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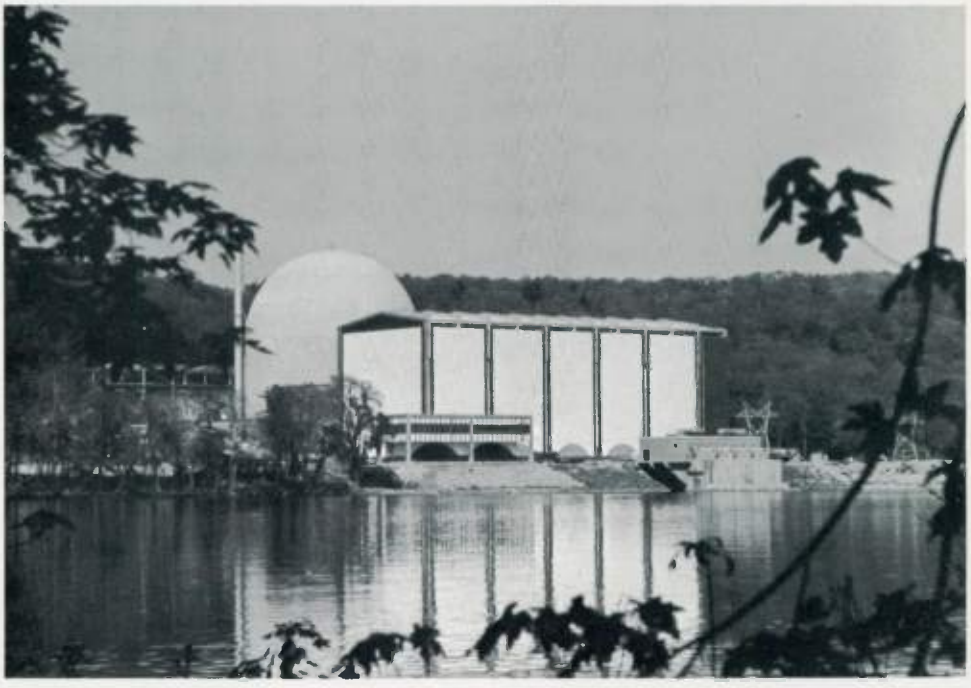

Seven years of progress in nuclear power is demonstrated by these two Yankee pressurized-water plants. In late 1960, the 185-MW prototype plant went into service at Rowe, Massachusetts, and helped demonstrate the economic feasibility of nuclear power.

The second Yankee plant, at Haddam Neck, Connecticut, went into commercial operation January 1, 1968 with an initial capacity of 490 MW. The Connecticut Yankee Atomic Power Company is sponsored by eleven New England electric utility companies.

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Cover design: More than half the electrical generation capacity contracted for in 1967 by the electric utility industry was nuclear. This rapid growth of the light-water reactor has emphasized the desirability of introducing the high-gain breeder reactor as soon as possible. The benefits of the breeder to the overall nuclear fuel cycle and the type of fast breeder required to provide these benefits are the subjects of this issue. The cover design by artist Tom Ruddy symbolizes the fast breeder reactor.

A Partnership Approach Will Assure Timely Development of the Breeder Reactor

Ten years ago the first industrial demonstration reactor plant achieved full power, the technical feasibility of commercial nuclear power was established, and the electric power industry was acclaimed "on the threshold of economic nuclear power." During the following ten-year period, that economic threshold appears to have been crossed; the electric utility industry now feels that the economic potential may be great enough to warrant nuclear plants for more than half of the generating capacity being ordered.

From the initial reactor rating of 60 megawatts at Shippingport, plants now going into service have ratings as high as 500 MW, and plants are being designed for service in the 1970's with ratings beyond 1000 MW. This rapid progress in nuclear plant size has been accompanied by corresponding reductions in the cost of nuclear power: light-water plants being sold today will generate electricity for less than five mills per kilowatthour. And the development of the light-water plant will not stop there. The research effort now going into water reactor technology ensures that water reactors will continue to improve.

The speed with which the water reactor has been transformed from technical feasibility to economic reality has been possible only because of the joint effort of the interested parties—the electric utilities, the Atomic Energy Commission, and the manufacturing industry. And now that rapid transition from technical to economic potential has brought the nuclear industry to another threshold—the threshold of the next generation of nuclear plants—much faster than anyone had predicted just a few years ago. That next generation is the high-gain breeder reactor which will make fissionable fuel faster than it burns it; thus, the breeder is an ideal working partner for water reactors. The urgency for establishing this reactor partnership soon is purely a matter of economics; breeder reactors should be operating in the 1980's to keep nuclear fuel cycle costs and operating expenses at their most attractive level. If developed on that timetable, the breeder can minimize nuclear fuel costs well into the next century; if development of the breeder reactor is delayed, a large economic penalty will result.

Unfortunately, the technology of the breeder reactor is much more difficult than that of the water reactor. Although development work on water reactors and breeders started together, the simplicity of the water reactor held the promise of an earlier economic payoff. Thus, more emphasis has been given to water reactor development, and breeder technology still lags far behind economic need. Although the basic technical feasibility of the sodium-cooled fast breeder reactor has been established, it remains to be demonstrated that the proper combination of parameters meeting the economic objectives will perform reliably in a total system.

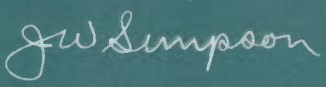
To overcome the technology gap that still exists, the same cooperative effort is now required on breeder development that so successfully launched the water reactor—a partnership of utilities, the Atomic Energy Commission, and the manufacturing industry. The justification for a joint development program is a matter of mutual economics, because the fast breeder venture is desirable for all: utilities will benefit from the low-cost fuel cycle that the breeder can provide, the public (represented by interested government agencies) will be guaranteed continuing and abundant low-cost electric power, and the manufacturing industry will be provided expanding market opportunities for new kinds of equipment.

No one beneficiary can be expected to carry the full burden of development—especially on a scale large enough to guarantee timely introduction of the breeder. In fact, even with full three-way participation, sacrifices will be required. For the electric utility and the manufacturer, the present-worth value of revenue and profits that will be earned 30 or 40 years from now (the length of time that will be required for the breeder industry to develop fully) is less than five cents on today's dollar—but the development expenses that must be incurred now are in today's dollars. On the other hand, in a free-enterprise economy the public can hardly be expected to bear the total burden of development and research for privately owned industry. Thus, a cooperative program is the only logical approach—a program in which all potential beneficiaries share the cost of risks of development, so that all can share in the gains that will eventually accrue.

To support this approach, Westinghouse has committed more of its resources, including capital, to breeder development than to any other single technological development. The Westinghouse approach is a three-phase program leading to construction of a sodium-cooled fast breeder reactor. This demonstration plant, with a rating of 200 to 400 megawatts, will be the prototype design for a full-scale 1000-MW plant. The program's first phase, which will continue until 1970, encompasses the study and research needed to commit the demonstration plant to detailed design; the second phase is plant construction, expected to take about five years; in the final phase, the plant will be operated to optimize the design and determine the technological and economic feasibility of the full-scale fast breeder plant.

This development program can be successful only if it receives the active support of both the utility industry and the Atomic Energy Commission. The first phase of the program, already under way, is receiving financial support from several utilities, some of which may also provide manpower. A number of other utility companies, along with the Atomic Energy Commission, have been invited to join the first phase. Similar cooperation will be required to complete the second and third phases.

There is little doubt that the breeder plant will be developed eventually—the ever-increasing needs for energy leave future generations no other choice. But the economic optimization of the development is a matter of timing, and it is this timing that is now of concern. The economic incentive for breeder development has arrived—for utilities, government, and manufacturers. Although it is possible that the fast breeder could be brought to economic fruition by individual effort, only a full partnership can make the possibility a probability.



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Only High-Gain Breeder Reactors Can Stabilize Uranium Fuel Requirements

J. C. Rengel

The rapid growth of water reactor capacity has accelerated the economic desirability of the high-gain breeder reactor. The breeder will be used in partnership with water reactors to supply base load, and will breed more fissionable fuel than it consumes for use in other reactors.

The present state of affairs in commercial nuclear power can be summarized briefly with two basic observations:

First, the growth of nuclear power in the electric utility industry over the past two years has exceeded all predictions. And second, because of this rapid growth, the long-range future of nuclear reactors can best be assured by taking steps now to stabilize future fuel cycle costs. Those costs are sensitive to the market price of uranium and plutonium. As will be shown, nuclear fuel cycle costs can be kept reasonably low by introducing the next generation of nuclear reactors—the breeders, which produce more fissionable fuel than they consume. A mix of tomorrow's breeders and today's burners (water reactors) can guarantee continuing low fuel cycle costs and at the same time provide the optimum economic generating plant combination for a utility system.

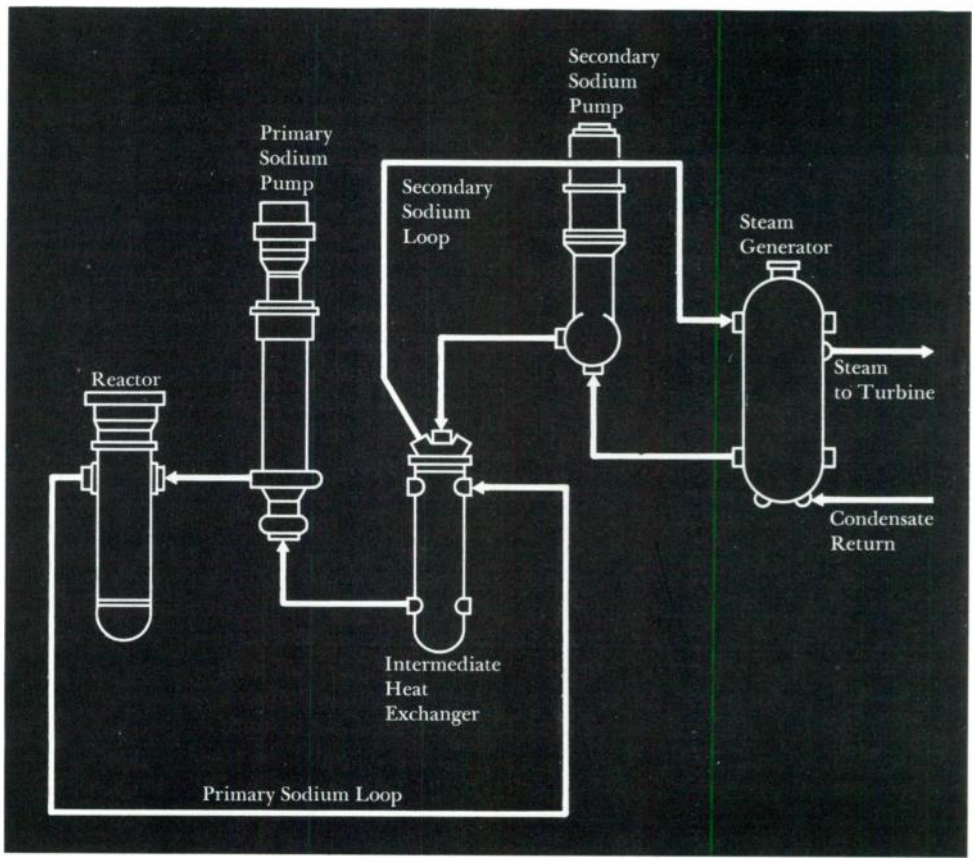
In 1966, over 40 percent of the total generating capacity ordered by the electric utility industry was for nuclear plants; in 1967, more than 50 percent of the capacity contracted for was nuclear. And there is no reason to expect that this trend will not continue—if anything, the rate of growth should increase. The nuclear plants that are now being designed will generate electricity for less than five mills per kilowatthour, a rate that makes them economically competitive with fossil-fueled plants in all but the lowest-cost fossil fuel areas.

The shape of nuclear power in the years to come can be predicted from today's trends. The growth of electric utility load in the United States continues to increase at a rate of about 7 percent per year; hence, utility system generation capacity doubles in size every ten years. Predictions of the part that nuclear power will play in this growth have been changing rapidly, but present estimates range from 120,000 to 170,000 MW of nuclear capacity installed in the United States by 1980; by the year 2000, it is estimated that nuclear power plants will constitute at least 50 percent of our total electrical generating capacity of 1.5 billion to 1.8 billion megawatts.

Economic Questions

To accomplish the anticipated nuclear growth with today's burner reactors as cited—reactors that consume more fissionable fuel than they produce—a number of economic hurdles would have to be surmounted:

First of all, the presently used fuel cycle, while economically competitive, has a "combustion efficiency" of less than one percent—more than 99 percent of the energy potentially available from uranium ore cannot be released. This limited utilization comes about because light-water reactors require enriched uranium obtained by processing natural uranium through gaseous-diffusion enrichment plants; approximately six units of natural uranium are required to obtain one unit of uranium at the proper enrichment level. (This enrichment procedure is demonstrated for a typical 1000-MW plant, Fig. 2.) The other five units must be set aside as "depleted uranium," not usable in water reactors. Furthermore, even from the one usable unit of enriched fuel, only 3 to 5 percent of the potential energy con-



1—Present sodium-cooled fast breeder reactor designs use a primary sodium loop and a secondary sodium loop to isolate reactive primary sodium from the steam loop.

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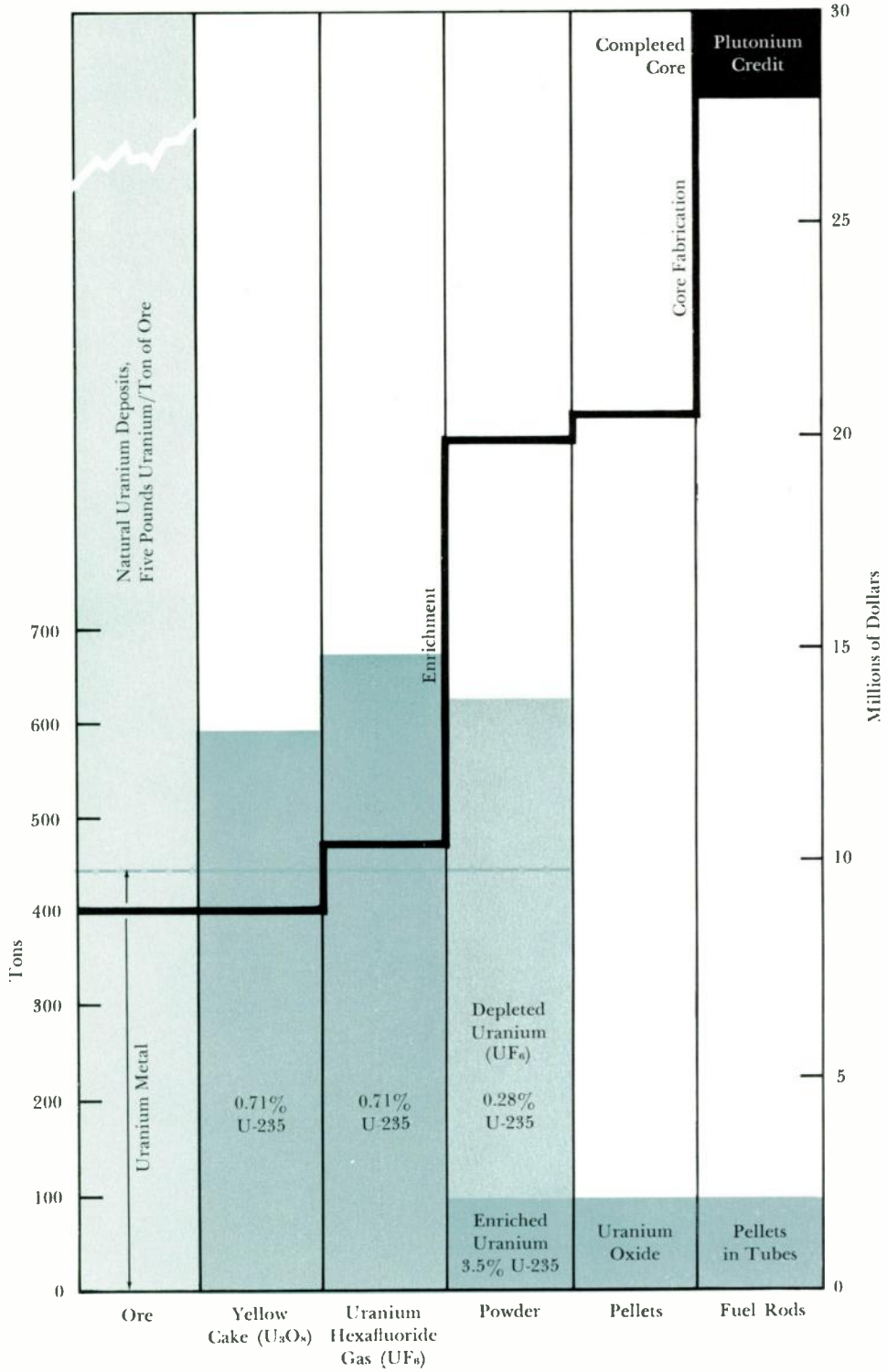


2—Uranium enrichment process is demonstrated for a 1000-MWe nuclear water reactor. The $30-million cost of a typical reactor core is divided into three approximately equal parts—cost of natural uranium, uranium enrichment, and core fabrication.

tent can be released. Certainly, the energy reserve now being accumulated in the depleted uranium stockpile will always be available, waiting for recovery. But this recovery cannot be accomplished with burner plants.

Although there is no shortage presently envisioned in the supply of uranium obtainable at reasonable cost, an indefinite expansion in nuclear power capacity using only water reactors will certainly place huge demands on future requirements for both uranium ore and uranium enrichment facilities.

Another economic question concerns plutonium credit. Today's water reactors create sizable quantities of fissionable plutonium, which affects the economics of the fuel cycle because sale of this plutonium reduces power generation costs. For example, today's water reactors receive a plutonium credit of about ¼ mill per kilowatthour. But this credit will depend on the market value of plutonium. It is technically feasible and will soon be economically attractive to recycle plutonium in water reactors.

Westinghouse currently is conducting research in plutonium critical experiments, neutron lifetime behavior, fabrication development, and optimum plutonium fueling schemes for water reactors. This research program is supported not only by Westinghouse but also by electric utility companies and by atomic development agencies of the United States and Europe. This effort will ensure a sound economic value for plutonium by the early 1970's when the guaranteed Government buy-back of plutonium will no longer exist.

However, plutonium can have an even greater value as an efficient fuel for fast breeder reactors. For example, it has been estimated that the 300 tonnes (metric tons) of plutonium that will have been produced in light water reactors by 1990 will be worth about $3 billion as water reactor recycle fuel, but could be worth as much as 50 percent more as a breeder fuel. Thus, optimum water-reactor fuel cycle costs can really be achieved only if the fast breeder is developed to provide a strong *premium* market for plutonium. (In addition to plutonium, the breeder would

be loaded with uranium from the depleted-uranium stockpile.)

In the final analysis, all of these economic hurdles reduce to one basic problem—the need to stabilize nuclear fuel cycle costs for the future. Even though presently known domestic supplies of low-cost uranium are continuously enlarged by massive exploration efforts, fuel costs of water reactors will always be subject to the uncertainties of uranium ore costs if water reactors are the only type built. Thus, the really significant potential benefit of the breeder reactor to the electric utility industry will come from its ability to ensure continuing low nuclear fuel costs.

Breeder Reactors and Uranium Requirements

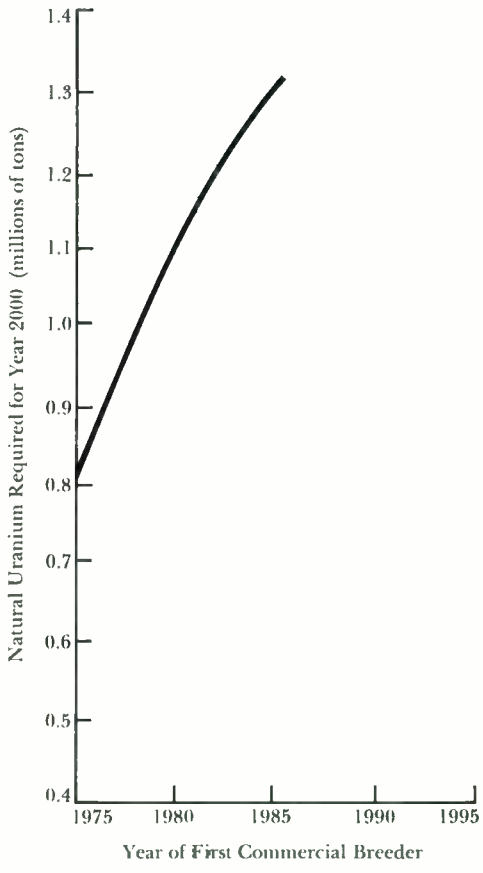
The most significant factor in stabilizing uranium fuel supply requirements with the breeder cycle is the *doubling time* of the breeder. Any reactor that produces more fissile material than it destroys can be called a breeder, but only a *high-gain* breeder can have a significant effect on uranium requirements. A high-gain breeder is a reactor that has a fissile-fuel doubling time shorter than the doubling time of nuclear capacity growth. Today, this fissile-fuel doubling time should be no more than eight years. The reason a low-gain breeder would be economically unsatisfactory is obvious: since it would not produce plutonium fast enough to permit reactor capacity to expand at the same rate as load demand, the low-gain breeder would limit its own growth to a rate slower than the increase in demand, and it could never provide sufficient plutonium to feed an expanding industry.

For example, it has been estimated that for a given set of starting conditions (150,000 MWe of nuclear capacity installed in 1980), uranium requirements with water reactors and low-gain breeders would be more than 3-million tons of uranium ore over the next 50 years. With high-gain breeders and water reactors, uranium ore requirements for these same starting conditions can be reduced by almost 50 percent; even more significantly, in the combination of high-gain breeders and water reactors, the plutonium balance would reach a point within this same period after which *no additional uranium* would be required. This does not mean that no uranium would be mined, but simply that utility management would then have an economic choice as to whether to mine uranium or use plutonium and the depleted-uranium stockpile.

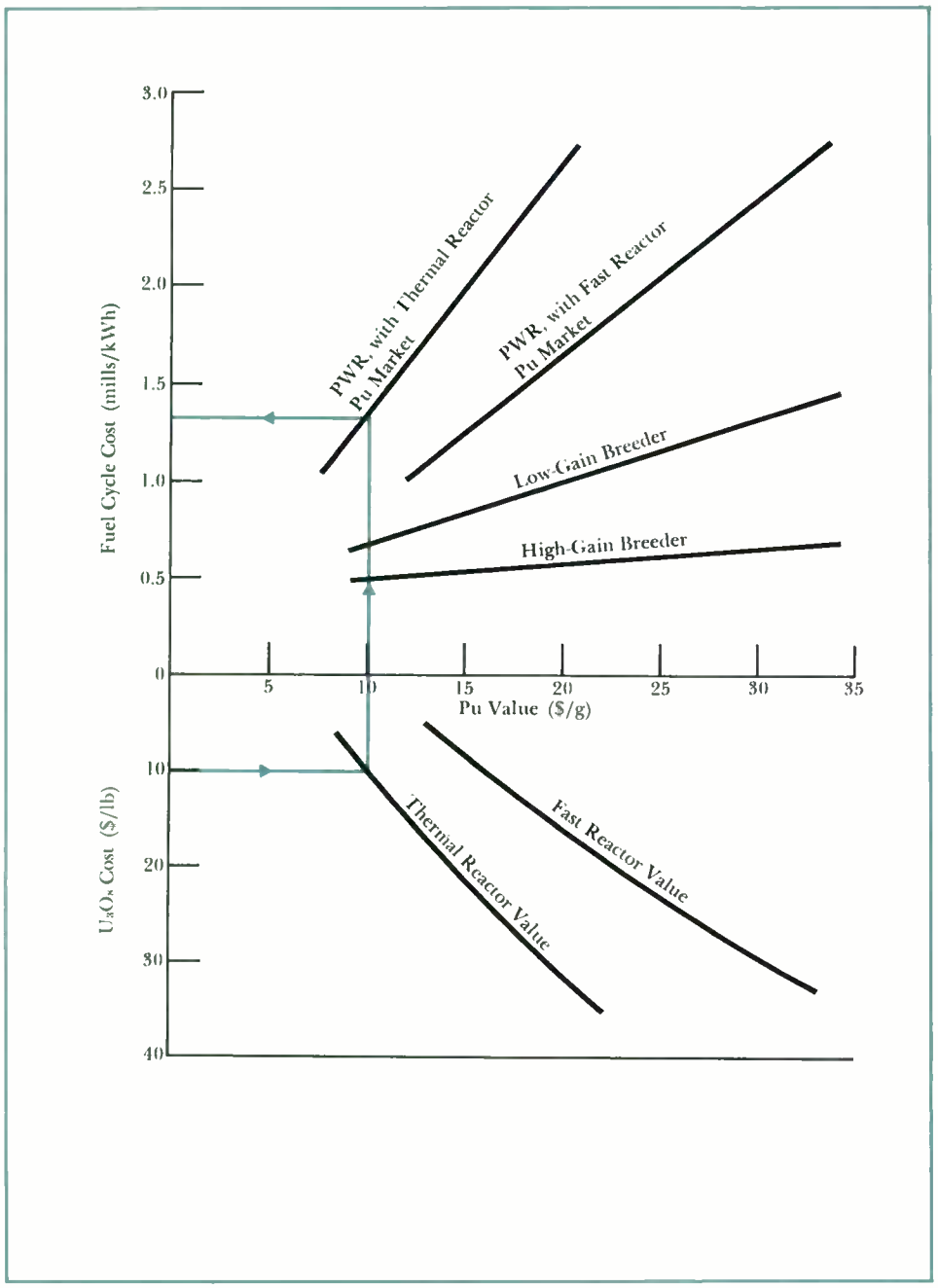
Assuming that a high-gain breeder can be made economic, the date that it is introduced will also have a significant effect on overall fuel requirements. For example, assume again that a water reactor capacity of 150,000 MWe exists by the year 1980, and that with the introduction of the breeder reactor, continuing additions of high-gain breeders and water reactors are built to satisfy total load demand thereafter. An estimate of total uranium requirements through the year 2000 can then be plotted as a function of the date that the high-gain breeder is introduced (Fig. 3). Thus, if the present target date of 1980 for economic breeder power can be achieved, estimated uranium requirements by the year 2000 are 1.1 million tons; however, if this breeder schedule should slip five years, an additional 200,000 tons of uranium would be required by the year 2000, the result of not producing excess breeder plutonium over this five-year period. Thus, the uranium requirements that result from a lag in breeder schedule could be in the neighborhood of 40,000 tons of additional ore per year of breeder delay—or approximately a billion-dollar penalty in fuel costs per year of delay!

And this delay cannot be made up, because making fissionable fuel in a breeder cycle is a "bootstrapping" operation that requires time—doubling time, to be exact. And since breeder doubling time will be limited to some value limited by technology and the overall economic combination of breeder plant and fuel cycle, plutonium production can expand only at the rate permitted by the product of unit production rate and the number of breeder units in service.

The effect of uranium cost and plutonium credit on fuel cycle costs can be summarized graphically (Fig. 4).



3—Total uranium requirements by the year 2000 will be affected by the year the first commercial breeder reactor is introduced. This curve assumes a water reactor capacity of 150,000 MWe in 1980, and continuing additions of high-gain breeders and water reactors after the year of breeder introduction.



4—Fuel-cycle costs for pressurized-water reactors in 1985 will depend on uranium ore cost, but they will be less sensitive to ore costs if there is a breeder cycle to create a premium market for plutonium. High-gain breeder fuel-cycle costs will be independent of uranium and plutonium costs because of the value of plutonium bred.

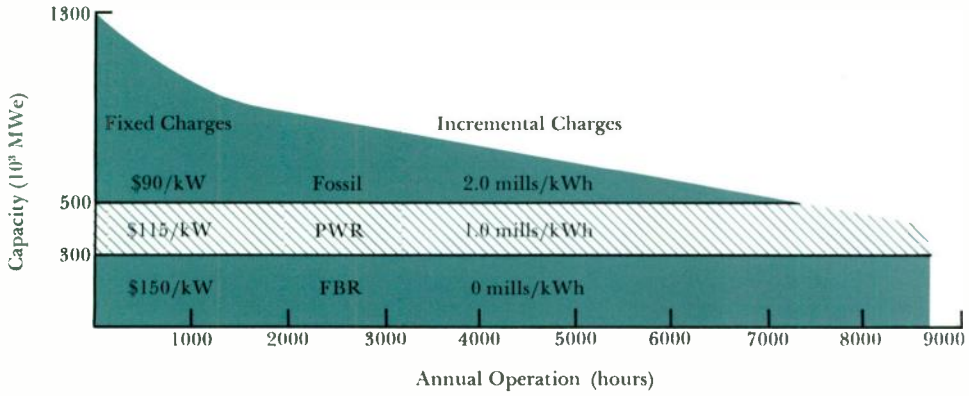
These relationships are shown as they might appear in 1985. Thus, if natural uranium costs $10 per pound, as shown on the lower left-hand scale (U_3O_8 *Cost*), and plutonium value for recycling in water reactors is about $10 per gram credit to the water reactor cycle, total fuel cycle cost is about 1.3 mills per kilowatthour.

The curves illustrate the greater value of plutonium for breeder-reactor fuel. For example, if water-reactor recycle provides the only market for plutonium, the PWR fuel cycle cost will be rather sensitive to increasing uranium prices—for each $5 per pound increase in uranium cost, PWR fuel cycle cost will increase about 0.3 mill per kWh. However, if the breeder reactor has been successfully developed, the premium price that the breeder will create for plutonium should raise the plutonium credit to the PWR fuel cycle, making it less sensitive to uranium price increases.

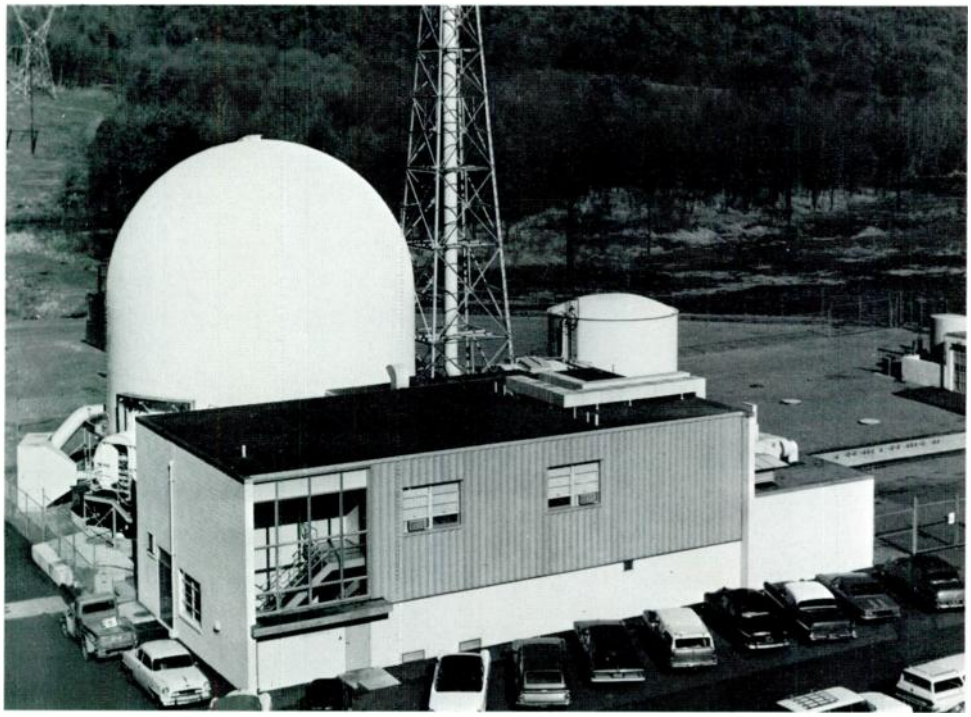
Fig. 4 also shows the economic potential of the high-gain fast breeder itself. Because the value of the plutonium bred in the high-gain breeder will offset the cost of plutonium inventory, the high-gain breeder fuel cycle cost curve is nearly flat, independent of the value placed on uranium or plutonium. This high-gain breeder requires no enriched uranium, only plutonium and the low-cost fertile uranium-238 isotope, which is the 99 percent of natural uranium that is unusable in light-water reactors. Already, the depleted-uranium stockpile contains enough U-238 to feed the breeder industry for more than 100 years—if we had the breeder plants in service.

Breeder Plant Operation

Breeder plants will probably have higher fixed costs than water reactors because of the additional plant complexity. But they will have no incremental costs, since the value of the plutonium they breed will cover fuel reconditioning expenses. Thus, to gain maximum advantage from the breeder plant cost characteristics, the breeder reactor will be most economic when operated in the base-load portion of the load-duration curve, with a load factor of 80 percent or better as shown in Fig. 5.



5—Electric utility load-duration curve for 1995 demonstrates the anticipated generation capacity makeup, assuming that an economic fast breeder reactor has been introduced in the 1980's. Fast breeder plants will operate in the base-load portion of the curve.



The Saxton experimental reactor, owned and operated by the Saxton Nuclear Experimental Corporation (SNEC), has been the testing ground for many of the advanced concepts used in today's PWR plants. SNEC and Westinghouse are currently engaged in a mutual program of reactor development, a major part of which is the evaluation of plutonium as a fuel for water reactors. The Saxton reactor has attained its full power rating with a core containing plutonium fuel. This is the first application of plutonium as fuel in a commercial type reactor. This achievement is part of an extended program of plutonium operation partially financed by the Joint United States—Euratom Research and Development Board under a contract administered by the United States Atomic Energy Commission. Other studies in plutonium utilization have been carried out as a joint effort of Westinghouse, the EEI and E.S.A.D.A. (Empire State Atomic Development Associates).

Water reactors, with lower capital costs and fuel costs stabilized by the breeder fuel cycle, will provide load-follow capacity, operating in the intermediate portion of the load-duration curve. Fossil-fuel plants, with the lowest capital costs and highest incremental costs, will continue to make up more than 50 percent of the installed capacity; they will be used in this mixed system for peak-load generation and system reserve duty.

Summary

The anticipated growth of nuclear power, if it is to follow its most economic route, can be outlined briefly: Water reactors, benefiting from the premium plutonium credit that will result from the demand for plutonium for breeder fuel, will continue to dominate nuclear plant additions until 1985. During this period, these plants will generate the initial charge of plutonium required for starting the high-gain breeder cycle. A mix of high-gain breeders and water reactors, constituted to best satisfy the utility load duration curve, will make up the nuclear growth from that point on.

The timetable for nuclear growth hinges on successful development of a breeder system that is truely economic. Although various breeder concepts may be developed, the only ones that will be useful to the utility industry will be those that serve three important and related functions:

1) They must be economically attractive compared to all other power sources at the time of installation;

2) They must have fuel-cycle characteristics that provide continuing favorable economics independent of future uranium or plutonium prices;

3) They should create a more stable fuel cycle environment for all in-service reactors. The breeder can accomplish this goal if it reduces the future demand for large quantities of uranium, and if it provides a truly premium market for the plutonium generated by other reactors.

A breeder system that fills all of these needs will guarantee abundant and inexpensive nuclear power for generations to come.

Fast-Neutron Breeder Power Reactors... Some Basic Concepts

Plutonium-239, the fissile material produced from uranium-238, is the basic fuel for breeder reactors. The nuclear physics parameters of plutonium dictate a fast neutron spectrum for breeder reactor operation.

The significant differences between the nuclear reactions in today's burner reactors and those in the proposed fast breeder reactors (FBR) concern the mean energy of neutron flux spectra ϕ (neutrons per square centimeter per second, Fig. 1) and its effect on the availability of neutrons for breeding.

All neutrons produced in a fission process are fast neutrons, i.e., they have kinetic energies around a million electron volts, and speeds of some 5,000 miles per second. However, fission of uranium-235 in today's burner reactors is accomplished predominantly with neutrons that have energies of less than one electron volt (thermal energies) and speeds of about one mile per second. A water moderator slows the fast neutrons down to thermal speeds by means of billard-ball type collisions between hydrogen atoms and neutrons. Emphasis in burner reactor design, therefore, seeks to obtain a population of thermal neutrons in the core sufficient to maintain the fissioning chain reaction at the desired power level.

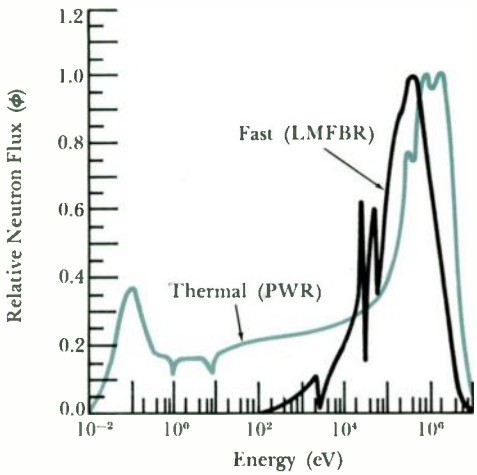
In contrast, breeder reactor design has two main nuclear objectives: maintain a chain reaction in fissile (fissionable by neutrons of all energies) plutonium-239, and provide enough additional neutrons to convert fertile uranium-238 to fissile plutonium-239. The reactor is a "breeder" when more plutonium-239 is made than is burned. As with a burner reactor, neutrons deteriorate the fuel elements and fission produces atoms that absorb neutrons. Hence, this is not perpetual motion; the fuel elements must be replaced periodically.

Uranium-238 can be bred from plutonium-239 only in a fast neutron spectrum. This is because only fast neutrons can produce enough excess neutrons for breeding when absorbed in plutonium-239. Fast neutrons are about 30 percent more efficient than thermal neutrons in fissioning plutonium-239 because the latter have more than twice the probability of being parasitically captured by plutonium-239.

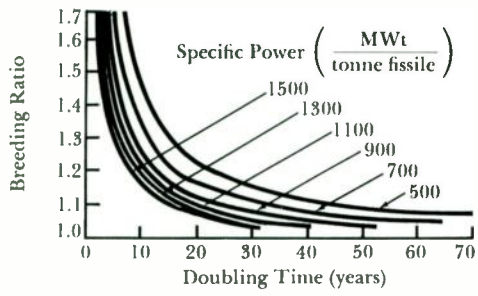
Experience to date (Table I) indicates that fast neutron breeder power plants can be designed, built and operated successfully. Sodium, NaK, and mercury have been used as coolants in fast breeder reactors. Both uranium and plutonium have been used for fuel. Such fuel has functioned satisfactorily in the form of metal alloys, oxides, and carbides and there has also been successful operation of components. While there have been some failures, no one type of failure has occurred consistently in all plants. Available theory, basic constants, and critical experiment data to date provide a basis for fairly accurate prediction of criticality and nuclear performance.

Despite the fact that the first electrical power generated from nuclear energy was by the fast breeder EBR-1 on December 20, 1951, the fast breeder development program was not pushed as aggressively as the water reactor programs. To date, five fast reactors have operated in the United States and four abroad compared to the 50,000 MWe of water reactors built, building or committed in the U.S.

Although demonstration FBR plants are planned for the 1970's, not until the early 1980's will large economic fast breeder power plants be constructed in the United States.



1—Typical neutron flux spectra in fast and thermal reactors.



2—Fast breeder reactor doubling time is a function of breeding ratio and specific power.

Breeding Ratio

Breeding ratio is defined as the ratio of fissile material produced to fissile material destroyed. For example, the breeding ratio is obtained by dividing the number of captures in fertile material which produce fissile material by the total number of fissions plus captures which destroy fissile material. The following is an expression for breeding ratio (BR) in terms of nuclear constants:

$$BR = \eta - 1 + \frac{F_u(\nu_c - 1) - C}{F_p(1+\alpha)}$$

where as given in Table II: η is the number

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of neutrons produced per absorption in the fissile material; ν is the neutrons produced per fast fission in the fertile material; α is the ratio of capture to fission cross sections for fissile material; F_u and F_p are fractions of all fissions or power which comes from fertile and fissile atoms respectively; C is the fraction of neutrons absorbed in other than fertile or fissile material per fission, and L is the fraction of neutrons absorbed in other than fertile of fissile material per neutron generated.

Substituting values for plutonium-239 and uranium-238 from Table II and $F_u = 20$ percent of power from uranium fast fission yields:

$$BR = 2 - 1.1C = 2 - 3.2L$$

Doubling Time

Even more important economically than breeding ratio is the time (doubling time) required to double the total fissile inventory in the reactor plus that being reprocessed outside the reactor.

This doubling time (DT) can be expressed:

$$DT_s = \frac{1000}{0.365\, P_{st}\, g f F_p (1+\alpha)(BR-1)}$$

where DT_s is simple doubling time in years; P_{st} is specific power (See Fig. 2) in megawatts thermal per tonne of fissile fuel in and outside reactor (750 MWt is a typical value); g is grams fissioned per megawatt day (1.0); f is plant factor (0.85); F_p is fraction of power coming from fissile fissions (0.8); α is the ratio of capture to fission cross sections (0.15); BR is breeding ratio (1.4). Substituting the numbers in parentheses (which are typical values) gives:

$$DT_s = 11.7 \text{ years}$$

The simple doubling time (DT_s) is the time required to reproduce the inventory. It is distinguished from the "compound doubling time" (DT_c) wherein excess fissile material inserted in another operating reactor would provide compounding similar to a bank compounding interest. Compound doubling time DT_c (8.5 years in example) is related to simple doubling time (DT_s) as follows:

$$DT_c = \frac{0.693}{\ln\left(1+\frac{1}{DT_s}\right)}$$

Fig. 2 shows the compound doubling time plotted versus breeding ratios for various specific powers. This gives a general picture of the relative importance of breeding ratio and specific power.

It is economically desirable that the doubling time for fissile fuel should be shorter than that for nuclear power application of breeders. This would provide a fissile Pu-239 inventory for new breeders. Based on present cost estimates, use of U-235 as fissile fuel for breeders may cost 0.3 to 0.5 mills/kWh more than Pu-239 but might be used for first fuel inventory of a new breeder if the economics throughout plant life justify it.

Practical factors tending to increase doubling time are the time required to build up breeder blanket fissile inventory after startup, time for fuel reprocessing, and the economics of reprocessing blanket fuel elements having little plutonium.

Table I—Fast Reactors

United States	*Fuel*	*Criticality*	*Electrical Power*	*Coolant*	*Design kWe*	*Power kWt*	*Operation kWt*
Clementine*	Pu	Nov. 1946	None	Hg	0	25	25
EBR-1*	U, Pu	Aug. 1951	Dec. 1951	NaK	250	1200	200
LAMPRE-1*	Pu	1961	None	Na	—	1000	1000
Fermi	U	Aug. 1963	Aug. 1965	Na	67,000	200,000	100,000
EBR-2	U	Nov. 1963	Mar. 1965	Na	20,000	62,500	45,000
SEFOR	U-PuO_2	1968	None	Na	—	20,000	—
FFTF	U-PuO_2	1972–73	None	Na	—	400,000	—
Foreign							
UK-Dounreay	U	Nov. 1959	Oct. 1963	NaK	15,000	72,000	60,000
UK-PFR-Dounreay	U-PuO_2	1971	—	Na	250,000	—	—
USSR BR-2*	Pu	1956	—	Hg	—	100	100
USSR BR-5	U-PuO_2; PuC	1959	—	Na	—	5000	5000
USSR BN-350	U-PuO_2	1969	—	Na	150,000	A	—
USSR BOR	U-PuO_2	1968	—	Na	—	50,000	—
France-Rapsodie	U-PuO_2	1967	None	Na	—	20,000	20,000
France-Phenix	U-PuO_2	1973	—	Na	250,000	600,000	—
Germany-NA-2	U-PuO_2	1973	—	Na	350,000	750,000	—

**Project now terminated*

A) 150 MWe plus 12,000 tons fresh water per day.

Nuclear Cross Sections

The cross section (σ_x) of a particular nucleus (Table II) can be interpreted as the probability that a neutron passing through a cubic centimeter containing one atom will interact by process x with the nucleus of that atom.

In general, the effective cross section for neutron interaction tends to become smaller as the speed of the incident neutron increases. Since the probability of nonfission capture becomes lower, a self-sustaining reaction remains possible.

The decrease in fission cross section (σ_f) in a fast neutron spectrum is significant, as shown in Table II. For example, σ_f for plutonium-239 is 790 barns in a thermal spectrum, but only 1.78 barns in a fast spectrum. Thus, the neutron flux, ϕ, must be greater in a fast reactor than in a thermal reactor to produce the same power per unit mass of fissile material. This is because power is proportional to $\phi\sigma_f\mathcal{N}$, where $\mathcal{N}$ is the number of fissionable atoms.

Even more significant is the fact that α (the ratio of capture cross section to fission cross section for fissile material) goes down with increasing neutron energy as is indicated in Table II. As a result, fewer neutrons are parasitically captured in a fast reactor.

The capture cross sections for coolant and structural materials increase relative to fission cross sections in a fast spectrum, as is shown in Table II for iron, sodium, and zirconium. In addition, each nonfuel nucleus has a scattering cross section which slows down the neutrons and thus lowers η—the neutrons produced per fuel absorption. For these various reasons, 4 or 5 times more fissile nuclei are required per unit volume to compete for neutrons in a practical fast reactor than in a thermal reactor. Table II illustrates how the greater (by a factor of 4) number of fissile atoms per cm³ in a FBR compared to the Yankee PWR makes σ_c/Σ_f, the relative effect per nonfuel atom, less for the fast reactor, even though the ratio of cross sections, σ_c/σ_f, is higher. (Thus, $\Sigma_f = \mathcal{N}\sigma_f$, where $\mathcal{N}$ is the number of fuel atoms per cm³ and is greater for the fast reactor.) Similarly, the effective absorption of neutrons by fission fragments is less in the fast reactor because more fissile atoms exist to compete for neutrons even at the burnup of 100,000 megawatt day per tonne fuel burnup (to be described later) for fast reactors. (This is about three times the energy produced per tonne of thermal reactor fuel before replacement.) Thus, the various neutron cross sections and atomic ratios must be used by the fast breeder designer if the "balance sheet" for neutrons is to show enough neutrons produced to account for all neutron absorptions.

Fission Reactions

The basic fissioning processes that occur in typical uranium and plutonium fissile isotopes can be indicated in equations as follows:

$$_{92}U^{235} + {}_0n^1 \rightarrow {}_{56}Ba^{137} + {}_{36}Kr^{97} + 2{}_0n^1 + 200 \text{ MeV}$$

$$_{92}U^{233} + {}_0n^1 \rightarrow {}_{56}Ba^{136} + {}_{36}Kr^{96} + 2{}_0n^1 + 200 \text{ MeV}$$

$$_{94}Pu^{239} + {}_0n^1 \rightarrow {}_{56}Ba^{137} + {}_{38}Sr^{100} + 3{}_0n^1 + 210 \text{ MeV}$$

In the first equation, a fissile uranium-235 nucleus is struck by a neutron and the nucleus splits to become a barium-137 nucleus and a krypton-97 nucleus. Two

Table II—Nuclear Parameters for Fast and Thermal Reactors*

	Fissile Fuel								*Fertile Material*			
	U-235		*U-233*		*Pu-239*		*Pu-241*		*U-238*		*Th-232*	
	Fast	*Thermal*	*Fast*	*Thermal*	*Fast*	*Thermal*	*Fast*	*Thermal*	*Fast*	*Thermal*	*Fast*	*Thermal*
Fission Cross Sect., Barns (σ_f 10^{-24} cm²)	1.4	577	2.20	527	1.78	790	2.54	1000	0.112	0	0.025	0
Neutrons Per Fission (ν)	2.5	2.4	2.59	2.51	3.0	2.90	3.04	2.98	2.60	—	2.4	0
Capture To Fission Ratio ($\alpha=\sigma_c/\sigma_f$)	0.15	0.17	0.068	0.10	0.15	0.5	0.114	0.4	—	—	—	—
Neutrons Per Absorption (η)	2.2	2.06	2.42	2.28	2.6	0.9	2.73	2.13	2.27	—	2.0	—
Delayed Neutron Fraction (β)	0.0065	—	0.0027	—	0.002	—	0.0053	—	0.0147	—	0.0204	—
Fission Threshold (MeV)	—	—	—	—	—	—	—	—	1.4	—	1.4	—

	Other Representative Materials							
	Fe		*Na*		*Zr*		*Effective Fission Product Pair*	
	Fast	*Thermal*	*Fast*	*Thermal*	*Fast*	*Thermal*	*Fast*	*Thermal*
σ_c/σ_f (U^{235}) 10^{-3}	6.1	4.4	1.8	0.87	—	0.31	0.08	0.1
σ_c/σ_f (U^{233}) 10^{-3}	4.0	4.8	1.18	0.96	—	0.34	—	0.1
σ_c/σ_f (Pu^{239}) 10^{-3}	4.9	3.4	1.46	0.68	3.3	0.24	—	0.1
σ_c/Σ_f Yankee 10^{-24}	—	15	—	3	—	1	—	236
σ_c/Σ_f fast breeder (Pu) 10^{-24}	4	—	1	—	3	—	120	—

Note: There are about 4 times as many fissile atoms per cm³ in the fast breeder as in Yankee thermal reactor. $\Sigma_f = \mathcal{N}\sigma_f$ where $\mathcal{N}$ is number of fissile atoms per cm³.
**These constants are approximate and change with neutron spectrum.*

fast neutrons are released, and 200 MeV of energy is produced. The energy and the neutrons are the useful products of the reaction.

The neutrons (ν) emitted per fission will vary with the energy of the incident neutron. Averages must therefore be considered for an energy range of incident neutrons, rather than for a specific reaction. Average fission cross sections (σ_f), the number of neutrons (ν) produced per fission, and the number of neutrons (η) produced per neutron absorbed in fissile and fertile fuel for fast and thermal neutrons are shown in Table II.

The number (η) of neutrons produced is particularly significant in breeder reactors because η neutrons must not only sustain the fissioning process, they must also provide for parasitic capture and breed fissile fuel.

Breeder Reactions

The breeding of fissile fuel by typical neutron reactions in fertile material can be written:

$$_{92}U^{238} + {}_0n^1 \rightarrow {}_{92}U^{239} \xrightarrow{24\ \text{min}}$$

$$-e + {}_{93}Np^{239} \xrightarrow{2.3\ \text{days}} -e + {}_{94}Pu^{239}$$

$$_{90}Th^{232} + {}_0n^1 \rightarrow {}_{90}Th^{232} \xrightarrow{22\ \text{min}}$$

$$-e + {}_{91}Pa^{233} \xrightarrow{27\ \text{days}} -e + {}_{92}U^{233}$$

In the first equation, a neutron is absorbed by a uranium-238 nucleus, which becomes a uranium-239 nucleus with a half-life of 24 minutes. When an electron is emitted, the nucleus gains a positive charge to become neptunium-239; some 2.3 days later another electron is released and the nucleus becomes plutonium-239. This reaction is similar to the breeding equation for conversion of thorium-232 to uranium-233, which is to be used in the thermal neutron breeders.

Parasitic Reactions

Parasitic reactions also compete for available neutrons and reduce the neutron population available for fissioning and breeding purposes. For example, plutonium-239 can capture neutrons without fissioning, as can the sodium coolant and the structural iron in the core. These reactions, respectively, can be written:

$$_{94}Pu^{239} + {}_0n^1 \rightarrow {}_{94}Pu^{240}$$

$$_{11}Na^{23} + {}_0n^1 \rightarrow {}_{11}Na^{24} \xrightarrow{14.8\ \text{hours}} -e + {}_{12}Mg^{24}$$

$$_{26}Fe^{56} + {}_0n^1 \rightarrow {}_{26}Fe^{57}$$

The plutonium-240 reaction is undesirable, even though plutonium-240 is a fertile nucleus and can be fissioned by fast neutrons. Although 3.17 neutrons are produced from fast fission of plutonium-240, thereby replacing the three neutrons required to produce and fission it, the production of plutonium-240 or higher isotopes represents a net loss to the neutron population because of the high probability of neutron capture without fission.

The sodium reaction destroys neutrons and creates radioactive sodium, which is an impediment to reactor maintenance and requires careful shielding until the sodium-24 decays with a 14.8-hour half life.

While the above equations indicate possible nuclear reactions, it is the relative probability (cross section) of a reaction occurring that is the significant factor. Such nuclear parameters have been measured experimentally and verified by operation of various types of reactors. The key parameters are listed in Table II.

Neutron Production (η) per Fissile Absorption

The basic nuclear physics parameter that dictates the fast neutron spectrum reactor for breeding purposes is the average number (η) of neutrons produced per neutron absorbed in fissile material.

The values of η for plutonium-239 are most important in breeder reactor design because this is the fissile material produced from fertile U-238 and therefore the basic fuel for breeder reactor operation.

In the breeding process, η must be greater than 2 (one for fission plus one for fertilization) to allow breeding, even if no neutrons are absorbed by the coolant and structure and none escape from the reactor. As indicated in Table II and Fig. 3, uranium-233 might breed in a thermal neutron spectrum but plutonium-239 will have an η significantly greater than 2 *only* in a fast neutron spectrum. A fast neutron spectrum produces more neutrons per absorption than a thermal spectrum because the additional neutron energy when absorbed in the fissile nucleus makes fission more probable. The difference between the neutrons (ν) produced per fission and the number (η) available per absorption in fissile material depends on relative neutron cross sections for fission and capture.

Neutron Balance

A typical neutron balance is given for a water reactor and a fast breeder in Table III. From this balance sheet, several facts can be inferred:

1) Twenty percent of the power generated comes from the fast fission of uranium-238 in a fast neutron spectrum, compared to only 7.5 percent in a thermal neutron spectrum. More uranium-238 atoms are fissioned directly in the fast spectrum because of the greater concentration of neutron flux with energy higher than the fission threshold for uranium-238 (1.4 MeV as given in Table II).

2) There is relatively less parasitic capture in the fast spectrum in the fissile material as indicated by a smaller fraction of captures in fissile material.

3) The fissile material destroyed in a fast neutron flux to produce 1000 neutrons is only about 70 percent of that required in a thermal flux, yet 1.8 times as much fertile material is converted to fissile material. The breeding ratios of 1.4 and 0.6 are typical of the two types of reactors. For example, the Yankee water reactor has a conversion ratio of about 0.5, whereas the Fermi fast breeder reactor has a breeding ratio of about 1.2 when fueled with uranium-235. The Fermi breeding ratio, incidently, could probably be increased to 1.4 with plutonium fuel. Such a switch to plutonium would require control changes because of fewer delayed neutrons (β) produced per fission of plutonium-239 (see Table II).

Delayed Neutrons (β)

Except for a small fraction of "delayed" neutrons, the neutrons produced by fission are emitted within one-tenth of a

microsecond. The delayed fraction (β) is listed in Table II for several nuclei.

Delayed neutrons are important because they alone have a long enough time constant to permit their control by mechanical means. For instance, delayed neutrons are emitted with half-lives from 0.4 to 55 seconds, which is long enough for neutron-absorbing control rods to move and thus control neutron level.

The fraction of delayed neutrons from plutonium-239 fission is approximately 0.25 percent, compared to the 0.7 percent delayed neutrons from uranium-235 fission. Thus, although the time constant for delayed neutrons in an FBR is similar to that for a water reactor, there are only one-third as many delayed neutrons in a plutonium reactor. This means for a plutonium reactor fewer controllable neutrons and the control band is narrower between a condition wherein the reactor is self-sustaining with prompt plus delayed neutrons (delayed critical), and the condition where it is self-sustaining with prompt neutrons *only* (prompt critical). In control terminology, the "delayed critical to prompt critical control band" is referred to as a "dollar of reactivity." The "dollar" has a ΔK reactivity worth which equals numerically the average value of β (see Table II), the fraction of delayed neutrons for the given reactor. For reactivity additions of less than a dollar, the fast breeder responds with the same time constant as the thermal reactor would if the fuel composition were similar. Transient response for more than a dollar of reactivity added depends upon prompt neutron generation times which are shorter for the FBR.

When EBR-1 fuel was changed from uranium-235 to plutonium-239, the control response was more sensitive to control rod motion, because only plutonium-239 ($\beta = 0.25$ percent) was involved. In contrast, in an economic fast breeder, uranium-238 will be mixed with plutonium-239. The delayed neutron fraction for uranium-238 fast fission is 1.47 percent (See Table II) which is more than five times greater than plutonium-239. The fast fission of uranium-238 thus provides additional control neutrons for the fast breeder.

For example, if 20 percent of the power comes from the fast fission of uranium-238 (Table III), more than half of the delayed neutron population will be supplied from this source and the total delayed neutron fraction would be doubled. This doubles the range of reactivity over which the fast breeder can be controlled on a delayed neutron basis.

Table III—Typical Neutron Balance

	Thermal PWR	*Fast Breeder*
Neutrons Produced		
Fissions in Plutonium-239 × ν_{239}	—	822
Fissions in Uranium-238 × ν_{238}	80	178
Fissions in Uranium-235 × ν_{235}	920	—
Total	1000	1000
Neutrons Absorbed		
Fissions		
Fissle Material (U^{235}, Pu^{239}, Pu^{241})	383	287
Fertile Material (U^{238}, Pu^{240})	31	83
Captures		
Fissile Material	65	63
Fertile Material	270	494
Structure & Coolant	89	23
Fission Products	100	20
Leakage and Control	62	30
Total	1000	1000
Breeding Ratio	0.6	1.4

Doppler Effect

Another beneficial process by which neutrons are absorbed by uranium-238 (the Doppler effect) yields additional control stability. The capture of neutrons in the intermediate (1-100,000 eV) energy neutron flux spectrum is a function of uranium temperature. Thus, as power increases, the temperature of the uranium increases and more neutrons are captured. This reduces reactivity and thus inhibits further power increase. Conversely, negative power transients are inhibited by the same effect since as power decreases the uranium cools, captures fewer neutrons, and provides reactivity that inhibits further power decrease.

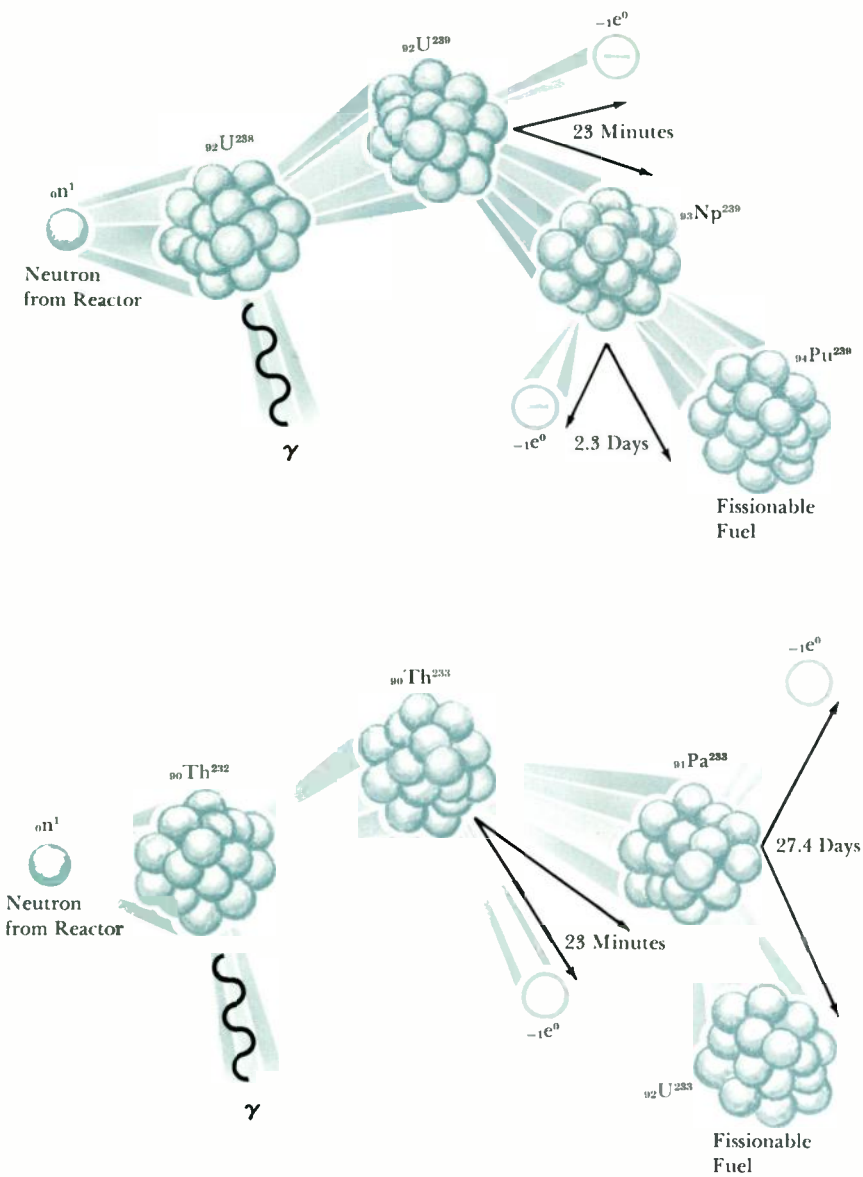
The change in neutron capture as a function of temperature occurs because of relative motion between the incident neutron and the thermal motion of vibration of the uranium in its crystal lattice. Since the neutron capture cross section as a function of energy has very sharp maxima (resonance peaks) in the (1-100,000 eV) resonance energy region (Fig. 4), more neutrons are captured if the peaks are broadened. Since thermal motion by uranium atoms means that an incident neutron can interact with uranium atoms with different relative velocities, the narrow energy peak for capture is broadened—thus producing more neutron captures. Since the effect depends upon relative motion, the analogy with frequency change of sound results in its being called the "Doppler effect."

Because the thermal motion is transmitted immediately with power rise, there is no delay in this degenerative feedback. As a result, the Doppler effect constitutes an important safety feature by inhibiting rapid transients which might otherwise involve larger power overshoots. The Doppler coefficient for the Fermi fast breeder reactor is -3×10^{-6} $\Delta K/°F$, which for 1000 degrees F change in fuel temperature would mean 0.3 percent ΔK. This is equivalent to 50 cents of control, or half as important as the delayed neutrons from uranium-235. In a plutonium-239 reactor, a similar Doppler effect would be worth more than a dollar of control. Hence, the effect is an im-

Breeder Reactors

Neutrons in a reactor can be used to manufacture, from fertile material, certain other nuclei that are fissionable. These artificially produced fuels can either be removed from the reactor and used as fuel for other reactors, or partially "burned" in place as a way of augmenting the lifetime of the reactor. Two typical fuel-producing reactions are illustrated.

The fission process emits about 2.5 neutrons (η) on the average per neutron absorbed in fissile material; one is required to cause another fission to perpetuate the chain reaction, leaving 1.5 neutrons to generate new fissile material. If a reactor is to make more fissionable material than it consumes, less than 0.5 neutron is left for leakage or capture by nonfission processes.



portant contribution to control stability to counteract possible positive reactivity transients, such as a sodium void coefficient.

Sodium Void Coefficient

The most efficient shape for a nuclear reactor core is one that minimizes the surface-to-volume ratio. This, in turn, minimizes neutron leakage—hence, the required fuel inventory. Therefore, typical thermal reactor cores are cylindrical with their height approximately equal to the diameter.

In a fast reactor, however, another consideration makes high neutron leakage desirable, so that breeder core designs are either pancakes, cylinders, or multiple cylinders (modular). The motive for these geometries is that neutron leakage, if accentuated, makes the reactor more sensitive, reactivity-wise, to loss of sodium from the reactor core.

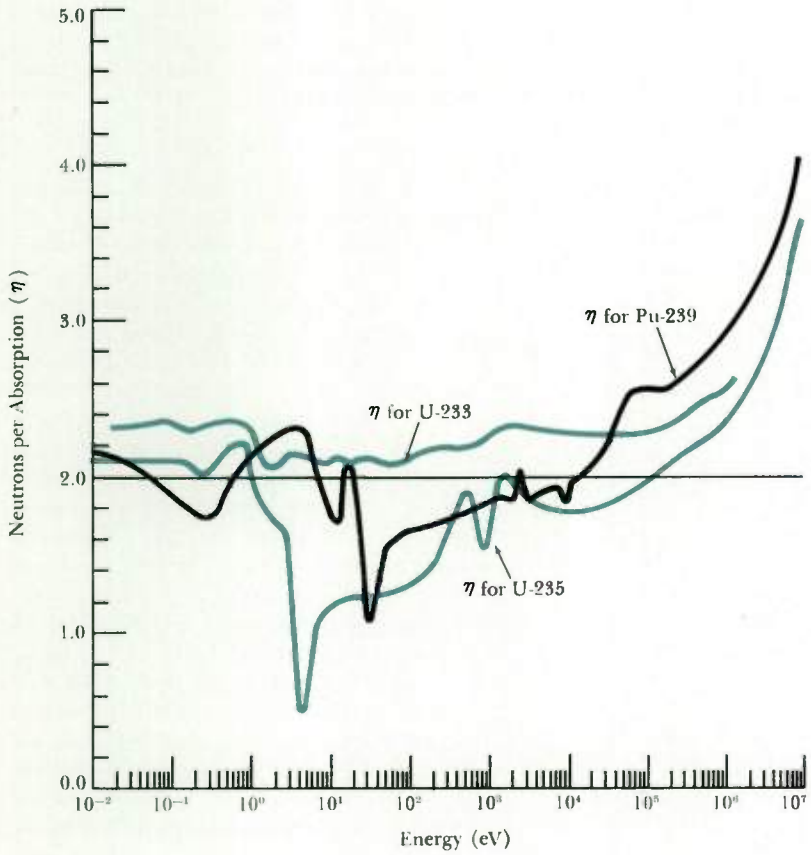
If sodium starts to boil and form a void in the reactor, it is desirable that the reactivity (and, hence, power) decrease so as to remove the cause and hence the void.

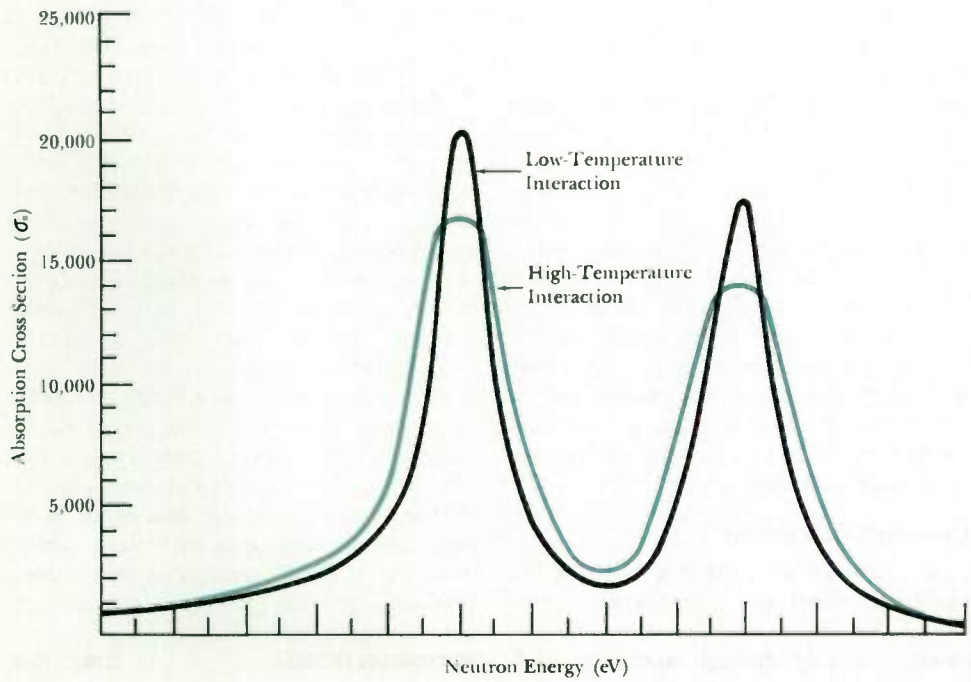
When a void occurs in the sodium, neutron leakage is increased through the void where otherwise the sodium scattering cross section would divert the neutrons and prevent leakage. The unusual geometries referred to above accentuate this leakage effect to cause reduced reactivity.

Conversely, reactivity can be gained because the decrease in scattering and hence moderation permits higher neutron energies and, hence, higher effective η. Sodium removal also reduces neutron capture, thus making more neutrons available which, in turn, increases reactivity.

If the net effect is a decrease in reactivity, the reactor is said to have a "negative void coefficient." This coefficient may be different for a local void or an overall loss of sodium, but is generally designed such that a large contribution in reactivity cannot be made. However, the Russian fast breeder designers believe the void coefficient can be handled by feedback control procedures based on instrumentation.

As a specific example of void coefficient, a certain fast reactor gains a maximum





amount of +17 cents if sodium is lost from the central 10 percent of the core volume, but will lose four dollars if the sodium is lost from the core and an additional six dollars if sodium is lost from the reactor vessel.

Fortunately, in normal operation at power the accentuated neutron leakage from a "pancake" or modular core surrounded by a blanket is not a total detriment to breeder reactor operation. This is because the leakage neutrons can be used to breed fissionable material in a surrounding uranium-238 blanket. However, the "high-leakage" geometry requires increased fissile fuel inventory and may thus increase power costs by perhaps 0.1 mill/kWh.

Expansion Coefficients

In addition to the Doppler effect and the sodium void coefficient, other coefficients influence reactivity and, hence, the kinetics of a fast breeder reactor. One group of such coefficients involves the effects of the mechanical expansion of reactor fuel, support structures, and coolant in the reactor.

The thermal expansion of fuel or structures increases reactor core size and thus lowers fuel density; it also increases the surface-to-mass ratio of the core, which increases neutron leakage and decreases reactivity. Since sodium may be squeezed out of the core by mechanical expansion of fuel or sodium, this may also cause a reactivity change similar to that described for sodium void.

Fuel temperature increases with power, so these various coefficients can also be related to power increases. Typically, the net effective coefficient for all these expansion effects might be in the order of −2¢/MW or −0.4¢/°F and can be influenced by design.

3—For a breeder reactor, neutrons generated per neutron absorption in fissile material must be greater than two. Thus, a Pu-239 breeder must operate in the high-energy spectrum.

4—Neutron absorption is a function of the temperature of the absorber atoms, which influence the width of absorption resonances by changing the energy of interaction between neutrons and atoms.

Fuel Burnup

One of the major problems with fast reactors is that fuel cycle economics requires that about three times as much energy be obtained per unit mass of fuel element as in a thermal reactor. A typical design value is 100,000 MWd/t (megawatt days per tonne of fuel). Since a megawatt day of energy requires the fissioning of 1.02 grams of plutonium, 100,000 MWd/t fuel burnup means that 100,000 grams per metric tonne or 10 percent of the fuel has been destroyed by fission creating 10 percent net new atoms. Fission fragment atoms occupy about the same volume as uranium atoms, so that about 10 percent atom volume increase might be expected for metal fuel, 5 percent for carbide, and 3 percent for oxide. Therefore, the mechanical structure of the fuel element is designed to accommodate swelling in addition to damage to the fuel. The fuel burnup (B) requirement in MWd/t is determined by the following factors:

1) Specific power P_s is given in terms of megawatts (thermal) produced per kilogram of fissile fuel. Since fissile plutonium nominally costs $10,000 per kilogram, about one (1) megawatt thermal per kilogram is desired so that the interest charges on the plutonium inventory will be reasonable (nominally 0.3 mills/kWh).

2) The ratio of the number of fissile atoms, F_p, per total fuel atoms is about 0.15 because about 15 percent enrichment is required for criticality.

3) Reloading cycle period is denoted by R_c and the fraction of fuel that is removed each cycle by F_c. Annual reloading gives an R_c of 365 days after which it is assumed that half of the core ($F_c=0.5$) is replaced.

4) Plant factor $P_f=0.85$ is a common target. The equation which relates these factors is:

$$B=\frac{P_s R_c F_p P_f}{F_c}\times 1000 \text{ MWd/t}$$

which gives 93,000 MWd/t for the sample factors given above. If only a third of the fuel were reloaded in the FBR each cycle, burnup requirements would be about 150,000 MWd/t. The out-of-core fuel inventory, however, would decrease from 50 to 33 percent. Such a decrease in inventory would decrease the compound doubling time of the total fuel inventory to nine years, assuming ten years as the former doubling time.

Shortening time between reloadings to less than a year could also decrease burnup requirements. Reloading time is subject to considerations of plant availability, utility load peaks, and time required for fuel reprocessing.

A fast breeder design for a 30,000 MWd/t burnup similar to water reactors could be based on replacement of half the fuel every 120 days.

In addition to the above considerations, the design burnup should be long enough to provide acceptable fabrication costs. For example, 100,000 MWd/t burnup and $200/kg fabrication is equivalent to 0.2 mills/kWh fabrication cost.

From a reactivity standpoint, burnup can be greater in a fast breeder than in a water reactor because fission products absorb a smaller fraction of the neutrons and breeding provides some reactivity help.

Equilibrium Fuel

Long burnup over many cycles will increase the heavier isotopes of plutonium as fuel is recycled. This continues until an equilibrium isotopic composition of fuel is reached. This equilibrium composition is dependent upon reactor design and upon the fuel feed material used to replenish core burnout. In general, breeding ratio improves with increase in the heavier plutonium isotope content because of the favorable nuclear properties of plutonium-241 (Table II) and because neutrons are not required to produce new net plutonium-240 and plutonium-241. The improvement in breeding ratio may be 5 or 10 percent as the plant operator recycles the fuel many times to utilize all the uranium-238 by fissioning all isotopes of plutonium which are produced.

Uranium Utilization

The utilization of a great fraction of the uranium mined is an important motivation for developing fast breeders because it is only through operating such breeders that a large fraction of uranium's potential energy can be utilized.

The maximum fraction of uranium (U_t) which can be fissioned in a reactor with a conversion ratio (c) less than unity is:

$$U_t=F/(1-c)$$

where F is the fissile fraction of the uranium as used, and recycling is assumed accomplished without loss of fuel.

Natural uranium has a fissile fraction of 0.007, of which about 0.002 is left in the gaseous diffusion plant waste, and F has a value near 0.005 for water reactors now in use. Thus, for today's thermal reactors with conversion ratios of 0.5, only one percent of the total uranium can be fissioned—even assuming plutonium recycle.

However, if the conversion ratio or breeding ratio is above unity (as would be the case for the breeder reactor) all of the core's uranium can be fissioned except for that lost during chemical reprocessing and fabrication of fuel. If f is the fraction of burnup per cycle averaged over all fissile-plus-fertile material in the reactor and L is the loss fraction per recycle, the fraction of uranium (U_t) which can be fissioned is:

$$U_t=f/(f+L)$$

For a core burnup of 100,000 MWd/t, the average burnup of fuel per fuel recycle (including the blanket) might be 30,000 MWd/t, or about three percent. If the loss per reprocessing and refabricating cycle is held to two percent, then 60 percent of the natural uranium can eventually be fissioned—a great improvement over the one percent fraction now achieved with water reactors. An increase of a factor of 60 in available economic energy thus constitutes a worthwhile objective for the fast breeder reactor development program.

By using thorium as a fertile material in a breeder in a similar manner, 60 percent of the energy potentially available from thorium ores can be added to that available from uranium. Since thorium may be more abundant than uranium, the available energy is more than doubled, providing further motive for developing breeder reactors.

Core Design and Performance Considerations of Fast Breeder Reactors

James H. Wright

The task of developing a truly competitive breeder reactor requires a careful selection and combination of design variables. Good economic potential and short doubling time are both served by high specific power and high breeding ratio. Material capabilities and safety aspects become the top considerations, leading to constraints on the parameters available to the designer.

All fast breeder concepts that have been considered to date probably have two things in common: higher capital costs than water reactors and lower fuel costs. Thus, there is considerable incentive for achieving the lowest possible fuel costs; otherwise, the breeder reactor may never become an economic addition to nuclear power generation. Design parameters that influence fuel costs become the key considerations. As will be shown, *breeding gain* and *specific power* are the real determinants of the plutonium fuel cycle cost. Therefore, these become the most significant design considerations in satisfying the economic objectives of the fast breeder reactor.

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Doubling Time

Assuming a steadily increasing demand for plutonium, the excess fissile material (breeding gain) produced by one breeder would be reprocessed, refabricated, and inserted in a new breeder as soon as possible. Thus, if plutonium dividends are continuously reinvested in new breeders that have the same rate of return as the original breeder, the doubling time is subject to compounding. Simple doubling time, on the other hand, does not include this "interest" on dividends. Doubling time can also be expressed either with or without out-of-core inventory. The broad definitions that cover the more commonly used expressions for doubling time are summarized in Table I.

Further refinements, such as *average bred* plutonium content, the *time-composition function* to reach equilibrium, reprocessing losses, etc., can be incorporated to improve accuracy, but these are more difficult to apply and require much more detailed information on the fuel cycle, thus losing the desirable characteristic of a simple and easily applied criterion for breeder performance evaluation.

Definition (4) in Table I is a particularly useful and fundamental criterion for breeder performance. Out-of-pile inventory is included because this most accurately reflects the *total* plutonium investment and the compound expression most accurately depicts plutonium gain when breeder reactors are continually being built in large numbers.

Breeder Fuel Cycle Economics

Breeder doubling time is more than a statement of plutonium gain potential; it is the index to the economic potential of the fuel cycle. Core doubling time determines the rate of excess plutonium production from a breeder, and results in a fuel cost credit as shown by the following approximate expression:

$$\text{Net Pu Credit (mills/kWh)} = \frac{0.045(BR-1)}{\epsilon} \times (\text{Pu Value, \$/g})$$

where $(BR-1)$ is the breeding gain and ϵ is the plant thermal efficiency.

Core inventory, the other major factor in plutonium economics, is expected to be the largest single component in the breeder fuel cycle cost. Inventory costs must include the in-core inventory, out-of-core inventory, and the inventory of

Table I—Doubling Time

Doubling Time Expression	*Approximate Formula*	*Example**
1) Simple Doubling Time, Core Inventory Only	$SDT^{**}_{\text{core}} = \dfrac{3.2}{S\,(BR-1)}$	8.0
2) Simple Doubling Time, Including Out-of-Core Inventory	$SDT_{\text{total}} = \dfrac{3.2}{S\,(BR-1)}\dfrac{(F_T)}{(F_c)}$	10.7
3) Compound Doubling Time, Core Inventory Only	$CDT_c = \dfrac{\ln 2}{\ln\left(1+\dfrac{1}{SDT_c}\right)}$	5.9
4) Compound Doubling Time, Including Out-of-Core Inventory	$CDT_T = \dfrac{\ln 2}{\ln\left(1+\dfrac{1}{SDT_T}\right)}$	7.8

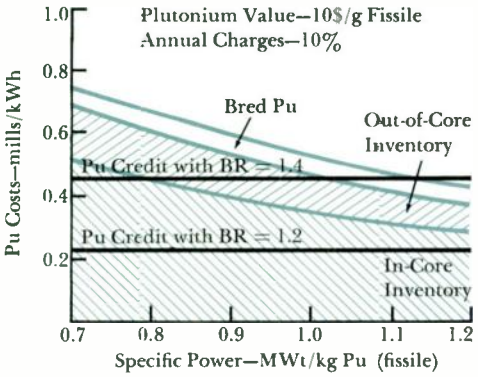
Where S = Specific power, MW Thermal/kg Fissile Pu in Core
BR = Breeding Ratio
F_T = Total Fuel in Cycle
F_c = Fuel in Core

*The example uses the same reactor design in all four cases:
Specific power (S) is 1 MW thermal/kg fissile Pu; breeding ratio (BR) is 1.4; and refueling cycle is ⅓ of core and blanket annually.

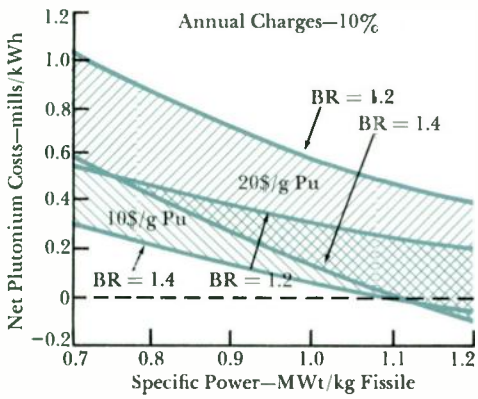
**Explicit form is:

$$SDT = \frac{2.6}{LF\,S\,(BR-1)}\left(\frac{1+F}{1+\alpha}\right)$$

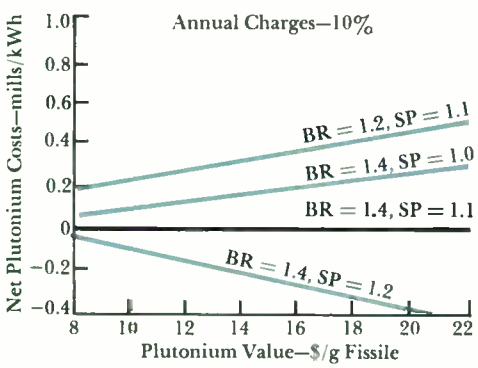
where F is fast fission effect in U-238 and α is capture to fission ratio in Pu. For simplifying formula, load factor (LF) is assumed 80 percent and $(1+F)/(1+\alpha) \sim 1.0$.



1—Plutonium costs (mills/kWh) are the sum of the in-core inventory, the out-of-core inventory, and the bred plutonium that is retained in the core.



2—Net plutonium cost is the difference between plutonium costs (in-core, out-of-core, and bred) and plutonium credit.



3—Net plutonium costs are a function of plutonium value, as determined by breeding ratio and specific power.

bred plutonium as defined in Table II. Total inventory cost is an inverse function of specific power, making it sensitive to some of the same design variables affecting the doubling time.

The relationships among inventory costs, plutonium credit, and specific power are shown graphically in Fig. 1. Plutonium credit is directly proportional to breeding gain ($BR-1$). (Plutonium was valued at 10 \$/g for this graph, and the inventory cost for non-depreciating assets was assumed to be 10 percent per year.) The *net plutonium cost* to the cycle is the difference between total inventory cost and net plutonium credit and is shown in Fig. 2 as a function of core specific power. The assumptions used in this graph include both 10 and 20 \$/g plutonium and indicate clearly advantages of both higher specific power and higher breeding ratio.

Similar curves can be drawn to determine net plutonium fuel-cycle costs as a function of plutonium value as illustrated in Fig. 3. Here, it is interesting to note that for a breeder having a specific power of 1.1 MWt per kilogram of fissile plutonium inventory and a breeding ratio of 1.4, the net plutonium fuel cycle costs are essentially zero for all values of plutonium. This corresponds to a high-gain breeder having a doubling time of 7 years according to definition (4) in Table I.

It is important that the breeder fuel cycle be examined over a broad range of plutonium values because the value of plutonium will, undoubtedly, increase. The two factors influencing this are (1) the greater value of plutonium in a fast breeder fuel cycle compared to plutonium recycle and (2) the possibility of escalating uranium prices.

A reasonable estimate of future plutonium value based on supply and demand is given in Fig. 4. The lower curve is based on a constant price of uranium of \$8 per pound and the upper curve assumes uranium cost increases of 50 cents per pound each year after 1980, beginning at \$10 per pound in 1980. The change in plutonium value with constant uranium price for the next few years is the result of:

1) Government price support of \$10 per gram through 1970;

2) a fluctuating but minimal supply and demand situation through 1972, during which research and development needs will be high compared to supply;

3) price stabilization with the beginning of commercial plutonium recycle in water reactors in 1973; and

Table II—Inventory Costs

In-Core Inventory (80% load factor):

$$I_c = \frac{1}{S} \times \frac{1}{\varepsilon} \times (x)\,(y)\,\frac{10^3}{7000} \text{ mills/kWh.}$$

S = Specific Power MWt/kg Pu Fissile
ε = Thermal Efficiency
x = Pu Value, \$/g Fissile
y = Annual Charge for Non-Depreciating Assets
10^3 = Mills/\$
7000 = Hours/Year at 80% Load Factor

Out of Core Inventory:

$$I_o = I_c\left(\frac{F_T - F_c}{F_c}\right) \text{ mills/kWh}$$

F_T = Total Fuel in Cycle
F_c = In-Core Fuel

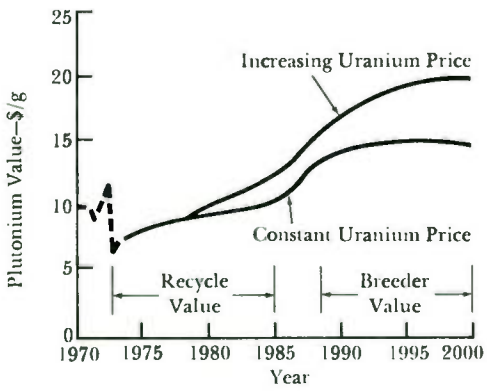
Inventory of Bred Pu:

I_B = Function of detailed fuel cycle characteristics influenced by residence time, burnup, breeding ratio, fuel relocation, etc.

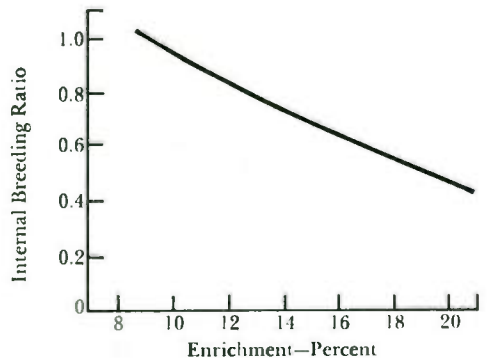
$$= \frac{0.12 \times 10^{-3}\,(BR-1)\,(1-\alpha)\,(x)\,(PF)}{(1+F)\,(\varepsilon)}$$

Total Inventory Costs:

$$I = I_c + I_o + I_B \text{ mills/kWh}$$



4—Future estimated plutonium values.



5—The internal breeding ratio in the enriched core is a function of plutonium enrichment.

4) price appreciation through 1980-85 as technology and volume in plutonium recycle are improved.

After 1985, a strong and growing demand for plutonium for use in fast breeders may equal or exceed the supply by 1990, a situation that will continue through most of that decade.

The upper curve shows the compounding effect of normally rising uranium prices on plutonium value with the same supply-demand criteria applicable to the lower curve.

It is reasonable to expect that plutonium values will reach $15-20 per gram by the time the fast-breeder fuel cycle becomes a significant part of utility system economics. For these conditions, only a high-gain breeder can have sufficiently attractive fuel costs as noted previously in Fig. 3.

From the foregoing discussion, it is evident that plutonium costs must be minimized by designing the breeder with high specific power and high breeding gain. To these plutonium costs, however, must be added the other fuel-cycle costs such as core fabrication cost, shipping and reprocessing charges, losses, etc. These other costs can be held to a minimal value through long-life, high-burnup fuel performance.

Design Considerations Affecting Breeding Ratio

Since fuel performance is related to the ratio of fissile and fertile fuel to other material in the core, the ideal core composition would be limited to only uranium and plutonium atoms. Metal cladding and structural materials downgrade performance in two ways—these materials soften the neutron spectrum by elastic and inelastic scattering, and they absorb neutrons.

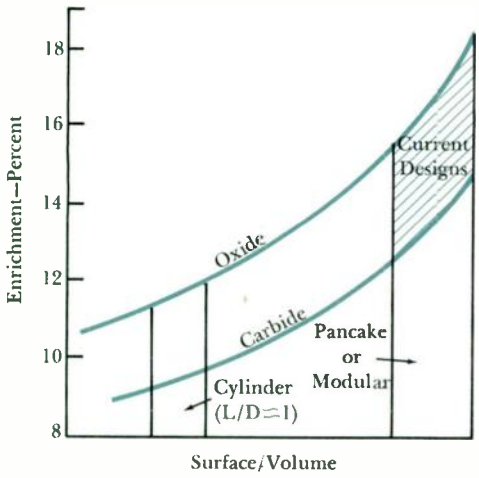
Spectral hardness is also affected by the composition of the fuel material. For example, the reduction in fissile and fertile atom density and increased scattering that occurs when fuel in the form of uranium-plutonium dioxide is used causes a reduction of the ideal or theoretical breeding ratio by as much as 45-50 percent; this dilution of fuel composition and the resulting reduction in breeding ratio requires a compensating increase in the ratio of plutonium (fissile) to uranium (fertile) material to maintain criticality for the desired fuel lifetime. The use of more dense fuel compositions, such as carbide or nitride compounds, reduces this downgrading effect.

One of the chief detractors from breeder performance is the moderating effect of the coolant on the fast neutron population. In this regard, cooling with helium gas is the least detrimental and steam cooling the most, with liquid sodium somewhere between.

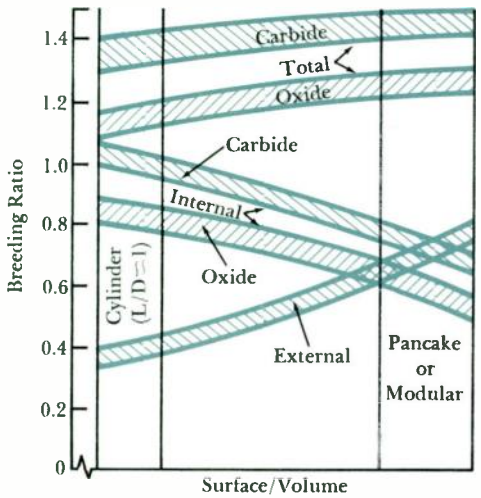
It is conceivable that breeder reactors will eventually be built in the United States using each of these coolants. Today, however, the sodium-cooled systems appear the most attractive for many reasons; furthermore, they are most likely to achieve early success because of their advanced technical position and because of large domestic and international development programs under way on this type. Similarly, ceramic fuels (oxide, carbide, or nitride compounds) appear to be the almost unanimous choice for breeder fuel, even though extensive and somewhat encouraging results have been obtained with metallic fuels. The advantage of ceramic fuels is that they offer the best potential for achieving high burnup with minimum physical (dimensional) change.

Within the framework of sodium-cooled ceramic-fueled breeders, the plutonium enrichment required to sustain the chain reaction is one of the most important parameters to breeder performance. When plutonium loading requirements go beyond the 8-10 percent minimum requirement, two adverse effects are noted on breeder performance:

First, the internal breeding ratio is reduced, as shown in Fig. 5. This occurs because the opportunity for interaction between the neutron flux and the fertile uranium atoms (U-238) is reduced as the concentration of uranium is reduced. Thus, breeding gain is reduced and a reactivity drop with burnup results. This latter effect requires additional initial plutonium loading to provide reactivity through the lifetime of the fuel charge. The excess reactivity at the beginning of core life makes reactivity control more



6—Current fast-breeder reactor designs are pancake or modular and require plutonium enrichment of 14 to 18 percent; optimized cylinder designs could reduce enrichment requirements to 10 to 12 percent.



7—The total fast-breeder reactor breeding ratio is the sum of the internal (enriched core) breeding ratio and the external (blanket) breeding ratio. Present modular or pancake core designs require large blanket regions to obtain the necessary total breeding ratio.

difficult over the total core life.

Second, the fissile plutonium specific power rating (MWt/kgPu) of a fuel having a defined upper thermal limit is reduced as plutonium is substituted for U-238. This occurs because the plutonium loading is increased but the thermal performance of the fuel rod is held constant at its maximum level (bigger denominator, same numerator).

Thus, increasing plutonium loading adversely affects specific power, breeder ratio, doubling time, and therefore economics. Why, then, should one not use a core design that would permit the minimum enrichment of plutonium? The answer, which is transitory in nature, is related to the present understanding of safety characteristics in large breeders. A large breeder core, cooled by sodium, will exhibit positive sodium temperature and void characteristics if designed for minimum plutonium content; this is because (to use the minimum plutonium content) it is necessary to minimize neutron leakage by minimizing surface-to-volume ratio. In such a core the internal breeding ratio is high but the sodium provides significant spectral softening during normal operation. For a loss of sodium, the neutron energy spectrum hardens, and reactivity increases. Present technology has not determined exactly how much reactivity increase from this source can be satisfactorily dealt with by other offsetting safety mechanisms (Doppler coefficient, thermal expansion, fast acting rods, etc.), nor how much reactivity increase will actually occur. Therefore, it becomes prudent to avoid the problem at this time by designing cores with higher neutron leakage resulting in negative sodium coefficients, which require higher plutonium loading.

A graph of the enrichment requirement as a function of surface-to-volume (leakage) ratio is given in Fig. 6. As shown, current designs require 14 to 18 percent plutonium loading (less enrichment being required for the carbide fuel). A general relationship of breeding ratio to surface-to-volume ratio (leakage) is shown in Fig. 7. The total breeding ratio is the sum of breeding in the core (internal) and the blanket (external). Although this figure indicates that the loss of internal breeding ratio can be compensated by additional breeding in the blanket, the blanket must be much larger and more costly; and the increased enrichment required for this abortive design reduces the specific power drastically.

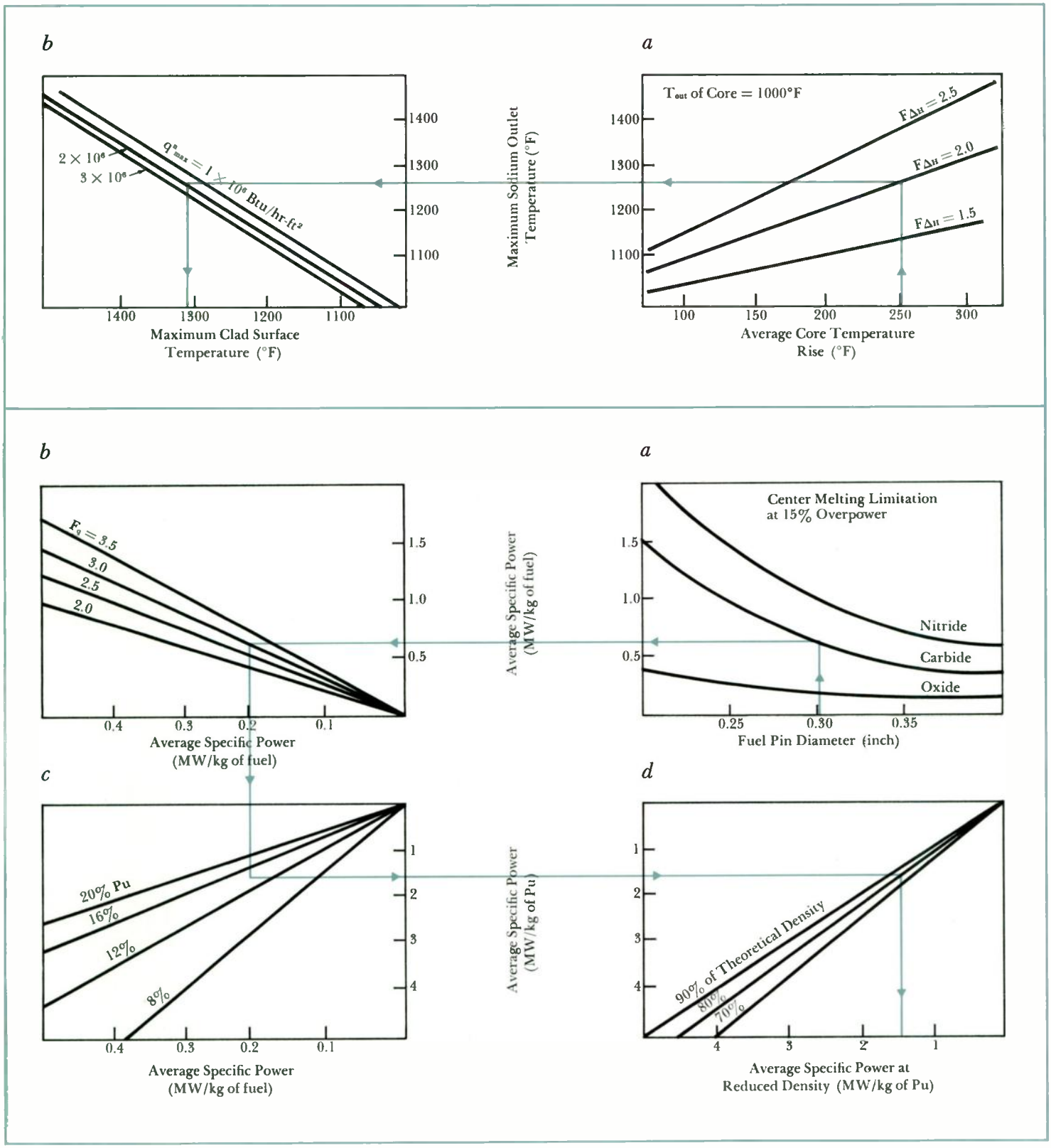
Thus, it is both likely and imperative that the necessary answers to the safety questions be forthcoming so that breeder design can be optimized. Eventually, the answer must be found in the overall power coefficient rather than in just the sodium temperature coefficient.

Design Considerations Affecting Specific Power

The specific power design of the breeder is subject to all of the usual materials constraints—fuel center temperature limit, maximum allowable clad temperature, fuel swelling and gas release, irradiation embrittlement of clad and structural materials, etc. The interrelationships between these limitations and the various core design parameters are much too complex to be considered here in detail. However, by way of illustration, the influence of two of these limitations, fuel-clad temperature and fuel-center temperature, will be discussed.

Although ceramic fuels have found widespread use in the growing nuclear power systems throughout the world, the performance demanded of these fuels has been modest compared to fast-breeder fuel requirements. Specific power and burn-up requirements for breeder fuels will exceed the performance previously demanded of nuclear fuels by a factor of three or four.

When fission heat is liberated within a ceramic fuel pellet, the temperature of the fuel must rise until all of the heat generated is conducted out of the fuel, through the cladding, and into the coolant. The temperature rise within the fuel is dependent upon the amount of heat being generated, the amount of fuel through which the heat must be conducted (pin diameter), the thermal conductivity of the fuel clad, the gap between the fuel and clad, and the heat transfer coefficient and bulk temperature of the reactor coolant.



With liquid sodium used as reactor coolant, excellent heat transfer from clad to coolant exists, but care must be taken to prevent the sodium temperature from rising to its boiling point. The clad material (usually 316 stainless steel) adds further constraints on the temperature profile by imposing maximum clad temperature limits. This limit for stainless steel is thought to be in the range of 1300-1400 degrees F and is determined, in part, by the temperature-dependent rate of nickel removal by sodium.

With a fixed maximum clad temperature, the reactor sodium temperature must be well below this point. The relationship between average sodium outlet temperature, average sodium temperature rise, deviation of hottest part of core from core average (hot channel factor), and maximum clad temperature is shown in Fig. 8. For example, with an average outlet sodium temperature of 1000 degrees F, a sodium temperature rise of 250 degrees F, a hot-channel factor ($F\Delta_H$) of 2.0, the maximum sodium temperature is 1250 degrees F for an inlet sodium temperature of 750 degrees F (Fig. 8a). With a sodium flow of 3×10^6 pounds/hr-ft^2 and a heat flux of 2×10^6 Btu/hr-ft^2, the maximum clad temperature will be 1310 degrees F (Fig. 8b).

To minimize the temperature rise across the clad to fuel pellet gap, this gap is filled with sodium.

The temperature rise from the outside to the center of a cylindrical fuel pellet is given approximately by the formula,

$$\Delta T_{\text{pellet}} = \frac{Q}{4k} r^2$$

where ΔT is the temperature difference between the outside and center of the pellet, Q is power density in the fuel (Btu/ft^3-hr), r is the radius of the fuel pellet, and k is the average thermal conductivity of the fuel material.

The thermal conductivity of many ceramic materials, including uranium and plutonium compounds, changes with temperature; therefore, a simpler expression, kilowatts (thermal) per foot of rod length, is often used to describe thermal performance capability of fuel rods.

If center fuel melting is to be avoided for all fuel during normal operation, a design basis for steady-state operation must be established that is well below the melting point. One such criterion might be to allow center fuel melting only at 15 percent over-power. In such a case, the maximum thermal rating per unit length of fuel rod can be established. The specific power of the total fuel then becomes a function of the fuel pin diameter. A plot of this relationship is shown in Fig. 9a for three ceramic fuel materials.

An example will illustrate the application of the curve of specific power to fuel-rod diameter. For a mixed carbide fuel with a fuel pin diameter of 0.30 inch, the maximum specific power, limited by the center melting criteria previously described, is 0.65 MWt per kilogram of *total* fuel material (Fig. 9a). This maximum specific power, divided by F_q, the heat generation hot channel factor (Fig. 9b) gives an *average* specific power for the fuel of 0.2 MWt. To get specific power per unit of fissile plutonium, average specific power of the rod is divided by the plutonium content of the fuel (Fig. 9c) to get 1.5 MWt per kilogram of *plutonium*. This value is corrected for the applicable fuel density (Fig. 9d).

At this time, all of these relationships are under development and therefore are being continually revised as more is learned in experimental work. For example, hot channel factors are empirical in nature, and vary from one reactor design to another with such factors as total size, core geometry, and fuel assembly design. Hence, a full-sized prototype fuel assembly can provide accurate values for a significant part of the overall performance evaluation.

Also, there are many other constraints on fuel pin diameter in addition to center temperature, which may turn out to be the controlling factors; for example, fuel swelling or gas release, because of the extremely high burnups that are being sought, may become limiting factors. However, the relationships illustrated by these curves point out the desirability of high fuel density, high thermal conductivity, low plutonium enrichments, and low hot-channel factor in obtaining larger, more economic fuel pin diameters.

Further Development

Fuel-center temperature and fuel-clad temperature are only two of many material constraints on the breeder thermal performance. Many others exist and each defines an exclusion envelope of parameters not available to the designer. Thus, one of the most critical aspects of attaining really high thermal and neutron performance will be to establish more favorable parametric boundaries, thereby making possible more economically favorable designs. Further study in the comprehensive three-year development program now under way will provide more accurate data from which many of these empirical relationships can be refined and proof-tested in detail.

Although much has been and will be done in the development laboratories, the next major step will be the design, construction, and operation of a fast breeder reactor prototype. A large-scale prototype breeder system should be built to demonstrate the technical features of full-scale fuel assemblies and plant components prior to embarking on the first commercial breeder plant.

Water reactors are setting nuclear power generating targets that will be difficult for any new system to meet or improve upon. The sodium-cooled high-gain breeder reactor presently offers the most hope for meeting these targets. In addition, it can provide in conjunction with today's water reactors a *total* nuclear fuel cycle that will be economically attractive through the turn of this century and many years beyond.

8—Maximum clad surface temperature is a function of average core temperature rise, hot-channel factor, and heat flux, as illustrated for a hypothetical case.

9—Large fuel pin diameters are desirable because they result in a core that is more economic to build; however, small pin diameters (*a*) produce higher values of specific power (MWt/kg of plutonium). The detrimental effect of hot channel factor (*b*), percent enrichment (*c*), and fuel density (*d*) on specific power of fissile plutonium is shown. Thus, the designer chooses the largest fuel pin diameter that will yield a satisfactory specific power.

A Prototype Plant Will Prove the Fast Breeder Reactor Concept

J. C. R. Kelly Jr.
P. G. DeHuff

A prototype fast breeder plant of 200 to 400 MWe can provide the information needed for the operational development of the reactor core and coolant system, and the fabrication of components for a full-sized plant. The development program now under way will permit a prototype plant to be committed to final design and construction by 1970.

A major step in the development of the high-gain fast breeder reactor concept is the design and construction of a prototype plant that will demonstrate the important features of the ultimate large commercial fast breeder. The first phase of this development, now under way, is the conceptual design of a 200-400 MWe plant that can be committed to detailed design and construction by 1970. For this prototype plant, the reactor will be a single-module core of the multimodule arrangement to be used in a full-scale plant. The coolant will be liquid sodium, circulated in one or two primary loops. The final coolant-loop arrangement will be determined as plant design progresses.

The prototype plant will be designed to attain several important objectives: Sodium coolant and steam conditions will be the same as those deemed necessary for a commercial fast-breeder plant; the fluid and mechanical systems will be similar to those projected for a full-scale plant; flexibility will be provided to permit modifications of core and internal structure as required during prototype operation; the components will be large enough to minimize future extrapolations to the large commercial plant, consistent with plant availability requirements; and the turbine cycle will have the modern temperature and pressure conditions necessary to produce low-cost power.

Development Goals for the Prototype

Development work is needed before the prototype is committed to construction because it is not possible to determine, on

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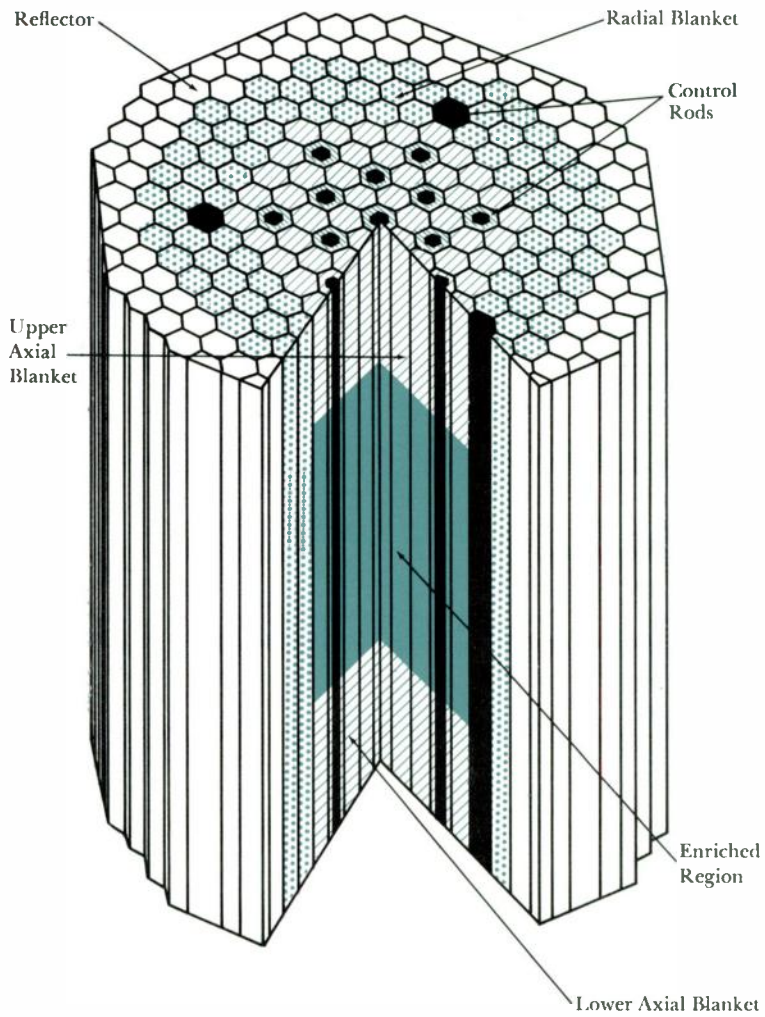


Prototype Core Data

Heat	500-800 MWt
Specific Power	123 kW/kg-metal
Plutonium Enrichment	14 Percent
Breeding Ratio	1.45
Doubling Time	<9 Years
Burnup	100 MWd/kg

1—Fast-breeder prototype core consists of enriched region surrounded by radial and axial blankets; control rods will be used for power regulation, to compensate for fuel deflection, and to shut down the reactor. The reflector region contains neutron flux within the core.

the basis of present technology, the exact nature and extent of the financial risk of the project. Even though the fast reactor design is based largely on prior demonstrated sodium- or water-reactor technology, there remain some key components and systems that require further development before the probability of success can be accurately assessed. These include:

1) Fuel assembly design, fabrication process development, and the irradiation of the mixed plutonium-uranium fuel.

2) Reliable sodium steam generator and other coolant-loop components within the range of satisfactory overall economics.

3) Techniques for refueling and fuel handling.

4) Fission product management systems and related maintenance procedures.

Solutions to these problem areas will require approximately two to three years of further development effort. A schedule for final construction will be formalized in accordance with the successful culmination of this work.

As discussed in the first article (p. 3), the economic advantages of the breeder cycle can be maximized if a full-scale fast breeder plant is available by 1980. To meet this schedule, the prototype plant should be operating in the mid-1970's so that design parameters can be optimized for the full-scale plant.

To have a prototype breeder ready in the mid-1970's, fuel fabrication for the first core will have to begin in late 1972 or early 1973. This provides some additional time beyond 1970 for completing the work of fuel development; on the other hand, the steam generator, because of the longer manufacturing lead time required, should be committed for fabrication shortly after the construction commitment is made in 1970.

Other engineering development work will be conducted before commitment of the prototype plant, for such items as sodium pumps, nuclear core design, and safeguard system. None of these are expected to limit the overall schedule.

The Prototype Reactor Core

The objective of the prototype 200-to-400 MWe core is to demonstrate one module of a 1000-MWe reactor. The following features will be practically identical to those of a core module for a full-size plant: Single-module core geometry (height, diameter, shape); fuel-rod and fuel assembly; thermal and hydraulic characteristics; and nuclear characteristics (except for multimodule coupling).

The prototype (single-module) core for a design shown in Fig. 1 contains an inner enriched region that is about four feet high and three and one-half feet in diameter, surrounded by a 13-inch-thick natural uranium radial blanket and a 15-inch axial blanket on top and bottom. The enriched region and blanket have a combined breeding ratio of approximately 1.4 to 1.5. In the enriched region, the fissionable plutonium enrichment is about 14 percent and the specific power will be approximately 120 to 125 kilowatts per kilogram of plutonium and uranium. The fuel is designed to heat the primary sodium coolant to approximately 1000 degrees F and achieve in later cores an average burnup of 100,000 megawatt days per tonne (MWd/t).

The core in Fig. 1 consists of 57 enriched fuel assemblies, each containing about 170 individual rods, and 98 radial blanket fuel assemblies, each containing about 90 individual rods. Some of the assemblies contain control rods as shown. The enriched fuel is a mixed uranium-plutonium carbide, in the form of cylindrical pellets, bonded by sodium to the steel tube. The carbide-fuel combination was chosen in preference to a plutonium-uranium oxide because of the excellent thermal characteristics of carbide, which will make possible a shorter doubling time. However, as a backup to the carbide development program, an oxide fuel will also be investigated for possible use in the first core loading. The advanced development work that has already been done with oxide fuel will make it possible to develop this composition sooner.

Fuel rods are held in their triangular array by a grid assembly that supports each tube. These grid assemblies are located along the length of the rods, forming a rigid structure that allows freedom for axial movement caused by differential temperature gradients. The upper and lower axial blanket sections of the prototype core consist of depleted uranium carbide pellets; the radial blanket consists of depleted uranium oxide pellets. All blanket pellets are clad in stainless steel. Control rods may be located in the radial blanket region to provide low-worth rods for power regulation.

The reflector section that surrounds the prototype core module consists of individual reflector assemblies, cooled by sodium flow. The reflector section improves neutron utilization and reduces radiation to structural components outside the core. Each reflector assembly is built up of Inconel plates to form a full or partial hexagonal cross section as required.

At equilibrium operating conditions, fuel will remain in the core for about three years. After each year of operation, one-third of the core and blankets will be replaced by new fuel. With an 80-percent load factor, the average fuel burnup will be 100,000 MWd/t.

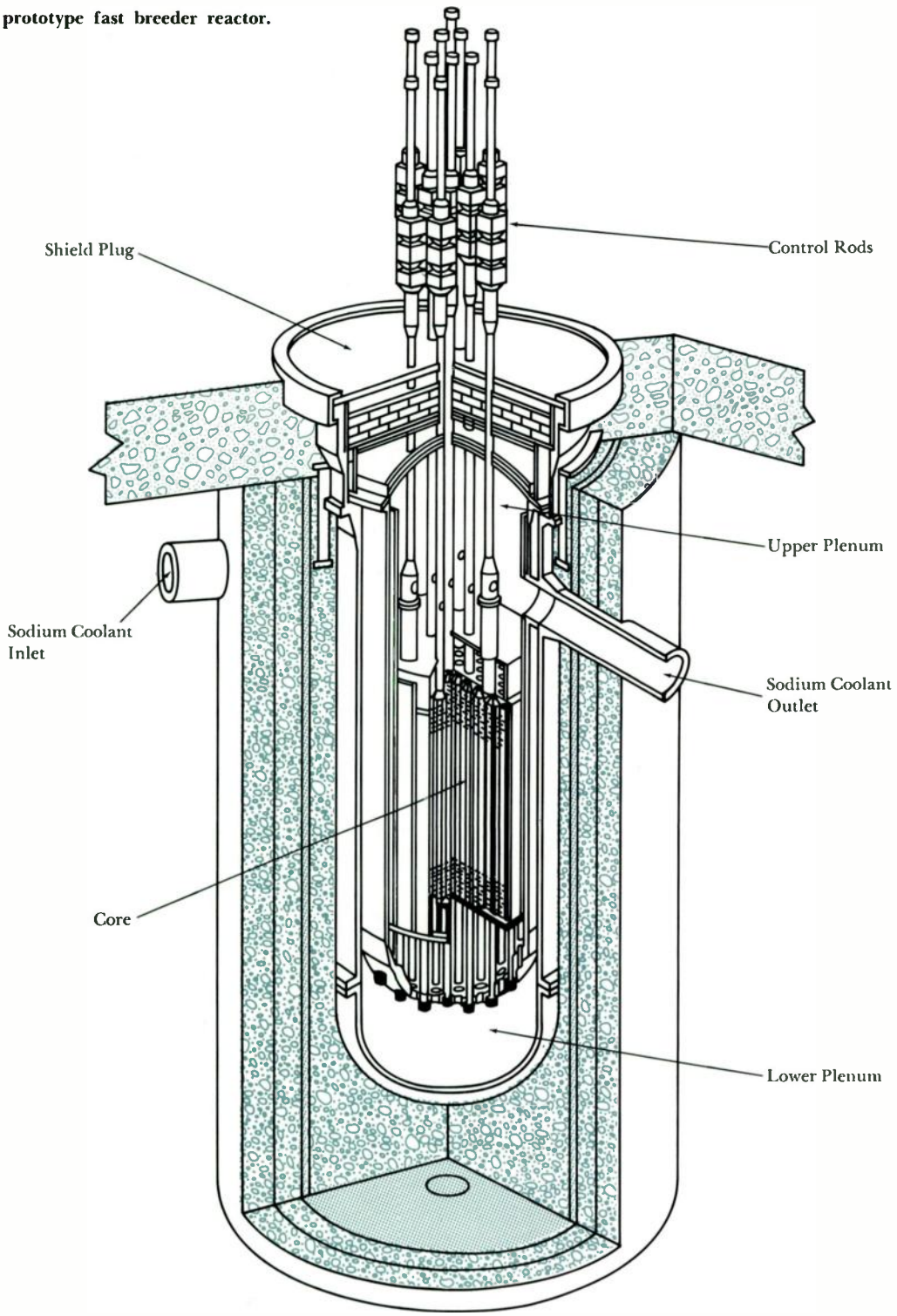
A cutaway view of the reactor vessel and core is shown in Fig. 2. The sodium coolant enters the vessel through a nozzle, flows downward through an annular region between the core barrel and the reactor vessel into a lower plenum, and flows upward through the core into an upper plenum region and exits through the outlet nozzle. As in pressurized water reactor practice, the core barrel is the main structural member.

The perforated domed structure at the top of the reactor vessel prevents excessive vortexing and turbulence of the coolant at the interface of the liquid sodium and cover gas by presenting a tortuous path to the fluid below. The argon cover gas above the sodium permits natural expansion of the coolant.

Sodium Loops

The prototype plant will employ conventional primary and secondary sodium heat-transport systems. The single-primary-loop plant shown in Fig. 3 represents neither the most optimistic nor the most pessimistic selection of the many

2—Cutaway of prototype fast breeder reactor.



design variables but serves as a reference design for future work. Further development work may indicate that a two-primary-loop arrangement will be desirable for the prototype plant.

The reactor heats primary-loop sodium from about 770 to 1000 degrees F; an intermediate heat exchanger transfers heat from the primary system to the secondary sodium system; and a once-through steam generator produces superheated steam at about 900 degrees F, 2400 psia. Turbine inlet steam conditions will permit a turbine cycle efficiency of 42 percent and an overall plant efficiency of about 40 percent.

Intermediate Heat Exchanger—The intermediate heat exchanger is a sodium-to-sodium shell-and-tube heat exchanger so designed that an entire tube bundle can be removed for maintenance and replacement without disturbing the shell, which contains the primary sodium. This intermediate heat exchanger should not be a severe development problem. Design uncertainties, such as unpredictable shell-side heat transfer, flow distribution, flow bypassing the tube bundle, and tube vibration should be eliminated in the preconstruction development effort.

Sodium-Circulating Pumps—Free-surface centrifugal pumps with gas seals around the shaft have been selected for the primary and secondary loops of the prototype plant. The basic concept of the vertical free-surface pump is well understood, but the largest size that has been operated with sodium is about 12,000 gal/min. Therefore, the development task is largely that of extrapolating existing designs upward to increase pump capacity by a factor of three to five and to develop larger shaft seals.

Steam Generator—Of all the major components required, the sodium steam generator appears to be the most critical from a technical (and perhaps economic) standpoint. A once-through steam generator has been selected because of its lower cost when compared with the equally acceptable recirculating-plus-superheater type. The steam generator will have sodium on the shell side; water will enter the steam generator tube bundle at the bottom through small headers, follow the tube flow path up through the sodium, and emerge as superheated steam at about 900 degrees F and 2400 pounds pressure.

The development problems in the steam generator are many. A satisfactory design will depend on successful completion of work in materials, design, and manufacturing. Materials must be suitable for both the sodium and water-steam environments; thus, they must resist stress corrosion attack on the water side and mass transport on the sodium side. Other problems include design of the tube-to-tube-sheet attachments, design of tube supports and thermal baffles to withstand the severe cycling and thermal shocks that will occur during operation, and development of economic manufacturing methods.

The unit must also be designed so that it can accommodate the possibility of a tube leak, and if one should occur, it must be possible to plug the defective tube.

Fuel-Handling System

The fuel-handling system is critical to a successful plant design because it must achieve a high degree of reliability in its performance so that it does not inflict extra down time.

Several types of fuel-handling systems have been advocated. Although no system has been fully proved in service, the hot-cell system has been selected for development because it appears to meet the most critical requirements of simplicity, plant availability, and maintainability.

The hot-cell system employs a large shielded cell, filled with an inert gas and located over the top of the reactor (Fig. 4). To refuel the reactor with this system, the shield plug is removed, completely exposing the surface of the core, which is submerged in the sodium coolant. The fuel-handling machine extracts fuel from the reactor sodium pool and transfers it for radiation decay. The fuel is then canned and transferred to a shielded cask for shipment to the processing plant.

The major development tasks are to provide specialized electrical, mechanical, and pneumatic mechanisms with a high degree of reliability to operate in the inert gas environment.

Sodium Contamination

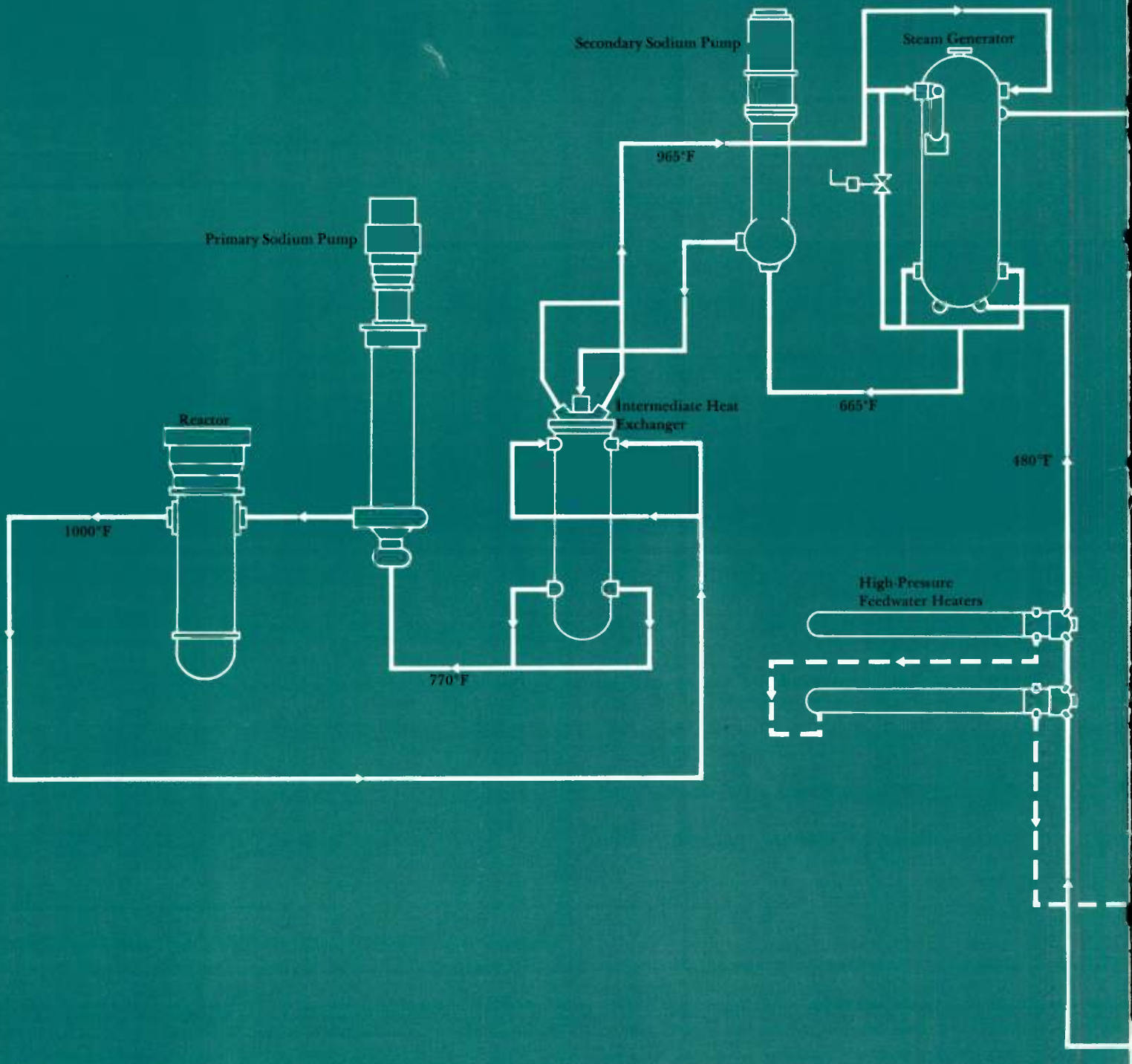
Any fast breeder plant that is cooled with sodium will have some radioactive contamination; the sodium itself becomes radioactive, and there will be mass transport of radioactive structural materials around the primary coolant loop. One of the prototype design concepts uses vented rods to provide escape for gases formed in the fissioning process. These radioactive fission gases either will be dispersed in the sodium coolant or will find their way to the cover gas above the reactor.

In the prototype design, radioactive fission gases in the cover gas will be removed by continously recycling the cover gas through a refrigerated activated-charcoal bed. Control of impurities in the sodium coolant will require development of purification systems analogous to the ion-exchange systems used in conventional water reactor systems.

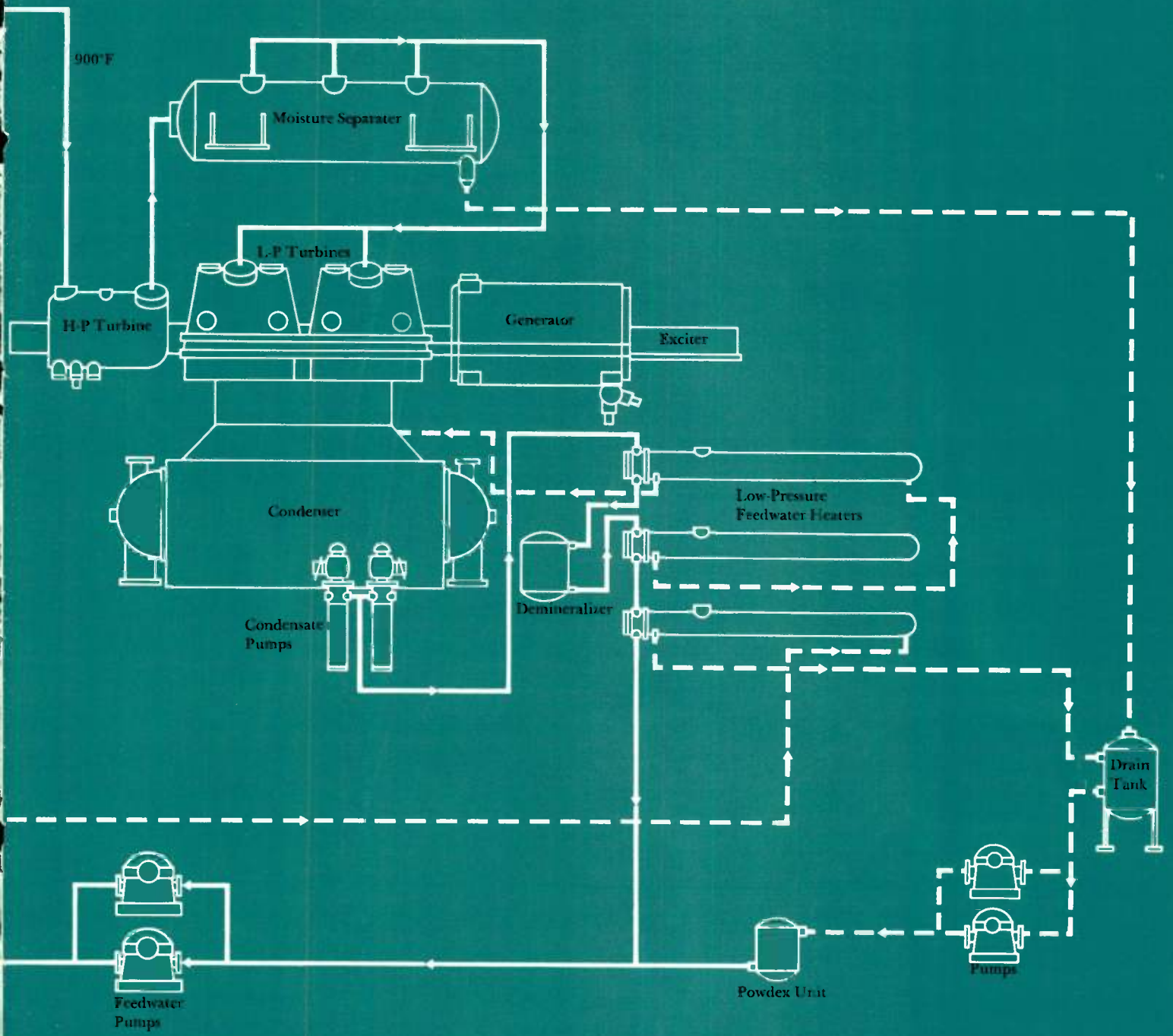
Impurities play a large role in sodium technology because of increased radioactive levels that can result, and because impurities may lead to the formation of plugs. The problem will be to identify and measure impurities quickly, and then control them. Identification and measurement might be accomplished by detecting oxygen in the sodium, which would indicate oxide content. Another method would be to measure electrical conductivity of the sodium, which is affected adversely by impurities.

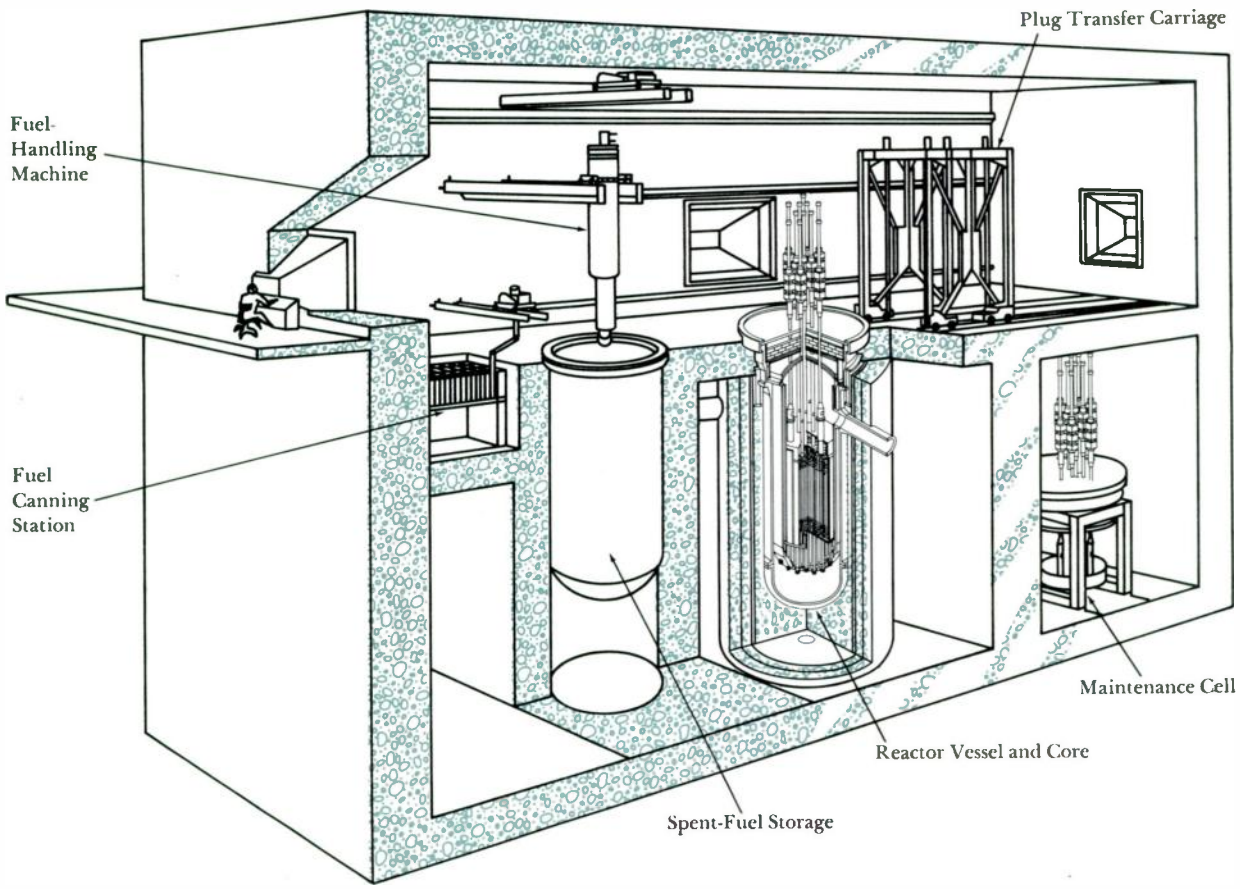
Possible purification systems are hot and cold traps. A hot trap would consist of a mesh of very active metal, such as zirconium, in a side stream from the main coolant loop. Since zirconium has more affinity for oxygen than sodium, the material coming out of the trap is stripped of oxygen by the chemical combination. This trap would also remove hydrogen because zirconium forms a hydride and has more affinity for hydrogen than does sodium.

A cold trap would consist of a side stream of coolant run through a heat exchanger where sodium temperature is dropped; cooling reduces the solubility of oxides in sodium, so that oxides precipitate out of solution.



3—Reference design for prototype fast breeder reactor plant has conventional primary and secondary sodium heat-transport systems.





4—Hot-cell system has been chosen for the fuel-handling system for the prototype fast breeder reactor. The remotely controlled fuel-handling machine operates in an inert gas atmosphere.

Many unknowns still remain, such as the fission gas release rate for the uranium-plutonium carbide fuel, and how this gas will disperse. However, designers now have a large background of information on how to approach the contamination problem, and they are confident that the fission products can be handled.

Conclusions

The breeder prototype plant is a vital final step in the overall program for developing commercial fast breeder reactors. Its construction and operation will provide certain key information not obtainable in any other way.

The design philosophy for the breeder prototype reactor can be summarized briefly: First, the plant should provide prototype designs for most of the fuel, core, component, and systems that will be used in future full-size commercial breeders; second, all prior experience from successful concept development in sodium and water reactors should be utilized wherever possible.

The technical areas requiring further development have been carefully reviewed and assimilated into an overall development program, with emphasis on the generation of information so that a prototype plant can be committed to design and construction by 1970.

The prototype plant will then provide critically important additional information needed through both the fabrication of components and the operational development of a complete core and coolant system. Information that can be obtained from operation of the prototype includes the reactivity lifetime characteristic of a carbide- or oxide-fueled fast breeder. The data obtained will provide a sound technical basis for further work in larger commercial-size fast breeder reactors with a minimum of technical and financial risk.

Light-Amplifying Picture Tube Used to Study Faint Stars

A television camera system with an ultrasensitive camera tube has been applied to an astronomical telescope for recording dim far-away objects in the universe with a sensitivity at least 10 times that of the photographic plates usually used. The system's SEC (secondary electron conduction) camera tube converts weak light into an electrical signal, amplifies the signal hundreds of times, and changes it back into a much stronger visible image. In addition, it can be used to build up even weaker light signals into bright images by accumulating the light collected over a period of time before releasing the signal.

This building-up process, called integration, is the mode of operation being used in astronomical research at the University of Pittsburgh's Allegheny Observatory, where the television camera is installed on the observatory's 30-inch refracting telescope. To orient the telescope, the camera is operated at the normal scanning rate of 30 frames per second. Then, to observe faint objects, the system is switched to the "integrate" mode for periods from 10 to 120 seconds. After the desired integration time, the strengthened image is read out within one or two frames (1/30 to 1/15 second) as a bright transient picture on the television monitor. During that period of bright display, a photograph of the object is made from the television screen.

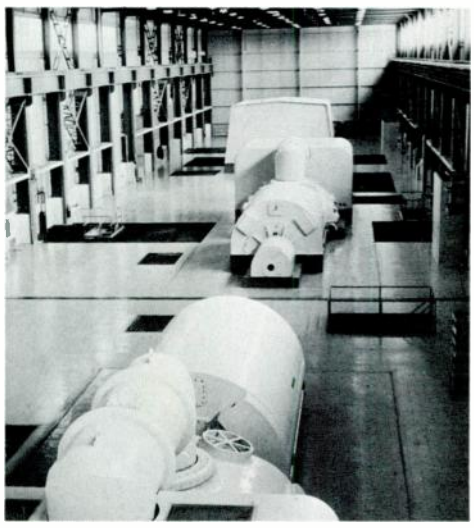
The greater sensitivity of the system over purely photographic techniques enables astronomers to record a faint astronomical object in a minute or two instead of the half-hour exposure needed on a photographic plate. Resolution is well within the standards required for good astronomical observations. And the red response is greater than that of photographic film, rendering the brightness of red stars and other objects more truly. Favorable results of the initial testing have led to the decision to use the SEC camera system for recording faint star spectra.

The SEC camera tube gets its ability to amplify light (as well as ultraviolet, x-rays, and other radiations) from release of electrons within a thin target film when the film is struck by the incident radiation. The target film has high electrical resistance, permitting signals to be built up and stored on its surface for hours without leaking away. Various types of SEC tubes are produced by the Westinghouse Electronic Tube Division, and complete SEC television camera systems are made by the company's Scientific and Electro-optical Department for such low-light applications as astronomy, product quality control, underwater photography, and security surveillance.

Flood-Proof Power Station

Not many power stations could operate in eight feet of flood water, but at least one can—the Holly Street Station of the Austin (Texas) Power Department. The reason for the design is that the station is located on a strip of riverbank land that has flooded in the past, and its basement is 20 feet below grade for economy in pumping water.

Double bulkhead doors are provided at all ground-level openings, and the structure has enough mass to prevent floating under the most severe flood conditions recorded at Austin. The station's boilers produce 800,000 pounds of steam an hour for each of its two 100,000-kW reheat turbine-generators and 1,300,000 pounds an hour for its 165,000-kW unit.



As much as eight feet of flood water could swirl around this Austin, Texas, power plant without disrupting operation.

Storage and Display System Stops Fluoroscope Action

A stop-action X-ray television system for medical fluoroscopy instantly freezes individual stored pictures on the TV screen at the push of a button. Or, if the physician desires, he can make a sequence of still pictures appear automatically at preset intervals, as in a continuous slide show. For both modes, the system automatically records the fluoroscopic TV pictures continuously and displays them on command, with X-ray exposure of the patient required only to change the picture.

The system combines the advantage of fluoroscopy—continuous and immediate viewing of the body's internal structures and functions—with the ability to let the doctor study an X-ray image as long as he wishes. While the stored image is on, no additional X-ray exposure is needed to keep it displayed. That feature, and the sensitive camera tube used, can reduce the patient's overall exposure to radiation 100 times or more.

The system also makes possible a new X-ray procedure called background cancellation. At the push of a switch, the last picture from the camera is stored and made to appear on the screen as a negative picture. Then, on the same screen, the real-time X-ray image in positive form is superimposed on the recorded image. By rotating a control knob, the two images are made to cancel each other to any degree desired. Anything introduced into the real-time picture then stands out in clear detail, uncluttered by unnecessary background. For example, if a radiologist wants to determine the outline of blood vessels in the head, he adjusts the negative and positive images so that they partly cancel each other. Then when dye is injected into the blood vessels, the vessels stand out in sharp contrast against a background having

only enough detail for orientation. Such cancellation can be achieved by purely photographic means, but generally the process is too long and involved to provide relevant pictures.

The system was worked out by the Westinghouse X-Ray Division and the Research Laboratories for use at Presbyterian University Hospital, University of Pittsburgh. It consists essentially of a sensitive TV camera, a magnetic recorder, electronic gating circuitry, a TV monitor, and a control panel. X-ray pictures picked up by the camera are fed to the recorder by way of the gate circuits, which are controlled from the control panel. The signals are fed to the monitor for display or to video recording equipment for a permanent record. Pictures are generated at the standard rate of 30 frames a second. When the control panel's "store" button is pushed, information recorded previously is automatically erased and one complete frame of video information is stored. A selector switch presents the automatic "slide-show" display of stored images at several frames a second or one every second, two seconds, five seconds, or 10 seconds.

Sign of the Times... Computer Controls Sign Lighting

The games computers can play have been known to only a few insiders until a Prodac 50 computer recently began playing a game for all the world to see. It controls an outdoor electric sign on Pittsburgh's North Side, facing the city's Golden Triangle across the Allegheny River. The sign departs radically from the simple—and sometimes irritating—on-off sequence of most flashing signs because it is programmed to be sophisticated, humorous, and challenging to the imagination. Thus, sequences credit the viewer's intelligence.

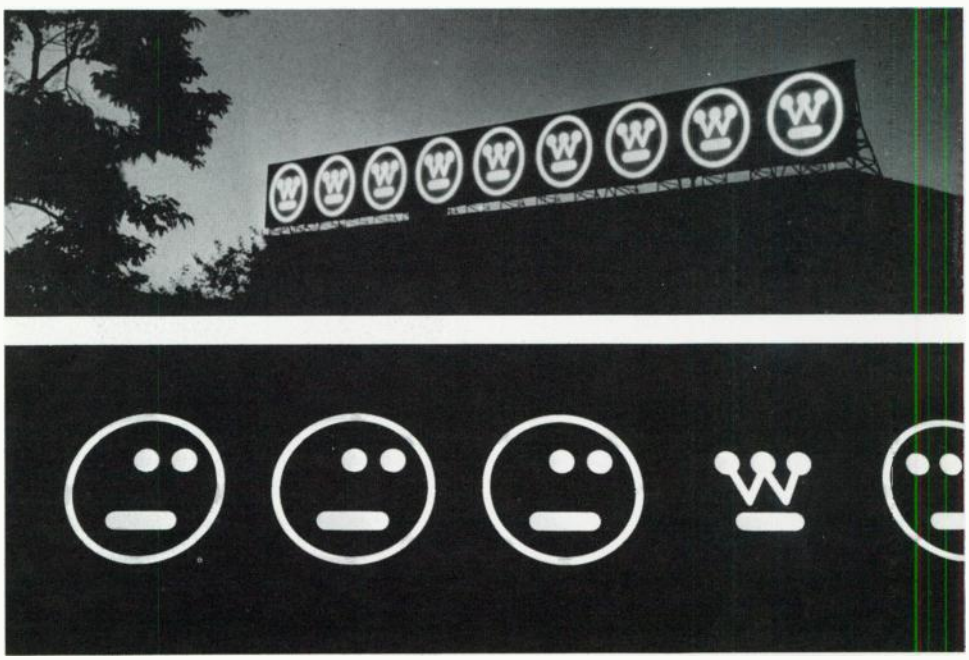
The sign consists only of a row of nine blue Westinghouse emblems. Each emblem measures 17.5 feet in diameter and is made up of nine geometric elements—the circle, three dots, four lines in the W, and the letter's underscore—for a total of 81 light circuits. This combination makes possible 81 factorial ways in which the lights can come on (81 times 80 times 79 times 78 and so forth), or 10^{120} ways, each completing the assembly of the nine trademarks. The lights are turned on in a number of sequences of instantaneous patterns at half-second intervals.

Each sequence of patterns is first programmed on sets of drawings. From these drawings, coding is generated on punch cards; the card data in turn is coded on magnetic tape and loaded into the computer. The computer transforms the data, which includes bit patterns and timing information, into its own language and stores the information in its 4000-word memory core.

Sequences are changed from time to time by reprogramming to introduce new and intriguing ways of assembling the nine trademarks. Because of the tremendous number of sequences available, they can be changed without repeating any that have been used before. Twenty different sequences were used initially in a nonrepeating cycle of five minutes. More can be added by enlarging the memory core, and space was provided for adding three more memory cores to permit use of nonrepeating programs more than half an hour long. Just as in industrial process control, the great memory flexibility of a digital computer permits program changes to be made easily as new sequences are devised.

Pumping a River Over a Mountain

Eleven 80,000-horsepower synchronous motors will help supply southern California with water from the north when the 444-mile California Aqueduct is completed by the state's Department of Water Resources. The motors will drive pumps that will raise water, collected from artificial lakes and the Sacramento and San Joaquin rivers, from the floor of the Central Valley to the top of the formidable barrier of the Tehachapi Mountains.



Computer-controlled sign consists of nine emblems, each made up of nine elements. With these 81 light circuits, the elements can be lighted in 10^{120} possible sequences ending with the row completely lighted, as at top. The control computer is programmed to assemble the emblems in sequences that are intellectually intriguing or humorous, as in the sequence at bottom where all but one of the emblems come on as faces that glare drolly at their nonconforming associate until even he gets with it.

From the summit, the water will flow southward to serve the counties of Los Angeles, San Bernardino, and Riverside through an aqueduct that will average 32 feet in depth and range from 110 to 252 feet in width.

Each of the huge motors will be 30 feet tall and 14 feet in diameter and will weigh 215 tons. The pumps they drive will move 1,800,000 gallons of water a minute through four tunnels to the summit of the mountain barrier—a lift of 1926 feet. Two m-g sets with wound-rotor drive motors will supply power to the pump motors. The installation, to be known as the Tehachapi Pumping Plant, will be located 29 miles south of Bakersfield; it is scheduled for completion in 1972. The giant motors and m-g sets will be manufactured by the Westinghouse Large Rotating Apparatus Division.

1000-Foot Deep Submergence Diving System Nears Completion

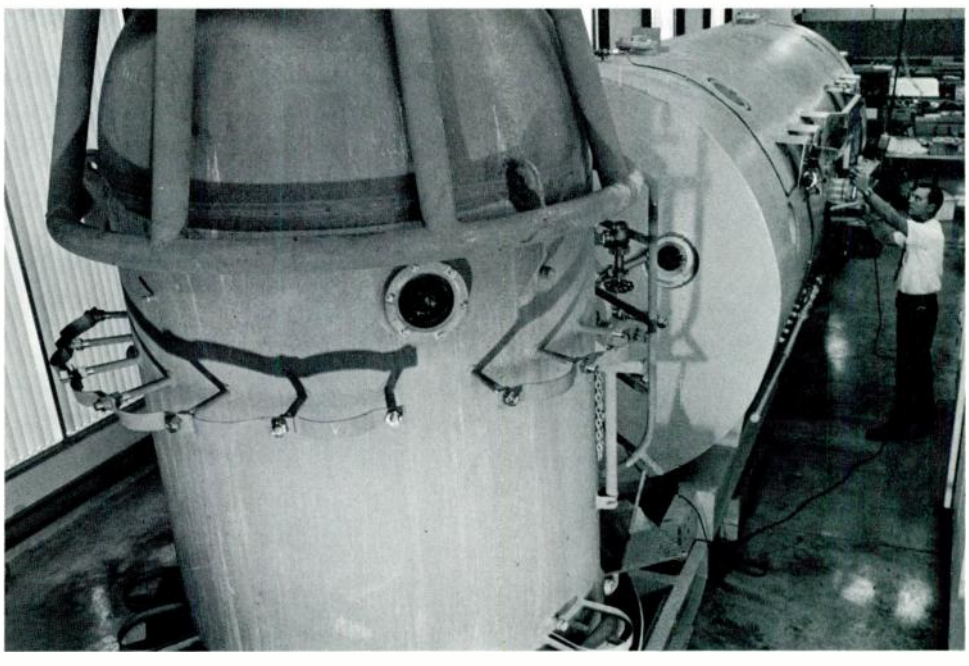
A diving system enabling divers for the first time to work for prolonged periods at water depths to 1000 feet is now being completed. The two-chamber system, called Cachalot-850, uses the principle of prolonged-submergence, or "saturation," diving—crews of four to six divers are kept at ("saturated in") working-depth pressure for a few days to two weeks. The divers live, sleep, and eat in a pressure chamber mounted on a surface barge between their working shifts. The other chamber of the system is a diving chamber through which pairs of divers are transported from the surface chamber to the work site. Air pressure in both chambers is kept equal to the water pressure at the working depth.

Prolonged-submergence diving is economically advantageous at depths below 150 feet. In conventional decompression diving at such depths, divers spend most of their time in decompression and relatively little (not more than an hour) at work. Prolonged-submergence techniques developed by the Westinghouse Underseas Division over the two-year history of the Cachalot-450, predecessor to Cachalot-850, have extended the actual working period for the divers to as much as six hours at a time.

The surface chamber includes an entry lock on one end and two internal compartments, all of which can be pressurized independently. In system operation, a two-diver team enters the diving chamber through a mated pair of pressurized transfer locks between the end of the surface chamber and the side of the diving chamber. When the divers close the lock hatches, the support crew separates the diving chamber from the surface chamber. The diving chamber is lifted from its pad, swung out, and lowered into the water by a crane. A heavy anchor makes the chamber sink, but it can be dropped in an emergency to allow the buoyant chamber to surface. While the chamber is being lowered, the divers don their equipment, which includes a special diving suit over which they place their breathing apparatus. The suit has internal tubes through which warm water is circulated to keep the diver warm.

When the diving chamber reaches the work site, the divers open the hatch in its bottom. Since the chamber is at working-depth pressure, no water enters. The divers leave the chamber through the hatch and connect long hoses, which supply their breathing mixture, to gas chambers on the outside of the chamber. Attached to each gas hose is a telephone line through which the diver can talk to and hear the surface support crew, an electric power line for lights if needed, an instrumentation cable, and the warm-water supply hose for the suit.

A special gas mixture is used in deep diving for two reasons. First, nitrogen, which makes up a high percentage of air, gives divers under pressure nitrogen narcosis—the "rapture of the deep"—so it must be replaced by another inert gas. Helium, although costly, is the safest known inert gas and is used most widely in deep diving. The other reason is that the oxygen of air also becomes denser under pressure, and too high a concen-



Cachalot-850 prolonged-submergence diving system is being outfitted for dives to depths of 1000 feet. The system consists of two chambers: a surface chamber, which is 27 feet long and 7 feet in diameter, and a diving chamber, 10 feet tall and 5 feet in diameter. The walls of the surface chamber are 1.25 inches thick, and those of the diving chamber are 1 inch thick. Both are formed from high-strength steel to withstand the high pressures encountered.

tration of oxygen is toxic. The greater the depth, the higher the percentage of inert gas that must be used in the mixture.

After working their six hours or less, the divers are brought back to the surface in the diving chamber, which is remated to the surface chamber. The men remove their equipment, clean up with a shower in the diving chamber, and reenter the surface chamber. Another team takes their place and descends to work.

Diving tables and techniques for maximum-depth use of the Cachalot-850 system will be developed in a new research facility at the Westinghouse Ocean Engineering and Research Center near Annapolis, Maryland. With this facility, consisting of three large pressure chambers, one of them partly filled with water, engineers and scientists will be able to simulate depths to 1500 feet. Research with this pressure system coupled with previous experience with the present Cachalot-450 system is expected to provide all the information necessary for 850-foot dives early this year and 1000-foot dives, if needed, by 1969.

Besides its use in prolonged-submergence diving, the Cachalot-850's diving chamber will also be useful for conventional decompression diving. It can be pressurized, on reaching working depth, from as low as one atmosphere to the working-depth pressure.

Products for Industry

Image intensifier tube provides brightness gain of 200 minimum, with low distortion and good resolution. The WX-30667 tube has a 40-mm fiber-optic input and a 25-mm fiber-optic output. Applications include low-light-level television, where it can be coupled to camera tubes with suitable input plates. In industrial and commercial X-ray applications, it can be used for direct-view image conversion. The intensifier is a single-stage electrostatically focused device. An image projected on the input causes emission of electrons, which are accelerated and focused onto an output phosphor by about 15 kV. *Westinghouse Electronic Tube Division, Elmira, New York 14902.*

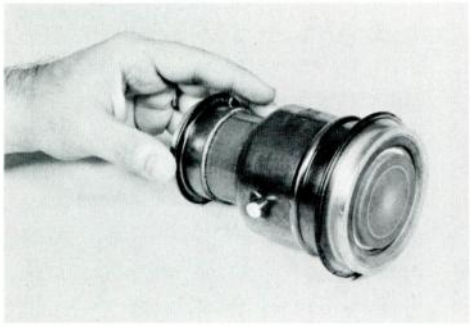
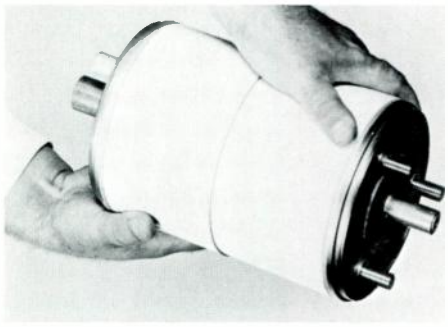


Image Intensifier Tube



Vacuum Circuit Interrupter

Vacuum circuit interrupter switch, rated 600 amperes continuous current, is capable of interrupting up to 12,000 amperes in a maximum of 0.018 second. The ceramic-insulated WL-23223 interrupter is designed for use on a 15.5-kV line. Applications are in power transmission and distribution equipment to remove short-circuit faults, high-voltage or high-current switching, transformer tap changing, capacitor switching, and motor controls. Mechanical life is 10,000 operations. *Westinghouse Electronic Tube Division, Elmira, New York 14902.*

High-pressure pneumatic transmitter—Optimac Model 332—accurately measures pressure of a wide variety of industrial gases and liquids, including steam. The pressure being measured is converted into standard air pressure signals that actuate local or remote indicating gages, recorders, control stations, or automatic controllers. Six different Bourdon tubes cover measuring ranges of 0 to 100 psig to 0 to 5000 psig; they can be interchanged and recalibrated by the user in a few minutes. Accuracy is guaranteed to plus or minus 0.5 percent and repeatability to 0.2 percent of the span of pressure being measured. *Westinghouse Hagan/Computer Systems Division, 200 Beta Drive, O'Hara Township, Pittsburgh, Pennsylvania 15238.*

IC voltage regulator combines a reference diode and sensing amplifier followed by Darlington-connected series regulator power transistors, all on a single silicon chip. The integrated circuit is a complete regulated power supply in a TO-3 power transistor package. By itself, the WM 110 T regulator provides regulation within 2 percent over a range of 8 to 48 volts output at currents to 2 amperes. If fed from a constant current source, regulation improves to within 0.2 percent; it can be further improved by using an integrated differential amplifier as a feedback element. *Westinghouse Molecular Electronics Division, Elkridge, Maryland 21227.*

About the Authors

J. W. Simpson, Electric Utility Group Vice President, directs the engineering, manufacturing, and marketing of basic electrical equipment for heavy industries. He graduated from the United States Naval Academy in 1937 and joined Westinghouse shortly after graduation.

The early part of his career, from 1938 to 1949, was spent in the Switchgear Division; during this period he earned an MSEE from the University of Pittsburgh (1941) and was granted a leave of absence to work at the Oak Ridge National Laboratory from 1945 to 1948. In 1949, Simpson transferred to the Bettis Atomic Power Laboratory as Assistant Manager of Engineering. He held successively more responsible positions there, was named Division Manager in 1955, and was elected a Vice President in 1958. At the Bettis Laboratory, which is operated by Westinghouse for the Atomic Energy Commission, he was responsible for the development of nuclear reactors for naval propulsion and power generation.

From this assignment, Simpson moved to Vice President and General Manager of the Atomic Power Divisions, which were involved in the development of commercial nuclear reactors as well as nuclear energy for space uses. He next became Vice President of Research and Engineering for the corporation, and in 1963 he was appointed to his present position.

Joseph C. Rengel, General Manager of the Atomic Power Divisions, has more than eleven years experience in pressurized water reactor technology, dating back to 1954 at the Bettis Atomic Power Laboratory. In 1955, he became project manager for the PWR Project, where he directed for Westinghouse the design, development, and construction of the reactor portion of the Shippingport Station.

Rengel is a graduate of the U.S. Naval Academy (1937), and he joined Westinghouse the same year. He subsequently held positions in the Marine Department and in the Transportation Department, and in 1952 he became assistant to the sales manager for the Defense Products Group.

After his assignments at the Bettis Laboratory, Rengel moved to the Atomic Power Division in 1959 as project manager for the development of a large commercial nuclear power plant. In 1961 he was made Manager of Advanced Development and Planning, and in 1962 he became Deputy General Manager of the Division. He was appointed General Manager in 1964. The division was split into three divisions in 1966 and Rengel was made Divisions Manager. He was elected a Vice President in 1967.

R. J. Creagan is Senior Consultant on J. C. Rengel's staff for the Atomic Power Divisions. He graduated from the Illinois Institute of Technology in 1942 with a bachelor of science degree in engineering. Creagan served in the U.S. Navy and was present at the Bikini A-bomb test. He next worked at the Argonne National Laboratory on the fast breeder EBR-1, earned his master's degree and his doctorate in physics at Yale University, and joined Westinghouse in 1950. He was sent to Argonne as a Westinghouse employe to work on the first criticals for the Nautilus reactor prototype. Creagan moved to the Bettis Laboratory in 1951 to head the group that did the initial reactor design studies for what later became the Shippingport Project. He next moved to the Atomic Power Department as Manager of Reactor Engineering where he managed the initial design and development of the Yankee reactor. In 1961, he was made Engineering Manager. He was appointed to his present position in 1966.

James H. Wright, like many engineers in the nuclear field, began his career in a totally different technical area and then migrated to nuclear engineering.

Wright graduated from Texas Technological College in 1948 with a degree in chemical engineering and promptly embarked on a career in the oil industry. He joined the Gulf Oil Corporation in their production division, where he served as a chemist and later as plant engineer and assistant plant superintendent. In 1952, he was awarded a petroleum fellowship at Mellon Institute of Industrial Research. During this period, Wright earned his master's (1954) and doctor's (1957) degrees in chemical engineering at the University of Pittsburgh.

In 1956, Wright joined the growing ranks of nuclear engineers; he became a fellow engineer in the reactor physics section of the Westinghouse Atomic Power Division. In 1958, he became Manager of Advanced Reactor Systems, where he had the responsibility for planning, implementing, and directing the reactor technology development program for advanced concepts.

Wright was appointed Technical Director of the Advanced Reactor Division when it was organized in 1966. Last year, he was appointed Senior Consultant on Rengel's staff in the Atomic Power Divisions.

John C. R. Kelly Jr. is General Manager of the Westinghouse Advanced Reactors Division.

He joined Westinghouse after receiving his Ph.D. degree in Physical Chemistry from Carnegie Institute of Technology in 1949.

His first assignment was to the Westinghouse Central Research Laboratories as an intermediate research chemist. In 1950, he moved to the Lamp Division to do research in metallurgy. He holds a number of patents in the fields of physical chemistry and extractive metallurgy of refractory materials.

Kelly returned to the Research Laboratories and advanced through a series of assignments to become Director of Materials Research and Development in 1964. He assumed his present position early in 1966. His Advanced Reactors Division has the responsibility for developing, designing, and marketing nuclear steam supply systems other than the pressurized water reactors.

Phillip G. DeHuff has been primarily concerned with technical management for many years, but his original training in metallurgy has served him well.

DeHuff graduated from the Department of Metallurgy of Lehigh University in 1940. His first job was with the Carnegie Illinois Steel Company where, among other things, he worked on armor plate problems. A year later he returned to Lehigh to take part in an industry-sponsored study aimed at producing better castings for ship propulsion units, and in 1942 he joined Westinghouse to work on similar problems.

In 1945, he moved to the newly formed Aviation Gas Turbine Division as head of the metallurgical engineering section and then of design engineering.

In 1955, DeHuff transferred to the Bettis Laboratories, where the Shippingport Project was just getting under way. Here he became manager of the section responsible for developing nuclear fuel elements for the plant. In 1956, he was made assistant manager of the project, with responsibility for reactor engineering.

DeHuff moved in 1959 to the Atomic Power Department, and in January 1960 was made Manager of the Carolinas-Virginia Tube Reactor (CVTR) Project.

From 1963 to 1966, he directed the design, development, and manufacture of the pressurized-water reactors. When the Advanced Reactor Division was organized in 1966, DeHuff was appointed Division Engineering Manager.

A turtle getting a patch test? No, it's a metal form being loaded with silicon wafers, each containing more than 20 power transistors in an early stage of manufacture at the Westinghouse Semiconductor Division. The loaded form is put into a vacuum chamber where a thin deposit of gold is evaporated onto the transistors to form contact areas for electrical connections.