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**THE PHYSICS OF THE FUSED-SALT
REACTOR EXPERIMENT**

By
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February 22, 1957

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THE PHYSICS OF THE FUSED-SALT REACTOR EXPERIMENT

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THE PHYSICS OF THE FUSED-SALT REACTOR EXPERIMENT

ABSTRACT

In 1954 the Oak Ridge National Laboratory operated, for demonstration purposes, a high-temperature circulating-fuel reactor, employing as fuel enriched uranium fluoride dissolved in molten fluorides of other cations. A BeO moderator was used. At a maximum power of 2.5 Mw, 100 Mw hr were obtained. A negative temperature coefficient resulting largely from fuel expansion yielded exceptional stability of the reactor, and made the reactor a slave of the power demand. Xe^{135} and some of the other fission fragments are not retained in the fuel; hence, there is no xenon poisoning and relatively little danger of spreading of gaseous fission fragments in case of an accident. The advantages of liquid-fuel reactors (elimination of radiation damage and thermal stresses in solid fuel elements, ease of fabrication and fuel reprocessing) are combined with low pressures at very high temperatures.

I. GENERAL DESCRIPTION

The Fused-Salt Reactor Experiment was operated at the Oak Ridge National Laboratory between October 30 and November 12, 1954 in order to demonstrate the feasibility of a high-temperature circulating-fuel fused-salt reactor. The basic ideas for this type of reactor are as follows: A fused salt is used to dissolve a salt of a fissionable material. The solution, the "fuel", has to have a high boiling point to permit operation at high temperatures, say, around 1200°F. In the reactor, the fuel is heated internally by fissions and this reduces the radiation damage problem which might become quite serious with solid fuel elements. The internal heating of the liquid salt also eliminates the problems of heat transfer, and associated stresses, with which solid-fuel elements have to contend. The fuel circulates through the reactor and carries the heat to a heat exchanger, where it gives off its heat to some other medium, which gets activated to a considerably lesser extent than if it had entered the reactor itself to pick up the heat. Compared to aqueous homogeneous reactors the fused-salt reactor can reach higher temperatures at low pressures, but the high melting points of the salt are some disadvantage. Compared to a liquid metal reactor, the fused salt reactor can obtain high concentrations of fissionable material.

Low neutron absorption cross section is one requirement on the elements which make up the salt. This, in a general way, points toward the light elements. As to the anion, considerations of chemical stability point towards the halides, hence fluorine appears to be the logical choice for the anion. As to the cation, the choice is not as unique, and, indeed, mixtures of fluorides of various cations are used, as they have lower melting points than pure fluorides.

All conceivably useful cations are either no better as a moderator than the fluorine, or they can be used only in relatively small concentration, and hence the moderation obtainable from the fluorine determines to a large extent the total moderation by the fuel itself. The fission-neutron age to thermal in the best fluoride fuels is about the same as the corresponding age in graphite. This means that very large reactors, such as those potentially useful for central power stations, could be operated as homogeneous reactors, with no other moderator in the core than the fuel itself; but for a reactor of a size suitable for a reactor experiment some good moderator had to be employed. The high temperature ruled out many of the common moderators, and beryllium oxide was chosen.

Figure 1 shows a photograph of a critical assembly of the Fused-Salt Reactor Experiment. The arrangement is essentially the same as in the reactor itself. The cylindrical assembly was made up mainly of hexagonal beryllium oxide blocks. The inner portion, up to a radius of 42 cm, constituted the core. The outer ring, 19 cm thick, was the reflector. The blocks in the core had longitudinal, 3.2 cm diameter holes along their axes, and tubes carrying the fuel were inserted into these holes. The reflector blocks had similar, but smaller, holes. Through these reflector holes liquid sodium was pumped in order to carry away the heat which is deposited in the reflector by neutrons and gamma rays. A large hole in the center accommodated the regulating rod, and three other large holes in the core housed the three safety rods. Instrument holes were provided in the reflector. The height of the core was 91 cm.

The Fused-Salt Reactor Experiment reached a maximum power level of 2.5 Mw (slightly higher during transients). The fuel volume in the core was 37,000 cm³, giving an average power density of 67 w/cm³, in the fuel, which is probably far below the power density that can be achieved in this type of reactor. By the time the reactor was shut down, having fulfilled its demonstration purpose, it had logged about 96 Mw hr.

In the following we shall discuss only those properties of the Fused-Salt Reactor Experiment which are specific to this type of reactor, or are only shown by a small number of other reactor types.

II. REACTOR STATICS

Since the fluorides are not very good moderators, and in any case occupy a large fraction of the core volume, a small-size fused-fluoride reactor is bound to have a substantial fraction of its fissions caused by epithermal neutrons. On the other hand, fluorine has at high energies a sizeable inelastic scattering cross section, which enhances its moderating power at high energies so that a fused-fluoride reactor will never be a very fast reactor.

In the experiment, no means were provided to determine the neutron spectrum and neither was it possible to determine the spatial distribution of the neutron flux. However, in a critical assembly the measurements shown in Fig. 2 were obtained. The critical assembly differed in some minor points from the actual reactor, and was operated at room temperature, but the general features of the measurements are undoubtedly applicable to the reactor experiment itself. The fast flux, as indicated by the "cadmium-covered activation", is highest at the center and falls off rather smoothly with increasing radius; the thermal flux, as indicated by the "bare activation minus cadmium-covered activation", is depressed in the center by the regulating rod mechanism and falls off towards the core-reflector interface, but reaches a peak in the reflector. From the figure it would appear that only about 20% of the activation was due to the fast flux, but the measurements were made with small foils, which did not depress the thermal flux very much. In the fuel tubes the thermal flux is considerably reduced, and hence a sizeable percentage of fissions is caused by epithermal neutrons, as expected for this type of reactor on the basis of the consideration given earlier.

In this connection, we list the means by which the fluoride fuel was represented in the critical assemblies. In the early room temperature critical experiments, including the critical experiment discussed above, the fluorides which were solid at room temperature were used as powders. This made it somewhat difficult to change the uranium concentration to accommodate different loadings. In later experiments uranium foils and teflon sheets were used as mockups for the fuel, and it was somewhat easier to change the ratio of uranium to teflon foils and hence the uranium "concentration". More recently, "hot criticals" have been operated which employed temperatures sufficiently high so that the fluoride salts could be used in the molten state. In these experiments it is relatively simple to change the uranium concentration by adding a concentrated solution of uranium fluoride in another salt.

In fact, the ease with which the uranium concentration can be changed in a fused-salt reactor is an important advantage. Fuel burn-up and fission product poisoning necessitate fuel additions during operation of any reactor. Besides the critical masses of reactors are notoriously not easy to predict and a wrong prediction is less serious when it can be compensated for by simple addition of liquid fissionable material than when it requires refabrication of solid elements.

Actually, the critical mass of the Fused-Salt Reactor Experiment was predicted fairly accurately, after the results of the critical experiment had been available. The experimental value turned out to be 14.9 kg U²³⁵, corresponding to 0.40 g/cm³ of U²³⁵ in the fuel. There is a somewhat diffuse upper limit for the desirable fuel concentration. This limit is about 0.45 g uranium per cm³, for the most common salt mixture. If this upper limit is exceeded, the melting point and viscosity of the fuel solution gradually becomes excess.

III. TEMPERATURE COEFFICIENTS

One of the most desirable features of the fused-salt reactors is their strong negative temperature coefficient. This negative temperature coefficient results in extraordinary stability of the reactor. The most important contribution to the negative temperature coefficient comes from the expansion of the fuel. The fuel salts have thermal volume expansion coefficients of $3 \times 10^{-4}/^{\circ}\text{C}$ (within a factor of 2), and this is considerably more than the volume expansion coefficients of the metals which serve as the walls of the container. Upon heating, the molten salts will expand out of the reactor, leaving less fissionable material, and less scatterer, in the reacting zone. If a fraction $\Delta M/M$ of the fuel mass is expelled from the reactor the reactivity is decreased by a fraction $\Delta k/k$, the ratio $\Delta k/k/\Delta M/M$ being one-third or less for most reactors. For the Fused-Salt Reactor Experiment the ratio was 0.24. Hence, we might expect on the basis of fuel expansion a negative temperature coefficient of $\sim 10^{-4}/^{\circ}\text{C}$ for fused-salt reactors. In the experiment the negative fuel-temperature-coefficient actually was greater, $1.75 \times 10^{-4}/^{\circ}\text{C}$, indicating that effects other than fuel expansion contributed to the coefficient.

The "fuel temperature" coefficient discussed above is observed if the reactor power changes fast enough so that only the fuel and not the rest of the reactor changes in temperature. A slow temperature change in which most parts of the reactor reach thermal equilibrium with each other yielded a negative "reactor-temperature-coefficient" of $1.1 \times 10^{-4}/^{\circ}\text{C}$, that is, only about $2/3$ of the fuel temperature coefficient. An attempt was made to study the influence of the coolant sodium temperature on reactivity. The sodium was cooled rapidly to prevent the moderator to follow its temperature. In this way, the sodium temperature coefficient was found to be negative. In another experiment the sodium was cooled so slowly that the moderator also dropped in temperature, while the fuel temperature was kept constant. This experiment gave the sodium temperature coefficient, plus the moderator temperature coefficient. Since the sodium temperature coefficient was already known, the moderator temperature coefficient could be determined. The moderator temperature coefficient turned out to be positive which explains why the above mentioned "reactor temperature coefficient" is smaller than the fuel temperature coefficient.

For the operation of the reactor, the temperature coefficient of the fuel is the most important since the temperature of the other components of the reactor does not change rapidly. As mentioned, the negative fuel temperature coefficient resulted in a great stability of the reactor. Furthermore, the power generated by the reactor is essentially determined by the power extracted: If the power demand is increased the reactor cools down a little, the reactivity increases, and the power goes up. The reactor is a slave of its power load. It is quite possible that a reactor of this kind could be operated without control rod.

These points were demonstrated in a number of experiments, one of which is illustrated in Fig. 3. The abscissa, to be read from right to left is the time, in minutes, and the ordinate are the print-outs from recorders reading the reactor inlet and outlet temperature. Initially, the reactor operated at low power. Then the heat extraction from the fuel was slowly increased which was first felt as a drop in the temperature of the fuel which reached the reactor inlet from the heat exchanger. This increased the reactivity and the reactor power, as indicated by the temperature rise at the reactor outlet. The spread of inlet and outlet temperatures corresponds to 2.5 Mw power. When the heat extraction was again reduced the inlet temperature rose and the outlet temperature fell until the two temperatures became nearly coincident. The control rods do not determine the power output, they only influence the average temperature. The shim rod insertion dropped the temperature. Rapid increase in the power demand on the fuel system again spread the inlet and outlet temperatures apart, and full insertion and the full withdrawal of the regulating rod depressed, and then raised both temperatures simultaneously. Next, the power extraction was stopped and the regulating rod inserted to make the reactor subcritical.

Then followed the demonstration that the reactor could be brought to criticality, without use of the rods, just by the power demand. It was power extraction from the sodium system

which cooled the reactor to make it critical and then the power extraction from the fuel again caused the spread of inlet and outlet temperature, until the experiment was terminated.

Among other experiments was the introduction of an excess reactivity equal to the excess reactivity held up in the delayed neutrons. The temperature coefficient compensated for this excess reactivity, there were no ill effects.

Most remarkable was the stability of the power level if the reactor was left alone. The power level showed none of the ripple which had been anticipated due to possible mechanical vibration, etc.

IV. ELIMINATION OF Xe^{135} AND OTHER FISSION FRAGMENTS

The remarkable stability made it unexpectedly possible to demonstrate that Xe^{135} was not retained in the fused salt and that there was no xenon poisoning in the Fused-Salt Reactor Experiment. Before the experiment was started it had been computed that the xenon poisoning after 25 hr of operation at full power should amount to 2×10^{-3} in Δk if all the xenon formed were retained. Actually, a 25-hr run at 2.12 Mw showed that the reactor lost no more than 10^{-4} in reactivity during that time, and this was close to the limit of error of reactivity measurements and could be explained by other effects. In any case, the observation puts an upper limit of 5% on the amount of Xe^{135} that might have been retained. Hence, in a fused-salt reactor there will be very little, or no, xenon poisoning or danger of a positive temperature coefficient due to xenon. Little or no excess reactivity has to be provided to override xenon after a shutdown, at least if the fuel can be circulated before startup.

Obviously, if Xe^{135} is not retained in the fused salt, other rare gas fission fragments will escape likewise. It is also possible that additional fission fragments get volatilized from the fuel, or get removed from the reactor by other processes. A considerable amount of fission fragment disappears from the fuel was indicated by the following experiment. A sample of the fuel was withdrawn from the Fused-Salt Reactor after a shutdown following reactor operation at 27 watts for 1 hr. This was early in the experiment and the reactor had only been operating at very low power prior to this 1-hr run. A similar sample was irradiated in the Bulk Shielding Reactor (Swimming Pool) under a power history which duplicated the power history of the Fused-Salt Reactor prior to the withdrawal of the first named sample, except for a proportionality factor of 0.37. The decay of the gamma activity of both samples was measured and is represented in Fig. 4.

Taking into account necessary corrections, but assuming that both samples retained all their fission fragments, the activity of the Bulk Shielding sample should have been 12.4 times as high as the activity of the sample from the Fused-Salt Reactor. Actually the ratio is 27 at 1 hr 40 min after shutdown, and 17.5 at ~38.5 hr. The discrepancy is easily explained by the fact that the Bulk Shielding sample was solid and did not lose any fission fragments whereas the sample from the Fused-Salt Reactor was stripped of some of its fission products. Since the most volatile fission fragments include strong gamma emitters with relatively short lives (such as 2.77 hr Kr^{88} and its Rb-daughter) it can be explained in a qualitative way why the fused-salt sample curve is depressed relatively more at short times after shutdown.

For the safety viewpoint, the loss of fission fragment gases is both a blessing and a curse. Since the volatile fragments are given off during operation and can be released under controlled conditions, or stored where they would not be liberated during an accident, such a hypothetical accident could reach essentially only the non-volatile fragments and could not release much airborne activity. On the other hand, during normal operation the requirements on leak tightness of the system are rather stringent.

V. CIRCULATING SLUGS

The elimination of gaseous fission fragments is undoubtedly due to a large amount of stirring of the fuel, but in spite of this stirring, liquid slugs of fuel retain their identity for several

passages through the reactor and external circuit. This was first noticed during the approach to criticality when U^{235} was added in slugs to barren salt. When the slug of high uranium concentration passed through the reactor the counting rate of the meter temporarily increased, as can be seen in Fig. 5, and the slug can even be recognized when it comes around the second time. A similar phenomenon is shown in Fig. 6 where the fuel circulation had been stopped while the reactor was at same low power, and where the pump started up after a scram. The slug which had been in the reactor while the power was on contained delayed-neutron emitters, and emitters of hard gammas, which can create neutrons by the Be-photoneutron reaction, and every time this slug went through the reactor again, the meters showed a "pip". At least five such passages could be recognized before mixing and decay of activity washed out the picture. Figure 7 shows six or more pips following a scram after a rapid raise in reactor power. The slug which filled the reactor at the moment of the scram is the most active one since the power had reached its crest at that moment.

The pips offered a convenient way of measuring the time it took the fuel to go through the reactor and the external loop, and pumps and flow meters could be calibrated by the use of this phenomenon.

VI. INHOUR EQUATION AND ROD CALIBRATION

The fact that slugs retained their identity through several passages was mildly gratifying because the inhour equation had been computed under this assumption in advance of the experiment. The inhour equation gives the relation between the reactor period and the excess reactivity, and it depends obviously on the delayed neutrons given off in the reactor. In a circulating-fuel reactor some delayed neutrons are born outside the reacting zone and this modifies the inhour equation in a manner which depends on the transit time through reactor and external circuit, and on the question whether fuel which is present in the reactor at a given moment returns to the reactor as a "slug" or gets mixed in the outside circuit with other fuel.

The higher the flow rate, the more delayed neutrons are lost, and the shorter the period corresponding to a given excess reactivity. This caused some misgivings prior to operation of the experiment, but the experiment dispelled these fears, largely because of the negative temperature coefficient, as discussed earlier.

The inhour equation, both with stationary fuel and with fuel circulating, was used to calibrate the regulating rod. This rod had also been calibrated by comparing its effect on reactivity with the effect on reactivity of addition of fissionable material, using the previously mentioned value of $(\Delta k/k)/(\Delta M/M)$. A fairly consistent set of data was obtained supporting the $(\Delta k/k)/(\Delta M/M)$ value as well as the inhour equation for flowing fuel. The regulating rod was worth 0.4% in $\Delta k/k$, or 0.033%/in. in the average over its 12-in. travel. The shim rods were similarly calibrated during fuel addition, and also intercalibrated with the regulating rod. Each shim rod was worth 5.8% in reactivity.

At some time prior to the experiment it was feared that the loss of the delayed neutrons might result in a loss in damping of any power oscillations of the reactor. However, it was possible to demonstrate mathematically¹ that the fuel circulation itself tends to damp oscillations, and the operation of the experiment did now show any tendency of power oscillations.

REFERENCE

1. See figure, H. V. Ergen, Kinetics of the Circulating Fuel, Nuclear Reactor Journal of Applied Physics, 25, 702(1954).

Figure 1

Figure 2

REACTOR INLET AND OUTLET TUBE TEMPERATURES

Figure 3

Figure 4

Figure 5

Figure 6

Figure 7