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Nuclear Energy Resources for Electrical Power Generation

by

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SUMMARY. "Nuclear Energy Resources" is interpreted as the nuclear power systems currently available commercially and those at an advanced stage of development, together with fuel and associated resources required to implement large-scale nuclear programs.

Technical advantages and disadvantages of the established power reactor systems are reviewed, and the uranium fuel situation is outlined in terms of supply and demand, the relationship of resources to the requirements of current reactor types, and the likely future implications of the Fast Breeder Reactor (FBR). Because of its importance for the future, the problems, status, and likely time scale of the FBR are discussed in some detail.

It is concluded that the most important areas for near-term attention in Australia are the criteria and conditions that would apply to nuclear installations, and the possible development of uranium fuel cycle industries. The pattern of development of reactor and fuel cycle strategies overseas is important for uranium industry planning, and in the long term plutonium availability may be a key factor in power and energy planning. Finally, acceptance of nuclear power includes acceptance that its radioactive wastes will have to be stored on earth, and recent developments to demonstrate that this can be done safely and economically are very important in terms of long-term public attitudes.

1 INTRODUCTION

This technical review covers nuclear energy resources available now and likely to be available over the next 2-3 decades, i.e. the nuclear power generating systems currently available commercially, and those under development, together with fuel and associated industrial resources required to implement large-scale nuclear power programs.

It has become customary to compare nuclear power with "conventional" forms of generation using fossil fuels or hydro-electric installations. If current plans for future capacity in Europe, the USA and Japan are realised (Ref. 1) nuclear power will become the conventional form in some areas by the turn of the century. The reactor types on which these plans are based are reviewed technically in terms of main design features, operational aspects, safety, and to some extent, costs.

Recent changes in availability and price of fossil fuels (particularly petroleum) have added emphasis to the future role of nuclear power. However they have also made it difficult at present to make economic comparisons between nuclear and "conventional" power; further, the cost effects of environmental policies seem likely to be important economic factors in future. Because of the magnitude of these unknowns no attempt is made in this paper to present quantitative economic comparisons.

A key question is the availability of nuclear fuels; it is considered that reasonable assumptions on both nuclear fuel resources and developments in technology indicate that nuclear power has the potential to supply man's needs for base-load power generation into the indefinite future, if required. At some future time it may well give way to other technologies - e.g. fusion, or solar energy - but this appears unlikely during this century.

2 HISTORICAL

Over the past 25 years many nuclear reactor types have been proposed and investigated to various degrees. Only three basic types have been successfully applied to large-scale power generation. It is interesting in retrospect that these followed the strongest lines of development being pursued 20 years ago, in the United States (light water moderated and cooled reactors, LWR's), in Canada (pressurised heavy water moderated and cooled, PHWR), and in the United Kingdom and France (gas cooled graphite moderated). Currently, advanced forms of gas cooled reactor are attaining commercial status in the USA and are also und development in Europe and Japan.

Light water reactors, which use enriched uranium fuel, have captured the largest share of the total market and at mid 1973 accounted for 87% of the total capacity (megawatts electrical, MW(e)) in operation or firmly committed.

LWR's are of two types, the pressurised water reactor (PWR) and the boiling water reactor (BWR), both of which originated in the United States. The PWR was developed initially using technology applied earlier in the first submarine of the US nuclear navy (Nautilus). Except for one unsuccessful sodium-cooled thermal reactor (in USS Sea Wolf), all subsequent maritime nuclear propulsion units have used PWR's. The first civil prototype of the PWR type (Shippingport) started up in 1957. At mid 1973 there were 139 PWR's operating or committed aggregating over 125,000 MW(e). The first experimental BWR operated at the Argonne National Laboratory in the USA in 1956 and the prototype power station (Dresden) started up in 1960. At mid 1973 there were 81 BWR's operating or committed, aggregating 76,000 MW(e).

The predominant heavy water reactor (PHWR) is

Canadian, often called CANDU (Canadian Deuterium and Natural Uranium) after the prototype Douglas Point (208 MW(e)), which entered commercial service in 1967. It was preceded by an experimental reactor (NPD, 18-22 MW(e) 1962) and by a research program on heavy water moderation which began in the mid 1940's; this included construction of the heavy water research reactor NRX (1947), for about a decade the world's most powerful research reactor.

After a difficult development period beset by many technical problems (e.g. fuel failures, heavy water leakage and losses, on-load fuel changing difficulties) the PHWR has become outstandingly successful in Canada. The Pickering generating station, consisting of four 500 MW(e) units, was commissioned over 1971-73 and is now the world's largest operating nuclear station. Its reliability record is excellent. The Bruce station now under construction consists of four 750 MW(e) units, and plans for a further 8,000 MW(e) have been announced. Small PHWR stations are operating in Pakistan and India; during 1973 Argentina ordered a 600 MW(e) station, and Korea has announced a similar intention.

In the United Kingdom the first nuclear power program (5000 MW(e)) utilised the "Magnox" series of gas-cooled (carbon dioxide) graphite moderated reactors, for which the technology was developed from earlier air-cooled military plutonium-producing reactors (Windscale). Prototype plants at Calder Hall (1956) and Chapel Cross were followed by nine increasingly larger stations, the latest being Wylfa (1180 MW(e), in two reactors). A smaller program using similar reactors was followed in France. Stations of this general type, using natural uranium metal fuel, are unlikely to be built in future because of their high capital cost, associated with their very large size. Nevertheless existing U.K. stations have given a notable performance both technically and economically in recent years, despite some downrating because of corrosion problems.

The Magnox series was followed by the Advanced Gas-Cooled Reactor (AGR) in the U.K. second nuclear program. The AGR uses uranium dioxide (UO_2) fuel clad in stainless steel with a gas (CO_2) exit temperature of about 600°C (to be compared with 380°-400°C for Magnox) but requires enriched uranium. A 32 MW(e) prototype (Windscale, 1963) was followed by orders for ten reactors in five stations. None of these are operating yet; the program has been beset by many delays and considerable rises in cost, and it seems unlikely that further reactors of this type will be constructed.

The future of gas-cooled thermal reactors appears to rest with higher temperature systems cooled with helium. Such systems have been under development for over a decade, principally in the USA (Gulf General Atomics, HTGR), Germany (AVR "pebble bed"), and OECD (the Dragon project). France and more recently Japan are also working on high temperature reactor (HTR) systems, the latter with particular interest in their potential to produce high temperature process heat for steel-making.

In the United Kingdom, a HTR design has evolved from a combination of AGR and Dragon technology; as in the USA and Germany, a major factor in this concept is the success of "coated particle" fuel development, the fuel being in the form of microspheres individually encapsulated in fission-product impermeable graphite. The HTR concept in the U.K. is based on a low enrichment fuel cycle, whereas the HTGR in the USA requires high enrichment and is directed towards the use of thorium feed to a fuel cycle involving conversion to U233. This cycle offers high conversion efficiency, and potential for very low fuel costs. The HTGR in the USA is the most advanced HTR in terms of commercial acceptance; after successful operation of a 40 MW(e) prototype (Peach Bottom), a 330 MW(e) station (Fort St. Vrain) was commissioned in 1973, and orders for over 7000 MW(e) have been placed.

Thus in terms of nuclear energy resources, the systems available to generating utilities as "established" technology today are LWR's, PHWR's, and HTGR's, the latter not yet demonstrated in a large sized unit. There are several other thermal reactor systems at the reactor experiment or prototype stage, but their future in commercial terms is unclear. The most advanced is the Steam Generating Heavy Water Reactor in the U.K. (Winfrith prototype, 100 MW(e)); similar systems, in the engineering sense, are under development in Japan (Advanced Thermal Reactors, ATR), and Italy (CIRENE). Canada built a natural uranium fuelled 250 MW(e) prototype (Gentilly 1). Research and development is still being funded in the United States on the last surviving "homogeneous reactor" concept, the Molten Salt Reactor, but again the future is unclear. The long term prospects for these systems, and indeed for the HTR, will depend on the technical and economic success, and the time scale, of the Fast Breeder Reactor (FBR), which is discussed below.

3 ESTABLISHED THERMAL REACTORS - ADVANTAGES AND DISADVANTAGES

Commercial ordering of LWR, PHWR, and HTR systems indicates that in their particular circumstances, the utility customers are satisfied that their performance will be satisfactory in economic terms. Attention will be concentrated therefore on advantages and disadvantages from the technical and operational viewpoints, including safety and environmental aspects and fuel cycle factors, with qualitative attention to future effects on economics.

a) Pressurised Water Reactors

"Nothing succeeds like success" is very apt for the PWR. Electrical generating utilities are naturally conservative in choice of equipment, giving considerable weight to experience and number of units in service. The PWR system acquired a flying start from naval propulsion technology, but probably the greatest initial advantage, still relevant, is that it uses the conventional coolant, high pressure water, with which utility engineers are familiar. Although heat transport equipment to special standards had to be developed, basically the technology is already understood by the customers.

Commercially the PWR has advantages in being available competitively from vendors in the USA, Germany, Sweden, Japan, the USSR, and recently France. Technically, the use of a closed primary circuit loop assists with control and containment of any radioactive leakages from faulty fuel and radioactivated impurities in the coolant (e.g. corrosion products). The PWR has excellent control characteristics, being to some extent self-regulating, accommodating minor load changes without control absorber movement.

There are some constructional disadvantages because the massive steel pressure vessel requires shop fabrication in special facilities and presents transport problems. The use of enriched fuel makes PWR (and BWR) more sensitive to uranium and enrichment price than PHWR systems. However probably the greatest disadvantage of the PWR is the controversy over the safety of its primary circuit - the pressure vessel and primary loop components (pipework, pumps, valves, steam generators) - and the effectiveness of the Emergency Core Cooling System (ECCS) in the unlikely event of a burst circuit depressurisation accident.

Despite safety criticism of the massive high-pressure steel pressure vessel, no country has actually refused to license PWR stations. Licences in all cases are subject to strict application of stringent criteria on design and construction, and of in-service inspection. This regular inspection by remote-controlled equipment (usually ultrasonic) at intervals throughout the life of a PWR is an operational disadvantage and expense, but since access to fuel requires shutdown, depressurisation, and removal of the pressure vessel head several times a year, in practice it can be combined with fuel changing down-time. Difficulty of access to fuel in a PWR becomes a major disadvantage and expense in the event of fuel failure severe enough to require its identification and removal.

Safety controversy in countries using PWR's appears to be largely resolved; in some cases this has involved some down-rating and limitations on overall size, fuel rating, and power density. While adequate quality, quality assurance, and in-service inspection have satisfied licensing bodies on pressure vessel safety, a potential economic problem remains, discovery of a new defect of safety significance during in-service inspection could mark the premature end of life of a PWR vessel.

b) Boiling Water Reactors

In many respects the BWR system has similar advantages to the PWR; in experience, units in service, use of familiar coolant, and availability from a variety of vendors. It has similar disadvantages in access to the core requiring prolonged shutdown. In principle it is simpler than a PWR, having no separate primary loop or heat exchangers (steam generators or boilers) - the coolant boils in the core. This gives rise to a number of advantages and disadvantages.

Elimination of the heat exchangers is a major advantage, because in practice these apparently simple devices have been a major source of trouble in PWR systems. However the core of a BWR reactor and the associated control system is very complex because of the need for extensive in-core instrumentation and controls to "power flatten" the reactor to ensure the highest possible steam exit quality.

There has been less criticism of safety in relation to the prospects for catastrophic accidents for BWR than for PWR. This is because operating pressures are lower (typically 900 psi, against 2500 psi for PWR), ingress of emergency coolant is less difficult, and fuel ratings are lower. However BWR systems attract more criticism in terms of operating safety in relation to the environment. Being a "direct cycle" system, any radioactive leakage from fuel, carried by the coolant, passes through the turbine and condenser. This leads to problems and costs in active area maintenance; it also necessitates additional attention and expense in design and operation of condenser off-gas systems to minimise radioactive gas releases. The BWR system also encounters problems (particularly in salt water cooled plants) in the event of condenser leakage, because coolant requires full-flow feedwater purification, and this leads to a disposal problem with radioactively contaminated ion exchange media.

c) Pressurised Heavy Water Reactors

The Canadian PHWR is now the only natural uranium fuelled system available commercially. It has suffered because of its relatively high capital cost, associated with its large size for a given power, and the high cost of its heavy water inventory. However the PHWR has very low fuel cycle costs; it is the least sensitive system to uranium price, and requires no enrichment services. Therefore as both uranium and enrichment prices are expected to rise in future, the fuel cycle cost advantage of the PHWR should increase.

The use of natural uranium carries a significant advantage for countries desiring maximum independence in nuclear fuel supplies. PHWR fuel is relatively simple and the manufacturing know-how is available from the vendors. A country possessing uranium resources therefore can become fully independent; Argentina is developing in this direction.

Design of a natural uranium PHWR requires an absolute minimum of parasitic neutron absorbers in the reactor core, and also facilities for on-load fuel changing. Fuel bundles are housed in zirconium alloy pressure tubes, and charged and discharged by remote-controlled machines. Pressure tube design has to be a compromise between minimising neutron absorption and ensuring adequate lifetime under "irradiation enhanced creep" conditions. Pressure tubes can be changed, which is an advantage; on the other hand, the operation and down-time are very expensive.

The PHWR system has constructional advantages in that much of the reactor core (the calandria and pressure tube assemblies) can be shop fabricated. Operationally it has a major advantage in that with on load fuel handling there are no nuclear reasons requiring shutdown.

Not all PHWR systems are entirely natural uranium reactors. To enable the reactors to start up quickly after unscheduled shutdown for any equipment malfunction or maintenance, booster rods containing enriched uranium (or plutonium) are required, to add reactivity to override the neutron poison xenon which builds up in the fuel within about 30-40 minutes of shutdown. Without xenon override boosters, shutdown may be extended to 30-40 hours, until the xenon decays. However if the reactor is operated without using the boosters for other purposes (e.g. to assist in load following), in theory they can be made to last the life of the plant without replacement.

The use of pressurised heavy water coolant is a deterrent for utilities, partly because it is unfamiliar, but mostly because of its cost and the economic penalties of any major leakages. Recent Canadian experience in minimising losses has been very impressive. There are some operational problems with heavy water because of buildup to an equilibrium concentration of tritium (by neutron capture in deuterium), and tritium is a health hazard. Consequently maintenance in areas subject to heavy water leakage may have to be done in fully protective clothing, and this is a deterrent in terms of cost and possibly industrial problems.

d) High Temperature Reactors

As indicated earlier, the only HTR systems ordered commercially to date are of the HTGR type, developed by Gulf General Atomic (now in partnership with Shell). Units of 770, 1100, and 1200 MW(e) have been ordered, but operational experience has been obtained only on an early 40 MW(e) reactor and more recently a 330 MW(e) station (Fort St. Vrain). The use of helium requires the development and proving of large new circuit components; so with this system scaling up is more likely to give rise to problems than in water-cooled reactors.

The HTGR has a considerable advantage in thermal efficiency. The high coolant outlet temperature (approximately 900°C) associated with virtually an all-graphite core structure and helium coolant results in overall efficiency of over 40%, in contrast with all water-cooled systems which are limited to about 300°C and about 30%. Consequently steam conditions similar to those in modern fossil fuel fired plants are attainable, and this has both economic and environmental advantages (minimising thermal pollution).

The system has safety advantages under both operational and accident conditions. The use of a pre-stressed concrete pressure vessel eliminates the problem of catastrophic failure; it has been demonstrated that extreme over-pressurisation to the failure point results in cracking, gas release, then relaxation. Helium is a "clean" coolant in that it does not transport released radioactivity from faulty fuel around the circuit to the extent shown by water; also the fission product retention integrity of coated particles in HTGR fuel is excellent.

The core of a HTGR contains very little parasitic neutron absorbing material, consequently the "neutron economy" is outstanding, and this results in high efficiency of utilisation of fuel. The HTGR is designed to use highly enriched uranium with thorium as fertile material to convert to U233 for recycling, and the thorium-U233 fuel cycle has potential for the lowest fuel costs of any thermal system. However the thorium-U233 cycle has not been implemented on an industrial scale and there is therefore uncertainty about costs of some stages, in particular the head-end treatment in fuel reprocessing. While U233 is in theory easier to handle than plutonium, being of lower specific activity as an alpha particle emitter, it contains some U232 which decays to gamma-emitting daughter products requiring shielding and possibly leading to high fuel fabrication costs.

The requirement for highly enriched uranium for HTGR fuel may cause difficulties and expense associated with the very stringent safeguards that will be necessary. The use of helium also carries some economic problems; it leaks easily, it is relatively expensive, and to some extent it is a strategic commodity.

In the long term, the HTR/HTGR is the only type of system with prospects for use as a source of high temperature process heat. A gas exit temperature of about 1000°C would not require major extensions of existing technology. However the transfer of heat from helium at this temperature to a process gas, probably reducing, presents difficult materials problems.

4 FUEL AVAILABILITY

a) Uranium Resources, Production, and Needs

Current nuclear power programs and most future planning are based on the uranium fuel cycle. Some use of thorium is foreseen in HTGR systems, and possibly in PHWR should uranium become sufficiently scarce or expensive, but no concern has been expressed about adequacy of thorium, which is fairly plentiful in beach sand deposits.

Traditionally uranium reserves and resources have been expressed in terms of short tons of uranium oxide (U_3O_8) in concentrate ("yellowcake"), or in tonnes of uranium, recoverable at less than US\$10 per pound of U_3O_8, plus higher cost resources in the range \$10-\$15/lb.

However in recent months base prices for uranium deliveries over the next 10-12 years have been reported in the ranges US\$9-\$10/lb for 1974/75, rising to \$19-\$20 in 1984/85. These are in 1973 dollars, and escalation will be applied. Thus prices in current negotiations for future deliveries already exceed the monetary values used to quantify resources, so it is difficult to interpret even recent published data (Ref. 2,3) to estimate the quantity of uranium now known to be available in terms of economically recoverable fuel for the next few decades.

In current power reactors, uranium price contributes 5-10% of the cost of power generated; in fossil fuelled plants the fuel cost contribution is much higher, perhaps 40% - this figure is also hard to estimate now. Nuclear generation is certainly much less sensitive to fuel price, but at present it is impossible to estimate the price a nuclear plant operator will be able to afford to pay for uranium in 15-20 years' time, in competition with fossil fuels.

Taking 1973 figures (Ref. 2) the sum of "reasonably assured" and "estimated additional" resources in the non-Communist world recoverable at less than \$15/lb (U_3O_8) amounted to about three million tonnes of uranium, of which over half was regarded as firm. Estimates of cumulative demand are in the ranges 1.5-2 million tonnes by 1990, and 3-4 million by the year 2000. While there appears to be no immediate problem in terms of overall reserves, it will be necessary to discover a lot more over the next 20 years; the discovery rate averaged 65,000 tonnes per year over the past eight years, but on present estimates will have to rise to about 230,000 tonnes per year over 20 years. In recent years an over-supply and depressed market prices caused decreases in exploration, through lack of commercial incentive.

These same factors have restricted growth in mining and milling capacity, and here there is serious concern for the near term future. Present production capacity can meet demand in the 1970's, but it will double (to about 100,000 tonnes uranium per year) between 1980 and 1985. In view of overall lead times of 6-10 years for proving reserves, raising capital, designing and constructing mines and mills, and often delays to satisfy environmental criteria, there is not much time for the industry to expand in time to meet this demand. The position is particularly critical in the United States, which is considering the removal of import restrictions for this reason.

Australia is in a fortunate position in

possessing uranium resources far beyond her likely needs for the next 2-3 decades. In the terms used above, these amount to about 240,000 tonnes of uranium, of which about 172,000 tonnes are firm reserves. There is thus considerable potential for export, and a key question is whether to export yellowcake, or proceed further into the fuel cycle, adding value by conversion to the uranium hexafluoride (UF_6) required for enrichment plant feed, and possibly by undertaking large-scale enrichment.

b) Uranium Enrichment

At present the main supplier of enrichment services is the United States, where the three existing gaseous diffusion plants are being improved and uprated to achieve a total capacity of about 27 million separative work units (SWU) per year. (In fuelling a typical PWR, about 100 SWU per year are required per MW(e)). This capacity plus smaller amounts available in Europe and from the USSR can satisfy projected demands until the early 1980's. Plans announced by URENCO (the U.K., German, Dutch tripartite company) for centrifuge capacity in Europe, and by EURODIF for a gaseous diffusion plant in France, will add sufficient capacity to meet demands until about 1984. After that time the annual rise in demand will require new capacity to come into production at a rate corresponding to the output of one large diffusion plant (10 million SWU/year) every 18 months.

Two large industrial consortia in the United States are expected to announce in July 1974 whether they intend to proceed from current feasibility studies to firm projects for commercial enrichment. Although both have announced that they are studying both diffusion and centrifuge technologies, Uranium Enrichment Associates (Westinghouse, Bechtel, Union Carbide) appear to favour gaseous diffusion, whereas the General Electric-Exxon study appears to be more inclined to centrifuge technology. Recent statements (Ref. 4) on US centrifuge technology indicate that it is considerably different from that developed by CENTEC-URENCO in Europe (Ref. 5). Although the URENCO partners have pilot plant capacity of about 50,000 SWU per year now in operation, commercial plants will not commence operation until 1976 at earliest, and this is also the time at which the USA expects to have a machine proven for commercial use.

Further prospects for new enrichment capacity are being studied in Canada, South Africa, Japan and Australia. Currently there appears to be no serious prospect of a shortage of enrichment services over the next ten years. After that if all potential enterprises now under study went ahead, there would be overcapacity (by a factor of about 2.5-3) by about 1985. Since enrichment plants are a highly capital-intensive industry, this is very unlikely to happen; each decision affects the others. As in the nuclear power and uranium mining industries, total lead times for enrichment plants are in the range 6-10 years (shorter for centrifuge technology), so major decisions on future capacity will have to be made soon. However the opportunity for Australia to enter the enrichment field is not going to disappear quickly; in fact as the demand for uranium resources (the feed material) increases over the next decade, there should be continuing opportunity for countries possessing such resources to break into the enrichment market.

The long-term viability of that market is particularly important to enrichment enterprises in view of the high capital cost of enrichment plants. Some protection is afforded by the fact that the economic lifetime of a thermal reactor nuclear power station will probably be at least 20 years; hence any fall-off in enrichment demand as a result of the adoption of new power generating technologies is unlikely, in less than a reasonable amortisation period for enrichment plants built in a situation of rising demand.

The most advanced contender among these new technologies is the Fast Breeder Reactor (FBR), which is at the prototype stage in three countries, and hence is at an advanced stage of development towards commercial adoption. The future roles of fusion (thermo-nuclear) power and solar energy are more difficult to predict because they are less advanced in terms of research and development towards commercial acceptance as energy resources.

c) Plutonium

The uranium fuel cycle involves recycling of two fissile materials, U235 and plutonium, into thermal reactors, and eventually the use of depleted uranium (with lower than the natural U235 concentration, 0.711%) as "blanket" material in which to breed plutonium using the escaping surplus neutrons from FBR cores.

Currently some partly depleted uranium from chemical reprocessing of spent thermal reactor fuel elements is being recycled, where the residual U235 concentration is still sufficiently above the natural level to make recycling economic. The actual figure depends on natural uranium price and availability. However to date plutonium is not being recycled anywhere (except in experimental quantities), and while plutonium recycle to thermal reactors appears as a significant factor in both US and European future fuel cycle planning - with effects of the order of 10% downwards in requirement for uranium enrichment services in the late 1980's - it is not at all clear when it will be introduced on a large scale, and how much will be recycled.

One major reason for this uncertainty is that the facilities required for the degree of recycling planned for the 1980's do not exist at present. In particular, light water reactor plutonium fuel fabrication plants required to implement US recycling plans have not been finally designed, because at present the criteria, regulations, and procedures that will be applied to transport, handling, and waste disposal have not been defined. Plutonium is a powerful alpha particle emitter and highly toxic, and must be handled at all times in isolation from workers and the environment; consequently plutonium fuel plants will be expensive. One estimate (Ref. 6) is that by comparison with a similarly sized uranium fuel plant, the plutonium plant will cost twice as much and produce one third of the output.

There is thus still uncertainty on the costs of recycling and as the total lead time for large-scale plants is foreseen at present as at least six, and possibly ten years, during which time costs of both uranium and enrichment are expected to rise, accurate economic prediction is difficult. Possible delays through regulation and licensing difficulties apply also to the use of plutonium in thermal reactors; to date only limited licences for experimental purposes have been issued, and there is considerable concern being expressed by intervenors and objectors to the large-scale transport and use of plutonium as a fuel because of both its toxicity and the possibilities of unauthorised diversion to make nuclear explosives.

The main interest for Australia in plutonium in the thermal reactor era is its effect on uranium and enrichment supply and demand. However in the long term the availability of plutonium as FBR fuel could be a critical factor in our fuel and energy economy. Examination and discussion of reactor and fuel cycle strategies overseas indicates that while uranium supply, and more recently enrichment services, are regarded as items of international commerce, planning for the recovery and use of plutonium in enriched fuel cycles is on a national basis.

Fast Breeder Reactors will require large initial plutonium inventories of the order of several tons. The only source of this material now is thermal reactor spent fuel, and this will apply until the breeding gain of advanced FBR's enables them to contribute plutonium for new reactors as discussed below.

There have been suggestions that since Australia will be relatively late in adopting nuclear power, we should by-pass the thermal reactors and await the FBR. By the time it is commercially acceptable and available there is a strong possibility that plutonium will be a premium fuel in short supply. Consequently this option may not be open to Australia, and this is a topic which merits continuing scrutiny by Australian power generating and energy planning authorities.

5 THE FAST BREEDER REACTOR (FBR)

The primary incentive for FBR development is to improve nuclear fuel utilisation. It was realised in the late 1940's that reactors operating with fast (unmoderated) neutrons had potential for a much higher conversion or breeding ratio - of fissile nuclei produced from fertile material, to fuel nuclei consumed - than moderated or thermalised reactor cores. The most favourable case involves the breeding of plutonium 239 from uranium 238.

This is achieved in the core and a surrounding "breeder blanket" containing fertile material (U238) in depleted or natural uranium. Fission of plutonium caused by fast neutrons in the core gives a higher neutron yield per fission than is possible with moderated neutrons. In order to work with fast neutrons a compact core with a high concentration of fissile material (typically, 20-25% plutonium) is required; the core has a high volumetric rating (power density) and high neutron leakage into the blanket. Because neutron capture in most elements decreases linearly as neutron energy rises, the FBR has low "parasitic" absorption of neutrons in constructional materials.

The development of FBR systems has presented many difficult technological problems, mainly in three areas. These are safety, heat transport, and control.

In most thermal reactor systems the nuclear chain reaction would be shutdown by a loss of coolant accident (which would also be loss of some moderator) but it would be important for an emergency core cooling system (ECCS) to act to prevent "shutdown heat" from fission product decay causing rupture of the fuel and release of radioactivity to the fractured primary loop. However even melting and agglomeration of fuel would be extremely unlikely to re-start a nuclear chain reaction because of the low enrichment in thermal systems. In the FBR the enrichment is much higher and so far it has not been possible to prove conclusively that an uncontrolled chain reaction would not occur in the event of fuel meltdown. In addition, the thermal rating of fuel in a FBR (megawatts per kilogram) will be much higher than in a thermal reactor, and therefore so would the shutdown heat and the prospects of meltdown if all coolant were lost. Consequently it is considered essential to design fast reactor core and blanket assemblies so that no conceivable combination of events can result in complete loss of coolant.

Engineering solutions to this problem are eased by the choice of liquid sodium as the preferred coolant for current FBR prototypes. No high pressures are involved in primary cooling circuits; e.g. while light water reactors operate in the range 900-2600 psi, sodium circuits in a FBR operate at about 60 psi. In U.K. and French designs the whole reactor structure and primary coolant circuit is contained in a multi-walled tank filled with sodium (about 1000 tons) so that while loss of pumped coolant flow may be possible, convective flow from a guaranteed source is available in emergency.

Sodium is chosen because the compact, highly rated FBR core requires a coolant of high specific heat, high thermal conductivity, and low viscosity (for low pumping power). Water or compressed carbon dioxide would be inadequate and would introduce excessive neutron moderation. Highly compressed helium is also suitable, and is the basis for the Gas-Cooled Fast Breeder Reactor (GCFBR) concept, which is under study in the USA and Europe, and which some suggest has better breeding prospects than the sodium-cooled (Liquid Metal, LMFBR) system. However the GCFBR poses additional safety problems because of the difficulty of providing guaranteed emergency cooling.

Sodium technology has been under intensive research and development over the past two decades to develop circuit materials and components (valves, pumps, etc.) together with the instruments and techniques necessary to apply this coolant safely and reliably to the FBR. Sodium burns in air and reacts violently with water, so containment and heat exchange to water and steam present problems; primary coolant sodium becomes neutron activated to sodium 22, a hard gamma ray emitter, with a half life of 2.6 years; sodium is opaque to light which poses maintenance problems. Consequently all equipment operating in sodium is required to be of high integrity, and particularly in radioactive sodium (reactor core and primary loop).

Most of the formidable control problems foreseen earlier in fast reactors have dwindled with increasing availability of basic nuclear data and understanding of the physics of fast neutron systems. The volume of sodium present in current designs has significant moderating effects, and this combined with dilution of plutonium (by U238) in the core results in a "softer" neutron spectrum than envisaged earlier. Hence the effectiveness of control absorbers has been increased, and this assists in control. However it is important to avoid the development of voidage in the sodium (e.g. by boiling under accident conditions) as this could give rise to rapid power excursions.

In addition the softer spectrum tends to reduce the neutron economy of the system. In the 1940's and 1950's, the breeding gain of fast reactors was optimistically predicted as high as 1.6-1.7. Consequently short "doubling times" for total plutonium inventory (of less than ten years) were forecast. However the current estimate of fuel doubling time for large reactors using oxide fuels is approximately 20 years. Oxides have been chosen for early FBR's to take advantage of the large amount of experience available from their use in water cooled reactors. It is expected that fuels with higher density and thermal conductivity based on uranium-plutonium carbides or nitrides will be developed over the next 10-20 years to give improved nuclear and thermal performance, including doubling times of less than ten years.

A fast reactor prototype is in operation in France (Phenix, 250 MW(e)); in the USSR the BN350 reactor (150 MW(e) plus process heat) has been operated but currently has heat exchanger problems; and in the United Kingdom the Prototype Fast Reactor (PFR, 250 MW(e)) is being commissioned. In the United States the construction of a "Demonstration" LMFBR has been authorised. Design studies are proceeding for full-scale commercial fast reactors in the range 1000-1300 MW(e) in France, Germany, and the United Kingdom; other European countries, and Japan and India have fast reactor development programs. In the U.K., Europe, and the USA the advent of genuine commercial FBR's is predicted by those working on FBR development as early as 1986. Most utility spokesmen are more conservative, and predict the mid 1990's for large-scale commercial ordering.

Bearing in mind the sophistication of this technology and the corresponding problems of "scaling up", it is difficult to foresee large-scale adoption before the 1990's. The performance of the current prototypes is still to be evaluated and time scales for the first large plants are unlikely to be settled until some operating experience has been obtained. Regulatory and licensing aspects have to be considered; although prototypes have been built, no regulatory and licensing authority has considered a specific proposal to build a large FBR for a generating utility, and this may be a source of some delay.

Early FBR's will have to compete with well-established thermal systems; their overall economics in the long term will depend considerably on their fuel costs (and hence on their breeding performance), and on fuel availability and cost for thermal systems. The objective is to achieve a total plutonium inventory doubling time less than the doubling time for FBR capacity. This appears achievable within the next 2-3 decades, i.e. at about the time that uranium supplies may become a limiting factor for thermal reactor capacity growth. This achievement would have three outstanding effects in terms of nuclear energy resources. Firstly, all stockpiles of depleted uranium now being accumulated as "tails" from enrichment plants, and as reprocessing plant output, would be usable in FBR blankets. Secondly, because the FBR would utilise about 70% of the energy inherent in uranium as mined, rather than 1-2% achieved by thermal reactors, FBR operators could afford to pay a much higher price for uranium; and finally, this low sensitivity to price would expand the economically recoverable uranium resources, very likely to the point at which extraction from granites and seawater became realistic - this would make uranium supplies virtually limitless.

Thus in the long term the FBR has the potential to supply man's needs for base-load power generation into the indefinite future, until superseded by some other technology. It is the only technology with this potential which is at the large-scale pilot plant stage. The other long-term contenders, fusion and solar energy, have not yet reached it, and their time scales are less definite. Hence the level of expenditure on the FBR overseas is considerably higher than it is on these other technologies.

6 RADIOACTIVE WASTES AND PUBLIC ACCEPTANCE

At present probably the greatest obstacle to general public and political acceptance of nuclear power is concern about potential ill-effects if a significant amount of the waste radioactivity it generates should escape and become an exposure hazard to man.

There is argument about what is significant and what is not. Very small releases of activity do result from normal operation of nuclear plants; "zero release" is technically feasible but would be costly. It has not proved possible to demonstrate quantitatively by direct experimental evidence whether or not there is a "threshold" below which an increase in annual radiation dose above the natural ("background") level is not significant in terms of health. However this background varies considerably from place to place, and despite considerable study, no consistent correlation has been obtained between health and genetic data, and natural dose rate. Release limits have been set therefore in terms of small fractions of the background dose rate; they are also low in relation to other "man-made" radiation effects, principally arising from medical X-rays and flying in jet aircraft.

While it cannot be argued that these limits have been proved to carry no risks at all, it is true to say that much careful work has not detected any ill-effects attributable to them. The question then arises whether the non-proven risks are acceptable in terms of being outweighed by the benefits of nuclear power. Numerically the risks are considerably less than in many other applications of technology - e.g riding in cars or aeroplanes - and because the benefits of nuclear power are increasing, it seems highly probable that its acceptance will become general.

This reasoning applies to nuclear plant operation and is being continually reinforced by demonstration, i.e. by successful safe operation of an increasing number of plants overseas. However it does not provide the answers sought by those concerned about possible dangers from the increasing amounts of waste radioactive materials.

The principal wastes are mixed fission product elements from which usable uranium and plutonium are separated during the reprocessing of nuclear fuel. In comparison with wastes from fossil fuel combustion, fission product wastes are small in volume and weight (about 3 grams per megawatt-day of electricity). If Australia's current generating capacity were all nuclear, we would produce less than a ton of mixed fission products per month; burning coal, the weight of carbon dioxide alone discharged would exceed eight million tons.

However these fission product wastes cannot be discharged. They contain some radionuclides with long half-lives; the most troublesome from the long-term hazard viewpoint are caesium 137 and strontium 90, with half lives of 30.2 and 28.9 years respectively. Effectively this means that the wastes must be kept apart from man and his environment permanently.

Unlike the operational release problem outlined above, the solution to this long-term isolation problem is not being demonstrated at present. In fact numerous "solutions" have been proposed but none have been put into effect on an industrial scale. In terms of allaying public fears and earning public acceptance, the nuclear industry and the relevant governments overseas appear to have made a tactical mistake in the 1960's through not demonstrating convincingly that acceptable solutions to this problem are available. The necessary technology exists at laboratory and pilot plant scales; but there has not been adequate economic incentive to put it into effect on an industrial scale, and the effects of the lack of demonstration on public and political attitudes have been sadly underestimated. In fact the interim practice adopted, of storage of liquid highly active wastes in shielded tanks has exacerbated the public relations problem because there have been some leakages followed by world-wide and sensational publicity.

Numerous laboratory projects in the USA and in Europe have demonstrated that it is technically feasible to convert these wastes into highly stable, unleachable solids, and to store them safely, at projected costs that would be of the order of 5-10% of today's overall fuel cycle costs for commercial reactors. The final solid forms studied are sintered ceramic oxides, glasses, and concrete and bitumen composites.

Until recently various proposals for permanent disposal of such "fixed" radioactive wastes under the earth's surface have been pursued as near-term solutions to the problem, including sinking them through the polar ice caps, and depositing them deep in the earth in man-made cavities, e.g. in salt mines, spent oil wells, and voids made by nuclear explosives. None of these have been received enthusiastically by those sectors of the public most concerned about the problem. Such proposals raise unanswerable questions about long-term dangers of release of part of the material to man's environment; nothing is "unleachable" for ever, and underground geological cataclysms do occur.

In terms of quantities of wastes, the United States has the largest problem, attending the world's largest nuclear power program. Current planning (Ref. 7) is on the basis of long-term storage at remote sites in surface "engineered storage" facilities in which fixed wastes in large double-containment capsules would be cooled and monitored under continuous surveillance. Capsule leakage would be highly unlikely and easily detectable at very low levels - one of the few advantages of waste radioactivity. In this event further encapsulation would be applied.

Firm plans for the first facility in the United States are expected to be announced later in 1974. This approach is considered to provide a safe and acceptable solution provided that the site is geologically stable, for the next 50-100 years; there is no apparent reason why it could not continue indefinitely. However better solutions may well arise in the long term, and a major advantage of the engineered storage concept is that all wastes are retrievable, accountable, and removable if some better final disposal method becomes available. Probably the best ultimate disposal site would be the sun; use of it would require highly reliable transport out of the earth's gravitational field, and it is reasonable to assume that this will be developed in the foreseeable future. Whether and when it is used for this purpose will require an interesting cost-benefit analysis exercise in which the performance of the facilities now being designed will be a major ingredient.

If man wishes to use nuclear power, he has to accept for the moment that fission product wastes will be stored on the earth. Demonstration that this can be achieved safely and reliably on a large scale should occur during the next decade. In weighing up the costs and benefits of nuclear power, some weight should be given on the positive side to the fact that uranium as a fuel does give the user the unique ability to collect all of the "combustion wastes" and store them, an ability unlikely to be conferred by fossil fuels. And in contemplating whether or not to use nuclear power in Australia, a positive factor is that our ability to provide remote, geologically stable sites for engineered storage is second to none.

7 CONCLUSION - IMPLICATIONS FOR AUSTRALIA

No predictions have been made in this paper on dates or rates for the installation of nuclear power as an energy resource in Australia. However the aspects of nuclear energy resources to which attention should be directed now and in the near future are emphasised below.

On nuclear power, the success of the technology overseas and the changing patterns of fossil fuel availability and prices indicate that this energy resource could become attractive for use in Australia in the near future. It is necessary therefore to keep under review and to assess the advantages and disadvantages of the various "established" nuclear power systems overseas, and those likely to enter commercial service in future. Since some of these benefits and penalties are site-related, State generating authorities have a role to play in such assessments.

It is vitally important that adequate and firm criteria should be applied to all nuclear installations to protect health and the environment. Compliance with such criteria at all stages is ensured by a regulation and licensing process, backed by competence to assess proposed designs and operating practices. Foreknowledge of these criteria, standards, and procedures is required for economic assessment, and this applies to uranium fuel cycle industries as well as to power stations. There is thus some urgency in Australia to define the conditions that would be applied to nuclear facilities.

In the nuclear fuel cycle, Australia is fortunate in possession of large uranium resources. The extent to which we should become involved in processing for export is of immediate importance; consequently it is necessary to assess the future market for uranium at various processing stages, and this involves study and analysis of both current nuclear power programs overseas, and the likely advent and future influence of advanced reactor types and other power generating technologies.

One aspect of the uranium fuel cycle which could be of particular long-term importance to Australia is the availability of plutonium for fast breeder reactors. Use of nuclear power efficiently will involve "reactor strategy", integrating the fuel cycle characteristics of different reactor systems to achieve maximum use of uranium, by closing the feedback loops in the uranium fuel cycle in the optimum way. Currently, it appears that planning to this effect for plutonium is proceeding overseas very much on a national basis. This situation and its implications for future planning in Australia deserve continuing attention, including at least a "watching brief" at this stage by State authorities.

Finally, the matter of public attitude and education requires attention. There have been notable examples overseas of complacency by power generation planners leading to severe public opposition when nuclear power was proposed (e.g. Switzerland). A major field of public uneasiness is the long-term fate of waste radioactivity, and fortunately positive steps are now being taken to demonstrate solutions to this problem. There would appear to be considerable merit in more positive public relations activities to bring such developments, and the continuing operational safety record of nuclear technology, to public attention.

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