Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 1

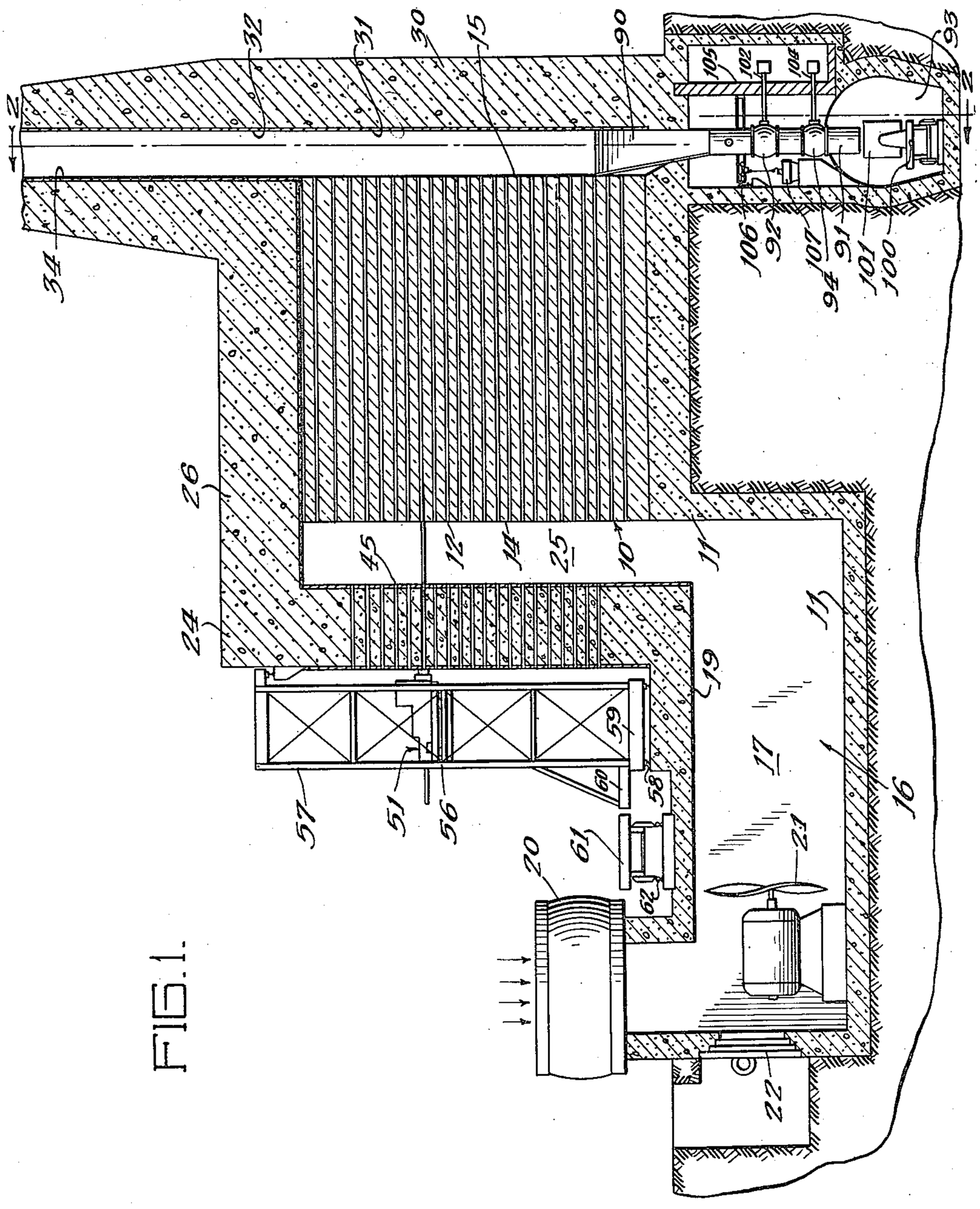


FIG. 1.

Witnesses:
Herbert E. Metcalf
Richard E. Burns

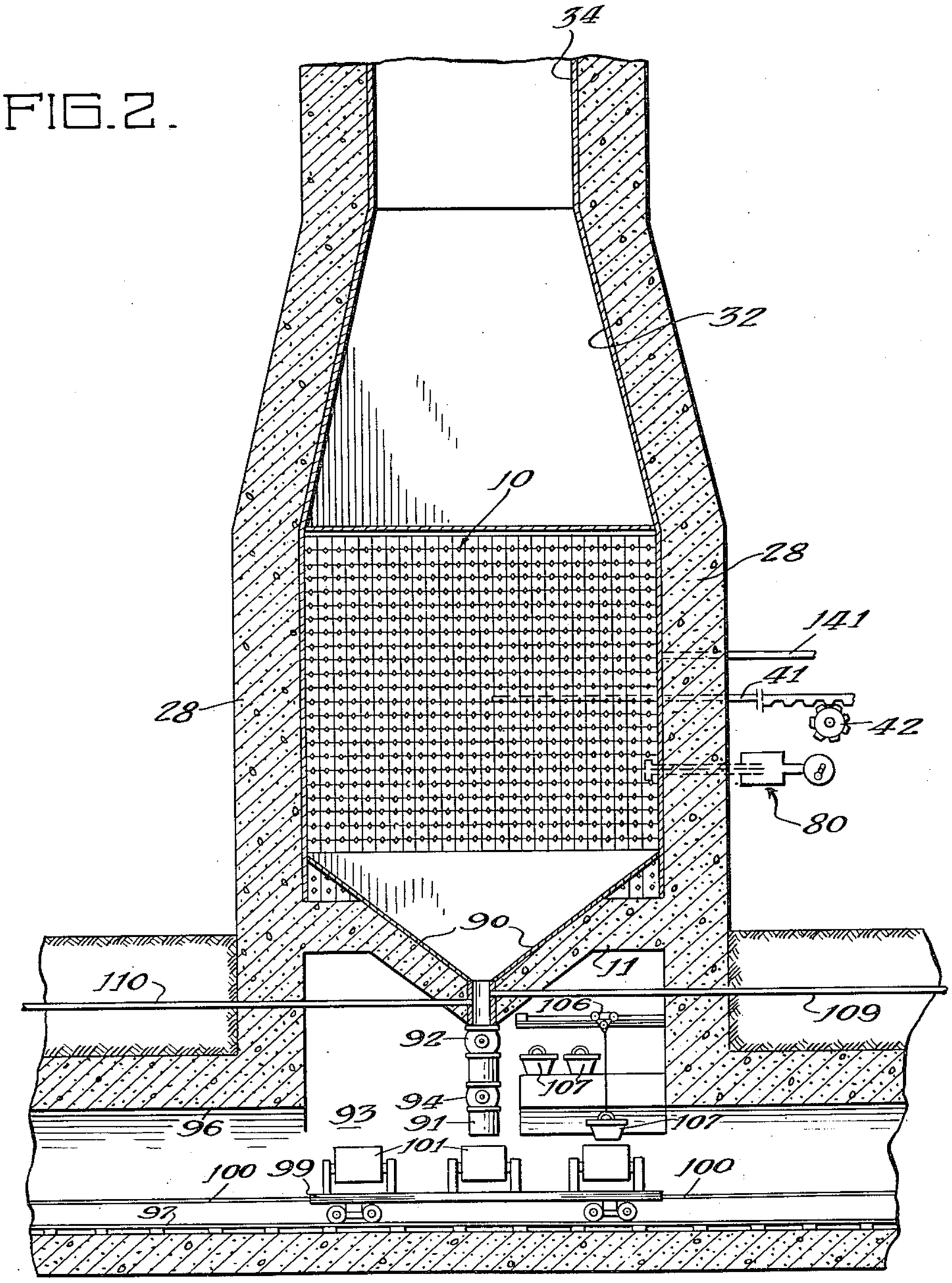
Inventors:
Enrico Fermi
Miles C. Leverett
By: Burt A. Fenander
Attorney

Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 2

FIG. 2.



Witnesses:
Herbert E. Metcalf
Richard E. Burn

Inventors:
Enrico Fermi
Miles C. Leverett
By: Robert A. Emmons
Attorney

Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 3

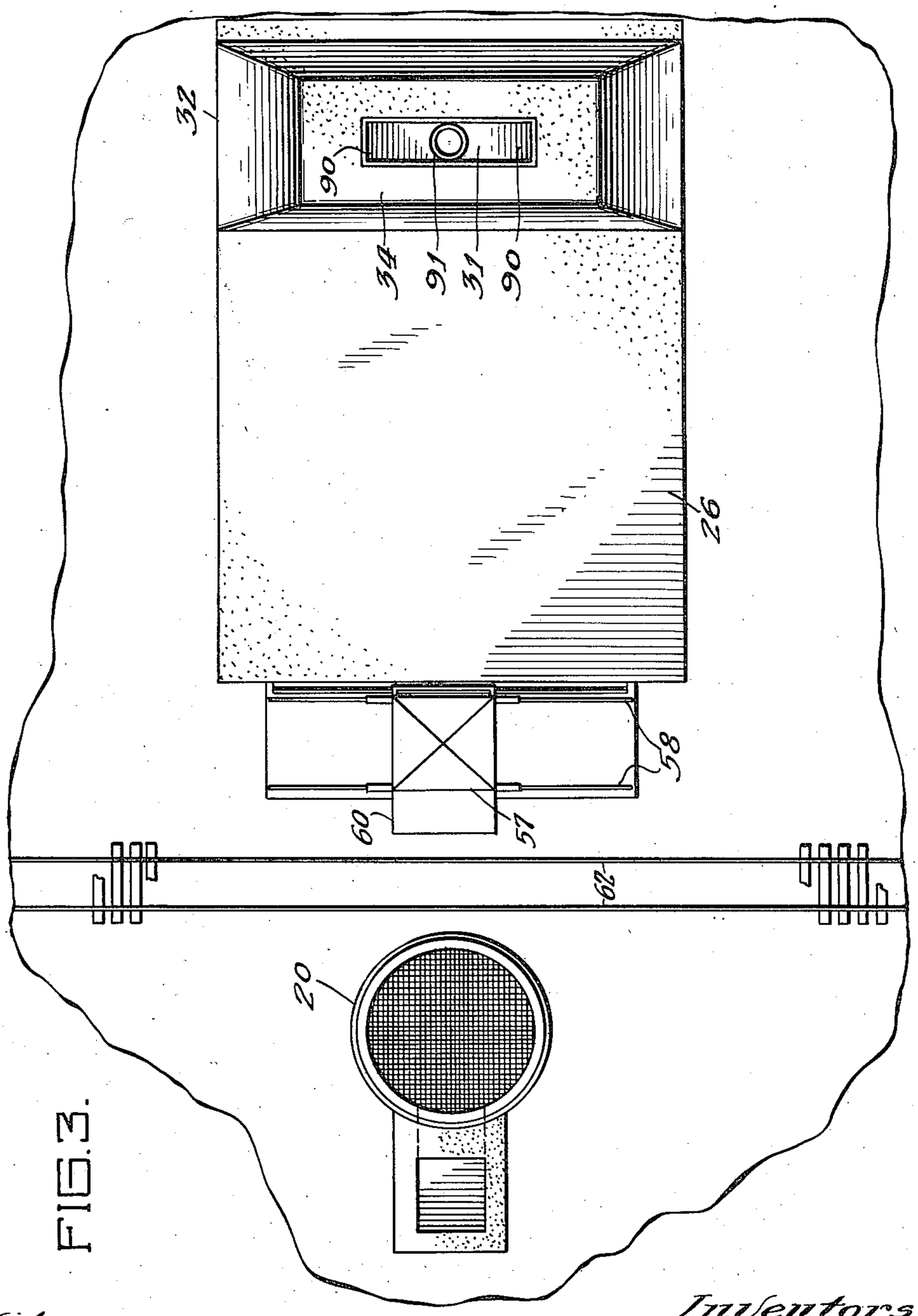


FIG. 3.

Witnesses:
Herbert E. Metcalf
Richard E. Burn

Inventors:
Enrico Fermi
Miles C. Leverett
By: Robert A. Emmons
Attorney:

Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 4

FIG.5.

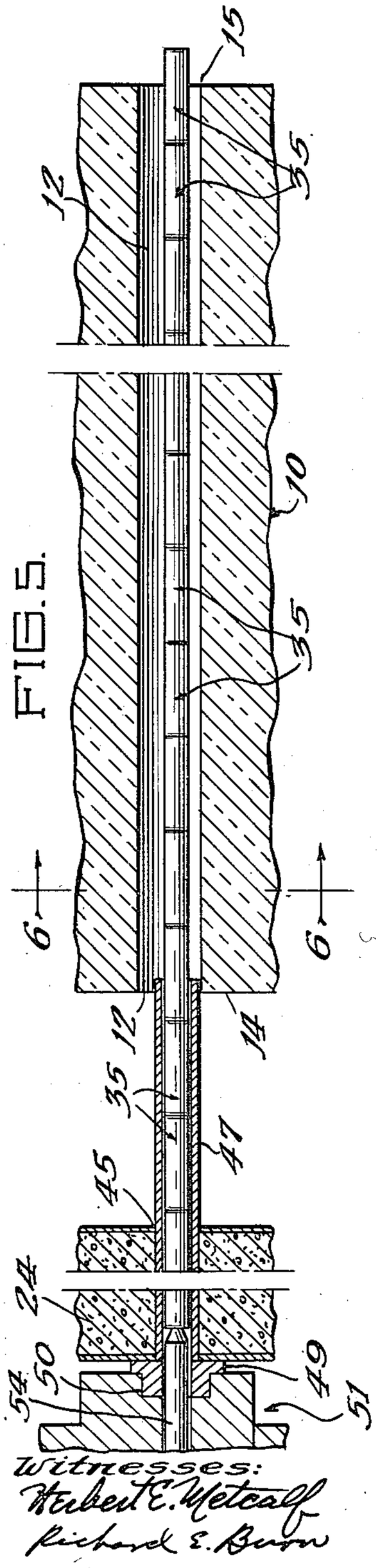


FIG.6.

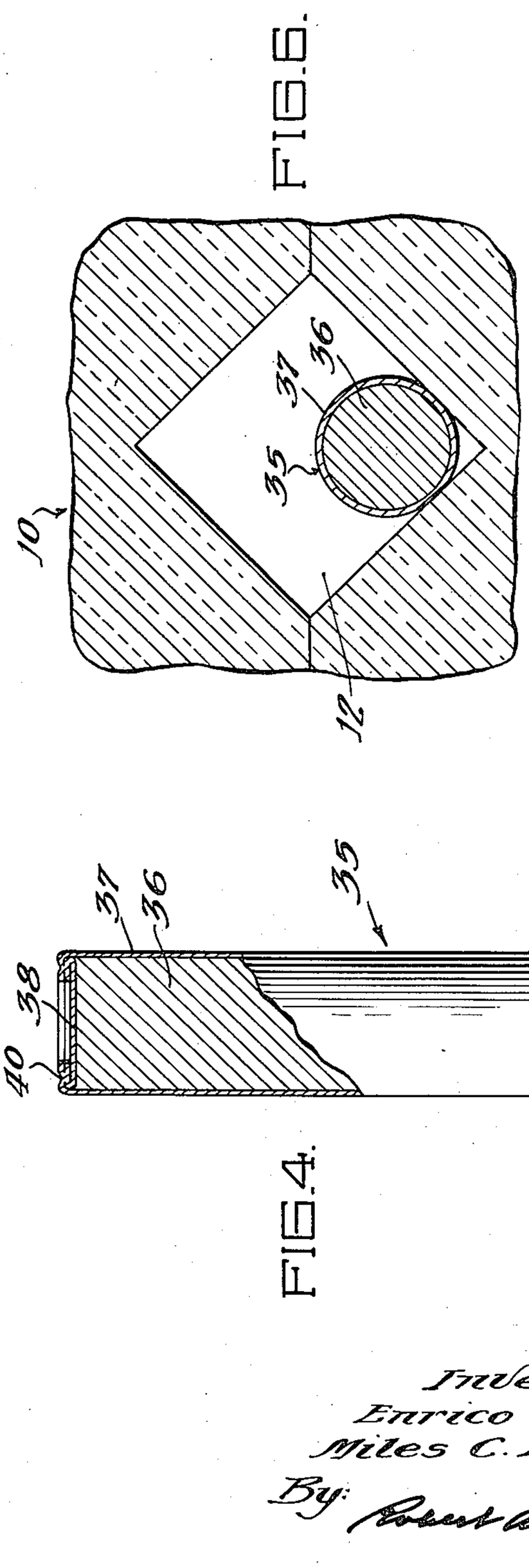


FIG.4.

Witnesses:
Herbert E. Metcalf
Richard E. Burns

Inventors:
Enrico Fermi
Miles C. Everett
By: Robert B. Townsend
Attorney.

Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 5

FIG. 7.

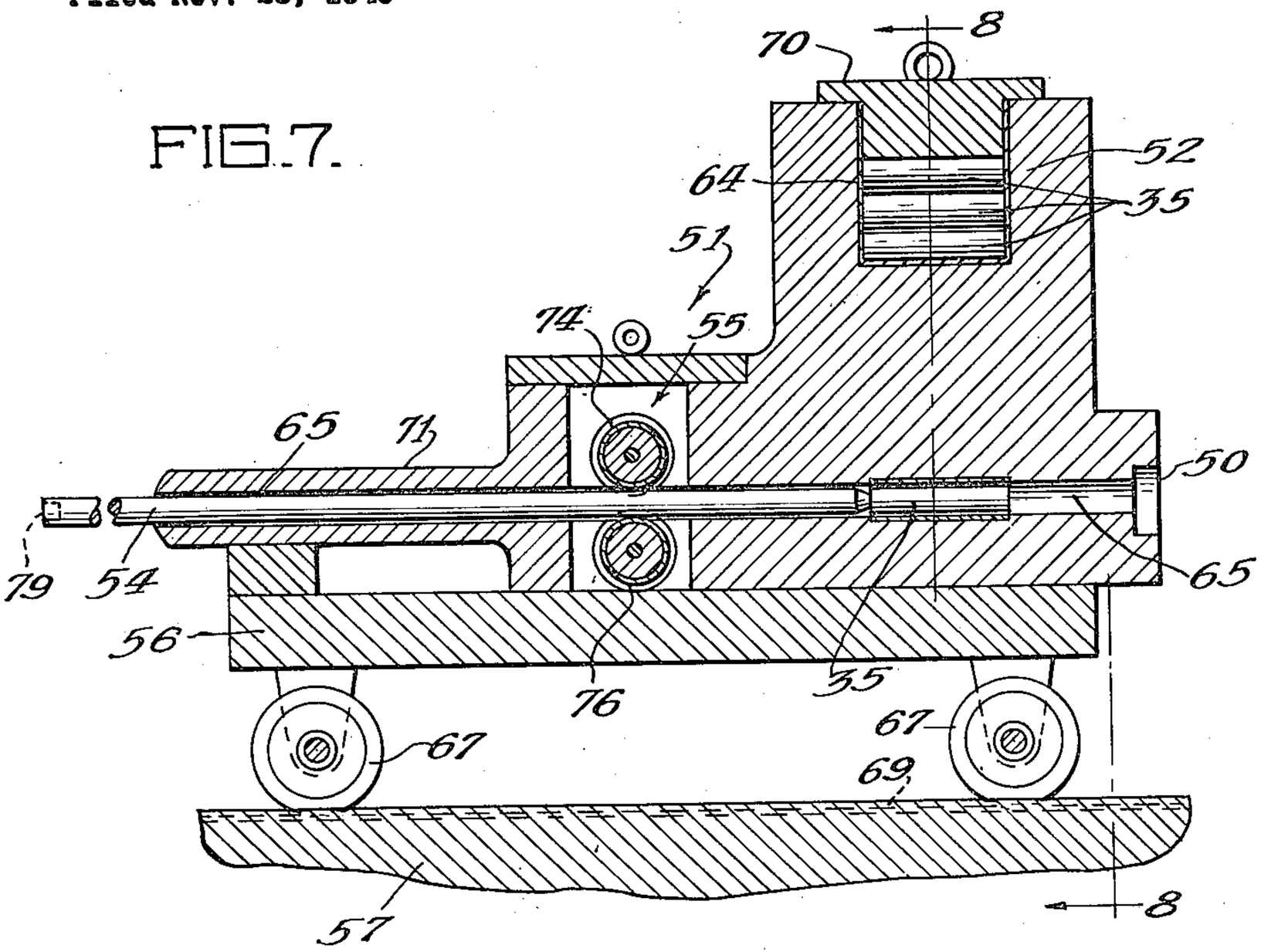
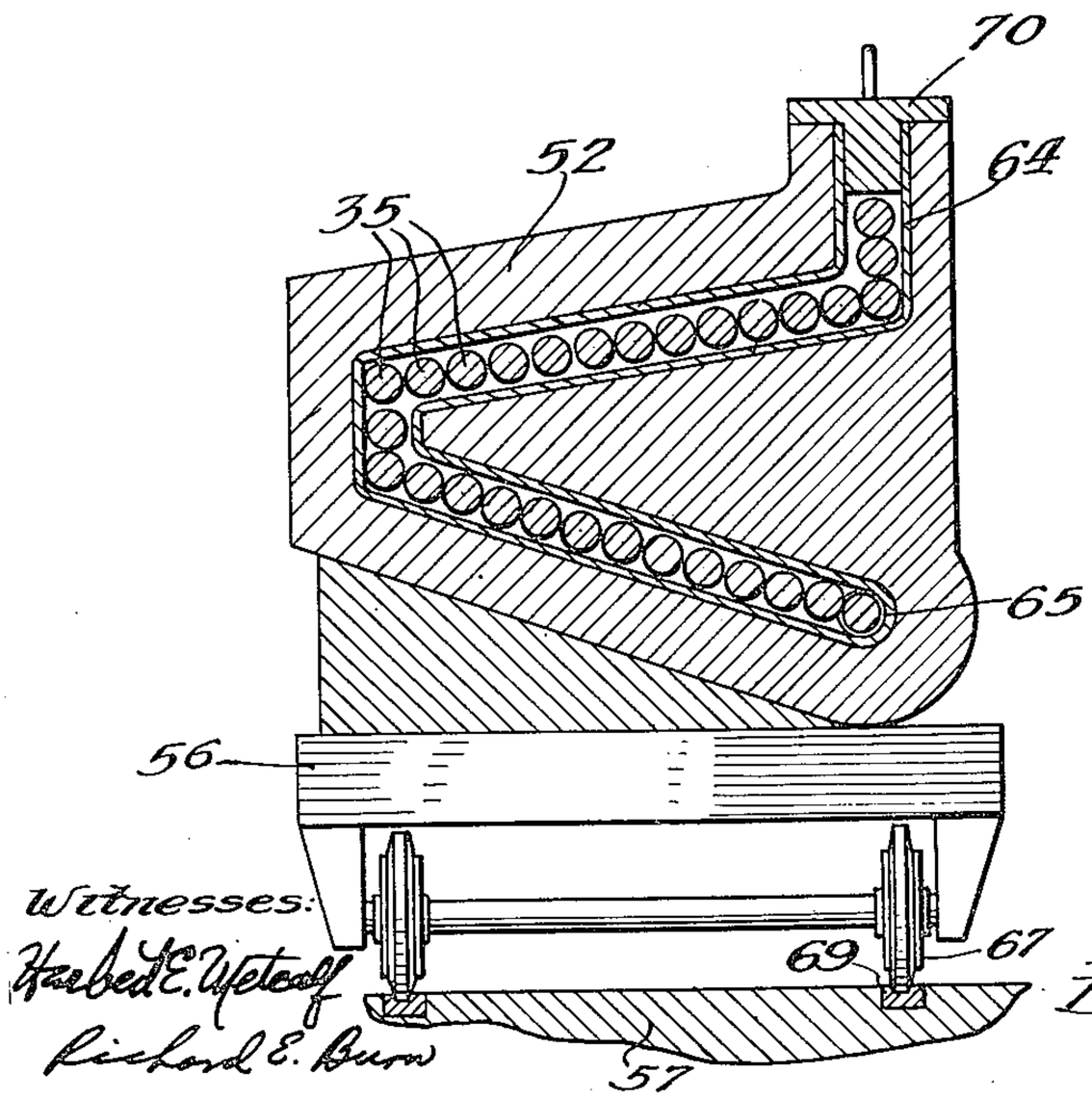


FIG. 8.



Witnesses:
Herbert E. Metcalf
Richard E. Burns

Inventors:
Enrico Fermi
Miles C. Leverett
By: Robert A. Saunders
Attorney.

Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 6

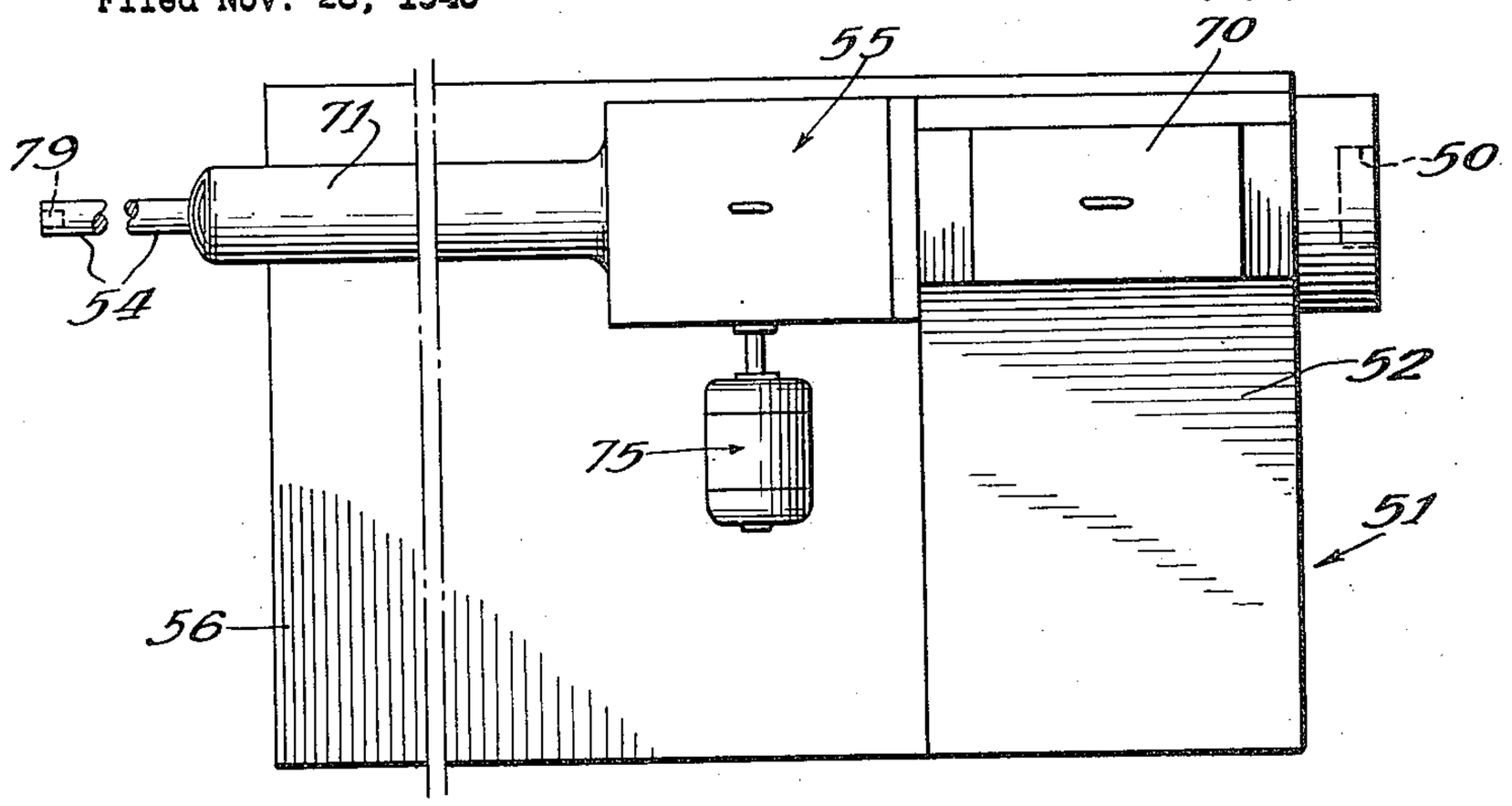


FIG. 9.

FIG. 11.

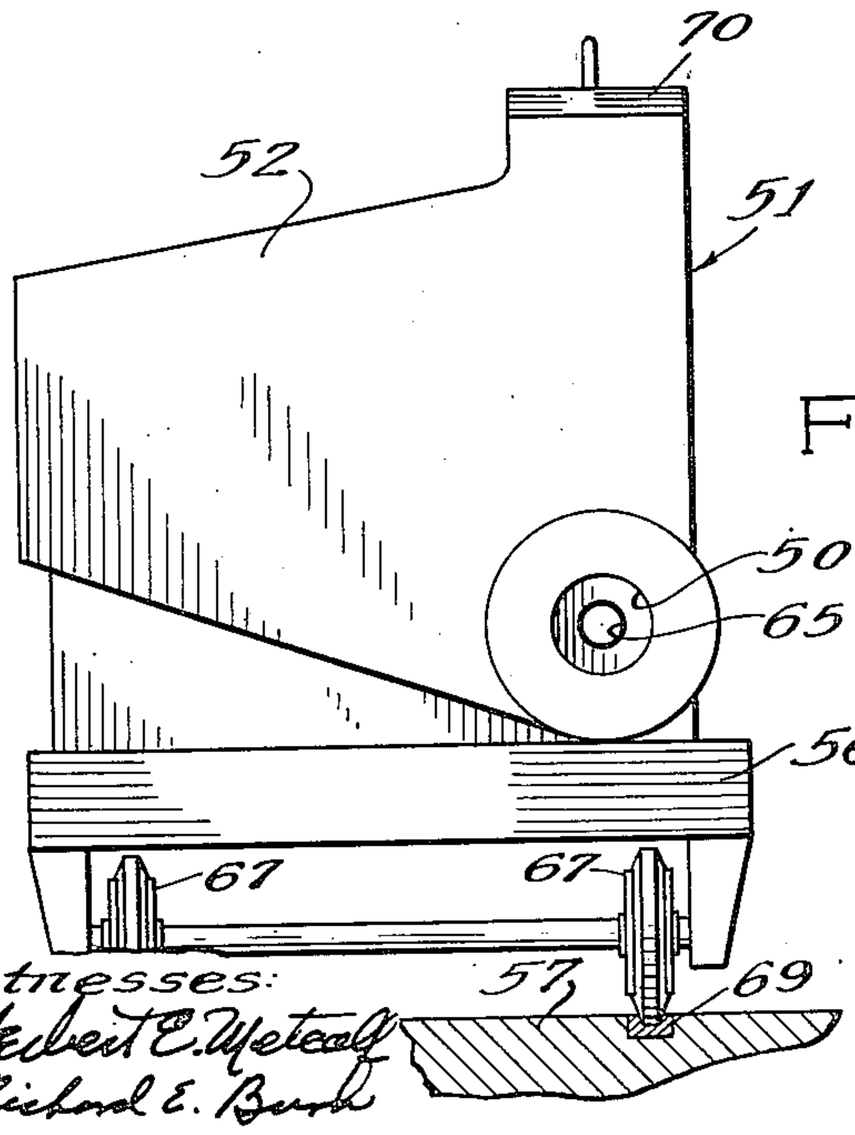


FIG. 10.

Witnesses:
Herbert E. Metcalf
Richard E. Burk

Inventors:
Enrico Fermi
Miles C. Leverett
By: Paul A. Saunders
Attorney.

Nov. 12, 1957 E. FERMI ET AL 2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Filed Nov. 28, 1945 7 Sheets-Sheet 7

FIG.12.

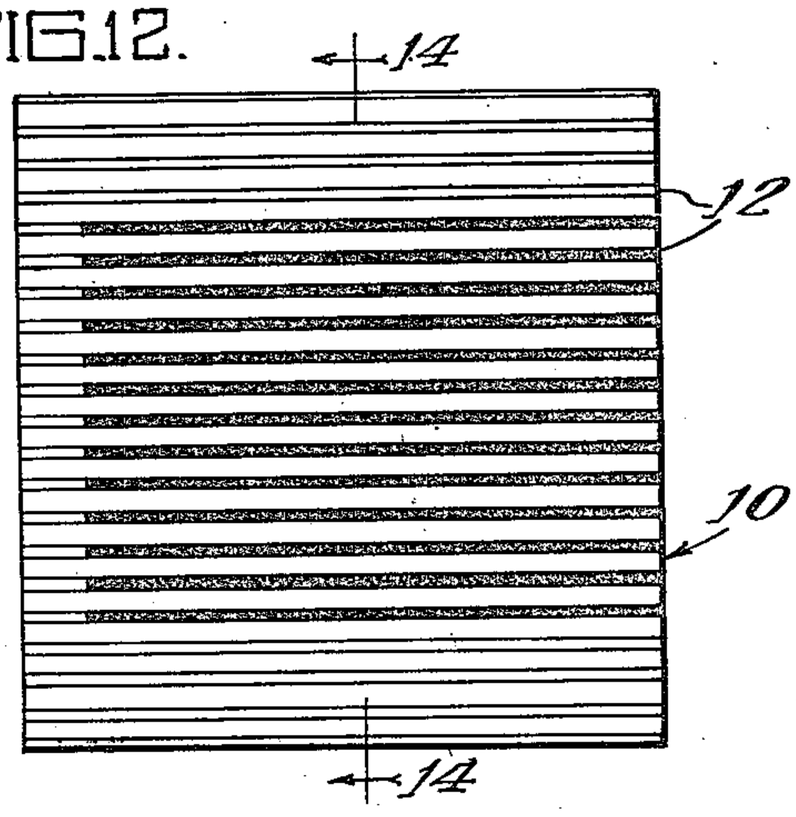


FIG.13.

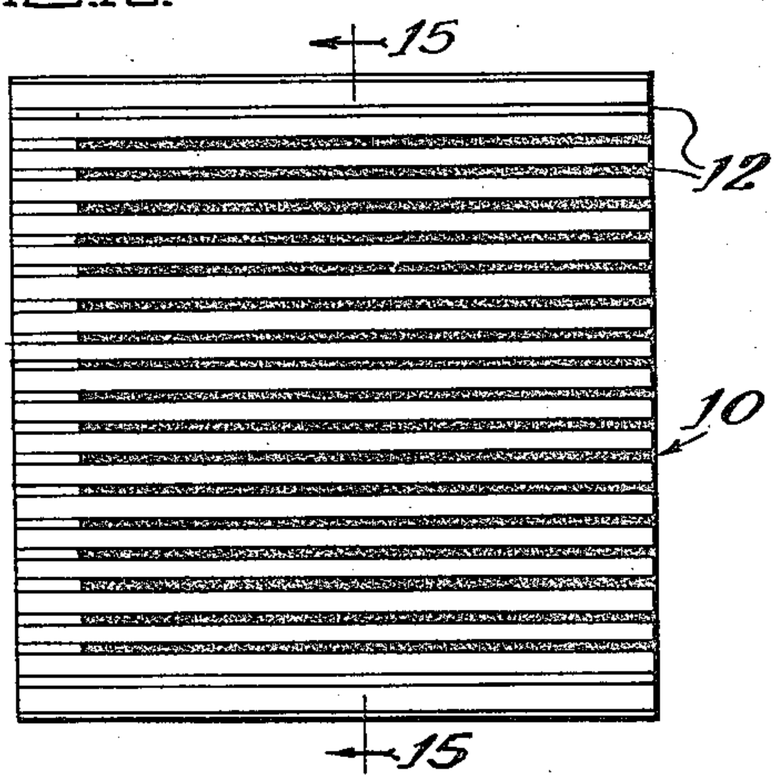


FIG.14.

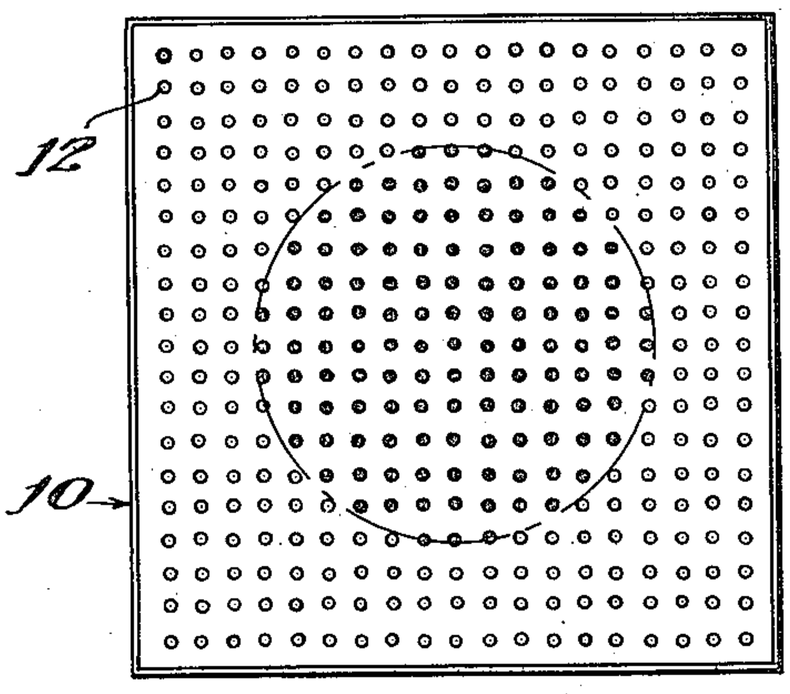


FIG.15.

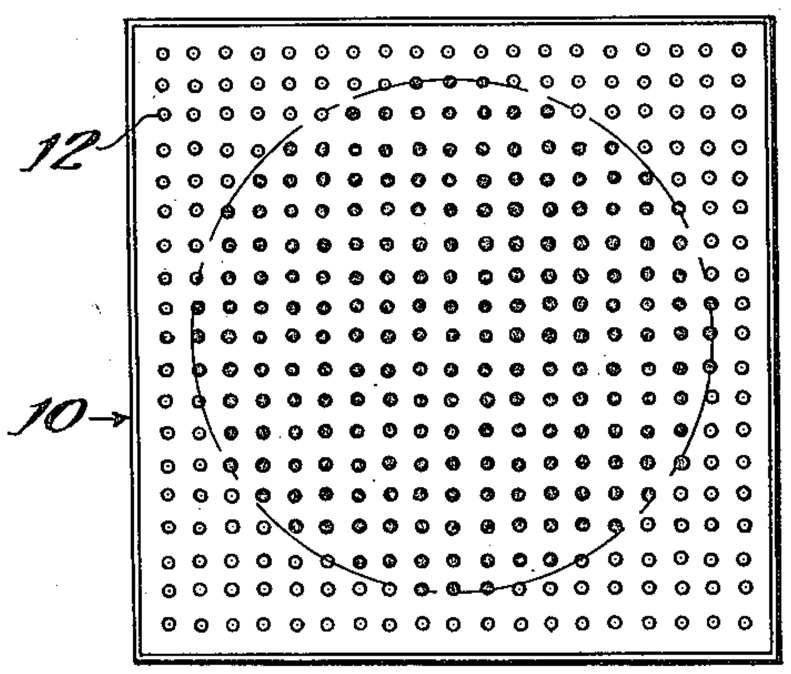
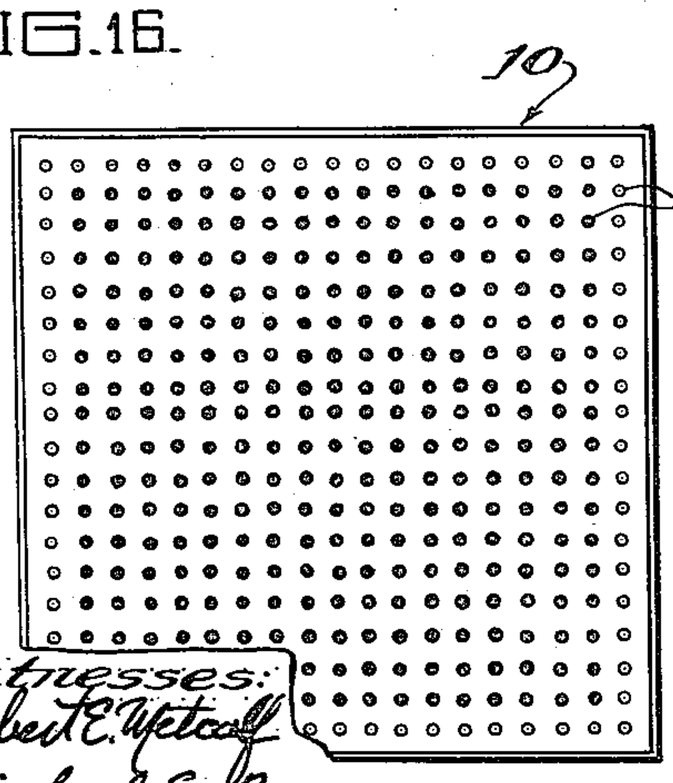


FIG.16.



Witnesses:
Herbert E. Metcalf
Richard E. Burns

Inventors:
Enrico Fermi
Miles C. Leverett
By: Robert A. Saunders
Attorney:

United States Patent Office

2,813,070

Patented Nov. 12, 1957

2,813,070

METHOD OF SUSTAINING A NEUTRONIC CHAIN REACTING SYSTEM

Enrico Fermi, Santa Fe, N. Mex., and Miles C. Leverett, Oak Ridge, Tenn., assignors to the United States of America as represented by the United States Atomic Energy Commission

Application November 28, 1945, Serial No. 631,406

1 Claim. (Cl. 204—154.2)

The present invention relates to devices of primary use for the production of neutrons by virtue of a self-sustaining chain reaction through fission of uranium or other fissionable isotopes with slow neutrons, known as neutronic reactors.

Natural uranium may be used in the reaction. Said natural uranium contains the isotopes 92^{238} and 92^{235} in the ratio of approximately 139 to 1. Hereinafter in the specification and the claim the term uranium is to be understood as referring to uranium and its chemical compositions of normal isotopic content unless otherwise indicated by the context.

In a self-sustaining chain reaction with slow neutrons, 92^{238} is converted by neutron capture to the isotope 92^{239}. The latter is converted by beta decay to 93^{239} and this 93^{239} in turn is converted by beta decay to the transuranic element 94^{239}. By slow neutron capture, 92^{235} on the other hand, undergoes nuclear fission to release energy appearing as heat, gamma and beta radiation, together with the formation of fission fragments appearing as radioactive isotopes of elements of lower mass numbers and with the release of secondary neutrons.

The secondary neutrons thus produced by the fissioning of the 92^{235} nuclei have a high average energy, and must be slowed to thermal energies by passing said neutrons through a material in which the neutrons are slowed by collisions. Such a material is known as a moderator.

Among the basic parts of a neutronic reactor system may be listed the following:

(1) A fissionable material such as uranium is properly dispersed in a moderator such as carbon or deuterium oxide.

(2) In most cases, a neutron reflecting or scattering layer is provided around the active portion for returning at least a portion of escaping neutrons into the active portion. The reflecting layer can be an extension of the moderator beyond the active portion.

(3) In all but the lowest power output reactors, coolant channels or pipes are passed through, or in heat exchange relationship to the active portion, in order that a stable temperature can be maintained in the reactor.

(4) In most cases, loading and unloading mechanisms are provided to charge and remove uranium from the active portion.

(5) A shield is almost invariably provided around the reactor to minimize escape of radiations biologically harmful to operating personnel. Such shields may comprise layers of iron and hydrogenous material closely surrounding the reactor, these layers in turn being surrounded with from 5 to 20 feet of concrete. Loading and unloading devices are usually operated from outside this massive shield. The size and weight of the shields make them permanent, and therefore it is very difficult to change the volume inside of the shields once they are constructed.

(6) A control rod of neutron absorbing material insertable into the reactor for the purpose of maintaining the neutron reproduction in an average state of balance is provided.

In the discussion to follow, the term neutronic reactor is used to mean the active portion and the reflector if used, and the term neutronic reactor system is used to mean the entire system including shields, charging and cooling systems, etc.

In considering the requirements for an operating neutronic reactor, the ratio of secondary neutrons produced by the fissions to the original number of primary neutrons initiating the fissions in a chain reacting system of infinite size using specific materials is called the reproduction or multiplication factor of the system and is denoted by the symbol K. If K can be made sufficiently greater than unity to create a net gain in neutrons, and the system made sufficiently large so that this gain is not entirely lost by leakage from the exterior surface of the system, then a self-sustaining chain reacting system can be built to produce power by nuclear fission of natural uranium. The neutron reproduction ratio (r) in a system of finite size differs from K by the leakage factor and operating poisoning factor, and it, too, must be greater than unity to permit the neutron density to rise exponentially. Such rise will continue if not stabilized at a desired density corresponding to a desired power output.

During the interchange of neutrons in a system comprising bodies of uranium of any size dispersed in a neutron moderator, neutrons may be lost in four ways; by absorption in the uranium metal or compound, by absorption in the moderator, by absorption in impurities present in the system or added as a result of operation, and by leakage from the system.

Natural uranium, particularly by reason of its U^{238} content, has an especially strong absorbing power for neutrons when they have been slowed down to moderate energies, these energies being termed resonance energies. The absorption of neutrons in uranium at these energies is termed the uranium resonance absorption or capture. It is caused by the isotope U^{238} and does not result in fission but creates the relatively stable nucleus 94^{239}. It is not to be confused with absorption or capture of neutrons by impurities, referred to later. Neutron resonance absorption in uranium may take place either on the surface of the uranium bodies, in which case the absorption is known as surface resonance absorption, or it may take place further in the interior of the uranium body in which case the absorption is known as volume resonance absorption. Volume resonance absorption is due to the fact that some neutrons make collisions inside the uranium body and may thus arrive at resonance energies therein. After successfully reaching thermal velocities, a large percent of the neutrons are also subject to capture by U^{238} without fission to produce 94^{239}.

It is possible by proper physical arrangement of the materials in the moderator to control the amount of uranium resonance absorption. By the use of light elements such as graphite and beryllium for example, fewer collisions are required to slow the neutrons to thermal energies, thus decreasing the probability of a neutron being at a resonance energy as it enters a uranium atom. During the slowing process, however, neutrons are diffusing through the slowing medium over random paths and distances so that the uranium is not only exposed to thermal neutrons but also to neutrons of energies varying between the energy of fission and thermal energy. Neutrons at uranium resonance energies will, if they enter uranium at these energies, be absorbed on the surface of a uranium body whatever its size, giving rise to surface absorption. Any substantial change of overall surface of the same amount of uranium will change surface absorption. Thus the volume ratio of moderator to uranium will control surface resonance absorption losses of neutrons in the uranium. The uranium is placed in the system in the form of spaced uranium masses or bodies

of substantial size, either of metal, oxide, carbide, or combinations thereof. The uranium bodies can be in the form of layers, rods or cylinders, cubes or spheres, or approximate shapes, dispersed throughout the graphite, preferably in some geometric pattern.

The term geometry is used to mean any pattern or arrangement wherein the uranium bodies are distributed in the moderator with at least a roughly uniform spacing and are roughly uniform in size and shape, or are systematic in variations of size, shape or spacing to produce a volume pattern conforming to a generally symmetrical system. If the pattern is a repeating or rather exactly regular one, the structure may be conveniently described as a lattice. The uranium bodies can be in the form of layers, rods, or cylinders, cubes or spheres, or approximate shapes, dispersed throughout the moderator. Optimum conditions are obtained with natural uranium by using metal spheres.

The resonance losses in uranium constitute one of the critical factors in coordinating the total losses permissible in a neutronic reactor.

The thermal neutrons are also subject to capture by the moderating material. While carbon and beryllium have very small capture cross-sections for thermal neutrons, an appreciable fraction of thermal neutrons (about 10 percent of the neutrons present in the system under best conditions with graphite) is lost by capture in the slowing material during diffusion therethrough. This means that when volume ratios are changed, the absorption in the moderator will also change.

In addition to the above-mentioned losses, that are inherently a part of the nuclear chain reaction process, impurities present in both the moderator and the uranium add a very important neutron loss factor in the chain. Such impurities may be originally present, or be formed during operation. The effectiveness of various elements as neutron absorbers varies tremendously. Certain elements such as boron, cadmium, samarium, gadolinium, and some others, if present even in a few parts per million, could very likely prevent a self-sustaining chain reaction from taking place. If impurities, solid, liquid or gaseous, and in elemental or combined form, are present in too great quantity, in the uranium bodies or the slowing material or in, or by absorption from, the free spaces of the system, the self-sustaining chain reaction cannot be attained. The amounts of impurities that may be permitted in a system, vary with a number of factors, such as the specific geometry of the system, and the form in which the uranium is used—that is, whether natural or containing more than the natural amount of a fissionable isotope, whether as metal or oxide—and also factors such as the weight ratios between the uranium and the slowing down material, and the type of moderating material used—for example, whether deuterium, graphite or beryllium. Although all of these considerations influence the actual permissible amount of each impurity material, it has fortunately been found that in general the effect of any given impurity or impurities can be correlated directly with the weight of the impurity present and with the K factor of the system, so that knowing the K factor for a given geometry and composition, the permissible amounts of particular impurities can be readily computed without taking individual account of the specific considerations named above. Different impurities are found to affect the reaction to widely different extents; for example, relatively considerable quantities of elements such as hydrogen may be present, and, as previously suggested, the uranium may be in the form of oxide, such as UO_2 or U_3O_8, or carbide, although the metal is preferred. Nitrogen may be present to some extent, and its effect on the chain reaction is such that the neutron reproduction ratio of the system may be changed by changes in atmospheric pressure. This effect may be eliminated by enclosing or evacuating the system if desired, or may be utilized by determining changes in

a particular system in the reproduction ratio as changes occur in the atmospheric pressure. A sensitive barometer is thus obtained. In general, the inclusion of combined nitrogen is to be avoided.

The effect of impurities on the optimum reproduction factor K may be conveniently evaluated to a good approximation, simply by means of certain constants known as "danger coefficients" which are assigned to the various elements. These danger coefficients for the impurities are each multiplied by the percent by weight of the corresponding impurity, and the total sum of these products gives a value known as the total danger sum. This total danger sum is subtracted from the reproduction factor K as calculated for pure materials and for the specific geometry under consideration.

The danger coefficients are defined in terms of the ratio of the weight of impurity per unit mass of uranium and are based on the cross-section for absorption of thermal neutrons of the various elements. These values may be obtained from physics textbooks on the subject and the danger coefficient computed by the formula

$$\frac{T_i}{T_u}\cdot\frac{A_u}{A_i}$$

wherein T_i represents the cross-section for the impurity and T_u the cross-section for the uranium, A_i the atomic weight of the impurity and A_u the atomic weight for uranium. If the impurities are in the carbon, they are computed as their percent of the weight of the uranium of the system. Thus the effect of all impurities originally in the system as placed in operation can be evaluated in terms of the K factor. This fact is used to insure control of the reactor, by deliberately inserting neutron absorbing materials to a greater or less extent in the reactor when built to an increase in neutron density, with subsequent stabilization of the neutron density at a desired value, as will be later explained.

The size of the system for proper operation will vary depending upon the K factor of the system, and upon other things such as, for example, the type of moderator and power of operation. If the reproduction factor K is greater than unity, the number of neutrons present will increase exponentially, provided the structure is made sufficiently large. If, on the contrary, the structure is small, with a large surface-to-volume ratio, there will be a rate of loss of neutrons from the structure by leakage through the outer surfaces, which may overbalance the rate of neutron production inside the structure so that a chain reaction will not be self-sustaining. For each value of the reproduction factor K greater than unity, there is thus a minimum overall size of a given active structure known as the critical size, above which the rate of loss of neutrons by diffusion to the walls of the structure and leakage away from the structure is less than the rate of production of neutrons within the system, thus making the chain reaction self-sustaining. The rate of diffusion of neutrons away from a large structure in which they are being created through the exterior surface thereof may be treated by mathematical analysis when the value of K and certain other constants are known.

In the case of a spherical structure employing uranium bodies embedded in graphite in the geometries disclosed herein and without an external reflector the following formula gives the critical overall radius when K is known:

Critical sphere of radius R ft., $K-1=C/R^2$

where C is a constant that varies slightly with geometry and for normal uranium-graphite lattices may have a value close to 7.4.

For a rectangular parallelopiped structure rather than spherical, the critical size can be computed from the formula

$$K-1=C\left(\frac{1}{a^2}+\frac{1}{b^2}+\frac{1}{c^2}\right)$$

where *a*, *b*, and *c* are the lengths of the sides in feet. The critical size for a cylindrical structure is given by the formula, irrespective of the shape of the uranium bodies

Cylinder height *h* ft.
Radius R ft.

$$K-1=C\left(\frac{1}{h^2}+\frac{0.59}{R^2}\right)$$

However, when critical size is attained by definition, no rise in neutron density can be expected. It is therefore necessary to increase the size of the structure beyond the critical size but not to the extent that the period for doubling of the neutron density is too short, as will be explained later. A desirable reproduction ratio (*r*) for an operating structure at a stabilized power with all control absorbers removed is about 1.005. This will permit the neutron density to rise in a reasonable time and yet permit control by insertion of sufficient neutron absorbers to bring the effective reproduction ratio to unity for continued operation at some predetermined neutron density. The size at which the reproduction ratio of 1.005 can be obtained may be computed from modifications of the above formulae for critical size when corrected for power of operation. For example, for spherical active structures the formula $K-r=7.4/R^2$ may be used to find R when K is known and *r* is somewhat over unity. The same formula will, of course, give *r* for given structures for which K and R are known.

It can be seen from the above discussion that if, after a neutronic reactor has been built to a fixed operating size, the K factor should change for any reason, critical size would change. The reproduction ratio *r* would then change in accordance with the difference between critical size and operating size. If K increases, critical size decreases, and *r* decreases. If K decreases critical size increases, *r* decreases, and if K decreases sufficiently, critical size may coincide with or become less than operating size and the reactor will become inoperative unless increased to a new operating size where *r* is again greater than unity.

During the lifetime of operation of a neutronic reactor, K may change for several reasons which may be grouped under long term effects and short term effects.

During the neutronic reaction fission products, including such extremely absorptive elements as samarium and gadolinium, for example, may form from the fission of the uranium isotopes U^{235} and 94^{239}. Samarium, as well as some other elements formed in this manner, has very high neutron absorption characteristics and may increase the total parasitic neutron absorption in the reactor, with consequent decrease in the reproduction factor. However, the absorption of neutrons by the uranium isotope U^{238} leads to the production of 94^{239}. Plutonium is fissionable to an even greater degree than the isotope U^{235} so that formation of plutonium tends to increase the reproduction factor K, notwithstanding decrease in the U^{235} content of the uranium in the reactor due to destruction by fission. The increase in K over a long period of production of 94^{239} may run as high as between 1 and 2 percent.

Still another factor tends to increase the reproduction factor in an operating reactor. Certain neutron absorbing impurities are present in graphite moderators, for example, sufficient to reduce K by from 1 to 1½ percent. A substantial portion of this reduction is due to boron inherently present in the graphite. This boron is gradually transmuted, probably to lithium, during the operation of the reactor, and as lithium has a very much less neutron capture capability than boron, K will increase.

The long term effects can be summarized by stating that K will probably initially increase due to production of 94 and then decrease as the fission products build up.

In addition to the long term changes in K to be expected, there are shorter and even more significant changes in K that take place in reactors operating continuously at high neutron densities, as for example, U-graphite reactors operating at from 5,000 to 500,000 kilowatts power output.

The short term effect of xenon 135 formed as a radioactive decay element from iodine produced in the reactor during operation, becomes significant after 4 to 5 hours of continuous operation and changes K by an amount equal to the equilibrium effect of the amount of xenon 135 produced and destroyed at the power of operation. Xenon 135 decays to barium, but is changed by neutron absorption to xenon 136. Xenon 135 has an extremely high neutron capture cross-section of about $2{,}500{,}000\times10^{-24}$ cm.^{2} whereas the capture cross-sections of iodine, barium and xenon 136 are relatively small. Thus while the xenon 135 is present, due to the operation of the reactor, K is reduced by an amount related to the continuous power output in a graphite moderated reactor as follows:

∞ kw.=0.03 K
500,000 kw.=0.02 K
100,000 kw.=.013 K
10,000 kw.=.0012 K

The end result of this xenon 135 effect is that a reactor having an actual size capable of supporting a chain reaction at low, intermittent powers, may not be large enough to support a chain reaction at higher continuous power outputs. In consequence, the active portion of the reactor must be made large enough in the first place, if continuous high power operation is to be attained.

While it is, of course, apparent that a decrease in K might cause a system fixed in size to become inoperative it is not so apparent that an increase in K would be undesirable. However an increase in K is undesirable for the reason that if K increases, *r* will also increase and the reactor may become dangerous to operate, if *r* reaches or exceeds 1.01.

In order that the significance of a reproduction ratio of 1.01 be more fully understood, the mechanism of fission will be discussed further. Not all of the fast neutrons originating in the uranium body, as the result of fission, leave the uranium immediately. With natural uranium about one percent are "delayed neutrons." These delayed fast neutrons may appear at any time up to several minutes after the fission has occurred. Half these neutrons are emitted within six seconds and 0.9 within 45 seconds. The mean time of delayed emission is about 5 seconds. The neutron reproduction cycle is completed by 99 percent of the neutrons in about 0.0015 second, but if the reactor is operating with a reproduction ratio near unity, the extra 1 percent may make all the difference between an increase or a decrease in the activity of the reactor. The fact that the last neutron in the cycle is held back, as it were, imparts a slowness of response to the pile that would not be present if the fission neutrons were all emitted instantaneously.

For cases in which the reproduction ratio (*r*) differs from unity by appreciably less than 1 percent, the rise of neutron density, or more specifically the value N to which the number of neutrons has risen from an original value N_0, after a lapse of time of *t* seconds during and before which the pile has operated at a fixed value of *r* (N_0 being the number of neutrons at the beginning of *t*, i. e., after disappearance of transient effects due to any preceding change in *r*) is given by —

$$N=N_0e^{wt.}$$

where

$$w=\frac{r-1}{\alpha-(r-1)}\cdot\frac{1}{T}$$

In this formula α is the fraction of the neutrons that are delayed, i. e., $\alpha=.0067$, and T is the mean time of delayed emission of the delayed neutrons=5 seconds.

The above formula is only approximate because it uses an average delay time.

As an example, suppose as a result of moving the control rod, r becomes 1.001, and assume that the system has settled down to a steady exponential rise in neutron density. Then

$$w=\frac{.001}{.0067-.001}\cdot\frac{1}{5}=\frac{1}{28.5}$$

that is, $N/N_0=2.72$ in 28.5 seconds. Hence doubling of the neutron density occurs about every 20 seconds and continues indefinitely. The above formula thus indicates the rate of rise for relatively low values of r and shows how the reduction of the rate of the delayed neutron effect is particularly significant in the stated lower range of r values. Strictly speaking, the given equation holds only for the steady state, i. e., where r has been held constant for some time; an additional transient term must be included to obtain an accurate representation of the neutron density during the first few seconds after a sudden change of r.

If r were made exactly 1.01, a more detailed theory shows that the neutron density would be more than tripled each second. However, if the reproduction ratio r is several percent greater than unity, so that the one percent delayed neutrons are unimportant compared with r—1, the density increases at a much more rapid rate as given approximately by $r^{t/l}$ where l is 0.0015 second, the normal time to complete a cycle. Thus if r were to be made 1.04, the neutron density would increase in 1.5 seconds by a factor of approximately 10^{17} over its original level. However, if r were 1.02 and 1.03, the factor by which the neutron density would be multiplied each second, would be 1100 and 700,000 respectively. It is thus apparent that too high a reproduction ratio in a practical system leads to the necessity of inserting what may be considered as an excessive amount of controlling absorbers to reduce the effective reproduction ratio to unity. An exceedingly dangerous condition could exist if by accident these absorbers were suddenly completely removed, as the time required for reinserting the absorbing material might be too long to prevent destruction of the system. As the same eventual density can be obtained with a reproduction ratio only slightly over unity, as with a higher ratio, only at a slower rate, the lower reproduction ratios which exceed unity by not substantially more than .01, an amount equal to the percentage of the neutrons formed which are "delayed neutrons" are preferred in practice in the interest of safety.

It is therefore desirable to provide a reactor in which compensation for variation in the reproduction factor may be made when necessary.

As massive iron and concrete shields, which may be of the order of 20 feet in thickness customarily surround an operating reactor, any requirement that a reactor be increased in size beyond a size limited by the shielding structures, could not be met because of the immobility of said shields. Furthermore, if a decrease in size is to be made, the entire system must be such as to be fully as efficiently operated after decrease as before.

It is the main object of the present invention to provide a novel neutronic reactor that may be initially constructed in such a manner as to allow for either an increase or decrease in the size of the active portion.

Broadly stated, compensation for changes in the K factor with a consequent requirement for change in operating size can be accomplished in accordance with the present invention by initially providing a moderator mass greater in size than would be required for a uranium-moderator structure of a given initial K factor, and then initially loading uranium into that mass only in an amount and with such distribution to provide a reactor of a predetermined operating reproduction ratio r. Then, if K should change, the amount of uranium in the moderator mass can be increased or decreased to provide continued operation with the predetermined r.

The amount of uranium loaded into a moderator mass can be changed because any portion of the moderator extending beyond the uranium containing mass acts as a reflector, returning by the scattering action of the moderator nuclei a large number of the escaping neutrons. The use of a reflecting layer of moderator 2 feet thick, for example, around a neutronic reactor can reduce the critical size of the reactor by several percent of the radius thereof.

It is not the reduction in size of the reactor that is important in this instance. The main consideration is that a reflecting layer one to two feet thick around a reactor is nearly as efficient a reflector as a reflecting layer of infinite thickness. As a close approximation, reflecting layers 2 to 10 feet thick, for example, may be considered equivalent. In consequence, when the moderator mass is larger than that portion of the moderator containing uranium, the active portion always has an efficient reflector, providing only that a moderator layer of sufficient thickness remains around the uranium bearing portion to give proper reflecting efficiency.

As a result, a mass of moderator can be charged or loaded with varying amounts of uranium to form a portion of operating size, as needed, in all cases being surrounded by a reflecting layer of substantially equal reflecting efficiency.

It is therefore another object of the present invention to provide a neutronic reactor system of fixed dimensions wherein the size of the active portion supporting the chain reaction can be changed to suit circumstances without loss of efficiency.

There are other advantages of a system wherein the active portion can be changed to suit conditions. As noted above, the active portions of neutronic reactors can be in the overall shape of a cube, sphere or cylinder. Further, it has been stated that the uranium can be placed in the reactor in the form of spheres, rods or layers. Thus, it is desirable that provision be made that a moderator mass be provided such that active portions can be loaded, not only to varying overall sizes, but to varying overall shapes, with varying types of uranium and with varying geometries of the uranium with respect to the moderator in the portion loaded.

It follows that another object of the present invention is to provide a neutronic reactor system in which uranium or other fissionable material can be loaded with varying shapes of the loaded portions, and with varying sizes and shapes of the uranium bodies themselves, as desired.

Other objects and advantages of the present invention may be more clearly understood by references to the following description and the attached drawings which illustrate, as an example, one form the invention may take. This example is not to be taken as limiting, as other forms within the scole of the appended claim will be readily apparent to those skilled in the art.

In the drawings:

Fig. 1 is a longitudinal view partly in section and partly in elevation of an air cooled neutronic reactor system illustrating the present invention;

Fig. 2 is a cross-sectional view, partly in elevation, taken as indicated by the line **2—2** in Fig. 1;

Fig. 3 is a plan view of the system shown in Figs. 1 and 2;

Fig. 4 is a longitudinal sectional view partly in elevation of a jacketed slug;

Fig. 5 is a longitudinal sectional view, partly in elevation, of a horizontal channel during a loading and unloading operation;

Fig. 6 is a cross-sectional view taken as indicated by the line **6—6** in Fig. 5;

Fig. 7 is a longitudinal sectional view, partly in elevation, of one form of loading device;

Fig. 8 is a view partly in section and partly in elevation taken as indicated by the line **8—8** in Fig. 7;

Fig. 9 is a top plan view of the loading device shown in Figs. 7 and 8;

Fig. 10 is an end view of the loading device shown in Figs. 7, 8 and 9;

Fig. 11 is a view partly in section and partly in elevation of a friction drive used in the loading device;

Figs. 12 and 13 are diagrammatic longitudinal sections showing loadings of different sizes, the section lines being left off for purposes of clarity;

Figs. 14 and 15 are diagrammatic cross-sections taken as indicated by lines **14—14** and **15—15** in Figs. 12 and 13, respectively, section lines being omitted for clarity.

Fig. 16 is a diagrammatic cross-section of a cubically loaded neutronic reactor, section lines being omitted for clarity.

Referring to the drawings, the present invention is illustrated by reference to an air cooled graphite-uranium reactor, sometimes known as a pile. The operation and construction of such reactors are more fully described in Fermi et al. Patent 2,708,656, dated May 17, 1955.

Such a reactor broadly comprises a mass of graphite blocks closely piled or stacked into a cube **10** as shown in Figs. 1 and 2. This graphite cube may be, for example, 24–26 feet on a side and rest on a concrete foundation **11**. The graphite cube **10** is pierced with horizontal air channels **12**, of square cross-section, with one of the diagonals vertical. The channels may be readily made by grooving adjacent blocks (Fig. 6). The channels are .175 inches on a side and extend completely through the reactor, from an inlet face **14** to an outlet face **15**. Two thousand channels may be provided, for example, and as will later be brought out, any unused channels can be plugged. Only a few of the channels are shown in the drawings for the sake of clarity.

Adjacent the inlet face **14** of the cube, the foundation is continued downwardly to form the floor of an inlet air duct **16** extending outwardly. The inlet air duct **16** is completed by concrete side walls **17** and top **19**.

At some distance away from the graphite cube **10** the inlet duct is turned upwardly to terminate in an air filter **20**, relatively close to the surface of the ground. A fan or blower **21**, here illustrated as electrically driven, is installed on the floor of the inlet duct just below the air filter, access to the fan being conveniently obtained through duct door **22**, behind the fan.

The concrete top **19** of the inlet air duct is continued upwardly as an inlet end shield **24**, positioned parallel to but spaced away from inlet face **14** of the cube **10** to form an inlet chamber **25** communicating with the air channels **12**.

Above the inlet chamber **25** and the cube **10** the concrete is continued horizontally to form a top shield **26**, and side shields **28** are built up from the foundation **11** to enclose cube **10**. Shields **26** and **28** closely approach the top and side faces of the cube, to minimize air flow around the outside of the cube. A small amount of air circulation, however, may be desirable over the top and side faces to cool these faces.

At the outlet face **15**, an outlet end shield **30** of concrete is provided. End shield **30** is parallel to and spaced from the outlet face **15** of the graphite cube to form an outlet chamber **31** communicating above with the base **32** of a stack **34** projecting upwardly and formed as a continuation of the concrete top side and outlet end shields. Thus the cube **10** is completely enclosed by concrete shields, with a duct system operating by virtue of pressure provided by fan **21** to conduct air from close to ground level through channels **12** into the stack and then into the atmosphere well above ground level at the top of the stack. The concrete shields may be from ten to twenty feet thick in accordance with the maximum desired operating power of the reactor and serve as shields to reduce escape of neutrons and gamma radiation.

As a neutronic reaction will take place when uranium bodies are properly spaced in a moderator mass of a certain finite size, the above described device can be made chain reacting by placing uranium bodies in the

horizontal channels in such a manner and in such an amount that a neutron reproduction ratio of slightly over unity is obtained, exclusive of all neutron losses within the reactor and from the exterior of the reactor. This reproduction ratio may be defined as the ratio of the number of neutrons gained by fission to the total number of neutrons lost by absorption in the uranium, absorption in the moderator, absorption by impurities in the reactor or added as a result of operation, and by leakage from the reactor for a reactor of finite size.

Using the graphite mass as the moderator to slow fast neutrons to energies where they again are able to create fission in 92^{235}, the device will have a reproduction ratio of unity when approximately 700 of the channels **12** in the graphite cube are each loaded with 68 aluminum jacketed uranium slugs **35** lying end to end, with a channel spacing of 7 inches measured center to center, and with the loaded channels roughly forming a cylinder (Figs. 14 and 15). Both graphite and uranium should be of highest possible purity. The diameter of the cylinder will be about 18 feet.

However, more than a unity reproduction ratio is required, as when the reproduction ratio is exactly unity no rise in neutron density will occur. Under such conditions the device will not develop high neutron densities or power in the form of heat. By loading additional channels, i. e., making the active portion greater than critical size, however, the reproduction ratio within the reactor can be brought above unity in order that a rise in density can occur. Then this excess neutron reproduction can be absorbed by neutron absorbing materials deliberately inserted into the reactor in order to hold the reproduction ratio at an average value of unity after a desired power output has been obtained, as a result of the initial rise in density.

Consequently, in accordance with the amount of excess reproduction ratio desired, about 1,000 channels may be loaded with uranium slugs. The diameter of the cylinder will now be about 21 feet, leaving a minimum of 3 to 4 feet of graphite around the cylinder except at the ends. Most of the channels not loaded with uranium may be closed by inserting plugs, preferably of graphite, in such channels in order to conserve air. Some of the channels, however, in the peripheral portions of the cube may be left open for cooling of the graphite in those portions.

One preferred form of slug construction is shown in Fig. 4. Each uranium metal slug is 1.1 inches in diameter and 4 inches long covered with an aluminum jacket approximately 20 mils thick in good heat conductive relation to the uranium. The slugs weigh about 2½ pounds each.

In forming the slugs **35**, the uranium portion **36** is machined to size, cleaned in trisodium phosphate and then washed in water. Aluminum or other non-fissionable metal jacket cans **37** are provided having an inside diameter somewhat larger than the uranium portion. This can **37**, open at one end only, is slipped over the uranium after being cleaned in benzine and hot water. The can **37** with the uranium inside is then passed through a sizing die of 1.134 inches diameter. This die, being of smaller diameter than the 1.1 inch uranium portion plus the two 20 millimeter walls, draws the can in tight thermal contact with the uranium.

A cup-shaped cap **38** is then placed base down inside the projecting portion of the can **37** and is seam welded to the can. The projecting portion is then cut off above the seam weld **40** and the remaining projecting portion including the weld, spun over the adjacent end of the slug. Thus each jacket completely encloses and seals the uranium, preventing air from corroding the uranium and, as will be pointed out later, also preventing fission fragments created by nuclear fission at the surface of the uranium from entering the air stream.

The channels are loaded with uranium until the reproduction ratio, with neutron absorbers removed, is about 1.005 to 1.006. This means that for every two hundred neutrons starting in each neutron generation about two hundred and one neutrons are produced in the reactor over and above all losses. Under these conditions and taking into account the fact that about one percent of the neutrons of fission are delayed in their emission for a mean time of about 5 seconds, the neutron density of the reactor will double every 8 to 15 seconds. With some part of the neutron absorbers inserted but with the insertion of less than the amount of neutron absorbers required to make the reproduction ratio unity, the rise is slower. When the neutron absorbers are almost but not entirely inserted the doubling of the neutron density may take several hours. Then when a desired density has been reached, the reproduction ratio can be reduced to unity so that the desired density is continuously maintained by the introduction of neutron absorbing material into the reactor.

The neutron absorbing material is introduced into the reactor by means of a control rod **41** as shown in Fig. 2. This control rod extends into the graphite cube, sliding in a channel therein and is operated from outside of side shield **28** as by rack and pinion **42**. The rod is made from, or incorporates therein, an efficient neutron absorber, such as for example, cadmium or boron. A sheet of cadmium riveted to a steel strip forms a satisfactory control rod. As the depth of insertion of the rod determines the amount of neutron absorbing material inside the reactor, the critical position of the rod is where the rate of neutron absorption by the rod balances the reproduction ratio at unity. Thus, by moving the rod outwardly from the critical position the neutron density in the reactor will rise. Moving the rod inwardly from the critical position causes the reproduction ratio to fall below unity, and the reaction stops. Thus the reaction is always under control, and as the rise in neutron density is exceptionally slow as the rod approaches the critical position, manual control is possible. Other and similar rods **141**, shown schematically in Fig. 2 may be provided, if desired, for rapid progression into the reactor to stop the reaction in case of failure of the control rod to stop the rise in neutron density for any reason. Such rods are termed safety rods.

During operation heat is released in the reactor in accordance with the neutron density therein. Most of the heat arises from the kinetic energy of the fission fragments and about 92 percent of the energy is released in the uranium. About 6 percent is released in the graphite due to neutron absorption therein and about 2 percent escapes from the reactor in the form of neutrons and gamma radiation. Consequently, the reactor can only be operated at a power dependent upon heat removal to the point where a stable temperature obtains. Otherwise, the reactor will accumulate heat to the point that the device may be damaged. Since aluminum melts at 658° C., stable temperatures below this value should be used although with jackets of other non-fissionable metals, such as beryllium, the stable temperature may be increased, although if the temperature should rise too high the uranium bodies might be damaged even when using beryllium jackets as uranium of the type used in neutronic reactors melts at about 1100° C.

A stable temperature is obtained in the device of the present invention by passing atmospheric air through the reactor, and in the specific example shown and described, the air is passed through the graphite channels and directly in contact with the aluminum jackets of the slugs. Under these circumstances the reactor can be operated continuously at 250 kilowatts electrical equivalent of heat by passing 32,000 cubic feet per minute through the reactor with a maximum temperature rise of the slugs to about 100° C., and at 500 kilowatts continuously with about 50,000 cubic feet per minute of air with a maximum

metal temperature of 200° C. The output of the reactor can be stabilized at still higher powers by the use of larger fans if desired.

Having discussed generally the operation of the reactor, and the temperature stabilization thereof by air cooling at elevated powers and neutron densities, there is now described one means and method by which the reactor can be loaded and unloaded, in order that the neutron irradiated uranium can be removed for further processing such as the recovery of 94^{239} formed in the uranium, and fresh uranium inserted for subsequent operation of the reactor, and in any amount desired.

To accomplish loading of the slugs **35** into the various air channels **12**, the concrete of the inlet end shield **24** is pierced with a plurality of loading apertures **45**, as shown in Figs. 1 and 5, each aperture being aligned with the axis of slug positions in the air channels **12**. Normally, during operation of the reactor, each aperture **45** is closed by a removable lead plug extending through the shield **24** only.

When it is desired to load a channel with new slugs, the lead plug in shield **24** for that channel alone is removed, and a charging tube **47** inserted, extending through the inlet end shield **24**, across the inlet chamber **25** and entering the corresponding air channel **12** as shown in Fig. 5. The outer end of charging tube **47** is provided with a flanged nipple **49** shaped to engage a nipple recess **50** of a loading mechanism indicated generally by numeral **51**. It will be noted that the charging tube is smaller than the air channel **12** and that air can pass through the channel being unloaded. The air should circulate during unloading, although it may be at reduced velocity.

Loading mechanism **51** comprises a loading magazine **52**, a loading plunger **54**, and a plunger drive **55**, as shown in Figs. 7 to 10 inclusive.

The loading mechanism **51** is mounted on an elevator platform **56** mounted to be raised and lowered in an elevator frame **57** capable of moving along the outside of inlet end shield **24** on elevator tracks **58**. Base **59** of the elevator frame is provided with a platform **60** projecting outwardly on the same level as the top of a supply car **61** travelling on supply car tracks **62**. Supply car **61** is used to bring a supply of slugs to the elevator for use in the loading mechanism **51**.

The slugs **35** when received at the elevator, are loaded into an inclined loading channel **64** in the loading magazine, in side by side relationship and feed by gravity to the bottom thereof. The bottom of loading channel **64** is a part of a plunger bore **65** extending through the loading magazine ending in the nipple recess **50** cooperating with flanged nipple **49** on charging tube **47** so that the plunger bore **65** and the loading bore in guide tube **47** are in concentric alignment. To provide engagement and disengagement of nipple recess **50** and nipple **49**, the entire loading mechanism is movable with respect to elevator platform **56** on wheels **67** running in guides **69** on the elevator platform.

It will be noted that loading magazine **52** is massive. In some instances it may be desirable to load slugs already partially irradiated and in consequence radioactive. The thick walls of the magazine then act as a shield for the radioactive slugs, and in this case a heavy cap **70** may close the upper opening of the loading channel **64**. Iron or lead may be used for the body of the magazine. In addition, the use of thick metal in the magazine, particularly around the plunger bore **65**, reduces radiation that might pass through the interconnected loading aperture **45** and guide tube **47** either from radioactive slugs therein or from the irradiated slugs in the reactor when charging tube **47** is empty.

The slugs are fed from magazine **52** by a reciprocating motion of plunger **54** operating in plunger bore **65**. Plunger **54** may be of iron to act as a shield when inserted into charging tube **47** and is supported outwardly by

plunger bearing 71 on the opposite side of a plunger drive 55.

Plunger drive 55 in simplified form, may be a friction wheel 74 driven by motor 75 as shown in Figs. 7, 9 and 11 opposed by an idler wheel 76 pressed against plunger 54 by spring 77. Motor 75 is reversible and under control of the loading operator. Plunger 54 is sectional, having a threaded end 79 capable of making connection with additional plunger sections. Sufficient sections are provided to insert the plunger entirely through an air channel 12 when required.

In the initial loading of the graphite cube 10, loading is started with the more central air channels, and 68 slugs are placed in the guide tube 47 and pushed into each connected channel. Plunger 54 is operated to push the slugs into the channel until the outer end of the first slug is at the outlet face 15. The plunger is then withdrawn, leaving the outer end of the last slug about 16 inches from the inlet face 14, for purposes explained later.

Proceeding outwardly and preferably concentrically, additional channels are loaded, meanwhile checking the neutronic activity of the reactor. As the activity increases as the loading approaches a critical size, that is, the size where the reproduction ratio will be exactly unity, the approach to critical size can be predicted by extrapolation of observed neutron density values with respect to the volume of the cube loaded with uranium. The neutron density values can be obtained from an ionization chamber 80 (Fig. 2), for example, using any well known indicating circuit, or by measuring the radioactivity of indium foils, for example, induced by neutron irradiation when inserted into the reactor.

As the critical size is approached, the control rod 41 is inserted deeply into the reactor to prevent a self-sustaining chain reaction; and loading is continued until the desired maximum reproduction ratio of, for example, from 1.005 to 1.006 is attained. This ratio can be checked by removal of the control rod and measuring the time taken by the reactor to double its neutron density. From this period, the reproduction ratio can be mathematically computed.

When the desired number of channels are loaded the active core of the reactor may contain from 34 to 50 tons of urnaium, and will be ready for operation. Graphite plugs for the unused air channels may be loaded in a manner similar to that described for the uranium slugs.

It will be noted that on 4 sides of the graphite cube excess graphite will be present. On the fifth side, i. e., at the inlet face, graphite will also extend 16 inches beyond the uranium. On the remaining side, i. e., the outlet face, no graphite extends beyond the uranium. Thus, 5 sides of the active portion (the uranium bearing portion) are surrounded by graphite. This graphite constitutes a reflector and reduces the amount of uranium required to reach critical size.

Graphite and other neutron moderators can be used as reflectors around the active portions of a neutronic reactor, as such materials scatter neutrons passing through them and thereby change the direction of the neutrons. Statistically, due to the scattering action, some of the neutrons that otherwise would be lost to the active portion are returned to it, thus reducing exterior loss and thereby reducing critical size. In graphite-uranium reactors having roughly a spherical shape for the active portion there can be a reduction of 3 to 3½ feet in diameter of the active portion when it is surrounded by a layer of graphite from 3 to 5 feet thick. In the present instance the reduction in size is slightly less as only 5 sides of the active portion are surrounded with the reflecting layer.

It will be noted that more air channels are originally provided in the graphite cube than are required for the disposition of the uranium containing slugs. The manner in which the uranium is disposed in the graphite is known as the geometry of the system, and this geometry may be

varied to suit conditions, for example, the system as described may be said to have rod geometry with cylindrical loading as the end to end relationship of the slugs forms, in effect, a long rod of uranium. Further, the use of full length rods in all channels concentrically arranged around the center of the active portion, gives that portion a generally cylindrical shape.

Lump geometry may also be used in the reactor, and is accomplished by separating the individual slugs by graphite rods, for example 4 inches long. In this case additional channels will be loaded to bring the amount of uranium almost to the same value as that used in the rod geometry, although the lump geometry is slightly more efficient than rod geometry.

Furthermore, a sperical shape is somewhat more efficient for the active portion than a cylindrical shape, and can be approached with either rod or lump geometry by shortening the extent of uranium loading in the channels as the peripheral concentric layers are added, until an approximate sphere is formed. A suitable compromise is the use of one or more concentric rings of outer channels around a cylindrically loaded central portion, with only half the number of slugs in the outer channels and disposed with the ends of the slug row equally distant from the ends of the fully loaded rows. Other arrangements will be apparent to those skilled in the art, and can be attained in the structure described because of the provision of more channels than are to be used.

After the reactor is loaded the fan is started and the control rod is withdrawn until a rise in neutron density to a desired power output is attained. The control rod is then moved forward into the reactor until a neutron balance is obtained with the reproduction ratio at unity, thus maintaining the chain reaction at the desired operating power. Small variations from the unity reproduction ratio will occur during operation, due to temperature variations of the cooling air, and to change in barometric pressure and to minor variations in air pressure delivered by the fan. However, such variations are compensated by slight inward or outward corrective movements of the control rod, either by hand in response to indicated variations in neutron density, or automatically by direct linkage of the control rod to the output of the ionization chamber 80. However, such automatic control is no part of the present invention.

During operation of the reactor, the air passing through the reactor becomes radioactive due to the fact that it is subjected to intense neutron irradiation. Investigation has proved that the only significant radioactivity present in the air after having passed through the operating reactor is that of $Argon^{41}$, having a 110 minute half-life. At high power output, however, this activity may be present in the exhaust air to the point that it would be biologically dangerous to operating personnel unless highly diluted during its radioactive decay. For that reason the air passing through the operating reactor is not delivered to the atmosphere at ground level but is exhausted at a substantial distance above ground, such as for example from the top of a 200 foot stack, with the result that when and if any of the radioactive $Argon^{41}$ reaches ground level it is so dispersed in and diluted by fresh atmospheric air that less than 0.1 Roentgen per day will be received by any persons on the ground, either close to or away from the stack. Thus, the air is passed only once through the reactor and does not acquire excessive radioactivity.

The sole presence of the above noted type of radioactivity, however, is predicated on the use of the jackets sealed around the uranium bodies. Aluminum is preferred for the jackets, as aluminum has a relatively low neutron capture capability and, consequently, can be used in substantial amounts in the reactor without absorbing or capturing sufficient neutrons to prevent a self-sustaining chain reaction from occurring. Aluminum also corrodes very slowly in hot air.

The jackets have two functions, both of which reduce radioactivity of the cooling air. The first is to prevent oxidation of the uranium. While considerable oxide could be tolerated in the reactor itself if the uranium were to be used in unprotected condition, some of the oxide particles would be picked up by and exhausted in the cooling air. As these particles would be highly radioactive and relatively heavy, the proper dispersal thereof would be a difficult problem.

In addition, if operation should be accomplished in the reactor with bare uranium, fission fragments from nuclear fissions occurring on the surface of the uranium would also be projected into the air stream and would be carried out by the air stream. These fragments are exceptionally radioactive and could not safely be dispersed into the atmosphere. When jackets are used, these fragments are stopped by the jackets and cannot enter the air stream.

Thus, the jackets prevent corrosion of the uranium and prevent fission fragments and corrosion products of uranium from entering the air stream. As fission fragments will pass through an extremely small hole, one method of monitoring the reactor for jacket failure, such as for example, a weld crack is to monitor the radioactivity of the stack gas. If the stack gas shows any substantially radioactivity other than that of $Argon^{41}$ then it is clear that a jacket failure has occurred. Such monitoring of the stack gas is normally a routine procedure of an operating air cooled reactor, but forms no part of the present invention.

After operation of the reactor for a sufficient length of time for an amount of 94^{239} to be created sufficient for chemical separation, such as for example 100 days at 500 kilowatts, the reactor is shut down by inserting the control rod fully into the reactor. After about one half hour's wait, during which all delayed neutron omission will have ceased and the more violent radioactivity subsided, the reactor may be unloaded.

The unloading may be accomplished in two ways, either by using the plunger to push the slugs out of the channels so that they fall by gravity out of the outlet face **15**, or by using the plunger to insert new slugs in the channels, each slug so inserted pushing an irradiated slug out of the outlet face **15**. In the first instance the graphite cube will be left empty after unloading. In the second instance the insertion of new slugs is continued until all or a predetermined part of the irradiated slugs are out of the reactor, having been replaced by fresh material. Thus, the reactor is left ready for the next run. Under ordinary circumstances the latter manner of unloading is preferred.

In either case the slugs drop by gravity from the outlet face into outlet chamber **31**, falling on to two angularly disposed pad plates **90** positioned to intersect the falling slugs, in the bottom half of outlet chamber **31** as shown in Figs. 1 and 2. The two plates slant to a centrally disposed outlet pipe **91** extending downwardly through foundation **11** and provided with spaced valves **92** and **94**. The slugs fall by gravity into pipe **91** above valve **92**.

Outlet pipe **91** opens into a lower coffin chamber **93** that in turn connects with a tunnel **96** carrying car tracks **97** on which a coffin car **99** may be moved by means of cable **100**. Coffin car **99** supports a plurality of slug coffins **101** in position to be successively positioned beneath the lower opening of pipe **91**.

Valves **92** and **94** are operated by means of rods **102** and **104**, respectively, from behind a heavy lead shield **105**, as shown in Fig. 1. A crane **106** is used for placing coffin caps **107** on each coffin after it has been filled with irradiated slugs.

Before unloading is started, both valves **92** and **94** are closed, and the upper portion of pipe **91** is filled with water from water inlet pipe **109**. A proper water level is maintained above valve **92** by water outlet pipe **110**. The air circulation is maintained, although it may be reduced to about 25 percent of the operating value. Slugs are then pushed out of the reactor to fall on to pad plates **90** and then roll by gravity into the water in the upper part of outlet pipe **91**.

In order that there be no material damage to the jacketing of the slugs, plates **90** are preferably padded with a soft material that does not deteriorate under neutron irradiation, and that will be able to withstand the slug impacts. A satisfactory pad has been found to be ¼ inch cotton duck on felt laid on wood backed by steel. Combinations of various synthetic elastic materials have also been found satisfactory.

After a number of slugs have been collected above valve **92** sufficient to fill a coffin **101**, unloading is stopped and valve **92** is opened, permitting the slugs and the water to drop through the valve and remain in the space between valves **92** and **94**. Valve **92** is then closed, the water level re-established and unloading continued. In the meantime, valve **94** is opened permitting the slugs and water to fall into one of the coffins **101**. The car is then moved to register the opening of the next coffin with the end of the outlet pipe and the first coffin is capped. The procedure is continued until all of the irradiated slugs are in coffins. These coffins may then be taken to a soaking pit (not shown) to remain until the radioactivity has decayed to a point where the slugs can be submitted to chemical removal of the products formed therein by irradiation. After 100 days' operation the aging period may be about 30 days.

Removal of the irradiated slugs under the conditions specified is performed for two reasons. Firstly, the slugs are so highly radioactive that they cannot be safely approached by personnel without adequate shielding being interposed, and, secondly, for some time after removal from the reactor this radioactivity is so intense that self absorption of the emitted radiations causes self heating of the slugs possibly sufficient to melt the slugs if not cooled in some manner. By unloading during maintenance of the air stream, by dropping the slugs at once into water, and by keeping the slugs in water until the more violent radioactivity has subsided, melting is prevented, as the slugs are cooled as they boil the water in which they are immersed. The slugs are then stored or aged under water until ready for chemical treatment, as for example, for thirty days.

It is to be noted that reactor operation at 250 to 500 kilowatts heat equivalent has been mentioned. These powers, however, are in no way maxima as the operating powers are dependent solely on the air supply available and the permissible maximum slug temperatures. Ordinarily slug temperatures of from 100° C. to slightly over 400° C. are permissible without special treatment of the uranium bodies before jacketing. In case, however, it is desired to operate the reactor with a maximum slug temperature above about 430° C., then precautions should be taken to remove occluded hydrogen from the uranium to less than .0002 percent in order to prevent swelling of the jackets by release of this hydrogen. At temperatures below 430° C. no swelling occurs from the release of occluded hydrogen, as uranium hydride is formed, preventing rise of internal pressure. Above 430° C., however, internal pressure from released hydrogen may, with unprocessed uranium, swell the jackets. The occluded hydrogen, however, is easily removed from the uranium bodies by heating them to a temperature of from 580° C. to 600° C. under continuous evacuation by a vacuum pump until equilibrium is reached. The uranium bodies are then cooled 10 to 20 hours, preferably in an argon atmosphere, and then jacketed. When processed in this manner no swelling will occur at any temperature. Thus, if desired, the reactor can be operated at powers of from 1000 kilowatts to 5000 kilowatts, when the proper amount of air is supplied, and the hottest slugs permitted to rise in temperature to from 400° C. to 500° C.

Having described the reactor with one type of loading,

changes in loading required by operating conditions can be made, as illustrated by the diagrams shown in Figs. 12 to 16.

In Fig. 12 and 14, a loading is shown, for example, where only minor amounts of intermittent power are required. For example, 800 tubes can be loaded providing an excess *r* of just enough to reach the required power, the power being lowered or the reaction stopped before the appearance of the xenon^{135} effect. The solid black areas indicate uranium loading.

If, however, even higher powers are desired, the moderator blocks can, for example, be almost completely filled up, as shown in Figs. 13 and 15, as for example 1600 channels can be loaded to provide an excess *r* of about .01. This will permit, other conditions being consistent, a large continuous power output with some excess *r* available after xenon^{135} poisoning is taken care of, for control of minor variations by the control rod.

Thus by making the moderator blocks large, so that at least sufficient channels are provided to allow enough channels to be loaded for the maximum power contemplated and for maximum expected poisoning both short and long term, the reactor structure except for the loading, need not be changed.

Furthermore, the moderator blocks pierced with more channels than may be initially required, permits loading to be made in various patterns.

The loading in Figs. 12 and 14, and 13 and 15 is what is known as cylindrical loading, in that the active portion defines a cylinder. However, cubical or parallelopiped loading may be desirable, as shown by the diagram of Fig. 16, in cross section. The longitudinal section will be the same as shown in Fig. 13. It will be seen that the cubical loading can also be contracted or expanded in size to meet operating conditions. By progressively shortening peripheral rods an approximation of a spherical overall shape of the active portion can be obtained.

By so providing space for increasing or decreasing the size of the active portion, all presently known operating conditions requiring change in critical or operating size can be obtained. Thus, when the neutron reproduction ratio decreases due to impurities formed during operation of the reactor some plugs of graphite are removed and additional bodies of thermal neutron fissionable material are added to the passages thus provided.

While the theory of the nuclear chain fission mechanism in uranium set forth herein is based on the best presently known experimental evidence, the present invention is not to be bound thereby, as additional experimental data later discovered may modify the theory disclosed. Any such modification of theory, however, will in no way affect the results to be obtained in the practice of the invention herein described and claimed.

What is claimed is:

The method of sustaining a neutronic chain reacting system comprising operating a reactor containing a solid neutron moderator of graphite having a plurality of passages adapted to receive bodies of thermal neutron fissionable material, and containing sufficient bodies of thermal neutron fissionable material in certain of said passages and solid moderator plugs of graphite in other of the passages to achieve a chain reaction having a neutron reproduction ratio greater than unity, removing some plugs of solid moderator from some of the other said passages, and adding more bodies of thermal neutron fissionable material, said additional bodies being added to the passages thus provided when the neutron reproduction ratio decreases due to impurities formed during operation of said reactor and said additional fissionable material being sufficient to maintain the neutron reproduction ratio at greater than unity.

References Cited in the file of this patent

UNITED STATES PATENTS

2,708,656	Fermi et al.	May 17, 1955

FOREIGN PATENTS

114,150	Australia	May 2, 1940
114,151	Australia	May 3, 1940
861,390	France	Oct. 28, 1940
233,011	Switzerland	Oct. 2, 1944

OTHER REFERENCES

Power, July 1940, pp. 56–59.

Smyth: "Atomic Energy for Military Purposes," Aug. 1945.

Goodman: "The Science and Engineering of Nuclear Power," vol. 1, page 275, Addison-Wesley (1947.)

Kelly et al.: Phy. Rev. 73 1135–9 (1948.)