Electrical breeding

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Some methods for breeding fissile fuel by means of electrically generated free neutrons are reviewed. These methods are of interest because rapid growth of the nuclear power industry may cause a near-term shortage of nuclear fuel. The outlook for auxiliary breeding schemes is discussed briefly. There is a brief analysis of the economics of electrical breeders.

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1. The rapid depletion of fossil fuels is forcing a progressively greater use of nuclear power, at least in the industrialized nations. Figure 1 gives an idea of how rapidly the known reserves of oil, gas, and coal may be depleted unless a new long-term energy source, which could only be nuclear energy, is developed and adopted widely.[1] The future growth of nuclear power entails not only an increase in the use of nuclear energy to generate electrical energy (at present, 20–30% of the electrical energy in the industrialized nations is generated from nuclear energy) but also the use of nuclear energy to heat cities, for large-scale (marine) transportation, and for industrial purposes: to generate heat on a large scale and to produce synthetic fuels and reducing agents, in particular, hydrogen. The wide variety of future production problems which can be solved only through the use of nuclear energy will mandate a diversified inventory of reactors, including both fast reactors and a large number of reactors of various types based on thermal neutrons.

Light-water reactors will apparently dominate the inventory for the next 20–30 yr despite their low thermal efficiency, their low fuel burnup (at present, they burn only about 0.5% of natural uranium), and thus their high uranium consumption rate. Since the nuclear power industry as it exists today is based on the burning of uranium-235, for which the reserves are small (uranium-235 amounts to only 0.71% of the natural mixture of uranium isotopes), the widespread use of light-water reactors might cause a shortage of uranium as early as the beginning of the next century. There would thus be a further increase in the cost of uranium, because the comparatively inexpensive uranium reserves are limited. (The curve for uranium-235 in Fig. 1 would be similar in shape and position to the curve for oil.)

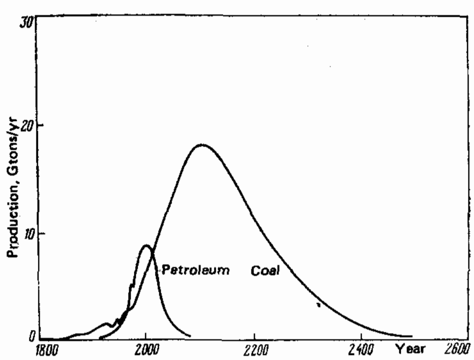

FIG. 1. Annual worldwide production of coal and petroleum [in gigatons (metric tons) of coal equivalent] in the absence of any other large-scale, long-term energy source. The total reserves of coal and petroleum (the areas under the curves) are estimated to be 5000 and 550 Gtons coal equivalent; the annual increases in production are 3.5% and 7%, respectively. The curves for gas and for uranium-235 are similar to that for petroleum.[1]

There are, of course, ways to increase the efficiency of the fuel cycle in thermal reactors: reprocess the spent fuel, recovering the unburned uranium-235 and the plutonium-239 generated during the cycle; use fuels denser than UO_2; switch from light to heavy water; etc. These approaches, however, while generally adding to the complexity and cost of reactors, fail to provide a fundamental solution to the problem of the nuclear-fuel balance. They can do no more than prolong the uranium-235 era for some 15–20 yr.

A fundamental solution to the fuel problem will come from an expanded production (breeding) of nuclear fuel in reactors with a conversion ratio greater than unity (breeders). Let us examine the various breeding schemes.

Fast-neutron reactors ("fast breeders") with a sodium coolant and an oxide fuel are presently being developed. The underlying idea is to make use of the plutonium-uranium cycle,

$$^{238}\mathrm{U} + \text{neutron} \xrightarrow{\gamma} {}^{239}\mathrm{U} \xrightarrow{\beta^-} {}^{239}\mathrm{Np} \xrightarrow{\beta^-} {}^{239}\mathrm{Pu},$$

$$^{239}\mathrm{Pu} + \text{neutron} \to \eta \text{ neutrons } + 150\ \mathrm{MeV} + \text{waste}.$$

Here η is the average number of neutrons emitted by a fissile nucleus per neutron-absorption event. This number may be regarded as the fuel breeding parameter, since the maximum possible conversion ratio is η minus the one neutron required to continue the chain reaction. In a thermal reactor with a good neutron economy we would actually need $\eta \approx 2.2$ for breeding (Fig. 2). This value of η reflects the need to maintain criticality, radiative capture of neutrons in the fissile material which does not result in fission, and absorption of neutrons in structural materials and in fission fragments. For plutonium-239 the value of η is well above the actual threshold for breeding in a fast reactor, in which, furthermore, the threshold itself is reduced by the fission of uranium-238. Breeding by means of thermal neutrons, on the other hand, could produce only uranium-233, which is not found naturally in usable concentrations (the same is true of plutonium-239). Uranium-233 can also be produced from natural

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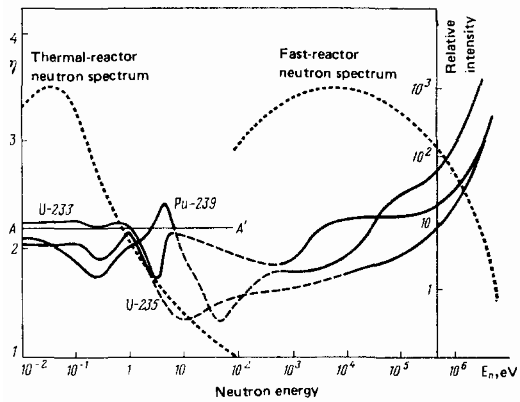

FIG 2. The parameter η as a function of the neutron energy for three fissile nuclei which are being used, or which might be used, in nuclear fuel cycles.[2] Line AA' is the actual breeding threshold in thermal reactors. The dotted curves are the neutron spectra in thermal and fast reactors (the corresponding ordinate scale is at the right).

thorium-232, whose abundance is comparable to that of uranium. A breeder cycle using uranium-233 (the uranium-thorium cycle),

$$^{232}\mathrm{Th} + \text{neutron} \xrightarrow{\gamma} {}^{233}\mathrm{Th} \xrightarrow{\beta^-} {}^{233}\mathrm{Pa} \xrightarrow{\beta^-} {}^{233}\mathrm{U},$$
$$^{233}\mathrm{U} + \text{neutron} \rightarrow \eta \text{ neutrons } + 170\ \mathrm{MeV} + \text{waste},$$

could be realized only in reactors having a very good neutron economy.

Present-day fast breeders have conversion ratios roughly in the range 1.15–1.35 and can support an annual growth of 2–6% of the nuclear power industry, but the excess plutonium which is bred in them is insufficient to fuel a growing inventory of thermal reactors. This would require fast breeders with a conversion ratio of up to 1.6–1.8 and, in addition, a reduction of the plutonium inventory in the reactors themselves and at plants for reprocessing spent fuel. So far, we have seen nothing approaching a plausible plan for a fast breeder which can breed at this rate. The work which is presently being carried out to improve fast breeders, their fuel, and the fuel reprocessing technology, with the goal of raising the conversion ratio and achieving short doubling times (≤7–9 yr), is still far from completion. If we had some other, reasonably economical method for converting nuclear raw materials (uranium-238 or thorium-232) into a nuclear fuel (plutonium-239 or uranium-233) then our efforts to improve both fast and thermal reactors could be concentrated on improving their economy, reliability, and safety rather than on striving for the limiting fuel-balance characteristics.

The solution to the basic problem of fission power—the problem of providing a fuel base—thus requires a search for, and comprehensive study of the best methods for making use of nuclear fuel. These methods would have to provide fuel for the foreseeable future at an acceptable cost. The most important task is thus to study methods for large-scale production of neutrons (for processing the huge stocks of uranium wastes and natural thorium) which would not involve uranium-235, which is being depleted.[1)] Breeding nuclear fuel improves the utilization of natural uranium—when the unavoidable losses are taken into account—by a factor of about 100 and thereby solves the fuel problem for at least a thousand years. This possibility is of fundamental importance in its own right, but in this paper we are interested not only in the fundamental aspects of the approach but also (and even more) in the directions which will be taken in the development of breeding schemes. This effort should solve the fuel problems for the next 50, or at most 100, years.

2. So far, two approaches have been suggested for this large-scale generation of "nonuranium" neutrons: the fusion of light nuclei and reactions involving a deep disintegration of heavy nuclei induced by accelerated light ions (spallation reactions). In the fusion approach, the idea is to obtain usable energy through the thermonuclear fusion of heavy hydrogen isotopes in magnetically or inertially confined plasmas. (See the preceding article[62] for a survey of the present state of the magnetic-confinement approach to controlled fusion.) The development of pure fusion reactors—which is the ultimate goal of the rapidly developing effort toward controlled fusion in the leading industrialized nations—has proved to be a technically more complicated matter than it seemed at first. Nevertheless, a recent analysis of the various concepts for fusion power reactors shows that, if the necessary effort is made, a test reactor can be constructed in 10–12 yr, at least for the most advanced concept, the tokamak. On the other hand, the same analysis shows that at our present technological level a fusion reactor would be much more expensive than a fission reactor and would furthermore require a long time for solving engineering problems and for developing the industrial and technological base to make the fusion reactor economical.

Interest has accordingly been attracted to hybrid or symbiotic fusion reactors, in addition to the pure fusion reactors. In a hybrid reactor the fusion and fission of nuclei would take place in a common unit, so that such a reactor would have several advantages not shared by installations in which fission and fusion occur separately. For example, a hybrid reactor could breed several metric tons of fissile material per year from spent uranium or thorium while generating useful electrical power of 1–2 GW. The hybrid schemes are based on the established experimental fact (see Fig. 3, taken from Ref. 4) that 14-MeV neutrons are multiplied in uranium through fission reactions and (n, xn) reactions. Breeding in uranium thus "recycles" the energy of the fusion neutron instead of utilizing it directly to heat a moderator. Since the energy of a fission reaction (≈200 MeV) is about an order of magnitude greater than that of a fusion reaction (17.3 MeV), wrapping a uranium blanket around a fusion reactor would increase the reactor power level by a large factor (about 7 for the studies which have been made). This advantage, combined with the possibility of breeding large amounts of

[1)]We will not discuss breeding by means of underground thermonuclear explosions, which are discussed in Ref. 3, for example.

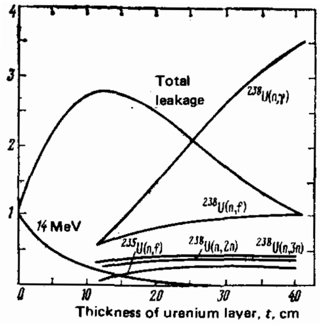

FIG. 3. Total number of reactions and neutron leakage as a function of the thickness of a blanket of natural metallic uranium in which a 14-MeV neutron is absorbed.[4]

nuclear fuel for fission reactors, significantly eases the technical problems in developing an economical fusion reactor: It would be possible to use fusion installations of relatively modest size and power; the radiation and heat load on the reactor chamber (the first wall) would be lowered; the reactor would not have to be operated at the extreme values of the "power gain" by the plasma; etc. In summary, the hybrid fusion reactor may be regarded as the earliest application of controlled fusion capable of contributing to the solution of the fuel problem of a fission-reactor power industry. At the same time, the hybrid fusion reactor would be a practical step toward the pure fusion reactors of the future. However, the development of hybrid reactors (more precisely, of their fusion neutron sources) also involves some serious difficulties, and at this point it is not possible to say with any certainty just when they will begin to furnish a significant fraction of the secondary nuclear fuel. At the moment, at least, it appears that a hybrid fusion reactor can probably be developed from the approach of an open confinement system[5] or, as mentioned above, a tokamak.[6]

Another approach in the fusion effort makes use of inertial plasma confinement, i.e., the isentropic compression of a spherical target (which contains a DT fuel and which has a more or less complicated multilayer structure) by intense laser beams or beams of relativistic electrons or ions. It is now becoming clear that in order to ignite such targets with laser beams we will apparently need very high power densities ($\gtrsim 10^{16}$ W/cm^2); at these power densities, which take us into a region of unknown physics, some interrelated nonlinear processes may operate to limit the absorption of the laser beams by the target. Unfortunately, it is at these high power densities that the pertinent analytic methods which are available become progressively less reliable and can furnish only qualitative estimates, hardly of practical use. Furthermore, for reactor purposes we would need an intense laser with an efficiency of at least 0.1–0.15 whose active medium (probably gaseous) emits in the wavelength interval 300–3000 nm. The threshold energy of such a laser is presently estimated[7] to be at least 300 kJ. The characteristics of the existing laser systems are much inferior to this, although lasers for ablation compression and ignition of DT targets will be constructed in the near future to demonstrate the physical feasibility of a fusion reaction with a net energy yield. If this assessment of the situation is correct, however, we cannot expect to see even a hybrid version of a laser-ignition fusion reactor until the next century.

The approach using megampere relativistic electron beams cannot provide the required power density at the target. It can be provided by ion beams, especially beams of heavy ions ($A > 150$) accelerated to energies of several tens of MeV per nucleon.[8,9] Because of the better energy transfer from the ion beam to the target, the energy of the ion bunch required to achieve a gain of about 100 is estimated to be comparatively modest (a few hundred kilojoules). Furthermore, we may say that the technology of high-current linear accelerators with a reasonably high efficiency ($\gtrsim 0.20$–0.25) already exists, especially when we take into account the invention of spatially homogeneous focusing[10] and the MEQALAC structure.[11] Again in this direction, however, scientific feasibility must be proved; the first experiments along this line are planned for the late 1980s.

Even cloudier is the outlook for the old concept of ballistic fusion, which has recently been revived.[12] The idea here is to bombard a DT target with a macrosopic particle (a magnetic dipole) with a mass of 0.1–1 g accelerated to a velocity ~10^8 cm/s in a magnetic accelerator with a length ranging from tens to thousands of meters. As the accelerated magnetic dipole (which would probably by a superconductor, and about 10% of its mass should be a DT mixture) is absorbed in the target a shock wave arises; the target is heated intensely during the strong compression, with the result that the DT fusion reaction is ignited. At the conceptual level the accelerating apparatus appears quite simple, but the technology has not yet been developed. So that at this point we could hardly expect a demonstration of physical feasibility of the concept in the near future.

We will not discuss here the various new approaches which have been proposed for achieving fusion reactions. Although the many new suggestions have some clear advantages over the earlier approaches, it is an expensive business to do the research and development on each possibility; plasma physics, despite its indisputable progress, is not yet capable of quickly identifying which of these suggestions would be the simplest to implement and economically acceptable. Furthermore, and very significantly, the development and widespread dissemination in the economy of any new technology for large-scale, long-term production of energy will require many years (decades), so that those approaches which (as we now believe) will begin to be adopted by industry in, say, the first few decades of the next century must be completely developed by the end of the present century. Accordingly, just how a particular direction will develop must become clear in the next few years—before the turn of the decade. It is thus difficult to imagine that, in the face of the formidable materials-science and engineering problems, it would be possible to do all the work to prove the technological feasibility of an inertial-confinement fusion reactor and also to put this technology in place in the economy in the 18 years remaining until the turn of the century (even if the physical feasibility of heavy-ion fusion is demonstrated in

1985–1990). It is also difficult to believe that hybrid reactors using inertial-confinement systems would begin to have any appreciable effect on the production of nuclear fuel before the end of the first quarter of the next century.

3. There is the possibility that alternative methods for producing neutrons and nuclear fuel (methods making use of intense ion accelerators) may be implemented earlier and at a lower cost. One possibility is the catalysis of DT fusion by negative muons,[13] and another is the electronuclear method of producing free neutrons in heavy media by means of accelerated light ions. In the muon method, a funneling (under the Coulomb barrier) occurs in the mesomolecular system $DT\mu^-$, in which the D and T nuclei are very close because of the screening effect of the negative charge of the muon. The recoil momentum of the He^4 nucleus which is produced in the reaction is sufficient to "shake off" the muon, which is then recaptured into a common mesomolecular orbit of the next pair of D and T atoms. The muon manages to trigger about 100 fusion events (according to calculations) over its lifetime ($2.2\cdot 10^{-6}$ s). It has been suggested that deuterons accelerated to 2 GeV might be used with a beryllium target to produce the negative muons; the thickness of the beryllium target would be chosen such that the deuteron beam would lose about 30% of its energy in the beryllium (the muon-production efficiency falls off rapidly with decreasing deuteron energy), and the residual beam of particles (a mixture of deuterons, protons, and neutrons) leaving the beryllium would enter an electronuclear target (more on this below) in which it would be completely absorbed. The negative pions produced in the beryllium by the deuterons would be trapped in a magnetic confinement system which would confine the decay muons and transport them to the reactor chamber. The reactor chamber would contain a gaseous mixture of deuterium and tritium at a pressure of several hundred atmospheres, at the density of liquid hydrogen. The 14-MeV neutrons produced during the DT fusion would be multiplied in a blanket around the reactor chamber containing depleted uranium and/or thorium. Pursuing this possibility, Gershteĭn *et al.*[13] have suggested a version of the hybrid reactor which uses "mesonnuclear" neutrons; this reactor would be combined with an electronuclear reactor (discussed below).

For such a reactor to be economically attractive, the source of mesonuclear neutrons would have to provide some 10^{19}–10^{20} neutrons/s; i.e., the current of accelerated deuterons would have to be several hundred megamperes (Gershteĭn *et al.*[13] assumed the efficiency at which thermalized negative muons are produced, i.e., the average number of muons per primary deuteron, to be about 0.3). The amounts of nuclear fuel produced by the mesonuclear and electronuclear targets would have a ratio of about 3:2, so that at a deuteron beam power of 200 MW such a system could, in continuous operation, generate about 1.2 metric tons of nuclear fuel per year.

At first glance the muon-catalysis method might seem to offer a radically simpler way to achieve DT fusion (there is no need to deal with a hot plasma!), and "proof of principle" would be simpler, since the reaction occurs (this point has been demonstrated experimentally[14]) as soon as the thermalized negative muon finds itself in a moderately compressed DT mixture. It appears, however, that an effort toward muon catalysis would have to overcome some very serious engineering problems in order to develop this complicated target–(magnetic confinement system)–(DT reactor)–blanket assembly. Even before that it would be necessary to examine the practical usefulness of muon catalysis: It would be necessary to obtain reliable and comprehensive experimental data on the number of DT fusion events per muon, on the average number of negative ions produced in a thick beryllium (or lithium) target by a deuteron with an energy of 1.5–2.5 GeV, on the attachment of muons to helium and to other possible impurities, etc. Even at this point, it is clear that "pure" muon catalysis, unassociated with nuclear fission, would have to be rejected because of the overly high effective threshold for the DT fusion reaction (a single 2-GeV deuteron causes about 33 fusion events each with an energy yield of 17.6 MeV; i.e., there is a more than threefold energy loss even if we ignore the efficiency of the apparatus). The combination of the muon-catalysis method with the electronuclear method improves the fuel-energy balance by a factor of 1.5–2, since the yield of fissile material per unit energy expenditure is estimated to be 60–80% higher for this combination than for each method separately (the yields of the fissile products for the electronuclear and muon-catalysis targets separately will be roughly the same if we assume that two plutonium nuclei are produced in the blanket per 14-MeV neutron, which is the usual assumption in the designs for the blankets of hybrid fusion reactors). These two technologies are thus synergistic.

Let us estimate the energy which would be expended on producing a fissile nucleus by the hybrid, muon-catalysis, and electronuclear methods; we will ignore the multiplication in the fission reactor for the moment, since it is the same for all the methods. A multibeam laser system which transfers an energy of 60 kJ to a pellet of a DT mixture will produce $6.4\cdot 10^{17}$ neutrons, according to the calculations of Ref. 15. This means that about 0.6 MeV will be expended per 14-MeV neutron, and if the laser efficiency is, say, 10% (for a gas laser) then the electrical energy expended will be about 6 MeV per neutron. Making the further assumption that the absorption of this neutron in the uranium blanket will produce two plutonium-239 nuclei, we find an expenditure of about 3 MeV per nucleus; we are ignoring heat evolution in the blanket. Increasing the laser efficiency and/or the power gain of the target could lower this value to a few MeV per nucleus, while a severalfold increase in the threshold energy of the laser (see Ref. 7, for example) would correspondingly raise the energy expended on the production of a fissile nucleus. Roughly the same energy expenditures are apparently typical of a heavy-ion system whose accelerator part has an efficiency of 15–20% and for which the power gain of the target reaches several hundred.

In the muon-catalysis approach the energy expended per fissile nucleus can be estimated in the following way. A 2-GeV deuteron produced about 0.33 of a μ^- meson, i.e., 33 DT fusion events, and thus about 80 plutonium-239 nuclei in the uranium blanket. This process consumes 30% of the power of the deuteron beam; the other 70% enters the electronuclear target (the depleted uranium), where another 70 or so plutonium nuclei are produced [for simplicity we are assuming that undissociated deuterons leave the beryllium target with an energy 5% lower than the initial energy and with an intensity of 0.75, while the expenditure of kinetic energy on the production of a fissile nucleus in the depleted uranium is some 15% lower than for a proton of the same energy (Fig. 4); as usual, we are assuming that the neutrons are used at an efficiency of 0.85]. As a result we find that it is necessary to expend 2000/150 $\simeq$ 13 MeV of kinetic energy to produce a single fissile nucleus; i.e., at an accelerator efficiency of 0.6 the expenditure of electrical energy will be $2000/150\cdot 0.6 \simeq 22$ MeV per nucleus. It can be seen from Fig. 5 that for a 1-GeV proton the expenditure of electrical energy on the production of a plutonium-239 nucleus will be $17/(0.6\cdot 0.85) \simeq 34$ MeV, and about 20% less for a 2-GeV deuteron. When we take into account the energy evolved in the blankets and the targets (an additional 15–20 MeV of electrical energy is generated per fissile nucleus) we find a net energy gain in the case of the hybrid devices, in the muon-catalysis case the energy gain is 3–4 MeV, and that in the electronuclear method is 6–7 MeV, per fissile nucleus (in the most favorable case, in which the beam of accelerated particles is absorbed in depleted uranium). These estimates are good enough for a qualitative picture of the situation, although we do not claim any great accuracy for them: The particular numbers may change significantly, depending on the efficiencies, the target enrichments, etc., adopted in assessing the energy balance. We recall that the burnup of a fissile nucleus in a thermal reactor yields $E_f\cdot \eta_T \sigma_f/\sigma_a$ MeV of electrical energy, where E_f is the energy of nuclear fission ($\approx$200 MeV). η_T is the thermal-cycle efficiency of the power plant, and σ_f/σ_a is the fraction of nuclei undergoing fission in the neutron spectrum of the thermal reactor ($\approx$0.9), i.e., the energy yield is about 50 MeV.

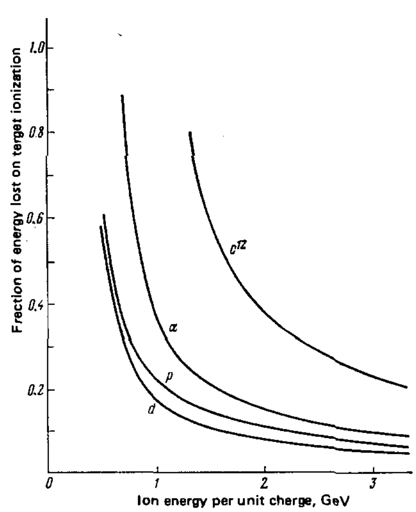

FIG. 4. The fraction (Δ) of the kinetic energy lost by an accelerated ion on ionization of the target before the first inelastic nuclear collision as a function of initial kinetic energy E_0. The target is made of lead, and

$$\Delta = -\int_0^{\lambda_{in}} dx \frac{dE}{dx}/E_0.$$

This fraction is calculated from the data of Refs. 17–19.

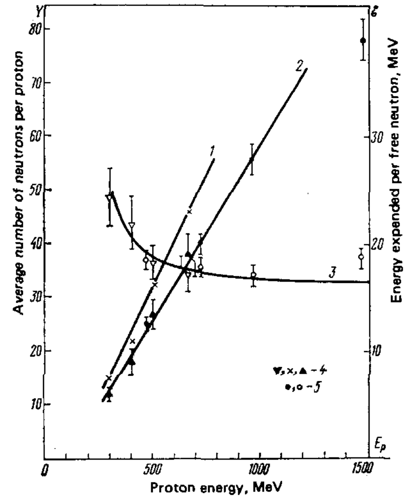

FIG 5. Plutonium-239 yield in a quasi-infinite target of natural metallic uranium (1) and of depleted metallic uranium (2). Curve 3—Energy expended on producing a single plutonium-239 nucleus (the ordinate scale at the right) in depleted uranium. The experimental values are taken from Refs. 21 (4) and 22 (5). The data from Ref. 22, obtained for a uranium cylinder 10 cm in diameter and 60 cm long with an enrichment of 0.22%, have been converted to correspond to a target of infinite dimensions by multiplying by 1.375 per unit charge of the ion ($\blacktriangledown \equiv \triangledown$).

4. The electronuclear method is another, completely independent, possibility for using electrical energy for large-scale production of neutrons,[16a] in particular, for producing nuclear fuel (see Refs. 16b and 16c for a summary of the method and its history). An electronuclear reactor would combine an accelerator which produces an intense beam of protons, deuterons, or—less probably—α-particles with a target reactor containing heavy materials: lead, bismuth, thorium, uranium, or combinations thereof. The target should probably consist of two parts: a primary target in which the particles accelerated to $E_0 \approx 1.5-2$ GeV are converted into neutrons, and a subcritical blanket, which multiplies these neutrons and produces nuclear fuel as well as a certain amount of energy. Actinide reactor waste products (uranium-236 and plutonium-240) might be added to the primary target; the impurity of low-A elements, on the other hand, must be minimized to avoid softening the neutron spectrum. As the beam of accelerated particles is absorbed in the heavy-material target, spallation reactions are excited and followed by a cascade-evaporation multiplication of neutrons, which includes (n, xn) reactions. In fissile materials there will also be fission by nuclear nucleons with energies ranging from a few MeV to several hundred or thousands of MeV. In the case of a light primary target (lithium or beryllium)

and a deuteron beam the neutron multiplication mechanism entails an initial stripping of the deuteron, followed by absorption of the resulting nucleons in the heavy medium around the primary target, accompanied by the excitation of a cascade of nuclear reactions.

The neutron yield (the average number of neutrons emitted by the target per bombarding particle) is proportional to the kinetic energy of the bombarding particle at the time of the first inelastic collision with a nucleus of the primary target. Consequently, the maximum neutron yield per unit energy expenditure is reached when the minimum fraction of the initial energy of the particle is expended on ionizing the medium before the first inelastic collision. Working from data on the ionizational energy loss rate dE/dx (see Ref. 17, for example) and the experimental total inelastic cross sections for the interactions of accelerated light nuclei with matter,[18,19] and calculating the kinetic energy of the ion (per unit charge of the ion) in the first inelastic collision, we find that the maximum neutron yield per unit energy expenditure (or, looking at it the other way around, the maximum energy expended per liberated neutron, $\mathscr{E}$) will probably be reached at the minimum of the ionizational energy loss rate. With a further increase in E_0, a logarithmic increase of dE/dx begins, and—a far more important point—the fraction of the energy which is carried off by the electron-photon component of the cascade (through neutral pions) and which thus essentially does not participate in the production of free neutrons increases monotonically. Wilson[20] cites calculations by Van Hinneken which predict that the energy expended on the liberation of a neutron from a massive target increases for this reason by a factor of nearly 1.5 as the energy of the primary proton is increased from 1 to 100 GeV. For a proton and a deuteron, for example, the fraction of the energy lost on ionization (Fig. 4) is a few percent at 2–3 GeV, while for ions with atomic numbers $Z \geqslant 2$ this fraction increases $\propto Z^2$, so that the use of accelerated ions heavier than α particles is less advantageous from the standpoint of the energy balance. In a real electronuclear installation, of course, the energy to which the ion should be accelerated will be determined by optimization studies.

At accelerated-ion energies of hundreds or thousands of MeV, at which the ionization range becomes far longer than the nuclear range (the total inelastic cross section remains roughly constant above 200–300 MeV), essentially all the ions undergo inelastic interactions with the target nuclei over a distance equal to the ionization range. The density of these interactions along the axis of the ion beam is described approximately by $\exp(-x\Sigma_a)$, where Σ_a is the macroscopic inelastic cross section of the medium (the geometric cross section). Experiments show that at these ion energies the neutron yield over the first two nuclear ranges is about 80% of the total yield; the fifth nuclear range contributes no more than 1.5% of the total number of neutrons. The axisymmetric neutron source which arises during the absorption of the ion beam in the target and which decays exponentially along the axis reaches a maximum in the first nuclear range. The heat evolution in the target has a similar distribution, since the charged products of reactions in which neutrons are liberated have a short range. The difficulties which stem from this nonuniformity of the neutron field along the axis of the particle beam and from the very high heat-evolution densities (at high beam densities) can be eased significantly by modifying the density of the primary target along the beam axis to flatten out the distributions of the sources of fast neutrons and heat.

Figures 5 and 6 show experimental results[21,22] for a quasi-infinite target of depleted metallic uranium bombarded by accelerated protons. It follows from these figures that the absorption of a 1-GeV proton in several metric tons of depleted uranium results in the liberation of about 55 neutrons and the evolution of thermal energy of 4.5–5 GeV. The neutrons are ultimately captured by uranium-238 and lead to the production of plutonium-239. Results of this type are basic for the electronuclear method, since they make it possible to estimate the neutron productivity of the method and its energy balance, but it should be borne in mind that these results were obtained for pure uranium without a heat-exchange medium and without structural materials.

There is a corresponding picture in the case of thorium-232: The capture of neutrons in the thorium results in the formation of uranium-233 (the best fuel for thermal reactors), but since the ability of thorium to undergo fission is lower than that of uranium-238 (the fission cross section is smaller by a factor of about five for neutron energies from 1 to 15 MeV) the yield of uranium-233 is estimated to be 30–40% lower, some 35–40 nuclei per proton at a proton energy of 1 GeV. For the same reason the energy evolution in a thorium target should also be lower: by a factor of about three according to estimates. In lead, fission can be ignored, and the energy balance is determined by energy conserva-

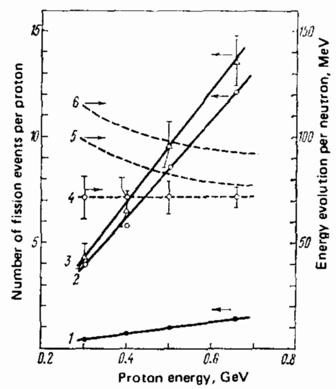

FIG. 6. The number of fission events of uranium nuclei in a quasi-infinite target of depleted metallic uranium (ordinate scale at the left). 1—Number of uranium-235 fission events; 2—the same, for uranium-238; 3—total number of uranium-235 and -238 fission events (all values are per proton). Ordinate scale at the right: 4—Energy evolution per plutonium-239 nucleus produced (the ionization energy loss of the primary proton is ignored, as is the fission of uranium by high-energy nucleons in the cascade-breeding stage); 5—the same as 4, but with allowance for the energy lost by the primary proton on ionizing the target; 6—the same as 5, but with allowance for the fission of uranium by the cascade nucleons (the number of such fission events was calculated in Refs. 26b and 37d).

tion. The average number of neutrons per proton with an energy between 500 and 1000 MeV ranges from 8 to 22 for lead cylinders 10–20 cm in diameter and 30–60 cm long.[22-25] Experimental data (Figs. 5–7) show that the dependence of the neutron yield on the energy of the primary proton, E_0, is linear from at least 300 MeV up to ≈1.5 GeV.

The proton kinetic energy expended on the production of a free neutron in a quasi-infinite block of depleted uranium reaches ≈17 MeV at a proton energy of 1 GeV (Fig. 5), while in natural uranium it is 14 MeV, and in spent fuel (with a 2% enrichment) it is estimated to be ≈11 MeV. It can be seen from the experimental results of Ref. 21 that the replacement of the central part of a uranium target, where the proton beam is stopped, by a lead block reduces the neutron yield and the number of fission events by a factor of about two; i.e., the values given above for the energy expended on the production of a neutron are doubled. This result is interesting for evaluating the version of an actual electronuclear target consisting of a breeding blanket around a primary target of liquid lead or a eutectic lead-bismuth mixture. If we now take into account the heat evolved in the target [about 80 MeV per fissile nucleus produced; this value varies significantly with the enrichment of the uranium target (Fig. 6)], then at an accelerator efficiency of 0.6 and at a thermal-cycle efficiency of 0.4 (the heat-exchange medium is a liquid metal) we find that the actual power gain for the electronuclear method ranges from ≈1.5 to 4. When we take into account the breeding in the thermal reactor, the gain ranges from roughly 3 to 8, i.e., some 3–4 times smaller than in the case of the hybrid fusion reactor with $Q=4$.

Several leading laboratories (Argonne, Brookhaven, Dubna, Oak Ridge, Los Alamos, and Chalk River) have developed computer programs for calculating the cascade multiplication of neutrons in massive extended media. These programs essentially combine an analysis of the nucleon-meson transport in the macroscopic medium (which may be heterogeneous) with a subsequent analysis of the neutron transport. In the nucleon-meson-transport part of the program, the number of neutrons and their spatial and energy distributions are calculated by a Monte Carlo method from the cascade-evaporation model. These neutrons are then followed down to a certain cutoff energy (10–15 MeV), below which the nature of the reactions induced by the neutrons (fission, elastic and inelastic scattering, capture, leakage, etc.) is studied by the neutron-transport program through the use of a many-group system of neutron cross sections. The agreement with experiment is generally satisfactory in terms of the calculated integral and average values (the discrepancy ranges from 5% to 20%), although in some cases programs of the same type used to calculate the multiplication under the same experimental conditions[21] yield greatly divergent results even in terms of the neutron yield.[26a,27] The agreement is generally worse in terms of the differential characteristics (see Ref. 27, for example). For a further refinement of these computation methods it will be necessary to expand the experimental study of both the integral effects (in thick targets) and the interactions at the elementary level, described by new theoretical approaches involving, for example, solution of neutron-transport kinetic equations.[29,30]

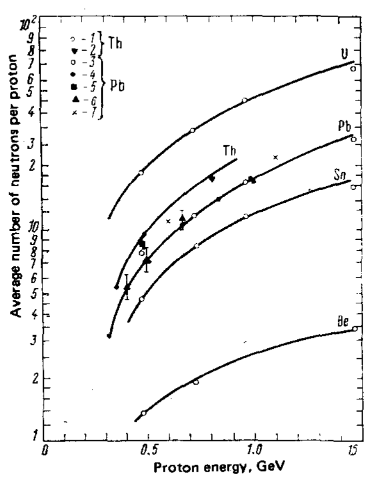

FIG. 7. The neutron yields from cylindrical targets 10 cm in diameter and 60 cm long (the length of the thorium target is 30 cm, and the length of the beryllium target is 90 cm). The results shown for Be, Sn, and U are the experimental results of Ref. 22. For Th (1), the results are data from measurements in the TRIUMF cyclotron[23a]; 2—results of Ref. 23d for Pb; 3—data of Ref. 22; 4—data of Ref. 24; 5—data of Ref. 23a; 6—data of Ref. 25; 7—data of Ref. 23c. The primary particles are protons.

It can be seen from Fig. 5 that the yield of plutonium-239 per 660-MeV proton is 38 ± 4 nuclei from a target of metallic depleted uranium. This figure does not include the neutron leakage from the target (10–12% for a target with a mass of 3.5 metric tons), but this leakage is not very important for the estimates below since the neutron yield in a real target is reduced 10–20% by the effects of the heat-exchange medium and the structural materials. Accordingly, at a proton current of, say, 100 mA the daily production of plutonium-239 may be 0.8 kg (corresponding to roughly three moles of neutrons per day), or the annual production (300 days) would be 240 kg. A thermal power of about 300 MW is evolved in the target; i.e., the 66-MW power of the proton beam is "multiplied" by a factor of about 4.5 in the depleted uranium (Fig. 6). At an efficiency of 0.4 for the thermal station of an electronuclear reactor, this multiplication could result in an electrical power of about 100 MW, sufficient to keep the accelerator in operation if its efficiency was at least 0.6. At present, such efficiencies look attainable in the next 10–15 yr (Refs. 31 and 32). We might note that about 1.4 g of depleted uranium is expended on the production of 1 g of plutonium-239.

These estimates of course refer to the most favorable case in which the beam of accelerated particles goes directly into the uranium, but this approach would hardly be possible at beam power levels of the order of hundreds of megawatts because of uranium's inadequate thermal conductivity and susceptibility to radiation damage.[33] Metallic thorium has slightly better properties, but at these beam power levels the density of the energy evolution in heavy solid-state targets is so high that adequate heat removal is probably impossible. For this reason, the energy of the accelerated-particle beam should probably be converted in a liquid-metal target (molten uranium salts, a lead-bismuth eutectic mixture, or, finally, lithium, if a deuteron beam is used). The target would play two roles: convert the primary particles into neutrons and transfer heat. This approach, however, presents some very complicated mechanical problems in the design of the target. Furthermore, the need to inject the beam of accelerated particles through a large surface would (if the nuclear range is short and also constant) result in poor transport of neutrons from the primary target to the blanket, unless the density of target material along the ion beam is reduced in some fashion. The method for injecting the beam into the reactor target thus strongly affects the neutron yield, and since the uranium and thorium will most probably be used in the form of oxides or carbides in the blanket of an electronuclear reactor the yield of fissile materials will be significantly lower than in the case of pure uranium (by about 20–30%; see, for example, Refs. 34 and 35). This decrease in the neutron yield and in the energy evolution, however, can be offset by the extensive multiplication in the blanket of an electronuclear reactor. Estimates based on the results of Refs. 21 and 30 show that the use of uranium with an enrichment of 2% in the blanket and the use of a lead-bismuth primary target do not change these estimates. A similar result is found for a target consisting of molten uranium salts into which a proton beam is injected directly.[36]

5. What does the energy balance of an electronuclear reactor look like? When we are interested in the liberation of a neutron for power purposes, i.e., to obtain a fissile nucleus from a fertile one which absorbs this neutron, it is natural to require that the energy expended in the process (E_{in}) be lower than the usuable energy (E_{out}) which can be extracted by burning the resulting nucleus in a reactor:

$$\frac{\mathscr{E}}{\eta_y} = E_{in} < E_{out}; \tag{1}$$

where $\mathscr{E}$ is the ion kinetic energy expended on liberating the neutron, given by $\mathscr{E} + E_0/Y_0$, where E_0 is the initial energy of the primary particle, Y_0 is the neutron yield from the bombarded target (Fig. 5), and η_y is the efficiency of the accelerator, so that $\mathscr{E}/\eta_y$ is the electrical energy expended on liberating the neutron. It can be reduced by making use of the thermal energy W_M evolved in the target. If the energy multiplication in the target is characterized by a factor Q, i.e., if $W_M = QE_0$, then we have (per free neutron)

$$\frac{W_M}{Y_0} = \frac{QE_0}{Y_0} = Q\mathscr{E}, \tag{2}$$

and inequality (1) becomes

$$\mathscr{E}\left(\frac{1}{\eta_y} - Q\eta_T\right) < E_{out}, \tag{3}$$

where η_T is the thermal-cycle efficiency of the target.

The quantity on the right side of (1) is

$$E_f \left\langle \frac{\sigma_f}{\sigma_a} \right\rangle \eta_T, \tag{4}$$

where E_f is the energy released per nuclear-fission event (≈200 MeV for both the fissile and fertile nuclei), and $\langle \sigma_f/\sigma_a \rangle$ is the ability of the nucleus to undergo fission, averaged over the neutron spectrum of the reactor. This factor reflects the circumstance that the absorption of a neutron by a fissile nucleus does not necessarily lead to a disintegration of the latter. The value of this factor depends on the neutron spectrum of the particular type of reactor and on which fissile material is burned in this reactor. For a fast reactor with plutonium fuel the quantity $\langle \sigma_f/\sigma_a \rangle$ is 0.83, while for uranium-233 in a thermal reactor it reaches 0.91.

Strictly speaking, expression (4) must still be multiplied by $1/(1 - \mathrm{CR})$, where CR is the fuel conversion ratio for the given reactor, to take into account the partial breeding of fissile material. Here it is implied that the bombarded fuel will be processed or that there will be a repeated "charging" (regeneration). At first, however, we will analyze inequality (1) without taking this factor into account, i.e., for an electronuclear reactor by itself. It consists of an accelerator, a primary target in which the accelerated particles are converted into fast neutrons, and a secondary target, which is a subcritical blanket which multiplies neutrons from the primary target. The breeding of the fuel in a power reactor during the burning of the fissile material produced in the blanket of the electronuclear reactor, on the other hand, can be taken into account by studying the symbiotic system consisting of an electronuclear reactor combined with one or several thermal reactors.

Inequality (3) thus becomes

$$\mathscr{E}\left(\frac{1}{\eta_y} - Q\eta_T\right) < E_f \left\langle \frac{\sigma_f}{\sigma_a} \right\rangle \eta_T; \tag{5}$$

for simplicity we are assuming that the thermal efficiencies of the reactor, the primary target, and the blanket are equal. This inequality becomes stronger if the neutrons from the primary target are multiplied in a blanket with a multiplication factor k_{eff}, so that the yield of fissile product and the energy evolution are increased. We know that the multiplication of the neutrons of a source in a medium with a factor k_{eff} is determined by

$$K = \frac{1}{1 - k_{eff}}; \tag{6}$$

this expression holds if the spatial and energy distributions of the neutrons of the target and the blanket are the same (for the primary multiplication mechanism in the blanket). At large values of k_{eff} this expression gives an accurate description of the multiplication of the neutrons from a source at the center of the blanket.

The neutron multiplication in the blanket involves a variety of reactions (including some which do not produce neutrons) with relative weights δ_i. We denote by δ_1 the relative number of neutrons which undergo radiative capture, (n, γ), in the original materials; by δ_2 the relative number which cause fission, (n, f), of the nuclei of these materials (fission by fast neutrons); by δ_3 the relative number which are absorbed in the fissile material (radiative capture and fission); by δ_4 the relative number which cause fission, (n, f), of the fissile material; and by δ_5 the relative number which are wasted (absorbed in the heat-exchange medium, the moderator, and structural materials or leaked from the blanket). The relative number of neutrons which escape from the primary target and which do not reach the blanket can be made arbitrarily small by choosing the target and blanket geometry appropriately. Accordingly, from the normalization condition

$$\delta_1 + \delta_2 + \delta_3 + \delta_5 = 1 \tag{7}$$

and the experimental fact (at least for thermal installations)

$$\delta_1 + \delta_2 + \delta_4 \approx 0.95 \tag{8}$$

we conclude that the relative number of wasted neutrons is $\delta_5 \lesssim 0.05$. In the estimates below, as above, we assume $\delta_5 \approx 0.15$.

Using these definitions and recalling that the multiplication factor of the medium is $k = \bar{\nu}\langle\sigma_f/\sigma_a\rangle$, i.e., $k = \eta\,(\delta_2 + \delta_4)$, we can write an expression for the effective multiplication factor of the blanket:

$$k_{\text{eff}} = \frac{\bar{\nu}\,(\delta_2+\delta_4)}{\delta_1+\delta_2+\delta_3+\delta_5} = \bar{\nu}\,(\delta_2+\delta_4); \tag{9}$$

here we are assuming that the average number of neutrons per fission event, $\bar{\nu}$, is identical for the fertile and fissile nuclei.

The blanket thus increases the total yield of fissile nuclei (per primary accelerated particle) to

$$Y = Y_0 K\,(\delta_1 - \delta_3). \tag{10}$$

Substituting this expression into the denominator on the left side of inequality (5), we find

$$\frac{\mathscr{E}\,(\eta_y^{-1} - Q\eta_T)}{K\,(\delta_1-\delta_3)} < \eta_T \left\langle \frac{\sigma_f}{\sigma_a} \right\rangle E_f. \tag{11}$$

Taking into account the energy released in the blanket,

$$W_{\text{bl}} = \eta_T E_f K\,(\delta_2+\delta_4) = \eta_T E_f \frac{k_{\text{eff}}}{1-k_{\text{eff}}}\,\frac{1}{\bar{\nu}}, \tag{12}$$

we can make inequality (5) even stronger, and the final expression for the energy expended on the production of a single fissile nucleus becomes

$$\left[\frac{\mathscr{E}}{\eta_y} - \eta_T \mathscr{E} Q - \eta_T E_f K\,(\delta_2+\delta_4)\right][K\,(\delta_1-\delta_3)]^{-1} \quad \text{(MeV/nucleus)}, \tag{13}$$

where

$$\eta_T Q \mathscr{E} = \eta_T \frac{W_M}{Y_0}$$

is Eq. (2) multiplied by η_T.

We rewrite our original inequality (1), substituting into it expressions (3), (6), (10), (12), and (13):

$$\frac{1-k_{\text{eff}}}{\delta_1-\delta_3}\left[\frac{\mathscr{E}}{\eta_y} - \eta_T\left(\mathscr{E}Q + \frac{E_f k_{\text{eff}}}{\bar{\nu}\,(1-k_{\text{eff}})}\right)\right] < \eta_T \left\langle \frac{\sigma_f}{\sigma_a} \right\rangle E_f. \tag{14}$$

Equating the expression in square brackets in inequality (14) to zero, we find the energy breakeven condition for an electronuclear installation (i.e., $E_{\text{in}} = 0$):

$$\mathscr{E}\,\frac{\bar{\nu}}{E_f}\left(\frac{1}{\eta_y\eta_T} - Q\right) = \frac{k^0_{\text{eff}}}{1-k^0_{\text{eff}}}, \tag{15}$$

where k^0_{eff} is the effective multiplication factor of the blanket for an accelerator-target complex which is autonomous from the energy standpoint; the other quantities were defined above. The minimum value of $\mathscr{E}$ is close to 16–17 MeV/neutron for a target of depleted uranium bombarded by protons (Fig. 5); for lead cylinders 10–20 cm in diameter, $\mathscr{E}$ is 55–45 MeV/neutron (Refs. 21–25). The energy multiplication Q can reach 4.4–5 for the same targets (this is several metric tons of metallic depleted uranium) about twice this figure for a target of uranium with a 2% enrichment (the spent fuel of light-water power reactors), 1.2–1.7 for a cylinder of depleted uranium (10–20 cm in diameter and 30–60 cm long), and 0.6–0.7 for a lead cylinder of the same size. All these estimates follow from experimental data[21-25] and the calculations of Refs. 26, 30, and 37.

Substituting numerical values into (15) ($\mathscr{E} = 47$ MeV, $Q = 0.65$ for $E_p \approx 1.5$–2 GeV, and $\eta_T = 0.35$; we are assuming that the utilization factor for the neutrons produced is 0.85), we can calculate the values of k^0_{eff} corresponding to an energy-autonomous electronuclear reactor for various values of η_y (Fig. 8). If, for a fixed value of η_y, we have $k_{\text{eff}} > k^0_{\text{eff}}$, the electronuclear reactor will transfer its excess power to the power grid.

Returning to inequality (14), we rewrite it in a form more convenient for analysis, making use of (9):

$$\mathscr{E}\,\frac{\bar{\nu}}{E_f}\,\frac{1-\eta_y\eta_T Q}{\eta_y\eta_T} < \frac{k_{\text{eff}}}{1-k_{\text{eff}}}\left(\text{CR}'\left\langle \frac{\sigma_f}{\sigma_a} \right\rangle + 1\right), \tag{16}$$

since for the blanket we have

$$\text{CR}' = \frac{\bar{\nu}\,(\delta_1-\delta_3)}{k_{\text{eff}}} = \frac{\delta_1-\delta_3}{\delta_2+\delta_4}. \tag{17}$$

It can be seen from the data of Ref. 21 that the value of CR′ is 2.5–2.7. Using inequality (16), into which we have substituted the numerical values given above, we can single out the explicit relationship between η_y and k_{eff} or CR′ for blankets of various types (thermal or fast). It can be seen from Fig. 8 that the substitution of realistic values into condition (15) leads to values $k^0_{\text{eff}} = 0.65$–0.85; for safety, k_{eff} could be kept at a level ≤ 0.9.

Multiplying blankets of this type have been studied quite comprehensively in connection with the develop-

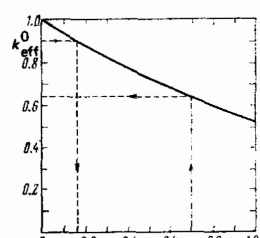

FIG. 8. Relationship between the effective multiplication factor of the blanket of an electronuclear reactor and the accelerator efficiency corresponding to an exact energy balance (the electronuclear installation does not draw energy from outside or furnish energy to the external grid). The primary target is a lead cylinder 20 cm in diameter, and the primary particles are protons. The dashed lines show the possible limiting values of k^0_{eff} and η_y. For example, if η_y reaches 0.7, the effective multiplication factor of the blanket would be about 0.63. If, on the other hand, k^0_{eff} does not exceed 0.9 the electronuclear reactor would reach energy breakeven at an accelerator efficiency ≈ 0.2.

ment of hybrid fusion reactors (see Refs. 6 and 38, for example). The blanket of an electronuclear reactor can be optimized to breed nuclear fuel with a subsequent radiochemical reprocessing and a refabrication of the bombarded material, to breed reactor fuel without reprocessing (the regeneration of spent fuel elements from power reactors), or to generate power. In the first case, the electronuclear reactor becomes a source of plutonium-239 for power reactors (water-cooled, water-moderated reactors) and even for fast breeders during their stage of rapid development. It also becomes a source of uranium-233 for thermal reactors. In the second case, it becomes a regenerator of the spent fuel of water-cooled, water-moderated power reactors and of more advanced thermal reactors. In the third case, it is an externally driven reactor. This flexibility in the design of the target of an electronuclear reactor and in its optimization stems from the mandatory subcriticality of the blanket.

The neutron yield in the target can be increased at a given beam power if protons are replaced by deuterons (Fig. 4): At an energy of 300 MeV, the ratio of the yields for a lead target bombarded by deuterons and protons is 1.21, while at 650 MeV it is 1.11 (according to the most accurate and most reliable of the experimental data available[24]). A linear extrapolation of the curves of the neutron yields measured in Ref. 24 as a function of the energy of the bombarding particle over the range 1.5–2 GeV yields a ratio of 1.07 (this is a lower limit); the calculations of Ref. 37 lead to roughly the same value at an energy of 2 GeV and to 1.2 at 1 GeV. The use of tritium for acceleration and conversion into neutrons could increase the neutron yield by about 40% in comparison with the case of protons (according to the estimates of Ref. 34), but this approach presents many new difficulties. The switch to other accelerated particles could, of course, raise the yield of the fissile product per unit expended energy and could ease several technical problems (for example, the combination of a deuteron beam with a lithium-beryllium primary target would seem to ease the target-cooling problem, and the large neutron yield per unit expended energy would make it possible to relax the requirement of a high accelerator efficiency). However, in the absence of systematic and accurate experimental data on the behavior of the neutron yield as a function of the energy of the various primary particles, it is not possible at this point to state with confidence just what the actual benefit will be and whether the benefit would offset the additional difficulties which the switch would introduce (for example, the difficulties involved in accelerating deuterons). In particular, in the absence of such information it would be impossible to optimize the energy and current of the beam of accelerated particles. In our estimates we are working from the basis that the energy cost of producing a free neutron, $\mathscr{E}$, in a heavy material bombarded by an accelerated ion tends toward a minimum, which probably lies near the minimum of the specific ionization energy loss of the ion. Clearly, if the minimum of the function $\mathscr{E}(E)$ is a broad one, it would be possible to achieve the same neutron yield per unit expended energy by using various combinations of the current, energy, and particle species. In particular, the idea underlying the electronuclear method is to work at the minimum of $\mathscr{E}$, because the power gain is then at a maximum.

As mentioned above, in the case of protons the minimum value of $\mathscr{E}$ for depleted uranium is 16–17 MeV, and that for lead is 40–50 MeV. About 80–90 MeV of thermal energy is released in a uranium target, or 25–30 MeV in lead. From these values we can get an idea of the energy balance and thus the efficiency of the accelerator and the target reactor. The requirement that the external energy drawn by the electronuclear reactor (because of the colossal beam power) be minimized (or eliminated completely) and that the accelerator be powered by the heat evolved in the target leads to a resultant efficiency $\gtrsim 0.21$ according to the data of Ref. 21 and of Fig. 6 (this efficiency is the product of the efficiency of the accelerator and that of the thermal cycle of the target). Under the assumption that the target heat is converted into electrical energy at an efficiency of 0.35–04, we find values 0.55–0.6 for the efficiency of the accelerator (the efficiency at which the grid power is converted into rf power, multiplied by the efficiency at which the energy of the rf field is converted into the kinetic energy of the proton beam). This is only a lower limit on the efficiency, however, since it refers to a pure uranium target; with targets of lead, thorium, or oxides or carbides of uranium and thorium, the energy breakeven condition will not be satisfied, and the energy deficit will have to be made up from the power generated by the reactors for which the fuel is being bred by the electronuclear reactor. Alternatively, as mentioned above, the blanket of the electronuclear reactor must be designed to have a larger multiplication factor.

6. What productivity would an electronuclear reactor have to have in order to be economical? Because of the high cost of the accelerator-target complex, fuel would have to be bred at a rate high enough to provide fuel to an inventory of thermal reactors whose net electric power would be much greater than the accelerator power and whose cost would be much greater than the cost of the electronuclear breeder. [We recall that the number of reactors supplied is proportional to the quantity $1/(1-\mathrm{CR})$ for a given amount of fuel.] The estimate given above for the productivity of a uranium target excited by a 100-mA beam of 660-MeV protons ($\approx$240 kg/yr) is useful only for getting an idea of the scale of the installation; neither the proton energy nor the current can be assumed optimized. The proton energy for a commercial electronuclear reactor would apparently be 1.5–2.0 GeV, where the power lost from the primary beam due to ionization is a few percent, and the current would have to be of the order of hundreds of milliamperes (the number of free neutrons in the system is $\sim 10^{20}\ \mathrm{s}^{-1}$). Space-charge effects would probably limit the current to 350–400 mA. A beam with these characteristics in a target of depleted uranium ($k_\infty = 0.38$, with a proton beam power of 300–400 MW) could provide a neutron productivity of about 4.5 moles per day per 100 MW of beam power (roughly speaking, a kilogram of fissile material). In other words, the productivity would be only 14–18 moles of neutrons (3–

4 kg of plutonium-239) at the total beam power. The thermal power of the target would be of the order of 1.5 GW. As mentioned above, the characteristics would be roughly the same for a target of lead (the primary target) combined with uranium with a 2% enrichment (the blanket). We might note that the overall energy balance of an electronuclear reactor could be improved by putting enriched material in the blanket (for example, a preliminary enrichment could be carried out in a conventional separation plant to a level about half the concentration required for charging a water-cooled, water-moderated power reactor; alternatively, as mentioned earlier, the residual enrichment of the spent fuel could be used); since, however, the materials bombarded in an electronuclear target become enriched in fissile material in one way or another, it by no means follows that we should reject the idea of a preliminary enrichment of the fertile material, especially since the economics of electronuclear breeding would benefit from this enrichment.[34]

An electronuclear breeder using an accelerator of this power level and with an efficiency $\gtrsim 0.6$ could produce 1–1.2 metric tons of plutonium-239 per year from depleted uranium or could regenerate, without a subsequent radiochemical processing and refabrication, 100 metric tons of spent fuel from commercial thermal reactors. The production of uranium-233 would be lower by a factor of about 1.3–1.5, with a much poorer energy balance for the target reactor, so that it appears at this point that the thorium would have to be used in combination with depleted uranium or with spent uranium at an enrichment nearly an order of magnitude higher than that of the depleted uranium.

At a productivity at this level, an electronuclear reactor would be able to fuel three water-cooled, water-moderated power reactors of the latest generation (CR $=0.6, \eta_T = 0.3$) with a uranium-plutonium fuel, or it would be able to fuel four such reactors with a uranium-thorium fuel (CR = 0.73). The power of each reactor would be 1 GW (electrical). In comparison with the CBR fast breeder being planned in the US, with a power of 1 GW (electrical) and a doubling period of no less than 15 yr, an electronuclear reactor with a proton-beam power of 300 MW could provide about 40 times more plutonium-239 (or uranium-233) per year, although, of course, it would not provide electrical power for the external grid. Thought of as a source of nuclear fuel, an electronuclear reactor is thus essentially analogous to a uranium separation plant. Each, drawing on an external energy source (or drawing little external energy, in the case of an electronuclear apparatus), produces fuel for power reactors and does not produce electrical energy for the external grid (or does produce some energy, but little, in the case of the electronuclear apparatus). However, the electronuclear reactor reprocesses depleted uranium (about 1.4 g per gram of plutonium), while a separation plant continually requires new natural uranium (about 200 g per gram of uranium-235 if 0.2% of fissile material is left in the tailings).

The interest in the production of uranium-233 by the electronuclear method does not derive entirely from heavy-water reactors which generate electric power; another consideration is the use of high-temperature thorium reactors to produce high-grade heat.[39] If research confirms that it is possible to accumulate uranium-233 in fresh thorium rods (to a concentration of about 3–4% in the rather hard neutron spectrum of the electronuclear target), this approach would substantially simplify the problem of utilizing thorium resources, since in the first charging, at least, it would be possible to skip the radiochemical processing, which is greatly complicated by the high activity of the decay products of the secondary uranium-232 (Ref. 40). Uranium-233 is the best fuel for thermal reactors, and an electronuclear reactor generating about 2.5 kg of uranium-233 per day could supply several heavy-water reactors with a CR = 0.9 (requiring about 0.1 g of uranium-233 per day per thermal megawatt) with a total power of more than 8 GW (electrical).

The high cost of reprocessing bombarded fuel and the technical and political problems associated with this technology will not promote its implementation, for there is the hope that the process of reprocessing and refabricating fuel can eventually be made less expensive and that, in addition, these other problems can be resolved (for example, the problem of preventing the proliferation of nuclear weapons). Under such conditions, if reprocessing of bombarded fuel is not widely adopted, an electronuclear reactor could play a key role in the nuclear power industry, because it might be used to enrich primary fuel and to reprocess spent fuel. This approach would make it possible to provide nuclear fuel to the existing and developing inventory of thermal reactors, to cut the demand for natural uranium several-fold, to eliminate the need for new uranium-enrichment facilities (by which we mean enrichment by the conventional methods), and to continue to develop the most acceptable methods for reprocessing bombarded fuel. On the other hand, it should be kept in mind that radiochemical reprocessing is absolutely necessary for complete use of uranium-thorium resources, in particular, in the fast-reactor cycle.

7. The idea of using an accelerator method to produce free neutrons on a large scale has recently been revived,[41,42] in no small part because of the progress in accelerator theory and practice which is transforming the accelerator from an apparatus unique to the laboratory into a reliable industrial machine. Experience shows that the utilization factor of accelerators exceeds 90%, which is higher than that of commercial power reactors. The interest in high-current ion accelerators as neutron sources arises from several problems: electronuclear breeding, the production of high flux densities of thermal neutrons, and the "afterburning" of certain long-lived radioisotopes (krypton-85, strontium-90, and especially, actinides) which are produced in large numbers and accumulate during the operation of nuclear power plants. (We will not discuss here the use of spallation reactions to produce intense pulsed neutron sources; these sources are presently being constructed in several laboratories on the basis of proton accelerators and linear accelerators devel-

oped especially for the purpose. Our reason for neglecting this topic is that we are interested here in a neutron production scale of the order of tens of moles per day at a minimum expenditure of energy.) The characteristics of the accelerator must be such that when it is combined with the target it must meet a natural requirement for a commercial installation: It must produce a maximum amount of plutonium-239 and/or uranium-233 at a minimum capital expenditure and at a minimum rate of consumption of materials. The consumption may be offset to a greater or lesser degree by the recovery of the high-grade heat evolved in the target reactor.

For several reasons, a cw linear accelerator is preferred as the base accelerator (driver) of an electronuclear reactor: It can accelerate high ion currents [pulsed proton currents $\gtrsim 300$ mA have been accelerated in several laboratories (CERN, Brookhaven, Fermilab)] and can thus provide beams with a power $\sim 10^8$ W at proton energies 1.5–2 GeV; it has the highest efficiency in cw operation; its size (about a kilometer at an accelerating-field gradient of 1.5–2 MeV/m) simplifies the injection of such high power levels into the accelerator and simplifies the servicing of the accelerator, since the induced radioactivity per unit length of the apparatus is reduced; it provides a lot of flexibility in terms of actual length and choice of construction site; and the particle beams can be completely extracted from linear accelerators.

The development of a high-current linear accelerator for large-scale production of free neutrons is becoming completely realistic (see Ref. 31, for example) thanks to two developments in the past two decades: a) The Vladimirskiĭ-Kapchinskiĭ-Teplyakov idea of combining a spatially uniform rf focusing with magnetically hard quadrupole lenses[31,45] (RFQ) avoids the difficulties inherent in ordinary Alvarez accelerators (the high injection energy at a large accelerated current and the low efficiency at which particles are captured into acceleration) and eases the problem of developing an injector and sections with drift tubes. b) Andreev structures[46] can apparently solve the problem of developing accelerating structures which can transfer at least 90% of the rf power to the beam. These are waveguides with disks, some with apertures, for which there is a strong coupling between the acceleration cells ($\pi/2$-coupling cells, with a coupling coefficient of about 0.5). The pion and neutron sources presently under development (FMIT, SNQ, PIGMI, ZEBRA, and the refined LAMPF linear accelerator) make use of these two important advances in accelerator technology. These devices[47-50] may be regarded as prototype breeder accelerators or as prototypes of units of such accelerators. It is pertinent to note in this regard that the development and use of linear proton accelerators for meson factories will make it possible not only to carry out nuclear-physics experiments with various targets and to study their radiation damage the nature of the heat evolution, etc., but also to obtain experience for eventually moving up to beam power levels higher by a factor of a hundred or so. The construction of high-power electronuclear neutron sources (on the scale of the ING or the SNQ) is a necessary intermediate step in the development of a large-scale electronuclear breeder.

The linear proton accelerator of the breeder has the standard arrangement (see Ref. 31, for example): a comparatively low-voltage injector (50-100 kV) and a spatially-homogeneous focusing section, where protons are accelerated to 2–3 MeV and then enter an Alvarez section (resonators with drift tubes). After reaching an energy of 150-200 MeV, the protons enter a section with Andreev structures, where they are accelerated to their final energy. Several hundred oscillator tubes,[51] klystrons (or gyrocons) with a total power of about 0.5 GW in continuous operation (an average power of 1–2 MW per unit) will be required for the rf pumping in a three-frequency linear accelerator of this type. The tubes will oscillate at $\simeq$70–100 MHz (for the spatially-homogeneous focusing section), 200–300 MHz (for the section with the drift tubes), and 500–1000 MHz (for the section with the resonators with the disks) at an efficiency of at least 0.7. The reliability and service life of this large number of high-power rf oscillators must be exceedingly good, especially since they are powering such high-power beams.

The cw power level of klystrons reached 0.2–0.3 MW back in the early 1970s at an efficiency $\simeq$0.4. The characteristics of klystrons have improved rapidly since then, particularly in connection with the construction of large storage-ring accelerators. Klystrons are presently being operated at a cw power of 0.6–1 MW and oscillating at frequencies of 100–500 MHz at an efficiency of $\simeq$0.6, so that there are good prospects for the development (with the help of computer simulations) of klystrons with a unit cw power of 1–2 MW which can operate at frequencies in the range 10^2–10^3 MHz at efficiencies up to 0.80–0.85 (Ref. 52).

The gyrocon can probably be regarded as an alternative power source for a section of a linear accelerator with Andreev structures.[53] Development of this rf technology is being stimulated by the gyrocon characteristics which have emerged from computer-assisted studies: efficiencies >0.8 for units with a power of 2–3 MW in cw operation and an optimum frequency interval $\simeq$300–1500 MHz (Ref. 54). Experience with the gyrocon is limited to that acquired at Novosibirsk and Los Alamos,[55] however, so that a more definite assessment of its applicability in the linear accelerators of breeders is still a few years away. Nevertheless, at the present state of accelerator science and technology it would be possible to design and construct a breeder linear accelerator with a beam power of the order of hundreds of megawatts and an efficiency of at least 0.5 (Refs. 31, 32, and 41), although it is hardly possible at this point to foresee the entire spectrum of potential difficulties.

The situation regarding the reactor target is less clear, and so far no technically feasible and economical design has been developed which best incorporates the properties of materials in intense radiation fields and at high temperatures, the production of neutrons and their effective transfer in the blanket, the evolution and removal of heat, the neutron physics of the blanket, and the distribution of the fissile material produced. The target apparatus will probably be a two-component structure (consisting of the primary target, in which the beam of accelerated ions is stopped, and the fast-neutron source appears, and the surrounding blanket with raw material and fissile materials) reminiscent in many ways of a fast-neutron reactor, so that the system could make use of many results obtained in research on fast breeders. The blankets might consist of the large units of the breeder reactors and improved converters which are presently under development (the LMFBR, the GCFBR, the HTGR, and the THTR, for example). The material and design of the primary target must provide a maximum neutron yield keeping the radiation damage low enough for at least a year of reliable operation; the target must also cope with intense and nonuniform heat evolution. For these reasons and also for maximum hardness of the neutron spectrum, it is apparently preferable to use liquid-metal coolants to remove the thermal power. It is possible that a primary target consisting of plates or rods of metallic thorium in a flow of liquid sodium would prove technically feasible. The neutron yield from a thorium target would of course be about 30% lower than from a uranium target, but the resistance to radiation damage, the mechanical properties, and the thermophysical properties of metallic thorium are much better than those of uranium.[33] Radiation damage to the materials of an electronuclear reactor is a problem no less acute than, say, that in the case of fusion reactors, since in the target of an electronuclear reactor the typical flux densities of the hard neutron spectrum will be of the order of a few times 10^{15} $cm^{-2} \cdot s^{-1}$ (Ref. 36). The cooling and radiation-damage problems would be eased greatly by sending a deuteron beam into a flow of liquid lithium or beryllium cooled with liquid sodium,[56] but there would be a substantial drop in the neutron yield.

Working from the present technical capabilities for cooling in power reactors (0.5–1 MW/liter), we conclude that at a thermal power of 1300–1800 MW the volume of the "active zone" of the target reactor would have to be at least 2–3 m^3. This result means that the volume in which most of the fast neutrons are produced (about 80% of the total number) in a continuous medium (liquid lead-bismuth or molten salts of uranium) would be, roughly speaking, a disk about two nuclear ranges thick (30 cm) and 1–1.5 m in diameter. The distributions of the neutrons and of the heat evolution in this disk would be described by exponential functions. Consequently, the conditions would be poor for the transfer of neutrons into a radial blanket, and the neutron and thermal fields would be nonuniform. As mentioned above, these difficulties could be avoided by reducing the average density of matter of the primary target along the beam of accelerated particles, thereby stretching out the neutron source and the heat evolution over space. The method for injecting the beam into the target can be chosen after several complicated auxiliary problems are resolved. One of these problems is the stability of a liquid Pb-Bi eutectic jet if the primary target is a set of heavy-liquid jets in free fall.[35] If, on the other hand, the beam is introduced into an array of thorium or uranium rods in flowing helium,[34] it would be necessary to develop vacuum windows capable of transmitting an ion beam with a power of hundreds of megawatts for at least a hundred hours of operation. It would also be necessary to develop an apparatus for automatically replacing these windows without interfering with the operation of the reactor.

Among the possibilities discussed above for using electrical energy for large-scale production of neutrons, the electronuclear method already appears to be a technically feasible method for processing nuclear raw material into nuclear fuel. Although it would not be a trivial matter, the target reactor and the linear proton (or deuteron) accelerator could be designed and constructed with existing technology. Many important characteristics of the target reactor and accelerator of an electronuclear breeder are shared with accelerators of the present generation and figure in designs for reactors proposed for the immediate future. There are no fundamental limitations of any sort (scientific feasibility does not have to be proved), although it will be necessary to carry out an extensive program of research and development, and it will also be necessary to study thoroughly the possible role of electronuclear breeding and its economics within the framework of the existing and developing nuclear power industry. The economics may be the main difficulty here, since with a comparatively narrow margin in the energy balance an electronuclear breeder looks rather expensive at present: an expensive accelerator plus an expensive target (a reactor in a new technology).

It will be necessary to measure the neutron and energy yields in targets of various compositions and configurations far more accurately. For reliable, detailed calculations we will need data on the neutron cross sections above 15–20 MeV and on the cross sections for (n, xn) reactions and reactions which do not produce neutrons. We will also need the experimental spatial and energy distributions of the neutrons which are produced in materials of interest by the accelerated and secondary particles. The target and blanket can be made of lead, bismuth, uranium isotopes, thorium, and plutonium; possible structural materials are stainless steel, titanium, vanadium, niobium, and zirconium; possible coolants are lithium, sodium, lead-bismuth, helium, and water (ordinary and heavy); and the accelerated particles could be protons or deuterons.

These experimental results would then have to be used to design and test the various components of the target, which will push the existing reactor technology to its limits. It will then be necessary—but now for the large elements of the target—to carry out detailed measurements of the cross sections, to carry out calorimetric and spectral measurements, to write computer programs describing the neutron breeding and transport in the blankets, to do the engineering, to search for the best methods for introducing energy into the target and removing it from the target, to develop a thermohydraulic system and the overall design of the actual target reactor, and to analyze in detail the neutron physics of all elements of the target. This work will of course have to be preceded by the development and selection of materials for the fuel shells, the liners in contact with the liquid part of the target, the ports through which the accelerated particle beam enters the target (there are possible systems for injecting beams without ports), and the reactor structure.

For a linear ion accelerator with an energy of 1–2 GeV, a current of hundreds of milliamperes, an efficiency of at least 0.6, and a particle loss ~1 nA/m in the course of the acceleration, we would need to develop, construct, and test a high-current injector (~0.5 A in continuous operation, with preliminary acceleration in fields $\lesssim$100 keV); to develop rf power oscillators with an efficiency $\gtrsim$0.7 in continuous operation with a unit power of 1–2 MW for pumping energy into the beam at frequencies ≈100–300 MHz (a spatially-homogeneous-focusing section and a section with drift tubes) and at frequencies three to five times higher in coupled-resonator sections (500–1000 MHz); to develop accelerating structures capable, at resonator loads of 75–80%, of transferring to the beam at least 90% of the energy of the rf field; to reach a detailed understanding of the dynamics of high-current ion beams; and to study the problem of controlling the loss of these beams during acceleration, in order to minimize the activation of the accelerator and thereby make it possible to service the installation at least partially without complicated remote-control systems. Moving up to beam powers roughly 1000 times those in meson factories will radically change the reliability requirements imposed on all parts of the accelerator.

8. At a thermal power of 1.3–1.5 GW for the target reactor and at a heat-cycle efficiency of 0.35–0.4, the electrical power (≈500 MW) would be sufficient to power a linear accelerator with an efficiency of 0.6 and a beam power of 300 MW. The cost of such a target is estimated to be about $600 000 000 on the basis of the present capital expenditure on constructing high-power reactors, $1030 per electrical kilowatt installed.[58] The cost of a linear proton accelerator (1 GeV, 300 mA, and efficiency of 0.6) is estimated to be about $350 000 000 (Refs. 34 and 35; see also Refs. 32 and 57, where costs ranging from $140 000 000 to $600 000 000 are cited). If the electronuclear reactor must be supplemented with a fuel-processing plant the cost of the latter is estimated to be $350 000 000 (Refs. 34 and 35). The cost of the fissile product based on these estimates ranges from $40 to $100 per gram[41] or, apparently more realistically, $257 per gram.[35] The cost of enriched uranium (93%) is presently about $50 per gram, of which half is the separation cost; the other $24 or so is the cost of a kilogram of U_3O_8 divided by the number of grams of uranium-235 which are extracted from this kilogram during the separation. Over the two decades from the late 1950s to the late 1970s the cost of natural uranium has increased roughly sixfold, primarily because of concern regarding shortage in the foreseeable future rather than because of any actual increase in the cost of mining it. (In 1978–79, uranium in fact started to become less expensive as soon as the demand for it began to fall off because of a temporary curtailment in orders for new atomic power plants, because of the uncertain situation, but this was a temporary phenomenon.) As the demand for uranium fuel increases, progressively greater use will be made of low-grade ore, and the increase in the cost of mining uranium will become more evident. The combination of these two factors will bring the cost to $200–$250 per kilogram of U_3O_8 in about 20 years (see Ref. 59, for example). The cost of the separation also increases, although rather slowly, so that in about 20 years the cost of enriched uranium will increase by a factor of two or three, reaching $70–$100 per gram. If the assumed cost of nuclear fuel processed by means of an accelerator is still regarded as intolerably high, as it was in the MTA project,[16] where it was $124 per gram ($250 per gram when referred to the conditions in 1976), the electronuclear method will be economically justified 20 years from now. Simple economic considerations show[60] that the electronuclear method for producing nuclear fuel becomes competitive when the cost of enriched uranium exceeds $80 per gram. If the cost of electrical energy increases, on the other hand, this limit will rise.

Let us take a cursory look at the economics of a breeder and at the main variables involved. We denote by k_B the specific cost of a breeder (the capital cost plus other costs, referred to the starting time, without any expenditure for fuel); we denote by k_T the same quantity for a thermal reactor (k_B and k_T are in rubles per kilowatt); and we denote by η_B and η_T the net efficiencies of these reactors. The quantity

$$n=\frac{r_B-\omega g_B}{q_T+\omega g_T}$$

is the number of thermal reactors which would be supplied fuel from one breeder of the same power (the growth dynamics is taken into account), where r_B (in tons per gigawatt per year) is the excess plutonium production by the breeder per gigawatt of themal power, g_T (in tons per gigawatt per year) is the plutonium requirement of a thermal reactor, ω (in reciprocal years) is the growth rate of the nuclear power industry and g_B and g_T (in tons per gigawatt) are the plutonium inventories in the breeder, in the thermal reactors, and in their external cycles. If we consider a system which is closed in terms of plutonium and which consists of a single breeder (1 GW thermal) and n thermal reactors, we conclude that it produces $\eta_B+n\eta_T$GW of useful power and costs k_B+nk_T million rubles. A system with the same useful power but consisting of thermal reactors exclusively would cost $(n+\eta_B/\eta_T)k_T$ million rubles. The difference, $k_B-(\eta_B/\eta_T)k_T$, is what we would spend on producing fuel for this system. The ratio of this cost to the cost for thermal reactors alone,

$$B=\frac{(k_B/k_T)-(\eta_B/\eta_T)}{n+(\eta_B/\eta_T)},$$

is the fraction of the total expense which goes for fuel in the system with a breeder. This ratio can be used for comparisons and optimization.

At present the corresponding quantity (the cost of enriched uranium) is about 10% of the total expenditure on the nuclear power industry, but it is tending upward. This tendency can only be accelerated as high-grade uranium reserves are exhausted, and the economic role of breeding is to prevent the nuclear power industry from losing its economic advantages in this eventuality. These advantages stem specifically from the low cost of the fuel; the values of k_T are relatively high for nuclear power. (Nuclear power will not be the unique energy solution in the foreseeable future, since coal will last another 100 years or more.) Consequently, the requirement made of breeding is that the quantity B be substantially less than 1, say no more than 0.3 (in this case breeding becomes advantageous when the cost of uranium increases to about three times its present level.

We can now find the tolerable cost of a breeder as a function of n and η_B/η_T:

$$\frac{k_B}{k_T}\leqslant 0.3n+1.3\frac{\eta_B}{\eta_T}.$$

For electronuclear breeders with unenriched targets, as for hybrid thermonuclear reactors, we would have $n \approx 3-4$ if we use existing thermal power reactors (of the water-cooled, water-moderated type with an oxide fuel), with the fuel balance characteristic of these reactors. We could of course choose for this purpose some reactor with a far better fuel balance, in which case n would be $\approx 10-15$ or perhaps even higher. If we do decide to make this switch, however, breeding will not be needed as soon and should be postponed. Actually, breeding is also serving the purpose of making it unnecessary to develop new types of reactors. With $n=4$ we would have

$$\frac{k_B}{k_T}\leqslant 1.2+1.3\frac{\eta_B}{\eta_T}.$$

This expression shows how strongly the energy balance of a breeder affects its economics, despite the fact that 80% of the energy in the system is generated by thermal reactors. The reason for this conclusion is that the fuel produced by the breeder for the nuclear power industry is (and must be!) cheap in comparison with the energy which is actually produced. It can be seen from this expression that at $\eta_B \approx 0$ [a breeder with a slim energy balance: an electronuclear breeder, a hybrid thermonuclear breeder with a low Q (sub-Lawson), etc.] the cost of a breeder can exceed the cost of a thermal reactor by no more than 20%, and the breeder can essentially be ruled out. If, on the other hand, $\eta_B/\eta_T=1$, then we would have $k_B/k_T \leq 2.5$, and it is probably feasible. When we get to $k_B/k_T \approx 2$, however, the expression for B shows that a breeder with $\eta_B/\eta_T \approx 0$ will produce plutonium at a cost 2.5 times that of a breeder of the same cost but with $\eta_B/\eta_T \approx 1$. We see that the difference is rather large.

An electrical breeder will apparently be desirable even earlier, since this fuel source, which would be entirely independent of the natural uranium reserves and which would make widespread use of depleted uranium, would stabilize the cost of nuclear fuel permit use of the thorium-uranium-233 fuel cycle, which is very advantageous for thermal reactors and would expand the fuel base even further. The electrical breeder would reduce the demand for new separation facilities and possibly for new radiochemical facilities. Furthermore, one approach in the accelerator method for producing free neutrons—producing thermal neutron fluxes $\sim 10^{16}\ \mathrm{cm^{-2}\cdot s^{-1}}$—could furnish a tool for converting (transmuting) certain of the fission products and actinides into short-lived and even stable nuclides.[61]

9. As the economic and ecological problems of the power industry become more acute, a progressively greater burden must be shouldered by nuclear power. It will hardly be sufficient to simply scale up the nuclear facilities; instead we will need new types of reactors for various branches of the power industry, and we will need new methods for breeding nuclear fuel. The electrical breeding methods which we have discussed here open up a new approach to the development of a large-scale, versatile nuclear power industry, whose extensive development is becoming particularly urgent and economically advantageous for the Soviet Union, with the large fraction of its petroleum and natural gas which is used in producing electricity and heat, with its well-developed system for central heating of cities, and with the long distances between its principal coal deposits and the regions with the greatest demand for power.

The electronuclear and hybrid thermonuclear reactors, by supplying fuel to fission reactors, would be capable of increasing the degree to which natural ore (uranium or thorium) is utilized from the present 0.5–1% to 40–70%, where the unavoidable loss is taken into account. In addition, these reactors raise yet another interesting possibility (which deserves study) for greatly postponing the exhaustion of our natural ore reserves without resorting to an intermediate radiochemical regeneration or refabrication of fuel. Specifically, it might be possible to use the uranium or thorium blankets in these reactors to enrich fuel elements (thorium or uranium) with plutonium-239 or uranium-233 to the concentration of 2–4% required for thermal reactors and then burn these elements in fission reactors. Even if such fuel elements are used only once, the degree to which natural uranium is utilized could be raised to 20–40 kg/ton, i.e., 4–8 times the present level. If the level of radiation damage to the fuel elements permits their return to the blanket of an electronuclear or hybrid thermonuclear reactor for a repeated enrichment, the degree to which the ore is utilized could be doubled again. When we take into account the demand on thorium reserves (in the absence of a special technology for reprocessing and refabricating thorium), this result means that we would have solved the problem of our nuclear fuel reserves essentially to the end of the next century, and we would have provided a substantial time interval for finding the best solution to the complicated technical, ecological, and political problems accompanying the reprocessing of spent fuel, the storage of high-activity waste, and the creation of large plutonium inventories.

The high productivity of electrical breeders in terms of the nuclear fuel (about 2 tons per electrical gigawatt per year in contrast with 0.2–0.4 tons for fast breeders) would make it possible to use different types of reactors—both reactors which produce fuel and those which consume it—each working in a particular sector of the nuclear power industry where it is most advantageous. There would be no need to attain the maximum possible characteristics in the production and consumption of fuel by fast and thermal reactors, and the effort devoted to the development of these reactors could be concentrated on improving their reliability and economy. There would be no need to reduce to the absolute minimum the time the fuel is stored after it is spent; this time could be instead the optimum time from the economic standpoint (1–2 yr). In fact, it may be possible to defer reprocessing for a substantial time, until the most economical and reliable technology has been developed.

Each of these schemes for electrical breeding has its own advantages and disadvantages. The development of the electronuclear breeder, for example, can draw on extensive experience in reactor and accelerator technology, and the underlying physical principles are quite clear. However, the economics of this approach is not clear, and although the energy balance is positive the margin is slim. The muon catalysis of the DT fusion reaction might improve the energy balance of electronuclear breeding, but at the cost of serious technical complications. Thermonuclear breeding has a good energy balance, at least for reactors with a high power gain (the tokamak, for example), but it will require the solution of many fundamental and engineering problems. A thorough study must be carried out to identify the most promising breeding scheme, since it will be too expensive to pursue all the reasonable possibilities to any great extent.

In examining the outlook for some particular energy technology, however, it is crucial that we avoid examining only a single aspect of the problem (the fuel reserves, the ecological qualities, etc.). The most graphic example here is solar energy, which would seem faultless in this regard. The fact is that the low intensity of sunlight severely restricts its direct use in the near future to certain particular applications. Approaching the question from a practical standpoint, i.e., considering time intervals within a century, we must start with the understanding that there exists not a single energy solution but a set of energy solutions, including coal and nuclear fuels. The choice of the best power technology or, more probably, the best balance of a variety of technologies will be determined by their various properties, summarized in economic indices.

Comprehensive study will reveal the most economical and technologically acceptable solutions for implementing the new industrial method of producing neutrons through the use of electrical energy. Since electrical breeding is a method for producing nuclear fuel which is not dependent on uranium reserves it could insure the nuclear power industry against a possible shortage of natural uranium and of separation facilities; it could stabilize the cost of uranium fuel; it could provide fuel to power reactors of the present generation (such as the water-cooled, water-moderated reactor and the channel-type high-power uranium-graphite boiling-water reactor); and in the future to improved converters. Electrical breeding could prevent a possible future shortage of plutonium which might result from the rapid and widespread adoption of fast-neutron power reactors. Although the present cost of producing nuclear fuel by electrical breeding is estimated to be several times the cost of enriched uranium from existing separation facilities, as the world returns to a rapid development of nuclear power the situation will change in about 20 years. This is just the time interval over which it would be possible to develop and to begin to install an optimum electrical breeder capable of not only generating fuel but also producing neutrons for other purposes (including, possibly, a partial reprocessing of radioactive wastes) before pure fusion reactors and asymptotic reactor systems (gas-cooled fast and thermal breeders) begin to find widespread use.

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