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THE TECHNICAL AND ECONOMIC CHARACTERISTICS OF THORIUM REACTORS*

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THE TECHNICAL AND ECONOMIC CHARACTERISTICS OF THORIUM REACTORS

Programs for the development of thorium fuel cycles and reactors which use thorium are most often justified by the argument that better "fuel utilization" is obtained with thorium than with uranium. Numerous analyses of expanding nuclear power economies have shown that less uranium ore must be mined using thorium converter reactors which are started-up with U^{235} and the bred uranium recycled than using uranium converters with recycle of plutonium. The argument for the use of thorium is extended further by the contention that thorium converters can develop into "break-even" breeders, or even breeders that produce enough excess fuel to inventory other reactors.

There is no doubt as to the importance of conserving fuel resources, but other factors must also be considered to determine whether thorium systems should be developed and which thorium reactor types are most likely to be built. These relate to the economics of thorium systems and to the technical difficulties that lie in the way of their development.

Two points should be made about economics. First, unless restraints are applied, power system operators are going to select the reactor type that they believe will yield the lowest power cost. In addition, they are going to operate at conditions producing the lowest cost, although the fuel utilization of a reactor can always be improved at the expense of higher cost. Conservation of fuel resources may be an important national or world objective, but it will be decisive only if it affects the economics of a reactor of if it is imposed by governmental constraint. Of course, in evaluating economics, future conditions must be properly weighted and attitudes of governmental agencies, as reflected in the regulation of financing and rates, or in tax provisions, taken into account.

The second point is the converse of the first. Some thorium reactors may have lower power costs than competing uranium reactors, and those that do may be developed and built even if their fuel utilization is not particularly favorable.

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Technical factors which determine whether a reactor system can actually perform as assumed in the analysis of fuel utilization and economics are of obvious importance. But the question is not just whether a system can be developed to perform as assumed, but also when the postulated performance can be achieved in full-sized plants. In addition, it must be recognized that before the calculated economics can be achieved, the fuel processing and fabrication industry postulated in the economics analysis must exist. One who is considering developing a new system that looks attractive, of course, must also consider the cost of development.

The questions regarding development are particularly applicable to thorium systems. Most of the reactors which appear attractive with thorium are less well developed than the competing uranium systems. In addition, the presence of U^{232} in recycle fuel requires that special fuel fabrication facilities be built; and in some cases, new head-end processing methods must be developed and the processing facilities built. Thus, in the United States at least, the thorium converters appear to be caught in a squeeze. On the one hand, the status of development of the thorium reactors is such that they will not be brought into operation in any quantity before the early 1970's, and by then the pressurized and boiling light water reactors will have a considerable head start. On the other hand, there are large programs directed towards the development of fast plutonium-uranium breeders, and the postulate of the United States program is that these reactors will reach the point of economic attractiveness before 1990. It would appear, therefore, if thorium converters are to play an important role, they must become competitive with other converters and be built in quantity in the 1970's. Thermal breeders with low fissile inventories, however, may be able to enter the competition later. If their costs are favorable relative to fast reactors, the period in which they are attractive could extend far into the future.

In view of the foregoing, a study was undertaken at ORNL to estimate the performance and power costs that will be achieved with various proposed thorium reactors if they are built in the 1970's. The design information was taken from conceptual designs of 1000 Mwe reactors provided by developers of the concepts. A set of economic factors that represent the

average conditions over the life of reactors built during that period was defined and physics performance, capital, operating, and fuel cycle costs for all of the reactors were estimated on a consistent basis. For comparison with the thorium reactors, similar estimates were made for a 1000 Mwe light-water moderated pressurized-water reactor. Although we attempted to make our cost estimates correct on an absolute basis, the emphasis in all of the estimates was on accuracy of the costs of the various systems relative to each other.

To determine that the reactors could attain the performance postulated in the design and to identify any development that is required, the design and predicted engineering performance of the reactor plants were reviewed and evaluated. Taking into consideration differences in design requirements revealed by this review, normalized estimates were made of the capital costs for all the plants.

In the sections which follow, the reactors considered are described briefly and their engineering performance requirements discussed, the economic context for the evaluation is outlined, some general comments are made on the physics of thorium reactors, the fuel utilization of specific reactors is presented, and the fuel cycle, capital, and operating costs of the reactor are reported.

Description of Reactors

Six thorium-fueled reactors have been evaluated, along with a large pressurized water reactor which serves as a basis for comparison. We have not attempted to design reactors ourselves but have taken designs proposed by groups who are the developers or proponents of the systems. The technical characteristics of the reactors are summarized in Table 1.

The Pressurized Water Reactor (PWR) design is based on a Westinghouse study[1,2] of a 1000 Mwe reactor and differs from the Westinghouse concept mainly in the use of Zircaloy fuel cladding rather than collapsed stainless steel. A 15.8 ft inside diameter by 11.2 in. thick reactor vessel contains the core, and the peak-to-mean power density is 3.3.

Table 1. Reactor Design Characteristics

Item	Pressurized Water	Spectral Shift Controlled	Heavy Water	High Temperature Gas Cooled	Seed Blanket	Molten Salt Converter	Molten Salt Breeder
Net electrical capacity, Mwe	1,002	1,000	1,010	1,008	1,000	1,000	1,000
Reactor power, Mw (thermal)	3,220	3,270	3,870	2,270	3,200	2,250	2,250
Steam temperature/pressure, °C/Atm.	257/43	254/40	241/33	565/238	247/37	538/238	538/238
Net station efficiency, %	31	31	26	44	31	44	44
Moderator	H_2O	D_2O-H_2O	D_2O	Graphite	H_2O	Graphite	Graphite
Coolant	H_2O	D_2O-H_2O	D_2O	Helium	H_2O	Fuel salt[a]	Fuel salt[a]
Coolant temperature, inlet-outlet, °C	285-314	279-312	264-304	382-800	271-306	566-732	566-732
Coolant pressure, atmospheres	140	150	130	30	140	7	7
Control method	Soluble poison	Moderator mix	Continuous fueling	Poison rod	Seed position	Continuous fueling	Continuous fueling
Core diameter x height, meters	3.9 x 3.4	3.1 x 3.3	7.9 x 4.6	9.5 x 4.8	5.3 x 2.2	5.1 x 6.3	2.6 x 2.6
Specific power, kw/kg fertile	32	49	48	27	17-24	40	---
Core power density, kw/ℓ	81	130	16	7	68	17	160
Fuel elements	Rod bundle	Rod bundle	Concentric tubes	Graphite cyl.	Rod bundle	Molten salt	Molten salt
Fuel rod or tube diam., in.	0.32-0.38	0.42	1.34, 2.68, 3.83	4.5	0.25, 0.33, 0.74	---	---
Fuel	UO_2	UO_2-ThO_2	UO_2-ThO_2	UC_2-ThC_2	UO_2-ThO_2	UF_4-ThF_4-LiF-BeF_2	UF_4-LiF-BeF_2
Fuel management	1/3 scatter refueling	Batch	On-line refueling	1/12 scatter refueling	Batch	Continuous	Continuous

[a]Circulated through external heat exchanger.

The Spectral Shift Control Reactor (SSCR), which utilizes a mixture of light and heavy water as the moderator and coolant, is based on a design[1] by Babcock and Wilcox. In this concept, the entire core is refueled at one time; fuel depletion and fission product build-up are compensated by increasing the H_2O/D_2O ratio to reduce the probability of neutron capture in thorium resonances. Twelve zones having different uranium-to-thorium ratios are used to flatten the power distribution, thus reducing the peak-to-average to 1.9 from the higher value of the PWR. This permits a high power density, and the SSCR reactor vessel size is reduced from that for the PWR.

The Heavy Water Reactor (HWR) is a pressure-tube type based on a design study[1] by E. I. du Pont de Nemours and Company (Savannah River Laboratory). Pressurized heavy water coolant passes through 688 Zr-2.5% Nb pressure tubes which have an inside diameter of 4.43 in. and are 0.16 in. thick. Concentric tubular fuel elements are used, and the reactor is refueled while at power.

The Seed-Blanket Reactor (SBR) is based on the movable-fuel concept proposed by Bettis Atomic Power Laboratory.[3,4] In this concept, fully-enriched uranium is located in annular seed regions distributed throughout a thorium blanket. The SBR is controlled by axial movement of portions of the seed, which changes the effective thickness of the seed annulus. This results in a change in the leakage of neutrons from the highly reactive seed regions into the subcritical blanket regions, thereby providing criticality control. Control poisons are thus eliminated, and the neutron economy improved.

In the "converter recycle" concept considered here, the seeds are initially fueled with U^{235}, and U^{233} is recovered from the blanket. After about three converter cycles using U^{235} seeds, the accumulated U^{233} is used to fuel the core. According to Bettis, self-sustaining recycle is achieved afterwards with no additional fuel makeup required.

The design evaluated here has 61 seed modules, and the reactor vessel is a 28-ft inside diameter by 9-in. thick sphere.

The High-Temperature Gas Cooled Reactor (HTGR) is a helium-cooled, graphite and BeO moderated reactor based on the "TARGET" concept[1,5] of General Atomics. Graphite fuel elements contain loose pyrolytic-carbon coated fuel particles in holes located in a "phone-dial" arrangement. Two sizes of particles are used, one containing only uranium and the other only thorium. The two types of particles are separated at the end of the cycle and processed separately. Material from the uranium particle is sold or discarded to reduce the buildup of U^{236} in the system. Bred uranium recovered from the thorium particle is combined with makeup U^{235} and refabricated into the uranium-only particles for a subsequent fuel cycle. Thus, the makeup U^{235} is never recycled and bred uranium is recycled for only one pass. A BeO spine in the center of the graphite body has a volume that gives a carbon-to-Be atomic ratio of 2.4.

Use of a 56-ft inside diameter pre-stressed concrete reactor vessel that contains the core, the steam generators, and the blowers is a major design feature of the plant.

The Molten Salt Converter Reactor (MSCR) is essentially a scaled-up design[6] of the MSRE now in operation at Oak Ridge. It consists of a Hastelloy-N vessel filled with graphite moderator. A fuel salt, consisting of fluorides of uranium, thorium, lithium, and beryllium, and melting at about 475°C, is circulated through passages in the core graphite and then through an external heat exchanger. A side stream of the fuel salt is processed continuously for fission product removal in a fluoride volatility and vacuum distillation processing plant integrated with the reactor.

The Molten Salt Breeder Reactor (MSBR) is similar to the MSCR, except that the fissile and fertile materials are maintained in separate streams. The fertile stream not only circulates through the core graphite, but also surrounds the core forming a large blanket. The fuel stream circulates through the core within graphite tubes. The two-stream design makes possible a smaller core, lower neutron leakage, and decreased neutron losses to protactinium.

Engineering Performance and Status of Development

Evaluation of the thermal designs of the solid-fuel reactors indicated that all appear to be adequate, with the reservation that the power peaking factors were not checked by physics calculations and the enthalpy rise factor of 1.8 claimed for the pressurized water reactor has not yet been demonstrated. With regard to fuel element requirements, the PWR and SSCR appear to be designed on the basis of current technology. The clad on the blanket rods of the SBR appears not to be "free-standing", but by proper selection of dimensions it can very likely be made to operate satisfactorily.*

The remaining reactors are designed on the basis of extrapolations of fuel element technology well beyond existing data. Tests of the nested fuel elements for heavy-water reactors have not progressed to the burnups required at the specified heat ratings, and neither the largest size tubes nor assemblies of tubes have yet been tested. The HTGR concept requires further testing of coated particles of large sizes and to high burnup, and demonstration of the operation of graphite fuel elements for a six-year life.

The PWR, SSCR, and SBR all require large heavy-walled pressure vessels that are beyond present experience, but the problems associated with their construction appear to be solvable. The HWR uses Zr-2.5% Nb alloy pressure tubes on which corrosion and creep information are favorable but which require excessive extrapolation to predict a 30-year behavior. An on-line refueling machine specified for the heavy water reactor will require high reliability under severe conditions. Adequately low losses of heavy water from the SSCR and HWR can probably be achieved, but care with closures, seals, and leakage recovery systems will impose some burdens during operation and maintenance.

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*It was announced recently that Zircaloy-clad ZrO_2-diluted UO_2 rods typical of seed elements for long-life SBR cores have failed in irradiation experiments, but we do not yet know the details of the failures and their applicability to the shorter life breeder cores.

Feasibility of the particular HTGR design proposed depends largely on successful demonstration of the prestressed-concrete reactor vessel that houses the entire primary system, including the core, helium-circulating blowers, and steam generators. While French and British experience with concrete reactor vessels is applicable, and both the French and British have "all-inside" plants under construction, successful performance of vessels of the high temperature gas cooled reactor type remains to be demonstrated.

A preliminary analysis of plant availability indicates that a high degree of reliability will be required of all reactor equipment. Because of the scheduled down-time required for the reactors which are shutdown for refueling, only a few percent of the operating time will be available for unscheduled equipment outages if those plants are to have an availability of 90%.

The state of development of the systems and components required for the operation of the reactors is far from equal. The light-water-cooled reactors have the most extensive operating experience to call on. Heavy-water reactors have the advantage of NPD, CVTR, PRTR, and HWCTR experience, with the operation of the 200-Mw(e) CANDU reactor anticipated soon. Until operation of the Peach Bottom, Dragon, EGCR, and Oldbury stations is under way, the technology of the HTGR must be considered the least firmly established among the solid-fueled reactors.

The molten salt reactors are in a separate line of development, represented at present by two small reactor experiments, but no full-scale plants. The feasibility of the molten salt concept was demonstrated in 1954 with the Aircraft Reactor Experiment, a 2.5 Mwt reactor with a 900°C maximum temperature. The Molten Salt Reactor Experiment, which went critical on June 1, 1965 is a 10 Mwt, graphite-moderated reactor operating at 680°C, and is designed to demonstrate the compatibility of the construction materials, Hastelloy-N and graphite, with the molten fluoride fuel, as well as the operation and maintenance of the pumps and other components.

The MSCR is essentially a scale-up of the Molten Salt Reactor Experiment but is designed for a somewhat higher temperature, 730°C, and for

a higher power density. The assumed on-line processing has not been demonstrated, but the extension of the well-developed fluoride volatility process[6] should be straightforward. A process for removing rare-earth fission products from the fuel by vacuum distillation has been tested on a laboratory scale. Rapid removal of xenon will be tested in the Molten Salt Reactor Experiment and uncertainties regarding the behavior of some other fission products clarified. The plant design is based on use of an intermediate coolant between the fuel salt and the steam system, but all of the coolants considered present some design or development problems.

The MSBR involves several features that are still under development. The use of graphite tubes to separate the fuel stream from the fertile stream has not been demonstrated, although a method has been developed for making brazed graphite-to-metal joints, remotely if necessary. Replacement of the tubes will be possible, but a design which allows adequate tube life to be obtained in spite of graphite shrinkage will be required. Another engineering problem is the close-coupling of the primary heat exchanger to the reactor that is required to minimize the fuel inventory.

Economic Context and Cost Factors for the Comparison of Reactors

As noted earlier, the period of interest for the introduction of thorium reactors appears to be the 1970's. We have, therefore, defined a set of economic assumptions that would represent the average conditions over the 30-year life of a reactor concept if a number of reactors of that design were built in the United States between 1970 and 1980. These are as follows:

1. The technology used in the reactor and power plant designs was, in general, restricted to that which would be available for smaller, prototype reactors to be built for startup in 1970.

2. Private ownership of the reactors, the fuel, and of the fabrication and processing plants was assumed as the reference case, and annual charges were taken as 10% on fuel, heavy water, and Li^7, 12% on the reactor plant, and 22% on the fuel fabrication and processing plants. Alternate plant fixed charges of 7% and inventory costs of 5% were used to represent public ownership.

3. Fuel fabrication and processing costs are based on single-purpose plants, each capable of serving 15,000 Mwe capacity of the concept under consideration. Because the plant size is specified in terms of electrical capability, the capacity in tonnes/day is different for each fuel burnup.

4. Levelized fuel cycle costs were computed by determining the "present worth" of all expenditures and receipts during a reactor lifetime of 30-years using a discount factor of 6%. The reactor plant availability was required to be 0.9 in. in order to achieve an average plant factor of 0.8 over the plant life. The unit costs of fuel fabrication and processing were held constant throughout the reactor life.

5. Assumed costs of nuclear materials are \$17.60/kg U_3O_8 and \$30/kg separative work (\$12/gm fully enriched U^{235}), \$11/kg ThO_2, \$14/gm U^{233}, \$10/gm fissile Pu, \$37.40/kg D_2O, and \$0.12 /gm Li^7.

6. The thorium reactors are fueled initially with fully enriched U^{235} and the bred uranium subsequently recycled with fully enriched U^{235} makeup as required. Plutonium from the PWR was assumed to be sold at \$10/gm, which studies have indicated gives about the same fuel cycle cost as if the plutonium were recycled.

7. The times for fuel fabrication, shipping, post-irradiation cooling, etc. were estimated for each case. Ownership of fuel during fabrication and processing was assigned to the reactor plant, and the inventory charge includes the full out-of-pile inventory. Shipping costs were estimated for each fuel element design and irradiation condition assuming that the fabrication and processing plants are located at a site 1000 miles from the reactor.

The ground rules and procedure followed are similar to those stated in more detail in reference 1, with however, some changes in the values of materials.

Physics of Thorium Cycle Reactors

The outstanding characteristic of thorium cycle thermal reactors is the favorable neutron economy which is obtained using U^{233} as the fissionable material. This results largely from the value of η, the number of neutrons produced per neutron absorbed, which at 2200 m/sec is 2.29 for U^{233} but only 2.07 and 2.10 for U^{235} and Pu^{239}. In the thermal spectrum of an actual reactor the η's of all three nuclides will be less than the 2200 m/s values because of the necessity of elevated moderator temperatures and because of hardening of the neutron energy spectrum. A significant fraction of neutron absorptions occur at epithermal energies where the values of η for all of the nuclides decrease. The effect of epithermal absorptions is generally least harmful with U^{233}, but there has been a troublesome uncertainty in the U^{233} epithermal cross sections. A cross section set derived from various differential cross section measurements, mainly fast chopper measurements, yields an average capture-to-fission ratio (alpha) of about 0.23. Integral epicadmium measurements by Halperin[7] at ORNL and by Feiner at KAPL[8] have indicated, however, that the average value of alpha in a 1/E spectrum above 0.5 ev is about 0.17. The weight of evidence appears to favor the lower value of alpha, and it has been used in this study.

Several other differences, less important than the value of η, occur in the comparison of the thorium cycle with the uranium cycle. One of these is that the fission product yield curve is shifted slightly toward lower mass numbers in the case of U^{233} fissions. This appears to give a favorable effect on the average absorption cross section, particularly in the vicinity of mass 149 where there are a number of high cross section nuclides, and the yield from U^{233} is only 0.0076 compared to the U^{235} yield of 0.0113.

An unfavorable and well-known aspect of the thorium cycle is the relatively long half-life (27 days) and high cross section (43 barns thermal and 925 barns resonance integral) of the Pa^{233} which is intermediate in the conversion of thorium to U^{233}. When a neutron is absorbed

in Pa^{233} the loss in conversion ratio is equivalent to two neutrons. In order that there be a favorable result in the competition between decay of Pa^{233} into U^{233} and absorptions in Pa^{233}, the flux in the regions containing thorium should not be too high. In most designs the specific power is kept lower than it would otherwise be for this reason. Small cores with breeding in external blankets or continuous movement of the thorium in and out of the high flux regions could also increase the fraction of Pa^{233} which decays into U^{233}.

The initial buildup of Pa^{233} and consequent delay in generation of U^{233} in a fuel element causes a rapid decrease in reactivity of thorium-cycle reactors during the first several half lives of the Pa^{233}. This reactivity is recovered when the Pa^{233} decays during a shutdown, and losses to the control poisons that are required to offset these "protactinium transients" can be important in a reactor which is shutdown frequently. A typical plot of reactivity vs burnup in Figure 1 shows the contrast of a U^{233}-thorium reactor with a partially enriched uranium reactor, which has an initial reactivity increase as the Pu^{239}, with its high fission cross section, builds up.

Although the ultimate goals of thorium reactor designs may include breeding, it is difficult to reconcile the other, often conflicting, demands of low fuel-cycle cost and low capital cost with the requirement of high enough neutron economy for breeding. Neutron leakage and parasitic captures in moderator, coolant, and structure can take an appreciable fraction of the available neutrons. The losses to fission products become important if the fuel fabrication and reprocessing costs require high fuel burnup. A further reduction in neutron economy comes from the fact that non-breeding systems must use U^{235} (or possibly plutonium) as makeup to supplement the bred U^{233}. The lower η of these isotopes plus the increased buildup of U^{236} and Np^{237} further reduce the breeding ratio. A rough rule-of-thumb is that the decrease in breeding ratio from any increased parasitic capture, such as in fission products, will be doubled when one takes into account the greater proportion of U^{235} in the recycled fuel. The use of thorium in non-breeding reactors serves principally to extend burnup and increase the utilization of fissile uranium.

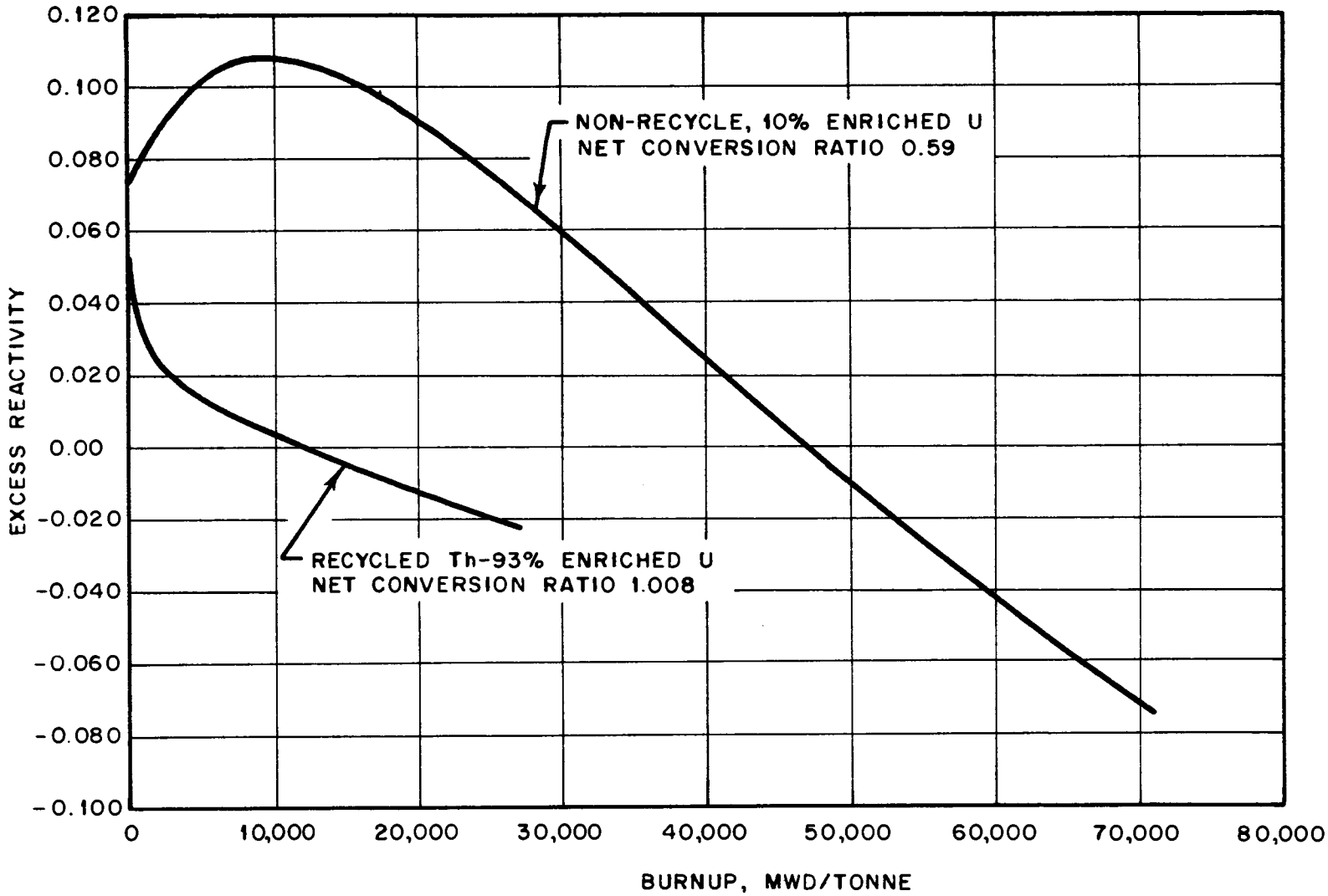

FIGURE 1. REACTIVITY CHANGES WITH BURNUP.

We have calculated the performance of the PWR, SSCR, HWR, and HTGR described earlier as to breeding ratios, burnups, and mass balances over the thirty year history. The isotopic depletion calculations were typically done with 15 energy groups using cross sections that were condensed from a 98 group set. The fuel management scheme recommended by the sponsor was used in the calculations with, in one case, as little as 1/12 of the core refueled at a time. An appropriate delay time was considered between removal of spent fuel from the core and recycle of the reprocessed fuel. Several cases were calculated for each reactor type with variation of fuel enrichment to determine the optimum burnup.

In the case of the seed blanket reactor, we performed no physics calculation ourselves for the specific design considered but used values provided by Bettis. However, we previously analysed an SBR core which was designed to have a breeding ratio greater than 1.0 and found that starting with pure U^{233}, it was capable of operating for over two years with no net consumption of fuel. Our physics calculations for the MSR's were made using an automated computer program that determined the economic optimum design and computed the physics performance and fuel cycle cost at equilibrium conditions. Thirty-year-average values were obtained by correcting the equilibrium results for the time variation in nuclide concentration assuming that the reactors are started with U^{235}.

Results are given in Table 2 for near-optimum cases of each reactor type.

The SSCR has the lowest breeding ratio of the thorium reactors. This is attributable to the low η in the undermoderated condition of the first part of the cycle and the parasitic captures in hydrogen in the last part of the cycle. The HWR gives the next higher breeding ratio in this comparison. The use of heavy water for both coolant and moderator can result in very good neutron economy since there are virtually no absorptions in the heavy water. The fact that the breeding ratio for the HWR was only 0.84 is caused by the parasitic absorptions in the pressure tube and by the economic necessity of reducing heavy water inventories. HWR's could give breeding ratios above unity if the economics favored such a design. The neutron economy of the HTGR is hurt slightly by the absorptions in the

Table 2. Reactor Performance

Concept	Feed Enrichment[a]	Exposure (Mwd/T of U + Th)	Breeding Ratio[b]	Index of Fuel Consumption[c]
PWR	2.2	21,000	0.60	1.29
SSCR	4.0	25,400	0.75	0.80
HWR	2.1	28,800	0.84	0.60
HTGR	2.1	52,300	0.90	0.22
SBR	1.4	12,000	1.00	0.00
MSCR	2.6	2,300	0.96	0.09
MSBR	(d)	(d)	1.06	-0.13

[a]Percent by weight, based on total heavy metal.

[b]Gross estimated for last cycle of 30-year reactor life.

[c](1.0 - breeding ratio)/thermal efficiency, a measure of relative fuel consumption.

[d]Not applicable because fertile and fissile streams have different cycle times.

graphite and BeO moderator, but is is helped by the (n, 2n) reaction in the beryllium and by a device of keeping the bred fuel separate from the makeup U^{235} and recycling only the bred fuel. After being fueled with U^{233}, the SBR has the highest breeding ratio of the solid fuel element reactors. It achieves a breeding ratio of 1.0 in light water by having a high moderator content in the seed for thermalization where most fissions occur and having a low moderator content in the blanket to reduce parasitic losses there.

The MSCR has good neutron economy by virtue of using graphite moderator and continuously stripping out the volatile fission products so that there is little poisoning from xenon. The MSBR also used graphite moderator and continuous stripping of volatile fission products. In addition it has separate fertile and fissile fuel streams and is able to minimize neutron losses to Pa^{233} by allowing most of the Pa to decay in low flux regions.

The cumulative ore requirements for inventory and makeup of the reactors have been computed and are shown in Figure 2. In each case the nuclear power industry of the U.S. was assumed to consist entirely of the reactor concept considered, and the power growth rate was taken from the estimate of the November 1962 "AEC Report to the President". The ore commitment at a specified date was taken to include the inventory in the operating reactors and associated processing plants as well as the makeup which would be required for existing reactors over the remainder of a 30-year life. In the case of the MSBR, however, the ore requirement at a particular time was not reduced by bred fuel to be produced in the future.

By the measure used in Figure 2, the ore requirement for the SBR over the next 50 years could be as high as that of a converter which has a lower conversion ratio but also a smaller fuel inventory. Hence comparison on this basis is somewhat unfair to the SBR, since at the same ore commitment it would actually consume less fuel than a converter. On the other hand, the converters were not optimized for high fuel utilization, and higher conversion ratios could be achieved in them at the expense of higher power cost.

An example of the breeding ratios that can be obtained when a thorium cycle reactor is designed for maximum breeding is shown in Figure 3. The calculations are for the HTGR described earlier, operating at an average core power density of 7 w/cm^3. If one assumes that all fission products are to be retained in the fuel until it is reprocessed (the current design aim) the maximum breeding ratio is 1.07 after deducting processing losses. If, instead, a fuel were designed to release instantaneously all of the volatile fission products (the noble gases and halagons) the maximum breeding ratio would be 1.11. The release of volatile fission products plus the lithium formed in the (n, α) reaction in the beryllium would give a maximum breeding ratio of 1.12. The corresponding doubling times, assuming an out-of-reactor holdup time for reprocessing and refabrication of 60 days, are 55 years, 30 years, and 24 years. Since complete instantaneous fission product release from solid fuel elements is probably not achievable, the maximum attainable breeding ratio for this concept is probably somewhere between 1.07 and 1.12.

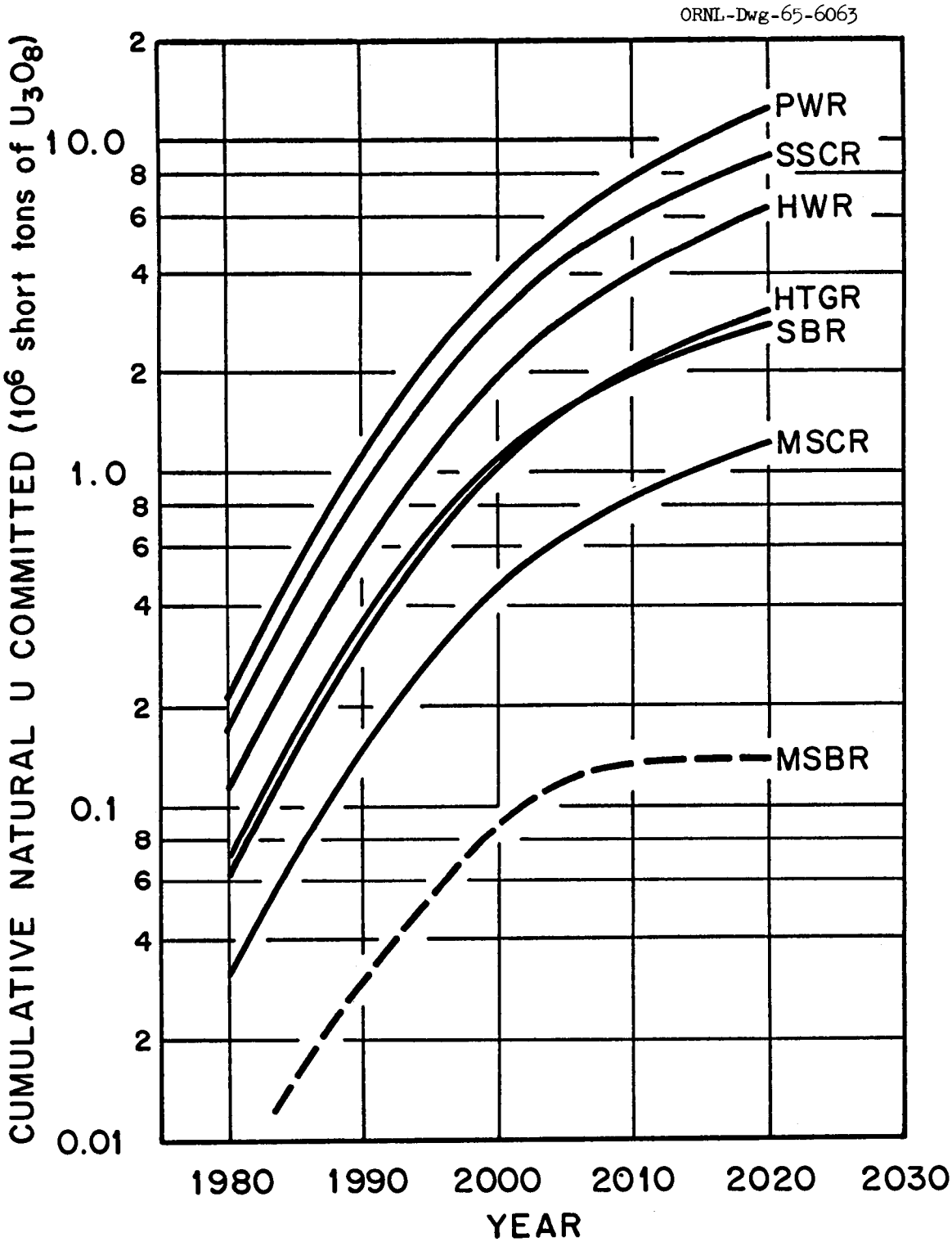

FIGURE 2. NATURAL URANIUM COMMITTED AS A FUNCTION OF TIME.

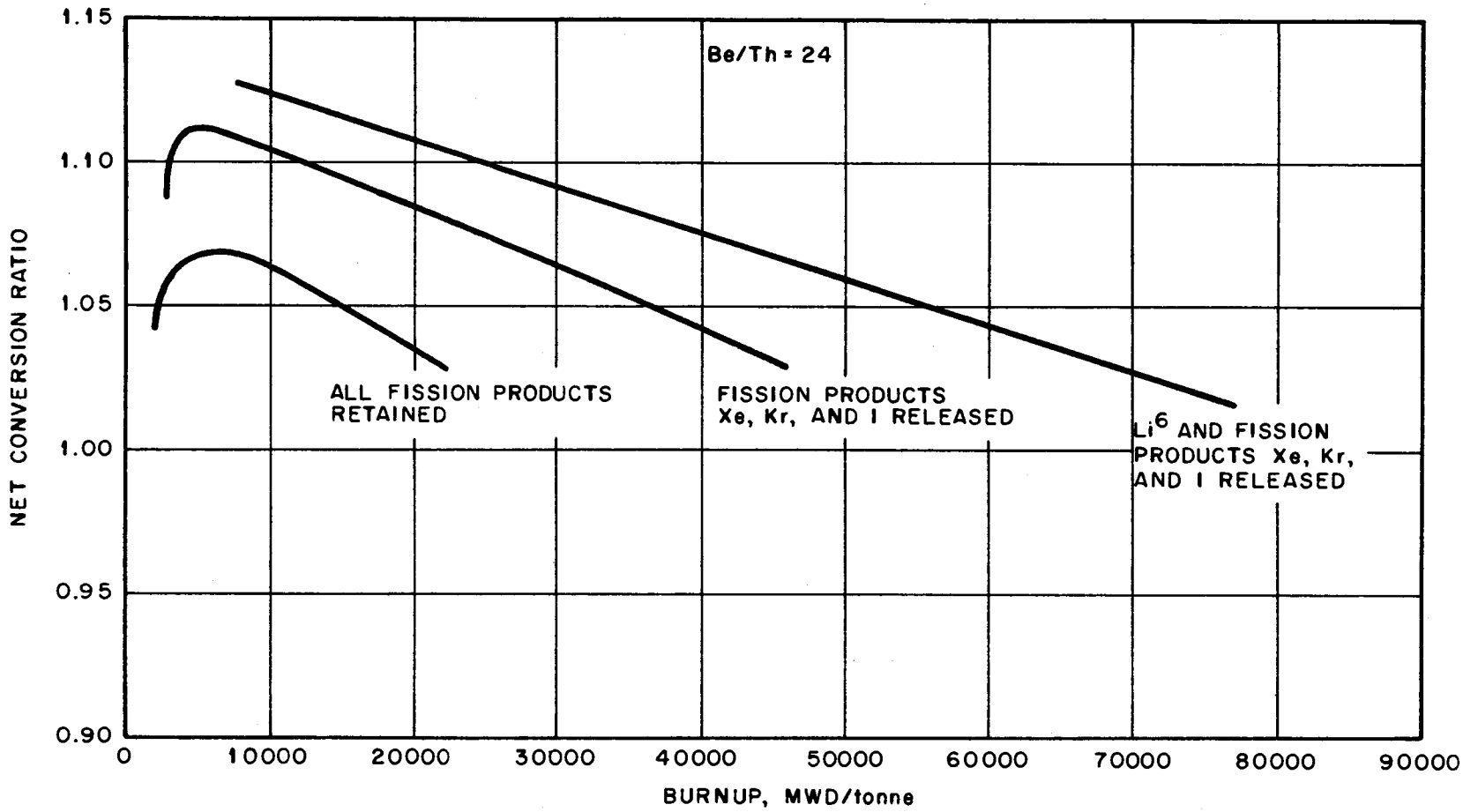

FIGURE 3. BREEDING RATIO FOR HTGR.

It should be kept in mind that the low fuel burnups for the cases shown in Figure 3 would give excessive fuel-cycle costs unless extremely cheap processing and fuel fabrication procedures which have low fuel holdup were developed. For this reason, fluid fuel reactors appear to be particularly apt candidates for thermal breeders. The MSBR referred to in Table 2 has, for example, a breeding ratio of 1.06 with a low fissile inventory (0.7 kg fissile U/Mwe) and is thereby able to achieve a system doubling time of 10 years.

Fuel Cycle Costs

Fuel fabrication, processing (including ultimate waste disposal), and shipping costs were estimated for each reactor as functions of plant throughput using the methods described in reference 1. For the PWR, SSCR, HWR, and HTGR, these unit costs were combined with the thirty-year mass balances and, using present worth accounting, levelized fuel cycle cost was obtained as a function of fuel exposure. This procedure yielded the minimum costs for the core designs considered, but it should be noted that dimensions, fuel-to-moderator ratio, plant design conditions, etc., were those specified by the reactor designers, and such factors were not varied.

In the case of the seed blanket reactor, we used only the single set of mass balances provided by Bettis and no optimization was done. As mentioned earlier, calculations for the MSR's were made using a computer program that selected the optimum design and computed the equilibrium fuel cycle cost. Thirty-year average costs were obtained by correcting the equilibrium results for the time variation in nuclide concentration assuming that the reactors are started with U^{235}.

A near-optimum example of the operating conditions and fuel cycle cost breakdowns for each concept is presented in Table 3. The inventory costs reported in the table include the fuel held up during processing and fabrication.

Table 3. Thirty-Year-Average Fuel Cycle Costs for Reference Economic Factors

Reactor	PWR[a]	SSCR	HWR	HTGR	SBR	MSCR	MSBR
Equilibrium or last-cycle data							
Exposure, Mwd/MT of fuel	21,000	25,400	29,400	52,300	12,000	2300	---
Fuel lifetime, full-power years	1.83	1.46	1.64	4.95	2.18	0.21	0.08
Feed enrichment, wt % fissile material	2.2	4.8	2.1	3.1	1.4	2.6	---
Fabrication and processing plant size for 15,000 Mw(e), MT of fuel per year	670	555	590	189	1,300[c]	(f)	(f)
Fabrication cost, $/kg of fuel	52.50	47.20	37.50	115.00	51.50[d]	0	0
Processing cost, $/kg of fuel	31.20	39.00	39.70	105.00	23.00[e]	---	---
Fuel shipping costs (fabricated + irradiated), $/kg of fuel	4.00	8.00	5.90	25.60	5.00	0	0
Out-of-pile holdup time, days	310	360	330	390	290	---	---
Initial fissile inventory, kg	2,060	2,910	1,450	2,910	3,500	1750	620
Value of first core loading, $\$10^6$	15.05	35.82	18.33	36.00	43.82	21.00	7.44
Fabrication cost of first core, $\$10^6$	5.44	3.20	3.00	16.40[i]	8.60	1.30[g]	4.39[g]
Fuel-cycle cost, mills/kwhr(e)							
Fabrication	0.34	0.25	0.22	0.26	0.61	0.04[g]	0.08[g]
Net fuel burnup and losses[b]	0.71	0.60	0.40	0.19	0.12	0.07[h]	-0.08[h]
Processing	0.20	0.20	0.24	0.19	0.27	0.08	0.12
Shipping	0.03	0.04	0.04	0.05	0.05	0.00	0.00
Fissile and fertile inventory	0.25	0.72	0.43	0.51	0.95	0.37	0.17
Interest on fabrication	0.08	0.03	0.05	0.17	0.08	0.02[g]	0.06[g]
Interest on processing	-0.01	-0.01	-0.01	-0.03	-0.03	0.00	0.00
Net fuel-cycle cost	1.60	1.83	1.37	1.34	2.05	0.58	0.35

[a]Fueled with partially enriched uranium.

[b]Losses = 1% in processing, +0.2% in fabrication.

[c]For breeder, pre-breeder throughput is 860 MT/year.

[d]For breeder, pre-breeder costs, $64/kg.

[e]For breeder, pre-breeder costs, $30.50/kg.

[f]Integrated processing.

[g]Costs associated with carrier salt, including Li^7.

[h]Losses = 0.1% per pass through processing

[i]Includes cost of initial purchase of BeO.

The costs for the solid-fuel reactors in Table 3 reflect the large capacity fabrication and processing plants assumed for this study. Costs associated with fabrication and processing range only from 26% of the total fuel cycle cost for the SSCR to 45% for the SBR. An effect of lowering the importance of the costs for processing and fabrication is to increase the importance of the costs associated with owning and consuming uranium, particularly for a 10% inventory charge. Consequently, the fuel cycles generally tend to optimize at lower burnups than those cited by the concept sponsors.

The HTGR achieves the lowest fuel cycle costs of the solid-fuel reactors by virtue of its good neutron economy and long fuel life, and its high thermal efficiency gives it a special advantage over the water cooled reactors with which it is compared. Still lower costs--down to perhaps 1.2 mills/kwh--could be achieved in the HTGR if the BeO were eliminated, but the breeding ratio would be lower and the fuel utilization poorer.

Good neutron economy and low fuel fabrication costs assist the HWR in maintaining low fuel cycle cost (if heavy water cost is not included). While the PWR has somewhat higher burnup costs than the SSCR, the difference is more than offset by the lower inventory cost of the PWR that results from use of partially rather than fully enriched uranium.

The high fuel cycle cost of the SBR results from a combination of large fissile inventory and short core life. The SBR was not optimized for minimum power cost as were the other reactors in the table but was designed specifically to obtain a breeding ratio above 1.0 after accumulation of a core loading of U^{233}. Probably lower fuel cycle costs could be obtained if it were optimized for the ground rules used.

The MSCR achieves low fuel cycle costs through the good neutron economy it gets from xenon removal and continuous fuel processing, from essentially eliminating fuel fabrication cost, and from high thermal efficiency. Although the fuel processing plant for the molten salt reactors is at the reactor site and integrated with the reactor, the capital charges (at 12% per year) and operating costs for the processing plant have been isolated and reported in the fuel cycle breakdown as processing costs. By use of separate fissile and fertile streams, with a thorium

blanket surrounding the core, the MSBR achieves improved neutron economy over the MSCR and a lower fissile inventory. Low processing cost is obtained even with quite short cycle times by the use of fluoride volatility and distillation processes; the estimated cost of an integrated processing plant for the MSBR is less than $5,000,000.

Because the inventory and interest charges used in this study may not apply in many cases, and because the price of uranium ore may differ from the reference value used, the fuel cycle costs reported in Table 3 have been recalculated for different economic assumptions. The cycles have not been reoptimized for the changed conditions, however, and somewhat lower costs than those given could be achieved by reoptimization.

As seen in Table 4, doubling the reference cost of U_3O_8 (making it $35.2/kg) would increase the fuel cycle costs of the MSBR by 0.03 mills/kwh, the MSCR by 0.15, the HTGR by 0.24 the HWR by 0.29, the SBR by 0.37, and the PWR and SSCR by about 0.46 mills/kwh. Reducing the inventory and interest charge from 10 to 5% would benefit the SBR the most because of its large fuel inventory and benefit the MSR's and PWR the least because of their low inventory costs.

Power Costs

Capital and operating costs of the reactors were estimated with the help of Sargent and Lundy Engineers from information provided by the sponsors of the concepts. Since design assumptions varied from one concept to the next, common features or conditions of plants were made comparable before the costs were normalized. A plant design was not available for the SBR and its capital cost was taken as that of the PWR. The costs for the molten salt reactors were obtained by pertubation of the costs estimated for the sodium graphite reactor in reference 1. A change from reference 1 for all reactors is the use of total indirect costs equal to 32% of the direct costs rather than 41%; the lower value seems more consistent with costs reported for large plants.

Table 4. Effect of Variation in Economic Factors on 30-Year-Average Fuel Cycle Cost*

Reactor	PWR	SSCR	HWR	HTGR	SBR	MSCR	MSBR
Fuel cost for reference conditions, mills/kwhe	1.60	1.83	1.37	1.34	2.05	0.58	0.35
Effect of change in fuel inventory and fabrication interest charges to 5% per year, mills/kwhe	1.44	1.46	1.13	1.00	1.53	0.38	0.23
Effect of change in uranium ore cost, with bred fuel values changed in proportion to fully-enriched uranium, mills/kwhe							
At \$8.8/kg U_3O_8 (\$9.54/g U^{235})	1.30	1.55	1.20	1.19	1.83	0.47	0.33
At \$35.2/kg U_3O_8 (\$16.23/g U^{235})	2.07	2.29	1.66	1.58	2.42	0.73	0.38

*Values given are pertubations of the base cases without reoptimization.

Table 5 presents the estimated capital and operating costs and combines them with the fuel cycle costs given in Table 3 to obtain power costs. Values are given for both the reference economic condition and for inventory rates and fixed charges that are more appropriate for government ownership or for special financing and taxation arrangements. The cost of heavy water includes both inventory charges and an assumed 2% per year loss rate.

The low capital costs of the HTGR are attained largely through use of the prestressed-concrete vessel that contains the entire primary system. The SSCR cost is lower than that of the PWR because of savings resulting from the high core power density. Elimination of a mechanical fuel handling system, reduced control requirements, low system pressure, and small heat transfer areas in the heat exchangers kept the costs of the molten salt reactor from being high in spite of the high melting coolant and the intermediate heat transfer loop.

The capital cost values shown of course have considerable uncertainty associated with them. More confidence, however, can be placed in the costs of one concept relative to the others. An examination of the details of the estimates indicates that the capital cost of any of the solid fuel plants relative to the others could be as much as 0.1 mills/kwh lower than the values shown for the reference conditions. Only the HWR, the HTGR, and the molten salt reactors could be much higher than the estimates: as much as 0.1 mill/kwh in the case of the HWR, and up to 0.2 mills/kwh for the HTGR (largely because of uncertainties about the concrete reactor vessel). The cost estimates of the molten salt reactors are more uncertain than the others because less is known about the designs. Since much of the equipment in power plants is similar for all concepts, and since we have tried to be conservative in making the estimates, it would not appear that the capital costs of the MSR's could be underestimated by more than 0.2 mills/kwh.

Although the same operating cost is used for all plants, there will clearly be some variations, but insufficient information is available at present to make a significant distinction.

Table 5. Contributions to Power Cost

Reactor Concept	PWR	SSCR	HWR	HTGR	SBR	MSCR	MSBR
Plant capital cost, $/kwe	124	116	126	110	124	123[a]	128[a]
Power cost with 12% fixed charges on reactor plant and 10% fuel inventory and fabrication interest charges (mills/kwhre)							
Capital	2.1	2.0	2.2	1.9	2.1	2.1	2.2
Operating	0.3	0.3	0.3	0.3	0.3	0.3	0.3
Fuel cycle	1.6	1.8	1.4	1.3	2.0	0.6	0.4
Heavy water	—	0.1	0.5	—	—	—	—
Total	4.0	4.2	4.4	3.5	4.4	3.0	2.9
Power cost with 7% fixed charges on reactor plant and 5% fuel inventory and fabrication interest charges (mills/kwhre)							
Capital	1.2	1.2	1.3	1.1	1.2	1.2	1.3
Operating	0.3	0.3	0.3	0.3	0.3	0.3	0.3
Fuel cycle	1.4	1.5	1.1	1.0	1.5	0.4	0.2
Heavy water	—	0.1	0.3	—	—	—	—
Total	2.9	3.1	3.0	2.4	3.0	1.9	1.8

[a]Excluding on-site processing facilities, for which charges are included under "Fuel Cycle Cost".

Conclusions

The results of our evaluation show that there are thorium cycle reactors which can have both lower power cost and better fuel utilization than a PWR. As often occurs in comparative evaluations, however, the systems which are the least developed or about which the least is known ---the HTGR and the MSR's---appear to have the lowest costs. Since our estimates are intended to represent what the costs would be if the reactors were successfully developed, they do not reflect uncertainties in the technology.

The molten salt reactors have the lowest power costs of all the reactors studied as a result of their low fuel cycle cost, but there are a number of design and development problems which need to be solved. The MSBR, in fact, appears to violate the ground rule that the technology used in the designs be limited to that which could be used in a prototype to go on line in 1970. However, the use of on-site processing for the MSR's does eliminate one economic uncertainty in that achievement of low fuel cycle costs would not await the growth of a large support industry. In spite of the development problems which need to be solved, the MSBR offers a particular virture in that it can breed and at the same time could have low enough power costs to make it attractive to power system operators.

Of the solid fuel reactors, the HTGR appears particularly promising, but both the graphite fuel elements and the "all-inside" concrete pressure vessel require demonstration in a power reactor. Thus while the HTGR concept is flexible and alternate design approaches are available, its success will depend on the favorable outcome of forthcoming tests of its important features and on creation on an adequate scale of the somewhat unique fabrication and processing facilities it requires.

The heavy water reactor evaluated in this study has good fuel utilization but its high power cost makes it unattractive. However, studies by others of using organic or boiling light-water coolants for HWR's have shown that appreciable reductions in capital costs, including the D_2O inventory, can be obtained. By a substitution of coolant the power cost of the HWR thus possibly could be lowered significantly from the value we

obtained. Since the SSCR offers neither lower power costs nor significantly better fuel utilization than the PWR and has the disadvantage of having heavy water in the coolant, its further development seems unlikely.

The SBR has good fuel utilization but high power cost. It is difficult to compare it with the other reactors, however, since it was designed for a particular breeding ratio and was not optimized for the conditions of our study.

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