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Introductory Remarks

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Introductory remarks

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The fast-neutron breeder reactor is the principal means now envisaged of exploiting the very large resource of energy residing in the naturally abundant isotope of uranium, ^{238}U. Extensive research and development programmes are being carried out in a number of countries to realize this potential. There are about a dozen substantial reactors operating; and wide-ranging supporting programmes include fuel processing and development, and safety and environmental issues. The purpose of this Discussion Meeting is to present the principal recent scientific and engineering results of these world-wide programmes; and, in the final session, to discuss the future trends in this research and its utilization.

During the 50 years since the discovery of nuclear fission in uranium (Hahn & Strassman 1939; Frisch & Meitner 1939), research and development has been carried out world-wide to establish this form of nuclear energy as a major long-term source of energy for practical application. The fast-neutron breeder reactor (FBR) is the principal means now envisaged of exploiting the very large source of energy which resides in the naturally abundant isotope of uranium, ^{238}U. This potential contrasts with that of existing commercial nuclear power stations where the ultimate resource is limited by the availability of the comparatively rare (0.7 %) isotope ^{235}U.

By using the relatively abundant ^{238}U, some 50 times more energy can be extracted from a given amount of natural uranium than can be obtained in thermal neutron plant, and very much lower grades of uranium ore become economically exploitable, constituting a much larger resource of uranium. The average concentration of uranium in the granitic rocks of the continental crust is about 4 p.p.m. (by mass) (Taylor 1954; Spurgeon 1988). At this concentration, the total uranium fission energy in the rock is about 3×10^{11} J t^{-1}; this is approximately 10 times the energy of combustion of a tonne of coal, whereas that of ^{235}U is 10 times less (table 1); also shown are other sources of nuclear energy in granitic rocks. Thus the ultimate, but not unique, benefit of the successful development of FBRs and their associated fuel cycle, will be to make available a practically limitless source of energy, and one that can be fuelled by widespread low-grade sources of uranium.

As a consequence, many industrial nations have undertaken extensive research and development programmes for FBRs (Walter & Reynolds 1981; Weaver *et al.* 1988). Development has reached the stage of operating experimental power-producing plant of order 100 MW$_e$ in a number of countries, including the U.K. In France, a West European consortium started operating, in 1986, a 1200 MW$_e$ plant Superphénix, whose size is close to that of envisaged industrial plant.

The FBR uses the ^{238}U primarily by converting it by neutron capture to the more-readily-fissionable nuclide ^{239}Pu. This is the breeding process; it takes place in the reactor which consists of an assembly of fuel elements, containing ^{239}Pu and ^{238}U, immersed in a heat transfer

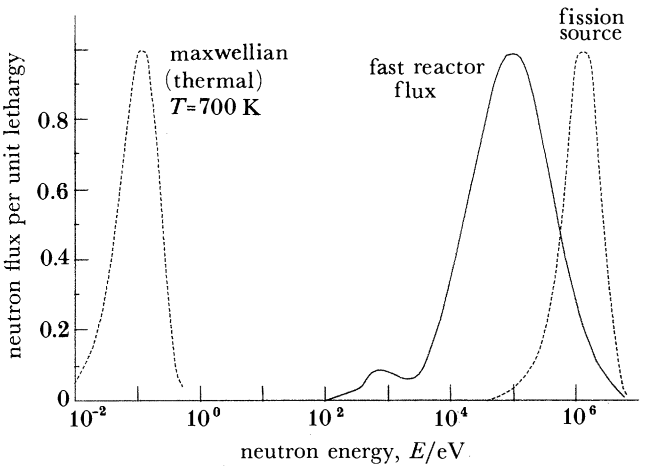

FIGURE 1. Neutron energy spectra. The spectra are normalized to the peak value. (From B. H. Mitchell, personal communication.)

TABLE 1. NUCLEAR ENERGY IN GRANITIC ROCKS

(Illustrative numbers based on mean abundances of Taylor (1964).)

isotope	mean concentration (p.p.m.)	related energy content[a] (J t^{-1})
^{235}U	0.03	2×10^9
^{238}U	4	3.5×10^{11}
^{232}Th	17	1.3×10^{12}
^{6}Li and ^{7}Li (fusion)	30	0.8×10^{13}
combustion of coal		3×10^{10}

[a] The related energy content is obtained by multiplying the mean concentrations by 200 MeV per fission for ^{235}U, 213 MeV for ^{238}U, and 195 MeV per fission for ^{232}Th. The achievable energy releases can be more, as in the case of ^{235}U where significant ^{238}U energy is also released in thermal neutron reactors; or less, as in the case of the fusion energy available in Li, where it is unlikely that all the Li can be used to breed tritium.

fluid. To ensure both that the neutron chain reaction – common to all nuclear fission reactors – is sustained, and that there are sufficient neutrons to be devoted to breeding from ^{238}U, the mean energy of the neutrons has to remain high compared with thermal energies, and close to tbeir initial energy; i.e. the neutrons are fast moving. In practice, the bulk of the neutron velocity spectrum corresponds to the energies of 1 keV to 1 MeV (figure 1). This spectrum, together with a breeding blanket containing ^{238}U surrounding the core, secures adequate ratios of the cross sections for neutron production, for neutron capture in ^{238}U, and for parasitic capture of neutrons in the coolant and structural elements of the reactor. It enables more ^{239}Pu to be bred from the ^{238}U than is consumed in sustaining the reactor output.

Table 2 lists some of the leading reactors that have operated, and from which much of the experimental data has been drawn. A marked feature of the development is the concentration on oxide-fuelled sodium-cooled reactors. They operate with high power densities in the core, of several hundred megawatts per cubic metre. Larger total power outputs are achieved by enlarging the volume of the core. This is done by expanding the diameter of the near-right cylindrical core of the smaller reactors to the pillbox-shaped cores of the higher-power reactors.

TABLE 2. SOME PRINCIPAL FBRS

(Walter & Reynolds 1981; Weaver *et al.* 1988; B. H. Mitchell, personal communication.)

name	country	date to power	thermal power MW	core height m	core diameter m	core fuel	coolant
Clementine	U.S.A.	1949	0.025	0.14	0.15	Pu metal	Hg
EBR 1	U.S.A.	1951	1.2	0.22	0.18	enriched U	Na–K
EBR 2	U.S.A.	1965	62	0.36	0.51	U metal	Na
SEFOR	U.S.A./F.R.G.	1970	20	0.93	0.88	UO_2–PuO_2	Na
FFTF	U.S.A.	1980	400	0.91	1.21	UO_2–PuO_2	Na
BR 5	U.S.S.R.	1959	5	0.28	0.28	PuO_2–UC	Na
BN 600	U.S.S.R.	1980	1470	0.75	2.06	UO_2	Na
DFR	U.K.	1962	60	0.53	0.53	U–Mo	Na–K
PFR	U.K.	1975	600	0.91	1.47	UO_2–PuO_2	Na
Phénix	France	1973	570	0.85	1.39	UO_2–PuO_2	Na
Superphénix	Fr/W. Eur.	1986	3000	1.00	3.7	UO_2–PuO_2	Na
KNK II	F.R.G.	1978	58	0.60	0.82	UO_2–PuO_2	Na
Joyo	Japan	1978–82	75–100	0.55	0.76	UO_2–PuO_2	Na
FBTR	India	1985[a]	40	0.32	0.45	UC–PuC	Na

[a] Date to criticality.

A wide range of disciplines collaborate in FBR research: the physics of heavy nuclei, and their neutron reactions; the engineering of fast neutron chain reactions; the use of liquid sodium as a coolant, which has involved many new developments in fluid dynamics, chemical engineering and metallurgy. In the reprocessing of fuel elements, the extraction of the ^{239}Pu from irradiated fuel elements with the efficiency needed has required major developments of chemical engineering. The demands of safety have involved researching diverse physical phenomena and extensive engineering analyses.

The purpose of this Discussion Meeting is to place before a wide audience the principal scientific and engineering results of these world-wide programmes. As shown in table 2, many of these experimental plant have been operating for a number of years. Now is therefore a good time to take stock of their performance and potential, and to review the major scientific and engineering results of a substantial public investment in long-term energy development.

The large-scale use of nuclear fission energy is a controversial issue in some countries: the safety of the plant and processes envisaged for FBRs, the large-scale use of plutonium with its implications for military potential, the production of long-lived radioactive waste, and the economics of electricity generation, are issues which, together with the benefits of this energy resource, command a major public interest in FBR research (Anon. 1976; Von Hippel *et al.* 1985).

Consequently, we have planned the programme to provide for presentations as follows. The main issues of the design and performance of the reactors themselves are given in sessions I and II; the results of work on the fuel cycle, and its environmental aspects are in session III; the safety and risks analyses that have been carried out in session IV. The disposal of radioactive waste has been discussed recently at another Royal Society meeting (Roberts *et al.* 1986) and is therefore discussed here only in respect of issues special to the FBR. We have reserved much of the fifth and final session for a discussion of the wider issues and of the future trends in this research and its utilization.

We are most grateful to the eminent speakers who have come, many from overseas, to contribute papers to this meeting, and to those who will join our discussions of this important topic. We hope that the outcome of this discussion meeting will enable our wider audience to assess better the potential of the fast-neutron breeder reactor. I am grateful to Mr B. H. Mitchell of Associated Nuclear Services for assistance in preparing this paper.

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