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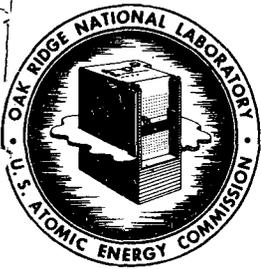
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FIVE LECTURES ON NUCLEAR REACTORS⁽¹⁾

Alvin M. Weinberg

~~In the following five lectures I shall try to summarize~~ the basic issues ^{are summarized.} which ~~are~~ involved in the physics and engineering of nuclear reactors. Since the time allotted for this summary is so short, it will be possible to touch only those phases of the basic technology which are central.

At the outset it should be stressed that nuclear technology has progressed to the point where the question "How can the nuclear fire be ignited" is much less important than the question "How can the nuclear fire be kept burning for a long enough time to be practically useful". The latter question is mostly technological, chemical, and metallurgical in character; the former question is primarily one of nuclear physics and the related reactor physics. Yet, just because the nuclear issues have hitherto received so much attention, we are now in a better position to identify the most important problems related to the nuclear rather than technological design of a chain reactor.

(1) Presented at California Institute of Technology, February 6 - 10, 1956.

Lecture I

Theory of Reactor Design, IThe Two Problems of Nuclear Reactor Design

Since chain reactors are energy or isotope producing devices, the central question to be answered by nuclear reactor theory is the distribution of power, $p(\vec{x}, t)$, or isotope production, $I(\vec{x}, t)$, as a function of position in the reactor, and of time. Once the power distribution is known, it is a matter of engineering (often difficult and demanding engineering) to arrange for cooling and shielding of the reactor. Since the energy is produced mainly as the result of fissioning in the reactor, the power is related to the flux $\Phi(\vec{x}, E, t)$ by

$$p(\vec{x}, t) = E_0 \int \Sigma_f(\vec{x}, E) \Phi(\vec{x}, E, t) dE \quad , \quad (1)$$

where E_0 is the energy released per fission and $\Sigma_f(\vec{x}, E)$ is the fission cross section per cubic centimeter at energy E and position \vec{x} . The flux $\Phi(\vec{x}, E, t)$ is the neutron density $n(\vec{x}, E, t)$ multiplied by the neutron speed; it represents the sum of the speeds of all the neutrons of energy E in a cubic centimeter at \vec{x} at time t . Similarly, the isotope production is related to the flux distribution as (1) but with $\Sigma_a(\vec{x}, E)$, the cross section for absorption leading to the isotope in question, replacing $E_0 \Sigma_f$ in (1).

Since a chain reaction is a device in which the chain carriers - i.e., the neutrons - are themselves produced by the reaction, the time history of the reaction can be subdivided into successive generation, much like human populations. Very special conditions must be fulfilled in order that the number of neutrons produced in the reaction be maintained at a steady state. Broadly speaking, in order for a chain reactor to operate steadily, the number of neutrons produced per second in the reactor must just balance the number lost from the reactor

either by leakage or by absorption. A reactor for which this is the case is said to be "critical". The neutrons are produced by successive generations of fissions in the reactor; thus, if N_1, N_2, \dots, N_n denotes the total neutron population in the reactor in successive generations, the reactor will be critical if N_n/N_{n-1} is unity. In general this ratio will not be unity; this ratio is called the "criticality constant" of the reactor and is denoted by $C (= N_n/N_{n-1})$. A critical reactor has a value of $C = 1$; if C is < 1 , the reactor is sub-critical and the reaction will die; if $C > 1$, the reactor is supercritical and the reaction will grow.

Since a steadily operating reactor must have $C = 1$, it is important to be able to compute C for a proposed arrangement of fissionable material and moderator. The general method for finding C in a non-critical reactor is to compare the reactor with a just critical one which is identical with the reactor in question except that the number of neutrons produced per fission has the value ν' . Since the increase or decrease in population per generation is proportional to the number of neutrons produced per fission, the actual ratio in two successive generations must be the ratio of ν , the actual number of neutrons per fission, to ν' , the number required for criticality - i.e., $C = \nu/\nu'$. By this stratagem it is sufficient in most cases to deal with only just critical reactors.

The Uniform, Bare Homogeneous Reactor

The flux $\Phi(\vec{x}, E, t)$ and the criticality constant C are determined by solving the neutron balance equations. This accurate balance equation is the Boltzmann equation which is an equation in phase space - i.e., it involves the angular distribution of the neutron flux. For most reactor purposes it is possible to use elementary diffusion theory in which the neutron flux is represented by only two spherical harmonics. All of our further considerations will be based on elementary theory.

The simplest reactor is bare, uniform, and homogeneous. "Bare" in reactor language means not only that the reactor has no reflector, but that the extrapolation distance - i.e., the distance beyond the true boundary at which the flux extrapolates to zero - is independent of energy. A reactor of this sort can be treated by considering an identical reactor which extends to infinity, and then seeking those solutions of the reactor equations which vanish on the extrapolated boundary. This technique is similar to that used in the Born-von Karman theory of crystal vibrations.

We therefore consider an infinite medium identical in all its intensive properties with the actual reactor, except that the ν of the comparison reactor is assumed to be such as to make the reactor just critical; hence, Φ does not depend on t . Then the equation of balance is

$$D(E) \Delta \Phi(\vec{x}, E) - \Sigma_a(E) \Phi(\vec{x}, E) + S(\vec{x}, E) = 0 \quad \text{for non-thermal neutrons} \quad (2)$$

$$D_t \Delta \Phi_t(\vec{x}) - \Sigma_{at} \Phi_t(\vec{x}) + S_t(\vec{x}) = 0 \quad \text{for thermal neutrons.} \quad (3)$$

In these equations $D(E)$ is the diffusion coefficient for neutrons of energy E , $\Sigma_a(E)$ the macroscopic absorption cross section, and $S(\vec{x}, E) dE dx$ is the number of neutrons produced in the energy range dE either directly from fission or as the result of slowing down (by elastic or inelastic collisions) of more energetic neutrons less the number leaving the energy element dE as the result of slowing down. Subscript t denotes thermal; in the schematic treatment used here all neutrons below, say, Cd cut-off are viewed as thermal.

The source $S(\vec{x}, E)$ is closely related to the so-called slowing down kernel $K(|\vec{x} - \vec{x}'|, E', E)$; namely, the number of neutrons that appear in unit volume at \vec{x} with energy in unit range around E per second following the production of a single neutron per second at \vec{x}' with energy E' in a medium whose absorption and scattering properties are like the reactor's. This function is in principle measurable though it is more practical to compute it. We denote the spectrum of fission neutrons, normalized to unity, by $f(E)$. Then the number of neutrons of energy E'' produced per second per cc. at \vec{x}' is

$$v f(E'') \left\{ \int_{E_t}^{E_1} \Sigma_f(E') \Phi(\vec{x}', E') dE' + \Sigma_{ft} \Phi_t(\vec{x}') \right\} \quad (4)$$

where E_1 is the top of the fission spectrum, and the number of neutrons produced per second at (\vec{x}, E) as the result of fissions taking place everywhere, and subsequent slowing down, is

$$S(\vec{x}, E) = v \int \int_E^{E_1} f(E'') \left\{ \int_{E_t}^{E_1} \Sigma_f(E') \Phi(\vec{x}', E') dE' + \Sigma_{ft} \Phi_t(\vec{x}') \right\} K(|\vec{x} - \vec{x}'|, E'', E) dE'' d\vec{x}' \quad (5)$$

It should be noted that the source $S(\vec{x}, E)$ given in (5) includes both the neutrons produced directly from fission at \vec{x}, E and those which have slowed in from above minus those slowed out of the energy range. The function $K(\vec{x}, E_1, E_1)$ - i.e., the source of undegraded neutrons - is therefore a delta function $\delta(\vec{x})$ since any neutron found away from the source is assumed to be degraded by scattering. We seek a solution of (2), (3), and (5) which is positive inside the reactor and vanishes on the extrapolated boundary.

To solve the set of equations (2), (3), and (5) it is necessary to use a mathematical theorem sometimes called the first fundamental theorem of reactor theory. This states that if

$$\Delta \psi(\vec{x}) + B^2 \psi = 0 \quad , \quad (6)$$

then

$$\int_{\text{all space}} P(|\vec{x} - \vec{x}'|) \psi(\vec{x}') d\vec{x}' = \psi(x) \bar{P}(B^2) \quad (7)$$

where $\bar{P}(B^2)$ is the Fourier transform of P ; i.e.,

$$\bar{P}(B^2) = 4\pi \int P(x) \frac{\sin Bx}{Bx} x^2 dx \quad . \quad (8)$$

This result will not be proved here. The significance of this result lies in the fact that for ψ 's satisfying (6), the production of neutrons at low energies at a point even after diffusion is proportional to the flux of neutrons producing the original fission.

To apply this result to the equations (2) thru (5), we assume

$$\begin{aligned} \Phi(\vec{x}, E) &= \psi(\vec{x}) \chi(E) \quad , \\ \Phi_t(\vec{x}) &= \psi(\vec{x}) \chi_t \quad , \end{aligned} \quad (9)$$

$$\Delta \psi + B^2 \psi = 0 \quad ,$$

and substitute into (2) thru (5). These equations become

$$\begin{aligned} \left\{ B^2 D(E) + \Sigma_a(E) \right\} \chi(E) &= \nu \left\{ \int_E^{E_1} \int_{E_t}^{E_1} f(E'') \bar{K}(B^2, E'', E) \Sigma_f(E'') \chi(E') dE'' dE' \right. \\ &\quad \left. + \int_E^{E_1} f(E'') \bar{K}(B^2, E'', E) \Sigma_{ft} \chi_t dE'' \right\} \end{aligned} \quad (10)$$

$$\left\{ B^2 D_t + \Sigma_{a_t} \right\} \chi_t = \nu \left\{ \int_{E_t}^{E_1} \int_{E_t}^{E_1} f(E'') \bar{K}(B^2, E'', E_t) \Sigma_f(E') \chi(E') dE'' dE' + \int_{E_t}^{E_1} f(E'') \bar{K}(B^2, E'', E_t) \Sigma_{f_t} \chi_t dE'' \right\} . \quad (11)$$

Upon putting

$$\bar{g}(B^2, E) = \int_E^{E_1} f(E'') \bar{K}(B^2, E'', E) dE'' , \quad (12)$$

we can write

$$\left\{ B^2 D(E) + \Sigma_a(E) \right\} \chi(E) = \nu \bar{g}(B^2, E) \left\{ \int_{E_t}^{E_1} \chi(E') \Sigma_f(E') dE' + \Sigma_{f_t} \chi_t \right\} . \quad (13)$$

$$\left\{ B^2 D_t + \Sigma_{a_t} \right\} \chi_t = \nu \bar{g}(B^2, E_t) \left\{ \int_{E_t}^{E_1} \chi(E') \Sigma_f(E') dE' + \Sigma_{f_t} \chi_t \right\} . \quad (14)$$

This pair of equations for $\chi(E)$ and χ_t has the solution

$$\Sigma_f \chi(E) = \frac{\nu \bar{g}(B^2, E) \Sigma_f(E)}{\Sigma_a(E) + B^2 D(E)} . \quad (15)$$

$$\Sigma_{f_t} \chi_t = \frac{\nu \bar{g}(B^2, E_t) \Sigma_{f_t}}{\Sigma_{a_t} + D B^2} . \quad (16)$$

provided the following characteristic equation holds ($L^2 = \frac{D}{\Sigma_a}$)

$$C = 1 = \nu \frac{\Sigma_{f_t} \bar{g}(B^2, E_t)}{\Sigma_{a_t} (1 + L_t^2 B^2)} + \nu \int \frac{\Sigma_f(E')}{\Sigma_a(E')} \frac{1}{1 + L^2(E') B^2} \bar{g}(B^2, E') dE' . \quad (17)$$

This is the most general expression for the criticality constant of a reactor in which fissions occur at any energy. The expression can be understood physically if we define first a thermal multiplication constant, k_t , as

$$k_t = v \frac{\Sigma_{ft}}{\Sigma_{at}} \quad (18)$$

and note that the probability that a fission neutron will not leak out of the reactor nor become absorbed before it becomes thermal is

$$P_t(B^2) = \frac{\bar{g}(B^2, E_t)}{1 + L_t^2 B^2} \quad (19)$$

Similarly, we define the multiplication constant at energy E' as

$$k(E') = v \frac{\Sigma_f(E')}{\Sigma_a(E')} \quad (20)$$

and note that the probability that a fission neutron will not leak out nor become absorbed before reaching energy E' is

$$P(B^2, E') = \frac{\bar{g}(B^2, E')}{1 + L^2(E') B^2} \quad (21)$$

Thus, the criticality constant is

$$C = 1 = k_t P_t(B^2) + \int k(E') P(B^2, E') dE' \quad (22)$$

which, in a sense, summarizes all of bare reactor theory.

The essential simplicity of bare reactor theory stems from the fact that for a bare uniform reactor the flux is separable in space and energy. Such a flux distribution is called "asymptotic".

Lecture II

Theory of Reactor Design, IIThe Two Classes of Chain Reactors

Chain reactors of technological importance fall roughly into two classes - the ones which use unenriched or slightly enriched uranium, and which are characterized by k close to unity, and the ones which use highly enriched fuel and are characterized by k closer to ν . The original Hanford reactors were of the low k type; the breeders are necessarily high k devices. The design theory of these two reactor types is necessarily quite distinct; for example, the low k systems are necessarily thermal reactors because only at thermal energy does the U^{235} fission cross section become large enough to compensate for the very large amount of U^{238} present in the reactor. The high k systems may have any energy spectrum depending on the ratio of moderator to fuel, although usually they are resonance reactors. The beauty of the general theory summarized in (22) is that the two entirely different reactors - resonance and thermal - can be discussed as special cases of (22).

Thermal, Low k Reactors

In a natural uranium reactor almost all of the fission is caused by thermal neutrons. What little epithermal fission occurs is primarily due to very fast neutrons which collide with U^{238} nuclei before the virgin neutrons have left the fuel lumps. The fuel usually is disposed in lumps in order to reduce the U^{238} resonance absorption, although if the moderator is extremely efficient, as D_2O , a natural uranium chain reaction can be sustained even without lumping.

Since the fast fissions occur with neutrons which essentially have not slowed down - i.e., have had no opportunity to diffuse - the non-leakage probability for these neutrons is essentially unity. Hence, the second term in (22) can be summarized by a single number

$$k_f = \int v \frac{\Sigma_f(E)}{\Sigma_a(E)} dE \quad (23)$$

and the critical condition may be written

$$1 = \eta f P_t(B^2) + k_f \quad (24)$$

where η = number of neutrons produced per neutron absorbed in fuel

$$= v \frac{\Sigma_{ft}(\text{fuel})}{\Sigma_{at}(\text{fuel})} \text{ and } f = \text{thermal utilization} = \text{number of neutrons absorbed in fuel/number absorbed in reactor} = \frac{\Sigma_{at}(\text{fuel})}{\Sigma_{at}}$$

We now define a "fast effect" constant ϵ as the ratio of all fissions to thermal fissions in an infinitely large system having the same microscopic properties as the reactor in question. For such a system the total multiplication constant $k = 1$; hence,

$$\epsilon = \frac{k}{k_s} = \frac{k}{k - k_f} = \frac{1}{1 - k_f}$$

or

$$k_f = 1 - 1/\epsilon \quad ; \quad (25)$$

and the criticality condition may be written

$$\eta \epsilon f P_t(B^2) = 1$$

Finally, since $P_t(B^2)$ includes the probability p that a neutron survive resonance capture, we can write

$$P_t(B^2) = p P_t'(B^2) \quad (26)$$

where $P_t'(B^2)$ is the non-escape probability in a system identical with the one in question except that there is no resonance capture. The criticality equation is therefore

$$\eta \epsilon p f P_t'(B^2) = 1 \quad (27)$$

in which the combination $\eta \epsilon p f = k$ is the usual multiplication factor. This equation is the ordinary four-factor equation for neutron multiplication in a low enriched, predominantly thermal reactor.

TABLE I

Comparison of high k , enriched systems, and low k , unenriched

	<u>High k, enriched</u>	<u>Low k, unenriched</u>
Size	usually small	large
Energy spectrum of fissions	usually high, though can be thermal	thermal
Flux distribution	usually non-asymptotic - hence, not separable in E, \bar{x} - because reflector saving large	nearly asymptotic (hence separable) because reactor large compared with reflector
Critical equation	$\int k(E) P(B^2, D) dE + k_t P_t = 1$	$\frac{k}{1 + M^2 B^2} = 1$
Main calculational problem	Computation of macroscopic distribution	Computation of microscopic distribution

A very important result for a low k reactor follows from the fact that if $k \approx 1$, $B \rightarrow 0$. Hence

$$P_t^i(B^2) \Rightarrow \frac{p\bar{g}(B^2, E_t)}{1 + L^2 B^2} ,$$

$$\begin{aligned} \bar{g}(B^2, E_t) &\Rightarrow \int_{E_t}^{E_1} f(E'') \bar{K}(B^2, E'', E_t) dE'' \Rightarrow 4\pi \int_{E_t}^{E_1} \int_0^\infty f(E'') K(|x|, E'', E_t) \left(1 - \frac{B^2 x^2}{6}\right) x^2 dx \\ &= 1 - \frac{\bar{X}_t^2}{6} B^2 \approx \frac{1}{1 + \frac{\bar{X}_t^2}{6} B^2} \end{aligned} \quad (28)$$

where \bar{X}_t^2 is the mean square distance for slowing down to thermal energy, and k is assumed so close to unity that $B^2 \bar{X}_t^2 \ll 1$. Hence for such reactors the criticality condition is the familiar

$$C = \frac{k}{1 + M^2 B^2} = 1 \quad (29)$$

where M^2 , the migration area, is roughly $\frac{\bar{X}_t^2}{6} + L^2 = \frac{1}{6}$ x mean square distance from birth as a fission neutron to death as a thermal neutron. Since B^2 , for say a spherical reactor of radius R is $\frac{\pi^2}{R^2}$, equation (29) immediately gives the critical radius of a sphere, namely $R = \pi \frac{M}{k - 1}$.

Since low k reactors are large, the conditions at the boundary hardly affect the neutron distribution. Hence, so long as the reactor is loaded

uniformly, the power distribution will be asymptotic over most of the reactor; for example, in a spherical reactor it will be $P(r) = P_0 \frac{\sin \pi r/R}{\pi r/R}$, R being the radius of the reactor.

Thus the problem of the gross neutron distribution in a Hanford-type reactor is almost trivial - the asymptotic distribution is very good except very near the edge. On the other hand, the values of the cross sections to be used in the original equations - or, in related manner, the values of f , p , ϵ , - are by no means obvious since on the small scale low k reactors are usually heterogeneous. The problem of choosing the correct averages for the cross sections in heterogeneous reactors is the subject matter of "microscopic" reactor theory.

High k , Enriched Reactors

At the opposite extreme are reactors which use highly enriched fissionable material as the fuel. The fuel may still be disposed heterogeneously, but, since the resonance absorption of U^{238} is no longer important, this is largely a matter of engineering choice, not of nuclear necessity. The energy spectrum in such a reactor may be anything from several hundred kilovolts (in an unmoderated reactor) down to thermal energy if the reactor is heavily moderated. This choice is usually dictated by the necessity for conserving neutrons - as in a breeder which, because of the favorable high E value of σ_f/σ_c in the fissionable isotopes, tends to be a high neutron energy device - or by the requirement that the reactor size be limited by engineering considerations. Thus, if the reactor is very small, a relatively high concentration of fuel is required to make it chain react, and this implies a high neutron spectrum if the moderator is not a very efficient one.

Since high k reactors are small, the asymptotic theory described above can hardly be accurate; the neutron distribution is often profoundly affected

by the boundary or the reflector. To calculate the distribution it is therefore necessary to have some means for explicitly computing the slowing down kernel in media which are not uniform. If inelastic scattering can be neglected, and if the moderator is non-hydrogenous, this is afforded by the "age" theory. We therefore turn to a description of age theory in the simplest uniform system, and later it will be applied to non-uniform systems.

In an infinite medium age theory predicts for the number of neutrons crossing energy E per second per c.c. at distance r from a point, monoenergetic source at energy E' , the expression

$$q(r, E, E') = \frac{1}{\left[4\pi \tau(E, E')\right]^{3/2}} e^{-r^2/4\tau(E, E')} \quad (30)$$

where

$$\tau(E, E') = \int_E^{E'} \frac{D(E)}{\Sigma_S \xi} \frac{dE}{E} = \frac{1}{6} \times \text{mean square distance from birth}$$

to energy E if the medium is non-absorbing, or

$$q(r, E, E') = \frac{1}{\left[4\pi \tau(E, E')\right]^{3/2}} e^{-r^2/4\tau(E, E')} - \int_E^{E'} \frac{\Sigma_a}{\Sigma_S \xi} \frac{dE}{E} \quad (31)$$

if the medium absorbs with absorption cross section Σ_a .

Herein ξ is the log energy decrement per collision $= 1 - \frac{\alpha^2}{(\alpha - 1)^2} \ln \alpha^2$,
 $\alpha = (M - 1)/M + 1$, $M =$ atomic weight. The Fourier transform of $q(r, E, E')$ is

$$\bar{q}(B^2, E, E') = e^{-B^2 \tau(E, E')} - \int_E^{E'} \frac{\Sigma_a}{\Sigma_S \xi} \frac{dE}{E} = e^{-\int_E^{E'} \frac{DB^2 + \Sigma_a}{\xi \Sigma_S} \frac{dE}{E}} \quad (32)$$

Now $q(r, E, E')$ being the number of neutrons crossing from above per second is related to $K(r, E, E')$, the number appearing in unit energy range per second by

$$K(r, E, E') = \frac{\partial q}{\partial E}(r, E, E') ; \bar{K}(B^2, E, E') = \frac{DB^2 + \Sigma_a(E)}{\xi \Sigma_s(E)} e^{-\int_E^{E'} \frac{DB^2 + \Sigma_a}{\xi \Sigma_s} \frac{dE''}{E''}} .$$

Hence, according to this age theory model,

$$\bar{g}(B^2, E) = \int_E^{E_1} f(E'') \bar{K}(B^2, E'', E) dE'' = \int_E^{E_1} f(E'') \frac{DB^2 + \Sigma_a(E)}{\xi \Sigma_s} e^{-\int_E^{E_1} \frac{DB^2 + \Sigma_a}{\xi \Sigma_s} \frac{dE'''}{E'''}} dE'' .$$

The thermal $\bar{g}(B^2, E_t)$, since no neutrons leave the thermal group by slowing, does not involve the energy derivative of \bar{K} ; i.e.,

$$\bar{g}(B^2, E_t) = \int_{E_t}^{E_1} f(E'') e^{-\int_{E_t}^{E_1} \frac{DB^2 + \Sigma_a}{\xi \Sigma_s} \frac{dE'''}{E'''}} dE'' .$$

Hence, in age theory the critical equation (17) is

$$C = 1 = \nu \frac{\Sigma_{ft}}{\Sigma_{at}} p_t \frac{e^{-B^2 \tau(E_t)}}{1 + L_t^2 B^2} + \nu \iint \frac{\Sigma_f(E')}{\Sigma_s \xi(E')} f(E'') p(E, E'') e^{-B^2 \tau(E'', E)} dE'' dE' \quad (33)$$

where $p(E, E'') = e^{-\int_E^{E''} \frac{\Sigma_a}{\Sigma_s \xi} \frac{dE'''}{E'''}}$, (the resonance escape probability from energy E'' to E), and p_t is the resonance escape probability to thermal energy. For a purely thermal fission reactor, the criticality constant in age theory approximation is simply

$$C = 1 = k e^{-B^2 \tau} (1 + L_t B^2)^{-1}, \quad k = v(\Sigma_{ft}/\Sigma_{at}) P_t .$$

The main purpose of introducing the age theory at this point is that it enables us to relate $q(\vec{x}, E)$ to the flux at (\vec{x}, E) ; namely,

$$q(\vec{x}, E) = \Sigma_g \xi E \Phi(\vec{x}, E) . \quad (34)$$

For small, non-uniform reactors this is very important since the integral theory just described is not applicable, while age theory makes possible the treatment of non-uniform reactors. The material balance equation for a non-uniform reactor in age approximation is therefore

$$\begin{aligned} \text{div } D(\vec{x}, E) \text{ grad } \Phi(\vec{x}, E) - \Sigma_a(\vec{x}, E) \Phi(\vec{x}, E) + v f(E) \left[\int_E^{E_1} \Sigma_f(\vec{x}, E) \Phi(\vec{x}, E) + \Sigma_{ft} \Phi(\vec{x}, E) \right] \\ + \frac{\partial}{\partial E} \left\{ \Sigma \xi E \Phi(\vec{x}, E) \right\} = 0 \quad \text{non-thermal} \end{aligned} \quad (35)$$

$$\text{div } D_t(\vec{x}) \text{ grad } \Phi_t(\vec{x}) - \Sigma_{at}(x) \Phi_t(\vec{x}) + \Sigma \xi E_t \Phi(\vec{x}, E_t) = 0 \quad \text{thermal} \quad (36)$$

where Σ replaces Σ_g in the source term; this can be shown to be a better approximation than (34). In general, these equations are not separable in space and energy. Thus, the asymptotic distribution no longer plays a central role in the theory; instead, the actual solution can be considered to be a linear superposition of numerous non-asymptotic solutions.

A special, and very important, example of a non-uniform reactor is one which is uniform over its central core and which is surrounded by a non-multiplying reflector. Far from the reflector-reactor interface the asymptotic solution will tend to prevail. As the interface is approached, the non-asymptotic solutions tend to become important: energy spectrum gradually changes from the spectrum

inside the uniform core to that characteristic of the non-multiplying reflector.

Numerical techniques must be resorted to in order to solve the non-uniform reactors in general. The most commonly used technique is the so-called multi-group method; in this scheme the energy interval is divided into n groups, and the parameters of the reactor are averaged over each group. In this way the differential equation in energy and space is reduced to a system of coupled linear differential equations in space alone. Such systems are amenable to numerous numerical techniques.

The most useful such technique is the method of iteration. In this scheme one starts with a single neutron in the most energetic group distributed uniformly in space, and successively computes the neutron distribution in each successive group resulting from this spatially uniform high energy neutron. Since the high energy group neutrons are produced by fission in the lowest energy group, the high energy group distribution can be recomputed from the derived low energy distribution. This corresponds physically to tracing the neutron through a complete life cycle of one generation. As this process is repeated the spatial distribution of the neutrons changes from the originally assumed distribution (in this case uniform) to the distribution actually characteristic of the steadily operating reactor; that is, eventually the spatial distribution of the neutrons ceases to change after iteration. The ratio of the neutron population in two successive iterations (i.e., in successive generations) is the criticality constant.

Lecture III

Technical Problems in Power ReactorsTechnical Problems in Reactor Design

A chain reactor from the technical standpoint is a combination of numerous systems - it is a nuclear system, a heat transfer system, a control system, a shielding system, a mechanical system. The design of a chain reactor involves optimization, in some sense, of each of these systems. Since all of the systems are inter-related, it is never possible absolutely to optimize each of the systems - rather, compromises must always be made; technical advantages in one system being given up to make up for technical disadvantages in other systems. In this respect - that reactor design involves compromises between conflicting demands of different systems - chain reactor design is much like the design of any other engineering device.

The dominant requirement in any chain reactor is that it indeed chain react; that is, that the criticality constant not be less than unity. Almost any measures which may be taken to rationalize the engineering design of a chain reactor - for example, the introduction of cooling fluid, or the dispersion of the fuel to make it easier to extract heat from it, or the introduction of structural members (which absorb neutrons) - tend to reduce the criticality constant. Thus, the amount of freedom available to the designer for the purpose of improving the non-nuclear systems depends primarily on the amount by which the criticality constants exceed unity before the nuclear system is compromised by the requirements of the other engineering systems.

Since the importance of the available criticality constant depends upon the amount by which the "uncompromised" criticality constant exceeds unity,

the two classes of reactors previously described - the low k unenriched ones and the high k enriched ones - are different in respect to the importance of nuclear design. The criticality constant is always of the form

$$C = k P ,$$

P being a non-leakage probability, and therefore less than unity; the maximum of C can therefore never exceed k. Hence the nuclear design dominates the over-all design of the low k, unenriched systems, to a much greater extent than it does the high k systems - in the latter system much more compromise is tolerable before C becomes less than unity and the system ceases to chain react. The importance of the nuclear design in a high k system arises mostly in situations where the device is size limited - this implies that P has a fixed (often small) value and therefore the maximum value of k may be quite far from what is actually available.

We consider now a few examples of how chain reactor design proceeds - our choice of example will serve to illustrate the difference between the small k, unenriched systems, and the large k, enriched systems.

1) A water-cooled, graphite-moderated, unenriched reactor. Assume that the maximum U^{235} isotopic enrichment available is 0.7% (compared with 0.72% in natural uranium).

In such a reactor the following design choices must be made:

a) Fuel-heterogeneous or homogeneous: This is simple since at 0.7% enrichment the reactor will not chain react unless it is heterogeneous.

b) Cladding material for the fuel elements: Of the possible

materials with low cross-section - Zr, Mg, Al - Zr has the best high-temperature properties, Mg the lowest cross-section, Al the lowest cost. Since the reactor will run at pressure and temperature of, say, 250°C, Al and Mg are probably marginal: this leaves the very expensive Zr as an almost unique choice.

c) Fuel element construction: The fuel element should have the highest possible surface to volume ratio, since this allows easiest heat transfer. However since high surface tends to increase resonance absorptions, and thus reduce k , there is a limit to how much the fuel can be subdivided.

For considerations of simplicity, it has been customary to cool fuel rods with a thin layer of water which passes rapidly over the rod surface. The maximum radius of the rod is fixed by the requirement that the maximum temperature at the rod center not exceed the $\alpha - \beta$ transition temperature in uranium metal (660°C). The temperature rise at the center of a slug is

$$\Delta T = \frac{q r_0^2}{4 \kappa}$$

provided the heat is produced uniformly inside the rod - herein q is the heat production per c.c. per second, κ is the thermal conductivity.

Now r_0 , from nuclear considerations can hardly be less than 0.5 cm.; otherwise the resonance absorption will be so high as to render the reactor non-chain reacting. Taking for κ the value .07, and for $\Delta T \sim 660^\circ\text{C} - 200^\circ\text{C} = 460^\circ\text{C}$, this gives 400 cal/cm³/sec or 20 cal/gm/sec as the maximum possible heat rate in the reactor.

This is the maximum possible heat rate. We now assume that the reactor is an unreflected cube so that its distribution is

$$\Phi(\vec{x}) = \cos \frac{\pi x}{H} \cos \frac{\pi y}{H} \cos \frac{\pi z}{H} ;$$

the maximum-to-average rate of heat production is $(\frac{\pi}{2})^3 = 3.24$. Thus the average heat production in such a reactor is reduced to only $20 \div 3.24$ or 6.1/cal/gm/sec.

It must be remarked that one of the aims of power reactor design is to maximize the output - of heat, or produced fissionable material - from any materials placed in the reactor. This is important from the economic viewpoint, since the heat producing elements are expensive, and the overall cost is reduced by "working" each element as hard as possible. It is therefore important in such reactors to try to "flatten" the flux - that is, to make the distribution Φ be as nearly uniform as possible. A reflector, of course, helps flatten the flux. A more drastic way of flux flattening is to distribute neutron absorbing poison (say, boron or U^{238}) preferentially where the flux is high. This has the effect of pulling the flux down in the high places, but at the expense of reducing the multiplication constant. How much such flux flattening can be tolerated depends most strongly upon what the originally available k is. Thus we have here one example of what the reactor designer can use "extra" k for - viz., flux flattening.

The actual design of such a reactor, which is necessarily heterogeneous, begins with a parameter survey of the system's nuclear properties. The primary variables to be investigated are the diameter of the fuel rod (if the fuel is in rod shape), and the rod spacing or lattice pitch; subsidiary variables are

the thickness of metal sheathing on the rods and thickness of coolant layers around the fuel.

The criticality constant for such a reactor is of the form

$$C = \frac{\eta \epsilon p f}{1 + M^2 B^2} = \frac{k}{1 + M^2 B^2} .$$

Of the quantities entering in k , p and f depend strongly on the moderator-to-fuel ratio, as well as on the fuel rod size, while ϵ depends primarily on the fuel size. The quantity η is essentially independent of the lattice geometry. Of the two quantities that make M^2 , τ , and L^2 , the age is rather independent of the fuel-moderator ratio, while L^2 , being inversely proportional to the slow neutron lifetime in the lattice, is given by $(1 - f) L_0^2$ (L_0 is the diffusion length in the moderator).

The resonance escape probability and the thermal utilization vary in opposite senses as the moderator-to-fuel volume ratio is changed: p increases as the moderator-fuel ratio is increased since the stronger the slowing down the less likely it is that a neutron will be caught in a U^{238} resonance; f decreases as the moderator-to-fuel ratio increases because the moderator competes more favorably for neutrons which would otherwise be absorbed in fuel. Thus, in a plot of f and p vs. volume ratio (V_m/V_u), one would have

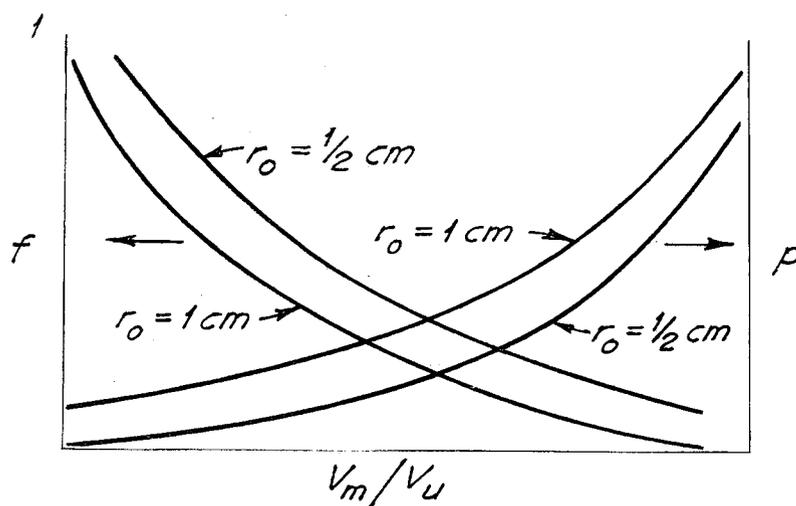


Figure 1

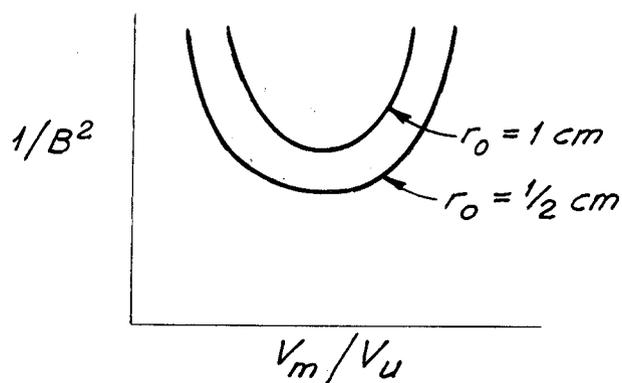


Figure 2

Since f and p vary in opposite senses, their product fp must go through a maximum - this corresponds nearly to the maximum k . Similarly $B^2 = \frac{k-1}{M^2}$ will go through a maximum at a point near to but not the same as the k optimum. The choice of design point on the V_m/V_u (i.e., lattice pitch scale) depends on what engineering parameter is to be optimized. Each such optimization implies a move off the k optimum, and the extent to which this

can be done of course depends upon what the original value of k is. Some examples follow.

- 1) Minimum size reactor - this implies maximizing $B^2 = \frac{k - 1}{M^2}$, not k . This occurs on the uranium-rich side of the maximum k .
- 2) Maximum Pu production - this implies a small p ; i.e., working on the uranium-rich side of Figure 2.
- 3) Maximum heat transfer - as has been mentioned, this requires very small r_0 ; however, as the curve shows, the optimum B^2 occurs where $r_0 = 1$ cm - at $r_0 = 1/2$, B^2 has already been compromised.
- 4) Stability against H_2O loss - the cooling water has two contrary effects: it increases resonance escape because of its strong moderating power and it decreases f because it absorbs neutrons. If one works far toward the uranium-rich side - where f is close to 1 so that loss of water would not improve f very much, but where p with H_2O is very low, say, 0.7 - loss of water would reduce p more than it improves f . Hence, on this side the reactor tends to be stable against inadvertent loss of H_2O . To exploit this, of course, it is necessary that fp be, at its maximum, very high; otherwise, such a drastic move off the maximum would make the reactor non-chain reacting. Only in D_2O -moderated systems is the available k high enough to make this stabilization work.

Design of Enriched Reactors

With enriched reactors the number of possibilities is so great that it is hard to establish design principles which are of general applicability. Enriched fuel is used in a chain reactor for two basic reasons:

- 1) to make the size of the reactor small;
- 2) to achieve very high conversion ratio - if possible, ultimately to breed.

The small reactors are generally used for propulsion or are portable; the breeders are expected to be central power producers. We consider these somewhat separately.

For a mobile power source, say, a marine power plant, the design restrictions are the following:

- 1) maximum size of the chain reactor;
- 2) temperature of outlet fluid;
- 3) over-all power of reactor.

The choice of coolant, moderator, fuel - these are matters on which there are hardly any general criteria. For example, H_2O has the great merit of combining both moderator and coolant functions; its low critical temperature and high critical pressure are distinct disadvantages. Sodium is an excellent heat transfer medium but it absorbs neutrons, it becomes extremely radioactive, and it cannot be used as a moderator. Since no very clear choice exists between these two engineering scale experiments, the STR and the SIR have been built to compare them.

Once the size of the reactor and its general configuration have been settled - essentially entirely on heat transfer and mechanical engineering grounds - the problem of making the "black box" chain reacting is taken up. This is accomplished by "salting" fuel into the device until $C = 1$. Now as more fuel is "salted", the probability of a resonance neutron causing fission increases - thus the energy spectrum is the last thing determined in such a chain reactor - it comes out automatically after the size and structural and thermal properties of the device have been settled.

Lecture IV

Physical Problems in Nuclear Power ProductionSome Physical Problems in Chain Reactor Design

The physical problems of chain reactors fall basically into three categories:

- 1) Problems concerned with ignition of the nuclear fire; i.e., questions of criticality, etc. These are mainly questions of relevant cross sections.
- 2) Problems concerned with the operability of the chain reactor; these mainly center on control of the reactor, and involve particularly the effect of the temperature on the chain reaction.
- 3) Problems concerned with the long term operability of the reactor. These are matters of fission product poisoning, of the cross sections needed for breeding (since, in a sense, a true breeder can operate indefinitely); and in an entirely different field, the questions of radiation damage to the solids of the reactor.

We consider a few of these problems in some detail.

The most significant nuclear cross sections are of course the fission cross sections, σ_f , and the number of neutrons per fission, ν . Second comes the absorption cross sections of fissile, structural, and fertile materials; and, finally, the transport cross sections. The cross sections are required at all energies in principle up to the top of the fission spectrum.

In the region of interest, the cross sections for capture can be well represented by the B - W one-level formula

$$\sigma_a(E) = \frac{\pi \hbar^2}{p^2} \frac{\Gamma_r \Gamma_n}{(E - E_0)^2 + \frac{1}{4} \Gamma^2} \text{ gc/g} \quad (37)$$

For fission, Γ_r is replaced by Γ_f .

However, for reactors in which the spectrum is fairly widely distributed, the average of (37) over many levels is adequate. This is (summed over both spin states)

$$\bar{\sigma}_a(E) = \frac{2\pi^2 \hbar^2}{p^2 D} \frac{\Gamma_r \Gamma_n}{\Gamma_n + \Gamma_r}$$

Now the neutron width in terms of the reduced width γ_n^2 is

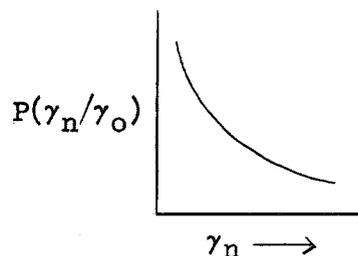
$$\Gamma_n = \frac{2\gamma_n^2}{|I_n|^2}, \quad I_n = \text{penetrability} = \left(\frac{\hbar}{p}\right)^{1/2} \text{ for small } pa/\hbar, \quad l = 0.$$

Hence $\bar{\sigma}_a(E)$ can be expressed, for $l = 0$ neutrons, in terms of the so-called "strength functions" $f = \gamma_n^2/D$ as

$$\bar{\sigma}_a(E) = \frac{1.8 \times 10^{+16} f}{\sqrt{E}} \frac{\Gamma_r}{\Gamma_r + 0.44 \times 10^{10} f D E^{1/2}} \left(= \frac{4\pi^2 \hbar^2}{p D} \gamma_n^2 \frac{\Gamma_r}{\Gamma_r + 2\gamma_n^2 p/\hbar} \right)$$

if all reduced widths are constant. Actually, the neutron widths fluctuate, and γ_n^2 is distributed like $P(\gamma_n/\gamma_0)$; hence, $\bar{\sigma}_a(E)$ should be changed to

$$\bar{\sigma}_a(E) = \frac{1.8 \times 10^{16} \Gamma_r}{\sqrt{E} D} \int \frac{P(\gamma_n^2) \gamma_n^2 d\gamma_n^2}{(\Gamma_r + 2\gamma_n^2 p/\hbar)}$$



This averaging tends to reduce the capture cross section since it stresses the large neutron width levels, and these, which have a large probability for compound nucleus formation, also have large probability for scattering (rather than absorption).

According to the above formula, the average capture cross section falls first as $1/\sqrt{E}$ until the neutron width exceeds the radiation width (at a few thousand volts for $A = 100$, at lower energy for $A = 200$), and then falls as $1/E$. However, this ignores the higher angular momenta. A formula due to Wigner which takes them into account is

$$\bar{\sigma}_a(E) = \frac{2\pi^2 \hbar^2}{p^2 D} \Gamma_r \sum_l \frac{2l + 1}{1 + |I_l|^2 \Gamma_r / 2\gamma_n^2}$$

$$I_0 = (\hbar/p)^{1/2}$$

$$I_1 = (\hbar/p)^{1/2} (\hbar/pa)$$

$$I_l = (\hbar/p)^{1/2} (\hbar/pa)^l [1.3 \dots (2l - 1)]$$

assuming D , γ_n^2 , and Γ_r are independent of l . The general effect of the higher angular momenta is to increase the cross section at higher energy, and to wipe out the $1/E$ resonance region. The effect is most pronounced for large γ_n^2/Γ_r ($= 5 \times 10^{-13}$ to 500×10^{-13} cm). Since $\gamma_n^2 \approx \hbar D$ (roughly) $\gamma_n^2/\Gamma_r \approx \hbar D/\Gamma_r$ the importance of the deviation tends to be greater for high D (low A or even-even nuclei). The effect of the higher angular momenta can be seen in the following table.

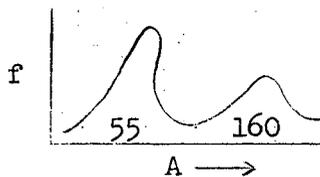
	E	$\bar{\sigma}_a(E)$ (all l)	$\sqrt{\sigma}_a(E)$ ($l = 0$)
A = 100	4 kev	1.5 (D \cong 100 ev)	
		1.0 (D \cong 1 ev)	
	40 kev	4 (D \sim 100 ev)	
		1.3 (D \sim 1 ev)	
	200 kev	\sim 20-30 (D \sim 100 ev)	
		3 (D \sim 1 ev)	

At extremely high energy all $|I_{l\ell}|^2$ become equal to \hbar/p , the sum then approaches the usual

$$\sigma_a(E) = 0.4 A^{2/3} \Gamma_r/D = \pi R^2 \frac{2\pi \Gamma_r}{D}$$

However this occurs at energies considerably above the fission spectrum energy for $A \sim 50$.

To apply these formulas one needs the strength function $f = \gamma_n^2/D$, Γ_r , and D as function of excitation energy, and $P(\gamma_n)$. The low energy resonance data give γ_n^2/D , D , and $P(\gamma_n^2)$; theoretically the cloudy crystal ball should give f (Fig. 1). For Γ_r there is the velocity selector work, and the danger coefficients. For D as function of excitation energy, the statistical theory has been used.



The best check with experiments is with U^{238} . Its level spacing is about 20 ev, and many of its levels are resolved. L. Dresner at Oak Ridge has fitted

the curves to $U^{238} \sigma_a(E)$ within about 20%. Fission product cross sections have been estimated this way independently by two groups - one at KAPL (Hurwitz and Greebler), the other at Varenna (Businaro, Gallone, and Morgan). The results of the two estimates are given in the following table.

TABLE I

$\bar{\sigma}_p$ = fission product cross section per fission

	E(ev) 10^2	10^3	10^4	10^5	10^6	10^7
$\bar{\sigma}_p$ (KAPL)	15	3.1	0.6	0.11	0.02	
$\bar{\sigma}_p$ (Varenna)	47	9.9	1.7	0.3	0.24	all σ 's in barns
$\bar{\sigma}_f$ (U^{235})	23	8.5	3.7	1.7	1.3	

The difference between Varenna and KAPL is that KAPL used much larger D for the even Z - even N nuclides. However, neither group took the higher angular momenta into account properly; and, as has been indicated, large D implies that the higher angular momenta are more important and would tend to increase the importance of the high angular momenta. As can be seen, the poisoning due to fission products in the intermediate energy range can indeed be catastrophic.

Fission Cross Sections - ν , η

Much has been learned about these cross sections in the last ten years. In particular, $\frac{\sigma_c}{\sigma_f} = \alpha \sim 0.5$ for Pu^{239} and U^{235} over the low energy spectrum; it falls off to about 0.15 at 200 kev, and lower at higher energy. Moreover, the Γ_f 's fluctuate widely from level to level. This fluctuation is attributable by Bethe to the fact that fission requires the nucleus at the transition state

to be in a single mode. The widths fluctuate between 0 and 180 millivolts.

Attempts have been made by Oleksa and Wigner to explain the fall in α as due to the preferential increase in scattering caused by the distribution of Γ_n 's - the distribution stresses the high values of Γ_n at the expense of the lower ones. This has been only moderately successful, and it is now believed that the average fission width increases strongly from 0 to 200 kev. This is suggested by the cross section in U^{234} , U^{236} , U^{238} measured at Oak Ridge - they all show a large increase in σ_f within 200 kev above threshold.

Since U^{233} does not show this reduction in α ($\alpha \leq 0.1$ throughout) the outlook for thermal breeding with U^{233} must be deemed good. For Pu^{239} , α becomes very large in the resonance region, and it is therefore necessary to go to high energy for a Pu breeder.

	U^{233}	U^{235}	Pu^{239}
α_{th}	~ 0.1	.18	.42
$\alpha_{\sim 100 \text{ eV}}$.52	.78
α_{103}	~ 0.1	.48	.50
α_{104}		.35	.43
α_{105}	~ 0.1	.13	.18
$\alpha_{5 \times 10^5}$.1	.1

Whether ν varies from resonance to resonance is a moot point. There is conflicting evidence from Brookhaven and Hanford on this point, and the final answer is not clear.

Fission Product Poisoning in Thermal Reactors

In thermal reactors the fission product poisoning is dominated by a few very large poisons - Xe^{135} and Sm^{147} . However these saturate rather quickly, and so, on the whole the thermal fission product problem looms as less serious

than the fast neutron fission product problem.

The nuclide Xe^{135} is of course entirely remarkable - it is one short of a magic neutron, and its level spacing ought therefore to be quite large. The detailed measurement of its cross section (at Oak Ridge) has given the result $\sigma_c \sim 3.5 \times 10^6$ barns. $E_1 = .08$ ev.

Lecture V

Future Developments in Nuclear PowerTrends in Reactor Development

The ultimate aim of nuclear reactor technology is to burn the two raw materials, uranium and thorium, economically and usefully. In the short run systems which burn only U^{235} without any conversion, or with little conversion, are bound to be of interest; even in the long run such systems will be useful for compact power plants.

Actually the present cost of U^{235} - between \$15 per gram and \$30 per gram - implies a power cost of about 3 to 6 mills/kwh if the U^{235} is converted to power with 25% efficiency, and if there is no regeneration of the fuel. Such a cost is already competitive in many situations, but it is achievable only if the U^{235} can be burned completely without expensive reprocessing, perhaps in a homogeneous system.

If regeneration is achievable then the fuel cost can be reduced appreciably below 3 mills/kwh. Thus if the U^{235} is burned as natural uranium selling, fabricated, at \$10/lb, the U^{235} cost is only about \$3.50/gram; and if all the U^{235} can be burned without reprocessing, the fuel cost would be about 0.7 mill/kwh. Usually fuel costs below 1.5 mills/kwh are deemed adequate and so it is usually suggested that burning of about half the original U^{235} - corresponding to irradiation of about 3000 mw days/ton of U - should be adequate to achieve economical power in low enriched power reactors.

The outlook for achieving burnups of 3000 mwd per ton is on the whole good, but not easy. The two major problems are the poisoning of the reactor by fission products and heavy isotope buildup - these losses in k must be balanced

by the gain derived from buildup of Pu^{239} - and the loss of structural integrity due to radiation damage.

The outlook for successful solution of both these problems is sufficiently good that a good segment of the nuclear technological effort is directed toward low enriched, high burnup, thermal neutron reactors. These include D_2O (Canadian NPD), graphite (British, Russian, U. S. SGR), H_2O as moderators, and gas (British), D_2O (Canadian), and H_2O as coolants.

Nevertheless, since such reactors attempt to utilize only a small fraction of the ultimate raw material, it is unlikely that they will represent the long term trend in power reactor design. Rather, some highly regenerative reactor - preferably one with conversion ratio > 1 - which uses Th and U as raw material will always be at least the aim of reactor design.

The choices available to the reactor designer at the outset are enormous. In the following table we list some of the possibilities, any combination of which can lead to a breeder reactor system.

<u>Fuel</u>	<u>Fertile Material</u>	<u>Energy</u>	<u>Moderator</u>	<u>Coolant</u>	<u>Heat Removal</u>	<u>Geometry</u>
U^{233}	Th^{232}	Fast	H_2O D_2O	H_2O D_2O	Boiling Forced Convection	Homogeneous Heterogeneous
Pu^{239}	U^{238}	Inter- mediate	Be BeO	Na CO_2	Circulating Fuel etc.	
		Slow	C etc.	etc.		

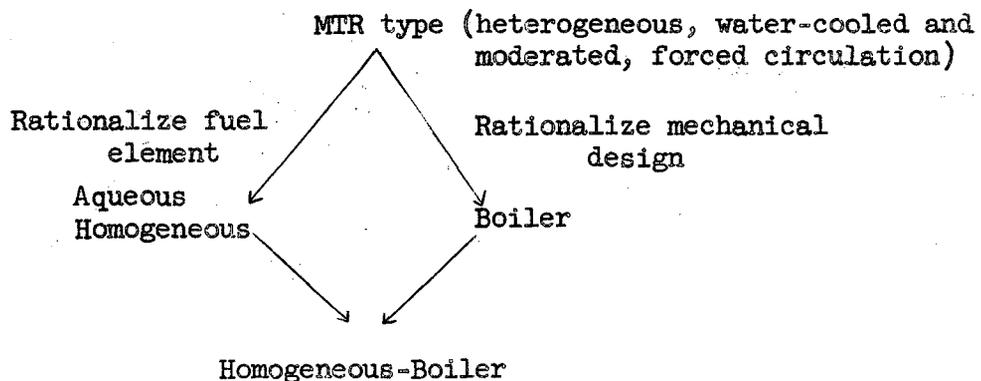
Not all the choices are free; for example, a fast reactor can have no or very little moderator. Nevertheless, there are of the order of a hundred distinct reactor types which must be considered seriously.

From the nuclear standpoint the Th and the U breeder systems are sharply different because of the different way in which α for Pu²³⁹ and for U²³³ change with U²³³, $\eta_{\text{thermal}} = 2.3$ and $\eta_{\text{fast}} > 2.3$. Hence, a Th breeder can be a fast or thermal system. With Pu²³⁹, $\eta_{\text{thermal}} \approx 2.0$, $\eta_{\text{fast}} \approx 2.9$; hence, a U-Pu breeder must be a fast reactor.

Thus the attempts to achieve a successful power breeder fall along two engineering and nuclear lines - fast, Na-cooled Pu-U systems, and thermal, D₂O-cooled U²³³-Th systems.

The fast reactor is beset with a profound dilemma whose solution is still unclear; namely, in order for the system to breed, its energy spectrum must be maintained very high, while in order to achieve high material efficiency and avoid radiation damage the reactor must be diluted with coolant or the Pu fuel must be diluted with U²³⁸. The presence of any diluting material tends to lower the average neutron spectrum, and this in turn lowers the breeding gain. Thus, the fast reactor must balance "unuseable compactness" against "unworkable diluteness".

The main line of development in the Th-U breeder system is the aqueous homogeneous. From the engineering standpoint this represents a rationalization of the water-cooled line of reactor development according to the following sequence.



As can be seen in the diagram, if one begins with the standard MTR-type pressurized water reactor, two directions of rationalization or improvement are possible. First, one can eliminate the intermediate heat exchanger and raise steam directly in the reactor. Such a device would be exemplified by the boiling water reactors. The other direction of rationalization involves subdividing the fuel element more and more until the reactor is homogeneous. This, in many respects, is a more basic kind of rationalization since the whole fuel element recycle problem is avoided.

Finally, one can merge the two rationalizations into a homogeneous boiler; such a reactor would consist simply of a boiling pot of liquid, and the intensely radioactive steam would be sent directly to the turbine.

The problems which the homogeneous reactor circumvents are primarily those concerned with the reprocessing of fuel elements; in exchange, this system must cope with the extraordinary engineering difficulty of handling high pressure radioactive liquids with absolutely no leaks, the formidable question of corrosion, and the handling of thorium slurries. The fact that there is no soluble thorium compound (aside from the nitrate, which absorbs too many neutrons) is one of the major disappointments in chain reactor technology.

Nevertheless, the incentive toward achieving a successful homogeneous breeder is very great, and much work is being devoted to this problem at ORNL, at Westinghouse - Pennsylvania Power and Light, at Foster Wheeler, and in the Netherlands.

So far one homogeneous power plant, the HRE, has been operated. This reactor produced 1000 kw in a circulating aqueous solution of UO_2SO_4 , at pressure of 1000 psi and outlet temperature of 250° C. The reactor was found to be completely stable, and since it was so strongly power demand responsive, the control rods were not used in the reactor. The HRE ran for about 1950 hours,

and was shut down because of mechanical failures which it was decided were not worth correcting before going to the next reactor.

A new homogeneous reactor, the HRT, is now under construction at ORNL. This is rated at 5 mw, and will circulate 2000 psi UO_2SO_4 in D_2O at outlet temperature of 300°C . This reactor is a two-region machine - inner core Zr, D_2O blanket, has no control rod, and it is hoped, will demonstrate the long-term reliability and feasibility of the aqueous homogeneous principle.