company, New Bedford, Mass.; Cambridge Electric Light Company, Cambridge, Mass.; Central Vermont Public Service Corp., Rutland, Vermont.

SUPPORTING RESEARCH AND DEVELOPMENT

The cost of power is composed of plant costs, operating costs, and fuel costs. For nuclear power the objective is to achieve the potentially low fuel costs without excessive increases in plant or operating costs. The experimental reactor projects will give much information on plant performance and operating problems. They also will provide valuable data on plant costs. They will serve as important irradiation test facilities to gain experience and data on fuel element behavior in actual power reactor conditions. Important as it is, the reactor is only part of a nuclear power system. Overall economics of the system depends upon other factors such as fuel preparation and fuel processing. Two of the experimental reactor projects include plans for fuel processing integrated with the reactor. These are the Homogeneous Reactor Experiment No. 2 and the Experimental Breeder Reactor No. 2. Extensive large scale component development work is being done on water systems, various liquid metal systems and homogeneous fuel systems. Of special note in connection with development of aqueous homogeneous reactors is the Los Alamos Power Reactor Experiment under final stages of construction at the Los Alamos Scientific Laboratory. It utilizes phosphoric acid fuel solution rather than the sulfuric acid of the Oak Ridge project and is designed to produce 2000 kw of reactor heat.

There is a continuing search for new materials, fabrication techniques, and for improved understanding of reactor physics and irradiation phenomena. New fuel systems, reactor coolants, and moderator materials are constantly being evaluated in terms of specific reactor concepts and the potential economics of that design in specific applications. The value of reactors for civilian uses other than production of electrical power is given continuous attention.

CONCLUSION

The civilian power reactor program in the United States is advancing on a wide front. It is focussed on building experimental reactors to exploit and extend promising technology with emphasis on factors which will reduce the cost of power from future nuclear power plants. The transition to a large scale nuclear power industry under industrial practices normal to the United States already has begun. An extensive research and development effort will continue to advance the frontiers of knowledge in the field of civilian applications of nuclear power. This effort recognizes the importance of early achievement of economically competitive nuclear power and provides a sound basis for effective assistance to other countries in the nuclear power field. The Atomic Energy Act of 1954 contains provisions for such assistance to other countries. The United States Atomic Energy Commission intends to make maximum use of those provisions.

Record of Proceedings of Session 14A

MONDAY AFTERNOON, 15 AUGUST 1955

Chairman: Mr. G. Randers (United Nations)
Vice-Chairman: Mr. J. V. Dunworth (United Kingdom)
Scientific Secretaries: Messrs. M. Trocheris and J. Goedkoop

PROGRAMME

P/499 The sodium reactor experiment ..... W. E. Parkins

DISCUSSION

P/406 The graphite-moderated, gas-cooled pile and its place in power production ..... C. Hinton

DISCUSSION

P/501 The engineering design of EBR-II, a prototype fast neutron reactor power plant ..... A. H. Barnes *et al.*

P/813 Operating experience and experimental results obtained from an NaK-cooled fast reactor ... H. V. Lichtenberger *et al.*

Mr. PARKINS (USA) presented P/499.

DISCUSSION ON P/499

The CHAIRMAN: Mr. Parkins' paper is now open for discussion. There is one question here from Mr. Wilhelmsen of Norway.

Mr. WILHELMSEN (Norway): Would the author care to comment on what is the melting point for the sodium potassium bond around the fuel elements; would thermocycling of this bond render the design unsuitable if frequent shut-downs had to be considered? What is the burn-up you expect before reprocessing the fuel elements, and what would be the duration of the start-up and shut-down periods?

Mr. PARKINS (USA): The sodium potassium alloy we are using is the eutectic. I do not remember the melting point of that, but it is below the freezing point of water. We could use a different alloy, but we happen to be using the eutectic. I do not see any particular difficulty with the bond material in so far as the cycling of the fuel elements is concerned. That is one advantage of a liquid-metal bond of this type. As far as the burn-up is concerned, we do not know what to expect. That is one of the main reasons for this program; the fuel element development is one of the primary objectives of the SRE program. I am afraid I cannot answer you as to the duration of start-up and shut-down periods. We expect that the experimental program, and the many different experiments we wish to carry out with this facility, will cause a great many start-ups and shut-downs, and we have not at this time laid down any particular operating schedule.

The CHAIRMAN: The next question is from Mr. Preece of the United Kingdom.

Mr. PREECE (UK): I note with interest that you have decided to use centrifugal-type pumps with mechanical drive rather than electromagnetic type pumps. Would you like to give us the reasons for making this decision?

Mr. PARKINS (USA): The primary reason was the cost factor. The purpose of this program, of course, is to develop a reactor suitable for central station power use. At the present time the electromagnetic pumps are an order of magnitude more expensive than mechanical pumps. There are also other difficulties associated with their use which we would like to avoid. We admit that there are difficulties in using these mechanical pumps, especially in connection with the frozen seal, but we felt that, all factors considered, there was more promise in the development of this type of pump than the electromagnetic pump.

Mr. PREECE (UK): My second question is, do you consider that the sodium-graphite type of reactor could be developed for use with gas turbine cycles? I have in mind the possible development with smaller units of this same basic type.

Mr. PARKINS (USA): I believe it could. At the present time the biggest limitation would be the

maximum temperature obtained. As I mentioned, we are expecting to obtain 960°F primary outlet temperature in the SRE. The secondary system operates at about 60° lower as now planned; so the maximum temperature there would be 900°F. This would not be very attractive for a gas turbine. Actually one of the primary purposes of this program is to try and push up the temperature. We have designed the reactor and the sodium system so that the parts are stressed for operation at 1200°F. We should like to be able to push up to that temperature. There are difficulties in going that high in temperature, one of them being the strength properties of the zirconium in that region, but with future developments and attaining such temperatures, I think the gas turbine cycle would become a possibility.

The CHAIRMAN: The next question is from Mr. Hall of the United Kingdom.

Mr. HALL (UK): Would the author care to comment on the possibility of the accumulation of gaseous fission products in the sodium potassium bond inside the fuel element, and the possible overheating of the fuel?

Mr. PARKINS (USA): I did not mention it, but the sodium potassium alloy extends up a few inches above the column of slugs. Above that is a helium atmosphere which is sealed within the fuel rod. The purpose of that helium is to permit expansion of the alloy, and also to provide some space for the accumulation of the fission products which may diffuse out into that region. We do not know how much to expect of this, but we feel that the amount of diffusion of the fission products will be sufficiently small so that there will not be an excessive build-up of pressure within the fuel-rod jacket.

The CHAIRMAN: The next question is from Mr. Shepherd of the United Kingdom.

Mr. SHEPHERD (UK): I should like to ask if you have any experience with the use of toluene in radiation fields. You mention it admittedly in not very strong radiation zones, but nevertheless I should be interested to know whether you have any experience with the use of toluene or other hydro-carbons in this connection.

Mr. PARKINS (USA): Actually, I think one could satisfactorily use water in all these locations where we are using toluene. This has been perhaps an extra cautious approach to the hazard problem. We have conducted some experiments on the radiation decomposition of toluene. It does, of course, decompose, but the amounts of it in the particular places where we are using it are sufficiently small and there appears to be no problem at all. I cannot give you the figures on the decomposition of toluene, but I believe we have unclassified reports which give those, and perhaps I could obtain them for you. I should mention one other factor here. While I have said "toluene" throughout, we have in the last couple of weeks almost decided to go to a different hydrocarbon called tetralin. The reason is the difference in toxicity in handling it and also the higher flash point. There are hazards associated with using toluene from the standpoint of inflammability and toxicity. The tetralin has a higher boiling point, a higher flash point, and lower toxicity, not in gram-per-gram quantities but because the vapor pressure at any temperature is somewhat lower. Again it is satisfactory from the cost standpoint and from other heat transfer standpoints.

The CHAIRMAN: There is one question further from Mr. Moore of the United Kingdom.

Mr. MOORE (UK): I should like to ask the author two questions. In the first place, would he give the reasons for the construction of the plant below ground level. Are they related to safety, or are they simply of an economic nature? The second question I should like to ask is somewhat dissimilar. Could he give the degree of the enrichment of the fuel and the conversion factor he expects initially on start-up of the plant?

Mr. PARKINS (USA): The reactor is below ground primarily as a safety measure, the point being that we wanted the reactor tank to be the lowest point in the sodium system so that a break nowhere in the system would result in draining sodium from the reactor core. There are also other factors which make it attractive to put it below ground, that is in connection with the shielding. We are providing no entry to the core tank below the pipes which come in above the graphite assemblies. All of the elements which actually go into the core hang from the top shield, so that there is no need really to provide access around the sides, and it turns out to be an economical thing to do to place it below ground as well as a good safety measure.

On your second question, I forgot to mention the enrichment of the fuel; it is 2.8 atom per cent U^{235}. I do not actually know the conversion factor. In this geometry it is rather low because it is a small reactor for a graphite-moderated type. I believe it is down around 0.7 or something like that.

The CHAIRMAN: Are there any more questions on this paper? If not, may I just ask one question in connection with what the last speaker said. Was the enrichment of 2.8 per cent chosen merely because of reactivity considerations, or from the economic standpoint?

Mr. PARKINS (USA): It was based entirely on reactivity. The core was sized for the removal of 20,000 kilowatts of heat at the 1200°F maximum fuel temperature, and with the fuel geometry that I have described we then calculated an enrichment which would be necessary with the spacing we are using of 11 inches center-to-center and with the various amounts of poison material, the zirconium, the steel and the sodium, within the core.

Sir Christopher HINTON (UK) presented paper P/406 and added the following comments:

The first graphite-moderated, gas-cooled reactors which are being built at present in Britain may have shortcomings. However, they have the advantage that they are inherently safe and they can generate electricity in large quantities on a guaranteed program and at prices which are comparable with those of modern conventional power stations.

They cannot stand on their own, and they are competitive only when there is a market for the by-product plutonium. That market must and can be found by recycling or by use of the by-product plutonium in advanced types of reactors, culminating in the fast fission reactor. But, considered in that light, the Calder Hall type of reactor has, I am quite certain, a considerable future.

One has to remember, in considering nuclear reactors, that the cost of power produced consists, in round figures, of something like two-thirds arising from capital charges and about one-third arising from fuel charges. The position in a conventional power station is approximately the other way round: approximately one-third of the cost of power arises from the capital charges, and about two-thirds from the fuel charges.

It is common experience that the cost of prime movers falls with the passage of time and with the growth of techniques. Calculating as nearly as possible—and the calculation into terms of real money value is always extremely difficult—into terms of today's money values, you will find that in 1920, if you wanted to buy a land-based oil engine, it cost you something of the order of £50 per horsepower. Today, you can buy a land-based oil engine at something like £10 per horsepower.

An even more striking illustration is provided by the way in which the cost of steam power stations has come down. You will see from the curve that, if you were a Cornish mine-owner wanting to buy a power station from Bolton and Watt in about 1780, you would have paid something of the order of £1000 a horsepower for it. Today, as you know, you can buy steam power stations at something of the order of £70 a horsepower. That is the way in which capital costs have come down, first of all in oil engines and secondly in steam power stations.

If you look at other forms of prime mover, you will find that the pattern of capital cost reduction is just about the same, and I am quite certain that in nuclear power that same pattern will be followed. First of all, we shall get our reductions in capital cost arising from higher ratings in our graphite-moderated, gas-cooled reactors; savings arising from our ability to build better heat-exchange surfaces; savings arising from our ability to achieve higher fuel element operating temperatures; and, after that, we shall follow on, as has happened in the case of the steam engine, to other types of reactors which give higher ratings and which will enable downward trend of capital costs to be continued.

But that reduction in capital cost at present is clearly of the greatest importance because, as I have already pointed out to you, the unit cost of electrical power generated in an atomic power station consists of capital charges to the extent of two-thirds, and therefore the reductions in capital charges which I am sure we can expect in nuclear power stations are going to bring the cost of nuclear power progressively downward.

But do let us be clear on one point: These stations which Bolton and Watt sold in the 1780's were sold to hard-headed Cornish mine-owners who, if they could have done so, would have been very pleased to buy their power stations for £100 a horsepower instead of £1000 a horsepower. However, if they had tried to do so at that time, they would have succeeded in nothing except landing themselves in very extensive trouble. The old slow-speed reciprocating steam engine that they were sold then represented the best engine that lay within the compass of the engineers as they understood engineering design at that time and as they could use the metallurgical and other materials which were available to them.

The downward trend of that curve had to be achieved, with the passage of time, by a patient progress in which the optimum movement in the direction of advanced technologies and reduced costs was achieved. Let us remember that the first movement onward from the Bolton and Watt engine was really made by Trevithick when he built his high-pressure steam engine on the Thames, with its cast iron boiler which blew up, killed eight men, nearly ruined him, and set back the development of the steam engine by a great many years.

We must make certain that we do not do that sort of thing. We must be certain that our movement in the direction of reducing capital cost and improving performance is made commensurate with our knowledge of materials and techniques which are available to us. We must, I suggest, be most careful that we do not on any account overstep the reasonable rate of progress and move forward into designs which cannot be fully supported by the technologies which are available to us.

It is in that respect that I suggest that the Calder Hall type of pile is so very sound, in that it uses technologies which are well established or well understood and which we can consider to be reliable.

Similarly, let us remember that even though we act as I suggest we should—with the greatest amount of caution—we will still find that the design, construction and operation of nuclear power stations is in an advanced engineering field. I think we ought to realize that for that reason the initial use of nuclear reactors ought to be made in those countries where engineering and other technologies, where industry as a whole, are reasonably well advanced. I think we should be misleading ourselves and I think

we should be doing a bad service to countries whose industries are not well advanced if we suggested to them that immediately nuclear reactors are going to be more serviceable than the more conventional types of prime movers, such as oil engines or steam engines.

It is reasonable to think of nuclear reactors in an ascending scale of specific heat-rating of their fuel elements. At the bottom of that scale you have the graphite-moderated gas-cooled reactor of which we have been talking. It is essentially a heavy land-based reactor, using low specific ratings, but it is economical because it does not need to make use of expensive and exotic materials of construction. It can, if you like, be regarded as the slow-speed reciprocating engine of the reactor world—reliable, almost conventional in design. At the other end of your reactor scale, you have the fast fission reactor —much more highly rated—using expensive and rare materials of construction. High ratings are necessary there in order to make that reactor economical, because the fissionable material which it uses as a fuel is extremely expensive. It can perhaps be regarded as the gas turbine of the nuclear reactor world. In between those two extremes, there are dozens of types of reactor. Which of those reactors will at future dates be found to be economical is more than I would attempt to predict today.

We in Great Britain are attacking the two ends of the scale. We have designed a graphite-moderated, gas-cooled reactor which will soon be producing large quantities of industrial power at Calder Hall. We are building the fast reactors at Dounreay. In between those two extremes we are applying our resources at those intermediate points which appear to offer the greatest promise. The development of highly rated reactors is needed in order to burn the by-product plutonium produced in the reactors of conservative rating which are built as the first step in our industrial program.

It would, however, be a great mistake to assume that these conservatively designed reactors of the graphite-moderated, gas-cooled type have only a limited future. With development, the reciprocating steam engine held its own over a period of nearly 200 years as an economical prime mover, and the simplicity and reliability of a gas-cooled reactor will entitle it to a similar position in the history of nuclear power.

DISCUSSION ON PAPER P/406

M. Taranger (France): I want to ask three questions: (1) What are the two pressures in the steam circuit compared to the pressure of carbon dioxide? (2) Is the carbon dioxide the commercial grade, or must impurities be carefully removed? (3) Would the speaker comment on the choice of vertical channels, particularly from the point of view of the fuel behavior and from that of the behavior of the pressure cyclinder.

Sir Christopher Hinton (UK): I should like to answer the second and third parts of the question first.

The carbon dioxide which is going to be used is the ordinary carbon dioxide that one gets from the solid carbon dioxide which is on the market and which one uses in a converter to gasify it.

The use of vertical channels is, of course, an extremely interesting point in design. There are a number of advantages to using horizontal channels. I would say straightaway that the overwhelming reason for using vertical channels is the problem of graphite support. Graphite, after all, is subject to thermal expansion and contraction. The channels must remain straight. Clearly, if one uses vertical channels, one can corset the graphite blocks in such a way that the expansion does not materially alter the straightness of the channel. If one attempts to use any other arrangement, such as horizontal channels, one finds that one has not nearly so elegant a solution of the graphite support problem available. That, as I have said, is the overwhelming reason for the use of the vertical channel arrangement. The use of horizontal channels does have advantages. With horizontal channels it is easier to support the fuel elements. I think, however, that the vertical vessel arrangement does give one an easier support of the pressure vessel. It produces less restraint on the vessel and enables the vessel to move more freely with the pressure and temperature strains, so that there are fewer secondary stresses.

The gas pressure is 100 psig, the steam pressure in the primary circuit was 200 psig and in the secondary circuit 50 psig.

Mr. Horowitz (France): Will the speaker please give some specifications for the fuel elements used at Calder Hall together with the maximum specific power in kilowatts per kilogram which he thinks will be reached?

Sir Christopher Hinton (UK): The fuel elements are of the finned type and are enclosed in light metal cans. I would say quite frankly that I think it reasonable to maintain a measure of what I would call industrial secrecy about fuel element design; this is the sort of thing about which one feels one may perhaps not tell one's competitors in detail.

The specific rating, in round terms, in these reactors is likely to be about 2.5 megawatts per ton, upwards.

Mr. Hering (France): My first question relates to the exhaust temperature of carbon dioxide. My second question is this: Is there no fear of the formation of carbon monoxide, and of its decomposition in the coolers? My third question concerns the elimination of the air originally present in the gas circuit.

Sir Christopher Hinton (UK): The outlet temperature of the gas is of the order of 320°–370°C.

The question of the formation of carbon monoxide —that is, of the equilibrium reaction between the

graphite and the carbon dioxide coolant—is one which has been extremely carefully studied. We are quite sure that the equilibrium condition lies close to the carbon dioxide and that the quantity of carbon monoxide formed is of no importance. It is an interesting question because, of course, it depends not on what would happen normally but on what happens under conditions of neutron irradiation. We are, however, quite satisfied on that point.

With regard to the purification of the carbon dioxide, once again we feel that no special action will be necessary.

Mr. BRUNI (Italy): What are the maximum temperature and pressure which can be used to produce energy by means of gas turbines?

Sir Christopher HINTON (UK): I suppose that if one wanted to use a gas turbine—and I am not a gas turbine expert—one would need to get up into the temperature range between 550° and 600°C; that is, beyond the temperatures which we can immediately expect to achieve in reactors of this type. In fact, what I am saying is that, initially at any rate, we feel that the correct thermodynamic cycle to use with the graphite-moderated, gas-cooled reactor is a steam cycle rather than a gas cycle.

Mr. DOAN (USA): I should like to ask what shielding is provided for the external heat exchangers. In the event of a fuel element failure, is it anticipated that any significant amount of carry-over of fission product activity into the heat exchangers will occur?

Sir Christopher HINTON (UK): The answer to the first part of the question is that there is no need for shielding around the heat exchangers. The answer to the second part of the question is that we are quite satisfied that we can prevent carry-over of any activity from the reacting core into the heat exchangers which might deposit on the heat exchange surfaces there.

Mr. SWARTOUT (USA): Sir Christopher has quoted the value of 6/10th pence per kilowatt hour for the power cost. In the paper, it is stated rather definitely, quoting the White Paper, that electrical power can be generated at a cost of 6/10th pence per unit sent out. Further, it is stated that this will be done by using techniques which are presently known. For clarification, I should like to be sure of what the term "sent out" means. Does it mean net electricity after providing the reactor and all auxiliaries with power? Furthermore, does the 6/10ths pence per kilowatt hour apply to Calder Hall as now designed at the power now specified for it?

Sir Christopher HINTON (UK): The answer to the first part of the question is that "units sent out" means units exported from the power station. All the power needed for the ancillaries, the blowers, the lighting, the feed pumps, the internal station usage, and so forth, has been taken out. This is the price for the electricity that is sold.

As to whether that price applies to Calder Hall, the answer is "no." That price is given for the White Paper stations; that is, stations built to produce electricity as a primary product with plutonium as a secondary product. The Calder Hall reactors are dual purpose reactors. They were not optimized for the production of electricity. The price is given for a station using the Calder Hall techniques, using the main design parameters employed at Calder Hall, but optimized for power production.

Mr. BENEDICT (USA): To elaborate further on the basis for computing the cost of power, could you tell us what burn-up of fuel was used in estimating the 6/10th pence per kilowatt hour figure?

Sir Christopher HINTON (UK): I think it is given in the White Paper: 3000 megawatt days per ton.

Mr. LICHTENBERGER (USA) presented paper P/813. As regards the fuel elements, (P/803) he remarked that the uranium slugs were bonded to the stainless steel tube, for better heat transfer, by means of sodium potassium alloy.

Mr. KOCH (USA) presented paper P/501.

DISCUSSION ON PAPERS P/813 AND P/501

Mr. KAZATCHKOVSKY (USSR): I wish to ask a question about the EBR-II reactor. Have there been cases of the fuel sub-assemblies breaking down and what method is used for recording such breakdowns?

A second question: Has the state of the fuel material been investigated after the reactor has been in operation for more than three years? Has the discharge of gas from the "hot" sections into the gas collector space been investigated?

Mr. LICHTENBERGER (USA): We have never had a fuel element stuck so that we could not remove it from the reactor. With respect to the problem of a burst fuel element, we do not have a monitoring system for detecting breaks in fuel elements. This is a somewhat difficult problem when one is using a circulating coolant which is somewhat radioactive to begin with. We have not had any problems with leakage of gas from the fuel elements.

Sr. John COCKCROFT (UK): Could the failure of coolant in EBR-II lead to melt out of the core and severe criticality conditions?

Mr. KOCH (USA): We have taken all of the precautions we could think of to avoid the loss of coolant. That is the primary motivation for adopting a primary system arrangement in which the entire system is contained in a single vessel. We believe that we can build into this unit the highest degree of integrity, preventing the loss of sodium, as contrasted to a system involving external loops with multiple piping connections and so forth in which the failure of a pipe or a system component could conceivably

permit the tank to drain. I agree with your statement that the loss of sodium would be a serious matter, and it is to be prevented at all costs. I tried to describe an approach which we have adopted in an effort to avoid such a circumstance.

Mr. HERING (France): My question relates to the measures which must be taken to prevent sodium leaking into the water, or water leaking into the sodium, in the sodium heat exchanger.

Mr. KOCH (USA): I believe you are referring to the steam generator. The steam generator design is not as yet completely established. However, we are planning to incorporate a unit using double wall construction of the boiler tubes with a detection system to detect leakage of either fluid. In addition, this is the primary purpose for utilizing a secondary sodium system which eliminates the possibility of radioactive sodium coming in contact with water and thus minimizing the seriousness of such a probability to a straightforward chemical accident rather than a chemical accident involving radioactivity in addition. I might comment that there are many designs of sodium-to-water heat exchangers or boilers, if you will, being advanced by industrial organizations of many types. It now appears as though we are reaching an era, at least in this part of the business, where we have somewhat of a choice in selecting this kind of equipment.

Mr. DALTON (Australia): Could Mr. Koch tell us what is the effect of the internal blanket on the neutron spectrum of the EBR-II as opposed to a single core system?

Mr. KOCH (USA): I should say, first, that the comments I will make are based on analysis, not on experiment; we have not as yet conducted critical experiments on these geometries.

The use of the central blanket softens the spectrum in the core somewhat—in other words, lowers the spectrum, as contrasted to a completely uniform core.

Mr. DALTON (Australia): Have you in fact checked on the effect on your breeding gain through putting the blanket internally?

Mr. KOCH (USA): Are you directing that question toward the effect of breeding in the central blanket as such? Are you asking whether you get more breeding. . .

Mr. DALTON (Australia): No, the effect on the softening of the spectrum, on your breeding gain, as opposed to the gain you get through having your blanket in the high flux system.

Mr. KOCH (USA): I think the differences here are so small that they are within the accuracy to which we know the cross sections, and also to the effect on spectrum of this rather small additional amount of uranium. As to breeding gain, we just do not know the effect.

Mr. JACKSON (UK): You have apparently only one mechanical type of control and shut-off mechanism. Have you any comment on the philosophy or the desirability of having two different types of mechanism, so that the thing that may inadvertently happen to one cannot prejudice the operation of another?

Mr. KOCH (USA): If I understand your question correctly, you are referring to the fact that we have 12 identical control units. I did not make this point clear—but, first of all, they are completely independently driven, so that there is no interdependence of units, nor are they in any way interconnected in multiples. We are very partial to the use of moving fuel as opposed to introduction of a poison, primarily because of consideration of neutron economy. I did not mention it, but we are also seriously considering the use of a second shut-down type of rod, which could very well be a poison, which would be used only for shut-down purposes but not for operation.

Mr. SHEPHERD (UK): I should like to ask Mr. Koch what degree of burn-up they are aiming at in the fuel elements—I mean, the over-all burn-up.

Mr. KOCH (USA): This is an experimental reactor; so we are going to try to get all we can. We have used 2 per cent burn-up of total atoms as a design basis. We do not expect to reach this at an early stage, and we expect that we will have to work up to it. We should like very much to surpass that number, of course. Two per cent burn-up in this reactor represents about a 120-day cycle.

Nuclear Reactor Catalog

Nuclear Reactor Catalog

Prepared for the United Nations by Herbert S. Isbin,* USA

Concise tabulations of unclassified reactor features were published in 1952 as the Nuclear Reactor Catalog[1] and a supplement was issued in 1953. These tabulations reflected the world-wide interest in nuclear reactors, and illustrated the diverse reactor conceptions and technologies. Another revision of the Catalog has been prepared to take into account the impressive array of information released for the In-

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Table 1. Thermal, Heterogeneous, Graphite

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
CP-1, Uranium Graphite Pile (West Stands, Chicago, USA)	200 watts (max.)	6 tons of natural uranium, 40 tons of natural uranium oxide (UO_2 and U_3O_8) (See CP-2)	1 ft graphite
CP-2 (Palos Park, Illinois, USA)	200 watts (up to 2 kw, 3×10^8)	3200 uranium metal lumps, 2¼ in. dia.; 14,500 uranium oxide lumps, 2¼ in. dia.; about 50 tons uranium; lumps spaced on 8¼ in. square lattice; fuel core 18 ft wide, 20 ft long, 19 ft high	1 ft graphite
GLEEP, Graphite Low-Energy Experimental Pile (Harwell, England)	100 kw; 3×10^{10}	12 tons of uranium bars, 0.9 in. dia., 12 in. long, sprayed with 0.003 in. Al; 21 tons of uranium oxide in Al cans 1.6 in. dia., 2 in. long; line lattice with 7¼ in. pitch; fuel core 5.24 m long, 2.86 m radius; metal in central portions up to 1¾ m radius	2 ft graphite in shape of octagon
OAK RIDGE (ORNL Reactor, X-10 Pile) (Oak Ridge National Laboratory, Oak Ridge, Tenn., USA)	Increased from 1000 kw to 3800 kw, 10^{12}	Uranium fuel slugs in 0.035-in. thick 2S Al jackets, 1.1 in. dia., 4 in. long, 2.57 lb per slug; 39 to 54 slugs per channel; 1248 fuel channels on 8 in. rectangular lattice, 821 used (30 tons req'd for criticality, 54 tons used); diamond-shaped channel, 1¾ in. square; fuel core about 20 ft cube	2 ft graphite
BEPO, British Experimental Pile (Harwell, England)	Increased from 4 Mw to 6 Mw; central thermal flux $\sim 1.4 \times 10^{12}$	0.9-in. dia., 12-in. long uranium bars, encased in Al; 28 tons for criticality, 40 tons for full load; 900 central channels used, 20 bars per channel, 7¼ in. between channels; cross section of empty channel is 3½ in^2.; fuel core 10 ft radius, 20 ft long	About 3 ft graphite
Brookhaven (Brookhaven National Laboratory, Upton, N.Y., USA)	30 Mw; 4×10^{12}	Al-clad uranium metal rods placed on 8-in. centers; 60 tons of uranium; 1369 circular fuel channels circular cross section of 36 cm^2; graphite moderator in two right rectangular prisms, $12\frac{1}{2} \times 25 \times 25$ ft, separated by vertical opening for cooling air	4.5 ft graphite

ternational Conference on the Peaceful Uses of Atomic Energy. Many, but not all, of the reactor papers were available for review and inclusion in the Nuclear Reactor Catalog.

Tables 1 through 5 represent a catalog of nuclear reactors placed in operation, and Tables 6 and 7 summarize reactor developments.[2] The following items are included in Tables 1 through 5: column 1, reactor name or classification, and location; column 2, power level (heat) and flux in neutrons/cm²-sec, designating average thermal flux if not otherwise noted; column 3, description of fuel, moderator, and fuel core arrangement; column 4, reflector; column 5, shielding; column 6, over-all size; column 7, control methods; column 8, coolant; and column 9, additional notes, including startup date in parenthesis.

REFERENCES

1. Isbin, H. S., "Nuclear Reactor Catalog," Nucleonics, Vol. 10, No. 3, pp. 10-16, 1952; *ibid.*, Vol. 11, No. 6, pp. 65-69, 1953.
2. The material is based upon a catalog being prepared for the "Nuclear Reactor Handbook" (McGraw-Hill Book Co., Inc., New York).

Moderated, Natural Uranium Reactors

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
None	24½ × 24½ × 19 ft high; 385 tons of graphite	10 rods	None	World's first reactor. Operated 2 Dec. 1942. After initial operation, assembly dismantled to form basis for CP-2
5-ft concrete walls; 6 in. lead; and 4 ft wood on top	30 ft wide, 32 ft long, 24½ ft high; 472 tons graphite; total weight about 1400 tons	Bronze strips covered with Cd; 2 regulating, 1 shim, 3 safety	None	Operations started 20 March 1943; in service until May, 1954
Cubical concrete shield with 5 ft thick walls	Large cube containing 505 long tons graphite	Rods containing Cd, 4 coarse control, 1 fine control and 6 safety	Air, at least 5000 ft³/min at subatmospheric pressure	$k_{ex} = 0.002$; max. uranium cartridge temp., 60°C (1947)
7 ft concrete	47 ft long, 38 ft wide, 32 ft high; 620 tons graphite	4 safety (each 0.003 k), 2 shim (each 0.007 k), 2 regulating (each 0.005 k); back-up safety consists of boron steel shot. Vertical rods 3½-in. square, 8 ft long, contain layers of Cd encased in steel; horizontal steel rods 1¾ in. square, 200 in. long, contain 1½% boron	Air, 120,000 ft³/min drawn through reactor, filtered, and discharged through 200 ft stack	Average highest fuel temp. ~ 270°C, average moderator temp. 135°C. An Al-silicon eutectic is used to bond Al jacket to uranium metal (1943)
6-in. cast iron plate, 6 ft (barytes aggregate, $\rho = 3.5$) concrete on sides and 7½ in. on top; 600 tons steel, 3000 tons concrete	Cubical core of 8 tons of graphite	4 horizontal and 10 vertical boron-carbide filled hollow steel rods, 2 in. dia.	Air; 260,000 ft³/min, filtered, and discharged from 200 ft stack; stopping power of filters down to 5 μ	Fuel elements operate at temp. of 220°C; exit air temp. 80°C; about 1.2 Mw of heat recovered in a hot-water system for heating a building. (First practical utilization of heat.) (1948)
6-in. iron plate, 4¼ ft heavy concrete, 3-in. iron plate	38 × 55 × 30 ft high; 20,000 tons (including foundation), 700 tons of graphite	Horizontal rods containing boron, entering from two adjacent corners in two arrays of 8 rods each; back-up safeties consisting of boron-steel shot to 4 tubes in graphite, trichlorobenzene to eleven tubes	Temperature ranges from Air, 300,000 ft³/min at subatmospheric pressure; each half of moderator complex cooled separately, filters on inlet and exit air	125°C to 175°C at experimental openings. Complicated, mechanical-electrical system for channel leak detection, He in fuel element serves to indicate leak by pres. drop, provides an inert atmosphere for the uranium, and improves heat transfer. (1950) Plan to replace natural uranium fuel with MTR-type fuel plate

Table 1. Thermal, Heterogeneous, Graphite

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
Hanford (Hanford, Washington, USA)	Classified	Natural uranium, graphite moderator	Graphite
Hanford 305 Test Reactor (Hanford, Washington, USA)	6 watts	Uncooled, graphite-moderated, natural uranium reactor	
Windscale (Sellafield, England)	Classified	Similar to BEPO, natural uranium, graphite moderated	Graphite

Table 2. Thermal, Heterogeneous, Heavy-

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
CP-3 (Argonne National Laboratory, Chicago, Ill., USA)	300 kw; 10^{12}	120 (max. 136) uranium metal rods 1.1-in. dia., 0.035-in. Al jacket, 6-ft long; 3 tons of natural uranium metal; square lattice, 5⅜ in. centerline to centerline; 6½ tons of D_2O contained in 72-in. dia., 105-in. high Al tank	2 ft graphite
CP-3′ (Argonne National Laboratory, Lemont, Ill., USA)	300 kw; 3 to 4 × 10^{12}	4.2 kg U^{235}, about 122 fuel rods 2.16 cm dia., 1.68- long; highly enriched in U^{235}, Al alloy of 98% Al and 2% U	See CP-3
ZEEP (Chalk River, Canada)	3.5 watts; 6 × 10^8 (30 watts max.)	Natural uranium metal in form of slugs, 1.285-in. dia., 6 in. long, jacketed with Al "stockings," 1.295 ± 0.005 in. id, 0.040 ± 0.003 in. wall, and 9 ft 6¾ in. long, holding 9 slugs; total of 148 rods; square lattice spacing of 6 in. 10 tons of D_2O contained in steel tank 6¾ ft dia., 8½ ft high	Graphite, 2½ ft thick under tank, 3 ft around
ZOE (Chatillon, France)	Increased from 5 kw to 150 kw, 2 to 3 × 10^{10}	Originally, UO_2 in tablets 3 cm high stacked in Al tubes of 66 mm id, and 180 cm effective height. Effective density of oxide is 8.3. D_2O (at 40 ± 6°C) contained in cylindrical Al vat, 181 cm id, 235.5 cm high. Max no. of fuel bars is 69 (3.55 tons UO_2), set hexagonally (sides of hexagon 18.6 cm). Uranium oxide replaced by metal rods	Graphite (diffusion length 45 cm), 15 cm of D_2O at bottom
Heavy-Water Research Reactor, USSR)	500 kw	Uranium fuel rods suspended in D_2O contained in 175-cm dia. tank, 195-cm high. Critical experiments performed include use of 2.2 and 2.8-cm dia. rods with a 0.1 cm Al envelope; square lattice spacing from 63 to 162.6 cm; no. of rods from 86 to 292; critical D_2O level from 120 to 181.6 cm; rod length ~ 160 cm	100 cm thick graphite reflector on sides and bottom

Moderated, Natural Uranium Reactors — *Continued*

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
			Water	Three Pu production reactors in 1945, and additional units have been added; secondary recirculating glycol system to furnish heat to buildings
	About an 18 to 20 ft cube	Control rod, shim rod, safety rod; BF_3 control system tested	None	Used to measure reactivity change due to a given amount of sample. Not sealed from atmosphere, and ambient pressure affects reactivity
Biological concrete shield, steel plate thermal shield		24 boron steel control rods, horizontally at right angles to fuel channels	Air, inlet and exit filters, discharge through 400 ft stack	Two Pu production reactors

Water Moderated, Uranium Reactors

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
4 in. Pb-Cd, and 7 ft 8-in. octagonal concrete wall 13-ft high; top shield contains Cd, 1 ft lead, and 4 ft wood and steel	Octagonal 26 ft across, 14 ft high	Rods containing Cd metal; 2 control; 3 shim; and 2 safety	200 gpm D_2O	World's first heavy-water reactor, 15 May 1944. Helium sweep for dissociated D_2O. Improved version CP-3′ with enriched uranium alloyed with Al
See CP-3	See CP-3	See CP-3	See CP-3	Improved design of CP-3. (1950) Replaced by CP-5 in 1954
Water in tanks, 3 ft thick		4 plates, 8 rods, Cd-coated stainless steel	None	Migration area, $M^2 =$ 237 cm^2, slowing down length $\sim \sqrt{118.5}$ cm; buckling factor $B^2 =$ 794×10^{-6} cm^{-2}; mean life of neutron 0.86×10^{-3} sec (1945)
Concrete 150 cm thick		Two sets of 2 types of Cd safety rods; Cd regulating bars	D_2O circulated through an external circulation system	Laplacian = 5.8 m^{-2} (1948)
Concrete side shields 2.5-m thick		4 Cd control rods	D_2O circulated	He atm above D_2O (1949)

Table 2. Thermal, Heterogeneous, Heavy-

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
P-2 (Saclay, France)	1500 kw, 7×10^{12} (max)	136 uranium rods, 1.1 in. dia., 7 ft long, triangular lattice with rod spacing of 5.93 in., fuel element consists of 4 concentric Al cylinders with innermost one as protective sheath for U rod, insulating space between 2nd and 3rd cylinders, 3.3 tons of uranium. Fuel immersed in Al tank, 8⅔ ft high and 6⅔ ft in dia., partly filled with D_2O (6.3 tons). With heavy water at 1794 mm, critical height $H_c = 224 \pm 3$ cm, critical radius $R_c = 132 \pm 2$ cm; $M^2 = 238$ cm^2, $B^2 = (5.30 \pm 0.15) \times 10^{-4}$ cm^{-2}	Graphite 3 ft thick outside tank and on sides and bottom
JEEP (Kjeller, Norway)	Designed for 100 kw, increased to 350 kw; 10^{12}	2200 kg uranium slugs in Al tubes; 35.5 kg per rod; rods 25.4 mm dia., 300 mm long, placed on 180 mm centers. Fuel core 1.9 m long, 7 tons D_2O contained in 2-m dia. tank. 65 to 76 fuel elements	Graphite, 700 mm thick
NRX (Chalk River, Canada)	30 Mw, $> 5.8 \times 10^{13}$, rebuilt reactor, 40 Mw	176 uranium rods, Al-clad, surrounded by two concentric tubes. Uranium rod dia. 1.36 in.; Al sheath thickness 0.079 in.; Outer Al sheath id 1.66 in., thickness 0.040 in. Uranium assemblies hang vertically. D_2O contained in cylindrical vessel 10 ft high, 8 ft dia. Core contains about 10.5 tons of natural uranium, and about 20 tons D_2O. Each fuel rod is 10 ft long. Coolant annulus reduced from 0.100 to 0.071 in.	Graphite
CP-5 (Argonne National Laboratory, Chicago, Ill., USA)	1000 kw, 3×10^{13} (1×10^{13} epithermal, 1×10^{12} virgin). Design is 4 Mw	Fuel elements are curved, sandwich plates, 0.05 cm fuel Al-U^{235} alloy clad with 0.05 cm 72S-Al; 12 plates mounted in a $3 \times 3 \times 24$ in. fuel box. Normal assembly has 12 fuel boxes and 16 req'd at 4 Mw. D_2O contained in Al tank 6 ft dia., 7½ ft high. Fuel core about 2 ft dia. by 2 ft high. Assembly contains about 1680 gm U^{235}, and 1 tons D_2O	2 ft D_2O and 2 ft graphite around and below tank. Zone is 10 ft high and 10 ft dia.
SLEEP, Swedish Low Energy Experimental Pile (Stockholm, Sweden)	300 kw, 3×10^{11} (max)	126 uranium fuel rods, arranged in a hexagonal lattice, spacing 145 mm, 29 mm dia., canned in reflectal (very pure Al alloyed with 0.5% Mg) by forcing through a die, reducing wall thickness of tube from 1.25 to 1.10 mm. D_2O contained in 1.85 m dia. and 2.54 m high reflectal tank	900 mm thick graphite surrounding bottom and sides of tank

Water Moderated, Uranium Reactors — *Continued*

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
Cast-iron thermal shield about 8 in. thick, concrete shield about 7 ft		Cd plate moving between tank and reflector, 2 Cd rods in tank	Originally, recirculation of nitrogen at 10 atm, flows down space between outer two Al cylinders and up space between first and 2nd cylinders. System now uses CO_2 under 7 atm	(1952)
Octagonal concrete shield, 2 m thick on sides, lined with Cd		4 Cd plates 1300 mm long, 350 mm wide, 1.7 mm thick, held between Al plates, placed between tank wall and reflector	4 liters/sec of D_2O (inlet 20°C, outlet 40°C)	Average lifetime of neutron 2×10^{-3} sec (1951)
8 ft concrete, steel thermal shield	34 ft dia., 34 ft high	Varying D_2O level, Cd and boron rods; 18 shutoff rods, 1 control	Light water flows downward in inner annulus (air flows upward in outer annulus); D_2O circulated to external heat exchanger; air cooling of graphite. 95% of heat removed by water flow in fuel channel	Helium purge system. Shut down in Dec. 1952 following accident and severe contamination; operation restored in Feb. 1954. Plutonium and U^{233} are produced. (1947)
¼ in. boral, 3½ in. lead, 4 ft 8 in. limonite-iron concrete ($\rho = 4.4$ gm/cm³)	Octagonal, 20 ft across and 13½ ft high	4 shim-safety rods (Al-clad Cd), signal arm type, operating between parallel rows of fuel assemblies; one regulating rod moving up and down outside the fuel assemblies; quick-opening valve to drain D_2O	About 1200 gpm of D_2O flows up through fuel assemblies (about 1 m/sec in central section max. heat flux about 4.5 cal/cm²-sec) and down in tank with a 5°C rise. D_2O circulated through external, light water heat exchanger and cooled from 126° to 118°F. Chilled water system for limiting temp. rise and to increase reactivity. Thermal shield cooled by light water. (Total shield heat load including graphite about 2 to 3% of total.) Helium purge system, graphite reflector cooled with helium	Reactor housed in an almost leakproof building. (1954)
Biological shield of poured concrete with iron ore, $\rho = 3.8$, 1.8 m thick, lined with a Cd-Al sandwich		Two security rods equiv. to about 2% in reactivity, two regulating plates between reactor tank and graphite reflector control about 0.6% in reactivity	D_2O recirculated, 1000 l/min to air-cooled heat exchanger, about 0.001 of circulation passes through mixed-bed ion exchange. Air used for cooling, passed upward between reflector and tank and downward between graphite and shield	Reactor built underground. Helium atmosphere above D_2O and circulated through palladium catalyst recombination unit. (1954) Effective mean life of thermal neutrons 0.71 ± 0.03 msec

Table 3. Thermal, Homogeneous, Light-Water Moderated, Enriched

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
LOPO, Low Power Water Boiler (Los Alamos, New Mexico, USA)	1/20 watt	Enriched uranium sulfate solution, 580 gm U^{235}, 3378 gm U^{238}, 534 gm S, 14,068 gm O, 1573 gm H; $\rho = 1.348$ gm/cm^3 at 39°C. Solution contained in 1-ft dia., 1/32-in. thick, stainless-steel sphere (15 liters)	3 × 3 × 6-in. bricks of BeO, $\rho = 2.7$; graphite on bottom
HYPO, High Power Water Boiler (Los Alamos, New Mexico, USA)	6 kw, 3×10^{11}	Enriched uranium nitrate solution, 896.6 gm U^{235}, 5341 gm U^{238}, 731 gm N, 13,780 gm O, 1312 gm H; 13.65 liters of $UO_2(NO_3)_2 \cdot 6\ H_2O$ in H_2O, $\rho = 1.615$, contained in 1-ft dia., 1/16 in. thick stainless steel sphere.	24 × 24 × 27-in. BeO surrounded by graphite to form a 60 × 48 × 60-in. rectangular parallelopiped
SUPO, Super Power Water Boiler (Los Alamos, New Mexico, USA)	45 kw, 1.7×10^{12} maximum. (est. max. intermediate flux 2.8×10^{12}; fast flux 1.9×10^{12})	Enriched uranium nitrate solution, 88.7% of U as U^{235}, 777 gm U^{235} for critical mass, 870 gm U^{235} used; $\rho = 1.10$; 12,700 cm^3 solution contained in 1-ft dia. stainless steel sphere	About a 55-in. cube of graphite
Water Boiler Neutron Source, WBNS (North American Aviation, Inc., Downey, Calif., USA)	1 watt, 5×10^7	1.5 lb of uranium fuel in form of U^{235} enriched uranyl nitrate in light-water solution; solution contained in a 1-ft dia. stainless steel sphere 1/16-in. thick	Graphite, 5-ft dia. and 6-ft high
Livermore Water Boiler (Livermore, Calif., USA)	100 watts, 10^9 (increased to 500 watts, 2×10^{10} max. thermal flux)	14.524 liters of light water solution of UO_2SO_4 containing 694.2 gm U^{235}; 798 gm-mols hydrogen; 420, oxygen; 5.64, sulfur; and 2.95, uranium. Solution contained in stainless steel sphere, 12½ in. od, 0.06 in. wall thickness	Right cylinder of graphite, 5-ft dia. by 5-ft high. (Brookhaven graphite with thermal diffusion length of 51.6 cm.)
North Carolina State College Reactor, NCSC (North Carolina State College, Raleigh, N. C., USA)	10 kw, 5×10^{11}	14 liters of light water solution of UO_2SO_4 containing 790 gm U^{235} of 90% isotopic enrichment, $\rho = 1.08$ gm/cm^3; solution contained in stainless steel cylinder, 1/16 in. thick, 11 in. high, 10¾ in. dia.	20 in. graphite; 105 ft^3, 5.4 tons; $\rho = 1.65$ gm/cm^3
HRE-1, Homogeneous Reactor Experiment (Oak Ridge National Laboratory, Oak Ridge, Tenn., USA)	1000 kw	Enriched uranyl sulfate (enrichment > 90%) dissolved in distilled water; solution contained in an 18 in. dia. 347 stainless steel sphere with 3/16-in. walls. Boiling of fuel solution suppressed by maintaining a pressure of 1000 psi (in pressurizer, solution heated to 545°F). Max. fuel temp. 482°F. Fuel consumption 0.002 lb/day at 1000 kw	10-in. layer of D_2O, pressurized with helium to within ± 100 psi of fuel solution; temp. regulated to 350°F

Uranium Reactors (Water Boilers and Homogeneous Reactor Experiments)

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
None		Cd cylinder, ¾-in. dia., 34 in. long, Cd safety curtain	Water; max. temp. 39°C	World's first water boiler reactor, 1944; replaced with HYPO
4 in. lead, 1/32 in. Cd, 5 ft concrete		Rods containing Cd, 1 shim, 2 control, 1 safety	50 gal/hr water through 6-turn, ½-in. id, 157-in. long coil. Max. solution temp. 185°F	About 50 cm^3/sec of air used to sweep out H_2 and O_2 from decomposition of H_2O. Replaced by SUPO. (1944)
½ in. of B_4C in paraffin; 2 in. steel; 4 in. lead; 5 ft concrete	About 15 × 15 × 11 ft	See HYPO, two additional control rods move in reactor core in reentrant thimbles	Water circulated through three 20-ft long, ¼ in. od, 3/16 in. id, stainless steel tubes	100 l/min air circulation, recombiner for H_2 and O_2; rate of radiolytic hydrogen evolution about 0.55 moles/kwh (1951)
2 ft of concrete blocks		Two safety rods, coarse control, fine control, move in reflector	None	Calculated $k_\infty = 1.561$ (leakage 0.360). (1951)
Graphite in cylindrical steel tank, 0.030-in. Cd shot and 5-in. lead, 3-ft concrete blocks	13 × 26 × 9 ft high	Two safety and two control rods: rods are flattened stainless steel tubing packed with 1 (control rods) or 2.2 lb boron carbide (safety rods)	Cooling coil in sphere, six turn helix of 5/16-in. id tubing, 160-in. long, distilled water used as coolant, and refrigeration unit used to reduce solution temperature	Closed gas handling system. (1953)
6 ft concrete, barytes ores coarse aggregate and colemanite ore as fine aggregate, $\rho = 3.4$ gm/cm^3; 4 to 6 in. lead around graphite	Octagon, 17 ft across, 12 ft high	Two control rods, stainless steel tubes of 2.5 gm/cm^3 sintered B_4C powder, in reentrant sheaths; two shim rods, 4-in. wide Cd strips, periphery of reactor cylinder	Light water (refrigerated city water) 1 gpm flow through each of 4 helical coils of ¼-in. stainless steel tubing, 7 ft immersion length	Max. reactor solution temp. 80°C. (1953)
Reflector and core contained in an outer pressure vessel of forged steel, 39-in. id with 3-in. thick wall; 7-ft thick concrete walls (barium sulfate ore used as aggregate to increase ρ to 3.5)		Reactor is self-stabilizing as a result of its large negative temp. coef., (and mechanical safety devices found to be unnecessary). Safety measures include: 2 safety plates fall in 0.01 sec and are completely effective within 0.2 sec, reduce reactor temp. from 250 to 243°C; reflector may be dumped within 6½ sec, drops operating temp. to 160°C; dilution of fuel requires 15 min and is normal shutdown procedure; reduction in power demand; and drainage fuel solution	Reactor solution circulation, 100 gpm; outlet temp. 482°F. and cooled to 410° in a U-tube heat exchanger, generating steam in the shell side at 3000 lb/hr at 200 psia. Generated 150 kw of electricity. Canned rotor centrifugal pump for circulating fuel solution. About 50 kw removed by recirculating D_2O reflector through boiler feed water preheater	Gas evolution of ($2H_2 + O_2$) at rate of 10 ft^3/min (S.T.P.) and 20 cm^3/day of fission-product gases. About 40 kw liberated in flame recombiner, followed by a catalytic recombiner. Radioactive gases absorbed in activated carbon beds. Catalytic recombiner for D_2 and O_2 gases. Radioactivity of fuel solution about 30 curies/cm^3. Reactor has been dismantled and to be replaced by HRE-2. (1952)

Table 4. Thermal, Heterogeneous, Light-Water Cooled

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
Bulk Shielding Facility (typical of swimming-pool types) (Oak Ridge National Laboratory, Oak Ridge, Tenn., USA)	100 kw, 5×10^{11} (10^{12} max. avail. thermal flux, 3×10^{12} max. available epithermal flux)	MTR-type fuel elements: 18 convex plates, each 3-in. wide, 24-in. long and 0.06-in. thick, comprise a fuel box. Plates of enriched uranium (> 90% U^{235}) encased in 2S Al sandwiches, clad with 72S (the latter corrodes preferentially). About 140 gm U^{235} per fuel box. Al (including structural members) to H_2O ratio is 0.7. Fuel boxes held in an Al grid 5 in. deep. Critical mass about 3 kg U^{235} in fuel core $12 \times 12 \times 24$-in.; with allowances for other factors including beam holes, burn-up, 3.5 kg; with BeO reflector, 2.4 kg U^{235} in fuel core $9 \times 12 \times 24$-in.	10 cm BeO on 4 sides of active lattice; or light water
MTR, Materials Testing Reactor (Reactor Testing Station, Arco, Idaho, USA)	30 Mw, 4×14^{14} (in reflector), 1×10^{14} fast flux	Fuel assembly is a $3 \times 3 \times 24$-in. box containing 18 curved, vertical, fuel plates; plates consist of uranium-Al alloy, 0.5 mm thick, sandwiched between 0.5 mm Al; 3.0 mm spacing between plates; U^{235} per plate increased from 140 gm to 200 gm. Lattice configuration can be varied from a 3×9 (27 fuel assemblies) to a 5×9 array. Initially about 4 kg U^{235}. Core of fuel assemblies held between 2 grids, and is contained in a 54-in. dia. Al tank extended to form a well 30 ft deep filled with H_2O. Al to H_2O vol. ratio 0.73	Primary reflector, Be 3-ft high, 54-in. dia., in water zone: secondary reflectors, graphite in two zones outside water tank; pebble zone of 700,000 1-in. dia. balls to form 7 ft 4 in. square by 9 ft high, outer stack of graphite to 12×14 ft by 9 ft 4 in. high
RMF, Reactivity Measurement Facility (Reactor Testing Station, Arco, Idaho, USA)	Zero power	30 MTR fuel assemblies installed in MTR canal	
LITR, Low-Intensity Test Reactor (Oak Ridge National Laboratory, Oak Ridge, Tenn., USA)	Increased from 500 to 3000 kw, 2×10^{13}	3.4 kg U^{235}. MTR fuel assemblies (16 plates per fuel assembly)	Loosely stacked Be blocks, minimum thickness 8-in.
Experimental Nuclear Reactor (RFT) (USSR)	300 kw, 2×10^{12}	Active core approximated by a cylinder 40-cm dia., 50 cm high; core consists of 32 units with 24 units each containing 16 fuel elements (remaining units for control rods); cylindrical fuel elements, 9 mm od, 50 cm high, square lattice with ~ 18 mm spacing; fuel elements contain 10% enriched U^{235}; 3.5 kg U^{235} in core. Light water moderator, reflector and coolant. Core in 500 mm dia., water-filled Al tank, and fuel units fixed at top and bottom in guide grids. Al and Al alloy used as structural materials in active core and for canning fuel elements	Water

(and Moderated) Enriched Uranium Reactors

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
16 ft of water above reactor, water and high density concrete below reactor	Pool 40 ft long, 20 ft wide, and 20 ft deep	Two Cd-Pb safety rods (each equivalent to 3 to 4% reactivity) and a Cd-Pb regulating rod (0.8% reactivity); cross sections of 1¼ by 2½ in., and travel inside special fuel boxes. Other rods used are two B-Pb shim safety rods (mixture of Pb and boral) and one control rod	Light water, convective flow	Addition of 50 ppm Na_2CrO_4 to inhibit corrosion of Al (1950)
Two 4-in. layers of steel thermal shield, 9 ft barytes concrete, water 15 ft above core and 5 ft below	About a 34-ft cube	Up to 8 vertical shim-safety rods, upper section containing Cd steel and lower containing fuel plates; Be-Cd shim rods; 2 vertical regulating rods in Be reflector	Light water recirculated, cooled by flash vaporization, 20,000 gpm, enters at 100°F at top and leaves at bottom at 111°F. About 1/6 heat generated in graphite and thermal shield, air cooled; 4% of heat generated in Be reflector	Reactor used for study of materials exposed to intense radiations. (1952) About 100 gm U^{235} equiv. to 1% reactivity. Fuel plates are curved laterally on a 5.5-in. radius to strengthen them and to define direction of thermal strain.
				Cold clean reactor used for precise measurements in reactivity effects
Unmortared concrete blocks, except for outermost 1-ft layer. Minimum thickness 10½ ft		Three shim-safety rods, containing a cadmium section and a fuel section	Light water recirculation system with water-air heat exchanger to dissipate heat. Cellulose filters used remove turbidity of hydroxides (Al and carbon steel piping); pH maintained between 5.5 and 6.5; by-pass line to demineralizer	Installation served as mock-up for MTR; converted to a training and research reactor (1950)
Cast iron and water used as shielding		Three boron carbide safety rods, one steel rod	240 m^3/hr water flow through core, enter at 30°C, 1°C rise. Max. fuel surface temp. 70°C	A 2 Mw thermal power nuclear reactor for research purposes has been designed with 52 units; fuel elements 10 mm od, fuel loading 4.5 kg U^{235}; water flow 900 m^3/hr; 8 boron carbide rods (3 safety, 5 shim) one steel regulating rod

Table 4. Thermal, Heterogeneous, Light-Water Cooled

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
RPT, Reactor for Physical and Technical Investigations (USSR)	10 Mw	Central core consists of 37 cylindrical ducts, 54 mm dia. with 14 cm spacing, piercing graphite layers, core 1-m dia., 1-m high. Tubes of Al alloy including fuel elements are inserted in ducts; fuel elements are hollow cylinders containing enriched uranium, covered with an Al casing; spacer ribs on Al end caps and on Al belt in the medium section. Both graphite and water act as moderator	Core and graphite reflector fill a cylinder 240 cm high, 260 cm-dia., 80 cm of graphite on sides, 60 cm on bottom. Graphite $\rho = 1.8$ gm/cm^3 thermal diffusion length $\sim$ 52 cm
APS, Atomic Power Station (USSR)	30 Mw (heat), 5×10^{18} (5 Mw electric power)	Enriched uranium (5% U^{235}) graphite and water moderated. 128 fuel channels pierce central part of graphite brickwork, forming a core 150-cm dia., and 170-cm high. Thin-walled steel used for fuel channel, and hollow fuel elements placed in channel (see RPT), 60 uranium rods required for criticality. Total uranium charge is 550 kg	Graphite encased in a hermetical steel jacket, clearance allowed in graphite brickwork, and jacket filled with He or N_2. Max. graphite temp. 650 to 700°C
Geneva Reactor Exhibit (Geneva, Switzerland)	10 kw (nominal) to 100 kw; 10^{11} (at center)	Swimming-pool type, about 4.5 kg U^{235}, enriched to 20%, 23 MTR-type fuel assemblies with 18 plates per assembly; Al to H_2O volume ratio 0.65; active lattice 15 × 15 × 24 in.	Light water
STR, Submarine Thermal Reactor (STR-Mark I, prototype, National Reactor Testing Station Arco, Idaho, USA) (STR-Mark II, U.S.S. Nautilus)			
TTR, Low power Thermal Test Reactor (Knolls Atomic Power Laboratory, Schenectady, N. Y., USA)	100 watts, 3.4×10^9 (Improved version up to 10 kw)	20 fuel slug tubes, each 2-in. dia., 24-in long, containing U-Al alloy disks with polyethylene spacers strung on 3/16-in. dia. rod; disks (slugs) immersed in light paraffin base oil contained in slug tube. Tubes in cylindrical array in an Al tank of 12-in. id and 18-in. od; tank filled with water. 2.7 kg U^{235} required for criticality; extra 0.1 kg provides about 0.6% excess reactivity	30-in. cylindrical ring of graphite (water, hydrocarbon)

(and Moderated) Enriched Uranium Reactors — *Continued*

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
Side shielding, iron layer/frame 2.5 cm thick; 320 cm concrete, $\rho = 2.4$ gm/cm^3; top, 150 cm graphite, 40 cm Pb, 20 cm-thick iron slab		2 automatic control rods, 3 manual, 3 slowly moving automatic rods of boron carbide, 2 systems of safety rods	Distilled water enters annulus between fuel tube and duct, and flows out through internal duct of fuel element; circulated in closed system. 20 to 30°C inlet, 55 to 65°C outlet; individual flows to ducts measured; 6 m^3/hr per duct	Use of He for heat transfer in graphite and to dry graphite if water should leak in. Calculated k with and without water is 1.59 ± 0.03. (1952) (Replica of APS)
Side water shield 100-cm thick, concrete wall 300-cm thick		18 boron carbide rods	Distilled water at 100 atm pumped down through the tubes and return up flow over surface of the uranium fuel elements. Recirculation with makeup system. Flow and temp. of each channel recorded. Inlet temp. 190°C, outlet temp. 260–270°C. Heat exchange to generate steam at 12.5 atm, 255–260°C. Separate cooling facility for graphite	U^{235} burnup about 15 to 20%. (1954)
Light water plus earth	Pool size, 10 ft dia. 21 ft deep; 13,000 gal capacity of demineralized water	Three safety and control rods, boron carbide, each about 2%	Light water, natural convection	Built by US for the International Conference on Peaceful Uses of Atomic Energy, 1955
			Light water	World's first mobile reactor unit; first major use of zirconium. (1953)
Reactor located in a room with 6-ft thick concrete walls	About a 5-ft cube	Coarse control, 6 Fe-clad Cd sheets, 4-in. wide, 18-in. long, at periphery of tank. Fine control, 2 Cd rods, 18-in. long, ½-in. dia., move between fuel slug tubes. Safety, 4 Cd rods ½-in. dia	(For 10 kw reactor, deionized water circulated through lattice)	Internal thermal column is a graphite cylinder 12-in. dia., 18-in. high. Test hole in center of column. Several reactors of this type have been built. (16 fuel assemblies, 3-in. dia. discs, Al alloy containing 90%-enriched U^{235}, clad with 2S Al. Four Cd safety sheets, 4 Cd safety rods, 3 control rods and neutron source.)

Table 5. Fast and

Name (1)	*Rating* (2)	*Fuel description* (3)	*Reflector* (4)
Los Alamos Fast Reactor, "Clementine" (Los Alamos, New Mexico, USA)	25 kw, 10^{13} fast flux	Rods of pure plutonium metal clad with steel, in 6-in. array, contained in a 6-in. dia. mild steel pot: rods of clad natural uranium interspersed with the fuel rods	Around fuel pot, 6-in. thick reflector of natural uranium built out of silver plated blocks, 6-in. of steel followed by 4-in. lead to form 38-in. cube
EBR-1, Experimental Breeder Reactor (Reactor Testing Station, Arco, Idaho, USA)	1400 kw, fast flux, center of core, 1.1×10^{14}	Close packed array of rods, each containing a U^{235} center section with a top and bottom natural uranium blanket section. Fuel slugs (2 per rod), enriched to $\sim$ 90% U^{235}, 4¼-in. long, 0.384-in. dia., jacketed in stainless steel. Provisions for 217 rods (rods not containing U^{235} are loaded with uranium) 48.2 kg U^{235} for criticality, 52 kg used. Double-walled reactor tank with gas space used for leak detection and insulation	Natural uranium blanket in two sections: first section, tightly packed array of rods 15/16-in. dia. 20¼-in. long, clad with 0.020-in. stainless steel jackets. Core and inner blanket contained in a 15⅞-in. dia., stainless steel tank. Outer blanket consists of an array of keystone-shaped bricks forming a cylinder around sides (and bottom) of reactor tank. Bricks are natural uranium, jacketed in 0.020-in. stainless steel. Outer blanket and shield mounted on hydraulic elevator. 18-in. of graphite for reflector
SIR, Submarine Intermediate Reactor (SIR-Mark A, prototype, West Milton, N. Y., USA; SIR-Mark B for U.S.S. Sea Wolf)			
Zephyr, Zero Energy Fast Reactor (Harwell, England)	Four watts	Cylindrical core with height = dia. $\simeq$ 15 cm, consisting of natural uranium and plutonium	Uranium

Table 6. Reactor Developments

ARGENTINA

Plans for a natural uranium fueled, beryllium-oxide moderated, heavy-water cooled reactor, patterned partially after Daniel's Oak Ridge design concept, to produce 17,000 kw.

BELGIUM

Heterogeneous, natural uranium fueled, graphite-moderated reactor under construction; US fabricated Al-clad fuel slugs; 18 cm lattice spacing, air cooled, 3.5 Mw (heat).

CANADA

NRU (Canada's third reactor)

To operate in 1956; produce plutonium as well for testing components for the prototype power reactor.

ITALY

CP-5 type reactor to be built in the Po Valley.

THE NETHERLANDS

Suspension Reactor

A reactor involving the suspension of uranium oxide in heavy water or fluidized uranium oxide under study.

UNITED KINGDOM

Calder Hall Piles (United Kingdom's first commercial scale power reactor)

To produce > 50 Mw electricity.

Natural uranium fueled, graphite-moderated reactor, with Co_2 under pressure as coolant; heat transferred in 70-ft high 18-ft dia towers to a double-pressure steam cycle; reactor contained in 40-ft dia by 60-ft high pressure shells, several inches thick.

Intermediate Reactors

Shielding (5)	*Over-all size* (6)	*Control* (7)	*Coolant* (8)	*Remarks* (9)
Alternating layers of 3-in. of iron and masonite, and later iron and boron-impregnated plastic, total thickness of 30-in. on 3 sides. 18-in. heavy aggregate concrete	11 × 15 × 9 ft high	Two safety and two regulating rods in uranium reflector, natural uranium in lower section and B^{10} in upper section, reactivity equivalent 0.004; safety block, large section of reflector, dropped to produce a reactivity change of 0.013	Mercury; water used to remove heat from uranium reflector	World's first fast reactor (1946). Dismantled 1954 after failure of fuel element
Thermal neutron shield, 6-in. iron (air-cooled) 9-ft ordinary concrete		12 control rods of natural uranium in outer blanket, 8 for shutdown (0.2% k), 4 for control (0.1% k), one bottom safety (0.07% k), 4¼-in. blanket travel (0.89% k) complete removal of outer blanket (8.9% k)	292 gpm NaK, temp. rise 88°C, 228°C in. flow upward in core and downward in first blanket. Outer blanket air cooled, 6000 ft³/hr maintains bricks at 200°C. 72% heat generated in core, 14% inner blanket, 14% outer blanket. Coolant adds a reactivity equivalent to 2 kg U^{235}	World's first production of electric power from a nuclear reactor (1951); 200 kw produced. Conversion ratio determined by two methods as 1.00 ± 0.04 and 1.01 ± 0.05. Second uranium core to be replaced by plutonium core. (First core produced over 3 × 10^6 kwh of heat.) Fission gases collected in top section above fuel slugs
			Liquid sodium	Prototype reactor enclosed in 225-ft dia., 1-in. thick, steel sphere, gas tight
No shielding, but reactor in small concrete room		Control and safety rods consist of uranium rods moving vertically in channels around core; uranium safety block can be pulled out	None	(1954)

Dounreay Reactor (United Kingdom's first breeder reactor)

To be built at Dounreay in Northern Scotland; reactor housed in steel sphere 135-ft dia. To produce 60 Mw (heat), core size 2-ft dia, 2-ft long, sodium cooled. Neutron shield 4-ft thick graphite containing boron.

Dimple, Deuterium Moderated Pile Low Energy (Harwell, England)

United Kingdom's first heavy-water reactor. (1954)

E.443 (Higher Power, Heavy Water Reactor) (Harwell, England)

Designed to remove 10 Mw from core, 1 Mw from experimental facilities. 10^{14} thermal flux. Core may contain as much as 2.5 kg U^{235} in form of uranium-Al alloy plates arranged in boxes, forming a cylinder 60 cm long, equivalent dia 86 cm. Core placed in center of 2-m dia. Al tank, 2-m high. Forced upward flow of D_2O. Graphite reflector outside Al tank, 60 cm thick on sides and bottom, contained in steel tank in He atm. Biological concrete shield.

UNITED STATES

Boiling Water Reactors

Borax I. Operated successfully at National Reactor Testing Station under large power excursions, but was destroyed in 1954 in a "runaway" simulated test with a power rise over 10^6 kw in 0.1 sec. MTR-type reactor core in reactor tank, 4 ft dia, 13 ft high. Reactor tank filled with water to within 3 to 4½ ft from top.

Borax II. Operating at 6000 kw heat under 300 psi. (Ten fuel plates per assembly.)

EBWR, Experimental Boiling Water Reactor (Argonne National Laboratory). Scheduled to begin operation in 1956 with output of 5000 kw electricity, 20 Mw heat. Light water coolant and moderator. Reactor and power generating equipment housed in gas-tight steel shell with a 400,000 ft³ volume. Fuel elements are mostly natural uranium with a few enriched elements necessary for attaining criticality.

SRE, Sodium Reactor Experiment (Santa Susana, California)

To produce 20 Mw (heat). Sodium cooled, graphite moderated, uranium enriched to 2.8% U^{235}. 6-in. long, 0.750 in. diameter uranium slugs stacked in 6-ft long stainless steel tube, 0.010-in. walls, with NaK in 0.010 in. annulus. He in gas space above slugs. Clusters of 7 tubes form the fuel elements. Core region 6-ft high, 6-ft dia, containing about 31 fuel clusters. Hexagonal graphite prisms, 10-ft long, are clad with zirconium. The core tank is stainless steel, 19-ft deep, 11-ft dia 5½-in. steel thermal shield surrounds tank, concrete biological shield. Sodium coolant inlet temp 500°F, avg outlet temp 960°C. Secondary sodium coolant employed.

Tower Shielding Facility (Oak Ridge National Labaratory, Oak Ridge, Tenn.)

Reactor operated at an altitude of 200 ft. Used for experiments requiring minimal scattering from the ground and adjacent structures.

EBR-2

To produce 62.5 Mw heat, 15,000 kw electricity.

To be fueled eventually with a uranium-plutonium alloy with a blanket of natural or depleted uranium.

Zero-power fast critical assembly is at Reactor Testing Station, Arco, Idaho and will furnish data for critical mass, breeding ratio, and power distribution in the core of EBR-2.

HRE-2 (Improved version of HRE-1)

Initial operation to be with D_2O blanket system, fuel solution to be dilute solution of uranyl surfate (about 90% U^{235}) in D_2O, 9.6 gm U^{235} per kg D_2O, power level of 10 Mw. Later operation will use thorium oxide suspension in D_2O in blanket system, and U^{233} to be substituted for U^{235}. Design bases: inlet fuel solution temp. 256°C, outlet, 300°C; fuel circulation rate 400 gpm; control by variable solution concentration, negative temperature coefficient (-2×10^{-3} per °C); core id 32 in., 2000 psia; 14-in. blanket, blanket id 60 in., 2000 psia, wall thickness 4.4 in.; core material Zircaloy-2, system, 347 stainless steel.

PWR, Pressurized Reactor (Shippingport, Pa.)

To produce ~ 230 Mw heat, 60 Mw electricity (full scale power plant). Light water cooled and moderated, enriched uranium in clad plates and natural uranium in tubes. Zirconium-uranium alloy plates clad with Zircaloy-2, 0.080-in. plate thickness, 0.080-in. spacing for water coolant. UO_2 pellets in Zircaloy-2 tubing, about 10-in. long, 0.413-in. od (0.028-in. wall); 100 tubes assembled in a bundle, 7 bundles fastened in a stack to form a total height of 6 ft. Reactor core is a 6-ft dia right cylinder, 6-ft high, containing both highly enriched "seed" and natural uranium blanket assemblies. Seed assemblies consist of the plates, and blanket assemblies contain the tube bundles. 52 kg of enriched U^{235} in seed assemblies and 12 tons of uranium in form of UO_2 in blanket. Reactor vessel over-all height 33 ft, 9 ft dia, nominal wall thickness 8½-in., 2000 psia. Water flow 45,000 gpm, inlet 508°F, outlet 542°F; 600 psia saturated steam produced. 24 hafnium control rods.

WTR, Westinghouse Test Reactor

To operate at 10 Mw and to be used for material and reactor component testing.

HPPR, High Performance Research Reactor (Oak Ridge Natianal Laboratory, Oak Ridge, Tennessee)

To be of MTR-type and to operate at 10 Mw.

Homogeneous Power Reactor Experiment (Los Alamos, New Mexico)

2000 kw, uses phosphoric acid fuel solution.

LMFR, Liquid-Metal-Fuel Reactor (Brookhaven National Laboratory)

Feasibility studies under way for development of reactor systems using a solution or dispersion containing U^{233} in molten bismuth.

Swimming Pool, Water Boilers, and CP-5-type Reactors

More than eight reactors of these types are planned for the national laboratories, universities and industrial groups. More than 25 universities, alone, are considering reactors for their programs in nuclear engineering education.

Savannah River Reactors

Five production reactors utilizing heavy water.

OWR, Omega West Reactor (Los Alamos, New Mexico)

1000 to 4000 kw reactor of MTR type.

TURKEY

US to loan 6 kg U^{235} (20% enrichment).

Table 7. Plans for Power Producing Reactors

FRANCE

French Five-Year Program

Two plutonium producing reactors, Marcoule, France.

G-1. Similar to Brookhaven reactor, contains 100 tons of natural uranium, elements 26 mm dia, 100 mm long, sheathed, in Mg. 1200 tons of graphite. Air cooled at atmospheric pressure. Under construction, to produce about 40 Mw heat and 5 Mw electricity in 1956.

G-2. Graphite-moderated; 100 tons of natural uranium, elements 26 mm dia, 300 mm long sheathed in Mg. CO_2-cooled in a pressurized closed circuit. Under construction, to produce 100 to 150 Mw heat, and 30 Mw electricity.

UNITED KINGDOM

United Kingdom's 10-Year Program

Type of reactor	*No. of reactors*	*Electric output, Mw*	*Estimated completion date*
Gas cooled, Calder Hall type	4	400 to 800	1960-1
Gas cooled, improved	4		1963
Higher power reactors	4	1000	1963-4
Liquid-metal-cooled reactors	4		1965
Production and power generation	6	Addition to above program	

UNITED STATES

AEC Program

Project	*Heat output, Mw*	*Electric power output, Mw*	*Estimated completion date*	*Notes*
Pressurized Water Reactor, PWR	~230	60	1957	See Table 6
Experimental Boiling Water Reactor EBWR	20	5	1956	See Table 6
Sodium Graphite Reactor	20	7.5	1955	See Table 6
Homogeneous Reactor Experiment No 2 HRE-2	10	2	1956	See Table 6
Experimental Breeder Reactor No 2 EBR-2	62.5	15	1958	See Table 6
Industrial Reactor Program				
Yankee Atomic Electric Co.	500	100	1957	Pressurized light-water moderator and coolant
Nuclear Power Group	692	180	1960	Boiling-water type

Atomic Power Devel. Assoc., Inc.	300	100	1960	Fast breeder
Consumer Public Power District of Nebraska	250	75	1959	Sodium-graphite reactor
Consolidated Edison	500 (140 Mw from reactor 96 Mw from oil-fired superheater)	236	1959	Pressurized water, uranium-thorium converter

Mobile Reactors

Submarines	See STR, SIR; SAR (Submarine Advanced Reactor)
Ship	LSR (Large Ship Reactor, Pressurized water type)
Aircraft	ARE (Aircraft Reactor Experiment) low power prototype of high temperature power producing reactor.

(Army Package Power Reactor, APPR, pressurized water type to produce 2 Mw electricity.)

OTHER REACTOR PLANS

Australia

10 Mw power reactor planned.

Canada

Atomic Energy of Canada, Ltd. and Hydro-Electric Power Commission of Ontario to build 20 Mw (electricity) prototype heavy water, natural uranium power reactor; to be in operation 1958.

The Netherlands—Norway

Dutch-Norwegian program. Ship reactor—prototype for pressurized heavy-water reactor to be built in The Netherlands.

Sweden

Plans for 20 Mw reactor by 1959 to be built south of Stockholm. 100 Mw reactor by 1965 to be built in western province of Bohuslan.

Switzerland

10 Mw (electricity) experimental heavy-water reactor to be built at Würenlingen.

USSR

One reactor with 5 Mw electric output. Second reactor being built with 100 Mw electric capacity.

Boiling homogeneous nuclear reactor for power.

PEACEFUL USES OF ATOMIC ENERGY:

Proceedings of the International Conference in Geneva, August 1955.

The following is a complete listing of the sixteen volumes which comprise this publication, together with the main subjects covered by each volume.

VOLUME 1

The World's Requirements for Energy: The Role of Nuclear Power

World Energy Needs
Capital Investment Required for Nuclear Energy
The Role of Thorium
The Role of Nuclear Power in the Next 50 Years
Education and Training of Personnel in Nuclear Energy
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