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By
F. T. Miles
D. W. Bareis
O. E. Dwyer
D. H. Gurinsky
I. Kaplan
R. J. Teitel
C. Williams

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Brookhaven National Laboratory
Upton, New York

Technical Information Service, Oak Ridge, Tennessee

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FLUID FUEL REACTORS WITH URANIUM-BISMUTH

F. T. MILES, D. W. BAREIS, O. E. DWYER, D. H. GURINSKY, I. KAPLAN, R. J. TEITEL, and C. WILLIAMS

Brookhaven National Laboratory

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Uranium-bismuth solutions or dispersions of USn_3 in liquid bismuth-lead-tin are proposed as fluid fuels for a power breeder or converter. These fuels offer advantages of high specific power, unlimited burn-up, good neutron economy, and low over-all operating costs which might lead to truly economical nuclear power. A simple and effective chemical process based on the extraction of liquid metal with fused salts will remove fission products in a concentrated form for use or storage. A process for the continuous separation of U^{233} from thorium in the blanket is described. Several design possibilities are discussed.

INTRODUCTION

In the last five years much attention has been focused on the problem of using the nuclear fission chain reaction as a source of energy for useful power. The fact is well established by now that stationary nuclear power plants are technically possible; thus far, however, from a power standpoint only, they cannot compete economically with modern power plants burning fossil fuels.

A reactor which produces power and fissionable materials for weapons might have the cost of operations partially subsidized to such an extent that the power could be sold as a profitable by-product. In the long run, however, any such artificial price support cannot be depended on, and fissionable material is worth no more than its value as fuel. This fact leads to the conclusion that ultimately fuel and operating costs must be low enough to be paid for chiefly by power, with excess bred fissionable material and fission products as by-products of unknown value. Breeding, therefore, will be necessary to achieve low fuel costs. The breeding rate, however, perhaps need be only fast enough to cover expansion of the power industry. The greater part of the operating cost results from the handling and processing of fuel and waste products. The use of solid fuel results in such expensive operations that it is hard to see how development of this type of reactor can ever yield economical power. Fluid fuels* offer possible economies which seem to be unattainable using solid fuel

*The term "fluid" as used here means any form of material capable of being reprocessed entirely by remote control. This might include gases, liquids, solutions, slurries, liquid alloys, powders, or even lumps of solids, provided that the latter could not be harmed mechanically by radiation or that damage could occur without affecting the handling of the fluid material.

rods. This article describes one class of liquid fuels based on uranium-bismuth and goes into their possible applications in integrated systems of reactors and chemical-processing plants.

Some of the ways in which liquid fuel reactors can reduce costs are as follows:

1. By substantially reducing the holdup of fissionable material, including material in the reactor, blanket, decay tanks, and separations and reprocessing plants, as made necessary by the reactors.

2. By rapid removal of fission products, resulting in less inventory of radioactive material in the reactor, greater safety, and therefore a smaller required exclusion area. (If located in an industrial region close to a power market, the area necessary for safety is very expensive; if it is moved farther away, there are additional power transmission costs.)

3. By reducing fuel handling and processing costs.

These costs are multiplied many times by low burn-up. Burn-up is limited by radiation damage and by undesirable secondary neutron captures, e.g., destruction of desired products or capture by fission-product poisons. From the point of view of neutron economy, separations need not be complete if the fuel is to be recycled, and a decontamination factor of 10 is sufficient to avoid serious fission-product poisoning. If the fuel must be handled in fabrication it must be decontaminated by a factor of 10^8, and any material going to waste must be very low in fissionable material. All this points to the need for a fissionable material cycle which is integrated with the reactor and is small, completely self-contained, and operable by remote control. Uranium-bismuth solutions can be processed very simply as described below and meet these requirements.

In a reactor system with an external fertile blanket there are similar limitations on the amount of conversion before reprocessing. A capture by a newly formed product may represent a double loss, a potentially fissionable atom and a neutron. Moreover, these captures add to the blanket cooling problem. Consequently, to get any breeding gain, the product must be removed after only about 0.1 per cent of the fertile material has been converted. A process for the continuous separation of U^{233} from a thorium blanket is being tested and is described below. Combined with the liquid fuel reactor this would give a breeder, or converter, system with integrated separations processes.

Finally, the waste disposal problems are simplified in a liquid fuel reactor since the fission products are removed frequently, continuously if desired, in a very concentrated form. This may allow a smaller area for safe operation.

DESIGN OF CORE

The design of a reactor core based on uranium-bismuth fuel is a relatively simple matter.

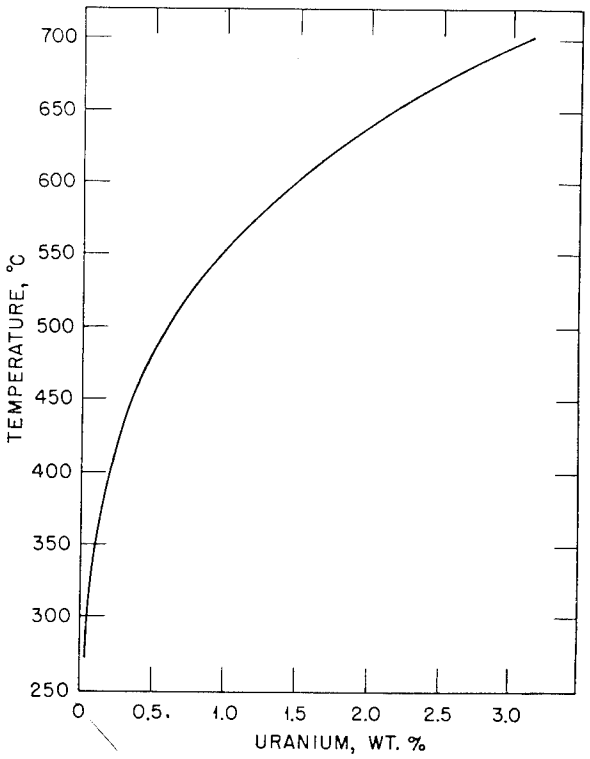

Fig. 1—Solubility of uranium in bismuth.

It can cover a wide range of uses to which the reactor may be put. A solution or slurry of uranium in bismuth as a fuel was suggested by Young[1] in 1947. Since then the solubility curve has been established[2-5] and is shown in Fig. 1. The solubility of uranium in bismuth is appreciable; for example, at 345°C it is approximately 0.1 wt. %. This is sufficient for the operation of

a thermal reactor with a solution of enriched U^{235} or pure U^{233}, but it is not enough for operation with natural uranium. The solubility of thorium in bismuth is very low. Therefore a breeder or converter must consist of an enriched core surrounded by a blanket. Since there is no heat-transfer limitation from the uranium to the bismuth in such a fuel, high specific power is possible, and the total power output can be high and is limited only by the capacity of the external heat exchanger. However, as the size of the heat exchanger is increased, the fuel holdup is increased.

As the first illustration we consider a thermal reactor which might be used either to fission U^{235} or U^{233} in the core and to produce U^{233} in an external blanket. For the sake of simplicity we will consider the case with an external heat exchanger. The reactor-core design is based on the use of a solution of U^{235} or U^{233} in molten bismuth at about 400°C as the fuel. The reactor proper consists of a cylindrical chamber containing beryllium or graphite rods as the moderator.

Exploratory calculations[6] on a reactor of this type were done over a range of values of the beryllium/bismuth ratio and of the uranium concentration in the bismuth. For purposes of comparison, the U^{233} concentration was chosen to give the minimum doubling time for a U^{233} breeder.

Initially, only neutron losses to beryllium and bismuth were considered. Assuming a fixed rate of heat removal per ton of bismuth, a constant determined by the flow rate and heat-exchanger design, we define the quantity a as the number of neutrons lost to beryllium and bismuth in the core per neutron captured by uranium.

$$a = \frac{N_{Bi}\sigma_{Bi} + N_{Be}\sigma_{Be}}{N_U\sigma_U}$$

Then, the minimum doubling time[7] for a breeder is obtained when

$$a = \frac{\eta - 2}{2} = 0.18$$

$$g = \frac{\eta}{2} = 1.18$$

$$k = 2.00$$

Under these conditions one reactor core has the following values: height, 5 ft; diameter, 5½ ft; 2½ tons of beryllium in the core; 440 beryllium rods, 2 in. diameter, on 3-in. centers in a triangular grid; 670 parts of U^{233} per million parts of bismuth; 14 kg of U^{233} in the core.

The thermal liquid fuel reactors have been emphasized because they represent a smaller extrapolation of our present knowledge than the fast neutron reactors, but this is not a necessary limitation. It may not be possible to reach the concentrations needed for fast reactors in a simple solution of uranium in bismuth, but suspensions of solids containing uranium in liquid metals may prove useful. An ideal suspension, or slurry, is one that is composed of a solid and a liquid having the same densities. The solid must be completely wetted by the liquid. A type of slurry which has been investigated and which has promising properties has USn_3 (solid) suspended in lead-bismuth-tin. The intermetallic compound USn_3 has essentially the same density as most lead, bismuth, and tin alloys and has been found to be thermodynamically stable with respect to the range of compositions of lead, bismuth, and tin solutions. Moreover, fine dispersions (0.5- to 5-μ particle size) have been produced on a laboratory scale, and a uranium content of 0.5 to 5 per cent can be obtained depending on the particle size. The stability of these fine dispersions with respect to growth at elevated temperature is being studied. The effects of radiation will also be investigated. These dispersions have other important properties: density, 10 g/cm^3; solidification at about 100°C; and low capture cross section of components other than uranium.

An apparent advantage of a fast U^{233} breeder is the larger number of neutrons produced per neutron absorbed by the U^{233}. If it is assumed that there is no parasitic capture at high neutron energies, η is 2.59 instead of 2.36, which should help the breeding gain. Preliminary estimates of the uranium concentration indicate that the fast reactor would be about 10 times as concentrated as the thermal reactor, so that the specific power would be only about one-tenth that of the thermal reactor. The over-all breeding rate of the fast reactor may, as a result, be less than that of the thermal reactor. However, the fast breeder with a slurry type of liquid fuel may still be a significant development.

Stability of Thermal Reactors

A preliminary calculation of the temperature effect on reactivity was made on the assumption that all cross sections varied as $1/v$. The change in reactivity is caused by the relative expansions of the tank, the beryllium, and liquid bismuth. This leads to a value

$$\frac{dk}{dt} = -2 \times 10^{-4}/°C$$

which can be compared with the values $-2.5 \times 10^{-5}/°C$ for the Experimental Breeder Reactor (EBR); $-6 \times 10^{-4}/°C$ for the Homogeneous Reactor Experiment (HRE) at 50°C; and $-2.5 \times 10^{-3}/°C$ for HRE at 250°C.

This indicates that the calculations on the stability of HRE should apply qualitatively to the liquid bismuth reactor. The power level would follow the demand smoothly, and any reactivity changes would be damped out quickly.

The negative temperature coefficient is composed of two parts, a large negative term from the expansion of the liquid partially compensated for by the expansion of the tank. For very rapid reactivity changes, the former part will be immediately effective, whereas the tank expansion will be delayed by heat transfer. Thus from the point of view of sudden runaways, the liquid reactor has an effective negative temperature coefficient even larger than that shown above. Mills[8] classified runaways as "dynamic" or "quasi-static." The latter applies to the case where the rate of power rise is slow compared to the velocity of sound across the reactor core. A rough calculation shows that the bismuth reactor could never have a dynamic runaway. Coupled with the immediate temperature coefficient, which is negative and large, this should make the reactor very safe from nuclear disasters.

CIRCULATION OF FUEL

The power of the reactor is proportional to the temperature rise and the rate of circulation of the bismuth. Several alternate methods of circulating the fuel have been considered: thermal convection alone, thermal convection aided by a lift consisting of gas or light liquid pumped into the bismuth, and mechanical or magnetic pumping. The heat exchanger design is greatly affected by the type of circulation because its capacity is a strong function of the circulation rate.

Conventional Type of Heat Exchanger

For the simplest case we consider a conventional tubular heat exchanger and thermal convective flow (see Fig. 2).

The heat-transfer calculations were based on the following assumptions:

1. A pressure head of 0.1 ft of bismuth (0.43 psi) across the reactor and across the heat exchanger.
2. A temperature rise of 170°C in the bismuth per pass through the reactor.
3. An average temperature of 400°C in the reactor.
4. Kinetic losses are small compared to friction losses (an assumption which may be incorrect).

These, together with conservative assumptions with respect to the heat-exchanger tubes themselves, led to the following results: U^{233} in the heat exchanger, 110 kg; total U^{233}, 124 kg; total bismuth, 190 tons; heat produced, 450 megawatts; 3600 kw per total kg of uranium; cycling time for the fuel, 10 sec; velocity of bismuth in the reactor, 5 ft/sec; burn-up time, 280 days.

Because kinetic losses were neglected, this set of figures may be too optimistic for the case of unaided convective flow. It is a self-consistent set of values for the given flow rate, but to attain this rate some pumping may be necessary. Nonuniformity of heating of the bismuth during its passage through the reactor, resulting from the fact that some of the metal may be held up by surface-drag effects, should be self-corrective because the more it overheats the less its density and viscosity will be and the faster it should flow up and out of the reactor.

The heat-absorbing liquid in the exchanger is mercury, which flows by thermal convection from the heat exchanger to a flash boiler. The mercury vapor is cooled in a steam superheater and then is condensed in a boiler condenser. The steam will be at about 125 psig and 500°F, which is suitable for direct use in a turbine. The mercury intermediate coolant can possibly be eliminated, with the heat being transferred

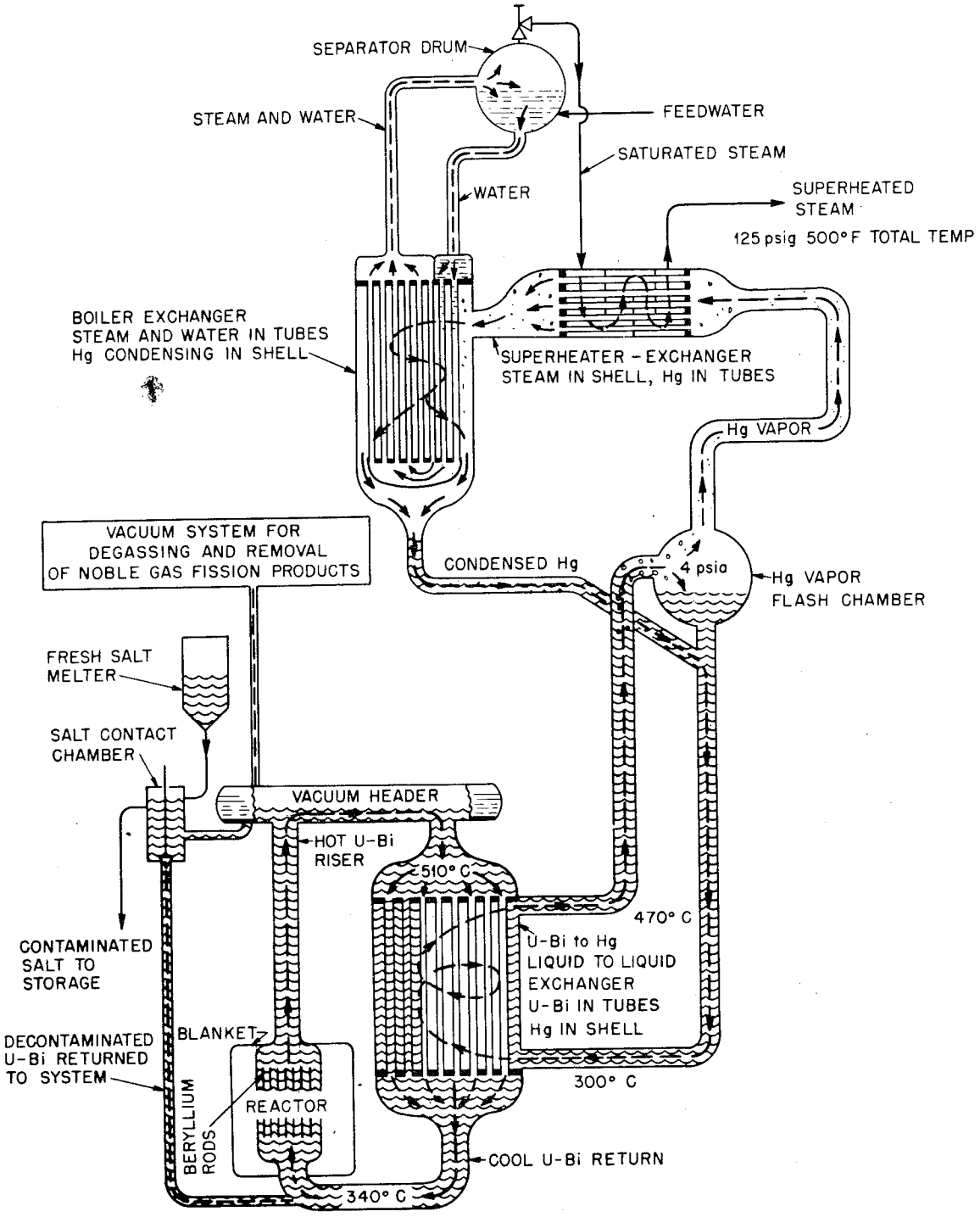

Fig. 2—Simplified flow diagram for power system with thermal convective flow and tubular heat exchanger.

directly from the bismuth to the steam. This procedure, however, has the disadvantages of (1) exposing the bismuth-filled tubes to high pressures and (2) producing radioactive steam. These disadvantages notwithstanding, some consideration is being given to this alternate method of heat removal, possibly involving the use of concentric tubes with an intermediate fluid.

On this basis the minimum doubling time turns out to be 1600 days. This value will, of course, be increased by other losses and holdups. Thus, if we take account of the time required for protactinium decay, the minimum doubling time is increased by about 15 per cent. If to this is added a holdup of one month, estimated for the recovery of protactinium or uranium from the breeder blanket, the combined effect is to increase the minimum doubling time by about 25 per cent. In the case of fission-product poisons the experimental results on the extraction of fission products from bismuth with fused salts, discussed below, indicate that the ratio of poison to uranium absorption may be kept as low as 1.5 per cent. This would increase the doubling time by about 10 per cent. Other neutron losses which must be considered include losses through the ducts; loss of delayed neutrons outside the core; and losses to the container walls, to nonfertile materials in the blanket, and to fissile materials that have been formed in the blanket and have not yet been removed. A combined neutron loss of 10 per cent in the container and the blanket would have the following effects on the breeder: The concentration of uranium in the bismuth should be raised from 670 to about 950 ppm in order to minimize the effects of these losses. The doubling time would still be increased by a factor of 2.3. If neutron absorption in the container and blanket reaches 0.265 of the absorption in uranium, breeding becomes impossible. The need for keeping container and blanket losses low is thus highly important. Preliminary calculations indicate that about 75 per cent of the leakage neutrons will leave the core before they are slowed down to thermal energies, which will reduce the capture by the container wall. It may also be possible to increase the fast neutron leakage by omitting the beryllium rods from the outer region of the core close to the wall. In this region thermal neutrons would be captured in uranium, giving fast neutrons, some of which would return to the moderated part of the core while others would pass through the tank wall while still fast. In any case it will be necessary to have an additional moderator in the blanket that will be kept away from the wall.

Finally, a large part of the above discussion has been based on the use of the liquid fuel reactor as a U^{233} breeder. Since sufficient U^{233} probably will not be available, this reactor would be started up using U^{235} as fuel. This would require the uranium concentration to be about 870 ppm, and a total of 290 kg of U^{235} would be needed to operate the reactor for 280 days. At the end of this time, assuming an optimistic doubling time of about 1600 days, a full charge of U^{233} would have been produced and the same amount of U^{235} would have been destroyed. The core could then be emptied, and the remaining 165 kg of U^{235} could be returned. The U^{233} would then be placed in the core, and the reactor would proceed as a breeder. If U^{233} should turn out to be more valuable than U^{235}, however, it would be more economical to operate as a converter rather than as a breeder. In this case the optimum reactor core should be larger and more dilute in order to increase specific power at the expense of neutron economy.[7]

All the discussion thus far has been based on the example of a liquid fuel reactor core which has a conventional heat exchanger and utilizes thermal convection for circulation of the fuel. The results must be considered as rough approximations because of the assumptions involved in the calculations. However, they are good enough to indicate the possibilities of a liquid fuel reactor and provide a base line for comparing alternate designs. From the standpoint of process design the most serious shortcoming of this type of reactor complex is the large holdup of fuel in the heat exchanger as compared to that in the reactor core. This is due both to the low circulation resulting from thermal convection and to the high volume of the conventional heat exchangers. Thermodynamically, the lowering of the temperature from the 500°C uranium-bismuth solution, which leaves the reactor core to the 260°C (500°F) steam produced, is not desirable. On the other hand, the mechanical simplicity which results from convective thermal circulation is advantageous, and the use of a conventional heat exchanger would seem to require less development. From

the viewpoint of over-all efficiency, however, the need for improvements in the system is apparent.

An obvious improvement is forced convection of the fluid fuel, i.e., the use of a pump to increase the rate of flow. A reasonably sized pump would probably reduce the holdup in the heat exchanger from 8 or 10 times that in the core to about the same amount. This would reduce the doubling time by a factor of 3 or 4. In the case of forced convection it is necessary to provide for shutdown cooling by thermal convection.

sion products. Part of the heated salt goes to a holdup chamber where the heat from the radioactive decay of the fission products raises the temperature of the salt before it goes to the boiler. The rest is pumped into the liquid fuel stream which leaves the reactor vessel. This removes some heat and some of the freshest fission products and, being the hottest salt, is used to superheat the steam. Some salt is removed at intervals to radioactive-waste storage and is replaced by new salt. Although the use of fused salts in the heat-transfer portion of the system may introduce further corrosion prob-

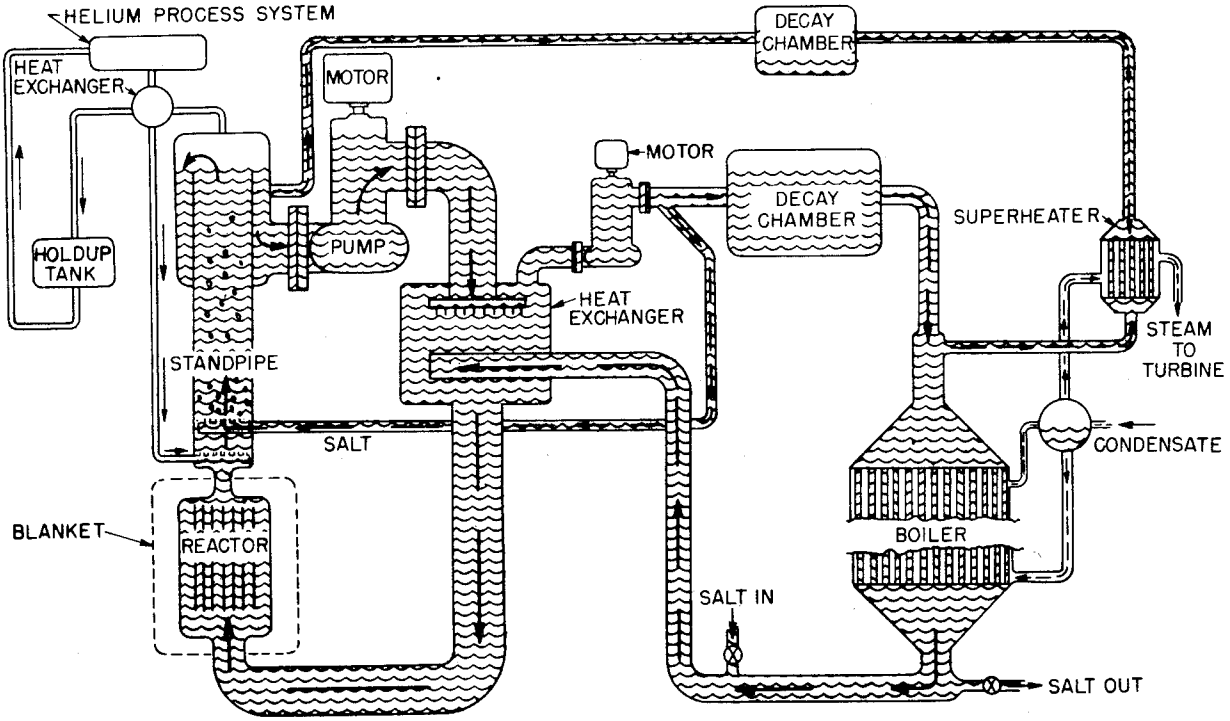

Fig. 3— Simplified flow diagram for power system with liquid lift and mechanical pumping and heat exchange to molten salt in spray type heat exchanger.

Alternate Heat Exchanger, Droplet Type

It may be possible to avoid the conventional type of heat exchanger and at the same time to combine the heat exchanger with the chemical processing. A flow diagram of a system under consideration is shown in Fig. 3. The liquid fuel is pumped to the heat exchanger where droplets are formed and contacted with a fused salt mixture, which serves to remove both heat and fis-

lems, the advantages of combining high-efficiency heat transfer and chemical processing with the accompanying decrease of the holdup outside the reactor make such a system worthy of consideration.

CHEMICAL PROCESSING

One of the main attractions of a liquid fuel reactor is the possibility of simple and efficient

chemical processing of the fuel. Since there is no radiation damage to the fuel, there is no limit to the burn-up on this account. The growth of fission product poisons reduces neutron economy.[9] For the thermal-neutron reactor operating at 0.3 per cent uranium burn-up per day, the major poisoning effects are caused by the 9.2-hr Xe^{135} and the stable rare earth fission products. It should be possible to remove Xe^{135} by continuous degassing and to remove the rare earths by a newly developed form of solvent extraction.

Gas Removal

In the first example above, the uranium-bismuth fuel would circulate through the system at 4 to 6 cycles/min. As the heated fuel rises from the reactor and passes over to the exchanger, it will flow over the bottom of a partially filled chamber, the upper section of which will be maintained at a moderately high vacuum. The reduction of xenon poisoning will depend on the fraction removed in each cycle as shown in Table 1. (The Xe^{135} yield was obtained from the U^{233} fission yield curve and is slightly higher than from U^{235} fission.)

Table 1—Xenon Poisoning

Xenon removed per 10-sec cycle, %	Xenon poisoning, %
0	6.1
0.1	4.5
1.0	1.4
10.0	0.17
25.0	0.07

Under conditions of 10 per cent efficiency, giving 0.17 per cent poisoning, the Xe^{135} concentration would be $\frac{1}{10}$ part per billion, and the total xenon concentration (all fission isotopes) would be ½ part per billion. Since the solubility of noble gases in bismuth at 500°C under low pressure should be very low, the efficiency of removal will probably depend on the rate of diffusion to the free surface. It would therefore seem desirable to supply additional surface by bubbling helium through the liquid. This would also aid the natural convection if used as in Fig. 3. About 30 per cent of all fission products are rare gases at some time, and gas removal will reduce their concentrations. The rate of removal of xenon from molten bismuth is being studied experimentally.

The removal of Xe^{135} would be greatly aided by removal of its parent I^{135}. The solubility of iodine and bismuth iodides in bismuth is being studied. Polonium can be removed also by a vacuum degassing cycle.

Extraction with Molten Salts

For the removal of the rare earth fission product poisons, extraction with fused salts offers an ideal process. The salt is the eutectic mixture of lithium and potassium chlorides (41 mol. % KCl, m.p. 352°C). When this is contacted with the uranium-bismuth solution containing rare earths, the following results are obtained: The bismuth dissolves in the salt to the extent of only 1 to 5 ppm. The lithium and potassium of the salt are found in the bismuth at about the same concentration (1 to 5 ppm). The two liquids are thus virtually immiscible. The uranium remains in the bismuth at 400 to 1300 ppm with not more than 2 to 4 ppm (the limit of the analytical method) transferred to the salt. The rare earths, however, are extracted by the salt.

The results of about 200 tests[10] may be summarized as follows: U (bismuth)/U (salt) > 600, rare earths (bismuth)/rare earths (salt) < 0.01, polonium is not removed by the salt.

This shows that a very simple batch or continuous extraction system will keep rare earth poisoning down with low uranium losses. A number of cases have been calculated;[10] for example, if about 10 per cent of the fuel is processed each day with 90 per cent removal of the rare earths, the total rare earth poisoning can be held to about 0.4 per cent, whereas without processing the poisoning would build up to about 5 per cent after two years of operation at 0.35 per cent burn-up per day. The saving in fissionable material can be calculated for the 450-megawatt unit with 190 tons of bismuth containing 670 ppm of uranium. If 19 tons, or 10 per cent of the bismuth, are treated each day, 90 per cent of the rare earths can be removed by using 2 tons of fused salt. Of the 126 kg of uranium in the whole fuel system, 2 g will be lost by solu-

tion in the salt. During this period (one day) 80 g of excess uranium would be produced, assuming an optimistic doubling time of 1600 days. The loss would then be 2.5 per cent of the excess uranium produced. With no processing and a 5 per cent poisoning, 23 g of excess uranium would be lost per day to production. The net saving by fuel processing is, therefore, 21 g of excess uranium per day. Although the assumed value of a 1600-day doubling time is optimistic, even with longer doubling times the saving in uranium due to fuel processing is significant.

Thus far, no reuse of the fused salt has been assumed. For the distribution ratios found experimentally, recycling of the salt is a very probable method of operation leading to further economy. The apparent limit here is the rate of heat output of the fission product–fused salt solution. The effluent salt, 2 tons or 250 gal/day, will have to be cooled until the short-lived fission products have decayed. As a matter of fact it may be a useful source of heat. After cooling, it might be desirable to purify the salt for reuse. This could be done by a simple oxidation and slag removal.

The reuse of the salt would cause further reduction of the waste volumes, providing reduced costs of storage and treatment of wastes.

A study is now being undertaken to determine what effect, if any, the presence of fission product iodine and bromine will have on the uranium losses in the processing cycle.

Salt Extraction of Uranium

Addition of about 1 per cent beryllium chloride to the salt causes essentially complete extraction of the uranium into the salt. This offers a way to remove the uranium if this is necessary, e.g., to reduce the concentration or to remove objectionable heavy isotopes of uranium caused by successive neutron captures.[11]

Slagging

Large batches of bismuth are purified commercially by treating with various oxidants at about 500°C and removing the slag. Uranium and polonium, which are not extracted by the salt, would be removed by this treatment. This process would be used as a less desirable alternote if the salt extraction proved unfeasible for any reason. It could also be used like the $BeCl_2$ to remove uranium.

Blanket Processing

If a liquid fuel reactor is used to produce U^{233} from thorium, either as a breeder or converter, it is important that the product be removed frequently. The holdup of U^{233}, added to the unavoidable 40-day holdup as Pa^{233}, increases the doubling time.[12] In addition, fission of the newly formed uranium adds to the heat-removal problem.

We are therefore working on a blanket of ThF_4 from which the U^{233} or Pa^{233}, or both, would be continuously removed as volatile fluorides in a fluorine-helium stream. The great economy would result from leaving the thorium fluoride in place continuously and essentially processing only those atoms which had captured neutrons.

A number of ThF_4 preparations and extraction conditions have been tried, and it has been found that the PaF_5 can be removed at a satisfactory rate (τ = one month) at 600°C and the more volatile UF_6 is removed as low as 450°C. In all these cases, however, the rates have fallen off with time, presumably because grain growth or sintering results in slower diffusion. It has been found that the particle size can be reduced and the rate increased by converting the fluoride to oxide with steam and then reconverting to fluoride with HF. This conversion, carried out *in situ*, offers a way to regenerate the blanket and to counteract the effects of grain growth.

The equilibrium

$$ThF_4 + 2H_2O \rightleftharpoons ThO_2 + 4HF$$

is temperature dependent. By varying the temperature and gas composition, it is possible to regenerate the solid at convenient temperatures, 300 to 500°C. The $HF-H_2O$ mixture can be condensed and reused without separation.

The blanket around a core with high power and high leakage (about half the neutrons) would generate a large amount of heat, roughly 2 per cent of the total. Removal of this heat in the helium-fluorine gas stream would require large equipment. A more easily cooled blanket would be a slurry of ThO_2 in D_2O.

The same series of reactions, conversion to fluoride, removal of UF_6 with fluorine, and con-

version to oxide, could also be used to process irradiated ThO_2. For example, a ThO_2-D_2O slurry could be filtered, dried, and processed with gases as above. All operations would be remote, and only small amounts of fluorine would be consumed. Such a process would be, of course, only semicontinuous.

METALLURGY AND MATERIALS

The metallurgy of materials is of great importance in a liquid fuel reactor system. The fuel system itself has already been discussed, and satisfactory fuels are either available, as in the case of the uranium-bismuth solution, or show promise, as in the case of suspensions.

The requirements placed on materials for the reactor system are determined by the chemical environment and temperature-stress conditions to which they are subjected. Heretofore, liquid bismuth has not been used extensively as a heat-transfer medium in industrial applications. Some experience in the handling of bismuth has been obtained in the purification of bismuth where batches of the order of 600,000 lb are reacted with various oxidants at 480 to 540°C. These vessels have usually been made of cast iron and have a life of three to four years. Their failure is apparently due to growth with attendant progressive oxidation from the outside. There is, however, but little experience with which to predict the specific effect of liquid bismuth on various structural materials.

From the work which has been done at Brookhaven National Laboratory (BNL) and at various other Atomic Energy Commission, industrial, and university laboratories, it is possible to say that certain materials appear promising at this time, namely, beryllium or graphite for the moderator, chrome-molybdenum silicon steels or molybdenum for the reactor, and molybdenum or 347 stainless steel for the salt equipment. These recommendations are based largely on static tests which may not give a true picture for these applications. There are indications that only dynamic tests incorporating a temperature gradient give significant corrosion results. A fair amount of work has been done on dynamic bismuth systems at American Smelting & Refining Co., the laboratories at Berkeley and Stanford, and at the California Research Corporation. Some cases of plugging have been noted where a temperature gradient existed. It is questionable whether this work is pertinent to systems which have uranium dissolved because it is possible that the presence of this component may change the nature of the corrosion attack and the surface characteristics of the container walls. Work on the dynamic systems has been started. It is planned to study the effect of addition agents, which, it is hoped, will act as inhibitors. An extensive development program is required in the field of the metallurgy of materials for a liquid fuel reactor.

COMPARISON OF URANIUM-BISMUTH WITH AQUEOUS HOMOGENEOUS REACTORS

In the first issue of this journal J. A. Lane[13] discussed the advantages and disadvantages of aqueous homogeneous* reactors. It is interesting to use his list to evaluate uranium-bismuth reactors.

Advantages.

1. High specific power: both reactors share the advantage that there is no limit of rate of heat transfer within the fuel.

2. High power output: both reactors are limited by the size and fuel holdup in the external heat exchanger.

3. High burn-up: generally burn-up is limited by radiation damage or fission poison build-up. There is no radiation damage in liquid bismuth, and the proposed chemical process should hold the fission-product poisons to satisfactorily low limits.

4. High neutron economy: the capture cross section of bismuth, $\sigma_a = 0.032$ barn, is not as low as that of D_2O but is low enough to allow breeding.

5. Simple fuel-reprocessing systems: the processes outlined above are very simple and offer practically ideal solutions to this requirement.

*A true homogeneous reactor has no structure in the core; the fuel and moderator are combined in a fluid. A fluid fuel reactor may have a stationary moderator structure in the core through which the fluid fuel flows. Since the flow of the moderator is generally less important than that of the fuel, these two types have many advantages and problems in common and can be considered together.

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6. Less complicated control system: both reactors appear to be inherently self-regulating. Reactivity can be controlled by changing the uranium concentration.

7. Inherent safety: the continuous removal of fission products cuts down the inventory of radioactivity in the reactor and thus minimizes the danger from any disaster.

8. Utilization of recombination energy: this does not apply to the bismuth system since it is not decomposed.

9. Simple internal structure: the core of the thermal liquid fuel reactor is not as simple as that of a homogeneous reactor. The additional moderator structure, however, does not have to be leaktight or very carefully dimensioned.

10. Easier fuel handling: this great advantage is shared by both bismuth and aqueous homogeneous reactors.

Disadvantages.

1. High cost of moderator: the cost of moderator is roughly the same for beryllium plus bismuth as for heavy water. If graphite is substituted for beryllium, this cost is substantially less.

2. Induced activity of pumps and heat exchangers: both systems have induced activity of pumps and heat exchangers, though one possible arrangement (see Fig. 2) avoids the use of pumps.

3. Corrosive nature of reactor solutions: the corrosive problems are very different but must be solved for both reactors.

4. High cost of pressure shell: the uranium-bismuth solution does not require an expensive pressure shell because the only pressure is the head of liquid metal.

5. Gas problems from decomposition of moderator: like advantage 8, this does not apply to the liquid bismuth reactor.

In addition to Lane's list there are some other special aspects of the uranium-bismuth system which should be considered.

The heat is available at a convenient temperature for recovery of useful power.

Fission products are produced in concentrated form suitable for use or for cheap storage.

Polonium is produced in quantity as a possible valuable by-product. This may be considered as a disadvantage, but it does not appear to add to the fission product hazard already present. If polonium is to be separated, its value should be enough to pay for the additional precautions which would then have to be taken.

On the other hand, the limited concentration in uranium-bismuth solutions (and even in the slurries) makes it necessary to use U^{233} or enriched U^{235}. The fertile material, U^{238} or Th^{232}, cannot be present in the core without making it very large and exceeding the solubility in bismuth. Thus, natural uranium as a fuel does not seem practical. Conversion or breeding must therefore take place in an external blanket. Even the enriched reactor is necessarily large, containing tons of bismuth. It cannot be scaled down easily to make a small reactor experiment like HRE.

The start-up of such a reactor will be difficult, involving melting the bismuth, dissolving the uranium, and filling the system while hot. After the reactor has run, the presence of fission products will keep the fuel melted even after shutdown, but then the problem is to have dependable cooling during shutdowns.

This comparison of aqueous and liquid metal fuels shows relative advantages and disadvantages on both sides. The aqueous system, of course, has been developed much further. On the basis of present knowledge, both should be pushed because the advantages which they have in common may be important for economical nuclear power.

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