

## THERMAL BREEDER REACTORS

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Nuclear breeders are reactors that produce more of the essential fissile fuels  $^{233}\text{U}$ ,  $^{235}\text{U}$ ,  $^{239}\text{Pu}$ , or  $^{241}\text{Pu}$  than they consume. They do this by neutron capture in the relatively abundant source materials  $^{232}\text{Th}$  and  $^{238}\text{U}$  (or the derivative materials  $^{234}\text{U}$  and  $^{240}\text{Pu}$ ), which cannot themselves sustain a neutron chain reaction. Fast breeders operate with neutrons primarily in the energy range  $10^8$ – $10^7$  eV—that is, close to the source energies for fission neutrons,  $10^6$ – $10^7$  eV; they customarily employ the  $^{238}\text{U}$ – $^{239}\text{Pu}$  fuel cycle. A comprehensive review of fast breeders was presented by Häfele in 1970 (1).

Thermal breeders operate with neutrons primarily in the energy range below 1.0 eV, though of course some reactions do occur in the process of slowing down from fission source energies. Thermal breeders must employ the  $^{232}\text{Th}$ – $^{233}\text{U}$  cycle, for reasons that will be discussed in the section “Reactor Physics Considerations.” These breeders thus complement the fast breeders by exploiting the other nuclear source material, thorium, as well as by employing quite different nuclear engineering technologies.

Most of the world’s nuclear energy community is now devoting its effort to the development of the liquid-metal fast breeder. Yet there is a small but persistent minority that continues to keep alive the spark of thermal breeding. Despite the far fewer resources that have been devoted to thermal breeders than to fast

breeders, there have been several significant new developments to report since the last review of thermal breeders appeared in 1964 (Alexander 2).

The two aims of breeder reactor development remain the same as always: conservation of resources of uranium and thorium, and production of prime energy at roughly our present cost essentially forever. It will be our purpose to re-examine these two historical motivations from today's perspective, and to estimate how realistically one can expect them to be achieved with thermal breeders. The review will consider three separate, though related, aspects of this question: (a) the reactor physics of thermal breeders; (b) general economic considerations, including conservation of mineral resources; and (c) the engineering of thermal breeder systems.

### REACTOR PHYSICS CONSIDERATIONS

Throughout this review, we shall refer to the isotopes  $^{233}\text{U}$ ,  $^{235}\text{U}$ ,  $^{239}\text{Pu}$ , and  $^{241}\text{Pu}$  as the fissile materials, because they undergo fission at all neutron energies and at least one of them is required to establish a fission chain reaction. We refer to the isotopes  $^{232}\text{Th}$ ,  $^{234}\text{U}$ ,  $^{238}\text{U}$ , and  $^{240}\text{Pu}$  as the fertile materials; neutron capture in these leads to production of the fissile material. We shall consistently use the term "breeding ratio" to mean the rate of production of fissile material relative to its rate of consumption, whether that ratio is greater than or less than unity. Occasionally, we may also use the term "conversion ratio," defined in the same way.

In fast breeder reactors, breeding ratios are characteristically much greater than unity—e.g. 1.3–1.5. In thermal breeders, however, the margin for breeding is extremely small. Hence the physics of thermal breeder reactors is dominated by the details of the neutron economy.

A necessary condition for breeding is that  $\eta$ , the number of fission neutrons produced per neutron absorbed in the fissile isotope, must exceed two, one neutron being required to maintain the chain reaction and the other being absorbed in the fertile material. This condition applies to the average value of  $\eta$  over the spectrum of neutrons in the reactor,

$$\begin{aligned}\bar{\eta} &= \int \nu\sigma_f(E)\phi(E)dE / \int \sigma_a(E)\phi(E)dE \\ &= \int \eta(E)\sigma_a(E)\phi(E)dE / \int \sigma_a(E)\phi(E)dE\end{aligned}$$

where  $\nu$  is the neutron yield per fission,  $\sigma_f$  and  $\sigma_a$  the fission and absorption cross sections of the fuel, and  $\phi$  the neutron flux. Since  $\eta(E)$  is a function of the energy of the neutron inducing fission, the quantity  $\bar{\eta}$  depends not only on the fissile nuclide itself, but also on all factors that influence the energy distribution of neutrons in the reactor—i.e., on the temperature of the moderator, on the moderator-to-fuel atom ratio, and on the presence of other materials which may modify the neutron spectrum.

[Actually, the condition  $\bar{\eta} > 2$  may be modified if additional neutrons are pro-

duced in reactions that do not destroy fissile atoms—e.g., fast fission in  $^{238}\text{U}$  or  $^{232}\text{Th}$ , or  $(n, 2n)$  reactions in beryllium. The modified condition is  $\bar{\eta}\epsilon > 2$ , where  $\epsilon$  is the ratio of total neutrons produced to those produced by fission of the fissile atoms.]

A recent, widely accepted, set of thermal-neutron cross sections for the four principal fissile nuclides is shown in Table 1. Note that only  $^{233}\text{U}$  has a value of  $\eta$  appreciably greater than two. It should be remembered also that  $^{233}\text{U}$  and  $^{239}\text{Pu}$  are the nuclides of primary importance as breeder reactor fuels, since they are the principal products of neutron capture in  $^{232}\text{Th}$  and  $^{238}\text{U}$ , respectively.  $^{235}\text{U}$  and  $^{241}\text{Pu}$  are third-order products, each requiring two additional neutron captures, and are therefore of much less importance in the behavior of breeder reactors. Nonetheless, they are inevitably present, in relatively minor amounts, and their nuclear properties must be taken into account in an accurate assessment of breeder reactor performance. In addition,  $^{235}\text{U}$ , as the only naturally occurring fissile material, may be required to start a breeder.

Unfortunately,  $\eta(E)$  for most of these nuclides is generally lower than the 2200 m/sec values over much of the neutron energy range of importance in thermal-neutron reactors. This may be seen from the curves shown in Figure 1. The value of  $\eta(E)$  tends to be low especially in neutron resonances where the cross section is high. Thus, the average value of  $\eta$  tends to be reduced by increases in moderator temperature and by increases in the fuel-to-moderator atom ratio, both of which enhance the relative importance of near-thermal and epithermal neutrons.

The trend of  $\bar{\eta}$  with fuel/moderator ratio was discussed by Chernick and Moore for the case of  $^{233}\text{U}$  at room temperature (3). We have repeated parts of their study, using more recent cross-section data (4), and have included results for  $^{235}\text{U}$  and  $^{239}\text{Pu}$ , to illustrate the special status of  $^{233}\text{U}$  for breeding in thermal reactors.

In Figure 2, we show spectrum-averaged  $\eta$  values (5), (plotted as  $\bar{\eta}-1$ ), for a series of binary mixtures of  $^{233}\text{U}$ ,  $^{235}\text{U}$ , and  $^{239}\text{Pu}$  in a graphite moderator at room

TABLE 1. NEUTRON CROSS SECTIONS (IN BARNs) OF THE PRINCIPAL FISSILE NUCLIDES  $^{233}\text{U}$ ,  $^{235}\text{U}$ ,  $^{239}\text{Pu}$ , AND  $^{241}\text{Pu}$ <sup>a</sup>

(Neutron energy = 0.0252 eV, velocity = 2200 m/sec)

	$^{233}\text{U}$	$^{235}\text{U}$	$^{239}\text{Pu}$	$^{241}\text{Pu}$
$\sigma_a^b$	$578 \pm 2$	$678 \pm 2$	$1013 \pm 4$	$1375 \pm 9$
$\sigma_f$	$531 \pm 2$	$580 \pm 2$	$742 \pm 3$	$1007 \pm 7$
$\sigma_\gamma$	$47 \pm 1$	$98 \pm 1$	$271 \pm 3$	$368 \pm 8$
$\alpha$	$0.089 \pm 0.002$	$0.169 \pm 0.002$	$0.366 \pm 0.004$	$0.365 \pm 0.009$
$\eta$	$2.284 \pm 0.006$	$2.072 \pm 0.006$	$2.109 \pm 0.007$	$2.149 \pm 0.014$
$\nu$	$2.487 \pm 0.007$	$2.423 \pm 0.007$	$2.880 \pm 0.009$	$2.934 \pm 0.012$

<sup>a</sup> Hanna, G. C. et al 1969. *At. Energ. Rev.* 7:3-92. Figures in the referenced article were all given to one additional significant figure.

<sup>b</sup>  $\sigma_a = \sigma_f + \sigma_\gamma$ ;  $\alpha = \sigma_\gamma / \sigma_f$ ;  $\nu$  = neutrons per fission =  $\eta(1 + \alpha)$ .

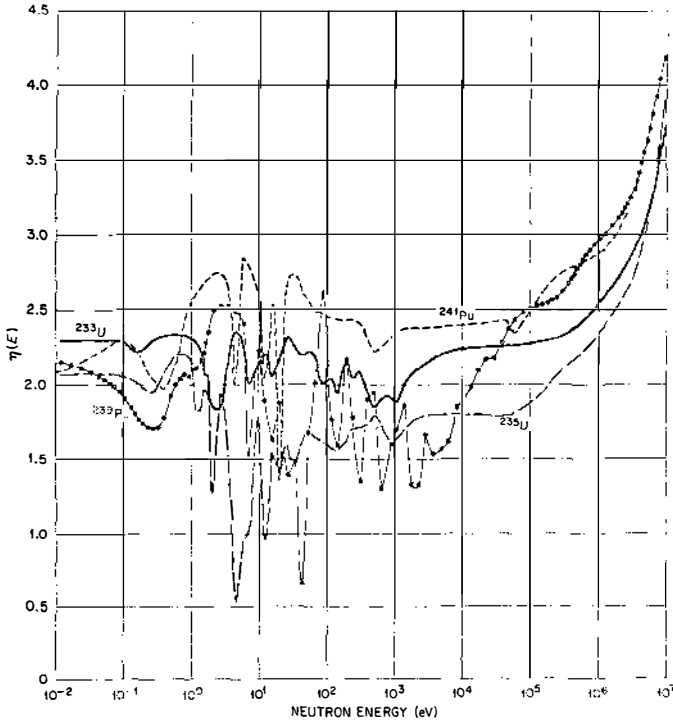


FIGURE 1. Energy dependence of eta for the principal fissile nuclides (4).

temperature, at 573°K, and at 900°K. No other materials are present, nor is the spectrum modified by neutron leakage. Thus, the overall spectra are not precisely those of a series of critical reactors. However, similar calculations for reactor core compositions having thorium, structural materials, and leakage show very similar variations of  $\bar{\eta}$  with fuel concentration. Figure 2 may thus be taken as a good indication of the possibilities for breeding in thermal reactors, and it may be seen that only  $^{233}\text{U}$  has values of  $\bar{\eta}$  appreciably larger than 2.0. (Also shown in Figure 2 is the "thermality" or fraction of all neutron absorptions in fuel that occur at neutron energies below 0.45 eV.)

Of course, not all of the  $\bar{\eta}\epsilon - 1$  neutrons theoretically available for breeding new fuel can actually be used for that purpose. Some neutrons will be absorbed in the moderator itself and others will be lost by capture in structures and coolant, in control rods, or in fission-product poisons; still others may be lost by diffusion to the surroundings. The breeding ratio actually achieved is thus given by

$$BR = \bar{\eta}\epsilon - 1 - L$$

where  $L$  represents all the "losses" per neutron absorbed in fissile fuel. In Figure

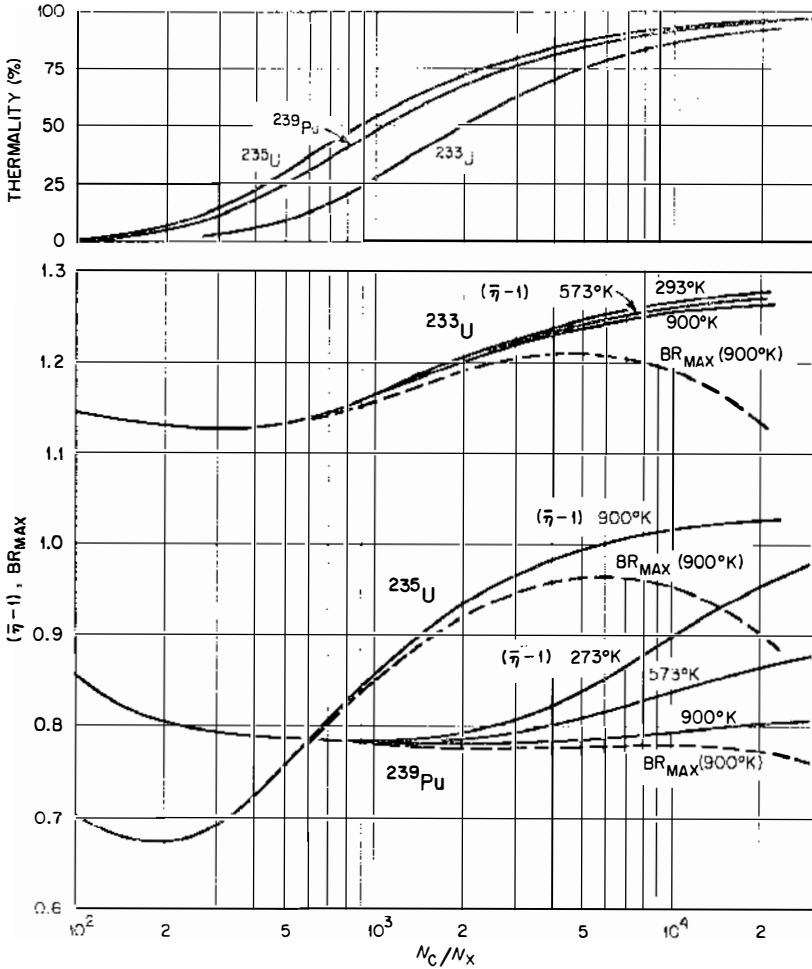


FIGURE 2. Spectrum-averaged  $\eta$ , maximum theoretical breeding ratio, and "thermality" (fraction of absorptions below 0.45 eV) as functions of moderator-to-fuel atom ratio. Carbon moderator. Temperatures as indicated (5).

2, we have also plotted the maximum attainable breeding ratio, with neutron absorptions in the moderator taken into account, but with no other losses postulated.

The attainable breeding ratio, in a thermal reactor, depends somewhat on the choice of moderator. The principal moderators to be considered are water, heavy water, beryllium, beryllium oxide, and graphite. The maximum breeding ratio for  $^{233}\text{U}$  in each of these moderators, allowing only for losses in the moderator itself, is shown in Figure 3 as a function of the slowing-down power (Glasstone 6,

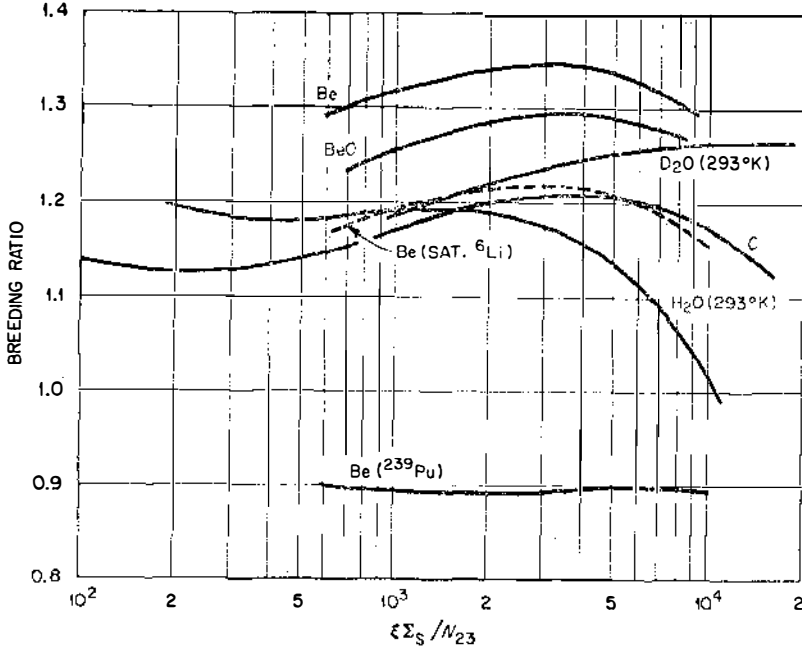


FIGURE 3. Maximum theoretical breeding ratio for various moderators, as a function of moderator-to-fuel atom ratio. Fuel is  $^{238}\text{U}$  and temperature is  $900^\circ\text{K}$  unless otherwise indicated (5).

Weinberg & Wigner 7),  $\xi \sigma_s$ , per fuel atom. ( $\sigma_s$  is the free-atom scattering cross section of the moderator, and  $\xi$  is the mean logarithmic energy loss of neutrons in collision with moderator atoms.)

The curves generally exhibit a maximum, resulting from the opposing effects of rising  $\bar{\eta}$  and increasing moderator loss as moderator-to-fuel ratio increases. Losses in  $\text{D}_2\text{O}$  are very small, even with an allowance (which is included in the curve) for 0.14 percent  $\text{H}_2\text{O}$  in the  $\text{D}_2\text{O}$ . The maximum breeding ratio in  $\text{H}_2\text{O}$  is only 0.02 less than in carbon; however, as with  $\text{D}_2\text{O}$ , losses in structure may be important. Beryllium would appear to be especially suitable as a moderator for thermal breeder reactors; its large ( $n, 2n$ ) cross section is only partly offset by a low-threshold ( $n, \alpha$ ) reaction, yielding a net fast-effect factor of about 1.07 (for Be)<sup>1</sup> or 1.04 for  $\text{BeO}$ .<sup>1</sup> Unfortunately, the Be ( $n, \alpha$ ) reaction produces  $^6\text{Li}$ , which has a neutron absorption cross section of 940 barns at 0.025 eV, and therefore reaches saturation rather quickly—more quickly than the fuel burns up. In Figure 3 we therefore indicate the reduced breeding ratio that would follow saturation of the  $^6\text{Li}$ . (Higher-order gaseous products,  $^3\text{H}$  and  $^3\text{He}$ , which would result from

<sup>1</sup> Based on ENDF/B-Version III cross sections for Be (8). Version II for other nuclides.

neutron capture in  ${}^6\text{Li}$ , are presumed to be unimportant, as poisons, because of the long life and mobility of the  ${}^3\text{H}$ .)

Fast neutron multiplication can also result from fissions or  $(n, 2n)$  reactions in  ${}^{238}\text{U}$ ,  ${}^{232}\text{Th}$ , or other even-even nuclides such as  ${}^{234}\text{U}$ ,  ${}^{236}\text{U}$ , or  ${}^{240}\text{Pu}$ . In contrast to the situation in fast-breeder reactors, however, these reactions make only minor contributions to the overall neutron production in thermal breeders. Fast fission in  ${}^{232}\text{Th}$  is much less important than in  ${}^{238}\text{U}$  because the cross section above threshold is much lower for  ${}^{232}\text{Th}$  than for  ${}^{238}\text{U}$ .

Control of the neutron loss due to leakage is largely a matter of economics. Leakage can be reduced by surrounding the active core by a blanket region containing mainly the fertile material—e.g.,  ${}^{232}\text{Th}$ —the extent of the reduction depending in part on the thickness of the blanket. Increasing the blanket thickness, we reach a point beyond which a further increase would cost more than the value of the additional neutrons saved. Indeed, it may be found that no blanket is economically justifiable. In any event, as a general rule, leakage losses in a reactor designed for minimum power cost are not likely to be less than 0.01 to 0.02 (relative to  $\eta$  source neutrons).

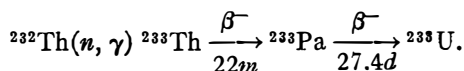
Neutron losses to the high-cross-section fission product  ${}^{135}\text{Xe}$  are well known (Glasstone & Edlund 9). The xenon poison fraction—i.e., neutron absorptions in xenon per absorption in fuel—may be related to the fuel specific power,  $S$  [MW(t)/kg fissile], which is a useful generalized measure of the neutron flux level in a reactor. Using the xenon yield for thermal-neutron fission of  ${}^{235}\text{U}$  (0.060) and cross sections appropriate to a graphite core at  $600^\circ\text{C}$  with  $N_c/N_{235} \sim 9000$ , we find for the xenon poison fraction

$$P \simeq 0.054S(0.44 + S)^{-1}$$

For typical in-core inventories of fissile fuel, values of  $S$  of 1 to 3 MW(t)/kg will normally be attained, corresponding to values of  $P$  of 0.037 to 0.047. Thus, a reduction of about 0.04 in breeding ratio will usually be associated with equilibrium concentrations of  ${}^{135}\text{Xe}$ .

Following a reactor shutdown or reduction in power, the xenon poisoning temporarily increases (9), passing through a maximum 10 to 12 hours after the shutdown. The magnitude of this transient additional poison fraction also depends on the fuel specific power, and is approximately 0.01, 0.04, or 0.07 for  $S=1, 2$ , or 3 MW(t)/kg, respectively. Although the temporary loss is not significant by itself, a reactivity reserve for xenon override, if normally compensated by control rods, would represent a permanent loss of neutrons. This must of course be avoided in a thermal breeder.

A potentially significant neutron loss in thermal breeders is that due to capture in  ${}^{233}\text{Pa}$ , which is an intermediate in the breeding reaction



${}^{233}\text{Pa}$  has a thermal-neutron cross section of about 43 b and a resonance integral

of about 850 b. The loss of neutrons by absorption in  $^{233}\text{Pa}$  is similar to the  $^{135}\text{Xe}$  loss, in that it involves a competition between neutron capture and radioactive decay, and is roughly proportional to fuel specific power for  $\sigma(\text{Pa})\phi/\lambda \ll 1$ . However, since absorption of a neutron by  $^{233}\text{Pa}$  destroys a nascent  $^{233}\text{U}$  atom, as well as removing a neutron that might have created yet another  $^{233}\text{U}$  atom, the loss to  $^{233}\text{Pa}$  is double the simple ratio of absorptions in  $^{233}\text{Pa}$  to absorptions in  $^{233}\text{U}$ .

We relate the  $^{233}\text{Pa}$  loss to specific power in a way similar to that used for xenon. While the ratio of spectrum-averaged cross sections,  $\bar{\sigma}(^{233}\text{Pa})/\bar{\sigma}(^{233}\text{U})$ , does depend on the reactor spectrum, a value of one-third may be taken as typical. Assuming that the breeding ratio is close to unity, and noting that the decay constant of  $^{233}\text{Pa}$  is 0.0257/day, we find that the loss in breeding ratio is given approximately by

$$\delta BR \sim 2S(64 + S)^{-1}$$

with values of 0.03, 0.06, and 0.09 for  $S=1, 2,$  and  $3 \text{ MW(t)/kg}$ , respectively.

This loss may be reduced by partial segregation of the thorium and fissile uranium, so that the thorium, and hence the protactinium, experiences a low neutron flux, while the fissile uranium is exposed to a higher flux. The factors involving the specific power in the above expression would then be multiplied by the ratio of effective flux in the thorium to that in the fuel.

An interesting consequence of the relatively long mean life of  $^{233}\text{Pa}$  (39 days) is that a significant reactivity addition can occur during a prolonged reactor shutdown. During normal, steady-state reactor operation, the ratio of  $^{233}\text{Pa}$  inventory to fissile uranium inventory is approximately  $S/20$ , where  $S$  is, again, the in-core fuel specific power in  $\text{MW(t)/kg}$  (fissile). Thus, for  $S$  in the range 1 to 3  $\text{MW(t)/kg}$ , an increase of 5 to 15 percent in fuel inventory would occur, with a time constant for approach to saturation of 39 days. While the reactivity effect of this additional  $^{233}\text{U}$  would depend on its location—i.e., on the initial degree of segregation of the fissile and fertile materials in the reactor—the effect could be as much as 40 percent of the fractional increase in fuel inventory; thus a reactivity increase  $\delta k/k \sim 0.02$  to 0.06 could occur. Such a reactivity increase need not be a problem, but appropriate control devices would be required to compensate for it. After the reactor is brought back to power, some loss of neutrons to control poisons might be involved, while equilibrium concentrations of fuel and protactinium are reestablished. This loss would not occur in the light-water breeder reactor, where movement of the seed is used to control reactivity (see the section “Engineering of Thermal Breeder Systems”), nor in the molten-salt breeder reactor, with essentially continuous control over the distribution of fuel inventory between the reactor and the chemical processing plant. Unfortunately, the reactivity increase associated with  $^{233}\text{Pa}$  decay cannot provide xenon override capability, since the time constants for the two processes are very different—i.e., 39 days vs 10 hours.

One of the most important sources of neutron loss, from the standpoint of breeding in a thermal reactor, is the loss to slowly saturating or nonsaturating fission products. In contrast to  $^{135}\text{Xe}$  and  $^{149}\text{Sm}$ , whose very large neutron-absorp-



tion cross sections cause them to reach saturation very quickly, the great majority of the fission products have cross sections which are comparable to or smaller than that of the fuel itself. Thus, the aggregate poisoning effect of these fission products is roughly proportional to the fractional burnup of the fuel prior to its removal from the reactor for chemical processing. The fission product poisoning depends also on the neutron spectrum, on the predominant species of fuel in the reactor, on the fuel-replacement strategy employed, and on the flux level, or fuel specific power (England 10). It is hardly possible, therefore, to exhibit a single universal relationship between fuel burnup and fission-product poisoning. Nonetheless we show a few typical points in Figure 4 in which the fractional fuel burnup is expressed in terms of *fifa*—i.e., fissions per initial fissile atom in fresh fuel. (Note that, with fuel regeneration by breeding, exposures greater than one *fifa* are possible.) It may be inferred from Figure 4 (with due allowance for the effects of other variables) that neutron losses in the neighborhood of 0.10 (per neutron absorbed in fissile atoms) may be expected for fuel exposures of 1 to 1.5 *fifa*.

Another rather important factor that tends to reduce the already small margin for breeding in a thermal reactor is the presence of higher isotopes of uranium, resulting from successive neutron captures in the chain starting with <sup>233</sup>U. The reduction results in part from the weighted contribution of the lower  $\eta$  of <sup>235</sup>U and in part from the added neutron loss in <sup>236</sup>U and <sup>237</sup>Np. Since  $\bar{\eta}_{25}$  is less than  $\bar{\eta}_{23}$

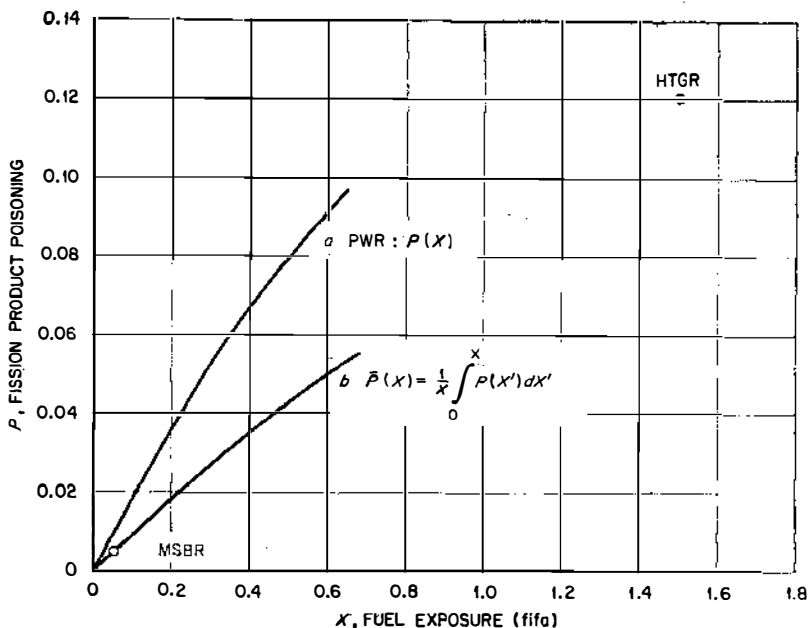


FIGURE 4. Fission product poisoning, excluding <sup>135</sup>Xe and <sup>149</sup>Sm from 149 chain fission yield. (Neutron absorptions in fission products per absorption in fissile fuel.)

by at least 0.2 (see Figure 2 for example), and since  $\sim 10\%$  of the neutron absorptions in fuel are in  $^{235}\text{U}$  (more, if the breeding ratio is less than unity), then the reduction in  $\bar{\eta}$  is at least 0.02. Similarly, taking into account the capture-to-fission cross-section ratios for both  $^{233}\text{U}$  and  $^{235}\text{U}$ , the losses in  $^{236}\text{U}$  and  $^{237}\text{Np}$  together may equal an additional 0.03 or more. This loss may be reduced if  $^{237}\text{Np}$  is removed from the fuel during chemical processing, particularly if the fuel is processed at low fractional burnup. The entire effect of the higher isotopes may be further reduced if the breeding ratio is appreciably greater than unity, since these nuclides may then be removed from the system, along with the fissile isotopes removed for sale.

It should also be noted that the buildup of  $^{236}\text{U}$ , and of any nuclides beyond it in the chain, may be rather slow, owing to a rather small cross section of  $^{236}\text{U}$  relative to that of  $^{233}\text{U}$ . For a fuel specific power of 1 MW(t)/kg, the time constant for the approach of the  $^{236}\text{U}$  concentration to equilibrium is something like 50 years at 0.8 plant factor (40 equivalent full-power years). Of course a higher specific power would produce a shorter time constant. (For this calculation, the specific power must be based on the entire inventory of fissile uranium chargeable to the reactor—i.e., including the out-of-pile as well as in-pile inventories.) On the other hand, it should also be noted that if a breeder reactor system is started up initially with  $^{235}\text{U}$ , owing to a lack of  $^{233}\text{U}$  for startup, then an amount of  $^{236}\text{U}$  much greater than the equilibrium amount would be produced early in the life of the system, and the equilibrium concentration would be approached from the high side. This extra poisoning effect must be experienced somewhere in the nuclear power complex, whether or not the extra  $^{236}\text{U}$  is retained in the breeder reactor.

Quite apart from the cost factors involved, rapid chemical processing may prove to be undesirable if the recovery of fissile material from exposed fuel elements is incomplete. If a small fraction of the fuel is lost during each fuel processing cycle, an effective reduction in breeding ratio is experienced which is inversely proportional to the discharge fuel exposure, expressed in fissions per initial fissile atom. For example, at an exposure of one fifa (neglecting a small correction due to radiative capture), a processing loss of 0.5 percent would give rise to a 0.005 reduction in breeding ratio, while at 0.1 fifa the same processing loss would lower the effective breeding ratio by 0.05. In Figure 5, we see how the combined loss of breeding ratio due to fission products and processing losses might vary with fuel discharge exposure, for a postulated linear loss due to the fission product aggregate (excluding  $^{135}\text{Xe}$  and  $^{149}\text{Sm}$ ).

For solid fuel elements, material losses in processing and refabrication are customarily supposed to be in the neighborhood of 1.0 percent (11). Figure 5 would suggest an optimum exposure of about 0.3 fifa for this rate of loss; but in fact, because of fabrication and processing costs, the economic optimum exposure would typically be much greater than this. For a fluid-fuel reactor, with an integrated chemical processing plant, very much smaller loss rates are believed to be attainable (see the section "Engineering of Thermal Breeder Systems").

The combined effect on breeder performance of the several losses discussed

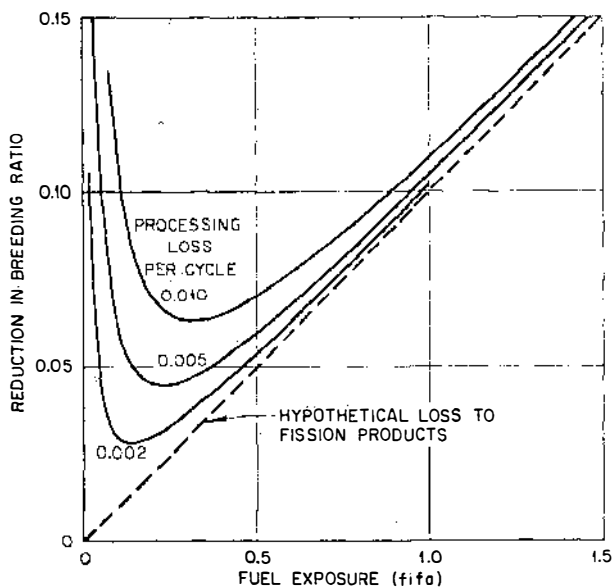


FIGURE 5. Combined reduction in breeding ratio due to fission products and chemical processing losses.

above is illustrated in Table 2, in which we compare neutron balances for a fast breeder, a thermal breeder, and some thermal converters. These “neutron budgets” show the number of neutrons absorbed in each nuclear species or reactor component relative to a single neutron absorbed in all fissile nuclides. Other neutron balances will be displayed in the section “Engineering of Thermal Breeder Systems.”

### GENERAL ECONOMIC CONSIDERATIONS

The two great goals of nuclear breeding are efficient utilization of uranium and thorium, and low-cost energy. Unfortunately these two goals of nuclear breeders often do not go together. In this section, we shall review briefly the measures of breeding performance that are related directly to resource utilization and will touch on the impact of cost considerations on the performance of thermal breeder reactors.

### UTILIZATION OF RESOURCES

The importance of fuel utilization, is usually illustrated by noting that breeder reactors, when able to satisfy all requirements for fissile material, burn a very large fraction of the mined material—i.e., 50% or more (depending primarily on processing losses). On the other hand, light-water reactors similar to present commercial reactors may burn approximately 1 to 2%, depending on whether the plutonium produced is recycled or not. Current projections (12, Whitman 13) for

TABLE 2. TYPICAL NEUTRON BUDGETS FOR NUCLEAR REACTORS

	Thermal converter <sup>a</sup>	Thermal breeder <sup>b</sup>	Fast breeder <sup>c</sup>
Fissile nuclides	1.000	1.000	1.000
Fertile nuclides	0.847	1.079 <sup>d</sup>	1.636 <sup>e</sup>
Protactinium	0.019	0.005	—
Other heavy nuclides	0.012	0.014	0.016
Structure, coolant, control	0.082	0.097	0.191
Xenon	0.042	0.005	—
Other fission products	0.136	0.010	0.073 <sup>f</sup>
Leakage	0.040	0.022	0.032
<b>Total = <math>\eta\epsilon</math></b>	<b>2.178</b>	<b>2.232</b>	<b>2.948</b>
Conversion ratio <sup>g</sup>	0.83	1.07	1.42
$\bar{\eta}(^{238}\text{U})$	2.23	2.23	—
$\bar{\eta}(^{235}\text{U})$	1.98	1.98	—
$\bar{\eta}(\text{fuel})$	2.17	2.21	2.33
$\epsilon$	1.003	1.008 <sup>h</sup>	1.27

<sup>a</sup> HTGR; An Evaluation of High-Temperature Gas-Cooled Reactors, 1969. *USAEC Doc. WASH-1085*.

<sup>b</sup> MSBR; MSR Program Semiann. Progr. Rep., February 28, 1971. *USAEC Doc. ORNL-4676*, p. 42.

<sup>c</sup> LMFBR; General Electric Company, 1968. *USAEC Doc. GEAP-5678*, p. 91.

<sup>d</sup> Includes 0.003 fissions in  $^{232}\text{Th}$ .

<sup>e</sup> Includes 0.213 fissions in  $^{238}\text{U}$  and  $^{240}\text{Pu}$ .

<sup>f</sup> All fission products.

<sup>g</sup> Conversion ratio = captures in fertile material less absorptions in  $^{233}\text{Pa}$ .

<sup>h</sup> About equal contributions from  $\text{Be}(n, 2n)$  and from fast fission.

electrical energy generation in the United States indicate a generation rate of  $10^{13}$  kWhr(e)/yr by the year 2000. At that rate, if the energy were supplied by nuclear plants, the ore requirements just to cover consumption would be about 180,000 tons of  $\text{U}_3\text{O}_8$  each year, at 1% utilization, 33% electrical efficiency, but only 2700 tons per year at 50% utilization, 44% electrical efficiency. These figures are to be compared with United States reserves of uranium, which are estimated (12, Faulkner 14) to be around one million tons (of  $\text{U}_3\text{O}_8$ ) recoverable at approximately current costs (less than \$10/lb  $\text{U}_3\text{O}_8$ ), around 10 million tons recoverable at less than \$50/lb, and perhaps 25 million tons recoverable at less than \$100/lb (all expressed in terms of 1970 price levels). It may be noted that increasing ore costs from \$10 to \$50/lb  $\text{U}_3\text{O}_8$  would add approximately 3 mills/kWhr(e) to the cost of energy from contemporary light water reactors, and about half as much from natural-uranium heavy water reactors or high-temperature gas-cooled reactors. Thus, even without allowing for further growth in energy demand beyond the end of this century, it is very likely that utilization efficiencies much greater than 1% must be achieved if nuclear energy is to satisfy any appreciable

fraction of our long-term energy demands without a serious increase in the cost of energy.

In a nuclear-power economy having constant average-power output, it would be sufficient for the average breeding ratio of the system to equal unity. In this circumstance, the "nuclear catalysts" (the fissile materials  $^{239}\text{Pu}$  or  $^{233}\text{U}$ ) would be regenerated at the same rate they were consumed, and the only fuel materials required from outside the system would be natural uranium or thorium. These primary source materials would thus be completely burned, except for recycle losses.

In the case of a growing nuclear-power economy, however, each new plant added to the system requires a fresh inventory of the fissile "nuclear catalysts," as well as of the fertile uranium or thorium. Thus, ideally the breeder reactors should not only regenerate fissile material to compensate for their own consumption, but should produce additional material at a rate sufficient to provide inventory for the new reactors. Fertile material must of course be furnished from outside the system—e.g. mined.

The goal in a growing system, therefore, is for the annual fractional increase of fissile inventory (called the yield) to equal or exceed the annual rate of growth of energy generation by the nuclear system. The yield is proportional to the product of the breeding gain (the amount by which the breeding ratio exceeds unity) and the fuel specific power. As possible examples, a fast breeder reactor with a breeding gain of 0.4 and a specific power of 0.75 MW(t)/kg would have an annual fuel yield of about 10% per year (assuming a plant utilization factor of 0.8). A thermal breeder reactor, with a breeding gain of 0.07 and a specific power of 1.5 MW(t)/kg, would have a fuel yield of about 3.7% per year.

If the average annual fuel yield of the system is less than the system growth rate, then additional supplies of fissile material will have to be furnished from outside the system. The only naturally occurring fissile material is  $^{235}\text{U}$ , which constitutes 0.71% of natural uranium. Thus the thorium fuel cycle as well as uranium-based fuel cycles must rely on natural uranium to offset any shortages in the fissile-fuel supply from breeding. When processed through a gaseous diffusion (isotope separation) plant, only about 0.5% of the uranium is recovered as enriched  $^{235}\text{U}$ , the remaining 99.5% being discarded as diffusion plant tails. In a system relying on the  $^{232}\text{Th}$ – $^{233}\text{U}$  fuel cycle, these tails cannot be used as fertile material in the system and can therefore not be burned. Thus, if any appreciable fraction of the system inventory of fissile material must be furnished as enriched  $^{235}\text{U}$ , the fractional utilization of mined resource material would be seriously reduced because of the waste depleted uranium. For this reason, thorium-cycle reactors, including low-gain breeders, though burning primarily thorium, may require far more uranium than thorium to be mined.

These considerations are displayed in Figure 6, in which the average fraction of mined uranium and thorium actually converted into energy (called the "resource utilization efficiency") is shown as a function of the average conversion ratio of the system, for various combinations of specific inventory and system growth rate. [Specific inventory = (electrical efficiency  $\times$  specific power) $^{-1}$  =  $I$  kg (fissile)/MW(e).] The marked dependence of utilization efficiency on breeding

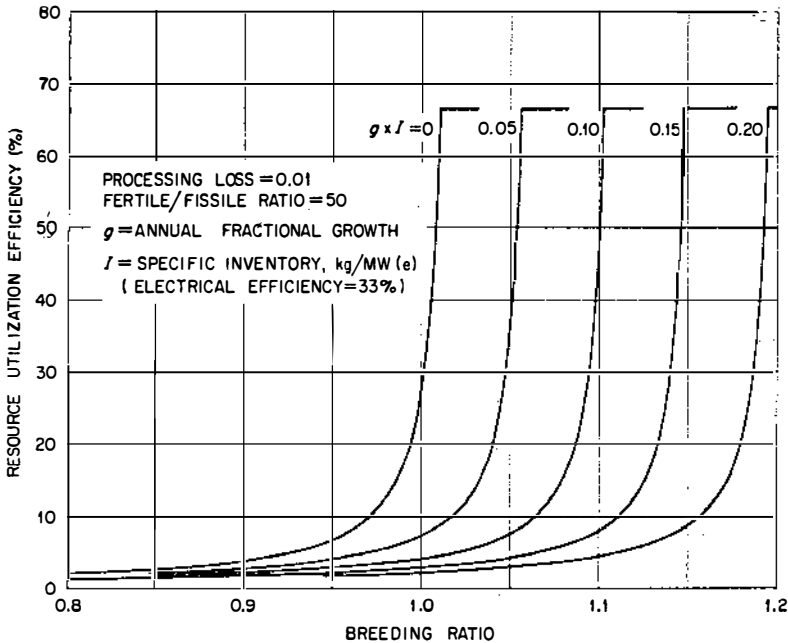


FIGURE 6. Resource utilization efficiency as a function of breeding ratio, annual growth rate of energy generation, and specific fuel inventory.

ratio, on system growth rate, and on specific fuel inventory is attributable primarily to the large mass of wasted diffusion plant tails associated with a small input of enriched uranium.

It is of course an oversimplification to think in terms of a constant fractional system growth rate. For a given type of breeder reactor to assume a major share of the electrical system load, it must initially undergo a very rapid rate of growth. But it is surely to be expected that eventually, possibly within the next hundred years, the rate of expansion of energy consumption must greatly decrease or fall essentially to zero. Current projections (9, 10) of nuclear-electric capacity by the Atomic Energy Commission and the Federal Power Commission suggest growth rates of the US nuclear-electric industry of around 12% per year by 1980, diminishing to perhaps 6% per year by the year 2000, and perhaps 2½ to 3% per year by 2030. These are, of course, only estimates, but they do suggest that it will be many years before the growth rate of the nuclear-electric system will fall below 2 to 3% per year.

It has been recognized for a number of years<sup>2</sup> that neither breeding gain nor annual fuel yield (or the doubling time, which is commonly calculated as  $\ln 2 \div \text{yield}$ ) adequately describes the performance of a breeder reactor in a growing

<sup>2</sup> An extensive discussion of this point is given by J. R. Dietrich (15, 16).

power system; specific fuel inventory plays a role beyond its effect on yield. It is in fact easy to show that for a linear system growth the cumulative requirement for mined uranium, up to the point where the system becomes self-sustaining with respect to fissile material, is proportional to the product of the doubling time and the specific inventory—i.e.,

$$Q \sim I^2/G$$

A recent AEC study (12) of potential nuclear power growth patterns showed that timely introduction of liquid-metal fast breeder reactors could limit cumulative requirements for uranium to about 1 to  $1\frac{1}{4}$  million tons of  $U_3O_8$ , while allowing nuclear-electric energy to assume nearly all of the projected electric energy generation. After about the year 2000, the breeder reactors would themselves fill all needs for fissile material, and uranium would thereafter be burned with the high efficiency characteristic of the self-sustaining breeder.

Briggs, Kasten & Rosenthal (17) have estimated cumulative ore requirements for several reactor types in a nuclear-electric system postulated to reach 140,000 MW(e) in 1980 and 930,000 MW(e) in 2000, and to expand at a rate of 100,000 MW(e) per year thereafter. Their estimates are shown in Figure 7. An underlying premise of their calculation was that only light-water reactors would be built initially, breeders would be introduced in 1982, and only breeders would be built after 1998. While these results are strictly applicable only to the assumed conditions, they illustrate the importance of specific fuel inventory in a growing nuclear-power economy.

Briggs, Kasten, & Rosenthal's curves show why, despite their long doubling time compared to that of fast breeders, thermal breeders may require no more resources than do fast breeders. Let us compare a fast breeder with doubling time 10 years and specific inventory of 4 kg/MW(e) with a thermal breeder whose doubling time is 20 years and specific inventory 2 kg/MW(e). According to Figure 7, the total resource required by the fast breeder core is nearly as large as that required by the thermal breeder— $2.0 \times 10^6$  tons compared to  $2.2 \times 10^6$  tons (including in each case the ore committed to the early converter reactors). This explains why thermal breeders of low specific inventory have always been tantalizingly attractive to reactor designers: their lower specific inventory compensates for their poor breeding gain. The engineering problem of course is to achieve specific inventories as low as, say, 1 to 2 kg/MW(e).

#### ECONOMIC TRADE-OFFS BETWEEN BURNUP AND BREEDING PERFORMANCE

It was noted earlier that neutron losses to fission product poisons are of great importance when compared with the small neutron excess available for breeding in a thermal reactor. Since these losses (excluding the high-cross-section, rapidly saturating poisons) can be reduced by frequent chemical reprocessing of the fuel, there is a great incentive for short fuel exposures in thermal breeder reactors. However, the cost of fabricating and reprocessing fuel elements, per unit of energy generated, rises sharply with decreasing fuel exposure, providing a strong

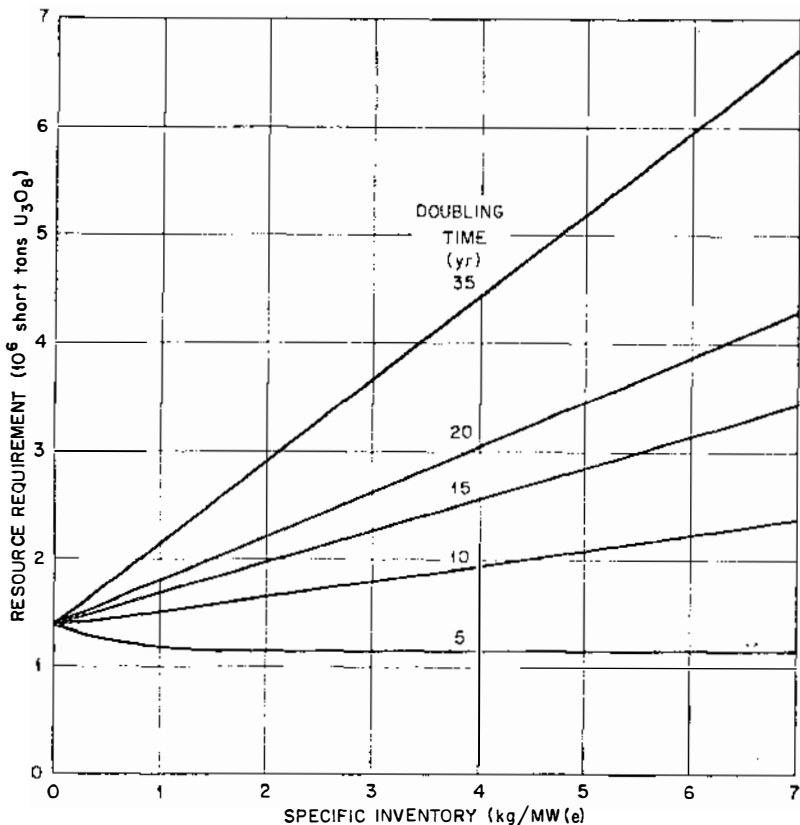


FIGURE 7. Maximum uranium ore requirements; power growth as indicated in text.

economic incentive for long exposures. Although fabrication and processing costs, fissile fuel loadings, the value of the fuel, and other factors which influence the economic optimization may vary considerably from one reactor type to another, it is nonetheless possible to make some useful generalizations.

Combined costs for fabrication, processing, and shipping of reactor fuel elements typically lie in the range \$100 to \$200/kg of total uranium and thorium (11, Ref. a of Table 2; 18); the lower figure is representative of light-water reactor (*LWR*) fuels and the higher figure more nearly representative of high temperature gas-cooled reactor (*HTGR*) fuel. At exposures of 30 MWd/kg for *LWR*s, or 60 MWd/kg for *HTGR*s, these costs contribute approximately 0.4 to 0.3 mill/kWhr(e) to the power cost (the difference, in this example, arising from the difference in electrical efficiency of the two reactor types—i.e., about 33% for *LWR*s and about 43% for *HTGR*s). With equivalent fuel enrichments of roughly 2½% and 5% for the *LWR*s and *HTGR*s respectively, these ex-



posures correspond to  $\sim 1.2$  to  $1.3$  fissions per initial fissile atom. (Either higher or lower figures may be found in specific actual cases.) To reduce these exposures by a factor of three to four would add roughly one mill/kWhr(e) to the energy cost. Yet, for many advanced converter reactor systems which are optimized for minimum energy cost, that is the sort of reduction in fuel exposure that would be required to approach or achieve breeding.

This is the classic dilemma faced by the thermal breeders: the cheapest fuel cycle is normally one in which the fission product poisons are allowed to accumulate to a point where the reactor no longer breeds. It was this basic concern which led to the original conception of the fluid-fuel breeder. The hope was that fission products could be removed so cheaply that the economic optimum might be close to the breeding optimum. How well this hope has been realized will be discussed later.

## ENGINEERING OF THERMAL BREEDER SYSTEMS

### GENERAL CONSIDERATIONS

Engineering of thermal breeder systems has always reflected the central importance of (a) suppressing neutron losses in  $^{233}\text{Pa}$  while maintaining a reasonably high fuel specific power; and (b) reducing neutron losses to fission products by removing them from the reactor as quickly as possible after they are produced.

The first of these objectives is achieved by somehow providing for the  $^{233}\text{Pa}$  to spend its life in a lower flux than the average for the fissile  $^{233}\text{U}$ . This can be done in solid-fueled reactors by segregating at least part of the fertile thorium from the  $^{233}\text{U}$  in separate elements, advantageously arranged. In the extreme case, all the fuel would be confined to a core region surrounded by a fertile blanket.

The second objective, quick removal of fission products, especially  $^{135}\text{Xe}$ , poses serious problems in solid-fueled reactors. In almost every instance, to achieve breeding, one must reprocess the fuel very quickly, and this imposes severe economic penalties. The economic optimum always turns out to be at long burnup and low chemical recycle rate: in short, every solid-fueled thermal breeder would operate more economically as a nonbreeder than as a breeder. Very broadly speaking, it would require something approaching an order-of-magnitude increase in uranium ore cost to move the optimum breeding ratio above unity.

It was this dilemma that led designers early to the idea of fluid-fueled thermal breeders. Here, in principle, one could sequester the  $^{233}\text{Pa}$  outside the neutron flux until it decayed to  $^{233}\text{U}$ , as well as remove the fission products very quickly. The breeder of course then becomes an integrated system, consisting of reactor and chemical plant. Four major breeder systems based on these principles have been investigated: (a) the  $\text{D}_2\text{O}$  solution system in which uranium is carried as a salt dissolved in  $\text{D}_2\text{O}$  (2, 19, Lane, MacPherson & Maslan 20); (b) the  $\text{D}_2\text{O}$ -slurry system in which both uranium and thorium are carried as oxide slurries (2, 19, Hermans et al 21, Went & Hermans 22); (c) the liquid bismuth-fueled reactor in which metallic uranium is carried as a solution in liquid bismuth (2, 19, 20, Thomas 23); (d) the molten-salt system in which uranium and thorium are

carried as fused halide salts in a mixture of LiF and BeF<sub>2</sub> (2, 19, 20, Rosenthal 24, Rosenthal et al 25). Of these four systems, only the aqueous slurry and the molten-salt have survived, and they alone will be described in the following.

Though in principle the liquid-fueled systems represent the most rational approach to dealing with the two originally stated objectives, they obviously face major technological problems. It is on this account that there continue to be serious attempts to adapt existing solid-fueled reactor technology to the demands of thermal breeding. Here the line between the breeder and nonbreeder is blurred: in principle, any heterogeneous reactor can be fueled with a mixture of <sup>233</sup>U and <sup>232</sup>Th, and it will then operate for at least a time as a <sup>233</sup>U converter. One then hopes by clever design (for example, by exploitation of fast neutron reactions, by partial segregation of fissile and fertile materials, or by advances in fuel element technology) to push the breeding ratio above the magical value of unity. This is the philosophy which underlies the light-water breeder, which is based on pressurized-water reactor (*PWR*) technology; and, in a less explicit sense, the breeder version of the HTGR and of the heavy-water-moderated reactors. These approaches are under active development, especially the LWBR, and will be described in some detail.

#### FLUID-FUEL THERMAL BREEDERS

We confine our review to the two systems, aqueous-slurry and molten-salt, that are still under active development. Reviews of the aqueous solution and of the liquid bismuth reactors are to be found in the literature (2, 19, 20, 23).

*Aqueous-slurry systems.*—The D<sub>2</sub>O-slurry system, in which urania and thoria are suspended in heavy water, was originally examined at the Metallurgical Laboratory during the early 1940s. However it was dropped, in the United States, in favor of the aqueous-solution reactor. Though many aqueous-solution “water boilers” have operated at low power and low temperature very successfully for years (Bunker 26, 27, Nozawa 28, Reiffel 29), solutions of uranyl salts exhibit potentially troublesome instabilities when operated at temperatures (~300°C) required for generating electricity (19). It was basically on this account that development of aqueous-solution reactors was dropped in the United States in 1961.

Many of the corrosion and fuel stability problems of aqueous-solution reactors can be avoided if a chemically inert slurry fuel is substituted for the salt solution. Moreover, the whole system acquires an admirable simplicity if both uranium and thorium are mixed: there is then but a single fluid to handle. To be sure, the single-fluid system has lower specific power or lower breeding ratio than does a system in which uranium and thorium are segregated into separate streams. Nevertheless, the aforementioned gain in engineering simplicity is so attractive that this concept is under active development at the KEMA Laboratory in The Netherlands (21, 22).

The KEMA concept of a single-region homogeneous-suspension reactor involves spherical (U, Th)O<sub>2</sub> fuel particles carried in D<sub>2</sub>O at about 300°C. A mean particle diameter of 5μ is presently preferred; however, both larger (30μ) and

smaller (0.02–0.05 $\mu$ ) particles have been studied and remain possible future choices. With 5 $\mu$  particles, essentially all fission fragments escape from the fuel, and may be adsorbed on fine ThO<sub>2</sub> scavenger particles; these are separable from the fuel by hydroclones, which are also used to control the fuel concentration. Extensive damage at the surface of the fuel particles appears to be caused by the escaping fission fragments; the original 5 $\mu$  particles are broken down into fine debris ( $\sim$ 0.02 to 0.05 $\mu$ ), the fines constituting a colloidal system with quite different physical-chemical properties than the initial suspension. The rate of degradation is very temperature dependent (22): at room temperature, the particles do not survive exposures of 200 MWd/t, but at 300°C preliminary evidence indicates a degradation rate only one percent as great. Larger particles, especially if coated with a layer of ThO<sub>2</sub>, would presumably not be susceptible to this form of radiation damage, but would preclude separation of fission products by the recoil process. A colloidal system, though studied less extensively than the suspension, appears feasible; here too, however, the recoil process for fission product removal would not work, in this instance because of recapture of the fission products associated with the smaller separation and larger surface area of the colloidal particles as compared with the 5 $\mu$  particles.

In addition to radiation damage, other potential problems of suspension fuels include agglomeration, settling, caking, and erosion (21). Extensive research, including operation of several high-temperature, high-pressure loops and a near-critical, low-temperature reactor, has revealed acceptable ranges of operating variables—e.g. temperature, flow velocity, pH, etc.—within which these problems can apparently be avoided. It is now necessary to test the fuel and the technological components of the system such as pumps, gas separators, heat exchangers, and concentration controllers under actual reactor operating conditions. For this purpose, the KEMA Suspension Test Reactor (*KSTR*) (21, 22) has been constructed, and is scheduled to begin operation this year (1972). For practical reasons pertinent at the time the design was fixed, the *KSTR* cannot be operated above 250°C. Thus, the further clarification of temperature-dependent effects between 250 and 300°C will require supplementary loop and capsule experiments. Except for this limitation, operation of the *KSTR* will show whether suspension fuels are feasible for aqueous homogeneous reactors and will help to identify any additional problems that may have to be tackled for further development to proceed (22).

Characteristics of the *KSTR* and of a Suspension Power Reactor (*SPR*) are given in Table 3. This *SPR* is not a breeder, primarily because it has an extremely high specific power—i.e. 4.8 MW(t)/kg—and a correspondingly large Pa loss (as stated earlier). It is evident that breeding ratios well above unity could be achieved by reducing the specific power. However, we are not able to identify the economic optimum performance. Such studies will undoubtedly be encouraged by successful operation of the *KSTR*.

*Molten-salt systems.*—Molten-salt breeder reactors (*MSBR*) comprise a varied class of fluid-fuel reactors in which the fuel usually is a mixture of metal

**TABLE 3. CHARACTERISTICS OF KEMA SUSPENSION TEST REACTOR AND OF A SUSPENSION POWER REACTOR (22)**

	KSTR	SPR
Fuel	25% <sup>235</sup> UO <sub>2</sub> /75% ThO <sub>2</sub>	1.5% <sup>233</sup> UO <sub>2</sub> /98.5% ThO <sub>2</sub>
Concentration, (U, Th)O <sub>2</sub> , g/liter	280	200
Moderator	H <sub>2</sub> O	D <sub>2</sub> O
Reflector	BeO-graphite	Graphite
Power, [MW(e)]/[MW(t)]	0/1	250/788 <sup>a</sup>
Power density, kW/liter	50	35
Maximum temperature, °C	250	300
Specific inventories:		
[kg <sup>233</sup> U/MW(e)]		0.65
[kg D <sub>2</sub> O/MW(e)]		200
Conversion ratio		0.99
Volume of fuel, liters:		
In core	20	22,000
External	50	26,400

<sup>a</sup> Four units per 1000-MW(e) station.

halides including the fissile and/or fertile materials as dissolved ThF<sub>4</sub>, UF<sub>4</sub>, or PuF<sub>4</sub>. Both fluorides and chlorides have been considered, but for thermal reactors the fluorides are preferred because of the high cross section<sup>11</sup> of Cl (33 barns). (Other salts have been considered as well,<sup>12</sup> but they<sup>13</sup> are not suitable for thermal breeder reactors and have not received any development effort.) Fluorides of Li, Be, Na, K, Rb, and Zr have been studied extensively, as well as binary and ternary mixtures of these with each other and with thorium and uranium (Thoma & Grimes 30, Grimes 31). As a class, these salts are stable, have low vapor pressures in the temperature range required to drive a modern steam cycle, have adequate solubilities for thorium, uranium, and plutonium, and are chemically compatible with a satisfactory moderating material, graphite, and with a number of commercial nickel-based alloys.

Molten-salt reactors were first considered for the Aircraft Nuclear Propulsion Project, because of the very high temperatures (~900°C) that could be attained with moderate system pressure. The Aircraft Reactor Experiment (*ARE*), operated at the Oak Ridge National Laboratory in 1954, employed a beryllium oxide moderator, Inconel structure, and a mixture of NaF, ZrF<sub>4</sub> and UF<sub>4</sub> as fuel. Work on a beryllium-moderated reactor, the Aircraft Reactor Test, continued until 1957, and resulted in much valuable experience with materials and components. However, these systems could not directly form the basis for a breeder reactor because of the high neutron-absorption cross sections of some of the materials, and because of the necessity of separating the salt from the moderator by Inconel. In 1959, studies (MacPherson et al 32, MacPherson 33) of civilian

power applications of the molten-salt technology (demanding rather lower temperatures than the ANP application—e.g. 700°C versus 900°C) led to the selection of graphite as the preferred moderator, Hastelloy-N (Ni, 16% Mo, 7% Cr, 4% Fe, 0.05% C) as the preferred structural material, and  ${}^7\text{LiF}\cdot\text{BeF}_2$  (66-34 mole %) as the basic fuel carrier. Isotopic separation of the lithium is required, but it was established that the cost (\$120/kg of lithium containing 50 ppm of  ${}^6\text{Li}$ ) was entirely acceptable. An important factor in this selection was the determination that salt does not enter the pores in graphite provided the pores near the surface are small enough.

The evolution of design concepts for molten-salt breeder reactors has been intimately related to developments in chemical-processing technology. In the mid-60s, attention was focused on a two-stream concept (Robertson et al 34), with fissile and fertile materials completely segregated in separate salt streams. In this concept, the fuel stream, containing no thorium, flowed through the core in two-pass, re-entrant graphite tubes. The blanket salt flowed at a much lower rate through interstitial passages between the fuel tubes and also surrounded the core region. The fuel salt was processed by fluorination to remove uranium, and the fission products were then removed from the salt by vacuum distilling the  $\text{LiF}\cdot\text{BeF}_2$  carrier salt away from the less volatile rare earth fluorides. Bred uranium was removed from the blanket salt by continuous fluorination. No assured process was at hand for separation of the  ${}^{233}\text{Pa}$  from the blanket salt, but the large volume of the blanket salt, mostly in low flux zones, served to keep Pa losses at a low level. The vacuum distillation process, though satisfactory for the thorium-free fuel salt, did not appear suitable for salts containing thorium, as an adequate separation between thorium and fission products could not be achieved. Thus, consideration of single-fluid reactors or of two-stream designs containing some thorium in the fuel stream was inhibited by lack of a suitable low-cost chemical treatment.

By 1967, laboratory tests of a liquid-metal extraction process (35) had progressed to the point that satisfactory separations of fission products from thorium-bearing salt seemed attainable, and continuous, rapid isolation of  ${}^{233}\text{Pa}$  from a single fissile-fertile salt stream became feasible. The process involved the selective chemical reduction and extraction of salt constituents into liquid bismuth at 600°C, by multistage counter-current extraction. Good separation was achieved between uranium, protactinium, and thorium. Separation factors between thorium and the important rare earth fission products were much smaller, but appeared adequate.

At about the same time, it was realized (36) that if, in a single-fluid reactor, one used a high C/U ratio in the middle of the core and a low C/U ratio near the edge, the single fissile-fertile salt could serve the functions of both a fuel salt and a blanket salt. Fissile-nuclide absorptions predominate in the well-moderated central zone, and thorium captures predominate in undermoderated peripheral zones where the neutron spectrum is much harder. Recent attention has thus been concentrated, in the interest of greater simplicity and reliability of the core

structure, on the single-fluid, multizone concept (36). Though it is much simpler than the two-fluid reactor, the single-fluid design has a specific fissile inventory perhaps 50 percent larger than the two-fluid reactor.

Within the last two or three years, remarkable improvements in the chemical-processing flow sheet have been made (25). The principal features of the new flow-sheet are shown in Figure 8. A side stream of fuel salt is withdrawn from the reactor fuel circuit. Uranium is first removed from this stream by fluorination, with 99% efficiency, and added to the processed salt returning to the reactor. The small residual amount of uranium and the protactinium are extracted into a bismuth stream from which they are transferred by hydrofluorination to a closed recirculating salt loop; here the protactinium is allowed to decay and uranium is recovered by fluorination. Plutonium, produced in very small amounts, is removed from the salt along with the protactinium, and is eventually discarded from the Pa decay tank. Rare earths are countercurrently extracted into a bismuth stream which also contains thorium at close to the solubility limit. The rare earths subsequently transfer to a countercurrent LiCl stream, which does not accept appreciable amounts of thorium. In this way, separation factors of  $10^8$  are achieved between thorium and divalent rare earths (Sm, Eu) and alkaline earths (Sr, Ba), while separation factors of  $10^4$  are observed between thorium and the trivalent rare earths (e.g. La, Ce, Nd). This process is economical in its use of reductant; the major effect is the replacement of rare earths in the fuel salt by an equivalent amount of  $^7\text{Li}$  as LiF. The amount of lithium added to the fuel salt in this manner in 30 years of operation would be about the same as the lithium inventory in the reactor. The amount of thorium lost from the system in the present chemical process flow sheet (25) is approximately four times as great as the amount converted to fissile fuel, leading to a thorium utilization efficiency of about 20%. However, expected modifications of the flow sheet should reduce thorium losses and increase the utilization efficiency to more than 50%, even at these very high processing rates.

Present plans call for processing of the entire fuel-salt inventory on a ten-day cycle [corresponding to a process-stream flow rate of about 3 1/2 liter/min for a 1000 MW(e) reactor]. Effective cycle times for the various nuclides depend on the distribution coefficients between the salt streams and the bismuth. The most important ones range from ten days for  $^{233}\text{Pa}$  to about 30 days for most of the rare earth poisons. Characteristics of a single-fluid MSBR are listed in Table 4, and a neutron balance appropriate for the above processing conditions is given in Table 5.

Recent detailed cost estimates (37) indicate that fixed charges for the chemical processing plant may be 0.5 to 0.7 mill/kWhr(e), corresponding to a unit cost of \$2/kg of thorium and uranium processed. The overall fuel cycle cost, including fixed charges, is estimated to be 0.9 to 1.1 mills/kWhr(e). While these costs for a breeder cycle are comparable to or less than those anticipated for most other reactors (Ref. a, Table 2; 18, 38, 39), it is possible that somewhat lower fuel costs could be achieved by batch-processing the fuel salt at intervals of several years (Perry & Bauman 40), with an average conversion ratio above 0.9. Cost savings



TABLE 4. CHARACTERISTICS OF A 1000-MW(e)  
MOLTEN-SALT BREEDER REACTOR<sup>a</sup> (25)

Reactor thermal power, MW(th)	2250
Overall plant thermal efficiency, %	44
Fuel salt inlet and outlet temperatures, °C	566, 704
Coolant salt inlet and outlet temperatures, °C	454, 621
Throttle steam conditions	240 atm, 538 °C
Core height/diameter, m	4.0/4.3
Radial blanket thickness, m	0.5
Graphite reflector thickness, m	0.8
Number of core elements	1412
Size of core elements, cm	10.2 × 10.2 × 396
Salt volume fraction in core, %	13
Salt volume fraction in undermoderated zones, %	37 and 100
Salt volume fraction in reflector, %	< 1
Average core power density, W/cm <sup>3</sup>	22
Maximum thermal neutron flux, n/(cm <sup>2</sup> -sec)	8.3 × 10 <sup>14</sup>
Graphite damage flux (> 50 keV) at point of maximum damage, n/(cm <sup>2</sup> -sec)	3.3 × 10 <sup>14</sup>
Estimated graphite life, years <sup>b</sup>	4
Total salt volume in primary system, liters	48,700
Thorium inventory, kg	68,000
Fissile fuel inventory of reactor system and processing plant, kg	1470
Breeding ratio	1.07
Fissile fuel yield, %/year	3.6
Fuel doubling time (exponential), years	19

<sup>a</sup> Design selected to minimize the ratio I<sup>2</sup>/G (Section on General Economic Considerations), subject to constraint on maximum fast neutron flux.

<sup>b</sup> Based on 80% plant factor and a fluence of 3 × 10<sup>22</sup> neutrons/cm<sup>2</sup> (> 50 keV).

for the breeder cycle might be realized by serving two or more reactors at a single site or a larger reactor with one processing plant. It is therefore not yet clear whether there is much difference in cost between the cycle that maximizes breeding performance and the economic optimum cycle.

Additional details of current reactor designs, plant designs, chemical processes, and system performance may be found in (24), (25), and Robertson (41).

*The molten-salt reactor experiment (MSRE).*—The 8-MW(t) MSRE was operated at Oak Ridge National Laboratory from 1965 to 1969. Its purpose was to demonstrate the basic compatibility of materials, the operability of components, and the maintainability of complex, highly radioactive equipment under representative operating conditions, including the effects of radiation and fission products. Detailed descriptions of the reactor have been published (Robertson 42, Haubenreich & Engel 43). Basically, it comprised a cylindrical graphite core, ~1.4 × 1.6 m, in a Hastelloy-N tank. The core consisted of vertical rectangular graphite stringers, 5 × 5 cm, with shallow channels machined on each longitudinal surface to give a salt volume fraction of 0.22. As with other molten-salt reactors,



TABLE 5. NEUTRON BALANCE IN A SINGLE-FLUID MOLTEN-SALT BREEDER REACTOR WITH TEN-DAY PROCESSING CYCLE<sup>a</sup>

	Absorptions	Fissions
<sup>232</sup> Th	0.9968	0.0030
<sup>233</sup> Pa	0.0045	
<sup>233</sup> U	0.9242	0.8245
<sup>234</sup> U	0.0819	0.0004
<sup>235</sup> U	0.0758	0.0618
<sup>236</sup> U	0.0074	
<sup>237</sup> Np	0.0064	
<sup>6</sup> Li	0.0032	
<sup>7</sup> Li	0.0161	
<sup>9</sup> Be	0.0070	0.0044 <sup>b</sup>
<sup>19</sup> F	0.0203	
Graphite	0.0510	
Fission products	0.0150	
Leakage	0.0221 <sup>c</sup>	
$\eta\epsilon$	2.2317	
Breeding ratio	1.0708	

<sup>a</sup> MSR Program Semiann. Progr. Rep. February 28, 1971. *USAEC Doc. ORNL-4676*, p. 44.

<sup>b</sup>  $n, 2n$  reaction in beryllium.

<sup>c</sup> Including delayed neutron losses.

the primary fuel circuit and the heat sink were separated by an intermediate heat transfer circuit; in MSRE the intermediate coolant was <sup>7</sup>LiF-BeF<sub>2</sub> (66-34 mole %). Heat was rejected to the atmosphere through an air-cooled heat exchanger in place of the steam generators of a power station.

Operation of the reactor with <sup>235</sup>U (33% enrichment) began in June 1965, and 9000 equivalent full-power hours of operation were accumulated by March 1968. During the last 15 months of that time the reactor was critical 80% of the time, and, during a six-month demonstration run, the reactor was critical 98% of the time.

The reactor was then shut down, and the uranium very efficiently removed in an on-site fluorination facility. <sup>233</sup>U was then added to the carrier salt, making the MSRE the first reactor to be fueled with this material. About 4200 equivalent full-power hours were accumulated with <sup>233</sup>U fuel. For the last three months of operation, additional reactivity for burnup was achieved by adding about 180 grams of plutonium, as PuF<sub>3</sub>; this was in addition to over 500 grams of Pu already present in the reactor, as a result of earlier operation with <sup>235</sup>U—<sup>238</sup>U.

In addition to demonstrating a reasonable technological base for molten-salt reactors, the MSRE served as an experimental tool for the study of chemical and metallurgical problems of molten-salt reactors. While substantial amounts of the noble gases were removed continuously during operation, significant quantities—typically 10% of the <sup>135</sup>Xe—migrated to the graphite moderator, which in MSRE

had a diffusion coefficient of about  $10^{-6}$  cm<sup>2</sup>/sec, large enough to provide very little effective resistance to diffusion of xenon. Thus, better methods for gas stripping and for preventing xenon diffusion into the graphite will be needed. Considerable data were also accumulated on the behavior of solid fission products, especially those that tend to remain in the free metallic state, such as Nb, Mo, and Ru. Deposition of such fission products on primary loop surfaces could cause serious afterheating in a large reactor; and some fission products, notably tellurium, may be detrimental to Hastelloy-N. The distribution of tritium in molten-salt systems was also studied in MSRE.

*Further technological developments.*—Notwithstanding a very large background of component testing (valves, seals, pumps, heat exchangers), materials development, and the successful operation of MSRE, there remain important areas in which potential problems have not been fully resolved or in which improved performance should hold promise for reduced costs.

(a) At present, it appears that available graphites suitable for an MSBR can retain acceptable properties up to neutron doses of about  $3 \times 10^{22}$  nvt ( $E > 50$  keV). In the present single-fluid MSBR design, the core power density is chosen so that this exposure will be reached in four years; thus, replacement of core graphite at four-year intervals will be required. (Selective replacement of graphite could increase the average life of the graphite in the core.) Graphites with greater endurance would be desirable, and recent experimental evidence (25) suggests that the resistance of graphite to radiation damage can be increased. Improved radiation-resistant surface coatings are needed to limit migration of xenon to the graphite. Failure to achieve such improvements, however, would add at most 0.015 to the expected <sup>135</sup>Xe poison fraction.

(b) Hastelloy-N, the structural alloy used in the MSRE core vessel and piping, suffers a severe loss of ductility as a result of helium accumulation due to ( $n, \alpha$ ) reactions. These include <sup>10</sup>B( $n, \alpha$ ) reactions induced by low-energy neutrons. It has been recognized for some time that small additions of Ti ( $\sim 0.5\%$ ) would largely prevent this ductility loss at temperatures up to 650°C, but the protection is lost when the temperature is raised to 750°C (25). In order to extend the temperature range over which good ductility can be maintained, larger additions of Ti have been investigated, as well as additions of Hf, Zr, or Nb. Experiments on small laboratory-prepared samples of these alloys show very promising results, but the improved radiation resistance must be demonstrated on large commercial heats of the modified alloys (25).

(c) Evidence from specimens of Hastelloy-N exposed to fuel salt in the MSRE indicated extensive micro-cracking or intergranular attack to average depths of about 0.1 mm and maximum depths of 0.2 to 0.3 mm. Subsequent laboratory tests have duplicated this behavior and seem to indicate that fission-product tellurium is associated with the effect; however, the specific mechanism has not yet been established. Although Hastelloy-N is still regarded as satisfactory for thick-walled components such as vessels and piping, an alternative material

may be required for primary heat-exchanger tubing. Some preliminary tests indicate that stainless steel (SS 304), while having less high-temperature strength than Hastelloy-N, is much less subject to attack by tellurium than is Hastelloy-N nickel is intermediate in its resistance to this attack. Resolution of this question will depend on results of further research.

(d) While tritium is produced to some degree in all reactors, the  $^3\text{H}$  production in an MSBR is unusually large (Briggs 44), owing primarily to the reactions  $^7\text{Li}(n,n,\alpha)^3\text{H}$  (threshold about 2.8 MeV) and  $^6\text{Li}(n,\alpha)^3\text{H}$  (all neutron energies). Total production in a 2250 MW(t) MSBR is calculated to be 2420 curies per day, of which only 30 curies per day come directly from fission. The tritium tends to escape rapidly from the salt, partly by diffusion through the large heat-exchange surfaces. Tritium in the steam system would be difficult to control; hence methods must be found to collect and store the tritium and to prevent it from reaching the steam system. Several methods (44) offer promise of adequately controlling the distribution of tritium in the MSBR plant, and investigations are under way to determine their relative merit and effectiveness.

(e) While engineering components in the MSRE performed well, much larger components will be needed for full-scale power plants; and no steam generator, with heat supplied by a molten salt, has been tested. Scale-up of components, development and testing of a steam generator, and further development of the off-gas system will be major development requirements of the program. Similarly, tools and techniques for maintenance or replacement of large activated or contaminated components will require substantial further development effort, notwithstanding the considerable progress already made in connection with the MSRE.

(f) Although much important information concerning the behavior of the "noble metal" fission products (Nb, Mo, Tc, Ru, and Te) was obtained from the MSRE, further understanding of the basic processes is needed in order to predict confidently the behavior of these nuclides as a function of reactor operating conditions.

It would thus be incorrect to suggest that all major problems of molten-salt breeders have been solved. On the other hand, the progress made in this system since the 1957 task force chose it over the other fluid-fuel systems has been gratifying. Though the program is now conducted on a technology level, there is substantial prospect that if a larger molten-salt breeder experiment were built in the next few years it would have very good chances for success.

#### SOLID-FUEL THERMAL BREEDERS

*Heavy-water reactors.*—Because of its superior neutron economy, the heterogeneous  $\text{D}_2\text{O}$ -moderated reactor with  $\text{D}_2\text{O}$  coolant has been considered as a possible thermal breeder reactor. However, two central conclusions have always been reached in such studies:

(a) The thorium cycle is found to be somewhat more expensive than the natural or slightly enriched uranium cycle (Rosenthal 45, 46, Kasten 47, 48).

Although optimum fuel exposures about twice those with uranium cycles are achieved (45, 48) (in part because the reactivity of the bred  $^{233}\text{U}$  is greater than that of the  $^{235}\text{U}$  feed), the higher cost of enriched uranium feed more than offsets the reductions in fabrication and processing costs.

(b) Although breeding is possible with the thorium cycle, it can be achieved only at high fuel processing rates—i.e. at a substantial cost penalty. Thus, optimum costs for the thorium cycle occur at conversion ratios less than unity.

From data presented by Naudet (49), for example, in a preliminary analysis of thorium cycles for  $\text{D}_2\text{O}$ -cooled and -moderated reactors (including fuel inventory charges implicit in Table VI of reference 49), we may estimate that for a  $^{235}\text{U}$  price of \$12/g and combined fabrication and reprocessing costs of \$100/kg, the optimum fuel exposure would be about 35 MWd/kg corresponding to a conversion ratio of about 0.9. To achieve a conversion ratio of unity would require reducing the exposure to less than 10 MWd/kg, and would increase the fuel-cycle cost from about 0.7 mill/kWhr(e) to about 1.5 mills/kWhr(e). Even with an increase of  $^{235}\text{U}$  price to \$50/g (\$80/lb  $\text{U}_3\text{O}_8$ ), the optimum conversion ratio is still only about 0.95. Thus, the “leverage” exerted by fabrication and processing costs still favors operation of the reactor in a sub-breeder mode. If combined fabrication and reprocessing costs of \$50/kg are assumed, Naudet’s data imply an optimum conversion ratio of about 0.93 for  $^{235}\text{U}$  at \$12/g, and about 1.00 for  $^{235}\text{U}$  at \$50/g.

In connection with the first general conclusion, Lewis (50) has pointed out that a mixed cycle using thorium in conjunction with slightly enriched uranium avoids the high fueling cost associated with fully enriched uranium, while profiting from the high unit value of the bred  $^{233}\text{U}$ . Such a “valu-breeder,” though having a conversion ratio of 0.8–0.9, transforms a less valuable fuel into a more valuable one, and may be expected to yield net fueling costs that are reasonably low (though not necessarily below the fuel cost for the natural uranium cycle) and are relatively insensitive to changes in raw material prices.

*Gas-cooled, solid-moderated reactors.*—For heterogeneous graphite-moderated reactors, such as the natural uranium reactors developed in France and in the United Kingdom, or the slightly enriched advanced gas-cooled reactor (AGR) developed in the United Kingdom, conclusions similar to those just stated for heterogeneous  $\text{D}_2\text{O}$  reactors apply—that is, because of the much higher cost of fissile material, the thorium cycle cannot compete with the natural or slightly enriched uranium cycle. Thus, the only serious attention to the possibility of breeding has been given to the more nearly homogeneous high-temperature gas-cooled reactors in which the fuel is dispersed in the moderator. The great advantage of the HTGR as a possible breeder derives from the elimination of fuel cladding. However, a major consequence of the dispersal of fuel in the moderator is a marked enhancement of resonance neutron capture in the fertile material e.g.  $^{238}\text{U}$ —which makes necessary the use of higher fuel enrichment than is required for the heterogeneous reactors. In these circumstances, the thorium cycle, re-

quiring highly enriched  $^{235}\text{U}$  or its equivalent in  $^{233}\text{U}$  or fissile Pu, does not suffer the relative disadvantage of high fissile material cost that was observed for reactors designed to use natural or slightly enriched uranium.

A wide range of possibilities confronts the designer of the "solid-homogeneous" reactor (which may, in fact, retain an appreciable degree of heterogeneity): the choice of moderator, the choice of fissile and fertile materials, the extent of fuel dispersion in the moderator, the average composition of the core (governed in part by the considerations mentioned earlier), the power density and specific power, the duration of the fuel exposure, etc.

For the moderator, graphite has become the universal choice for gas-cooled reactors, because it is relatively inexpensive, it has excellent physical properties at high temperature, and it is more resistant to radiation damage than Be or BeO. Helium is the preferred coolant, partly because it has good heat transport properties and has negligible effect on the neutron economy, but primarily because it is chemically inert; otherwise attractive candidates such as  $\text{CO}_2$  react with graphite at temperatures of interest in HTGRs (800–900°C) (51). Though the low-enriched uranium cycle has received considerable attention in Europe, the  $^{235}\text{U}$ - $^{232}\text{Th}$ - $^{233}\text{U}$  cycle is generally found to be economically advantageous if an assured supply of enriched  $^{235}\text{U}$  is available (Carlsmith, Podeweltz & Thomas 52, Jaye, Fischer & Lee 53). However, the thorium cycle in HTGRs also requires the development of an economic recycle technology, whereas the slightly-enriched-uranium reactor can more readily operate on the throw-away cycle.

After many years of development in Europe, in the United Kingdom, and in the United States, the HTGR has achieved commercial status, although further development work is required in connection with the recycle of  $^{233}\text{U}$ . Descriptions of current designs of HTGRs and the status of their technology may be found in Ref. a of Table 2, Johnston (54), Shepherd (55), Mattick et al (56).

For purposes of the present discussion, we summarize in Table 6 a few characteristics typical of HTGRs optimized for economical energy generation. A typical neutron balance, compatible with the conditions of Table 6, is shown in Table 7.

It will be noted that the optimum fuel exposures are large (65,000 MWd/T, 1.7 fira), and the average conversion ratio ( $\sim 0.83$ ) is therefore appreciably lower than might be physically realizable with this system. It has been suggested by Gulf General Atomic that a substantial increase in conversion ratio could be achieved without appreciable penalty in fuel-cycle cost: (a) if some BeO were incorporated in the fuel (and incidentally reprocessed with the fuel) to gain some advantage from the ( $n,2n$ ) reaction, and (b) if fuel elements were developed that would release volatile fission products (Xe, Kr, possibly I) and also the  $^6\text{Li}$  produced by ( $n,\alpha$ ) reactions in Be. Carlsmith and Thomas at ORNL have calculated conversion ratios and fuel-cycle costs as functions of fuel exposure for four HTGR fuel concepts—that is, all-graphite elements or elements having 2.4 parts graphite to one part BeO (by volume), in each case with or without release of the abovenamed neutron poisons. The results of their study (57, 58) are shown

**TABLE 6. CHARACTERISTICS OF A 1000-MW(e) HIGH-TEMPERATURE GAS-COOLED REACTOR<sup>a</sup>**

Reactor thermal power, MW(th)	2318
Overall plant thermal efficiency, %	43.1
Fuel	ThO <sub>2</sub> /UO <sub>2</sub>
Moderator	Graphite
Coolant	Helium
Coolant inlet and outlet temperature, °C	428, 829
Throttle steam conditions	240 atm, 565 °C
Core height/diameter, m	4.4/9.5
Average core power density, W/cm <sup>3</sup>	7.4
Thorium inventory, kg	41,700
Fissile loading, kg	1290
Fuel exposure, years, fifa, MW(th)d/kg(Th+U)	4, 1.7, 65
Conversion ratio	0.83

<sup>a</sup> From Ref. a, Table 2.

**TABLE 7. NEUTRON BALANCE FOR HTGR (EQUILIBRIUM RECYCLE)<sup>a</sup>**

<sup>232</sup> Th	0.7747
<sup>232</sup> Pa	0.0190
<sup>233</sup> U	0.7458
<sup>234</sup> U	0.0686
<sup>235</sup> U	0.2505
<sup>236</sup> U	0.0085
<sup>237</sup> Np	0.0031
<sup>238</sup> U, <sup>240</sup> Pu	0.0041
<sup>239</sup> Pu, <sup>241</sup> Pu	0.0037
Fission products	0.1784
Carbon	0.0370
Leakage and control	0.0852
Total ( $\eta_e$ )	2.1786
$\eta$ , <sup>233</sup> U	2.230
$\eta$ , <sup>235</sup> U	1.979
$\eta$ , average for system	2.173
Conversion ratio	0.826

<sup>a</sup> From Ref. a, Table 2.

in Figure 9. Curve *A* in Figure 9 represents essentially the present concept of all-graphite, fission-product-retaining fuel. Curve *C* indicates that addition of BeO, without release of fission products, is probably not economically worthwhile under present economic conditions, though it does appreciably increase the

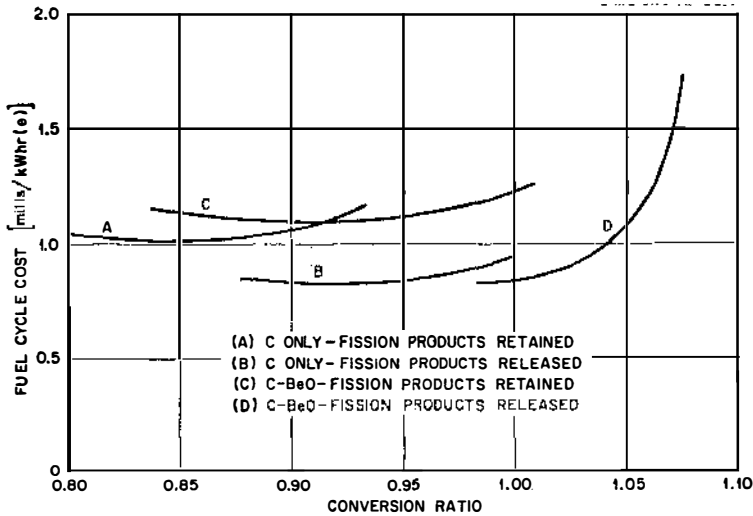


FIGURE 9. HTGR fuel-cycle cost.

optimum conversion ratio. Curves *B* and *D* indicate that release of volatile fission products could lead to both increased conversion ratio and decreased cost, while the combination of added BeO and releasing fuel would bring the optimum conversion ratio close to unity.

It must be noted that the costs for the more speculative cases (representing fuel concepts that have not yet been developed) are only estimates, and do not in fact include any additional capital or operating costs that might be associated with collection and disposal of the released materials or with a more radioactive coolant circuit. Nonetheless, it seems possible to conclude: (a) that current economic and technical conditions favor breeding ratios appreciably less than unity, and (b) that potential technological developments of HTGR fuel can now be recognized—i.e., addition of BeO, release of volatile poisons—which if successful would allow conversion ratios of unity or above, and which would largely protect the HTGR from the effects of higher fuel prices.

*Light-water breeder reactor.*—The light-water breeder Reactor (*LWBR*) is a pressurized-water reactor, conventional in most respects except for the design of its core. The LWBR would operate on the Th-<sup>233</sup>U cycle, and is conceived as a break-even breeder—that is, it would make just enough fuel to satisfy its own needs and to compensate for chemical processing losses. The underlying purpose is to enhance very greatly the utilization of natural uranium resources by pressurized-water reactors (*PWR*) while relying to the greatest extent possible on established *PWR* technology. The reactor is being developed by the Bettis Atomic

Power Laboratory, under the direction of the Naval Reactors Division of the USAEC.

Essential factors that make such self-sustaining operation possible in a pressurized-water reactor are the seed-blanket concept and the development of low-cross-section Zircaloy-4 as a cladding material for water reactors. The seed-blanket concept (see Figure 10) (59) comprises essentially a series of fissile islands (seeds) embedded in a sea of fertile material. The seed composition and geometry

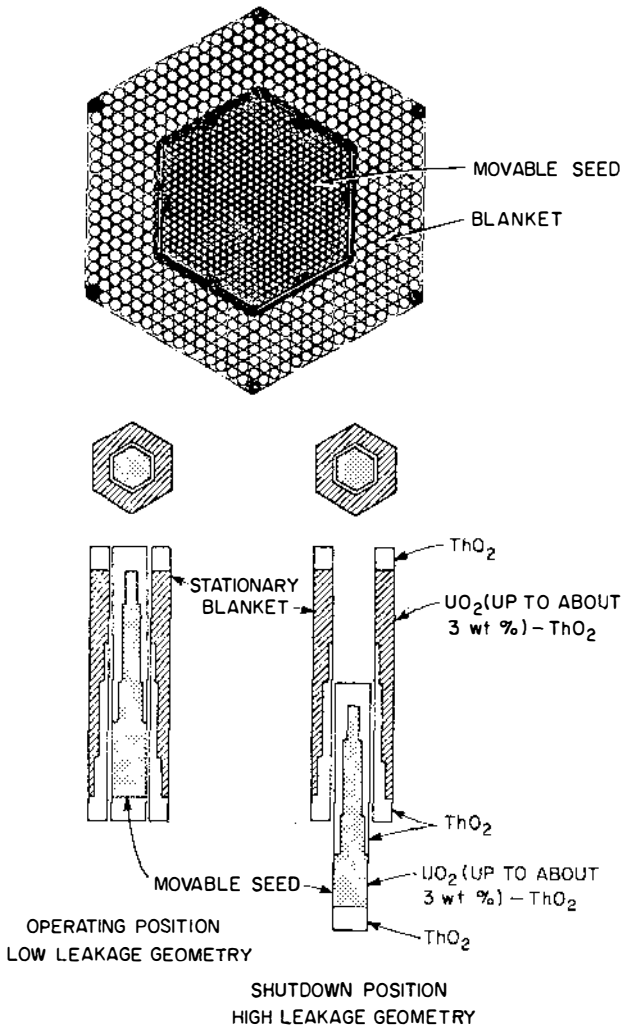


FIGURE 10. Typical LWBR fuel module.



may be optimized for neutron production and energy generation, while the surrounding blanket may be optimized to minimize neutron losses and enhance the production of fuel. For example, the seed has a very low volume-fraction of water, compared to more conventional PWRs, reducing parasitic capture in the hydrogen and enhancing the production of neutrons by fast fission and ( $n,2n$ ) reactions. As may be seen in Figure 3, under-moderation in a water lattice does not impair breeding performance as much as with other moderators; this behavior is associated with the particular energy dependence of the hydrogen scattering cross section (5). At the same time, the low moderator fraction causes an increase in resonance neutron capture in thorium, requiring a correspondingly high fissile loading. While this decreases the fuel specific power, it also reduces neutron losses in  $^{238}\text{Pa}$ , in  $^{135}\text{Xe}$ , and in fuel cladding.

The seed is made movable with respect to the blanket, so that reactivity can be varied by altering the geometry of the core, and hence the relative neutron absorption rates in the seed and blanket regions. In this way, thorium is used as the control material, eliminating the need for nonproductive neutron captures in conventional control poisons.

Operation of the LWBR has been proposed in two phases.

(a) An initial, pre-breeder phase during which the reactor would be fueled with  $^{235}\text{U}$  and would accumulate  $^{238}\text{U}$  in the blanket regions for subsequent use as fuel for the breeder phase. This initial phase would last for about eight years. The exact duration is not well defined, since a gradual transition would be made in the period six to ten years after initial startup.

(b) An asymptotic breeder phase, in which the reactor would be fueled with  $^{238}\text{U}$  and would end each fuel cycle ( $2\frac{1}{2}$  to 3 years at 0.8 plant factor) with 1 to 2% more fuel than at the beginning of the cycle.

Some relevant characteristics of a 1000-MW(e) LWBR (60, 61) during these two phases of operation are summarized in Table 8. These characteristics are not necessarily those of current LWBR designs, for which we have found no detailed information. However, the performance shown is consistent with recent official statements concerning the reactor (59, 61), and we believe these data to be fairly representative of the potential of this concept.

The most important entry in Table 8, from the point of view of breeding, is the fissile inventory ratio—i.e., the ratio of fissile fuel content at the end of the fuel cycle to the initial content. The design goal for the LWBR is for the fissile inventory ratio to exceed unity by a margin sufficient to allow for losses in fuel reprocessing and refabrication, while achieving an adequate reactivity lifetime from the standpoint of fuel-cycle costs and plant availability. The attainment of this goal, with fuel exposures of nearly one fission and with  $^{135}\text{Xe}$  and  $^{238}\text{Pa}$  retained in the fuel, is the result of very careful optimization of core-module design, the unavoidable neutron losses in fission products being balanced by careful control of other losses and by maximizing the small neutron production in materials other than the fissile fuel.

Neutron balances for the initial breeder cycle (fueled with  $^{238}\text{U}$ ) and for an

**TABLE 8. CHARACTERISTICS OF A 1000-MW(e) LIGHT-WATER BREEDER REACTOR (60, 61)**

	Pre-breeder	Breeder
Reactor thermal power, MW(t)	3200	3200
Overall plant thermal efficiency, %	31.2	31.2
Fuel	ThO <sub>2</sub> /UO <sub>2</sub>	ThO <sub>2</sub> /UO <sub>2</sub>
Coolant-moderator	H <sub>2</sub> O	H <sub>2</sub> O
Coolant inlet and outlet temperatures, °C	271,306	271,306
Throttle steam conditions	37 atm, 247 °C	37 atm, 247°C
Core height/diameter, m	2.2/5.2	2.2/5.2
Number of seed assemblies	61	61
Average core power density, W/cm <sup>3</sup>	68	68
Thorium inventory, kg	150	280
Fissile loading, kg	3500	3200
Fuel lifetime, equivalent full-power hours	19,000	17–20,000
Maximum exposure, seed/blanket, MWd(t)/kg	77/18	77/18
Fuel burnup, fissions/initial fissile atom	0.8	0.8–0.9
Fissile inventory ratio		1.01–1.02

equilibrium cycle are shown in Table 9, taken in each instance at about 10,000 full-power hours. It should be noted that these data were developed for an early LWBR design (60), and are not necessarily precisely applicable to more fully developed designs; they are to be regarded as indicative of the approximate distribution of neutron absorptions among the materials present.

**TABLE 9. APPROXIMATE NEUTRON BALANCE FOR A 1000-MW(e) LIGHT-WATER BREEDER REACTOR AT 10,000 EFPH<sup>a</sup>**

	First breeder cycle	Equilibrium cycle
<sup>233</sup> U + <sup>235</sup> U	1.00	1.00
<sup>232</sup> Th + <sup>234</sup> U	1.03	1.00
<sup>238</sup> U	—	0.02
<sup>233</sup> Pa	0.02	0.02
H, O, Zr	0.08	0.07
<sup>135</sup> Xe	0.03	0.03
Other fission products	0.09	0.08
Leakage	<<0.01	<<0.01
Total = ηε	2.26	2.23
Conversion ratio	1.01	0.98

<sup>a</sup> EFPH = equivalent full-power hours.

Because the fuel specific power is relatively low, the approach to equilibrium fuel composition following initial operation with  $^{238}\text{U}$  will be correspondingly slow (as mentioned earlier).

The basic technology of the LWBR is essentially that of contemporary PWRs, and no substantial new developments would appear to be required. There are, of course, numerous differences in detail, and these have required careful assessment, both in design and by numerous development tests. A demonstration core is scheduled to be installed in the Shippingport PWR. It will be fueled with  $^{238}\text{U}$ , and is intended to provide overall confirmation of the anticipated breeding performance, as well as of fuel and component characteristics.

Project estimates of capital cost and of fuel cycle economics, as affected by the unique characteristics of the LWBR design, have not been published recently. We are unable to say at this time whether the self-sustaining breeder design is close to an economic optimum.

### THE CASE FOR THERMAL BREEDERS

The world's decision to go primarily for fast breeders was based on the high value of  $\bar{\eta}\epsilon$  for the  $^{238}\text{U}$ - $^{239}\text{Pu}$  cycle in a fast reactor, as compared to the marginally low value of  $\bar{\eta}\epsilon$  for the  $^{232}\text{Th}$ - $^{233}\text{U}$  cycle in a thermal reactor. There is no question that a fast breeder that breeds can be built; there will always remain some doubt as to whether a thermal "breeder" will breed until an actual reactor of this type is operated.

What, then, are the justifications for continuing the efforts to develop thermal breeders? We would list them as follows.

(a) Once the initial fuel inventory is supplied, thermal breeders make a different raw material, thorium, usable as nuclear fuel. To be sure, there is plenty of low-grade uranium in the rocks to last mankind for eons; yet there may be economic advantage in providing an alternative fuel option. Though there seems little doubt that thorium could be burned in a fast breeder, the breeding performance of the fast reactor is substantially reduced with thorium (47, 48).

(b) Though fast breeders enjoy high breeding gains, this is partly compensated by their relatively low specific power, especially in a growing energy economy. A low inventory thermal breeder conserves raw material about as well as a fast breeder, even though its doubling time may be much larger.

(c) The foregoing are rather theoretical arguments: today the thermal breeder is looked upon primarily as an insurance policy. If for some reason the fast breeder, despite its promise, does not work out as planned, the thermal breeder, depending as it does upon totally different and in some cases well demonstrated technology, would be expected to take over. This is the primary justification for development of thermal breeders in the face of the very nearly total commitment to fast breeders.

It will be interesting to see whether, by the 1980s when the full returns on fast reactors are in, the nuclear community will be grateful for the protection

afforded by its thermal breeder insurance policy, or will consider thermal breeders to have been an unfruitful diversion from the main line of development.

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## LITERATURE CITED

1. Häfele, W., Faude, D., Fischer, E. A., Laue, H. J. 1970. *Ann. Rev. Nucl. Sci.* 20:393-434
2. Alexander, L. G. 1964. *Ann. Rev. Nucl. Sci.* 14:287-322
3. Chernick, J., Moore, S. O. 1959. *Nucl. Sci Eng.* 6:537-544
4. Evaluated Nuclear Data File, National Neutron Cross Section Center, Brookhaven National Laboratory, *ENDF/B-Version II*
5. Perry, A. M., Ragan, G. L. 1972. *USAEC Doc. ORNL-TM-3827*
6. Glasstone, S., Edlund, M. C. 1952. *Elements of Nuclear Reactor Theory*, p. 45. New York: Van Nostrand
7. Weinberg, Alvin M., Wigner, E. P. 1958. *The Physical Theory of Neutron Chain Reactors*, p. 288. Chicago: Univ. of Chicago Press
8. Evaluated Nuclear Data File, *ENDF/B-Version III*
9. Glasstone, S., Edlund, M. C., op. cit: 332-338
10. England, T. R. 1963. *USAEC Doc. WAPD-TM-333; Add. No. 1*. 1965
11. Reactor Fuel-Cycle Costs for Nuclear Power Evaluation 1971. *USAEC Doc. WASH-1099*
12. Potential Nuclear Power Growth Patterns 1970. *USAEC Doc. WASH-1098*
13. Whitman, M. J., Tardiff, A. N. 1971. *A/Conf. 49/P-72*
14. Faulkner, R. L. 1971. *A/Conf. 49/P-59*
15. Dietrich, J. R. 1963. *Power Reactor Technology* 6(4):1-38
16. Dietrich, J. R. 1967. *Nucl. News* (September):38
17. Rosenthal, M. W., Kasten, P. R., Briggs, R. B. 1970. *Nucl. Appl. Tech.* 8:107-118
18. Current Status and Future Potential of Light-Water Reactors 1968. *USAEC Doc. WASH-1082*
19. Report of the Fluid Fuel Reactors Task Force 1959. *USAEC Doc. TTD-8507*
20. Lane, J. A., MacPherson, H. G., Maslan, F., Eds. 1958. *Fluid Fuel Reactors*. Reading, Mass: Addison-Wesley
21. Hermans, M. E. A., Kersten, J. A. H., Warschauer, K. A., Went, J. J. 1966. *Conf. Utilization Thorium Power Reactors*, 146-165. Vienna: Int. At. Energ. Agency
22. Went, J. J., Hermans, M. E. A. 1971. *A/Conf. 49/P-020*
23. Thomas, C. E. et al 1960. *USAEC Doc. BAW-1171*
24. Rosenthal, M. W. et al 1970. *Nucl. Appl. Tech.* 8:101-219
25. Rosenthal, M. W., Haubenreich, P. N., McCoy, H. E., McNeese, L. E. 1971. *At. Energ. Rev.* 9:601-650
26. Bunker, M. E. 1963. *UASEC Doc. LA-2854*
27. Hahn-Meitner-Institut für Kernforschung Berlin GmbH. Jahresbericht 1970. *HMI-B 112*
28. Nozawa, R. 1964. *A/Conf. 28/P-120*
29. Reiffel, L. 1958. *A/Conf. 15/P-418*
30. Thoma, R. E., Grimes, W. R. 1957. *USAEC Doc. ORNL-2295*
31. Grimes, W. R. 1970. *Nucl. Appl. Tech.* 8:137-156
32. MacPherson, H. G. et al 1959. MSR Project Quar. Progr. Rep., (February 28). *USAEC Doc. ORNL-2723*
33. MacPherson, H. G. 1959. *Conf. Phys. Breeding*: 335-341. *USAEC Doc. ANL-6122*
34. Robertson, R. C., Briggs, R. B., Smith, O. L., Bettis, E. S. 1970. *USAEC Doc. ORNL-4528*
35. MSR Program Semiann. Progr. Rep. 1967. *USAEC Doc. ORNL-4191*: 248
36. MSR Program Semiann. Progr. Rep. 1968. *USAEC Doc. ORNL-4254*: 66
37. MSR Program Semiann. Progr. Rep. 1971. *USAEC Doc. ORNL-4728*: 179
38. An Evaluation of Advanced Converter Reactors 1969. *USAEC Doc. WASH-1087*
39. An Evaluation of Gas-Cooled Fast Reactors 1969. *USAEC Doc. WASH-1089*
40. Perry, A. M., Bauman, H. F. 1970. *Nucl. Appl. Tech.* 8:208-219
41. Robertson, R. C. et al 1971. Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor. *USAEC Doc. ORNL-4541*
42. Robertson, R. C. 1965. *USAEC Doc. ORNL-TM-278*
43. Haubenreich, P. N., Engel, J. R. 1970. *Nucl. Appl. Tech.* 8:118-136
44. Briggs, R. B. 1972. *React. Tech.* 14: 335-342
45. Rosenthal, M. W. et al 1965. *USAEC Doc. ORNL-3686*
46. An Evaluation of Heavy-Water Moderated Organic-Cooled Reactors

1968. *USAEC Doc. WASH-1083*
47. Kasten, P. R. 1970. *At. Energ. Rev.* 8:473-534
48. The Use of Thorium in Nuclear Power Reactors 1969. *USAEC Doc. WASH-1097*
49. Naudet, R. 1966. *Conf. Utilization of Thorium in Power Reactors*, 77-90. Vienna: Int. At. Energ. Agency
50. Lewis, W. B. 1968. At. Energ. Can., Ltd., *Doc. AECL-3081*
51. Gas-Cooled Reactor Coolant Choice 1959. *USAEC Doc. ORNL-2699*
52. Carlsmith, R. S., Podeweltz, C. M., Thomas, W. E. 1965. *USAEC Doc. ORNL-TM-1112*
53. Jaye, S., Fischer, P. U., Lee, D. H., Jr. 1965. *GA-6367*
54. Johnston, T. A. 1970. *GA-10068*
55. Shepherd, L. R. 1971. *A/Conf. 49/P-738*
56. Mattick, W., Müller, H. W., Kramer, H., Schulten, R. 1971. *A/Conf. 49/P-370*
57. GCRP Semiann. Progr. Rep. 1966. *USAEC Doc. ORNL-4036*
58. GCRP Semiann. Progr. Rep. 1967. *USAEC Doc. ORNL 4133*
59. Naval Nuclear Propulsion Program (1969), Hearings Joint Comm. At. Energ., April, 1969, 97-113, 199-228
60. Performance Potential of Large Seed Blanket Breeder Reactors 1965. *USAEC Doc. WAPD-LSBR-300*
61. AEC Auth. Leg. 1966. Hearings Joint Comm. At. Energ., March, 1965 Pt. 3:1785-1827