The Future of Nuclear Fuel Reprocessing in Australia

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An attempt is made to predict the nature and scale of the nuclear fuel reprocessing industry which might be expected to develop in Australia up to about 1975, and to indicate the kind of technical advances which will be required to achieve this.

INTRODUCTION

There are many different stages involved in the production of nuclear fuel and its recovery after irradiation in a reactor. These include:

- (a) mining and beneficiation of ores of uranium and thorium;
- (b) extraction from ore concentrates;
- (c) purification of crude uranium and thorium oxides;
- (d) reduction to metal;
- (e) fuel element fabrication, including containment;
- (f) recovery of spent fuel after irradiation; and
- (g) refabrication of fuel elements.

It is with stage (f), the reprocessing of irradiated fuel, that this paper is primarily concerned. This includes the separation of fissile from fertile fuel, such as uranium 233 from thorium and, of course, decontamination from fission products.

A number of the chemical processes and chemical engineering operations are common to more than one stage in the fuel cycle, but the reprocessing of irradiated fuel involves special technical and economic problems. The technical problems are mainly associated with the intense radioactivity of spent fuel and the need for highly efficient separations and fuel recovery. The economic problems arise from the high costs of nuclear fuel, and the fact that fuel reprocessing changes may represent a substantial proportion of the overall cost of power generation.

The reprocessing of irradiated fuel gives rise to a variety of radioactive effluents, and their treatment, storage and disposal is an essential part of any reprocessing plant. The problems associated with effluent treatment and disposal are considered in a paper being presented to the Symposium by Temple, and only a limited discussion of the possible scale and cost of effluent storage is included here.

The principal aim of this paper is to provide a general picture of the nature and scale of the nuclear fuel processing industry which might be expected to develop in Australia up to about 1975, and to indicate the kind of technical advances which are required.

In order to assess the processing requirements it is necessary to make some assumptions regarding the future development of nuclear power in Australia. This is the subject of a Symposium paper by Baxter and Watson-Munro. These authors are primarily concerned with the longer term developments (up to the end of the present century), whereas in this paper an attempt is made to predict developments in the transition period leading to the establishment of a fully integrated and competitive nuclear power industry.

For convenience, the stages of possible development have been considered at three dates —1965, 1970 and 1975. The principal assumptions used in the subsequent discussion of processing requirements are collected together in Table 1.

TABLE 1:—ASSUMED DEVELOPMENT OF NUCLEAR POWER IN AUSTRALIA

Date	Reactor Type	Installed Capacity (MW-Heat)
1965	Natural uranium	400
1970	Natural uranium	2,000
1975	Natural uranium	8,000
1965	Small unit — 10-25 MW (E) Probably high temp. high neutron efficiency.	50
1970		200
1975		500
1965	Prototype power reactors	50
1970	" "	100
1975	" "	100
1965	Research reactors excluding prototypes	10
1970		30
1975		50

POSSIBLE PROCESSING REQUIREMENTS. PART I—PROCESSING IN RELATION TO POWER SYSTEMS.

Natural or slightly enriched uranium reactors

These are the only power reactors which have, at present, been developed to the stage where they afford early promise of competitive power at costs comparable with those of large coal-fired stations. When account is taken of the fact that Australia is a producer of uranium, but is unlikely to invest in U235 enrichment, it seems likely that the first large-scale use of nuclear power will involve natural or slightly enriched uranium reactors. These

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* Australian Atomic Energy Commission Research Establishment. Manuscript received, March 14, 1958.

TABLE 2:—NATURAL URANIUM INVENTORY AND PROCESSING REQUIREMENTS

Year	Fuel Investment total to given date	Annual Processing Requirements*
1965	200 tonnes	40 tonnes
1970	1,000	200
1975	4,000	800

* Based on 300 days full operation.

could, for example, be a development of the Calder Hall gas-cooled reactors.

It may be assumed that the reactors are fuelled by uranium metal which could be contained in magnesium alloy or beryllium metal cans. While the initial supplies of fuel will probably be imported, it is to be hoped that the purification of the raw materials and fabrication of uranium and beryllium would eventually be carried out in Australia.

Unless supplies of pure fissile material are readily and cheaply available from abroad in adequate quantities, it will be necessary for Australia to manufacture in natural uranium reactors the pure fissile material that will be required for the initial investment in the potentially attractive breeding reactors that should be developed by 1970.

On the basis of Table 1 the scale of supply and processing of fuel would be as set out in Table 2. It is assumed that the fuel irradiation limit is 3,000 MWD/tonne and the fuel investment is based on a heat rating of 2 MW per tonne of uranium.

Excluding any re-cycled plutonium, the plutonium content of the irradiated uranium would be about 0.2% maximum. (In addition to Pu239 this plutonium would contain a considerable proportion of the higher isotopes Pu240 and Pu241.)

Natural uranium reactors with re-cycled plutonium

Although plutonium is not a good thermal reactor fuel, it may be desired to re-cycle at least a proportion of the plutonium recovered from the uranium. This could be added to the uranium fuel, but in view of the additional fabrication problems, would more likely be used as a separate alloy with, for example, aluminium. This arrangement might be used either to reduce the size of a reactor or to facilitate the inclusion of thorium so that the reactor is used as a uranium 233 producer.

The amount of plutonium which might be re-cycled in this way is hard to predict. If all the plutonium generated was fed back into natural uranium reactors, it would take about 15 to 20 years to build up the plutonium content to double the original concentration of fissile material. After plutonium has been re-cycled two or three times the conversion factors drop substantially because of the high Pu242 content. It has been assumed that it is unlikely that significant amounts of separated plutonium would be re-cycled in 1965, and that by 1975 the amount of re-cycled plutonium would only total a few kilograms per day.

Production of uranium 233 in natural uranium reactors

It may be possible to adapt the Calder Hall type of reactor to the production of uranium 233. In order to increase the amount of thorium which could be inserted in a natural uranium fuelled reactor, it may be necessary to provide some additional enrichment with plutonium. Two alternative approaches are possible:

(a) to burn the U233 in the reactor and to convert gradually from uranium to a U233-thorium fuel; and

(b) to recover the U233 for use in some other reactor system having a higher conversion factor.

Alternative (a) is unlikely to be adopted in the early stages of a power reactor program, because of the long time required to effect such a conversion. It is, however, quite likely that thorium would be inserted in a natural uranium-fuelled reactor with a view to producing U233 for small power reactors, which would operate on an approximately self-sustaining U233-thorium fuel.

The amount of uranium 233 which could conveniently be produced from natural uranium reactors, without substantial extra enrichment, is probably limited to about 10% to 20% of the plutonium production. This would represent:—

in 1965 approx.	10 Kg/year
in 1970 approx.	50 Kg/year
in 1975 approx.	200 Kg/year.

Assuming an irradiation limit of 0.1% (U+Pa) 233, the quantities of thorium to be processed would be 10, 50, and 200 tonnes per year, respectively. A total thorium investment of less than 1,000 tons would probably be sufficient to achieve this output of uranium 233 by 1975. The thorium would probably be irradiated either in the form of metal or oxide.

Small thorium-uranium 233 systems

The current research programme at Lucas Heights is concerned with two possible reactor systems, both based on uranium 233-thorium fuel, and offering the prospects of good neutron economy and high fuel burn-up for relatively small power installations. One of these systems —the high temperature gas-cooled reactor— would use either a ceramic type of fuel (oxide or carbide), or uranium-thorium-beryllium intermetallic compounds in beryllium metal. The other system is based on the use of a suspension of uranium or a uranium compound in liquid sodium. These two reactor projects are the subject of Symposium papers by Dalton and Alder, respectively. It is possible that one of these reactors may be developed by 1965, and several will be in use by 1970. By 1975 more than one type of small nuclear power unit is likely to be in operation.

The processing requirements are difficult to predict at this stage. The quantities of fuel to be handled, calculated on the basis of 200% burn-up, would be as shown in Table 3.

Beryllium recovery will be essential on grounds of cost and availability if this is incorporated in the fuel.

TABLE 3.—ESTIMATED PROCESSING REQUIREMENTS FOR SMALL Th-U233 SYSTEMS

Date	Annual Processing Requirement*		
	Fissile Fuel U233	Fertile Fuel (Th)	Moderator — If incorporated In Fuel. (as Me)
1965	10Kg	1 tonne	1 tonne
1970	40Kg	4 tonnes	5 tonnes
1975	100Kg	10 tonnes	12 tonnes

*Assuming U: Th : Be = 1 : 100 : 3,000 atoms.

The processing requirements would involve the separation of uranium and thorium from fission products, and possibly from beryllium or sodium.

It has occasionally been suggested that it may be possible to develop a long-burning fuel for this type of reactor system, for which processing may be unnecessary, in the sense that the fuel could be discarded without greatly affecting power costs. It is interesting to consider the value of the fuel and moderator in terms of units of power produced by a long burning reactor.

If we assume that for an approximately self-sustaining reactor (initial conversion factor about unity) the atomic ratios U : Th : Be are 1 : 100 : 3,000; and the efficiency of conversion of heat to electric power is 30%. Then for a factional burn-up "b", the consumption of fuel (discarded) would be approximately one kilogram of fissile material per 300b MW days of electric power produced. If the U233 is valued conservatively at £10/gram, the cost per unit of power of the fissile fuel discarded would be 1/3b pence per unit.

This only covers the replacement of the uranium 233 in the initial charge. The value of the thorium in the initial charge is relatively small—about 10% of that of the uranium 233, or £1 per gram of U233 in the original fuel charge.

The value of the beryllium which might be associated with each gram of uranium 233 is about £6.5, taking the value of recovered beryllium oxide as £20 per Kg. This assumes that the fuel is incorporated in the moderator. Thus, when allowance is made for the recovered value of thorium and beryllium, the cost per unit of power of fuel discarded could be as high as 1/1.7b pence/unit.

The preceding discussion only takes account of the value of the spent fuel and moderator in terms of the electricity generated. In practice, the scarcity value of these materials would probably make recovery essential.

These savings must be offset against the processing costs involved in fuel and moderator recovery. The maximum permissible processing cost—beyond which it would be uneconomic to carry out processing—is, of course, determined by the fuel and moderator costs, but on the preceding assumptions, should not exceed about £17,500 per kilogram of fissile material contained in the fuel, or about £80,000 per tonne of total fuel and moderator (assumed as metal). This only covers recovery of beryllium as oxide. It is clear that processing costs would have to be very high to justify complete rejection of the spent fuel.

POSSIBLE PROCESSING REQUIREMENTS. PART II—PROCESSING IN RELATION TO RESEARCH PROJECTS.

Prototype power reactors

It is clear that the investigation of future power projects will involve processing effort at two stages. Firstly, it will be necessary to carry out laboratory and subsequent pilot plant studies to assess the feasibility of processing particular reactor fuels which may be under development. In fact, the ease of processing and recovering the fuel will be an important factor in determining the final selection of a particular type of fuel element.

Secondly, it will be necessary to make arrangements to process and recover the fuel from prototype reactor experiments. While this will probably involve a small scale of operation, substantial research effort may be involved. After 1965 it may be assumed that there will always be one or two prototype reactors in operation in Australia. The scale of processing, based upon a heat rating of 50 MW and 200% fuel burn-up, would amount to only about 10 Kg. per year of fissile fuel, which might be associated with several hundred kilograms of thorium.

Processing of research reactor fuel

HIFAR

At present the fuel for the HIFAR research reactor is being supplied by the U.K.A.E.A., and will be returned to the north of Scotland (Dounreay) for processing. This arrangement has several obvious disadvantages, and it may be necessary to make arrangements for local processing of HIFAR fuel. This would involve the separation of highly enriched uranium from aluminium, and fission products. The scale of

TABLE 4 — SUMMARY OF PROCESSING REQUIREMENTS

Type of Reactor		Estimated Annual Throughput—Tonnes/Year 1965 Fissile Fuel	1965 Total Fuel	1970 Fissile Fuel	1970 Total Fuel	1975 Fissile Fuel	1975 Total Fuel	Type of Separation Required (Removal of F.P.'s in all cases)
Power (Large)	Natural uranium		40		200		800	U-Pu from metal
	Converted Pu	.08		.4		1.6		
	Recycled Pu	—		.1		.5		Pu from Pu/Al
	Natural uranium and thorium conversion							
	Th		10		50		200	Th-U233 from metal or oxide
	U233	.01		.05		.2		Possibly Pu-Th-U233 separation
Power (Small)	Small Th-U233 Systems							
	Th		1.0		4		5	Th-U233 and possibly recovery from Be and perhaps Na—from metallic or ceramic systems
	U233	.01		.04		.1		
Research	Prototype reactors	.01	.5	.02	1	.02	1	Various and awkward U-Th-Be separations from a variety of fuels.
	Research reactors	.02		.03		.05		U from U/Al, etc.

processing in terms of fissile material is quite small—only about 15 Kg. per year, but the fission product content is substantial, amounting to about 10^5 curies/Kg. of uranium after 150 days cooling.

OTHER RESEARCH REACTORS

It is probable that other research reactors will be constructed in Australia before 1975. The most urgent requirement will probably be for low power reactors of the swimming pool type, although late in this period further high flux facilities will probably be required. The present trend in research reactor design is towards the use of uranium-aluminium fuel and ordinary or heavy water as moderator.

Taking account of these possible developments, it is possible that the processing requirement for this type of reactor will rise to about 50 Kg. of fissile material per year by 1975, including HIFAR requirements. It would obviously be desirable to standardise the design of research reactor fuel elements to a minimum number of types, and to provide some central processing facility.

SUMMARY OF PROCESSING REQUIREMENTS

The requirements which have been described earlier are summarised in Table 4. This table only gives an estimate of the quantities of fissile and fertile material involved and only brief notes on chemical separations are included. It should be noted that about 10^5 to 10^6 curies of fission product activity would be associated with each kilogram of fissile fuel to be processed.

PRESENT STATUS OF SEPARATION PROCESSES

Introduction

Processes for the separation of nuclear fuels may be considered in two groups. Firstly, the aqueous processes (solvent extraction, ion exchange, etc.), which are in general fairly well established, and secondly, the non-aqueous processes, including pyrometallurgical and fluorination methods.

Processing requirements may fall into one of two distinct categories. In the first, virtually complete decontamination of fuel is required. This means that the active isotope content must be reduced to a level such that direct handling of the product is possible. In the second category, we have processes which aim only to yield a partial decontamination of the fuel and separation of the principal neutron poisons. As reactors are developed with higher specific power ratings, and using separated fissile fuel, it becomes economically necessary to reduce fuel cycle times to a minimum. Even if adequate chemical separations can be achieved, the presence of short lived isotopes such as U237 and Th234 may necessitate remote fabrication of fuel. In these circumstances partial decontamination processes may be adequate.

Aqueous processes

FUEL DISSOLUTION

The first research and production reactors to be constructed used, for the most part, metallic fuel which was contained in aluminium or magnesium cans. These cans could generally be stripped by mechanical means or dissolved in nitric acid with the fuel. The search for higher working temperatures and better corrosion resistance has led to the development of less easily soluble cans which may be tightly bonded to the fuel material. The present trend is for the difficulties of fuel element dissolution to increase as power reactor development progresses. The present status of various dissolving techniques is summarised in Table 5. This and the subsequent table is based on published information, and may err on the conservative side. However, it is clear that considerable further study of dissolving techniques will be essential if the Australian nuclear power program is to develop along the lines suggested in the preceding sections.

FUEL—FISSION PRODUCT SEPARATIONS

This stage includes the separation of fissile and fertile fuels from one another, as well as

TABLE 5—DISSOLUTION OF NUCLEAR FUELS FOR AQUEOUS PROCESSING

Fuel Type		Method of Dissolution	Technical Status*
Metal and Separable	U U-Al	HNO_3	A
	Pu-Al	HNO_3	A
Can	Th	HNO_3 + HF	A
Can Dissolution	Al	HNO_3 or NaOH	A
	Mg	HNO_3	A
	Zr	HNO_3 + HF or Chlorination	A
	Stainless Steel	HNO_3 + HCl	A
	Be	HNO_3 — HCl or Chlorination	C
	C	HNO_3 (anodic disintegration)	C
Oxides	UO_2	HNO_3 Fluorination	A
	ThO_2	HNO_3 + HF	A
+ Moderator	+ BeO	High temp. Chlorination	C
Carbides	UC_2/UC	HNO_3	C
	ThC_2/ThC	HNO_3	C
+ Moderator	+ C	HNO_3 (anodic dissolution)	C

*Technical Status:

A—Process operated on technical or production plant scale;

C—Small-scale laboratory studies only.

their separation from fission products. Table 6 attempts to summarise the types of separation required, and estimates their present stage of technical development.

It should be noted that solvent extraction techniques have been more extensively developed than other methods of nuclear fuel separation. A number of basic solvent processes for the separation of uranium, plutonium, fission products and of thorium, uranium 233, and fission products are well established. These processes are highly efficient in recovery, and further refinements in the chemistry of the processes are unlikely to have any revolutionary effect on process economics. Important chemical requirements are:

(a) to adapt the processes to quick turn-round of fuel, with consequently much higher concentrations of short-lived fission products, and

(b) to make provision for separation from beryllium and its subsequent recovery.

In addition, further advances may be expected in the development of more specific solvents, which could, for example, reduce the number of extraction cycles.

Considerable improvements in solvent processing can be anticipated in the field of chemical engineering. For example, substantial economies should be possible in the design of shielded facilities by extending the provision for maintenance.

Considerable pilot plant experience has been gained in the U.K., Canada, and U.S.A., on the use of ion exchange resins for basic separation processes, particularly at the later purification stages. The major industrial use of ion exchange resins in the nuclear field has, however, been in the recovery of natural uranium from leach liquors, although a pilot plant has been operated in Canada for processing irradiated plutonium fuel. Two major factors limiting the use of ion

TABLE 6.—AQUEOUS SEPARATION PROCESSES

Separation Method	Technical Status of Various Separation Processes*		
	U-F.P.'s U-Pu-F.P. Pu-F.P.'s	Th-U-F.P.'s	Th-U-Be-F.P.'s U-Pu-Be-F.P.'s
Solvent Extraction			
Tributyl Phosphate	A	A ?	C
Dibutyl-carbitol	A	—	—
Other Solvents	A	B	C
Ion exchange	B	B	C
Precipitation	A	B	C

* Technical status:

A—Process operated on production plant; or technical scale;

B—Pilot plant studies;

C—Small-scale laboratory studies only.

TABLE 7.—SUMMARY OF NON-AQUEOUS PROCESSES

Type	Variant	Objective
Melting metal	Simple remelting (self-slagging) oxidative slagging Fused halide extraction	Metal refabrication) Depoisoning) Depoisoning and F.P. removal
Metal distillation	Separation of diluent, Pu distillation	Na, Bi removal, etc. U-Pu separation
Halide distillation	UF_6 distillation $Zr\,Cl_4$ distillation $PaCl_5$ distillation	U-Pu separation Fuel element dissolution Pa separation
Liquid metal extraction	Uranium/Ag. Mg., etc. Zone melting	U-Pu separation Depoisoning
Metal Reconstitution	Chlorination and reduction Electrolysis	) Depoisoning

exchange resins for the processing of irradiated fuels are:

(a) radiation decomposition of the resins (which makes them unsuitable for use with the more active plant solutions); and

(b) technical problems associated with the emptying and recharging of resin columns.

The ion exchange processes are relatively simple to operate and often afford a convenient means of reducing solution volumes. They may well play an important part in future processing in Australia.

Separation by precipitation processes has special advantages for aqueous homogeneous reactor fuel, and also in cases where complete decontamination from fission products is not required. An advantage of precipitation processes which has not been fully exploited is their adaptability to very highly active solutions. Although the first large-scale separation of plutonium from uranium was achieved by a precipitation process, it seems unlikely that this kind of process will ever compete with solvent extraction and ion exchange unless it is associated with "active" fabrication of fuel in remotely operated plant. The principal difficulties are associated with solid-liquid separations.

Non-aqueous processes

During the past few years considerable research effort has been devoted to the investigation of a variety of non-aqueous separation processes—these are listed in Table 7. The processes are mainly designed to meet a requirement for quick turn-round of fuel, with limited decontamination from fission products, but with removal of the bulk of the neutron poisons.

Most of these processes have not passed beyond the laboratory or pilot plant stage. The greatest effort has been concentrated on the fluorination processes, partly because of the experience gained with uranium hexafluoride in diffusion plants, and partly because fluorides are common intermediates in metal production. Some of the fluorination processes show promise of yielding fairly complete decontamination of fuel from fission products, but for the most part these non-aqueous processes are more likely to find application in cases where "active" remote fabrication of fuel is possible.

It does not seem likely that these processes will find a great deal of application in an Australian nuclear power program of the type discussed in the preceding sections. However, two possible exceptions should be noted. Firstly, it may be necessary to use some dry fluorination or chlorination method to dissolve ceramic fuels should these be used in high temperature thorium—U233 reactors. Secondly, if liquid metal reactors are developed to the prototype construction stage, it is likely that some pyrometallurgical method of fuel processing will be necessary.

FISSION PRODUCT EFFLUENTS

No discussion of nuclear fuel processing would be complete without some consideration of the problems of the treatment and disposal of radioactive effluent. The estimate of the effluent volumes and fission product activity arising from the processing program just discussed can only be very approximate. However, an attempt has been made in Table 8 to indicate the order of magnitude of fission product effluents which may be expected.

The bulk of the fission products will arise from the natural uranium reactor program, and it is assumed that a solvent process is used which permits concentration by evaporation of the main fission product stream. On this basis, fission products may be concentrated to about 100 litres per tonne of uranium processed. The fission product activity is calculated on the basis of 10^{-3} curies-watt-year of reactor operation for long lived fission products (mainly Cs137). For short cooling (about three months), the activity will be about twenty times greater.

It seems that the total volume of highly active effluents which is likely to arise in the period under consideration will not constitute a serious problem. It should be feasible to provide for safe storage in underground tanks, on a semi-permanent basis.

The more difficult problem will be to arrange for the treatment and disposal of the larger volumes of intermediate activity effluent containing tens or hundreds of curies of fission

TABLE 8:—ESTIMATED ARISING OF CONCENTRATED F.P. EFFLUENTS

Year	Volume of Annual Arisings of F.P's	Arisings of Long Lived F.P's (curies after 6 months cooling)	
		Annual	Cumulative to given date
1965	5 m^3	5×10^5 C	10^6 C
1970	25	2.5×10^6	5×10^6
1975	100	10^7	3×10^7

products. There are only two practicable methods for the ultimate disposal of large volumes of active waste:

(a) ground disposal in desert areas; and
(b) sea disposal.

The latter appears the more convenient and safe.

It is almost certain that some steps will have to be taken to treat intermediate effluents and to drop the fission product concentration to a level such that disposal is possible. Methods of effluent treatment are discussed in other Symposium papers.

CAPITAL INVESTMENT IN REPROCESSING PLANT

On the basis of Table 4 the author has made estimates of the capital investment which would be required to meet the costs of the various stages of plant construction. As these estimates are highly speculative and cannot at this stage be based on local experience, it is not considered worthwhile to reproduce them in detail. However, the main conclusions can be summarised as follows:

(i) During the next five years it will be necessary to devote a substantial effort to the construction of small "pilot" plants. These should cover—
  (a) U233 separation from irradiated Th metal or oxide;
  (b) processing of HIFAR fuel;
  (c) recovery of Th, Be, and U233 from H.T.G.C. fuel;
  (d) recovery of U and Na from L.M.F.R. fuel; and
  (e) U-Pu separation.

Some of these requirements might be met by multi-purpose plants. The total investment involved at this stage would probably be of the order of £1.5 million to £3 million.

(ii) After about five years it will be necessary to develop a central processing site. Initial site development would cost about £3 million to £5 million, including provision of control laboratories, offices, effluent facilities and normal site services.
(iii) Plant construction on the processing site will develop gradually. Initial processing costs will be relatively high because of the wide variety of different types of process required. Over the period up to about 1970 the total expenditure required for reprocessing plant would be of the order of £5 million, excluding fuel element fabrication.
(iv) After about 1970, the emphasis would be on economically competitive larger-scale plants. Initially the major capacity will be required for the separation of plutonium from natural uranium type fuels, but separation of uranium 233 from thorium will become increasingly important.
(v) The total expenditure on nuclear fuel reprocessing required to meet the requirements listed in Table 4 can only be guessed at this stage, but seems likely to be somewhere of the order of £15 million to £25 million. This exludes provision for fuel refabrication, which could involve another £5 million to £10 million.

While a total expenditure of £25 million on fuel reprocessing by 1975 seems by itself a high figure, in fact it only represents an expenditure of about £10 per installed kilowatt of electrical generating capacity. This expenditure, on present indications, will be less than 10% of the total capital cost of the reactors and associated power generating equipment.

CONCLUSIONS

It is realised that much of the data presented in this paper is highly speculative, and that the quantities, costs, and time-scale presented may be substantially in error. However, if we accept the two principles:

(i) that Australia's requirements for nuclear power will follow approximately the pattern shown in Table 1 and
(ii) that Australia will endeavour to become self-sufficient in the provision and processing of nuclear fuels;

then within the next 15 to 20 years it is clear that a substantial new chemical industry must be established in this country. This will require careful long term planning to ensure that an adequate reserve of trained staff and experienced industrial support is available.

ACKNOWLEDGEMENT

During the preparation of this paper, the author has had many helpful discussions with the Chief Scientist and members of the Chemistry and Chemical Engineering Sections of the A.A.E.C. Research Establishment. However, it should be made clear that the views expressed are those of the author, and do not necessarily represent the policy of the Commission.