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Tr. School

PILE TECHNOLOGY

Lecture 39

POSSIBLE APPLICATIONS OF UF_6 IN PILES

Lecture and notes by D. E. Hull

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Lecture 39 of Pile Technology has been expanded and issued by Dr. Hull as a report MonN-336, in order to give the material wide circulation. The following pages, comprising the official lecture notes for Lecture 39, are taken directly from, and are identical with, the body of report MonN-336.

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SUMMARY

It is shown that the use of UF_6 as a pile fuel is feasible from the standpoints both of nuclear physics and of engineering.

A power pile is described which employs UF_6 gas at a temperature of 800° C and a pressure of 10 atm. contained in a porous beryllium metal matrix, moderated additionally by graphite, and cooled by helium at 10 atm. Such a pile would contain 16 kg of 25, and 1100 kg of beryllium, and would deliver 96,000 kw of heat, with a gas outlet temperature of 1250° F and other characteristics similar to the Daniels high-temperature pile. Solid fission products are accumulated in the porous metal, from which the UF_6 may be pumped out quantitatively at suitable intervals into fresh tubes. Possible means for removal of xenon by the circulation of the UF_6 are discussed. A breeding ratio of 0.99 is found for this pile with allowance for maximum poisoning by xenon and samarium, and 1.05 if the xenon is removed. With 23 for fuel, the corresponding figures are 1.18 and 1.24. This does not allow for end losses, for losses in the tank containing the pile, nor for losses in the breeding blanket.

A modification using aluminum instead of beryllium as the porous metal is considered in order to permit discarding the used tubes instead of reworking them by chemical and metallurgical processes. However, this would operate at a lower temperature and at lower specific power, while requiring more fissile material and giving poorer breeding.

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A water-cooled pile using similar porous rods of aluminum and UF_6 at 50 atm. is described. This is designed to give a power of 32,000 kw with 3.2 kg. of 25, and yields a thermal neutron flux of 2×10^{14}. Cooling water, pumped through the pile at 17 ft/sec. undergoes a temperature rise of 25° F and a pressure drop of 5 psi in the pile.

Several types of circulating UF_6 piles are discussed more briefly, in which the UF_6 is liquid, gaseous, or in solution.

A breeding blanket consisting of tubes containing finely divided ThF_4 in a bed of graphite is described. Removal of the 13 and 23 formed by neutron capture in the thorium would be accomplished by slow circulation of fluorine through the hot bed, leaching out volatile PaF_5 and UF_6. Cooling by 10 atm. helium at the surface of the tube is shown to be feasible in a flux of 10^{14}. A breeding efficiency of 96% is calculated for the blanket, exclusive of leakage to the shield or at the ends.

From the preliminary considerations entertained in this paper, it is concluded that the use of UF_6 would have several marked advantages, among them the following:

Ease of separation of fission products

Elimination of chemical and metallurgical treatment of fissile material

Simplicity of remote control operations

Small hold-up of fissile material outside the pile

Possible continuous removal of xenon

Continuous replacement of fuel to compensate for depletion

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Summary (concluded)

Elimination of fission damage to fuel rods

Large depletion factor possible

Small losses in reprocessing

At the same time the UF_6 pile compares not unfavorably with other thermal piles from the standpoints of neutron flux, specific power, critical mass, and breeding gain. The questions of stability of UF_6 under radiation and of corrosion resistance of structural materials, while appearing favorable from present information, require further study before it can be concluded that the UF_6 pile is feasible also from these standpoints.

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POSSIBLE APPLICATIONS OF UF_6 IN PILES

D. E. Hull

I. INTRODUCTION

Uranium hexafluoride offers certain obvious advantages for use as a pile fuel by virtue of its unique properties. Because of its volatility it would be possible to separate it from most of the fission products without performing any chemical operations upon the uranium. This would eliminate the costly and complicated chemical and metallurgical plants included for reprocessing the fuel in most present pile designs. Such processes would be replaced by purification methods handled by gas pumps, with perhaps some devices for heating or cooling. Loading and unloading would be accomplished similarly by use of pumps. Remote control of such processes would be relatively simple, and high decontamination factors would be unnecessary. The possibility of circulating the UF_6 for continuous removal of fission products is also attractive.

II. CHOICE OF MATERIALS

However, the great chemical reactivity of UF_6 with most materials requires for its use in piles a selection of other pile materials from a narrow field. In addition to meeting the criterion of low cross section for absorption of neutrons, any materials used must also have a high stability against corrosion by UF_6 and fluorine. Also, the possible effects of radiation on UF_6 and other materials used in contact with UF_6 and fluorine must be considered.

The technological problem of handling UF_6 in metal containers at high pressures and temperatures much above critical is well in hand. Also the handling of fluorine at pressures around atmospheric and temperatures up to 200° C is well understood for some metals. An extension of such knowledge to higher temperatures and to metals which can be used in piles will be necessary before an engineering design of a hex pile can be made.

Among the structural materials which might be used to contain UF_6 and fluorine, nickel and monel (a Ni-Cu alloy), which would be best from the corrosion standpoint, are probably ruled out by their high cross sections. (σ_{Ni} = 5 barns; σ_{Cu} = 3.5 barns.) Aluminum, magnesium, and beryllium have cross-sections such that it is worth considering the possibility of using them. (See Table I.)

TABLE I

	σ	Melting Point	Thermal Conductivity
Beryllium	0.01 barns	1280° C	0.2 cal/cm sec deg C
Aluminum	0.24	660°	0.5
Magnesium	0.35	650°	

Aluminum and magnesium are known to have good resistance to corrosion, although not as good as nickel.[2] Their corrosion rates at 150° C are of the order of a milligram/square foot/day. However, their relatively low melting points put them at a serious disadvantage for power pile applications. Of beryllium, which would be more desirable both from the standpoint of cross section and melting point (but less so from the standpoing of cost), less is known in regard to effects of UF_6 and fluorine; but it is to be expected that it will be qualitatively similar to aluminum and magnesium in forming a protective film.

Very few quantitative data on these corrosion rates at high temperatures are available, but it is known that the rate increases rapidly with rising temperature, and it would probably be necessary to work considerably below the melting points. Increase of pressure above atmospheric has little effect on the rate of corrosion, which appears to be limited by the rate of diffusion of ions through the crystal lattice of the protective film.

Among the possible solvents of UF_6 which might be used as moderators in a homogeneous thermal pile are HF, DF, and fluorocarbons. Aluminum and magnesium are rapidly attacked by HF, but nickel is satisfactory to contain such a solution. Fluorocarbons will probably be seriously decomposed by radiation, although there is no experimental certainty as to this point, nor is it certain that the decomposition products, if any, would not merely be other fluorocarbons equally acceptable as a solvent. Among possible coolants, water and liquid metals react vigorously with UF_6, and if their use is considered it must be behind an intervening layer of a material resistant to both UF_6 and the liquid. Fluorocarbons would be worth considering for coolants if their radiation stability were demonstrated. The low cross-section of fluorine as well as the short life of its radioactive isotopes would make it advantageous for this application. Among gases, hydrogen probably should be avoided for the same reason as in other piles, because of its explosive hazard, and it also reacts vigorously with UF_6. Helium, of course, is inert and would be just as useful in a UF_6 pile as in any other.

The use of graphite as a moderator is admissible if it is physically separated from the UF_6, but it cannot be used in contact since it reacts rapidly and completely.

III. RADIATION EFFECTS

The effect of radiation on UF_6 itself has been considered in detail in an earlier report.[3] The conclusions drawn from the rather fragmentary data available from the bombardment of liquid UF_6 by electron beams are:

(1) That liquid UF_6 in a pile would probably develop a fluorine pressure of several hundred atmospheres. This would be much lower at higher temperatures.

(2) That gaseous UF_6, at temperatures above its critical point (230° C) would be only slightly decomposed by radiation, and a pressure of a fraction of an atmosphere of fluorine would probably be sufficient to prevent decomposition.

For these reasons, it has seemed worthwhile to explore more thoroughly the possibility of using gaseous UF_6 in piles, although the possibility of using the liquid is not to be excluded, because of the large experimental uncertainties in the data.

As to the possible effects of radiation on the corrosion rate of UF_6 on metals, nothing is known. In studies of corrosion under the influence of radiation, some tendencies have been observed, as follows:

(1) If the corrosion product tends to be detached from the metal, corrosion is hastened by radiation.

(2) However, if the corrosion product forms an adherent protective film, radiation appears to have no effect.

Now it is well known that in most cases the corrosion of metals by fluorine and UF_6 belong to the second group. It is therefore not unreasonable, in the present absence of experimental data, to proceed tentatively on the assumption that corrosion will not be accelerated by radiation in the case of the metals considered here.

IV. SUGGESTED PILE ARRANGEMENTS

As an illustration of the possible arrangements of piles using UF_6, five different combinations of fuel, moderator, and coolant are set out here. The uranium has been assumed to be 100% 25 except in a few cases, where 23 has been taken. Physical characteristics of these piles are given in Table II.

1. Circulating Liquid UF_6. Pure UF_6 is not a practical possibility for a homogenous pile, because the moderating effect of the fluorine atoms is not sufficient to make it a thermal pile, and because of the low density of uranium atoms (compared to that in a metal pile) an excessively large critical mass, hundreds of kilograms, would be required for a fast pile. A pile of liquid UF_6 moderated by beryllium has been considered. This might consist of a cylindrical block of beryllium metal with holes parallel to the axis containing the UF_6. Such a pile would contain 4.5 kg. of 25 and 725 kg. of beryllium in the reactor. If the circulation of UF_6 be considered as the means of taking out heat, this value would have to be multiplied by an appropriate factor, depending upon the desired power output, to include the amount in the heat exchanger, pumps, and connecting lines. Assuming that in such an arrangement the fission products could be removed continuously, a limiting theoretical breeding gain of 1.05 could be approached.

Indications are that such a pile would develop a large pressure of fluorine at high power levels. The fission product fluorides which are solid might possibly be deposited on the walls of the holes in the beryllium, and their removal from the inside of the pile would be difficult. The problem of instability due to evolution of fluorine bubbles inside the pile fluid, and the difficulty of control due to removal of delayed neutrons with the circulating medium, would have to be faced.

The power which is to be taken from such a pile is limited by the capacity the heat exchanger. Figures of the order of 1 kw/kg of UF_6 have been quoted [4,5] for piles using circulation of ordinary UF_6, both by natural convection and by forced circulation. While this could undoubtedly be pushed up by proper engineering design, it is evident that enriched UF_6 is far too expensive to use as a coolant in any real power pile.

For both these reasons, the use of liquid UF_6 has not been pursued further.

2. Circulating Gaseous UF_6. In order to avoid the radiation decomposition of UF_6, its use in the gaseous state at high temperatures is considered. Temperatures approaching or higher than the critical

TABLE II

No.	State	Temp.	Press.	Moderator	Atoms/Atom U	Structure	Atoms/Atom U
1	Liquid	100°C.	4 atm.	Be	4,200		
2	Gas	230°	20	Be	4,200		
3	Gas	750°	100	He	10,000		
4	Solution	100°	15	DF	5,000		
5	Solution	100°	—	$(CF_2)_n$	3,000		
6	Gas	800°	10	C	6,400	Be	1,800
7	Gas	400°	10	C	11,700	Al	550
8	Gas	260°	50	Be	2,070	Al	126
				H_2O	290		

No.	Volume pile / Volume UF_6	a	k	g	$\mathcal{T}$	L^2	1/ -B
1	200	0.07	1.99	1.05	128 cm^2	34 cm^2	15.3 cm.
2	11	0.07	1.99	1.05	156	44	17.0
3	1	0.00	2.12	1.12	15×10^6	8.1×10^6	5,550
4	1	0.08	1.96	1.04	429	431	35.6
5	1	0.11	1.90	1.01	626	228	36.5
6	7	0.13	1.88	0.99	293	140	26.2
7	17	0.35	1.57	0.77	307	174	32.7
8	20.0	0.39	1.52	0.73	63	12.8	13.4

No.	R_o	α	R	V	M_c	M_m	Power	KW/kg	Flux
1	45 cm.	0	40 cm.	390 l.	4.5 kg	725 kg			
2	50	0	44	535	5.6	900			
3	14,400	0	850	3×10^6	84	14,400			
4	105	0	90	4540	9.6	4,300			
5	108	0	99	5830	16.8	10,700			
6	77	0	69	2000	7.6	2,400			
		0.3	99	5900	15.5	4,900	96,000 kw	6000	9×10^{13}
7	96	0	88	4100	9.8	6,200			
		0.3	126	12000	20.0	12,700	50,000	2500	4×10^{13}
8	39	--	34	240	3.2	260	32,000	10000	2×10^{14}

temperature, 230° C, would probably be effective in reducing the pressure of fluorine to very moderate values.

Here again, because of the low density, a fast pile, without moderator, is out of the question. As a moderator, for a thermal pile, a beryllium cylinder with holes for the gas may be used again to illustrate the magnitude of the quantities involved. At 230°C and 20 atmospheres, such a pile, without provision for cooling other than the circulation of the UF_6 itself, would require 5.6 kg. of 25 and 900 kg. of beryllium. The neutron economy of this pile is the same as in the preceding case. This pile would avoid the difficulties associated with the formation of gas bubbles in a liquid phase. However, it is even more likely that the solid fission products would settle out of the gas on surfaces inside the pile.

The possibility of mixing UF_6 with another gas to serve as moderator, such as helium or CF_4, in a homogeneous gaseous pile has been suggested. For the case of helium, calculations show that very large volumes would be required, even at 100 atmospheres pressure. But with a graphite reflector, the amount of 25 required at a partial pressure of 0.01 atm. is calculated by two-group theory to be less than 100 kg. This would require a spherical reactor with a diameter of 17 meters. A pressure vessel of this size and strength is a little on the fantastic side, but perhaps not entirely so for future consideration. The value of the breeding gain would be the highest possible for a thermal pile, although an enormous blanket would be required. In this pile, without any internal structure, it is quite reasonable to expect that the solid fission products could be collected largely outside the pile, and a system for the continuous removal of xenon is not too difficult to visualize.

3. <u>Circulating Solution of UF_6</u>. To obtain a homogeneous pile in the liquid state with UF_6, it would be necessary to dissolve it in an inert solvent which could act as moderator. One such possibility is a liquid fluorocarbon. A typical pile of this kind would contain 16.8 kg. of 25 and 10,700 kg. of fluorocarbon. The pile fluid would be suited to a continuous fractionation process for removal of fission products, both gaseous Xe and I and solid fission product fluorides. However, the decomposition of the fluorocarbons would probably complicate this process, both by introduction of gaseous fluorine at a considerable pressure, and by formation of higher and lower boiling fluorocarbons.

A system using DF as a solvent would probably be immune to radiation damage. This solvent would have to be held under pressure, however, as the vapor pressure of HF at 100°C is about 15 atm. A pile made of these components would contain 9.6 kg. of 25 and 4300 kg. of DF (which would require about 2000 kg. of heavy water for its production). This system could not be contained in aluminum, but nickel would be satisfactory chemically, though with some sacrifice in breeding gain from the theoretical figure of 1.04, and magnesium might be satisfactory.

UF_6 piles of the solution type do not offer nearly as great an advantage from the standpoint of chemical reprocessing over homogeneous piles of the aqueous solution type as over metal fuel rod types from the standpoint of chemical reprocessing, since chemical cleanup in the aqueous type can be accomplished by either extraction or ion-exchange processes, quite conceivably on a continuous basis and with maintenance of a low level of fission product poisons. Also, the use of D_2O has a marked advantage from the standpoint of heat removal.[6]

V. <u>A Gas Cooled Power Pile with Gaseous UF_6 in Porous Metal Rods</u>.

A pile arrangement which appears to most advantageously exploit the properties of UF_6 is one in which the gas is contained in the pores of a porous metal rod, encased at the ends and sides in an impervious metal shell. Such a rod may be filled with UF_6 gas, together with F_2 at a small partial pressure, and used in the pile like a conventional solid fuel element. The rod would be made by powder metallurgy techniques with pores small enough that the UF_6 would be in good thermal contact with the metal at all points. With small enough pores, the fission fragments formed in the gas would impinge upon the metal surface and dissipate most of their energy inside the metal. Thus most of the heat would be produced in the metal, and also the radioactive fission products would be lodged in the metal, thus automatically separating them from the unburned UF_6, which remains in the gas. Those fission fragments finding the end of their range in the gas would be deposited later on the surface of the metal, except for those which form gaseous products. After such time as the accumulation of fission products or corrosion of the fuel rods made it desirable to replace them. purification of the UF_6 would be accomplished simply by pumping it from the used tube into a fresh one, This would remove all the solid fission products and all but a small fraction of the gaseous ones. After any treatment that might be found necessary, such as, for example, passing fluorine through the tube to take off the last traces of UF_6 which might be in an adsorbed or reduced layer, the spent rod might be discarded or used for the extraction of fission products.

An alternative application of this scheme would be to use larger pores, such that most of the fission fragments would come to rest in the gas phase. The solid fission products would still be collected on the surface, but now it is possible to consider a slow circulation of the UF_6 through the porous tube, carrying with it the iodine and xenon, to a point outside the pile where the poison could be removed continuously. This offers the advantage of the greater neutron economy associated with the absence of the high cross-section xenon.

The porous rod pile appears to have interesting enough possibilities that it has been explored in more detail than those described above. These considerations are set out in the following:

A. <u>Structure</u>. The unit comprising the fuel element in a tube of beryllium, with a porous core 1 cm. in diameter containing 75% open space, enclosed by a solid shell of the same metal 1.1 mm. thick. Metals with as high as 90% open space can be made by powder metallurgy methods. The mechanical strength of the metal with such high porosity is low, but strength is of no importance in the present application. The porous rod is pretreated to remove impurities, then fluorinated to put on a protecting film, then filled with UF_6 in such quantity that at a uniform temperature of 800° C its pressure will be 10 atm., plus a small pressure of elemental fluorine. The moderator consists of graphite rods located at the center of a triangular lattice of fuel rods, or of annular tubes of graphite around the fuel tubes, and occupies a volume in the pile 2.5 times that of the fuel rods. 30% open space in the pile structure is attained by spacing the fuel tubes 2.6 cm. center to center. This open space is used for the flow of a gaseous coolant, helium. The pile is surrounded by a breeding blanket of graphite and thorium, described below.

B. Nuclear Properties. With the constituents of the pile in these proportions, the pile constants work out as follows, without voids:

TABLE III

	Density	Moles/cc.	Volume	Atoms	$N\sigma_a$	$N\sigma_t$	$N\xi\sigma_s$
UF_6	0.00383	0.000114	1	1	645 (565)	---	---
Be	1.86	0.206	1	1800	18	6,650	1510
C	1.7	0.141	5	6200	28	28,000	4770
FP's					38		
Total			7	8000	729	34,600	6280
Macroscopie cross-sections (cm^2/cm^3)					0.0071	0.336	0.061

TABLE IV

Fuel	U^{235}		U^{233}	
Fission Products	Kept	Removed	Kept	Removed
Parasitic losses: a	0.13	0.07	0.14	0.08
Multiplication: k	1.88	1.98	2.03	2.15
Naximum breeding: g	0.99	1.05	1.18	1.24
Age of fast neutrons τ	293 cm^2	293	293	293
Diffusion length L^2	140 cm^2	148	157	168
Buckling $1/\sqrt{-B}$	26.2 cm	25.5	25.2	24.5

[redacted]

Now, using a void fraction of 0.30, the dimensions of a bare cylindrical pile, based on the constants in the first column above, are 110 cm. radius and 203 cm. height. With the breeding blanket described below, these figures are reduced to 99 cm. radius and 192 cm. height (taking only half the calculated saving at the ends, to allow for heat removal). The volume of this pile reactor is 5900 liters and it contains 15.8 kg. of 25. There are 5150 tubes, each containing 3.07 grams of 25. The formulas used in these calculations are illustrated in Appendix I.

It is seen from the above table that the theoretically possible breeding gain is only slightly less than unity in the non-circulating pile, and if provision is made for continuous removal of the fission products, the gain may be pushed above 1. For comparison, the corresponding figures for a pile using $U^{233}F_6$ are given. When this material becomes available, breeding gains greater than unity ought to be attained readily. There is only a small effect on the critical dimensions of the pile, whether the fission products are kept or removed, and whether 23 or 25 is used as fuel, as shown by the close similarity of the values of the buckling coefficient.

C. <u>Power</u>. The power which can be drawn from this pile is calculated by use of the formula given by Amorosi.

$$KW = 336\,\alpha\,D^2 P \Delta T \sqrt{\frac{R\,\theta}{T_p\,T_b}} \sqrt{\frac{\eta_B}{F_c\,F}} \quad \frac{\frac{\gamma}{\gamma-1}\sqrt{\frac{9\gamma-5}{2(\gamma-1)MW}}}{B_r\sqrt{B_a}}$$

α is the percent voids.
D is the diameter of the pile core. Its value depends upon the value chosen for α in such a way that the specific power of the pile (kilowatts per kilogram of fissile material) is a maximum when $\alpha = 0.6$. However, this leads to a value of the critical mass 6 times as great as the minimum value of Table II. At a value of $\alpha = 0.3$ the specific power is 75% of the maximum, and the critical mass is only twice its minimum value. This seems an acceptable compromise, making D = 6.5 feet.
P = 10 atm. is the pressure of the coolant.
$\Delta T = 750^\circ$ F is the temperature rise of the coolant in the pile.
R = 0.15 is the fraction of the power output devoted to pumping coolant.
$\theta = 360^\circ$F is the temperature difference between the fuel rod surface and the coolant at the center of the pile.
$T_p = 1335^\circ$ R (875°F) is the average temperature of the coolant in the pile.
$T_b = 960^\circ$ R (500° F) is the temperature of the coolant in the blower.

$\sqrt{\frac{\eta_B}{F_c F}} = 0.23$ is a factor involving blower efficiency and pressure drops.

$\frac{\gamma}{\gamma-1}\sqrt{\frac{9\gamma-5}{2(\gamma-1)MW}} = 3.4$ is a factor involving only the ratio of specific heats and the molecular weight of helium.

[redacted]

$B_r = 1.5$ is the ratio of the maximum to average radial flux of thermal neutrons.
$B_a = 1.25$ is the ratio of maximum to average axial flux of thermal neutrons.

These figures are all taken as close to the Daniels high-temperature pile as reasonably possible. The chief difference is the smaller outlet coolant temperature, taken as 1250° F instead of 1400°, as a concession to the properties of beryllium.

Putting in these values we find for the pile described a power of 96,000 kw, which corresponds to a specific power of 6000 kw/kg. The average flux corresponding to this power level is 9×10^{13}.

D. Temperatures in Fuel Rods. The conductivity of the 75% porous beryllium rod is taken as 1/4 the conductivity of solid beryllium, in line with the experimental fact that the electrical and thermal conductivity of a porous metal is proportional to its density.

$$K = 0.20 \times 0.25 = 0.05 \text{ cal/cm-sec-deg C.}$$

The average heat flux in the pile may be computed from the average density of uranium of (0.027 g/cc, times the open space of 0.59 cc/cm length of rod, equalling)0.016 g/cm , and the specific power of 6000 w/g or 1440 cal/sec-g to be 23 cal/sec-cm along the fuel rod, or 7.3 cal/sec-cm^2 through the fuel rod surface. The temperature drop across the beryllium shell then is

$$\Delta T = 7.3 \times 0.1/0.2 = 4^\circ \text{ C.}$$

Inside the porous structure, the formula for temperature distribution in a medium of constant heat production per unit volume may be used, which, for the case of a cylinder, takes the form

$$T = q\, r^2/4\, K$$

Here q is the quantity of heat produced in cal/sec-cc.

$q = 0.027 \times 1440 = 39$ cal/sec-cc in the pores, or
$q = 39 \times 0.75 = 29$ cal/sec-cc average throughout the porous structure. Hence the temperature rise calculated for this case is

$$T = \frac{29 \times (0.5)^2}{4 \times 0.05} = 36^\circ \text{ C.}$$

between the inside of the shell and the center of the rod. These values are very moderate, and do not give rise to any important thermal stresses.

Now if the UF_6 were of uniform density in the porous rods throughout the pile, the maximum flux and temperature drops would be 1.87 times the figures quoted above. However, an interesting effect comes in at this point, which is unique to gaseous piles. Assuming that uniform pressure will obtain throughout the porous rod, the differences in temperature will give rise to differences in density of UF_6 from point to point. This will tend to drive the UF_6 from the hotter part of the rod ot the cooler part, and will tend to raise the rate of production of heat in the cooler part of the rod. The net effect of this will be to reduce the temperature differentials in the rod. The magnitude of this effect has been calculated for the radial distribution of temperature and density within a rod, assuming uniform neutron flux over this small region. At the hottest part of the pile, where the temperature of the shell is taken as 800° C, the temperature at the center of the rod would be 867° with uniform density, but only 860° with the density distribution according to 1/T. The effect, rather small in this case, will be much more pronounced in the axial direction, where larger temperature gradients obtain. The effect has not been calculated in this case, but qualitatively it is easy to see that the result will be to shift UF_6 from the hotter end of the rod, toward the cooler end, thus heating the coolant faster at the end where it enters, and slower where it leaves the pile. A more uniform temperature will thus extend throughout the pile. This will make possible a higher specific power than that calculated above, but, of course, it will also require a greater quantity of 25 for criticality, since the uranium is concentrated in the pile at points of lower neutron flux, on the average. If this effect turned out to be very desirable, it could be further exploited by extending it to cover the radial distribution throughout the pile by connecting the fuel rods together by a manifold at either end to permit pressure equalization throughout the entire porous rod assembly.

The temperature rise in a fuel rod in case of failure of the coolant circulating system is of interest. Neglecting the contribution of the UF_6, the heat capacity of 1 cm. of the porous beryllium rod is 0.47 cal/deg. The maximum rate of heat production is 2700 cal/g-sec 0.016 g/cm = 43 cal/sec in each cm. of rod. Without cooling, the temperature rise would be 43/0.47 = 90° C. per sec. This is not so fast but that the pile could be shut down with control rods before the fuel elements burned out.

D.1 <u>Pressure stresses on fuel rods</u>. While the pile is designed to operate with approximate pressure balance between the gaseous UF_6 inside the shell and the coolant outside, it is important to know what pressure the shell will withstand under conditions of filling or loading, or of abnormal operation. The tensile strength of beryllium at 800° C is of the order of 10,000 psi. Applying the usual formula for a thin cylindrical shell, the internal pressure required to rupture the 1 mm. thick 1 cm. diameter shell would be

$$\frac{2 \times 10{,}000 \times 0.1}{1} = 2000 \text{ psi.}$$

This provides a large safety factor over any foreseeable stresses.

[redacted]

[redacted]

E. Range of Fission Fragments. This quantity is interesting because of its relation to the fraction of the gaseous fission fragments left in the UF_6. Taking the experimental value of the range of the recoils in aluminum as 0.00137 cm., the range in UF_6 under the conditions assumed in this pile can be computed by multiplying by the ratio of the numbers of atoms per cc. and by the ratio of stopping powers.

$$R = 0.00137 \times \frac{0.10}{0.000114} \times \frac{1}{7.67} = 0.156 \text{ cm.}$$

Similarly the range in beryllium is

$$R = 0.00137 \times \frac{0.10}{0.206} \times \frac{1}{0.55} = 0.0012 \text{ cm.}$$

A comparison of these figures shows that a pore size of 0.1 mm. will permit almost complete entrainment of fission recoils in the metal, but only the outer 1/10 of the metal particles will be utilized and subject to fission damage.

F. Circulation of UF_6 through the Fuel Rods. The alternative method mentioned, involving the use of large pores and trying to keep the fission fragments in the gas, makes it necessary to consider the maximum size of UF_6 filled pores which will permit sufficiently good thermal contact between the center and the metal boundary to prevent dissociation of the UF_6. Considering a cylindrical shaped pore of 0.05 cm. radius containing UF_6 with a thermal conductivity of 2×10^{-5}, producing heat at the average rate of 1440 cal/g-sec, the temperature rise from the metal surface at 860°C to the center of the pore, taking into account the change in density, is 740°. Actually the pores would have to be about 10 times this large in order to contain most of the fission fragment ranges, for which size the calculated temperature rise is 11,900°. Hence it appears that if conduction is the only means of heat transfer, pores large enough to permit retention of gaseous fission products would result in greatly overheating and dissociating the UF_6. If thermal radiation were found to be an important mechanism of heat transfer at a temperature lower than the dissociation temperature of UF_6, this design might still be practicable.

It is easy to show that the pressure drop across such a circulating system would be nominal. Using the equation for pressure drop in a fluid flowing through a porous bed, with a circulating rate of 1 ft/min. and a pore size of 0.1 mm. the pressure drop turns out to be only 0.4 atm. With larger pores, the pressure drop would be smaller in proportion to the square of the pore size.

Also, it appears feasible to clean up the UF_6 from xenon and iodine with this rate of circulation. Table V shows the boiling points of the substances involved:

Table V - Boiling Points

UF_6	56° C. (Sublimation temperature)
Xe	-109°
IF_7	5°
F_2	-187°

From a mixture of these gases the xenon could be simply removed by condensing the UF_6 in a cold trap at dry ice temperatures and pumping the xenon on through. The fluorine in the system would be removed by this process, and would have to be replaced with the UF_6 is pumped back into the pile. The iodine, which would exist in the gas as IF_7, would not be so simply removed, but with the difference in boiling points between it and UF_6, a fractionating column ought to be feasible. If six minutes be allowed for the process, and a capacity of 1/10 of the UF_6 in the pile for the traps, the pile could be stripped of its xenon and parent iodine on the average once an hour, giving 99% elimination of the xenon poisoning. If only the xenon, and not the iodine, where removed, the whole pile would have to be processed every six minutes to keep the poisoning down even to 10% of its maximum value, at a flux of 10^{14}.

G. Adsorption of UF_6 on the Walls. It has been suggested that this effect might enhance the concentration of UF_6 within the rod at a given pressure. A calculation shows that this effect would be negligible. Even with pores of only 0.001 cm. radius, giving a surface of about 750 cm^2/cc, a unimolecular (which would be expected at these pressures) film of UF_6 molecules, each with an area of 2×10^{-13} cm^2 would put only 4×10^{15} molecules per cc in the adsorbed film, whereas there are 6×10^{20} in the gas in the same volume.

J. Porous Aluminum Rods. The use of aluminum for the porous metal for containing the UF_6 offers the attractive advantage of cheap dispensability of the spent rods, after pumping off the UF_6. A pile constructed with this material is illustrated in the figures in line 7 of Table II. With the same size fuel rods it is necessary to use a much larger proportion of moderator to keep the critical mass within reasonable limits. With this arrangement, the neutron economy is, of course, very much poorer. The power to be taken from this pile is sharply limited by the low melting point. Taking a maximum fuel rod temperature of 400° C (750° F), which is probably optimistic, and dropping the blower temperature to 350° F, a coolant temperature rise of 300° can be used. With a corresponding value of 144° for θ, the power of the pile is found to be 50,000 kw. This corresponds to a specific power of 2500 kw/kg and an average flux of 3×10^{13}.

The lower power obtainable with the aluminum rods is an inherent disadvantage, which would have to be balanced in economic considerations against the cost of chemical and metallurgical treatment of beryllium

[redacted]

rod pile. The greater conductivity of aluminum could be used to advantage in making a higher porosity, perhaps as high as 90%. A thinner shell on the tube, or higher pressure of the UF_6 would improve the breeding. Circulation of the fuel gas for removal of xenon, more frequent substitution of fuel rods, and the use of 23 instead of 25 could all be considered.

VI. A WATER COOLED HIGH-FLUX PILE WITH UF_6 GAS IN POROUS METAL RODS

A variation of the porous metal rod pile to provide for water cooling as an approach toward a high flux has also been considered in some detail. Such an arrangement is found to be feasible only at higher pressures of UF_6. Although a high power output is involved, the heat is taken out in liquid water at too low a temperature to have any important economic value. Also, breeding gain is sacrificed in this arrangement as being of secondary importance. If neither high temperature nor high breeding is considered, aluminum is a logical choice for the fuel matrix because of its low cost and ready availability. It is found that the porous metal must have a higher density than in the cases discussed above, to carry off the heavy heat load safely.

A. Structure. The fuel element in this pile consists of a 1 cm. diameter porous aluminum rod with 50% open space, encased in a 0.5 mm. aluminum shell. The pores in the metal, smaller by a factor of 5 here than in the pile above, are filled with UF_6 at a pressure of 50 atmospheres and a temperature at the boundary of 260° C (30° above the critical temperature). This whole unit is to be slipped snugly into a retaining beryllium cylinder with 5.5 mm. wall thickness. These elements are mounted in a hexagonal pattern with 3 cm. center to center separation. Beyond a 3 mm. annulus around each tube, the interviewing space is filled with solid beryllium. The tubes will be cooled with water in these annuli flowing at 17 feet/sec. The whole pile will be surrounded with a 12" beryllium reflector to keep the neutron distribution in the reactor fairly uniform. Such a device may be necessary in order to keep the rate of heat evolution uniform throughout the pile and prevent condensation of UF_6 near the edges. The difference in temperature between the liquid water outside the beryllium and the gaseous UF_6 inside is maintained by a heat flux across the beryllium tube and the water film of 150 watts/cm^2.

B. Nuclear Properties. The proportions chosen and the resulting cross-sections are summarized in Table VI and on line 8 of Table II.

TABLE VI

	Density	Moles/cc.	Volume	Atoms	N_a	N_t)th	N_t(f	N_s
UF_6	0.40 g/cc	0.00114	1	1	645	neg	neg	neg
Al	2.7	0.10	1.44	126	30	neg	neg	neg
Be	1.86	0.206	11.46	2070	21	7,700	7,700	1,740
H_2O	1.00	0.055	6.05	290	164	16,900	2,120	6,400
FP's					38			
	Totals		19.95	2487	898	24,600	9,820	8,140
	Macroscopic cross-sections (cm^2/cm^3)				0.308	0.844	0.337	0.279

The beryllium reflector reduces the critical radius from 39 to 34 cm, and makes the reactor volume 240 liters and the critical mass of 25, 3.25 kg. Each centimeter of fuel rod contains on the average 0.39 cm^3 of UF_6, or 1.05 g. of 25. Each tube, running the length of the pile, 72 cm, contains 7.6 g. of 25. The total number of tubes in the pile is 430.

C. Power and Fuel Rod Temperatures. Operation of the pile at a specific power of 10,000 kw/kg corresponds to a production of 250 cal/sec. cm. along the fuel tube, or 150 watts/cm^2 at the outer surface of the tube. This makes the total power of the pile 32,000 kw, and its flux 2.1x 10^{14}.

At this power, the rate of heat production in the body of the fuel rods is 320 cal/sec-cm^3 and with a conductivity of the porous aluminum of 0.25, the temperature at the center of a rod rises above that at the edge, 260° C, to 329°. The temperature drop through the aluminum shell is 8° and through the beryllium 141°. Thus the temperature at the outside of the fuel rod is 111° C, or 232° F. To prevent boiling against this surface requires only a small pressure head.

Using water at 100° F for cooling, the heat transfer coefficient comes to 3600 and this, with an equivalent diameter of the coolant stream of 0.24 in. requires a velocity of 17 ft./sec., according to the formulas given above. The water undergoes a temperature rise of 25° F and a pressure drop of 4.5 psi in going through the pile reactor.

With cooling water at a low pressure, some special precautions might have to be taken in starting up this pile to get the UF_6 into the gaseous state before turning up the power, if the production of fluorine from the condensed UF_6 proved to be troublesome. Also, the range of power might be limited by the necessity of maintaining the temperature drop across the beryllium, although the self-regulating effect of rapidly increasing heat production at a point of incipient condensation might prove to be very useful in the operation of this pile. If it turns out to be necessary, these limitations could be removed by use of 750-pound water, which boils only at 266° C, above the critical temperature of UF_6. Such hot water would serve to preheat the fuel rods on starting up, and also would permit a continuous range of levels up to maximum power.

Since beryllium has a tensile strength of 30,000 psi at these temperatures, the 5.5 mm. wall is abundantly able to bear the high pressure of UF_6.

The rate of temperature rise in case of failure of coolant circulation is calculated from the heat production rate in 1 cm. of rod, 250 cal/sec., and the heat capacity (including the water) in the same cm. of rod, 5.4 cal/deg., to be 46° C/sec.

VII. BREEDING BLANKET

The breeding blanket used as a reflector with the power pile considered in the foregoing is conceived to be made of graphite with hollow cylinders of thick beryllium containing finely pulverized thorium fluoride. A hot mixture of fluorine and helium is allowed to flow through the powder in the tubes. The fluorine serves to convert transmutation products to fluorides, and the helium, present in larger amounts, serves to cool the ThF_4. If the particle size of the powder is small enough, and if the temperature is high enough, it is to be expected that the PaF_5 and UF_6 formed by neutron capture and decay of the thorium will diffuse through the solid and be volatilized at the surface, being then carried away by the circulating gas. This would be the only chemical treatment which the thorium would require. Its gradual depletion could be made up merely by filling up the tube with fresh material. The PaF_5 and UF_6 could be separated in a fractionating column, perhaps with continuous operation under fluorinating conditions to remove the UF_6 as fast as it is formed by the decay of Pa. This would be in a form immediately useful for fuel in a UF_6 pile employing 23.

Beryllium is chosen to contain the breeding material both because of its low cross-section and because it appears likely to withstand the high temperatures which will probably be required for diffusion in the solid ThF_4. Beryllium tubes, 2 cm. in diameter and 0.5 mm. wall thickness, would be fitted into graphite in a traingular pattern 4 cm. center to center. The relative properties of the constituents of this blanket are shown in Table VIII.

TABLE VIII

	Density	Moles/cc.	Volume	Atoms	N_a	N_t
Th	4	0.0078	1	1	6.5	13
F	---	---	---	4	0.04	16
Be	1.86	0.206	0.2	3	0.03	11
C	1.7	0.141	3.2	35	0.16	157
TOTAL			4.4	43	6.73	197

These figures lead to a breeding efficiency of 6.5/6.73 = 0.965, which represents the fraction of the neutrons absorbed in the blanket which will be useful in converting fertile atoms to fissile atoms. This must be further reduced, of course, to allow for escape of neutrons through a blanket of finite thickness. The mean square diffusion length of slow neutrons, L^2, is 8.0 cm. in this medium, and the mean free path for transport, λ_t, is 2.86 cm. These figures have been used in calculating the pile savings in the foregoing sections.

VIII. CONCLUSION

A. Advantages of UF_6 Piles

From the preliminary type of considerations of feasibility discussed in this paper, it is evident that the use of UF_6 for a pile fuel, in comparison with other types of piles now receiving most study, would offer numerous advantages, some of which would be of great economic value. At the same time, the UF_6 pile does not compare unfavorably in respect to flux, specific power, critical mass, and breeding gain. Some of the advantages are common to fluid piles in general, others are unique to UF_6.

1. One of the most important advantages of this type of pile is the elimination of the damage from fission fragments. The loss in thermal conductivity and mechanical strength in solid fuel rods does not appear here. Such damage as is produced by the radiation is expected to be self-healing.

2. Temperature stresses, which are a limiting factor to the power in many piles, are reduced to negligible proportions here.

3. Fluid fuel may be loaded and unloaded by pumping. In some of the piles described above, this would have to be augmented from time to time by replacement of the fuel matrix.

[redacted]

4. Circulation of the fuel permits continuous removal of xenon, giving the advantages that go with higher neutron economy and avoiding the difficulties associated with the shut-down of high-flux piles.

5. With the fission damage eliminated and with the possibility of replenishing the fuel during operation, it becomes possible to go to a very high depletion ratio before replacing the fuel rods. Probably the limit on the length of time a fuel rod can be kept in the pile will be set by corrosion rates, or, if these can be sufficiently reduced, by the accumulation of stable fission product poisons of low cross-section. In such case, 50% depletion is not unreasonable.

6. The separation of the UF_6 from its fission products without the need of chemical operations is an outstanding advantage. Thereby the expensive processes of chemically handling "hot" material and of metallurgical procedures are eliminated. The pumping and fractionation operations which take the place of these more complicated procedures are simply handled by remote control.

7. Since the metallurgical operation is excluded, high decontamination is unnecessary; 99% removal of the fission products would be quite adequate.

8. With all operations handled by remote control, the long cooling period associated with most reprocessing schedules is eliminated.

9. For long time operation, the gradual accumulation of heavy isotopes of uranium becomes an important factor in diluting the fissile material. This may ultimately lead to the need for isotopic separation in some cases, and here the use of UF_6, which is the form of uranium processed by the separation plants, holds a natural advantage.

10. Without the involved reprocessing and the cooling periods, the time necessary to purify the UF_6 and return it to the pile becomes a small fraction of the time it spends in the pile. Thus the total inventory of uranium required for operation of the pile is only a fraction greater than the quantity actually in the pile. This fact should be kept in mind in comparing critical masses calculated in this paper with those stated for other types of piles. The calculations of specific power, also, would appear to be on a more rational basis were the power of the pile divided by the total quantity of fissile material required to keep it in operation. On this basis, the UF_6 pile is ahead of other piles by a large factor.

11. The simple method of recovering UF_6 from the spent rods will result in low recovery losses. In the first place, the amount

[redacted]

of UF_6 from the spent rods will result in low recovery losses. In the first place, the amount of UF_6 left on the walls after pumping out a porous metal is at most a fraction of a monolayer, which is not more than 0.001%. Any corrosion product not already picked up by the fluorine in the system is easily and quantitatively removed by fluorination at 200° C. This gives reason to hope that recovery losses can be pushed even below 0.001%, although the possibility remains that recoil of unburned U atoms from fission fragments may drive a small fraction of them into the interior of the metal.

12. The breeding gain of these piles is comparable with that of other piles. However, the net breeding gain during an extended period of operation involves two additional factors, the fraction of depletion allowed before reprocessing, and the loss per cycle in reprocessing. Since depletion can be large with a UF_6 pile, the number of cycles is small, and the loss per cycle being also small, the over-all breeding gain will be better than in other piles with similar pile constants.

B. <u>Disadvantages</u>. 1. The difficulty of handling fluorine and UF_6 immediately leads to resistance in the minds of many persons, almost to the point of prohibiting any consideration of these materials in piles. However, experience with these compounds in the isotope separation plants shows that this is more of a bugaboo than a real drawback. It is true that they are dangerous materials, but no more than many other materials handled every day in the chemical industry. With proper precautions they may be handled with a high degree of safety. The disadvantage on this score lies more in overcoming the widespread prejudice against handling fluorine than in the actual use of the material.

2. The fact that in these piles the fuel material and a considerable fraction of its fission products exist at a high pressure is a definite disadvantage. It means that any accident would result not only in loss of the fissile material, but in scattering radioactivity. Therefore, such piles must be built with safety factors assuring that no such accident will occur.

C. <u>New Problems</u>. Several avenues of investigation must be explored before the conclusion can be drawn that UF_6 piles are practical.

1. Corrosion rates on metals of interest by fluorine and UF_6 throughout a wide range of temperatures must be determined. Means for reducing the corrosion rate to minimum values will be most valuable.

2. The decomposition of UF_6 by radiation must be studied under a wider range of conditions than heretofore, particularly in the gas at high temperatures.

[redacted]

3. The phase relationships of UF_6 and fission product fluorides, with and without solvents such as DF and fluorocarbons, must be explored.

4. The transfer of heat through gaseous UF_6, by radiation as well as by conduction, must be investigated

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APPENDIX I

DETAILS OF PILE CALCULATIONS

Number of moles of UF_6 per cm^3 at 10 atm. and 800° C:

$$\frac{10 \text{ atm.}}{82 \text{ cm}^3\text{-atm/mole-deg.} \times (800 + 273) \text{ deg.}} = 0.000114 \text{ moles/cm}^3$$

Number of moles of carbon per cm^3:

$$\frac{1.7 \text{ g/cm}^3}{12 \text{ g/mole}} = 0.141 \text{ moles/cm}^3$$

Number of carbon atoms per uranium atom:

$$\frac{0.141}{0.000114} \times \frac{5}{1} = 6200$$

Number of atoms of uranium per cm^3 of UF_6

$$0.000114 \times \text{moles/cm}^3 \times 0.6 \times 10^{24} \text{ atoms/mole} = 0.000068 \times 10^{24}$$

Macroscopic absorption cross-section:

$$\Sigma_a = \frac{0.000068 \times 10^{24} \text{ atoms/cm}^3 \text{ of } UF_6}{7 \text{ cm}^3 \text{ of pile/cm}^3 \text{ of } UF_6} \times 729 \times 10^{-24} \text{ cm}^2\text{/atom} = 0.0071 \text{ cm}^2\text{/cm}^3$$

Parasitic losses:

$$a = \frac{18 + 28 + 38}{645} = 0.13$$

Multiplication factor:

$$k = \frac{2.12}{1.13} = 1.88$$

Breeding gain:

$$g = 2.12 - 1.13 = 0.99$$

Age of fast neutrons;

$$\tau = \frac{6}{0.336 \times 0.061} = 293 \text{ cm}^2$$

[redacted]

Diffusion length of thermal neutrons:

$$L^2 = \frac{1}{3 \times 0.0071 \times 0.336} = 140 \text{ cm}^2$$

Buckling coefficient:

$$1/\sqrt{-B} = \frac{293 + 140}{\ln 1.88} = 26.2 \text{ cm}$$

Radius of bare pile:

$$R_o = 2.95 \times 26.2 = 77 \text{ cm}$$

Height of bare pile:

$$H_o = 1.84 \times 77 = 142 \text{ cm}$$

Pile savings:

$$R_o - r = \frac{140}{\ln 1.88} \tan^{-1} \frac{\ln 1.88}{140} \times \frac{8.0}{2.86 \times 0.336} = 8 \text{ cm}$$

Volume of zero-power pile:

$$V_o = (69)^2 \times 134 = 2.0 \times 10^6 \text{ cm}^3$$

Critical mass of zero-power pile:

$$M_o = \frac{2.0 \times 10^6}{7} \times 0.0267 = 7.6 \times 10^3 \text{ g.}$$

Radius of power pile

$$R_p = \frac{69}{0.7} = 99 \text{ cm.}$$

Critical mass of power pile

$$M_p = \frac{7.6}{(0.7)^2} = 15.5 \text{ kg.}$$

Volume of power pile

$$V_p = \frac{2000}{(0.7)^3} = 5800 \text{ l.}$$

Specific power:

$$\frac{96{,}000}{15.5} = 6000 \text{ kw/kg}$$

[redacted]

Average value of flux:

$$\frac{3 \times 10^{10}/\text{watt-sec} \times 96 \times 10^{6} \text{ watts}}{545 \times 10^{-24} \text{ cm}^2/\text{atom} \times 0.0000097 \times 10^{24} \text{ atoms/cm}^3 \text{ of pile} \times 5.8 \times 10^{6} \text{ cm}^3} = 9 \times 10^{14}$$

TABLE

Nuclear Constants Used in Pile Calculations

	σ_a	$\sigma_t)_{th}$	$\sigma_t)_f$	$\xi\sigma_s$
H_2O	.62	58	7.3	22
D_2O	.0023	11.2	8.6	5.7
DF	.010	7.6	6.4	3.0
He	0	1.25		.63
C	.0045	4.5		.77
CF_2	.024	12.2		1.57
Be	.01	3.7		.84
Al	.24	1.33		.10
U^{235}	645	8		.07
U^{233}	565	8		.07
Th	6.5	13		.07

APPENDIX II --- OPTIMUM VALUE OF VOID FRACTION

The void fraction α may be varied in pile design to optimize the specific power, KW/M. The critical mass M, because of its dependence on the cube of the pile buckling factor, which goes as $\frac{1}{1-\alpha}$, and also on the density, which goes as $1 - \alpha$, varies as $\frac{1-\alpha}{(1-\alpha)^3} = \frac{1}{(1-\alpha)^2}$. D, the pile diameter, is proportional to $M^{1/3}$. Hence the factor $\frac{\alpha D^2}{M}$, which is proportional to the specific power, varies with α as $\alpha(1 - \alpha)^{2/3}$. This factor has a maximum value of 0.325 when $\alpha = 0.6$. However, the critical mass at this point is 6.25 times the mass for $\alpha = 0$ (M_o). Inspection of the function $\alpha(1 - \alpha)^{2/3}$ shows that it reaches a value of 0.237 already at $\alpha = 0.3$, where M is only 2 times M_o. This seems to be an acceptable compromise between large KW/M and small M. Hence, $\alpha = 0.3$ has been used in all calculations in this paper.

APPENDIX III

DISTRIBUTION OF GASEOUS UF_6 IN A POROUS PILE ROD

The differential equation for the heat flux in a volume element of a medium throughout which heat is being produced at a uniform rate of q cal/cm^3/sec. and which has a thermal conductivity K cal/cm-sec-deg. is, in cylindrical coordinates,

$$\frac{d^2T}{dr^2} = \frac{-q}{2k}$$

This must be modified for the present case because the UF_6 will distribute itself radially according to the temperature gradient. With a uniform pressure throughout the cylinder, the density of UF_6 will be inversely proportional to the absolute temperature. Since the rate of heat production is proportional to the density of the UF_6, we may write

$$q = \frac{Q}{T}$$

This equation defining Q. Then the differential equation becomes

$$\frac{d^2T}{dr^2} = \frac{-Q}{2kT}$$

This may be integrated by the substitution

$$p = \frac{dT}{dr}$$

$$pdp = \frac{-Q}{2k}\frac{dT}{T}$$

Whence $$p^2 = \frac{-Q}{k}\ln T + \frac{Q}{k}\ln T_c$$

The constant of integration T_c is chosen to meet the boundary condition that

$$\frac{dT}{dr} = 0 \text{ when } r = 0.$$

Solving for p,

$$p = \frac{dT}{dr} = -\sqrt{\frac{Q}{k}\ln T_c/T}$$

[redacted]

The negative root is taken to conform to the physical reality that the temperature gradient is negative.

$$\frac{dT}{\sqrt{\ln\ T_c/T}} = -\sqrt{\frac{Q}{k}}\ dr$$

This is put in convenient form for graphical integration by the substitution

$$x = T_c/T$$

Integration from the center of the cylinder, where $r = 0$, $T = T_c$, to the edge, where $r = r$, $T = T_s$,

$$x \int_1^{T_c/T_s} \frac{dx}{x^2 \sqrt{\ln x}} = \sqrt{\frac{Q}{k}}\ \frac{r}{T_s}$$

This expression has been integrated in the range $1 < x < 1.1$ by the approximation

$$x \int \frac{dx}{x^2 \sqrt{\ln x}} = \sqrt{x - 1} + x \tan^{-1} \sqrt{x - 1}$$
$$= \sqrt{x - 1}\ (x + 1)$$

and graphically in the range $x > 1.1$.

The values of the function are set out in the accompanying graph.

For the particular case of the porous beryllium fuel rod at 800°C, the term on the right has the value, based on the maximum power of the pile, with

$$Q = \frac{2700 \times 235 \times 10 \times 0.75}{82} = 58{,}000 \text{ cal-deg/cm}^3 \text{ sec.}$$
$$k = 0.2 \times 0.25 = 0.05 \text{ cal/cm-sec-deg.}$$

and $r = 0.5$ cm,

of $$\sqrt{\frac{Q}{K}}\ \frac{r}{T_s} = \sqrt{\frac{58{,}000}{0.05}}\ \frac{0.5}{1073} = 0.50$$

[redacted]

The value of x which gives this argument is 1.056. Hence the temperature rise

$$T = 1073 \times 0.056 = 60^{\circ}\ C.$$

The corresponding temperature rise if uniform density were assumed is

$$T = \frac{54 \times (0.5)^2}{4 \times 0.05} = 67^{\circ}\ .$$

The effect is seen to be of almost negligible importance in the radial distribution under these conditions. It will be more important when the factor $\sqrt{\frac{Q}{k}}\ \frac{r}{T_s}$ is larger. For example, with $T_s = 373^{\circ}$ A, and with the same specific power, the value of x is 1.34, and the temperature drop is 127° compared to 164° which it would be if the uranium were not free to move.

APPENDIX IV -- GROWTH OF XENON

Starting with a pile containing no fission product xenon nor iodine, the xenon grows at a rate depending on the flux of the pile and by two mechanisms:

(1) Formed by the decay of iodine $f_1 = 0.056$

(2) Formed directly in fission $f_2 = 0.003$

The rate at which it grows by decay of iodine is given by the impression

$$P_1 = \frac{f_1 \sigma \phi}{\lambda_{Xe} + \sigma \phi} \left[1 - Ae^{-\lambda_I t} - Be^{-(\lambda_{Xe} + \sigma\phi)t} \right]$$

where $A = (\lambda_{Xe} + \sigma\phi)/(\lambda_{Xe} + \sigma\phi - \lambda_I)$

$B = \lambda_I/(\lambda_I - \lambda_{Xe} - \sigma\phi)$

$\sigma = 3 \times 10^{-18}\ cm^2$ = adsorption cross-section for xenon

ϕ = thermal neutron flux

$\lambda_I = 2.87 \times 10^{-5}\ sec^{-1}$

$\lambda_{Xe} = 2.09 \times 10^{-5}\ sec^{-1}$

[redacted]

The rate at which it grows by direct formation in fission is

$$P_2 = \frac{f_2 \sigma \phi}{\lambda_{Xe} + \sigma \phi} \left[1 - e^{-(\lambda_{Xe} + \sigma\phi)t} \right]$$

If the iodine is not removed, but only the zenon, the growth of xenon follows the formula

$$P = \frac{(f_1 + f_2)\sigma \phi}{\lambda_{Xe} + \sigma \phi} \left[1 - e^{-(\lambda_{Xe} + \sigma\phi)t} \right]$$

With a flux of 10^{14}, numerical values of P_1, P_2, and P calculated from these formulas for various times are listed in Table .

TABLE

Time	P_1	P_2	$P_1 + P_2$	P
6 min.	0.0000	0.0003	0.0003	0.0060
12	0.0001	0.0006	0.0007	0.0114
18	0.0003	0.0008	0.0011	0.0162
24	0.0005	0.0010	0.0015	0.0201
30	0.0007	0.0012	0.0019	0.0242
36	0.0010	0.0014	0.0024	0.0276
48	0.0015	0.0017	0.0032	0.0333
1 hr.	0.0021	0.0019	0.0040	0.0378
2	0.0062	0.0025	0.0087	0.0497
3	0.0104	0.0027	0.0131	0.0535

[redacted]

APPENDIX V

HEAT REMOVAL FROM THORIUM BLANKET

Assuming a thermal neutron flux of 10^{14} and an energy release of 7 MEV per neutron captured in thorium, with a cross section of 6.5 barns, the heat produced in porous ThF_4 is

$$0.0078 \times 0.6 \times 10^{24} \times 6.5 \times 10^{-24} \times 10^{14} \times 7 = 21 \times 10^{12} \text{ MEV/sec-cm}^3$$

or

$$0.8 \text{ cal/sec-cm}^3$$

Assuming a value for the thermal conductivity of solid ThF_4 of 0.01, which is a typical value for salts, and assuming that the porous solid is sintered together for good thermal contact, the conductivity would be 0.005. Then the temperature drop across the tube is

$$\frac{0.8 \times (0.5)^2}{4 \times 0.005} = 10^\circ \text{ C.}$$

Thus even if the conductivity is overestimated by a large factor, it is seen that the temperature gradient will be quite reasonable.

To permit the removal of this heat, the blanket structure is opened up to place a 2 mm. annulus around the tube ($\alpha = 0.16$), through which helium at 10 atm. is pumped at 100 ft/sec. At an average temperature of 500° C this corresponds to a flow of 0.4 moles per sec. through each tube. In a tube 192 cm. long the heat production per second is

$$0.8 \times 0.79 \times 192 = 120 \text{ cal/sec.}$$

With a molar heat capacity of 5 cal/deg, the helium will be heated

$$\frac{120}{0.4 \times 5} = 60^\circ \text{ C.}$$

Its pressure drop will be about 1 psi.

The gas used for flushing out the transmutation products from the ThF_4 may be circulated at a rate of 1 ft/hr through a bed of particles a few microns in diameter with a pressure drop of the order of 1 atm. This gas might be chiefly helium, with a small partial pressure of fluorine.

Drawing # 4382

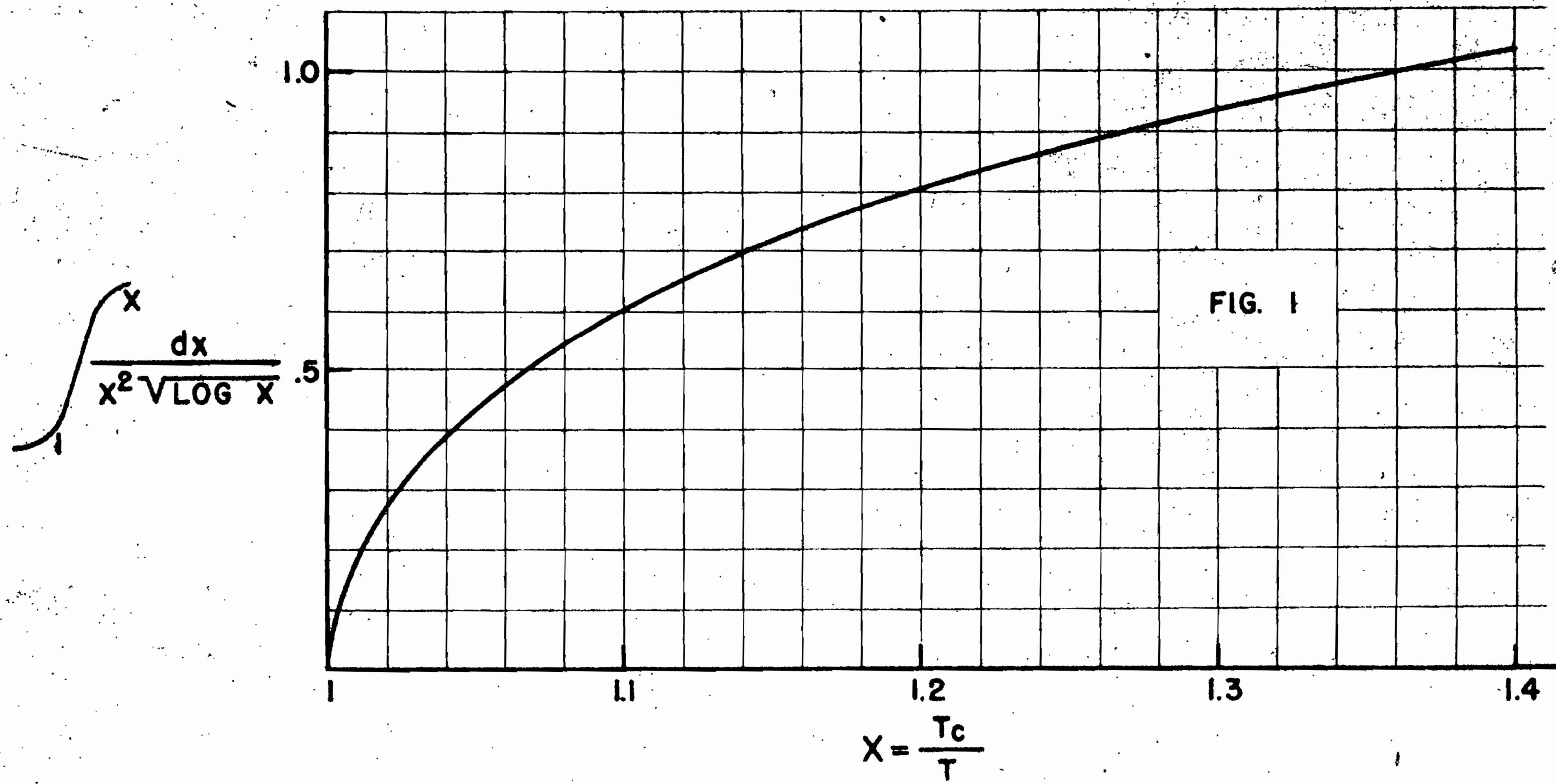

FIG. 1