IAEA-158

FACILITIES AND TECHNOLOGY NEEDED FOR NUCLEAR FUEL MANUFACTURE

PROCEEDINGS OF A STUDY GROUP MEETING ORGANIZED BY THE INTERNATIONAL ATOMIC ENERGY AGENCY GRENOBLE, FRANCE, 4–15 SEPTEMBER 1972



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The increasing role which nuclear power is playing in supplying the world's electrical energy requirements has led to a burgeoning demand for nuclear fuel. A recent study by the Nuclear Energy Agency of the Organization for Economic Co-operation and Development and the IAEA has estimated that the world's annual uranium requirements for nuclear fuel will increase from a minimum of 16,000 tonnes in 1973 to 100,000 tonnes by 1935. An important consideration in the national economy of countries with nuclear power programmes is whether their indigenous industries should manufacture any part of the fuel needed for power reactors. The objective of the Study Group was to present pertinent information on the technical and economic aspects and requirements of this part of the fuel cycle. The lectures given at this meeting which was held at the Centre d'Etudes Nucleaires, Grenoble, France in September 1972 are published in this volume. The meeting was primarily intended for senior technical personnel and for national energy policy planners, particularly from countries which have nuclear power programmes under active development or with major uranium resources.

In 1971 the Agency convened a group of consultants to assist in the drawing up of an agenda and to suggest countries who might usefully be invited to the Study Group. At that time it was agreed that the agenda should be restricted only to oxide fuels. It was also suggested that lectures from countries which are entering the field of fuel technology should be sought, as these would serve usefully as case histories.

The Agenda was sub-divided into four sections according to the type of activity described and was as follows:

- I. The Production of Power Grade Fuel.
- II. The Fabrication of Canning and other structural Materials.
- III. The Manufacture of Fuel Elements and their Assembly.
- IV. Economic Criteria for Fuel Fabrication.

For Group IV, certain countries who had initiated fuel programmes or were actively planning to do so had kindly agreed to contribute summary review lectures on their programmes. These countries were India, Korea and Pakistan.

In addition to the lectures, visits to nuclear fuel fabrication facilities and to the laboratories and other research departments of the CEN, Grenoble had been arranged.

During the discussions following the lectures and at a round table seminar it was the consensus of the meeting that countries planning to undertake nuclear fuel fabrication technology could do so by the following means:-

- 1. Joint Ventures.
- 2. Licencing under patents held by established fuel manufacturers.
- 3. Development of an indigenous fuel fabrication technology, probably with assistance in the form of financial loans combined with technical advice from developed countries on a bilateral basis. International organizations such as the IAEA could render assistance at the pre-industrial development stage by arranging for the advice of experts, the placement of fellowships and the supply of equipment on the principles of fabrication processes. In industrially sensitive areas where proprietary information may be needed, the placement of experts and fellowships would in general, be part of bilateral and licencing agreements.

The licencing agreements which have been concluded to date appear to follow no general guidelines, but have varied according to the needs and technological capability of individual licencees. There is a trend for such licences to fall into two categories, one for countries with a significant infra-structure of relevant industrial technology and the other for countries having little or none. In the former case the licencee may adopt a combination of joint venture together with licence(s) to use specific processes or equipment. For the developing countries licences might cover a wider spectrum of the relevant technological processes with eventual plans for such countries to develop their own fuel technology, over relatively long period (5-10 years). Licensing agreements might well include provisions for a training programme for indigenous staff, as well is for the loan of experts. They may cover the whole or part of the fuel fabrication process. There was support for the view that countries entering the field of nuclear fuel manufacture should do so in stages rather than to attempt to undertake the entire fabrication at once. Decision criteria on which phase of the fabrication process to allocate priority would include:-

(a) Availability of skilled manpower;

- (b) Extent and nature of industrial infra-structure existing;
- (c) Financial resources;
- (d) Size and type of nuclear programme under development or planned;
- (e) Export possibilities.

Several speakers from the developed as well as the developing countries emphasized the broad spectrum of training required in fuel fabrication technology. Apart from relevant skills in the nuclear field, trained personnel would be required in such fields as vacuum technology, welding, electronics and chemical and metallurgical analysis including metallography.

It was emphasized that close liaison between the reactor manufacturer, the designer and the fabricator was necessary at all times, not only because the fuel produced must meet the specifications set by the reactor manufacturer but also because of requirements for a "built-in" flexibility to allow for improvements in technology and design and for possible expansion in plant capacity.

Estimates for the cost of producing nuclear fuels should allow for such diverse factors as:-

- (a) Raw material costs including where applicable, mining and processing costs.
- (b) Design costs.
- (c) Development costs.
- (d) Licensing fees where pertinent, including regulatory fees.
- (e) Inventories.
- (f) Insurance.

The feasibility of regional cooperative projects, where associated countries could undertake complementary tasks by carrying out different fabrication steps was discussed. It was pointed out that the requirements for nuclear fuel in the developing countries was relatively small at present and that such regional association of countries in a common effort could make indigenous fuel fabrication more economically viable.

After some discussion on the role of the lASA in the field of nuclear fuel fabrication there was no decision on the degree of priority the Agency should allocate to activities, as the type of assistance requested by each country may vary considerably from country to country. There was some support for the view that current efforts to standarize methods of quality control including non-destructive testing and chemical analyses should be encouraged and where necessary, should be expanded. It is in such fields that the Agency could play a useful supporting role.

It is hoped that the lectures will be of interest to those concerned with the launching of nuclear fuel technology and fabrication programmes, as well as to international organizations giving technical assistance in this field.

The Agency wishes to express its appreciation to the CEA for hosting the meeting, to the authors of the papers, to all who participated in the discussions and to Messrs. Rogan, Flipot, Schaus and Jonkheere for guiding the individual sessions.

The meeting was closed by Mr.Gerbier, Deputy Director CEN Grenoble, who extended an invitation to delegates to participate in a follow-up meeting at a CEN Centre two or three years hence in order to issess the progress made by countries in the development of an indigenous nuclear fuel technology.

The countries participating and the number of delegates at the meeting are as shown on the attached list.

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ORE TO CONCENTRATES

A Discussion of the Processing of Uranium Bearing Ores

by

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Abstract

The paper provides a broad background discussion of the principal activities in the production of uranium concentrates. Estimates of market requirements and uranium delivery commitments through 1985 are given. Generalized costs for exploration, development and mining of uranium ore in the United States are presented.

The principal methods of ore processing for the recovery of uranium are described and their principal advantages and disadvantages are mentioned. Data on mill capital costs and operating costs are presented for the four principal processes used, and for the range of mill operating rates found in the United States. Labor productivity is also given for various milling rates.

The conclusions are that all of the processes in use today are capable of producing uranium concentrates at reasonable and competitive costs. No one process has a clear economic advantage. As would be expected, larger mills show some advantages in terms of lower capital and operating costs. These advantages are not conclusive, however, and mills of modest size have been built and operated at competitive costs.

I. Introduction

The market for uranium has been below the productive capability of the uranium producing industry for about 10 years. The commercial market got underway in 1969 when a change in the Atomic Energy Act permitted for the first time the private ownership of enriched uranium for use in nuclear electric plants. Since that time the U.S. requirements for uranium have grown steadily, and will be at a level of about 9,200 tons in 1972. The annual requirements are projected to reach 18,000 tons by 1975, and 37,000 tons by 1980. The annual requirements and existing sales commitments to meet those requirements are shown in Fig. 1. Table 1 projects annual requirements and delivery commitments through 1985. Peak production in the U.S. was about 17,000 tons in 1961. The projected uranium requirements of the other non-Communist countries as a group is about the same magnitude as for the U.S. as is indicated in Fig. 2. As these illustrations show, a very large increase in the world production capability for uranium must begin in the next few years. It is not surprising, therefore, that interest is growing again in the technology and economics of uranium production.

II. Production Cost

The broad components of the cost of uranium production include exploration in search of ore deposits, mining, transportation and milling the ore. Let us look briefly at the activities contributing to the cost of the ore delivered to the mill.

The exploration phase of uranium production may begin with regional reconnaissance over broad areas covering literally tens of thousands of square miles. This is followed by geologic and geophysical studies of more limited favorable areas. Finally, areas are selected for intensive investigation and land is acquired. The cost of this preliminary phase of the search is very difficult to assess, but it is considered that a company to be competitive in uranium exploration today should consider expending a minimum of \$1,000,000 annually over a period of possibly 8-12 years. The costs of surface drilling have varied over a wide range depending on type of ground, depth, and type of drilling. Thus, non-core drilling in soft sandstones at depths up to 500 feet may be as low as \$0.25-0.35 per foot. Core drilling at 2,000 feet has cost \$15-20 per foot. During 1971, the domestic industry reported costs of \$17,000,000 for 11,800,000 feet of exploration drilling, an average of \$1.44 per foot. Development drilling last year averaged \$1.29 per foot on about 3,100,000 feet.

If we assume for the purpose of illustration, that a successful exploration program were to result in the discovery of an ore body of 3,600,000 tons, which would support a 1,000 ton per day operation for 10 years, the cost of ore delivered to the mill in a typical operation might be about as shown in Table II.

The cost data shown in Table II should not be taken as necessarily very precise, nor are they derived from any single operation. They are reasonable approximations for open pit and underground operations in the U.S. of this size. However, as you will appreciate the cost and grade of individual operations vary widely from these data. Nevertheless, a cost of \$3.50 per pound of U_3O_8 in ore delivered to the mill is not uncommon.

Now let us examine the effect of the range of milling costs which have been observed in the industry on the total cost of production (see Table II continued). The range of capital costs on a 1,000 ton/day plant has been about 60%, while mill operating costs vary by a factor of about $2\frac{1}{2}$. In the example given in Table II, the range of mill capital and operating costs, about \$1.45 per pound of U₃O₈ can easily make the difference between a profitable and unprofitable operation. Therefore, considerable care and effort should go into mill planning and design to avoid the problems that can quickly run up costs. Before discussing further the factors contributing to milling costs, I would like to provide some basic information on the processing methods, and considerations that are involved in choice of process.

III. Milling Methods

The uranium ores processed in the United States are largely soft friable sandstones containing 0.1 to 0.5% of U₃08. The average grade is about 0.21%. The uranium occurs in various minerals, mostly in its chemically reduced form as coatings on sand grains, and mixed with clays and other cementing minerals associated with the sandstone. The sand grains, which make up about 2/3 of the ore, are essentially barren of values. Some of the ores, located in the Uravan Mineral Belt, contain important byproduct vanadium. Copper and molybdenum have also been recovered in minor amounts in a few plants.

A. Physical Methods of Uranium Concentration

These methods, which depend on such characteristics as mineral density, magnetic properties, electrostatic properties, radioactivity, and the surface wetting phenomena used in flotation have seen little use in the U.S., but have found limited use for a variety of ores encountered in other countries. As examples, a sink-float separation is used on Swedish shales to separate barren limestone from the uranium bearing shale. Jigs were successfully employed at Shinkolobwe in the Congo to separate out massive pitchblende after crushing of the ore. Radioactive sorting of ore passing along a conveyor belt was used to a limited extent in Canada, and has been further developed and used commercially here in France. It is not a useful method in the U.S. where the soft sandstones disintegrate too much in coarse crushing operations to make sorting worthwhile.

Considerable research work has been done to develop a selective flotatior method for uranium. To date the process is unattractive unless the uranium is associated with some other mineral which can be floated readily. This is the case for South African gold ores in which uranium is concentrated in pyrite. Several old slimes dams were reclaimed and processed by flotation to recover pyrite required for manufacture of sulfuric acid. The pyrite concentrate, constituting about 5% of the weight of feed, contained about 40% of the uranium. In most cases, however, when ore is ground finely enough for flotation the soft uranium minerals are so finely divided they are lost in the fine slimes. The separation achieved by flotation is then too incomplete to be useful.

In a few instances field concentrators have been set up at mines in the U.S. remote from milling facilities, where the ore bodies are not large enough to justify building a mill. The process has usually consisted of a crushing and grinding operation, followed by a sand-slime separation and attrition scrubbing of the sands, sometimes with a little acid added. The object is to remove the sands and reduce by a factor of 3 or 4 the amount of ore to be hauled to the nearest mill. It is difficult with most physical concentration methods to obtain a sufficiently high recovery of uranium so that it no longer pays to reprocess the reject material, and also to produce sufficient savings in reduced haulage costs to justify the cost of the concentrating operation. There are no longer any upgrading plants operating in the U.S.

B. Hydrometallurgical Methods of Ore Processing

1. Ore Freparation

In most cases, a mill is supplied with ore from a number of mines, the ore being brought to the mill by truck and railroad. After weighing, the ore is dumped on a pad or in bins at the If the ore is to be purchased from an independent mine, mill. it is crushed and sampled in lots of appropriate size and kept segregated until accepted for purchase. If the mill is not purchasing ore, the coarse ore sampling may be eliminated, and mill feed sampling may be done in connection with grinding In most mills, the ore on the receiving pad is operations. transferred to the primary crusher by a front-end-loader. The crushed ore is conveyed to one or more fine ore bins for storage. Crushing is commonly performed on one shift only, while the remainder of the plant operates continuously. High moisture content of sandstone ores frequently causes ore handling problems. The ores may be sticky and difficult to handle in the crushing and sampling operations. In severe winter weather it may freeze in unprotected ore bins. A number of mills have added equipment for drying ore, usually a part of the total ore feed to permit control of moisture content and improve handling characteristics.

The ore receiving, sampling, crushing and storage operations represent a large part of the mill installation and also of operating costs. It is in this portion of the operation that some of the most serious operating difficulties have arisen, and an area in which good design has the greatest potential for reducing overall production costs. One of the very recently built mills in the U.S., the Jtah International Inc. mill in the Shirley Basin of Wyoming, has taken a new and interesting approach to ore handling. The system has no primary crusher, conveyor belts or fine ore bins. A large autogenous mill is fed directly from the ore stockpiles by front end loader. The ground ore is pumped to a large holding pachuca which provides surge storage capacity between the ore preparation section and the leaching section. From all reports so far, the system appears to be working well.

2. The Basic Flowsheets

Uranium can be dissolved from its ores by treatment with either sodium carbonate solutions or with mineral acids. If the uranium is present in the ore in its chemically reduced form (+4 valence state), as is usually the case, an oxidizing reagent must be added. Heat is beneficial in increasing the rate of uranim dissolution in either system, and is essential to achieve adequate recovery of values in a practicable length of time in the carbonate leach process. The four basic flowsheets currently in use in the U.S. are shown in Fig. 3. Two of these processes begin with the sodium carbonate leach of the uranium from the ore. In one method the barren solids are next separated from the ore residue by 3-stage filtration and the uranium subsequently recovered by precipitation from the solution with sodium hydroxide. In the other method, the +325 mesh sands are removed by means of classifiers and cyclones. The uranium is then absorbed from the slime pulp onto an ion exchange resin and subsequently recovered from the resin.

The other two processes begin with an acid leach of the ore. Thereafter, a separation of sand and slime may be made, with rejection of the sand tailings. Uranium is then recovered from the slime fraction by the well known resin-in-pulp procedures. Alternatively, the pregnant solution resulting from the leach process may be separated from the barren ore residue by filtration or thickening and subsequently clarified to produce a clear solution. Uranium may be recovered from this solution and purified by either ion exchange or solvent extraction.

3. Carbonate Leaching

The carbonate leaching process has advantages in treating ores containing large amounts of acid consuming constituents such as limestone. It is also very selective for uranium and does not dissolve nearly the quantities of unwanted impurities from the ore that acid leaching does. However, to achieve good uranium recoveries and minimize leach time the ore is very finely ground, commonly 80% or more - 200 mesh size in a solution containing the leach reagents - sodium carbonate and bicarbonate. With pulp temperatures elevated to the boiling point in pachucas, or above the atmospheric boiling point by using autoclaves, the leaching time is usually in the range of 18-24 hours. Air or oxygen are continuously supplied, and sometimes chemical oxidants such as copper and ammonia are used as well. The pulp is discharged from the leach circuit through heat exchangers, transferring heat to the incoming leach feed, and then filtered on rotary drum filters. Usual 3 stages of filters are used. High concentrations of leach Usually reagents are used in this process. Only a fraction, about a third, of the reagents are consumed. Therefore, dilution in the filtering step must be kept low to permit recycling of the unconsumed reagents. Flocculating agents are necessary to get adequate filter performance.

Sodium hydroxide is added to the clarified pregnant solution to precipitate the uranium as an impure sodium diuranate. This step is not necessarily quantitative, and special measures are taken to improve recovery of uranium, including recycling of previously precipitated sodium diuranate.

The final product is usually high enough in uranium content to meet specifications, but may need to be redissolved and reprecipitated to remove sodium, or roasted and water leached to eliminate vanadium. The barren solution from the precipitation step is recarbonated with boiler flue gas and returned to the process.

4. Acid Leaching

For acid leaching, the ore is crushed, and ground in water only sufficiently to separate the sand grains. The pulp is leached at about 55% solids for 8-12 hours with dilute sulfuric acid and usually an oxidizing reagent. Manganese dioxide and sodium chlorate are most often used for this purpose. To withstand the corrosive acid leach solutions either wooden or rubber lined mild steel tanks are used. Leaching is done in a series of tanks, wich reagent addition made at several points in order to permit continuous control of reagent concentrations. Some plants provide for 2-stage leaching, the first stage being performed without oxidizing reagents. The savings in acid oxidizing and other reagents achieved by this method is largely offset by the added cost of two liquid-solid separation steps, and the added difficulties in maintaining solution balances. Consequently, the 2-stage system is seldom used.

5. Separation of Liquids from Solids

Filters are favored over other liquid-solid separation systems when:

- a) Dilution must be kept to a minimum
- b) Reagent recovery and recycle is important
- c) Leach pulps have low slime content and good filtering characteristics.

The principal disadvantages of filters are high labor requirements for operation and maintenance, and close control required to maintain solution balances in the system and obtain good washing of the filter cake. Carbonate leach pulps are handled by filtration to minimize dilution and keep reagent concentrations high enough to allow the solutions to be recycled after uranium precipitation. In addition, carbonate leach pulps generally have poor settling rates in thickener systems.

Thickener systems are usually favored when:

- a) Dilution on the order of 1 to 2.5 tons of solution per ton of ore can be tolerated
- b) Leach pulps have poor filtering rates, but will settle at acceptable rates, with suitable flocculating reagents.

In U.S. installations a variety of combinations of thickener systems with cyclones and classifiers are to be found. In these systems the slimes are handled in thickeners and the sands in either classifiers or cyclones. It is not clearly established that the separate handling of sands is advantageous, in view of the additional equipment and handling required. In several more recently constructed plants employing thickeners, no sand slime separation is made. The coarse sands have not been difficult to handle in the thickeners, and help to compress the slimes, resulting in better overall washing efficiency. Sand-slime separation followed by resin-in-pulp recovery of the uranium is necessary when filtration and settling characteristics of the ore pulps are poor. The procedure does not require the costly flocculating reagents usually employed in plants using filtration and thickeners. This type of plant handles a variety of ore types reasonably well. However, the dilution ratios in sand slime separation circuits are higher than other systems, about 2-4 tons solution per ton of solids.

6. Extraction of Uranium from Leach Solutions

a) Ion exchange

The ion exchange resins used are styrene polymer bead shaped particles, in a size range dependent on the intended use from 10 to 60 mesh. Coarser sizes are used in resin-in-pulp systems to make screening easier. In recovering uranium from clear solutions the finer sizes are satisfactory. The resins are anion exchange materials. That is they attract and hold negatively charged ions from solution. The resins are highly porous sponge like structures. As most metals form positively charged ions in solution, they are not held by the resin. Therefore, the resins offer a means of separation of positively from negatively charged ions. Further, the resins offer a considerable degree of selectivity among anions, and hold most strongly ions with higher charge. Uranium forms anion complexes with both sulfate and carbonate ions, and resins have been made which are reasonably selective toward these complexes. Thus, in practice a nighly concentrated and purified uranium bearing solution can be produced from a very impure acid leach solution in a single step. The resins are extremely tough and durable, and under difficult operating conditions have been known to maintain good performance over a period of several years.

In treating clear pregnant solutions resulting from filtration or thickening operations the ion exchange resins are contained in sets of cylindrical pressure vessels, 3 or 4 to a set. These tanks are 7 feet or more in diameter and with a height about twice the diameter. Either a sand bed or screen plate at the bottom of the column provides support for a bed of resin 5 feet or more in depth. The sets are operated continuously and fully automatically. Pregnant solution is passed through two or three columns in series for absorption of the uranium on the resin. Meanwhile one column is stripped by means of an acidified eluting solution containing a common anion such as nitrate, chloride or sulfate. Each reagent has its advantages and disadvantages. Recently, producers have tended to use a solution of a sulfate salt and dilute sulfuric acid (about 10%) as an eluting reagent. It leaves the resin in the sulfate form which is advantageous, and is cheaper than other reagents. The concentrated sulfate solution of uranium is then treated by solvent extraction. The uranium product recovered is very high grade, generally 95 to 98% U308 and meets all product specifications without difficulty. This system is known as the Eluex process.

In resin-in-pulp operations the principles of operation remain essentially the same as in column ion exchange. However, the resin is either confined in screen baskets, the system used in the first RIP plants built, or passed continuously through a series of 6-8 tanks containing desanded leach pulp, as in the more recently constructed RIP systems. The resin and pulp are airlifted over screens between stages and flow countercurrently through the absorption circuit. The resins survive amazingly well in contact with the slime pulps, as long as agitation is not severe. Solids content in these systems is The resins show little if any detectable wear about 5-1(%. from agitation in pachuca typ. tanks with air, or when the pulp is suspended by wide sweep agitators. Airlifts also do not cause appreciable wear of resin beads, nor do vibrating screens. However, the resins do show appreciable higher rates of loss when subjected to high speed agitators, or pump impellers regardless of the design of the pump. In optimum conditions resin losses in an RIP circuit from wear may be on the order of 20-30% of the operating inventory per year, very comparable with resin life in column ion exchange units.

The uranium is stripped or eluted from the resin in a series of smaller tanks, usually 10-14 in number. Screens have been replaced between stages by small settling cones. When the resin accumulates in the bottom of the cones to a pre-determined depth, an automatic valve discharges it to the next tank. The Eluex process is generally used.

b) Solvent extraction

Solvent extraction as practiced today in uranium ore milling is limited to the treatment of clarified acid-leach solutions, and as previously mentioned, acid eluates from ion exchange resins. No operating plant uses solvents to recover uranium from slime pulps. A number of devices have been tested for this purpose, but none has been used commercially. In view of some inherent obstacles, such as the strong absorption of some of the solvents on certain minerals, it is doubtful the solvents will ever be successful in slime systems. Also, solvents are not used () carbonate leach solutions, as they have appreciable solubility in dilute carbonate solutions.

For clear acid leach liquors, however, solvent extraction has some attractive characteristics. There are many solvents available with a wide variety of properties. The mixer-settler equipment used is simple and runs in continuous countercurrent flow of organic and aqueous liquids with controls only on the rate of solution feed to the system. The units are less expensive to construct than ion exchange units.

The solvents now used are either acidic or basic. The acidic solvents are alkyl phosphoric acids and act as cation exchangers. The basic solvents are alkyl amines and are anion exchangers. Chemically they perform in a manner analagous to ion exchange resins. There is a third solvent type, neutral solvents, that finds only incidental use in uranium ore processing. The active solvents are too viscous to be used directly, and consequently they are diluted in kerosene or other light high boiling point petroleum distillate before use. This does introduce a fire nazard, a factor which is often not given sufficient consideration. There have been two serious fires in solvent extraction units in the U.S. In one case, an electrical short in wiring mounted above the unit dropped hot metal sparks into the solvent. The resultant fire caused over a million dollars damage. In the other case, welding sparks ignited solvent flowing in an open launder. However, with adequate precautions and constant safety education these systems can be safely run.

Some solvent loss from the circuit is unavoidable. The principal source of loss is likely to be entrainment of fine solvent droplets in the barren solution leaving the system. In practice solvent losses have been kept in the range of 0.1 to 0.5 gallon per 1,000 gallons of solution treated.

Uranium bearing pregnant solution and the solvent are mixed together briefly in a small mixing compartment by means of a fairly high speed agitator. The centrifugal force of the mixing action raises the liquid level in the mixing compartment and the unit overflows into a long settling tank in which the water and oil phases separate again. The organic phase overflows from the settler into the next mixing unit, and the aqueous phase passes through a port at the bottom of the settler into an adjoining mixer. The agitators in the mixing units supply sufficient head to move the solutions through the system without pumps.

Uranium can be stripped from solvents with a number of reagents. A concentrated soda ash solution is often used for acidic solvents. For stripping uranium from amines the most used reagent combination is a fairly concentrated (1 molar) solution of ammonium sulfate acidified to a pH of 4.0 to 4.5. The uranium is precipitated from such a solution by addition of ammonia.

The precipitated uranium is thickened, further dewatered by filtering or centrifuging, then dried or roasted under closely controlled temperatures. The product is dense and a standard 55 gallon drum usually contains 700-800 pounds of concentrate.

The product is still an impure material, which generally contains 80% or more uranium calculated as U308. To be acceptable feed to a refinery converting concentrates of UF₆ the product must meet certain impurity specifications. The specifications used by the two commercial UF₆ plants operating in the U.S. are given in Appendix A.

IV. Milling Costs

A. Mill Construction

Most of the mills operating in the United States today were built in the late 1950's and early 1960's. In analyzing the available data, I have tried to eliminate situations where unique or non-typical circumstances have had a large effect on cost. However, there are variations in most instances that affect certain cost factors. The selection of a mill site may be influenced by availability of water, rail transportation, housing and other peripheral factors, as well as proximity to the mines. In a few cases, old mills originally built for other purposes have been remodeled into uranium mills.

Most of the plants have been expanded, or proved to have substantial additional capacity over their design capacity. Therefore, in examining unit costs, the demonstrated capacity, or highest sustained rate of operation was used rather than "rated" or "design capacity."

Figure 4 is a plot of capital cost versus plant size for a group of 17 mills. Costs are expressed as \$/ton of ore processed per day. The range of costs is about \$4,000 to 8,000 per ton per day. A curve or set of curves did not seem to fit the data well. To better show the significance of the range of costs, two areas of cost are shown. The inner area included the capital cost of 2/3 of the mills. The outer area includes them all. It is interesting to note that while there is a broader range of costs among the smaller mills, nevertheless some of them have capital costs that compare favorably with the larger installations.

The bars at the right of the chart indicate the range of costs noted for the four processes. They have a considerable overlap, and one can conclude that in terms of capital costs the four types of plants are quite comparable. On the average, assuming a 10 year operation the capital cost of the plant represents about 20% of the total cost of milling.

B. Operating Cost

A similar approach to that used for capital cost has been used in the analysis of mill operating costs. In Fig. 5, the operating costs in dollars per ton of ore processed are plotted against milling rate in tons per day. Here again the inner area includes the costs of 2/3 of the mills, and the outer includes the entire group analyzed, some 18 mills, and not entirely the same group as the one used in analysis of the capital costs. The range of costs estimated for the four principal processes is indicated by the bars on the right side of the chart. The overall range in processing cost is from \$3 to 11 per ton of cre. However, 2/3 of the operations are within the range of \$4 to 7.50 per ton. The range of costs is dictated by many factors including reagent consumption of the ore, handling characteristics, whether or not ores are purchased and must be sampled, etc. There is no clear cut difference in cost among processes used, except that one may note that the solvent extraction costs are rather consistently at the low end of the range. Costs are to some extent influenced by ore grade as it is economically attractive to increase effort in order to increase recovery from high grade ores.

In Fig. 6, the operating costs in \$ per pound of U308 are plotted against mill production rate in tons of U308 per year. As in the previous illustrations, the inner area represents the costs of 2/3 of the mills and the outer area includes the remaining 1/3 of the group examined. The overall range of costs is from about 90¢ to \$2.10 per pound of U308, but 2/3 of the values fall in the range of \$1.00 to 1.90 per pound. The bars on the right indicate a broad overlap in costs by the four principal processes. These ranges are somewhat flexible. They are necessarily based on an assumed ore grade for each plant. Ore grades vary, however, and to some extent are within control of the mine operator. The milling costs per pound U308 are very sensitive to ore grade, since the bulk of the costs incurred in milling are for ore handling, and relatively little for nandling the product.

Labor makes up about 30% of the cost of mill operations, reagents and supplies about 60%, and utilities, taxes and miscellaneous items account for the remainder. During calendar year 1971, the distribution of effort in the operating mills was as follows:

	Number Employed	% of Total
Office	302	18.4
Maintenance	500	41.4 30.5
Warehouse and Laboratory	159	9.7
Total	1,639	100.0

The labor productivity, expressed in tons of ore processed per man per day are indicated in Fig. 7. Again, the inner area designates the recent experience on two-thirds of the plants, and the outer area includes the remaining third. It is worthy of note that labor productivity has been rising, and the median figure is now about 20 tons per man per day for the full mill complement.

V. Conclusions

In conclusion, recent experience indicates that all of the processes in use in the U.S. today are capable of producing uranium concentrates from ore at reasonable and competitive costs. There is no one process that has a clear economic advantage. The data does indicate, as would be expected that larger mills have some advantage in terms of lower capital and operating costs. These advantages are not clear cut or conclusive, however, and mills of modest size have been built and operated at competitive costs. In view of the considerable variation that has been experienced in cost of plant construction and operation, considerable care and attention to detail in the planning phase is necessary. The choice of process to be used should be made on the basis of pilot testing of the various methods on adequate and representative samples of the ore to be processed. The persons with metallurgical expertise should play a major role in plant design. Some of the most serious errors in mill design have resulted from failure to give adequate consideration to these factors. Consequently, costly delays in reaching full production and added operating costs have been all too frequent. With good design, however, it appears that a mill of modest size, 800-1,200 ton-per-day capacity, can be fully competitive.

TABLE I

U. S. URANIUM REQUIREMENTS AND DELIVERY COMMITMENTS $\frac{1}{}$ (Tons of U₃0₈ - January 1, 1972)

Year of U ₃ 0 ₈ Delivery	Requirements	Delivery Commitments
1972	9,200 ^{2/}	11,700
1973	10,600	12,400
1974	14, 400	13, 100
1975	18, 200	13, 500
1970	20, 500	2,700
1977	23, 800	4,800
1978	28, 400	4,900
1979	32, 600	4, 100
1980	37, 000	2, 900
1981	42, 100	1,900
1982	47, 400	1,600
1983	53, 200	3/
1984 1985	59, 700 66, 600	

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Domestic producers to domestic buyers Commitments to foreign buyers are 1,600 tons. Deliveries were 4,700 tons before 1972.

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Pre-1972 domestic deliveries were 32, 200 tons.

31

Delivery commitments for 1983-1992 are 5,300 tons.

From: WASH-1169 Nuclear Industry Supply Survey, April 1972.

TABLE II

AN EXAMPLE OF EXPLORATION, DEVELOPMENT AND MINING COST FOR A 1,000 TON PER DAY OPERATION

	0. 20% U ₃ 0 ₈ Open_Pit		0. 25% U ₃ 0 ₈ Underground	
	\$/Ton	\$/Lb. Rec.	\$/Ton	\$/Lb. Rec.
Capital Costs				
Acquisition	0.60	0.16	0.75	0.16
Exploration Drilling U_	2.00	0. 53	2.50	0.53
Development Drilling	1.00	0.26	1.25	0.26
Mine Primary Dev.	5.40	1.42	3.25	0. 68
Mine Plant & Equip.	<u>0.20</u>	<u>0.05</u>	<u>0. 70</u>	0.15
	9.20	2.42	8.45	1.78
Operating Costs				
Mining	2.40	0. 63	12.00	2.53
Hauling	0.65	0.17	0.80	0.17
Royalty	1.35	0.36	1.75	<u>0.37</u>
Totals	13.60	3.58	23.00	4.85

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TABLE II (continued)

	0.20% U ₃ 0 ₈ Open Pit		0.25% U ₃ 0 ₈ Underground	
	\$/Ton	\$/Lb. Rec.	\$/Ton	\$/Lb. Rec.
Cost of Ore	13.60	3.58	23.00	4. 85
<u>Milling</u>	18 ang	98 96 40 40 40 40 10 ⁶ 0 ⁵ 0 ⁵ 0 ⁶ 0 ⁶⁰ ⁶⁰	****	40 40 40 40 40 40 40 40 40 40 40
Mill Plant & Equipment	1.40- 2.20	0. 37-0. 58	1.40- 2.20	0. 30-0. 46
Operating	<u>4.00-10.00</u> 5.40-12.20	<u>1.05-2.53</u> 1.42-3.11	<u>4.00-10.00</u> 5.40-12.20	<u>0.84-2.10</u> 1.14-2.60
Total Production Cost per Lb. U ₃ 0 ₈		<u>5.00-6.69</u>		<u>5.99-7.41</u>

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APPENDIX A

Uranium Concentrate Specifications

1. Allied Chemical Corporation

URANIUM CONCENTRATES

UF₆ CONVERSION SURCHARGES*

The following specifications and surcharges have been established for UF₆ conversion services. When uranium concentrates fail to meet the specifications set forth and designated as "standard" concentrate, surcharges will be assessed for any deviations up to the "maximum limit" concentrate specification according to the surcharge schedule, based on sampling, weighing, and assay of each lot. Surcharges will be computed to the nearest whole pound.

		"Standard" <u>Concentrate</u>	"Maximum Limit" <u>Concentrate</u>	Surcharge Schedule (\$/1b U)
1.	Uranium (U)	75%	65% min.	none
2.	Vanadium $(\dot{V}_{0}O_{F})$	0.10%	1.80% max.	see below
3.	Phosphorus (PO))	0.10%	0.70% "	see below
4.	Halides (Cl, Br, I)	0.05%	0.10% "	none
5.	Fluoride (F)	0.01%	0.10% "	none
6.	Molybdenum (Mo)	0.10%	0.30% "	\$0.005
7.	Sulfur (SO_h)	3.00%	8.00% "	none
8.	Iron (fe)	0.15%	1.00% "	none
9.	Arsenic (As)	0.05%	0.15% "	none
10,	Carbonate (CO_2)	0.20%	0.75% "	none
11.	Calcium (Ca) ^{D'}	0.05%	0.75% "	\$0.005
12.	Sodium (Na)	0.50%	7.50% "	see below
13.	Boron (B)	.0050%	0.10% "	none
14.	Potassium (K)	0.10%	3.00% "	\$0.005
15.	Water (H ₂ O)	2.00%	4.00% "	\$0.005
		•	•	• •

Specifications 1 through 14 are to be determined on a dry weight concentrate basis. Specification 15 is to be determined on a natural weight basis.

*Subject to escalation

Only uranium concentrates of natural origin (non-irradiated) containing 0.711% U_{235} will be acceptable under this agreement.

<u>Vanadium</u> $(V_2 O_5)$

If vanadium content is greater than 0.10%, a surcharge of \$0.005 per pound U will be assessed for each additional 0.10% vanadium or portion thereof up to a maximum allowable limit of 1.80%.

<u>Phosphorus</u> (PO_{j_1})

If phosphorus content is greater than 0.10%, a surcharge of \$0.004 will be assessed per pound of U for each additional 0.10% phosphorus or portion thereof up to a maximum allowable limit of 0.70%.

Sodium (Na)

If the sodium content exceeds 0.50%, a surcharge of \$0.02 per pound U will be assessed for each additional 1.00% sodium or portion thereof up to 3.5%. If the sodium content exceeds 3.5%, an additional surcharge of \$0.03 per pound U will be assessed for each 1.00% sodium or portion thereof from 3.5% up to maximum allowable limit of 7.5%. No blending of dry concentrates to reduce sodium levels will be allowed. Quantities of concentrates exceeding 0.50% sodium which will be accepted are subject to negotiation. Any deviations which exceed the listed maximum allowable limits are subject to negotiation in the determination of surcharges which may be assessed.

2. Kerr-McGee Corporation

Part I

CONCENTRATE SPECIFICATIONS

Foreign Matter. The concentrate shall be free of foreign material or objects (that is, any material or object not produced as a constituent of the concentrate in the milling of uranium ore) which would be detrimental to either the sampling of the concentrate or to the equipment used therefor.

Flowability. The concentrate shall be sufficiently free-flowing to allow it to be sampled by normal "falling stream" procedures.

Particle Size. The concentrate shall be capable of passing a U.S. Standard No. 3 Sieve (1/4 inch screen).

Amenability. The concentrate shall have a travel time not exceeding five (5) seconds under the amenability drop test procedure.

CONSTITUENT	SPECIFICATION (Limit Without Surcharge)	BASIS
Uranium	60.00% Min.	Concentrate as Received
Extractable Organic		
Material	0.10% Max.	Concentrate as Received
HNO ₂ -Insoluble Uranium	0.10% Max.	Uranium Content
Molybdenum (Mo)	0.15% Max.	Uranium Content
Vanadium (V)	0.10% Max.	Uranium Content
Calcium (Ča)	1.00% Max.	Uranium Content
Thorium (Th)	2.00% Max.	Uranium Content
Zirconium (Žr)	2.00% Max.	Uranium Content
Boron (B)	0.15% Max.	Uranium Content
Phosphorus (P)	0.35% Max.	Uranium Content
Halogens (Cl, Br, I)		
As Chloride	0.25% Max.	Uranium Content
Fluorine (F)	0.15% Max.	Uranium Content
Carbonate (CO ₂)	2.00% Max.	Uranium Content
Sulfur (S)	3.50% Max.	Uranium Content
Arsenic (As)	1.00% Max.	Uranium Content

Part II

SURCHARGES AND YIELD PENALTIES TO BE APPLIED TO (AND PROVISIONS FOR REJECTION OF) OFF-SPECIFICATION URANIUM CONCENTRATES

The following surcharges are assessed in lieu of rejection:

Uranium Content

For concentrates containing less than 60.00% uranium on a concentrate as-received basis, a surcharge of \$0.005 per pound of concentrate as-received weight will be assessed for each 1.00% absolute or portion thereof by which the measured uranium content differs from 60.00%.

Nitric-Acid-Insoluble Uranium

For concentrates containing greater than 0.10% nitric-acidinsoluble uranium, the guaranteed yield will be reduced by the amount by which the nitric-acid insoluble uranium exceeds 0.10% of the contained uranium.

Molybdenum (Mo)

For concentrates containing molybdenum in amount greater than 0.15% but less than 0.45%, a surcharge of \$0.005 per pound uranium will be assessed. If the molybdenum content is greater than 0.45% of the contained uranium, then in addition to the said \$0.005 per pound surcharge, a surcharge of \$0.02 per pound uranium will be assessed for each 0.10% or portion thereof by which the molybdenum content exceeds 0.45% of the contained uranium.

Vanadium (V)

For concentrates containing vanadium in amount greater than 0.10% of the contained uranium, a surcharge of \$0.005 per pound uranium will be assessed for each 0.10% absolute or portion thereof by which the vanadium content exceeds 0.10% of the contained uranium.

Other Specifications

For concentrates which fail to meet one or more of the remaining specifications of Part I of this Exhibit "C", a surcharge of \$0.01 per pound uranium will be assessed.

Note 1: Special handling fees are charged, subject to mutual agreement, for concentrates not meeting specifications for Foreign Matter, Flowability, and Particle Size. A fee of 5¢ per pound of uranium (U) is charged for concentrates accepted, but which fail to meet specifications for Amenability and Extractable Organic Material.

Note 2: The provisions of Part II of this schedule have been paraphrased for brevity.







Flowsheet 3: Acid leach, ion exchange or solvent extraction Flowsheet 4: Acid leach, resin-in-pulp, with "Eluex" alternative

Carbonate leach, sodium hydroxide precipitation

Flowsheet 1: Flowsheet 2:

Carbonate leach, resin-in-pulp

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FIGURE 5

FIGURE 8



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FIGURE 7









- Acid leach, ion exchange or solvent extraction Flowsheet 3:
- Acid leach, resin-in-pulp, with "Eluex" alternative Flowsheet 4:




FIGURE 5



FIGURE 6



FIGURE 7

FROM ORE TO CONCENTRATE

New Techniques and Prospects in Ore Processing

J.E. Léger and G. Boutonnet Pechiney Ugine Kuhlmann, France

Abstract

This article describes the reasons and circumstances governing the development of uranium hydrometallurgy in France, the objectives laid down for this industry, technological resources and expected future developments.

Two French processes stand out against the standard methods adopted elsewhere in the world: the "calcium" process and attack by forming a paste.

Chemical, technological and economic research has gone on since 1960, into the most logical form of connection between the concentration units on the mining sites and the fluoridation units preceding enrichment.

Yellow cake can no longer be justified except in special circumstances, while UF₄ appears as an appropriate and obligatory intermediate stage. Two processes have been tested up to the pilot stage, to prove the possibility of producing UF₄ on the site suitable for conversion into UF₆.

Resumé

On expose les raisons et les circonstances qui ont présidé au dévieleppement de l'hydrométallurgie de l'uranium en France: objectifs imparts à cette industrie, moyens technologiques et développements prévisibles de l'époque.

Deux procédés français se détachent des schémas classiques adoptés ailleurs dans le monde: le procédé dit "calcique" et l'attaque par empâtage.

Depuis 1960, ont été poursuivies des études chimiques, technologiques et économiques sur l'articulation la plus logique entre les unités de concentration établies sur les sites miniers, et les unités de fluoration précédant l'enrichissement.

Le "yellow cake" ne se justifie que par des raisons circonstancielles, tandis que UF₄ apparaît comme point de massage privilégié et obligatoire. Deux procédés ont été expérimentés jusqu'au stade pilote pour faire la preuve de cette possibilité de produire, sur le site minier, UF₄ apte à être transformé en UF₆.

FROM ORE TO CONCENTRATE

New techniques and prospects in ore processing

France started to extract uranium seventeen years ago, in January 1955, shortly before the First International Conference was held in Geneva.

Six years later, in February 1961, the fourth of the plants built by French industry came on stream in the Gaboon Republic, bringing production capacity up to two thousand metric tons of metal per annum.

In order to understand the development which has occurred in this industry's objectives, it is necessary to refer back to the circumstances which existed when this industry first came into being.

In France, as elsewhere, following the near-universal failure of physical methods of concentration, mining engineers took their ore processing problems to the chemists. The fact that processing units had to be designed and built as quickly as possible, combined with the then state of the art as concerns wet metallurgy and the security blackout on work done in other countries led to the adoption of simple process routes. At the time, the international "economics" of uranium did not come into the picture. Each of the countries concerned followed its own, strictly national, policy, which was frequently dictated by strategic considerations.

The industrial facilities built in response to these requirements responded remarkably well to the demands made upon them in the years immediately following their installation when they were having to produce to full capacity.

There can be no doubt that the original scheme of things would have undergone changes earlier if the size and scope of demand had continued to expand, instead of which the pattern of facilities installed in 1961 found itself frozen for a decade by the subsequent curtailment of demand.

While, therefore, there were no changes in terms of concentration and production facilities per se, process technology and economics underwent a radical transformation, involving (i) a modification of the original objectives and planned use of the finished products, (ii) improved production methods and (iii) greater emphasis on medium and long-term market prospects.

at the same time, the French ore processing industry took advantage of this pause to think out what it wanted to do and run pilot-scale tests on process routes which had been glimpsed as early as 1960 and were gradually being seen as more rational.

A - ORIGINAL TECHNOLOGICAL AND ECONO IC CONCIDERATIONS AND SUBSEQUENT DEVELOPMENTS

1. ... Concentration industry objectives and planned uses of products

In France, as elsewhere, uranium extraction and processing objectives were twofold :

(i) In terms of economics, the aim was to convert the ore - which could not economically be shipped - on the srot and in as simple a manner as possible into a technical-grade concentrate which could economically be shipped.

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Alkaline processes, where they can be used, are a simple answer. Acid processes, however, owing to the quantities of acids employed and the impurities inevitably taken into solution, entail an additional stage of purification by ion exchange prior to the precipitation of the final concentrate.

(ii) In terms of process technology, it was then desired to refine the technical-grade concentrate as a separate stage, which had above all else to ensure consistency of product quality rather than adherence to standards which could not always be expressed in terms of analytical limits. French industry sought a second stage designed with a view to the production of natural uranium metal.

The shift towards the use of enriched fuels has since introduced the conversion to the hexafluoride, and hence new standards for the UF₄ to be fluorinated. Certain specifications - e.g. molybdenum content - have increased in severity, while others - e.g. rare earths with large neutron cross-section - have been relaxed. At the same time, standards have been made international.

2 - Process technology

The solid and subsequently liquid ion exchange systems constituted the maximum feasible revolutionary development when the first industrial units were built. The French plants at Ecarpiere (Loire Atlantique) and Bessines (Haute Vienne) exemplified that trend: the former uses strong anionic resins, while the latter uses these resins

in combination with a liquid/liquid exchange process employing a secondary amine. Otherwise, the methods of wet grinding, digestion in stirred vessels, washing and grading continued to be based on what was most orthodox in ore processing practice.

As early as 1958, however, two new developments provided an insight into the potentialities of this infant industry.

The calcium process

The first of these developments found expression in the Forez plant in France and the Mounana plant in the Gaboon Republic, which came on stream in January 1960 and February 1961, respectively. This was the "Calcium process", which features a two-stage concentration flowsheet. Each of these stages involves a departure from earlier practice.

(Fig. 1)

Digestion of the ore is followed by the conventional liquid/solid separation, using a filter at Forez and separators at Mounana. The uranium-containing solution then undergoes a two-stage precipitation treatment :

(i) The solution is first treated at pH 3-3.5 to neutralise residual free acidity and precipitate the bulk of the ferric hydroxide. The latter is recirculated to the ore digestion stage to maintain a concentration of 6-8 g/l of iron and, after addition of an oxidising agent, a ratio of ferric to ferrous ion of 6-10/1. Ferric iron was known at the time to be able to oxidise U⁴⁺ and facilitate solution of UO_2^{++} , especially in the presence of inhibitors such as chelating phosphates. This process results in selective digestion of the gangue, so that, in the case of the ores concerned, solubilisation efficiencies of 96.5-97% are obtained with 25-30 kg/ton of sulphuric acid by maintaining a pH close to 1.8.

(ii) The second precipitation stage, from an almost neutral solution, yields a crude concentrate with the following typical composition (calcium process) :

(Calculated on
$$U = 40-45\%$$
,
dry weight) $AL_2O_3 = 5-7\%$
 $Fe_2O_3 = 2-3\%$
 $SiO_2 = 4-6\%$
 $CaO = 5-9\%$
 $CO_3 = 20-30\%$

This crude concentrate can then be looked upon as a rich ore, thus warranting the use of solution reagents which are more expensive than sulphuric acid, e.g. nitric acid, and highly selective purification techniques hitherto used only in refining processes, e.g. tributyl phosphate.

Concentration yields a uranyl nitrate of the following composition :

U = 400 g/l

$$..1 = 20$$
 ppm (culculated on U)
Cu = 10-20 ppm
Fe $\angle 20$ ppm
Si = 13 ppm
Mo = 0.5-3 ppm
Mn = 10 ppm
B equiv. = 20-100 ppm

Hence, as early as 1961, it had been demonstrated that a concentrate which was substantially within the analytical specification hitherto associated only with refined material could be produced on-site via a process flowsheet comprising no more individual stages than "conventional" routes.

The process exhibited another very important advantage, i.e. the natural "break" at the crude concentrate production stage. This was turned to account in the design and construction of the plant in the Gaboon Republic, where primary concentration facilities were installed on the extraction site almost four hundred miles from the coast and many more hundreds of miles from any industrial complex capable of supporting a sophisticated process unit. Production facilities are simple and orthodox, process feedstocks are few in number (sulphuric acid, flocculating magnesia or lime) and the plant can be operated by personnel with relatively little industrial experience.

The intermediate concentrate is subsequently processed in France, where improved resources and facilities - in terms of reagents, process control and servicing - are available.

The reaction paste process

The second of the developments referred to earlier was designed to reduce consumption of sulphuric acid, the major cost factor in the ore concentration process. As early as 1955, Kuhlmann had patented a process based on the use of dilute sulphuric acid to digest ore in the form of a paste.

Generally speaking, the digestion process requires the maintenance of an appreciable level of free acidity (10-40 g/1) in order to digest the gangue and ensure an adequate rate of solution of the uranium contained therein. Additionally, the normal method of solution, in stirred vessels, of a slurry obtained by a wet grinding process, produces 0.7-1 cu.m. of solution per metric ton of dry material. This means that 20-40% of the acid used - in the case of French plants - has to be neutralised and goes to waste.

The use of a diluted acid spray to produce a reaction paste goes at least part of the way to meeting three objectives, viz :

(i) The quantity of water employed is restricted to the absolute minimum necessary to enable the various reactions involved, which are basically of the ionic type, to occur. Depending on the type of ore concerned, and particularly the clays of which it is composed, the volume of water required can vary from 40 to 100 litres per metric ton dry weight of ore, so that, for the same free acid concentration in the final digestion medium, the amount of acid not used and which goes to waste is reduced by a factor of ten or twenty as compared to a slurry containing 700-1000 litres of water per metric ton of dry material.

(ii) An aqueous medium containing hot 50% sulphuric acid is itself oxidising enough that less, if any, of the usual oxidising agent is required.

(iii) The heat of dilution and reaction of the sulphuric acid, combined with the fact that the volume of water used is small, may in practice raise the temperature of the mass sufficiently to ensure a suitable rate of solution.

Additionally, the very small volume of water required to operate this process makes it particularly suitable for applications in arid regions. This idea could not, however, be tried out right away and was first put into effect in the Comair plant in the Niger Republic, which came on stream in November 1970.

In addition to these ideas, which had a substantial effect on process flowsheets, work went ahead patiently on the newer techniques based on liquid/liquid exchange. While it would take too long, and go too far beyond the scope of this presentation, to go over all the steps involved, the avenues opened up by refining processes incorporating these techniques will be considered in a moment.

3 - Production prospects and market-oriented trends

Between 1955 and 1958 nuclear energy had by no means shown itself to be competitive. It was hoped that this could be demonstrated between 1965 and 1970 and hopes were just high enough to support moderate forecasts for the next fifteen years.

Hence, world production capacity in terms of concentrate is new sufficient to cater for demand for at least two decades. France is no exception to the rule, so that there can be no serious thought of modifying the original overall pattern of concentrate production and refining facilities.

The nuclear programmes of every country concerned have now been established for as far ahead as 1985, or even 1990. Any slight uncertainty which may still persist due to environmental objections should not cause nuclear schedules to be modified by more than two years.

This means that by 1980, at the latest, world demand for concentrate amounting to 50,000 metric tons per annum - will be higher than the production capacity of plants now in operation, under construction or definitely to be built.

Longer term forecasts point to a level of demand of 120,000 tons by 1985 and 200,000 tons by 1990.

The prospect of having to increase production capacity by a factor of four within the space of fifteen years means that, however optimistic a view is taken, a long, hard look needs to be taken at the total industrial resources available in the chain extending from extraction to the hexafluoride.

TECHNOLOGICAL RESOURCES AND POTENTIAL HOUTES

The sequence of possible operations from the ore to the hexafluoride includes only one which is absolutely mandatory. This is the production of UF_4 . Logically, therefore, the pattern should be one of local mine-based units producing UF_4 for subsequent conversion to UF_6 by a central unit, assuming of course that no technical or economic barriers exist.

Again, it is a basic principle of process design that the most economic route is generally that which involves no changes of phase. In the particular case considered, therefore, the hydrometallurgist will seek a process sequence terminating in the production of UF_4 which can be carried through as far as possible in the liquid phase. Any stage producing solid intermediates will, unless it exhibits clear advantages such as those associated with the calcium process described earlier, inevitably entail additional costs in terms of separation, drying, packing and re-solution.

A final consideration based on the hydrometallurgist's experience of what nature can hold in store is that no process can claim to be of universal application. Hence, failing a unique process route able to cater for any and all types and ore, what has to be sought is the right combination of two or more process sequences.

Assuming that these basic premises are accepted, several proposals can be made.

(Table 2)

5.

The nature of the ores normally encountered is such that what first suggests itself is to employ the two process routes discussed earlier, viz :

I - Digestion, leaching and concentration by ion exchange, all operations being carried out on an open-run basis.

II - Digestion, leaching and precipitation of a crude concentrate and, possibly, recirculation to the digestion stage of part of the soluble salts.

Which of these two methods is employed will normally be determined by :

- (i) how difficult the ore is to bring into solution, since only the open-run process can ensure the very strongly acid conditions which may be required;
- (ii) alumina or soluble silica contents, since these will mean a leaner crude concentrate where the calcium process is employed; and
- (iii) the presence or otherwise of inhibitors, e.g. P₂O₅, which may be released in excessive amounts and thus build up in the calcium process, which employs recirculation.

<u>I.1</u> - This is the conventional route and has been improved in many respects, mainly as concerns the ionic concentration stage, e.g. use of more highly selective solvents, improved phase separation, use of synergistic agents, introduction of saturation or scrubbing stages.

These improvements have not, however, at least as far as the French techniques are concerned, enabled the THP refining stage to be eliminated. The amounts of sulphates, molybdenum, silicon, iron and rare earths present in the concentrates obtained are such that it scarcely appears feasible to pass directly to the stage of reaction with HF.

1.2 - The main improvement with respect to conventional techniques has been the combined use of resins and solvent, based on elution of the resins with sulphuric acid (Eluex or Bufflex processes).

(Fig. 3)

The pilot plant flowsheet used at the Bessines plant (Haute Vienne) does not feature anything particularly novel. It yields an ADU with an analysis comparable to data published with respect to similar processes. viz :

> U: 74-75% SO₄: 1.4 % Al: 20 ppm (calculated on U) Fe: 100-200 ppm Si: 70 ppm Mo: 20 ppm Mn: 3 ppm Ni: 5 ppm B: 0.4 ppm

In view of the fact that the ADU costs no more to produce than magnesium uranate, this production flowsheet would exhibit a decisive advantage as compared to any alternatives, were it not for its susceptibility to :

- (i) the Mo content of the ore, as can be seen from the analysis ;
- (ii) the presence of rare earths ; and
- (iii) variations in the physical appearance which are little compatible with the French technique of dry reduction and hydrofluorination.

<u>I.3</u> - Another and more decisive process, also being run on the pilot scale at the Bessines plant, employs a corbination of resins, sulphuric acid elution and a tertiary amine. This is based essentially on conversion of the organo sulfate complex to the much less stable hydrochloride and enables uranium to be obtained in comparatively concentrated solution containing no cations other than those arising from impurities (Fig. 4).

Conversion is a very simple two-stage process employing conventional mixing/separating equipment and 7-8N hydrochloric acia.

A 5-stage process of aqueous extraction of the unaryl chloride yields an eluate containing 150 g'l of uranium and a solvent which has been effectively stripped of uranium. The eluate is then concentrated to 200-250 g/l, reduced with SO_2 in the presence of a copper catalyst and then reacted with HF to precipitate UF₄, these operations being carried out in one and the same reactor.

The UF, obtained is of very high purity, viz :

U \ 74% SO₄ = 30-50 ppm A1 Z 3 ppm Fe = 7-12 ppm Si = 1-3 ppm Mo \angle 0.5 ppm Na \angle 3 ppm Rare earths \angle 0.4 ppm

The extra cost as compared to conventional "yellow cakes" amounts to 10-11% of the cost of concentration of the latter, including frinding.

The very small increase in capital cost required for this process flowsheet as compared to the one described previously (I.2), its convenience in terms of process operation and present-day progress in plant automation all support the contention that UF_4 of a quality suitable for higher fluorination should be produced on the extraction site.

<u>II.1</u> - As early as 1960, the calcium process was being used to produce uranyl nitrate with the analytical composition given earlier, although molybdenum and mare earth contents were still too high for the dry hydrofluorination process.

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11.2 - The wet process does away with this obstacle.

(Fig. 5)

This novel route was tried out in a pilot unit at the Forez plant and then successively scaled-up to 15 kg/day at Eurochemic's Mol plant and is to be extended to almost 2 tons/day at the Forez plant.

The sequence of operations involved is as follows :

- (i) The solution of uranyl nitrate containing 400 g U/l produced by the calcium process is denitrated, using stear and sulphuric acid, in a packed column
- (ii) The solution of uranyl sulphate thus obtained is diluted and put through three electrolytic reduction cells employing mercury cathodes, platinum anodes and polypropylene gauze membranes. Reduction efficiency (quadrivalent/total U) is as high as 99%.
- (iii) The next stage is continuous preciditation of uranium hexafluoride by addition of 20% HF to the uranous sulphate solution at 90°C.

Filtration, drying and dehydration in an atmosphere of nitrogen yields a dense UF_A of the following composition :

Where the calcium process can be used, it will produce uranyl nitrate at a cost which is substantially that of conventional yellow cake.

The cost of reducing uranyl nitrate electrolytically to UF_4 is less than F.F. 2/kg uranium.

<u>II.3</u> - A variant technique, taking the form of an improved method of purification of the concentrate obtained by the calcium process, has recently been patented by a French Group. This involves taking up the concentrate - obtained by precipitation of the acid liquors with lime or magnesia - with sulphuric acid and then carrying out a fractional separation sequence by successive dilution with water.

Final crystallisation yields uranyl sulphate with a purity entirely comparable with that of TRP-purified uranyl nitrate and which, when redissolved, can be put directly through the electrolytic reduction stage already described.

The direct cost of producing UF_4 by this latter route is - subject to final detailed assessments - only F.F. 1.20/kg higher than that of concentration in the form of yellow cake.

CONCLUSION

In very general terms, metal salt production flowsheets are based on a smooth sequence of operations designed to minimise any breaks in the pattern of the process.

The process of converting the uranium contained in an ore to the hexafluoride is no exception to this rule. The chemistry of uranium is such that it is logical or even mandatory to proceed through certain intermediate stages. It is suggested that one or more of these intermediates be produced on the extraction site and subsequently further processed in fluorination units immediately upstream of the enrichment plant.

UF₄ would appear to constitute the first choice in terms of intermediates, although this does not rule out other possibilities, e.g. semi-concentrates such as those obtained by direct precipitation of primary acid liquors.

A decision has to be made, with a view to future development work, on the basis of both technical and economic considerations, i.e. the nature of the ores concerned, the size of concentration and fluorination units, and the geographical location of these units.

In approaching this decision, the hydrometallurgist is in a stronger position than was the case fifteen years ago. Techniques which had then barely been developed to the industrial stage have now been thoroughly mastered and combined. Liquid phase operations lend themselves admirably to automated control and analysis. Industrial facilities and flowsheets based on this technology are distinguished by flexibility and convenience and have outgrown earlier problems.

The regularity of operation which can now be assured, combined with the potential savings to be realised in terms of overall production cost should prompt a wider examination of how the industry is to be organised during the current decade.





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RESINES / ELUTION SULFURIQUE / SOLVANT



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SCHEMA #3







Schéma de principe pour 1 kg d'uranium

PRODUCTION SCALE PROCESSES AND PLANTS IN THE UNITED KINGDOM-THE

CONVERSION OF URANIUM ORE CONCENTRATES TO NUCLEAR GRADE

URANIUM HEXAFLUORIDE AND TO ENRICHED URANIUM DIOXIDE

by H. Rogan

ABSTRACT

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Large scale production of nuclear fuels in the United Kingdom is carried out by British Nuclear Fuels Limited, a Company formed from the United Kingdom Atomic Energy Authority in 1971. The Springfields Works of British Nuclear Fuels Limited undertakes the conversion of uranium ore concentrates to uranium metal and uranium hexafluoride, the reconversion of enriched uranium hexafluoride to uranium dioxide and the manufacture of different types of nuclear fuel elements with a wide range of specifications.

The lecture describes in some detail the Springfields Works process for the conversion of uranium ore concentrates to uranium hexafluoride - dissolution of the concentrate, filtration, purification and evaporation of the liquor, followed by its conversion to uranium trioxide, uranium dioxide and then to uranium tetrafluoride. The capacity of the uranium ore concentrate to UO_3 stages of plant is 7,000 tonnes U per annum. The capacity of the UO_3 and UF_4 production stages is 5,000 tonnes U per annum, and is now being increased to match that of the preceding stages. Uranium hexafluoride is produced by fluorination of the UF_4 in a plant with a capacity of 3,000 tonnes U per annum.

After a brief reference to uranium enrichment, carried out at the Capenhurst Works of British Nuclear Fuels Limited, details are given of the processes used for reconversion of enriched uranium hexafluoride to uranium dioxide, including a new single-stage dry process with a capacity of 500 tonnes U per annum at enrichments up to 5% U235, in which UF6 is directly converted to ceramic oxide by reaction with steam and hydrogen in a kiln. The associated processes for recovery of uranium residues and treatment of waste hydrofluoric acid, analytical procedures, the operating costs of the processes and the safety measures that are employed, are also described.

1. INTRODUCTION

Springfields Works, part of the British Nuclear Fuels Ltd Company formed from the United Kingdom Atomic Energy Authority in April 1971, has been manufacturing nuclear fuels for over 20 years. This lecture gives an account of those fuel manufacturing stages involved in the conversion of uranium ore concentrate to uranium hexafluoride, and makes a brief reference to uranium enrichment and conversion to enriched oxide.

The Springfields Works process for the conversion of uranium ore concentrates to uranium hexafluoride can be conveniently split into six process stages, viz:

- i. Dissolution of uranium and soluble impunties in the ore concentrates in nitric acid, followed by removal of the insoluble impurities by filtration.
- ii. Purification by separation of uranium from the soluble impunties by continuous, counter-current solvent extraction.

- iii. Evaporation of the pure uranyl nutrate solutions obtained from the purification plant to concentrated uranyl nitrate, followed by thermal denitration to uranium trioxide.
- iv. Hydrogen reduction of the trioxide to uranium dioxide.
- v. Hydrofluorination of the dioxide to uranium tetrafluoride.
- vi. Fluorination of the uranium tetrafluoride to uranium hexafluoride.

The following describes these stages in some detail, discusses the associated processes for recovery of uranium residues and treatment of waste hydrofluonc acid, analytical procedures, the operating costs of the processes and the safety measures that are employed.

2. RECEIPT, SA' IPLING AND DISSOLUTION OF URANIUM ORE CONCENTRATES

The uranium ore concentrates used as feeds to the BNFL process at Springfields Works are obtained mainly from Canada, South Africa, Australia and the United States of America. The concentrates are derived by calcination of precipitates of ammonium or sodium diuranate and, depending on their source, have a uranium content of about 60% to over 80% uranium w/w After being sampled and assayed the concentrates are fed to the first stage of the process for production of uranium fuel.

2.1 Dissolution

Drums of ore concentrate are fed to ventilated mild steel tipping cubicles and the contents emptied into stainless steel hoppers. From the hoppers, the concentrate is fed to the primary dissolvers by stainless steel screw feeders. The dissolvers used in ore dissolution are, because of the corrosive nature of the nitric acid dissolving agent, fabricated in stainless steel. They are quadranted stirred vessels, the ore concentrate being fed to the first quadrant of the primary dissolver, co-currently with concentrated nitric acid. The contents of this quadrant are heated to 95–100°C by a steam coil, to produce a slurry containing approximately 45% w/v uranium and 4.5–6.0% w/v free acid. The slurry is then aged in the remaining three quadrants of this vessel and in a secondary vessel to give a coagulated slurry with improved filtration properties. Automatic control of the acid and water feeds to the primary dissolvers is effected by use of a gamma spectrometer giving the uranium content of the slurry and a free acid monitor measuring pH.

The aged slurry is pumped from the secondary vessel to the trough of one of three primary rotary vacuum filters. The drum of each filter is covered by two stainless steel support gauzes and is pre-coated with a $2\frac{1}{2}$ inch layer of a perlite-type filter-aid. A clean surface for filtration is maintained at all times by an automatically advancing, stellute tipped knife, which continuously cuts off the top layer of filter aid and undissolved solids in the slurry, so that they can be discharged to a reslurry vessel. Wash water, fed at the rate of 20% of slurry input, is applied to the drum at a point just above the knife. The nitric acid solution of uranium and soluble impurities, together with the wash water, is drawn through the filter medium into pipelines, then to a filtrate receiver from which it is pumped to stainless steel crude product tanks. The filtered liquor contains about 30% w/v uranium and 4 5-5.0% w/v free acid. The residue from the primary filters is reslurried with secondary filter filtrate to produce a slurry containing 1-3% w/v solids. This slurry is refiltered on a secondary filter, operated on a similar principle to that used on the primary filters; the solids discharged from this filter contain less than 0.10% w/v uranium, at which level they can be disposed of outside the works under s⁺ tutory regulations.

3. PURIFICATION OF CRUDE URANYL NITRATE SOLUTIONS

Purification of uranyl nitrate solutions is carried out in two continuously operated counter-current flow, mixer-settler extraction boxes, using a solution of tri-normal butyl phosphate in odourless kerosene as the extracting agent.

This process is the determining step in the production of nuclear grade uranium, that is uranium which is capable of sustaining the fission reaction responsible for the prime generation of power in the nuclear power station. Amongst the more important impurities removed by the solvent extraction process are :

- (a) those with high neutron capture cross-sections such as boron and cadmium,
- (b) those elements which form volatile fluorides that can be distilled along with, and therefore contained in, the UF_6 produced at a later processing stage. Some of these compounds have deleterious effects on the gaseous diffusion plants used for UF_6 enrichment, and a notable example of this type of compound is molybdenum fluoride,
- (c) those with similar chemical properties to uranium, eg thorium.

Crude uranyl nitrate is fed automatically to mixer 8 of the first extractor, which is a sixteen-stage horizontal, stainless steel unit, each stage comprising a mixer with independent stirrer, and a settler. Purification is effected in 8 stages by use of a 20% v/v solution of TBP in odourless kerosene, fed in at mixer 1. Further purification is achieved in the remaining 8 stages by stripping the solvent extract with pure uranyl nitrate solution, fed in at mixer 16. This strips impurities out of the solvent. Automatic control of the crude uranyl nitrate feed to this extractor is achieved by recording the specific gravity and temperature of the loaded solvent in settler 7, and relating these two readings to the uranium concentration in this settler. A check for the presence of thorium in the loaded solvent leaving the extractor is carried out. Solvent, containing about 7% w/v uranium as uranyl nitrate, leaves the first extractor, is preheated to 50°C, then fed to the second extractor, a twelve-stage stainless steel mixer/settler unit. Here it is backwashed with 0 02N nitric acid solution heated to 60°C and prepared by acidifying distillate from the evaporation section. At these temperatures, the complex formed etween uranyl nitrate and TBP becomes unstable with respect to its constituent parts, and the purified uranyl nitrate is, therefore, transferred back into the aqueous phase to give a final solution, emerging from the extractor to the storage tanks, of about 10% w/v uranium as uranyl nitrate at about 0.3% free acid.

Stripped solvent from this extractor is recycled to the first extractor but a small proportion is continuously "bled off" to be washed with sodium carbonate solution and water before rejoining the stripped solvent main-stream. This process is necessary to remove TBP hydrolysis products, such as dibutyl phosphate and phosphoric acid, and odourless kerosene degradation products. Stringent regulations are imposed on the quantity of uranium discharged to drain in the raifinate from the first extractor. The uranium level in the raffinate is continuously monitored. Uranium concentrations of 20–25 ppm uranium are normally encountered and at this level the raffinate is discharged to drain, but at higher levels the raffinate is recycled to a Uranium Saveall Plant for recovery of uranium.

4. EVAPORATION AND DENITRATION

Pure, dilute solutions of uranyl nitrate are concentrated by evaporation to a 100% w/v solution (uranyl nitrate hexahydrate contains 117% w/v U) which is then thermally denitrated at 300°C to uranium trioxide.

$$UO_2(NO_3)_2 \times H_2O \rightarrow UO_3 + NO_2 + O_2 + \times H_2O$$

The reaction is endothermic, the heat of formation of UO_3 from the hexahydrate being 570 Kcal/kgm of uranium.

4.1 Evaporation

The pure dilute uranyl nitrate solution is pumped from the stainless steel storage tanks, through a series of stainless steel preheaters, to raise the liquor temperature to 90°-100°C. The hot liquor then passes to the first of four stainless steel evaporation stages operating in series. The concentration of the product from the fourth stage is controlled by measuring the boiling point elevation of the liquor in this stage.

The evaporators work under a partial vacuum. There is a progressive decrease in pressure from the first stage to the fourth stage, which is maintained at 350 mm Hg.

4.2 Denitration

Concentrated uranyl nitrate liquor is maintained at a temperature of about 90°C in a stainless steel stock tank, where it is spiked with concentrated sulphunc acid to give a final concentration of 1200 ppm sulphate (on a uranium basis). The function of the sulphate is to impart a greater degree of porosity to the UO₃ and UO₂ particles, a property which markedly enhances the rate of reaction of UO₂ with hydrofluoric acid. The throughput of the denitrators is dependent on the rate at which the concentrated liquor is sprayed to the bed. Thus, it can be seen that the capacity of the unit will increase as the uranium concentration in the concentrated liquor increases, and it is of vital importance to keep the liquor strength at, or around, the 100% w/v mark.

The liquor is pumped to one of two guns and sprayed into the denitrator, containing a bed of UO_3 , fluidised by air, at a temperature of about 300°C. At this temperature, denitration to UO_3 takes place; UO_3 continuously overflows down a central pipe to a lift pot and is pneumatically transported, via a routing pot, to one of two hoppers.

Reaction off-gases pass through primary stainless steel filters and back-up rigi-mesh filters before being used in a heat exchanger to preheat the fluidising air stream to the denitrator. Offgases finally pass to the nitric acid recovery plant. This recovery plant basically consists of a pre-absorption condenser, six packed towers in which the fume is absorbed in circulating aqueous solutions, and a chimney stack A very rigorous statutory control is exercised over the nitrous content of the fume passing to the stack and thence to atmosphere. Most of the nitric acid recovered in this plant, which also deals with the fume from other areas of the Works, is re-used for dissolution of ore concentrate and residues.

The material of construction of the denitrator and ancillary equipment is stainless steel. The denitrator reactor is fitted with internal heaters, providing a total of about 170 kW, and external muff heaters, rated at 100 kW total. The reactor bed temperature is regulated by a controller which determines the voltage supplied to both sets of heaters, and the ratio of external to internal heat supplied is controlled by a second transformer. Thermocouples, positioned on each of the internal heaters, on the outside of the reactor wall, and in the external heater jacket enable continuous scanning, for temperature control purposes, to be carried out.

The capacity of the plant from uranium ore concertrate dissolution to UO_3 production is 7000 tes U/annum.

5. HYDROGEN REDUCTION TO URANIUM DIOXIDE

The hydrogen reduction of uranium trioxide to uranium dioxide, carried out at a temperature approaching 500°C in one of four fluidised bed reactors, can be expressed by the equation :

$$10_3 + H_2 = 10_2 + H_20$$

The reaction is exothermic, the heat of reaction being -24 Kcal/mole.

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The reduction reaction proceeds in a semi-continuous manner. The reaction is started with a heel of UO_2 heated to just under 500°C, then about 4 tonnes of UO_3 are fed to the stainless steel reactor by a rotary feeder over a period of several hours to maintain a constant bed temperature. After the UO_3 has been fed, the reaction with hydrogen is completed and the product UO_2 powder is pneumatically transported with compressed nitrogen via a routing pot to storage hoppers, leaving a heel for the next reaction. It is important to ensure that the correct reaction time and temperature are utilised and also that the quantity of hydrogen being fed to the reaction is sufficient for complete reduction. Otherwise, incomplete reduction, which will be followed by the undesirable production of uranyl fluoride, UO_2F_2 , in the hydrofluorination section, will probably occur.

Hydrogen is generated on site by electrolysis of sodium hydroxide solution, compressed and used as fluidising and reactant gas. The reduction reactor is fitted with sintered stainless steel primary filters, stainless steel back-up filters and two external heat exchangers. The composition of the fluidising gas is maintained in the range 70–85% H₂ in N₂ and the level of O₂ in N₂ is continuously monitored. Reactor heating is provided by electrically operated external muff heaters, the bed temperature being controlled automatically and reactor wall temperatures scanned continuously.

Gamma radiation from a cobalt-60 source inside the reactor, and operating in conjunction with a detector sited outside the reactor, is used for powder level indication.

6. HYDROFLUORINATION TO URANIUM TETRAFLUORIDE

Uranium dioxide reacts with hydrofluoric acid according to the equation :

$$UO_2 + 4HF = UF_4 + 2H_2O$$

The reaction is exothermic, the heat of reaction being -43 Kcals/mole. Any residual UO₃ in the UO₂ is converted to uranyl fluoride, thus:

$$UO_3 + 2HF = UO_2F_2 + H_2O$$

Every effort is made, by adequate control of the reduction process, to produce UF₄ which contains not greater than 1.2% by weight of UO_2F_2 . The presence of UO_2F_2 has two undesirable effects on the processes downstream of UF₄ manufacture, viz:

(a) decrease in the efficiency of fluorine usage in the UF₆ production plant,

(b) unstable conditions during the process for magnesium reduction of UF₄ to uranium metal.

Approximately 8 tonnes of UO_2 are fed by gravity from a storage hopper to one of the eight inconel hydrofluorination reactors, containing a heel of UF₄, fluidising on compressed nitrogen. After loading, the bed temperature is allowed to stabilise, then superheated anhydrous hydrofluoric acid (AHF) is introduced to the reactor whilst, at the same time, the nitrogen flow is reduced. Over the next 12 hours of reaction, the nitrogen flow is gradually reduced to zero, while the AHF flow is increased to a maximum. This flow rate is maintained until a sample of powder, drawn from the reactor, is shown by analysis to have a UO_2 content of 1% by weight, or less. The reaction time is usually of the order of 40 hours. The bed temperature rises to 450°C, at which it is controlled by means of a bed thermocouple linked to the top and bottom bed external muff heaters. Temperature control is

also effected by continuously scanned thermocouples on the reactor wall. Because of the exothermic nature of the reaction, it is necessary to use heaters only after the reaction has proceeded for 20 hours. At the end of the reaction, the bed is fluidised by nitrogen while it is cooled by forced air cooling of the reactor exterior. The UF₄, after being offloaded to hoppers by pneumatic transport, can be transported, again pneumatically, either to the UF₆ production plant or to the uranium metal production plant.

Gases from the hydrofluorination reactors consist of nitrogen, water and hydrofluoric acid and are exhausted to brine-cooled condensers where the condensible water and acid are removed, while the incondensible nitrogen passes to a scrubbing system before returning to the Nash compressors.

The hydrofluoric acid solution, ranging in concentration from approximately 60-80% by weight, is fractionally distilled in a monel, sieve-plate type of distillation column. The AHF product is returned to an AHF storage tank prior to re-use as fluidising gas. The azeotrope is collected and after further treatment is sold.

The main nitrogen generating plant, involving the separation of air into pure nitrogen and an oxygen enriched mixture by low temperature fractional distillation, supplies nitrogen to Nash Hytor compressors which are used to compress nitrogen to approximately 45 psig. This compressed nitrogen, after being dried, is used as reactor fluidising gas.

The capacity of the reduction and hydrofluorination plant is 5000 tes U/annum and is now being increased to match the capacity of the preceding stages.

7. FLUORINATION OF URANIUM TETRAFLUORIDE TO URANIUM HEXAFLUORIDE

Uranium hexafluoride (UF_6) is produced by the reaction of uranium tetrafluoride (UF_4) with elemental fluorine (F_2) . The production of fluorine is an integral part of the production process and is carried out by the electrolysis of hydrofluoric acid (HF) in an electrolyte of fused KF2HF. The chemical equation for the process is:

 UF_4 (solid) + F_2 (gas) = UF_6 (gas) + 112,000 BTU/lb mol U

and the reaction is very exothermic.

The present plant has a capacity of 3000 tes U/annum, and in 1973 will be extended to 8000 tes U/annum.

7.1 Fluorine Production

Fluorine is produced by the electrolysis of the fused salt KF2HF at 90–100°C in mild steel cells equipped with amorphous carbon anodes at a current density of approximately 1 amp/sq inch. The cell temperature is controlled by a system of cooling coils which act as cathodes to the individual anode assemblies. The hydrogen fluoride content of the electrolyte is controlled by the automatic addition of anhydrous hydrofluoric acid to maintain a pre-selected electrolyte level in the cell. The hydrofluoric acid used has to satisfy a demonding specification with regard to water and combined sulphur content. Too high an HF content produces rapid anode deterioration, whilst too low a content leads to high electricity costs. Experience has shown that for optimum efficiency the HF concentration must be kept within the range 40% to 429′. As HF is the only material consumed in the electrolytic process it was found that its concentration could best be controlled by monitoring changes in the level of the electrolyte. The measuring equipment needs to be robust because of the corrosive environment but very sensitive because the levels corresponding to the maximum and minimum HF content differ by only $\frac{1}{36}$ °. The necessary control is achieved using a transducer to monitor the change in pressure on a dip pipe caused by changes in level and to transmit a signal which actuates the HF feed valve. The fluorine leaving the cells passes through a surge vessel which minimises any tendency for pressures to fluctuate.

The fluorine produced contains 5–7% v/v HF which is removed by absorption in a primary absorber containing sodium fluoride powder, to reduce the HF content of the gas to approximately 0.5% v/v. This is further reduced to 0.1% v/v by passage through a secondary absorber of sodium fluoride pellets. Both HF absorbers and the process lines in the fluorine plant are fabricated in mild steel. The primary absorber is provided with electric heaters which enable the sodium fluoride to be regenerated and the HF returned to the cells. The pressure of the fluorine and hydrogen in the cells is controlled at 5 cms WG and a safety system is provided which automatically shuts down the plant should the cell pressures deviate markediy from 5 cms WG.

7.2 Uranium Hexafluoride (Hex) Production

The purified fluorine together with sufficient nitrogen for fluidisation is fed into an inert bed of calcium fluoride contained in a cylindrical monel reactor provided with a conical base and a carefully designed distributor plate to provide the required pattern of gas distribution within the calcium fluoride. The reactor temperature is automatically controlled by means of external

electric heaters and a forced air cooling system and the UF_4 is fed into the reactor using a screwfeeder. Under normal operating conditions the rate of reaction is so rapid that the uranium in the bed does not exceed 5%.

The transfer of the uranium tetrafluoride from the Production Plant, approximately a quarter of a mile away, is accomplished using an enveying system. At the uranium hexafluoride plant the uranium tetrafluoride is collected in a large storage hopper and then pneumatically transported as required to a smaller feed hopper sited alongside the reactor. A screwfeeder at the base of this hopper meters uranium tetrafluoride powder into a larger constant speed screwfeeder which delivers the pow or into the reactor.

The UFs produced together with the fine solids entrained in the gas stream, excess fluorine and the nitrogen carrier gas is then passed through two sintered monel filters in series to remove the entrained solids. Solids collected in these filters are periodically removed and additions of calcium fluoride powder made to the reactor in order to maintain a given reactor bed height.

The second filter is provided to prevent collection of solids in the condensers should the first filter fail.

Uranium hexafluoride contained in the gas stream leaving the filters condenses as a solid in two of four mild steel condensers arranged in series which are cooled with fluorocarbon 11 at -40°C. Each condenser is used sequentially for primary and backing up duties. The gas stream passes first through the primary condenser where the majority of the uranium hexafluoride is collected and then enters the back-up condenser to remove the remainder of the uranium hexafluoride. The condenser tail gases than enter an oil free recycle compressor and a sample of this gas stream is analysed for fluorine by an automatic on line monitor. The uranium tetrafluoride feed rate to the reactor is manually adjusted to keep the fluorine concentration in the gas stream in the range 5-10°, v/v A major proportion of the condenser tail gas stream is automatically directed back to the reactor in order to achieve high fluorine usage efficiencies. The remaining gas is discharged to a wet scrubbing system via a cold trap which removes the last traces of UF₆ from the gas stream.

The effluent gases from the plant are scrubbed free of traces of fluorine, hydrogen fluoride and traces of uranium hexafluoride with caustic alkali prior to discharge to atmosphere via a stack 60 feet high. The effluent released from the stack is monitored continuously. Equois containing precipitated uranium are futered and monitored botore. Escharge to urani, Vuinerable areas of the plant are monitored for uranium and fluorino levels in the air.

7.3 Hex Collection

When 6 tonnes of uranium tetrafluoride have been charged to the reactor a primary condenser is considered full and another empty cold condenser is brought into line on back-up duty and the existing back-up condenser is put on primary duty. The full primary condenser is then taken off line and heated with hot fluorocarbon 11 at 105°C. This raises the temperature of the uranium hexafluoride to 90–95°C converting the solid uranium hexafluoride to a liquid which can then be run out into transit cylinders. A sample of the uranium hexafluoride is collected as it enters the cylinder.

7.4 Process Control Procedures

The continuous nature of the process, the absence of intermediate storage, and the low pressure differential allowed throughout the plant which operates at less than atmospheric pressure necessitates fine control procedures and immediate availability of analytical results. One example of the special instrumentation is the fluorine in nitrogen analyser which enables the fluorine concentration in the recycle gas stream to be monitored. This instrument withdraws a sample of gas from the stream eight times an hour, passes it over heated sodium chloride and measures the chlorine liberated by means of a kathrometer. This actuates a potentiometric recorder linked with a slave recorder on the control panel so that the fluorine content of the gas stream is continuously known and adjustments can be made as necessary to optimise efficiency. A second example is the automatic fluorine impurity analyser. Fluorine produced electrolytically contains a number of volatile impurities it is necessary to know that the purification system for removing HF carried over from the electrolyte is working efficiently and be able to identify the presence of CF_4 and SO₂F₂ which indicate incipient disintegration of the anodes. The analyser isolates samples of gas from an appropriate position in the purification system and separates impurities along a temperature gradient in a liquid nitrogen cooled coil. The coil is then subjected to a controlled heating cycle whilst purging with pure helium. The impurities are individually swept by a molecular transpiration process through a kathrometer detector and the result is displayed as a sequence of peaks on a calibrated chart.

The plant's fluorine usage efficiency HF to UF₆ is in excess of 90%, and the uranium efficiency, UF₄ to UF₈, is better than 99%.

8. RECOVERY OF NATURAL URANIUM RESIDUES

Various types of uranium-containing residues arise continuously in all plants processing natural material. Economically, it is advisable to feed the residues back into the main line at the plant in which they arise and this is done, for example, in the case of UF_4 residues. However, many of the residues occur in such a form, or contain such impurities, that this is not possible and for recovery of uranium from these residues, a central plant exists alongside the ore dissolution and purification plant. The uranium is leached from the residues with nitric acid, the slurries filtered and the filtered solutions purified in a separate purification plant, smaller than the main line unit, but operating in a similar fashion.

Amongst the problems that have been encountered, and overcome, in the operation of the recovery line purification plant are recovery of fluonce containing residues, and maintenance of solvent quality. Fluoride ion forms a complex with uranyl ion and therefore interferes with solvent extraction. Addition of aluminium to the dissolver to pre-complex the fluoride, overcomes this. Solvent quality deteriorates due to the presence of entrained oil, grease etc. which is not readily filterable. Restoration of solvent quality can be effected by washing it with concentrated nitric acid, then filtering it through activated charcoal.

9. ANALYSIS

In addition to the in-line analysers and checks described already, a number of other analyses are carried out on both uranic and non-uranic process materials. Amongst the most important of these are:

- i. Purity analysis of uranium ore concentrates which are purchased according to agreed specifications.
- ii. Analysis of pure uranyl nitrate for uranium, free acid, molybdenum, U_{235} content (every batch) and for other elements cited in the specifications for metal and UF₆ (on bulk samples to a schedule).
- iii. Monitoring of the progress of the UO₂ to UF₄ conversion reaction by analysis of powder samples taken at intervals specified for each run as it proceeds.
- iv. Analysis of each batch of natural UF₄ for UO₂, UO₂F₂, Fe (to check on possible breakthrough of transition metals in the purification plant, and corrosion of stainless steel evaporation equipment), Ni (inconel corrosion product), Mo and IA (U₂₃₅).
- v. Analysis of AHF for sulphur-containing compounds such as SO₂ and H₂SO₄ which enhance the rate of corresive attack of HF on monel and inconel.
- vi. Analysis of the pure azeotrope for re-sale to a schedule agreed by the contractor.
- vii. Analysis of all the UF₆ to a rigorous specification, with particular emphasis on volatile gas content and concentrations of elements which form volatile fluorides.

10. SAFETY

Because of the corrosive nature of many of the raw materials handled, and the dangers that can arise from ingestion and inhalation of uranium-containing powders, strict procedures are defined to ensure sall by of personnel working in the areas concerned. All in ustrial personnel working full-time in the chemical plants undergo a complete change into what is known as "contact clothing" at the start of each shift, and make use of shower facilities before redonning their outdoor clothing at the end of the shift. Masks are used in certain areas to guard against dust inhalation or ingestion, goggles or visors and acid-resistant gloves afford protection against contact with nitric acid and caustic alkalis. In some cases, in particular in the UO₂ to UF₄ conversion, and HF recovery plants, a full suit consisting of PVC gloves, trousers, tunic and an integral hood and clean air mask (connected to an air supply), is worn over the contact clothing to ensure protection from the highly corrosive hydrofluoric acid.

On the Hex Plant the possibility of recombination of hydrogen and fluorine following their electrochemical separation, ingress of oxygen into the hydrogen containing vessels, and release of fluorine into the atmosphere is minimised by the provision of an integrated automatic plant safety circuit actuated by small changes in pressure.

Explosimeter tests for presence of hydrogen are carried out before any burning or welding work is carried out in, or in the vicinity of, equipment which normally contains hydrogen. Before testing, of course, the equipment is well purged with nitrogen.

11. COSTS

Dealing first with Capital Expenditure, the original book value of the plant and equipment so far mentioned was approximately £M9, and these will be supplemented by the previously mentioned additional plant items to be installed shortly at a capital expenditure of some £M3.

On manufacturing costs, the total cost of conversion of uranium ore concentrate to UF_6 can be split into three parts, viz:

- (a) uranium ore concentrate (UOC) to pure uranyl nitrate (UN)
- (b) pure uranyl nitrate to uranium tetrafluoride (UF₄)
- (c) uranium tetrafluoride to uranium hexalluoride (UF_6) .

The operating costs for the three parts of the process can be split into their main components as shown in the attached table. These exclude uranium ore concentrate costs, the purchase price of this material ranging from \$5-10/lb U₃O₈.

PERCENT OF OPERATING COST		
UOC TO UN	UN TO UF4	UF4 TO UF6
36 (of which about 90% is nitric acid)	34 (of which 97% is anhydrous HF)	35
17	17.5	16
13	21	18
12	5.5	2
8	20	24 (of which about 70% is electricity)
14	2	5
	PERC UOC TO UN 36 (of which about 90% is nitric acid) 17 13 13 12 8 14	PERCENT OF OPERATINGUOC TO UNUN TO UF436 (of which about 90% is nitric acrd)34 (of which 97% is anhydrous HF)1717.51321125.5820142

12. ENRICHMENT OF HEX

In the production of enriched uranium fuel by far the most costly stage is the enrichment of the natural uranium hexafluonde. All the large scale enrichment plants currently in use in the United States, United Kingdom, France and Russia use the gaseous diffusion process Because of the very small separation factor in the diffusion of U_{235} and $U_{2.3}$, many diffusion stages are necessary, leading to very large and expensive plants for the production of enriched uranium. At the current American enrichment charge of \$32/Kg of separative work, the value of the enriched uranium in an average thermal reactor charge is some 3 to 4 times the fabrication cost of the fuel, including cans and components.

In order to complete the description of processing stages from UOC to enriched oxide powder, the following recounts the reconversion of enriched uranium hexafluonide to UO₂ powder.

13. RECONVERSION OF ENRICHED HEX TO UO2 POWDER

The original process used in Springfields Works for the conversion of enriched uranium hexafluoride to ceramic oxide was first developed in 1951. This was essentially a batch process and involved hydrolysis of the hex in silver lined vessels followed by precipitation of ammonium diuranate with ammonia liquor. Conversion of the diuranate to ceramic oxide could then be carried out by pyrohydrolysis to remove residual fluoride and reduction in batch furnaces. Normally, however, it was preferable to dissolve the pyrohydrolysed material in nitric acid since this permitted both liquid blending to be carried out and the incorporation of dissolved residues into the process stream. Ammonium diuranate was then re-precipitated from the blended uranyl nitrate liquor. Final conversion to ceramic oxide was then carried out by calcination and reduction with hydrogen in batch furnaces. The sintering quality of the oxide was controlled by varying the precipitation conditions and the temperatures in the furnace. Criticality safety in the process was essentially controlled by mass limitation in the various units and by moisture control in the dry powders. Many of the stages could be operated up to 7% enrichment. This process was later developed into a large scale continuous plant with a throughput of 300 tonnes U per annum. In this plant cylinders containing 2 tonnes uranium hexafluoride were heated in steam baths and the gaseous hex converted to uranyl fluoride by contacting with steam in an inconel fluidised bed reactor. Control of the uranyl fluoride discharged from the base of the reactor was carried out by operation of level indicators in the bed. The uranyl fluoride was then fed to an inconel kiln in which it was reacted countercurrently with steam and hydrogen. The product oxide was virtually stoichiometric but with a very low specific surface area which made it unsuitable for the production of high density fuel. This first stage oxide was, therefore, dissolved continuously in nitric acid. Liquid blending and the incorporation of dissolved residues as necessary was carried out at this stage. The blended uranyl nitrate liquor was then precipitated continuously with ammonia and the ammonium diuranate filtered and dried continuously. The ammonium diuranate was passed through a granulator to convert it to a free flowing powder and then to a stainless steel kiln for calcination and reduction co-currently with hydrogen to produce ceramic oxide. Finally, the ceramic oxide was homogenised in 2 te batches in a double cone blender before passing to the granulation process.

Criticality control in this plant was generally by the geometry of the units which ensured safety up to 4% enrichment. Air contamination was kept to a minimum by the arrangement of many of the plant units in ventilated cubicles, the extract air being drawn through absolute filters before discharge to atmosphere All liquid effluents were precipitated and filtered before passing to the drainage system.

This multi-stage wet process has now in general been superseded by a single stage dry process in which the hex is converted directly to ceramic oxide by reaction with steam and hydrogen in an inconel kiln. This plant has a throughput of 500 tes uranium/annum and has been designed to handle enrichments up to 5%.

In this plant the cylinders containing 2 tes uranium hexafluonde are heated by hot water sprays to give gaseous hex at low pressure. The hex vapour is fed with steam into the base of the outlet gas filter hopper. The hex reacts with the steam producing uranyl fluoride which is fed to the rotating kiln by means of a scroll feeder. The kiln is heated by a number of zone heaters so that the temperature profile through the unit can be varied according to the process requirements. Reduction is carried out by hydrogen fed to the powder discharge end of the kiln.

The ceramic powder from the kiln passes in to check hoppers from which it can be discharged in to large transport containers. A moisture meter in the check hopper continuously monitors the powder in the hopper and should the moisture level rise the outlet valve of the hopper is closed automatically. The ceramic powder is homogenised in 2 to batches before further processing.

The plant has been designed for operational control from one centralised control room. The control equipment includes a computer programmed data logger which prints a complete log of all the process parameters at required intervals. The data logger also has an alarm facility which prints out abnormal levels of the main parameters as they occur.

Effluent gases from the kilns are passed to an HF Recovery Plant.

Since the construction of the UF₆ plant the conversion of other customers' UOC to natural UF₆ has been undertaken on a large scale and so far over 6,500 tonnes of natural uranum hexafluoride have been sent to the USAEC diffusion plants for enrichment on their behalf.

In the new plants for the manufacture of enriched UO_2 powder, almost 1,200 tonnes of enriched UO_2 have so far been produced for the British Nuclear Power Programme and for export to overseas customers, either as powder or as fuel for their reactors.

The plant described in this paper for the conversion of uranium ore concentrate to UF_4 has been in full operation since 1960. Since then over 35,000 tonnes of uranium in the form of natural and depleted UF_4 have been produced, for conversion to uranium metal for natural uranium fuel, or for conversion to uranium hexcfluoride.

THE TREATMENT OF UO2-POWDER

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Abstract

Fuel elements of modern nuclear power stations are expected to have a good performance in spite of frequent load cycles and increased rod power. This tendency to increased quality not necessarily involves higher costs. Among the various manufacturing steps for fuel elements the conversion of low enriched uraniumhexafluoride to UO_2 -powder is delt with as well as the pelletizing of the powder. Both, conversion and pelletizing must be seen together. This leads to lower costs for the pellets because the powder parameters needed for more economical pelletizing can be better adjusted.

In addition to various other steps necessary to meet the before mentioned higher quality the introduction of UO_2 -pellets of lower density is of importance. In the following proceeding from the manufacturing methods, for pellets with usual high density the possible ways for the production of such advanced lower dense pellets shall be discussed. Special emphasis will be given to the related powder parameters and to the manufacturing method of such powder. With the use of pellets of medium and, especially, of low density the possible moisture and gas content of fuel rods has to be watched very closely.

Both methods and the related equipment with regard to conversion of uraniumhexafluoride to UO₂-powder and concerning pelletizing will be discussed.

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Introduction

Basicly production of fuel elements for light water reactors is marked by the following fabrication steps:

Conversion of enriched UF₆ or uranyInitrate to sinterable UO₂-powder.

Production of UO2-pellets.

Loading of pellets into canning tubes and sealing of the tubes.

Assembling of fuel rods and structural components to fuel elements.

In the following the steps of conversion and, especially, pelletizing will be described along with the considerations leading to the distinct solutions chosen in the PWR-line of RBG.

<u>Conversion</u>

As starting material for the production of fuel elements, mainly the uraniumhexafluoride is used which is delivered from the enrichment plant.

For both, pressurized and boiling water reactors, at the time being, slightly enriched fuel, that means up to about 3 % U-235 is used. Uraniumhexafluoride is the only binary compound fairly volatile at room temperature which is one of the reasons that it is suited for isotopic separation and for purification.

All chemical treatments of this material lead, irrespective of the kind of procedure, to fluorine containing products; for example, by hydrolysis to HF. The corrosion of equipment influenced by these products must be avoided in order to maintain the purity of the fuel.

Other starting materials are solutions of uranylnitrate of natural isotopic composition or are solutions from reprocessing plants with compositions somewhat higher than natural. Finally, the scrap material from the fuel plant itself is converted into the uranylnitrate state with or without liquid liquid extraction for cleaning.

Some of the conversion methods to UO_2 -powder can easily handle both UF_6 and uranylnitrate, whereas other methods can only work with UF_6 and will need an additional line for nitrate conversion at least for internal scrap recovery. This fact is of importance when different methods are compared and evaluated.

It is of special importance that conversion and pelletizing are seen together, that means that all steps leading to the finished pellets have to be taken into consideration. This philosophy is necessary because of several reasons. As example for the first reason - quality of pellets one should consider the fluorine content of the pellets. It makes no sense to put costly efforts into reducing the fluorine content of the powder below a difficult to reach limit knowing that during sintering fluorine is remarkably reduced. As a second reason may be mentioned the fact that for the pellet-production-costs starting with UF_6 a l l working steps are of importance. In case that it is possible to save working steps by harmonizing chemistry and ceramics a reduction of expenses can be expected.

In the following the various possible processes for the conversion of UF_6 to UO_2 are briefly discussed:

Among the so-called wet processes the ammoniumdiuranat process, the ADU-process, is well-known. After evaporation the UF₆ is dissolved in water and reacted with ammonia. Usually the ADU-precipitate is dried and then treated with steam and hydrogen in a furnace to reduce the ADU to UO_2 and decrease the fluorine content to an acceptable level. The powder then is stabilized by a
slight oxidation in air, milled and sieved. To improve the pressing conditions the powder must be mixed with a small amount of binder-lubricant precompacted and then granulated, followed by crushing through a coarse sieve. The binder-lubricant, usually zincstearat, also helps to decrease internal friction during pressing of the pellets.

In principle, the most known dry process consists of the reaction of gaseous UF_6 with superheated steam at elevated temperatures, creating solid uranylfluoride. This UO_2F_2 now has to be reduced by hydrogen to UO_2 . The simplified equation

 $UF_6 + H_2 + H_20 - UO_2 + 6 HF$

shows that as effluents only hydrofluoric acid is created. By this the dry process could be more advantageous than the wet processes which have more effluents. On the other hand, the various working steps leading to sinterable UO₂-powder do not seem to be free of problems. Concepts using rotary furnaces or fluidized bed furnaces are in competition.

Among the numerous possible wet processes Nukem and RBG in Germany have chosen the ammoniumuranylcarbonateprocess. In this case one gets a free-flowing UO₂powder which can be pressed directly without any additional treatment, such as milling, precompaction and so on. It can, however, also be processed by any other pelletizing techniques.

At first the UF_6 -containers are placed in a pressure vessel and connected to the conversion line (picture 1). In this vessel the UF_6 is heated by steam at 100° C. The UF_6 is vaporized at this temperature and the gas over the liquid has a pressure of 3.2 atm. Thus, the UF_6 can be transferred as a gas via heated piping to the next process step, the precipitation.

The hydrolysis of the UF₆ and the precipitation as AUC, ammoniumuranylcarbonate, is performed batchwise with batch-sizes of approx. 250 kg oxid. The precipitation unit (picture 2) at the beginning of each batch is filled with demineralized water. UF₆, CO₂ and NH₃ are dosed as gases through a nozzle-syst m into the water in such a manner that there is always enough excess of ammoniumhydrogencarbonate to form immediately AUC from the hydrolysed UF₆.

The reaction is performed according to the following equation:

 $UF_6 + 5 H_2O + 10 NH_3 + 3 CO_2 - (NH_4)_4 UO_2(CO_3)_3 + 6 NH_4F$

The AUC precipitates as crystals with an average grain size of 0.06 mm. The suspension is cooled down in the precipitation tank in order to obtain a perfect precipitation with a low uranium content in the filtrate. Instead of UF_6 , also uranylnitrate solution can be used as feed material. The resulting UO_2 -powder is of the same quality.

The suspension is then filtered by means of a plane rotating vacuum filter (picture 3). On this filter the AUC is successively washed with a solution of ammoniumcarbonate and methylalcohol. The filtrates contain only up to 200 mg uranium per liter. This uranium, together with the contained chemicals, can be recovered in an effluent treatment plant. After washing , the AUC crystals have a very low fluorine content, usually less than 0.05 %.

The decomposition of the AUC and the subsequent reduction to UO₂ is performed in a fluidized bed furnace. The AUC is scraped off from the rotating filter continuously and transferred pneumatically into the fluidized bed furnace. As fluidizing medium dry steam is used to which hydrogen

is added as reductive component. The transformation into UO_2 is performed at temperatures of about 550° C within a few seconds after entry of the AUC into the furnace. After having fed the whole batch of about 500 kg AUC into the fluidized bed - that requires about 3 hours - the temperature is increased to 650° C in order to lower the fluorine content by pyrohydrolysis to less than 100 ppm.

From the fluidized bed the hot powder is transferred into a cooling drum in which the powder is stabilized for the further processing by partial oxidation with air. Then the powder is cooled to room temperature.

The cold UO₂-powder is transferred batchwise from the cooling drum into the homogenizer. The homogenizer can blend 8 charges which amount to about 2 tons of powder. By that method relatively large quantities of uniform material are obtained which means that testing costs can be kept on a rather low level. Samples of this 2-ton-lot are taken for control of the physical and chemical properties. For example samples are tested for BET area, the bulk density, the O : U-ratio, the fluorine content and, of course, the purity of the powder regarding boron-equivalent by analysing corresponding impurity elements. Finally a performance test is made to determine whether or not the powder can be pressed and sintered to pellets according to the specifications.

The powder from the AUC-process is a free-flowing powder. This is mainly due to the specific kind of precipitation of the ammoniumuranylcarbonate-crystals. The property of free-flowability has many advantages. Among other advantages this powder is able to be transported pneumatically and it can be easily mixed with plutoniumoxid.

All of the before-mentioned equipment is constructed in ever safe geometry. The related dimensions depend on the enrichment degree of the material to be handled. Based

on a batch size of 250 kg UO₂ each of the two 3 % enrichment lines has a 3-shift capacity of about 150 tons a year. The 4 %-line has a 100 tons capacity. The lines are semiautomated and all 3 lines require a total of eight operating personnel per shift. Up to now about 1000 tons of UO₂-powder have been produced by this AUC-process.

Pelletizing

The homogenized UO_2 -powder is pneumatically transferred from the homogenizers of the powder lines into the feed hoppers of the powder presses. In each of these hoppers approx. 200 kg UO_2 -powder can be stored, the amount necessary for one shift operation of the press. From each hopper the powder is transferred via special dosing equipment into the respective shuttle-box by which the powder is uniformly fed into the die cavities of the mechanical press.

Because of its special properties already mentioned before the UO₂-powder can be manufactured directly into pressed green bodies without intermediate production steps like milling, precompaction, granulating and addition of binderlubricants. This lack of binder-lubricant within the powder, however, necessita es lubrication of the die wall. This is done by an automatic lubricating system using common Diesel fuel injection pumps. By this method a certain amount of heavy duty oil is pressed through the hollow lower punch, wiping the punch sidewards, thus providing a thin film of lubricant on the die wall. Because of the small amount of lubricant and because the oil is only on the surface of the green body a special dewaxing furnace is not necessary.

Pressing can be performed with toolings with up to 9 dies in which 9 green bodies per stroke are pressed thus increasing the capacity of the press significantly (picture 4). In order to have only a minimum variation with regard to

the properties and dimensions of the sintered bodies the compacting conditions must be maintained over the whole powder lot for which they were fixed by a performance test.

Here a main factor is the green density for which the weight of the green body can be taken as a measure. Samples are taken for that purpose on a statistical basis. Especially in case of multitooling presses these tests are of importance.

Now the green bodies which are put on molybdenum sheets at the press are loaded in a molybdenum boat. Sintering is performed in a contineous pusher type furnace under hydrogen atmosphere at temperatures between 1600 and 1700° C (picture 5). By variation of powder properties, green density, sintering time and temperature, pellet densities in the range from 9.5 to 10.7 g per cm³ can be achieved with tight density tolerances and with uniform micro- and macrostructure. This is of great importance because more and more pellets of reduced density - that means reduced swelling in the reactor - are used for fuel elements with higher burn-up, frequent load cycles and increased rod power.

In a rough terminology, low pellet densities, for instance smaller than 9.8 g per cm³ are used for fast breeder reactors, medium densities between 9.8 and 10.2 g per cm³ for light water reactors and high densities, for example greater than $1^{\circ}.2$ g per cm³ for heavy water reactors. The possible moisture and gas content of the pellets depends to quite an amount on the density and the structure of the internal voids. Mainly it is the portion of open pores of the entire volume of voids and, therefore, the free surface of the pellets which determines the moisture content. Concerning pellets of low and medium density this portion of open voids can be kept relatively small up to a certain degree by means of special preparation of powder and conditions of sintering.

Water contents, higher than 20 ppm, have already shown detrimental effects in zircaloy fuel rods. The tolerable water and gas content of fuel rods may call for specific equipment and precautions. Considering various possibilities, the entire cost of production should not be overlooked. Thus, it can be profitable to reduce efforts toward lowering the portion of open pores in medium density pellets and instead to dry the pellets before loading. R- and D-work toward reaching sufficiently high tolerances of pellet diameter in the sintered state despite the remarkable shrinkage of about 15 % linear are under way. In order to meet the tight tolerances at the time being the cylindrical surface of the sintered pellets is ground by a centerless grinding machine. By an automatic feeding device the pellets are contineously fed to the grinder. Grinding to the required tolerances, normally \pm 0.01 mm, is performed in one step by a diamond grinding wheel.

Subsequently the pellets are automatically discharged, washed with demineralized water and dried by hot air. On the automatic discharge device the surfaces of the pellets are 100 % compared with standard pellets by the operator of the grinder. He removes defective pellets with chips and cracks. Notwithstanding the tight specifications the percentage of defective pellets is only in the range of up to 3 % of the throughput. The inspected pellets are automatically loaded onto storage trays (picture 6).

The material ground from the pellets, the grinding sludge, is separated from the grinding water in a centrifuge. This sludge is chemically clean. Because of using a diamond wheel it does not contain contaminants from grinding. So it can be directly recycled.

With the exception of chips and cracks all other important properties of the pellets are controlled on a statistical basis. Thus, the dimensions like diameter, height, dishing volume, parallelism of the faces, shoulder, hour-glassing,

are tested but also the density, porosity, the structure and finally the content of moisture and non-condensable gases are determined.

Besides the normal PWR and BWR pellets there can be manufactured more sophisticated forms (picture 7). Especially the hollow pellets could play a role in the future. In this case because of the lubricated die somewhat complicated punches have to be used.

<u>Scrap Recovery</u>

The scraps produced during fabrication of pellets and rods can be recovered by a dry method. These scraps are mainly chipped and cracked green and sintered bodies but also the grinding sludge. They are converted by oxidation into U₃O₈-powder. After sieving this powder is added to the original powder and both are homogenized in the homogenizers of the powder line. This method has two advantages. First, the whole amount of chemically clean scrap can be recycled without any wet chemical treatment and second, this addition of U308 decreases the pellet density. By properly adjusting the percentage of the U308 nearly each of the densities in the range of medium density can be achieved. But this method results in pellets with more or less open pores which must be dried entirely before loading. In case that closed pores are desired adding of a salt is recommended of which decomposition gases are created, forming closed pores. Suited for this method is for instance the AUC.

Chemically contaminated scraps and production tail scraps are dissolved in nitric acid and, after filtration of the obtained uranylnitrate solutions, precipitated as AUC. As the AUC precipitation has an excellent cleaning effect, cleaning of the solution by extraction methods is usually not necessary. In case of heavily contaminated scraps AUC precipitation, redissolving and second precipitation can be applied.

In summary: AUC-conversion leads to free-flowing powder which can easily be processed to UO_2 -pellets or optionally this process allows the flexibility of preparing mixed oxide fuel.



Picture 1

Loading of the UF_6 -container into the evaporation station

Unit for the precipitation of the uranium as ammoniumuranylcarbonate



Filtration of the AUC on a plane rotating vacuum filter; in the background precipitation units with control panel



Pressing of green bodies with a multitool press



Sintering furnaces



Pellets



Picture 7

Various kinds of pellets

FRENCH ACHIEVEMENTS

REGARDING THE CONVERSION OF UF, INTO SINTERABLE UO2

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ABSTRACT

The problem of converting UF₆ to UO₂ of sinterable quality arose rather late in France because the country's first nuclear power stations used natural uranium.

Beginning in 1960 the Commissariat à l'Energie Atomique (CEA) undertook in its own laboratories, and in those of industrial firms collaborating with it under contract, a study programme aimed at simplifying the process used up to that time to produce enriched oxide for experimental reactors from UF_6 .

Two parallel research efforts were launched, one concentrating on a dry process, the other on a wet process.

The present paper summarizes the different stages of this research and describes the patented processes which resulted and were subsequently put into operation on a semi-industrial scale. They are:

- A continuous process used by CEA at the Cadarache site in which the oxide is produced by hydrolysis of UF₆ followed by reducing pyrohydrolysis (in a rotary oven) of the intermediate uranyl fluoride; and
- A continuous wet process used by the Société des Usines Chimiques de Pierrelatte (UCP) which makes it possible to obtain, via intermediate precipitation of ammonium uranate followed by calcination and reduction in a rotary oven, an oxide of sinterable quality in the form of powder or grains.

The paper describes, in addition to the basic technological features of these processes, the principal operating conditions involved and the physico-chemical characteristics and sintering properties of the products obtained.

The advantages and disadvantages of the two processes are discussed, as are the effluent problems associated with them.

In conclusion, their economic aspects are touched on briefly.

RESUME

Historiquement, le problème de la conversion de l'UF6 en UO2 de qualité frittable s'est posé en France assez tardivement puisque les premières centrales nucléaires françaises utilisaient l'uranium naturel.

C'est à partir de 1960 que le Commissariat à l'Energie Atomique (CEA) a entrepris dans ses propres laboratoires, comme dans ceux d'industriels travaillant avec lui sous contrat, un programme d'études destiné à simplifier le procédé utilisé jusqu'alors pour l'obtention, à partir d'UF6, de l'oxyde enrichi destiné aux réacteurs expérimentaux.

Les recherches furent menées parallèlement dans deux directions :

- la Voie Sèche,
- la Voie Humide

L'exposé ci-après résume les étapes successives de ces recherches et décrit les procédés brevetés qu'elles permirent de mettre en oeuvre à l'échelle semi-industrielle, à savoir :

- un procédé continu exploité par le CEA sur le site de Cadarache qui permet d'obtenir l'oxyde par hydrolyse de l'UF6 suivie d'une pyrohydrolyse réductrice en four tournant du Fluorure d'Uranyle obtenu intermédiairement.

- un procédé continu de voie humide exploité par la Société des Usines Chimiques de Pierrelatte (UCP) qui au travers d'une précipitation intermédiaire d'Uranate d'Ammonium permet, après calcination et réduction en four tournant, d'obtenir un oxyde de qualité frittable sous forme de poudre ou de granulés.

Outre les caractéristiques technologiques essentielles des procédés décrits, l'exposé indique les principales conditions opératoires utilisées dans leur mise en oeuvre, ainsi que les gractéristiques physico-chimiques et l'aptitude au frittage des produits obtenus.

Mention est faite également des avantages et des inconvénients respectifs de chacun des procédés décrits ainsi que des problèmes d'effluents liés aux schémas de traitement retenus.

Enfin, l'aspect économique de ces procédés est brièvement abordé pour conclure.

1 - INTRODUCTION -

The present uranium enrichment processes result in the production of UF₆ which is used as the basic material for the production of reactor fuel elements using enriched uranium. However, the problem of the conversion of UF₆ into sinterable UO₂ appeared rather late in FRANCE

since the first reactors used natural uranium. Thus, the first preparation of enriched UO_2 for test reactors included the following steps :

- conversion through pyrohydrolysis of UF₆ into $U_{3}O_{8}$
- dissolution of this oxide in nitric acid
- conversion of the uranyl nitrate into UO₂, a process which was then well-known and industrially achieved.

However, to simplify this process, some studies were undertaken as soon as 1960 by the Commissariat à l'Energie Atomique (C.E.A.). either in its own laboratories or through some contractors. Parallel research works were carried out on two processes :

1.1. "Dry process"

The known processes consist in hydrolysing UF₆ by steam to obtain UO₂F₂ which is then subjected to a pyrohydrolysis under reducing atmosphere. The first studies were made by the UGINE company in its Centre de Recherches de LYON. They resulted in showing that it was possible to obtain some UO₂ by performing the two operations in two successive fluid beds. This process was then developed to a semi-industrial scale by the Ateliers de Traitement d'Uranium Enrichi du C.E.A., in CADARACHE.

Some technological difficulties at the level of the reducing pyrohydrolysis did not allow the continuous operating of the installation, thus impairing the quality of the resulting UO₂ which was then poorly "sinterable according to the process commonly used in FRANCE. Therefore, such method was left. But the research works were carried on in CADARACHE and resulted recently in the present process described further down (paragraph 2.).

1.2. "Wet process"

The various processes include the following successive steps :

- ammonia precipitation providing ammonium uranate
- filtration
- calcination and pyrohydrolysis to provide a defluorinated oxide
- reduction to UO2 by hydrogen

Further to some preliminary works carried out in the Centre de Recherches de LYON of the UGINE Company, the C.E.A. carried on in its laboratories of GRENOBLE some research works which resulted in the process flowsheet hereunder, drawn directly from the one applied to obtain UO₂ from uranyl nitrate :

- two-stage precipitation by ammonium hydroxide
- filtration of the precipitate
- calcination pyrohydrolysis in a rotary furnace providing the U₃O_A oxide
- reduction in a rotary furnace by the hydrogen proceeding from ammonia cracking, providing UO₂ whose characteristics can be adjusted by modifying the parameters involved in the various stages.

This process for which the C.E.A. was granted a patent was adopted by the Société des Usines Chimiques de PIERRELATTE which perfected it repidly. It was then looked for its optimization to improve its profitability. It could in some ways be simplified without impairing the properties of the final product :

- single-stage precipitation by ammonia
- single rotary furnace for the calcination, pyrohydrolysis and reduction operations.

The present process is described further on in this report (paragraph 3.).

2 - "DRY" PROCESS -

This patented process operated by the Ateliers de Traitement d'Uranium Enrichi du C.E.A. in CADARACHE allows the continuous operation the two reactions of hydrolysis of UF₆ and pyrohydrolysis of the resulting UO_2F_2 .

2.1. Operating flowsheet

The process includes the following operations :

2.1.1. Hydrolysis of UF₆

The reaction occurs according to the following equation : $UF_6 + 2H_20 \longrightarrow UO_2F_2 + 4HF$ $\Delta H = -27 \text{ Kcal/mole}$

It is carried out in a vertical static reactor with outside heating through a muffle casing allowing the adjustment of the reaction temperature. The gaseous UF₆ is fed at the higher part. The gases proceeding from the pyrohydrolysis reacotr and composed essentially of steam and nitrogen with hydrogen and hydrofluoric acid are injected into the lower part after having possibly been completed by additional steam or nitrogen.

The UO_2F_2 thus formed settles at the bottom of the reactor from where it is immediately removed by a screw to prevent it from being contacted with the gases and particularly UF_c .

The characteristics of the UO_2F_2 obtained and, further, those of the final UO_2 can be changed by acting on the operating parameters (reaction temperature, flowrates of the various fluids).

Thus the specific area of the UD₂F₂ can be increased by increasing the relative steam flowrate or raising the reaction temperature which however must remain lower than 650°C and commonly kept between 158 and 300°C.

2.1.2. Reducing pyrohydrolysis of U0₂F₂

The reaction is : $UO_2F_2 + H_2 + H_2O \longrightarrow UO_2 + 2 HF + H_2O \bigtriangleup H = +3.4 Kcal/moli$

The extraction screw of the first reactor feeds directly the pyrohydrolysis rotary furnace the temperature of which can be adjusted 'through a heating muffle. A mixture of steam, hydrogen and nitrogen - these last two gases proceeding from ammonia cracking - is run counter-currently to the UO_2F_2 stream. The reaction gases are sent to the hydrolysis reactor. The characteristics of the UO_2 obtained at the furnace outlet are influenced indeed by those of the former UO_2F_2 but also by the various parameters as adjusted for the operation, namely :

- reaction temperature
- residence time
- reagents flowrates

For example, the diameter of the particles can be decreased or the specific area of UO_2 increased by reducing the residence time inside the reactor or lowering the reaction temperature which however should always exceed 550°C.

2.1.3. Reception of UO2

Considering the characteristics of sinterability which are presently looked for, it is not necessary to stabilize the UC₂. The latter reacts little enough with air to be handled without risking a sharp reoxidation. For some special operations (for example thorough crushing), however, it is recommended to operate under inert atmosphere.

2.1.4. Treatment of effluents

The only effluents consist of the gases proceeding from the hydrolysis reactor and are composed of hydrogen, nitrogen, steam and hydrofluoric acid. They are removed from the reactor through sintered-metal filters to avoid any entrainment of uraniferous products. Then, they are run through a limestone bed which absorbs the HF as solid calcium fluoride. The outlet gases can then be safely vented to atmosphere after running through a safety-water scrubber.

2.2. Characteristics of the UD2 obtained

2.2.1. Physical characteristics

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The main characteristics of the UO_2 which can be obtained range as follows :

-	B.E.T. specific area	1 to 4 m2/g
-	mean particle diameter (BLAINE's method)	0.4 to 1 j
-	compressed density	1 to 2.5

2.2.2. Chemical characteristics

- uranium content	≥ 87.7 %
- O/U ratio	< 2.10
- moisture content	< 0.1 %
- fluorine content	≤ 10. 10 ⁻⁶ U
- carbon content	≤ 50. 10 ⁻⁶ U
- nitrogen content	≤ 50. 10 ⁻⁶ U
- thermal neutron capture section	< 1. 10 ⁻⁶ U
	boron equivalent

2.2.3. Fitness for sintering

The little flowability of the UO₂ powder obtained up to now requires to apply the "double cycle" method. Under normal sintering conditions, namely :

- temperature	≼ 1650°C
- bulk density	€ ⁶
- sintering time	≤ 2.30 hrs

it is possible to obtain sintered pellets with densities ranging between 90 and 98 % of the theoretical density with an excellent reproducibility.

2.3. Advantages and drawbacks of the process

2.3.1. Advantages :

- Very simple technology allowing a very easy operating of the installation and offering a great fiability.
- Production of effluents as solids only eliminating practically the problems of environment.
- Reagent saving Hydrogen only is expensive.
- Problems of criticity simplified by the exclusive use of solid or gaseous products with use of water only in vapour phase and small quantity. This allows more precisely to design highcapacity plants.

- The process has not, up to now, made it possible to obtain a "flowable powder" of UO, allowing a single-cycle shaping.
- The installation cannot be used for processing the uranyl nitrate polutions. The re-processing of the wastes more particularly involves their previous conversion into pure UF₆.

3 - "WET" PROCESS

Operated by the Société des Usines Chimiques de PIERRELATTE, this process allows the continuous development of the various reactions required to obtain UO_2 .

3.1. Operating flowsheet

The successive operations are :

3.1.1. Ammonium uranate precipitation

The precipitation of ammonium uranate involves many rather complex reactions. However, for an overall representation, the main reaction can be illustrated by the following equation :

$$2 \text{ UF}_{6} + 14 \text{ NH}_{4}\text{OH} \longrightarrow (\text{NH}_{4})_{2}\text{U}_{2}\text{O}_{7} + 12 \text{ NH}_{4}\text{F} + 7 \text{ H}_{2}\text{O}_{7}$$

It is achieved inside an agitated reactor provided with external heating to control its temperature and fed with water, UF_6 and ammonia. The characteristics of the precipitate obtained and, further, of the final UO_2 can be changed by changing the reaction temperature and the fluid flowrates.

Thus, the uranate specific area can be increased by increasing the relative ammonia flowrate, the latter, on the other hand, being always higher than three times the stoichiometric quantity to achieve the total precipitation of uranium. The reaction temperature may also be raised. However, the use of a temperature exceeding 90°C is not an advantage considering the ammonia losses through evaporation.

3.1.2. Uranate filtration

The precipitate is filtered and centrifuged using a vacuum rotary filter.

3.1.3. Calcination - Defluorination - Reduction

4

The thixotropic feature of the filtration cake allows it, further to a mere mixing, to be conveyed by pump into the single reactor where these three operations are carried out. This reactor is a rotary kiln heated by electric muffles. The necessary reagents, steam for defluorination, hydrogen for reduction, are fed counter-currently to uranium. By controlling the kiln temperature, the residence time of uranium inside the reactor and the fluids flowrates, it is possible to modify the characteristics of UO₂ obtained from a same uranate.

For example, the specific area of UO₂ can be decreased by increasing the residence time of the product inside the kiln or raising the reaction temperature. The latter must not be lower than 500°C to allow the complete reactions and is kept under 800°C to take into account the physical behaviour of the equipment.

3.1.4. Stabilization

The UO₂ at the kiln outlet is practically stoichiometric and very reactive in the presence of air. It is therefore necessary to stabilize it. This is achieved continuously at low temperature, by a controlled addition of oxygen in the counter-current circulation of reduction gases fed to the kiln, rated to reach the desired re-oxidation level. This operation has no effect on the physical features of UO₂.

3.1.5. Calibration of UD,

Two granulometric qualities of UO_2 can be obtained with this process :

- a quality called "direct granule" the flowability of which is sufficient to allow a single-cycle pelletizing. The calibrating consists of a 600 µ mesh screening followed by a reduction of the retained particles by forced screening.

- a quality called "powder" the use of which requires the doublecycle technique. The powder is adjusted to this operation by crushing.
- 3.1.6. Treatment of process effluents

These effluents include :

- the uranate filtration mother-liquors
- the gases proceeding from the calcination-defluorinationreduction kiln. They are extracted through sintered-metal filters to avoid uranium entrainments. The liquid ring of the vacuum extraction pump acts as scrubber for the absorption of the hydrofluoric acid, particularly before the vent gases are rejected to the atmosphere.

These effluents are then mixed. Their uranium content is controlled continuously. The treatment consists of :

- a lime precipitation of the fluorine ions and uranium traces
- decanting in pits where calcium fluoride containing traces of uranium is separated
- discharge after control of the clear water.

3.2. Characteristics of the UO, obtained

3.2.1. Physical characteristics

By control of the various operating parameters, it is possible to alter the main characteristics of the processed UO₂ within the following ranges :

- B.E.T. specific area	3 to 4.5 m2/g
- mean particle diameter (BLAINE's ma	ethod) 0.4 to 1 μ
- bulk density	> 1.5
- compressed density	≥ 2.5
- particle size :	·
direct granule 🔰 600 д	≤ 10 %
²⁰⁰ بر ²⁰⁰	≤ 15 %
pewder < ⁷⁴ H	≥ 90 %

3.2.2. Chemical characteristics

- U content	≥ 87.7 %
- O/U ratio	2.07 ± 0.05
- moisture content	< 0.1 %
- fluorine content	< 50. 10 ⁻⁶ ປ
- carbon content	< 50. 10 ⁻⁶ U
- nitrogen content	< 100. 10 ⁻⁶ U
- thermal neutron capture section	< 1. 10 ⁻⁶ U
	boron equivalent

3.2.3. Fitness for sintering

Whatever the granulometric quality of the UO₂ obtained with this process (direct granule or powder), the sintering results are satisfactory. Under normal sintering conditions, it is possible to obtain sintered pellets with densities ranging between 90 and 96 % of the theoretical density with an excellent reproducibility.

3.3. Advantages and drawbacks of this process

3.3.1. Advantages :

- A very simplified technology compared to the initial process allowing an easy operation and ensuring a great fiability
- many possible adjustments providing a great operating flexibility
- possibility of obtaining the "direct granule" quality allowing the simplification of pelletizing operations.

 possibility of treating the uranyl nitrate solutions in the same equipment, thus allowing the easy recycling of final operation recovered residues.

3.3.2. Drawbacks

- Important quantity of liquid ammoniacal effluents which may lead to require a special treatment
- limited capacity of the units resulting from the problems of criticity caused by the presence of great amounts of water
- great expenses of reagent The complete precipitation of uranium involves the use of approximately three times the quantity of stoichiometric amminia
- consumption of calories to remove the water entrained by the uranate precipitate.

4 - ECONOMIC ASPECT OF THE PROCESSES

As mentioned formerly, the problems of the conversion of UF_6 into sinterable UO_2 appeared only recently in FRANCE and, consequently, the production has not been operated long enough to provide definite data on the economics.

Nevertheless, the experience acquired by now has made it possible to ascertain that for yearly productions ranging around 100 m.tons of UO₂, UF₆ could be converted into UO₂ at a competitive price compared with the ones ruling commonly in the world.

5 - CONCLUSION

Both processes presently applied in FRANCE allow a wide range of specifications to be covered and the requirements of the sintering units to be met. Their application is recent and their optimization is being carried on which allows new improvements to be expected.

IMPROVEMENTS OF FUEL PEI LETIZING TECHNIQUES IN BELGIUM

A.J. Flipot.

Abstract

The manufacture of UO₂ pellets has been gradually simplified in order to reach a three-step fabrication scheme : pelletizing, sintering and control. The main improvements developed with the aid of a rotary press are described. Direct lubrication of the pelletizing tools is a first improvement : it is achieved by com pacting a dry lubricant mixed with a carrier. The second step deals with direct pelletizing of a non free-flowing powder thanks to an elaborated feeder. Finally, it has been tried to manufacture sintered cylindrical pellets in order to avoid centreless grinding. Three approaches are discussed. The key-factors are either the shape of the sinterability curve of UO₂ either the uniformity of the green density or the type of compaction method and the design of the die.

1. Introductio.

The main activity of the Joint CEN-BELGONUCLEAIRE Pu Group is related to the study of fabrication and behaviour of plutonium fuels. The best results obtained in these laboratories are then transmitted to the pilot plant and to the industrial fabrication line of BELGONUCLEAIRE which has to solve the further problems resulting from the increase of the fabrication scale. The studies, first devoted to plutonium recycling in thermal reactors have been extended gradually to fast reactor fuels. Fortunately, both objectives contribute to the improvement of fabrication techniques because the technological problems are basically identical. Moreover, the manufacture of UO₂-PuO₂ fuels is, in many respects, similar to this of pure UO₂ fuels so that UO₂ and UO₂-PuO₂ pelletizing techniques progress si multaneously and take each advantage of the improvements of the other one. This explains why the Joint CEN-BELGONUCLEAIRE Pu Croup has been developing for several years advanced fabrication procedures using rotary presses and direct solid lubrication of the dies.

Thanks to experiments carried out on UO2 since 1966, the earlier wetpreparation technique was abandoned progressively to the benefit of a dry pelletizing one. This method was then studied on UO2-PuO2 in 1969 and has been successfully applied since 1970 in the pilot plant of BELGONUCLE-AIRE. This was the first step to the simpliest pellet fabrication route which can be summarized in three words : pelletizing - sintering - control. This objective is no longer a dream for the manufacturers of UO₂ pellets as it has already been reached on a small scale.

The present paper comments the successive steps which have been overcome and describes some fabrication methods.

2. Progress in pellctizing technique.

The basic dry pelletizing technique is summarized by Fig. 1 : it is, in fact, constituted by two pelletizing cycles.

The first approach in improving and simplifying the fabrication route is linked to lubrication process. Homogeneous incorporation of dry lubricants in UO2 or UO2-PuO2 fine powders gives no satisfactory results. For a UO2 compact containing 0.3 wt. % uniformly distributed zinc behenate, the amount of lubricant squeezed on to the die wall during prossing is insufficient to ensure good lubrication. It seems necessary to increase the lubricant content up to 1 wt. % but this will decrease the fired density of the pellets. In consequence of this poor lubrication during precompaction, the wear of the tools will be fast. The green compacts are then granulated and mixed with 0.3 wt. % zinc behenate before pelletizing. During this second compaction, lubrication is effective because zinc behanate is concentrated on the surface of each granule and the quantity directly in contact with the die wall is thus increased. Unfortunately, these salts of fatty acids con glomerate easily and it is difficult to achieve a uniform blending on an industrial scale. Moreover, these additives must be removed before sintering to protect the furnace and thus require a dewaxing step. A direct lubrication method being effective and versatile would therefore improve the results and eliminate the two mixing steps and the dewaxing one.

Direct pelletizing of the raw materials constitutes the second simplification of the fabrication route. Besides the use of a direct lubrication technique, pelletizing a non-granulated powder requires to find the possibility to achieve a complete and reproducible filling of the die. Both objectives have been achieved with the aid of a rotary press which, in addition, gives a much higher output than conventional presses. These studies have been performed on UO₂ since 1968 and will be extended to $UO_2 - PuO_2$ fuels before the end of the year.

The third improvement which is the most difficult to succeed, is the manufacturing of accurate sintered pellets in order to avoid the grinding step. This will represent an important progress because grinding is expensive for many reasons. It damages some pellets and produces powder scraps which cannot be directly repelletized. Moreover, all ground pellets must be dried before filling the canning tubes. A new compaction method has been experimented on UO2 since the end of 1971. The results are very promising. The simplified fabrication route is illustrated by Fig. 2.

3. Rotary press with lubricated dies.

The machine is a 20-ton double-sided rotary-press as shown schematically by Fig. 3. It is formed by two completely independent parts and has

[&]amp; Press P 3 ~ Fette (Germany).

thus two sets of fill, compaction and ejection stations. One set is used for applying the lubricant whereas the other set yields the fuel pellcts. The cycle can be understood by following the various positions taken by dien^ol, for instance. The die enters first in the zone reserved for lubrication and then, due to the rotation of the table, reaches the other zone. It passes under the stationary feeder and takes UO2 powder which is then compacted and ejected. At this moment, this die has made one complete revolution and a new cycle starts. All tools are thus lubricated alternately and are working under normal operating conditions. Lubrication of the compacting tools is achieved by pelletizing a salt of fatty acids blended with an inert carrier (1, 2). The carrier is essential to obtain a good flow-rate and to guarantee that the lubricant tende to recover its initial shape after being pressed.

The first composite lubricant selected consisted of small polystyrene balls (0.2 to 1 mm diameter) covered with 3 wt. % zinc behenate (2). This mixture flows easily and lubricates as well as pure lubricant (3). More over, as the balls are somewhat elastic, the compact does not remain a pellet after ejection but returns spontaneously into elementary balls. These balls may therefore be directly reintroduced into the feeding boot of the press, allowing a closed cycle to be operated.

Though successful, this composite lubricant may be replaced by cork granules mixed with 10 wt. % zinc stearate in order to avoid electrostatic charges and the associated troubles. This feature is essential when pelletizing in a dry atmosphere as required, for instance, for carbidefuels. Moreover, the effectiveness of this latter composite lubricant seems rather insensitive to the compaction conditions. Zinc behenate was replaced by zinc stearate which is as effective and cheaper. It has been observed that lubrication efficiency of stearates depends on their physical properties and their impurity level (4). Lubrication is best when the tap density of the metallic soap is $\langle 200 \text{ g/l}$ and when the fatty acid content is kept as low as possible : these parameters are more important when the lubricant is admixed to the powder than when direct lubrication is used. In this case, a zinc stearate with a tap density of 400 g/l still gives excellent results.

4. Powder flow-rate,

Powder flow-rate can be evaluated according to standard specifications (5) using a Hall flow-meter. Similar results are obtained with the commercial equipment Erweka GDT which is more sophisticated and easier to operate in a glove-box. As shown by Fig. 4, this flow-meter is equipped with a control device. A well-determined powder quantity is poured into a funnel which is separated from an outlet tube by an electromagnetic valve. The flow-time is measured by an incorporated impulse counter which starts when opening the separation valve and stops when all powder has passed the light beam opposite to a photo-electric cell.

a) Free flowing powders.

This type of powder which flows rapidly and in a reproducible manner can be illustrated by the small spheres produced by direct conversion of UI'6. This method has been developed for some years by Métallurgie Hoboken and Centre d'Etude de l'Energie Nucléaire, Mol. It is now applied on a larger scale by Métallurgie et Mécanique Nucléaires in Dessel (Belgium). Some characteristics of free flowing UC2 pcwders available in Europe are summarized in Fig. 5. Such powders fill regularly the dies of conventional and rotary pelletizing preases. Most of these powders present no pelletizing problems if their particles are neither too hard nor too fine. Combination of hardness and fineness of the particles may lead to accelerated wear or even seizing of the tools.

For free flowing powders, the use of a rotary press still remains advantageous, especially when a large quantity of pellets must be manufactured. A fast conventional press produces a maximum of 30 pellets/min. whereas a low rotary press yields as many as 150 pellets/min. Fig. 6 shows how free flowing UO₂ is pelletized on a double-sided rotary-press using a dry lubrication technique. The foeding shoe is simple and effective. The powder moves above the dies and the excess turns with the table and then enters again the feeding shoe. The quantity of this powder in excess regulates the quantity of fresh powder admixed on the press so that the filling of the dies remains constant. On the other hand, an adequate separation exists between the two parts of the rotary press in order to avoid any mixing of fuel and lubricant.

Pellets manufactured with spherical UO2 particles prove to be very reproducible as the scattering in green density is below ± 0.5 %. Another free flowing UO2 constituted by coarse regular granules instead of small spheres can also give good pellets. It has been noticed that density reproducibility is linked to powder granulometry. A too coarse powder containing, for instance, 50 % of particles ranging between 0.4 and 1 mm fill the dies rapidly and completely but not regularly so that the scattering in weight and green density of 10 mm diameter pellets is as high as ± 2.4 %. On the contrary, this scattering does not exceed ± 0.7 % when powder has been previously granulated through 0.4 or 0.2 mm openings. Under these circumstances, both types of presses guarantee the

same quality of pellets.

) Sticking powders.

In opposition to free flowing particles the contour of which is regular, there are powders with a jagged outline (Fig. 7). These powders, often very fine, conglomerate easily and form bridges between each other so that they do not flow at all, even when vibrations are applied. Such a raw material is difficult to pelletize within close margins and must be precompacted and granulated. Precompaction can be per -formed either with the aid of a powerful hydraulic press, capable of making large compacts or by a rotary press equipped with an elaborated feeder.

Powders are sometimes so sticking that it seems almost impossible to manufacture free flowing granules. Batch number NL 36 (- 70 mesh) (Fig. 7) illustrates this exceptional case. The most recent filling techniques prove that, at the present time, thus UO_2 quality cannot be pelletized accurately in a 12 mm diameter die. The best results obtained with the aid of a rotary press show a \pm 10 % scattering in the green density.

After granulation on a 0.4 mm grid, the product does not flow. Unlikely as it sounds, seven out of ten flowing tests have pointed out that these granules form bridges and obstruct the funnel of the flow-meter. The product is not convenient for feeding conventional mechanical presses. However, it can be pelletized with a rotary press. Pellets manufactured with the aid of the feeder designed for granules or free flowing material are acceptable as the scattering in their green density is lower than ± 1.2 %.

c) Non free-flowing powders.

This third category is constituted by powders the characteristics of which lie half-way between the first two. These raw materials flow neither rapidly nor uniformly but their tendency to sticking is not very strong. Common ceramic grade UO_ powder manufactured first by Métallurgie Hoboken and presently by Métallurgie et Mécanique Nucléaires in Belgium is representative for this category (Fig. 7). Flowing tests give unreproducible results because 20 to 30 % of the tests show powder agglomeration in the flow-meter.

Powders of intermediate flow-rate have been simulated by mixing free flowing and non free-flowing UO2 qualities. The resulting flowrate depends on the properties of each constituent such as shape and granulometry of the particles. The evolution of the flowing charac teristics as a function of the composition of such mixtures is represented by Fig. 8. On the other hand, each of these composite batches was pelletized with a conventional mechanical press in order to evaluate the importance of the results of the flowing tests on the reproducibility of the green pellets. Fig. 9 gives the maximum weight and density variation measured on 250 pellets of each batch. If a 2.5 % scattering (or ± 1.25 %) is admitted, Fig. 8 and 9 indicate that a UO2 called free flowing must fulfil two criteria :

- flowing time/100 cm ³	:	
- A 1	:	\leq 5 seconds (usually $<$ 3 seconds).

These figures are valid only for defined working conditions and depend, for instance, on the type of the feeding shoe, the output of the press, the diameter of the die, the height of the pellets, etc... Beyond these two limits, powders can be called "non free-flowing" because their flow rate becomes rapidly unmeasurable and the scattering in weight and green density increases exponentially. Nevertheless, this scattering may be reduced by improving the performance of the feeder. Unfortunately, the flowing test appears to be poor and in sufficiently sensitive to classify the non free-flowing powders which remain the most currently available ones.

5. Pelletizing of non free-flowing powders.

A rotary press uses a stationary feeder which can be longer and more sophisticated than that of a conventional press. Moreover, the filling time is rather long. These both features are beneficial to the regularity of the pellets when powder flow-rate is insufficient.

A special feeding shoe with two rotating parts^{ik} (6) directing the powder into the die cavities, gives good results with Belgian ceramic grade UO₂. Fig. 10 displays typical histograms of the density of green and sintered pellets. These results represent 100 kg pellets with a diameter and height of 12 mm. Ten percent of the batch were controlled as a representative sample. These curves demonstrate that the reproducibility of the pellets

A Fill-O-Matic shoe from Fette-Hamburg (Germany).

is good. The scattering in green density reaches 0.13 or ± 1.4 %. After a two-hour sintering at 1650°C in an industrial continuous furnace the dispersion of the sintered density is 0.19 or ± 0.93 %. Fig. 11 shows the same results as a function of the fabrication sequence. As a rule, the densities are distributed at random round the mean value. This feature proves that the filling of the dies is uniform as a function of time. In turn, this supposes that the filling of the feeder itself is carefully performed and that the level of the powder is always kept between two predetermined limits which depend on the diameter of the pellets and on the quality of the powder. Fortunately, the effect of this limiting factor can be reduced by using a hydraulic compensation which tends to keep the compaction force constant and independent of small filling variations. This improvement should combine the advantages of mechanical and hydraulic presses : high output and good reproducibility of the green density.

The compaction force can be measured by strain gauges mounted on the compression rollers and compared to the nominal required value with the aid of an electronic device. In this way, it is possible to classify the pellets at the ejection station of the press and to eliminate the rejected pellets automatically. At the same time, the filling height can be adjusted by means of a motor operated electronically.

6. Manufacturing of accurate sintered pellets.

The above-mentioned improvements have solved the problem of manufacturing reproducible UO₂ pellets with non free-flowing powders. This technique now allows one to pelletize correctly most of the commercial powders but does not solve the distortion problem of the sintered pellets which rcsults from a heterogeneous compaction. In most cases, cylindrical green pellets turn into diabolo-shaped sintered pellets and centreless grinding becomes necessary to obtain the exact dimensions required.

The third objective was therefore the manufacture of sintered cylindrical pellets in order to avoid the expensive centreless grinding and to lower the fabrication costs.

There are three possible approaches in resolving this problem (7):

- to control the sinterability curve of the raw material ;
- to improve the uniformity of the green density by minimizing the frictional forces ;
- to cope with the distortion of the pellets by adapting the compaction technique and the shape of the die.
 - a) Effect of the sinterability curve on the regularity of the sintered pellets.

It can be demonstrated (7) that, for cylindrical pellets :

$$\frac{d_{g}}{d_{g}} = k \left(\frac{D_{g}}{D_{s}}\right)^{3} \qquad [1]$$

where D_g and d_g respectively stand for the diameter and the density of a green pellet while D_g and d_g represent the same parameters measured after sintering. If the sintered density is directly proportional to the green density $(d_s = ad_g)$, relation (1/2) becomes :

$$\left(\frac{D_g}{D_g}\right)^3 = \frac{a}{k} = b$$

This equation shows that, in this particular case, the sintered diameter is independent of the green density and is determined only by the diameter of the die. Consequently, a cylindrical green pellet will remain cylindrical after sintering whatever be the compaction characteristics. This result must mainly interest the powder manufacturer because he controls the characteristics of the raw materials. Usually, the pellet manufacturer does not process the powder to modify its green sintered density relationship but may sometimes choose its quality.

b) Factors affecting the uniformity of the green density.

As shown by relation $\lfloor 1 \rfloor$, the sintered diameter is affected by any change in green density. As a rule, the sintered diameter decreases or increases with the green density and this is responsible for the formation of the diabolo-shape in pellets compacted with double-acting presses. The extent of this diabolo is linked to the frictional forces during compaction. It has been pointed out that friction and diabolo size depend on the quality of the UO₂ powder and on the fabrication conditions.

The diabolo size (ΔD_s) can be expressed as a product of two factors:

$$\Delta D_{s} = \frac{\Delta D_{s}}{\Delta d_{g}} \Delta d_{g}$$

The first factor can be considered as the derivative of the sintered pellet diameter with respect to its green density and depends on the sinterability of the powder. Experience shows that this factor is usually a function of the first degree.

On the other hand, the second factor represents the density gradient existing in the green pellet. The diabolo size is therefore directly proportional to the green density of the pellet when the density gra dient is kept constant.

The density gradient results from the frictional forces and is thus affected by powder roughness, lubrication conditions and die surface. Table I illustrates the influence of these factors on the diabolo size and demonstrates that the best results are not always achieved with the same UO2 powder. Batches n° UO 70 and NL 30 FF are, for instance, constituted by free flowing particles but are different with respect to sinterability and surface roughness. The fine spherules of the UO 70 quality are macroscopically regular but an examination with a scanning electron microscope reveals that they are formed by microneedles. On the other hand, the regular blocks of the NL 30 FT UO2 quality are smooth and sinter more easily than the former quality. As expected, these features give a small diabolo when the experiments are carried out in a tungsten carbide die. Nevertheless, these results are not confirmed when working in a steel die having the same surface roughness. This is valid for new tools because a steel die is not as wear-resistant as a carbide die and deteriorates more rapidly. This increase in surface roughness is expected to decrease the regularity of the pellets. The role played by the die in the formation of the diabolo has not been explained yet but should be linked to the lubrication efficiency.

In this respect, a liquid lubricant such as LPS, a commercial ester of saturated fatty acids, is absolutely not suitable for a tungsten carbide die because it produces a big diabolo or even cracks from a low green density. On the contrary, this lubricant is effective for the pelletizing of some UO2 qualities in a steel die. The working conditions should therefore be determined according to the UO2 quality and conversely.

As seen in Table I, the combination of all favourable factors allows one to keep the diabolo of 12 mm pellets below 45 µm. This latter can reach as much as 92 µm when lubricant and die are not adapted 1 to the UO2 quality. Unfortunately, it is difficult, in practice, to control these parameters because the supply of the raw material is often imposed by non-technical requirements and a fabrication line is not sufficiently versatile to adapt the lubrication conditions to the characteristics of the UO2. As a rule, an important production supposes a rotary press equipped with tungsten carbide dies and a dry lubrication process whereas conventional presses, more convenient for small and medium fabrications, can be operated with a steel die lubricated by a liquid lubricant.

c) Choice of the compaction technique.

As the first two approaches are somewhat hypothetical because the key-parameters are not fully controlled by the pellet manufacturer, it has been tried to cope with the problem while accepting the quaity of the UO₂ powder supplied and the lubrication process adapted to the existing press.

In fact, a cylindrival green pellet compacted with a double-acting press turns into a diabolo-shaped sintered pellet. A better die design has no effect in correcting this distortion because of the complexity of the diabolo shape.

On the contrary, a cylindrical green pellet compacted with a singleacting press becomes conical after sintering and this elementary shape can be compensated for.

Fig. 12 represents the outline of a sintered pellet compacted with a single-acting press in a 15 mm diameter steel die. The diameter decreases rather regularly and the difference between maximum and minimum diameters reaches 68 μ m. It can therefore be expected that a pellet compacted this way in an adapted corical die is cylindrical after sintering.

As the shrinkage during sintering decreases when the green density rises and as the compacted pellet is ejected upwards, it is essential that the lower punch is moving while the upper punch remains stationary (8).

This technique is now being developed on a rotary press which has been slightly modified. The first results are encouraging. As an example, the maximum variation in diameter measured on 15 mm pellets ranges between 16 and 22 μ m. In the same conditions, pellets compacted with a double-acting press show a 60 μ m diabolo. It must be added that any method for manufacturing accurate pellets requires the use of a reproducible and uniform powder. Experience proves that such a quabty is difficult to obtain for two reasons. First of all, specifications are too vague because of lack of know ledge in the mutual effect of each powder characteristic on those of the sintered pellets and then, most of the powder manufacturers do not work according to a continuous process.

7. Conclusions,

Pelletizing techniques are in continuous progress but are already very advanced.

The earlier fabrication process has been simplified in order to follow the requirements of modern manufactures. Moreover, as the increasing demand in UO₂ pellets calls for high output polletizing presses, the performance of a rotary press was studied. The objective was to pelletize directly the commercial UO₂ powder and to obtain accurate sintered pellets so that the fabrication process is reduced to only three steps : pellecizing, sintering and control.

The first step was the development of a direct lubrication method : this has been achieved by pelletizing a dry lubricant mixed with a carrier. The technique has been successfully applied for more than three years on a double-sided rotary press. Half the stations is reserved for the lubrication of the tools whereas the other half yields the production of fuel pellets.

The second improvement achieved with the aid of a rotary press is the direct pelletizing of non free-flowing powders. The filling of the die is carried out by a special feeder equipped with rotating arms. This technique enables one to pelletize most of the commercial UO2 powders. The performance of mechanical rotary presses should still be improved by using a hydraulic compensation and by controlling the pelletizing parameters electronically.

The third step or the manufacture of accurate sintered pellets is being studied. The first results are satisfactory but have to be confirmed with different UO₂ qualities. The success depends on the pelletizing technique and on the quality of the raw material which must be uniform and reproducible.

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TABLE I

Influence of UO2 and lubricant quality on the diabolo size of sintered pellats

Pelletizing conditions	: double - a	cting press			
		diameter	:	12	mm
		height	:	15	mm

a) Direct solid lubrication (zinc stearate).

UO2 batch n°	Tungsten ç	arbide die	Steel die		
	$\frac{diabolo}{d_{g}} = 10.2$	<u>sizefo</u> r d _s = 10.4	diabo <u>le a</u> d _s = 10.2	d _g = 10.4	
	μm	ca y	րա	μm	
UO 70	79	92	53	61	
NL 30 FF	50	61	70	77	
VL 31 FF	66	73	80	92	

b) Direct liquid lubrication (LPS).

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IIOn hatch	Tungsten o	orbide die	Steel die		
n*	<u>diabolo size for</u>		diabolosize for		
	d _s = 10,2	$d_8 = 10.4$	d ₃ = 10.2	$d_{S} = 10.4$	
	μm	μm	11313	μm	
UO 70	cracks	cracks	55	60	
NL 30 FF	68	80	55	60	
VL 31 FF	76	cracks	39	43	





Fig. 2. Simplified fabrication route of UO2 pellets.



Fig.3. Double-sided rotary-press with lubricated dies.





Fig.5. Characteristics of some free-flowing UO₂ powders.

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UO2 mixtures.





Fig.10_ Reproducibility of 12mm diameter pellets (UO2_No.9080, Run_No.R 144).



Fig. 11. Evolution of the density of 12mm-diameter pellets as a function of the fabrication sequence.



Fig.12. Outline of a UO₂ pellet compacted with a single-acting press and sintered 4h at 1640°C.

PERSPECTIVES DE L'ENRICHISSEMENT DANS LE CYCLE DU CONBUSTIBLE :

OPTIMISATION D'UNE USINE DE DIFFUSION GAZEUSE

par

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ABSTRACT

Between the theoretical characteristics of gaseous diffusion cascades, an area that has been largely explored since the first publications of Karl Cohen in 1948, and the economic and industrial characteristics of present gaseous diffusion plants, an area that has received a good deal of attention at recent conferences, there is a less familiar middle ground where the theoretical and practical aspects meet: and it is here that one may hope to find ways of optimizing the enrichment link of the fuel cycle based on enriched uranium.

The paper describes this process of optimization and the prospects which it holds out.

It consists not of a single operation but of a series of operations based on the specific characteristics of the process itself and on what might be called the "economic environment" of the project. The elements of the optimization process relate to the construction and start-up phases as much as to the phase of commercial operation of the plant.

Attention is accordingly given to optimization of the cost of enriched uranium, optimization of the cost of separation work, and optimization of specific investment for a gaseous diffusion plant.

Since the quest for an over-all economic optimum requires numerous iterations, a summary of the mathematical tools developed by the C.E.A. for this purpose is given.

Finally, the prospects for practical application of this optimization work are reviewed.

RESUME

Entre le domaine des caractéristiques théoriques des cascades de diffusion gazeuse qui a été largement exploré depuis les premières publications de Karl COHEN en 1948 et le domaine des caractéristiques économiques et industrielles des usines de diffusion gazeuse qui a été fréquemment évoqué dans des congrès récents, il existe un domaine moins connu où ces divers aspects s'interpénètrent afin de permettre l'optimisation du maillon de l'enrichissement dans le cycle du combustible à uranium enrichi.

C'est la description de cette optimisation et les perspectives qu'elle laisse entrevoir qui feront l'objet de la conférence.

En fait, cette optimisation ne comprend pas une opération unique mais un ensemble d'opérations qui reposent sur les caractèristiques spécifiques du procédé d'une part, et sur ce que l'on peut appeler l'environnement économique du projet. En outre, ces optimisations portent aussi bien sur les phases de construction et de démarrage que sur la phase d'exploitation commerciale des usines.

C'est ainsi que seront successivement passées en revue et commentées : l'optimisation du coût de l'uranium enrichi, l'optimisation du coût du travail de séparation, et l'optimisation de l'investissement spécifique pour une usine de diffusion gazeuse.

Le recherche d'un optimum économique global exigeant de nombreuses itérations, un aperçu sera donné des moyens mathématiques qui ont été mis au point par le C.L.A. à cet effet.

Enfin, seront passées en revue les perspectives d'application que laissent entrevoir ces travaux d'optimisation dans le domaine de l'enrichissement.

I - RAPPEL DU CYCLE DU COMBUSTIBLE A URANIUM ENRICHI, DE SES PRINCIPALES CARACTERISTIQUES ECONOMIQUES, ET DE LA SITUATION PARTICULIERE DE L'ENRICHISSEMENT

Tout outil de production doit, compte tenu des préoccupations de rentabilité maximale, être optimisé. Ceci s'applique en particulier aux différents maillons du cycle du combustible à uranium enrichi qui est schématiquement représenté sur la figure 1. Pour faciliter la compréhension du jeu des optimisations qui sera repris en détail par la suite, le maillon de l'enrichissement qui figure dans cette chaîne et qui se rapporte à une usine de diffusion gazeuse est ramifié vers la droite de la figure, afin de présenter la décomposition de la valeur ajoutée correspondante.

Ainsi, il apparaît que le coût unitaire CP de l'uranium enrichi destiné aux réacteurs résulte de l'addition de deux termes qui représentent, d'une part une dépense en UF6 naturel, d'autre part une dépense en travail de séparation. C'est ce que traduit la relation :

$$\mathbf{C}_{\mathbf{P}} = \frac{\mathbf{F}}{\mathbf{P}} \mathbf{C}_{\mathbf{F}} + \frac{\Delta \mathbf{U}}{\mathbf{P}} \mathbf{C}_{\Delta}$$

dans laquelle F/p et CF sont respectivement la quantité et le coût de l'uranium naturel consomné et $\Delta U/p$ et CA la quantité et le coût du travail de séparation dépensé. La flexibilité de fonctionnement de l'usine d'enrichissement prête à une optimisation du coût du produit enrichi, optimisation qui dépend elle-même de l'enrichissement désiré.

Le coût de l'UF6 naturel CF dépend essentiellement du marché, étant donné que les frais de transport sont relativement négligeables.

Le coût du travail de séparation C_{Δ} est un compromis classique entre frais de réalisation et frais d'exploitation de l'usine de séparation isotopique. Dans le cas de la diffusion gazeuse, il faut distinguer parmi les frais d'exploitation la dépense en énergie qui résulte du produit de la consommation spécifique d'énergie W_{Δ} par la tarification de cette énergie t. D'où la relation :

 $C_{\Delta} = I_{\Delta} \cdot \mathcal{C} + W_{\Delta} \cdot t + autres frais d'exploitation$

Il est à noter que parmi ces différents termes, le taux d'intérêt et par conséquent l'annuité de remboursement γ , la tarification de l'énergie t, et dans une certaine mesure aussi le montant de l'investissement à consentir $I/\Delta U$ sont des paramètres qui dépendent de l'environnement économique du projet. Par contre, la consommation spécifique en énergie $W/\Delta U$, les frais d'exploitation et l'investissement $I/\Delta U$ dépendent on grande partie de la qualité de la tecnnique mise en oeuvre.

Nous verrons par la suite que I/AU est sujet à un effet de taille important et donne lieu à plusieurs optimisations intermédiaires. Nous verrons aussi que les frais d'exploitation de l'usine hors énergie sont pratiquement indépendants de sa taille.

De manière à situer l'importance relative des différentes dépenses qui sont engagées dans le cycle, des valeurs indicatives sont données qui permettent de constituer le diagramme de la figure 2. Bien entendu, on ne saurait trop insister sur le caractère très approximatif de cette décomposition puisqu'il faudrait en toute rigueur préciser le type de réacteur considéré, les hypothèses techniques et économiques prises en compte, la méthode utilisée, etc.. Mais du moins, les ordres de grandeur qui sont ainsi donnés aideront à situer les optimisations qui vont maintenant être examinées plus en détail.

II - OPTIMISATIONS MISES EN OEUVRE DANS LA POURSUITE D'UN PROJET

D'ENRICHISSEMENT

Les problèmes d'optimisation d'une usine de diffusion gazeuse interviennent à plusieurs stades.

Tout d'abord, lors de sa conception afin d'adapter l'usine au marché et au contexte technico-économique. Ensuite, lors de la construction afin d'en effectuer le démarrage dans les meilleures conditions. Enfin, au cours de son exploitation afin de l'adapter à l'évolution du marché tant en quantité qu'en enrichissement, ainsi que d'y intégrer les améliorations technologiques obtenues grâce aux programmes de recherche et développement.

Nous allons passer successivement en revue ces différents aspects.

II.1 - CONCEPTION DU PROJET : SES CARACTERISTIQUES PHYSIQUES ET SON ENVIRONNEMENT ECONOMIQUE

Dans un projet d'une usine d'enrichissement par diffusion gazeuse on commence par la définition d'un marché qui s'étend en principe sur la durée d'exploitation de l'usine. Dans le contexte actuel, on peut penser qu'une usine de diffusion gazeuse aura une durée d'exploitation voisine de celle des réacteurs nucléaires qu'elle dessert. Ceci permet donc de s'affranchir de considérations sur l'évolution des filières au cours du temps, étant entendu qu'une usine d'enrichissement peut être associée à un certain nombre de réacteurs qu'elle approvisionne en uranium enrichi.

Par contre, il faut dimensionner le marché à approvisionner afin de choisir convenablement la taille de l'usine. Ceci nous permet d'aborder un des premiers aspects importants de l'optimisation qui est basé sur l'effet de taille.

a) Effet de taille

Un étage d'enrichissement est composé principalement d'un compresseur, d'un échangeur, d'un certain nombre de réservoirs et de tuyauteries, ainsi que de barrières. Les principaux matériels, à l'exclusion des barrières, sont des produits classiques dans l'industrie chimique ou mécanique, et l'on sait que le prix relatif d'un matériel diminue lorsqu'on en accroît la taille et la puissance. On admet, en général, que le prix d'un matériel varie, toutes choses égales par ailleurs, avec la puissance 0,7 du débit qu'il traite. Ainsi, pour les matériels constitutifs d'une usine, l'effet de taille tend à la réalisation d'unités de grosse dimension puisqu'alors l'investissement spécifique diminue. En ce qui concerne les barrières, l'effet est différent puisque la barrière est un composant de petite dimension qu'il est nécessaire de rassembler en grand nombre pour constituer des diffuseurs. On est donc confronté dans ce cas avec un effet de série, le prix d'une barrière individuelle diminuant lorsque l'importance de la série croît. Cet autre élément joue également dans le sens de l'accroissement de la taille de l'installation. La figure 3 montre la variation du coût du travail de séparation avec la taille d'une usine de diffusion gazeuse. Ceci permet, en fonction du prix de marché, de définir une taille critique en dessous de laquelle une installation nouvelle n'est plus compétitive.

b) Nombre de tailles d'étages

L'optimisation doit également prendre en compte un facteur important qui est l'efficacité de l'usine. En effet, une usine d'enrichissement est constituée d'une chaîne d'étages, et le maximum d'efficacité résulte du non mélange des flux au niveau de ces différents étages. Ceci conduit au concept de la cascade idéale dans laquelle on a exclu, grâce à un choix judicieux des débits, tous les remélanges. Ceci a cependant comme corollaire que chaque étage devrait être différent des autres, ce qui est toutefois difficilement réalisable en pratique. En effet, une usine de diffusion gazeuse nécessaire pour produire un uranium enrichi à 3 % comporte en série plus d'un millier d'étages. Il est donc plus intéressant de perdre quelque peu sur l'efficacité de l'usine, mais gagner considérablement grâce à l'effet de série réalisable sur tous les matériels.

La figure 4 compare. des usines de structures différentes. et l'on voit, par exemple, que le passage de une à deux tailles permet un gain de l'ordre de 20 % sur l'efficacité de la cascade (cas a et e.). Par contre, l'examen du cas b montre que l'adjonction d'une troisième taille ne permet plus qu'un gain de l'ordre de 3 % sur l'efficacité. Une taille supplémentaire apporterait un gain encore plus faible. On pourrait aussi concevoir une usine constituée d'étages d'une seule taille dont certains seraient montés en parallèle pour donner un débit par exemple deux, trois ou quatre fois celui de la taille unitaire, afin de réaliser une meilleure approximation du profil idéal. L'efficacité que l'on pourrait atteindre dans ce cas (d) servit voisine de 97 %. Toutefois, l'usine serait constituée de petits matériels, et l'on se priverait du bénéfice de l'effet de taille réalisable avec des appareillages de tailles différentes. Les calculs montrent, du reste, qu'une usine réalisée par mise en parallèle de petits étages a une production moins économique qu'une usine identique réalisée en plusieurs tailles. Ceci amène à la conception d'une usine à plusieurs tailles d'étages.

c) Etages optimaux

Partant d'une technologie donnée : performances de la barrière et du compresseur, possibilité de réalisation le plus économiquement possible des divers matériels, on constitue, dans une première phase, un catalogue d'étages optimaux, c'est-à-dire d'étages produisant au meilleur compte des unités de travail de séparation.

En fonction des discontinuités sur certains matériels, par exemple : débit des compresseurs lié à la vitesse de rotation, possibilité d'utiliser ou non telle technique pour la fabrication des échangeurs, possibilité d'assemblage des matériels entre eux, on conçoit des étages pour chaque débit, et ce dans toute la gamme des débits nécessaires. Ensuite et, grâce aux fonctions de prix des divers composants et des caractéristiques retenues, on déduit le coût de l'unité de travail de séparation produite dans chacun de ces étages. Cette fonction n'est du reste pas monotone et fait apparaître des discontinuités lors de l'apparition de butées technologiques, ce que la figure 5 ne faisait pas apparaître.

Le point délicat réside dans la détermination préalable de l'ordre de grandeur des taillos probables, afin de faire des estimations aussi réalistes que possible. A cet effet, on fait appel à la constatation suivante : le coût minimum du travail de séparation est voisin de celui produit dans l'usine ayant le maximum d'efficacité pour un nombre de tailles déterminé. Cette règle permet ainsi, en fonction du nombre de tailles, de déterminer à la fois la valeur de ces tailles et les séries des matériels mis en jeu, donc d'établir les lois de prix dans des zones représentatives. C'est à partir de ces étages optimaux que l'on constitue l'usine, dont le coût global de l'unité de travail de séparation est déterminé à partir de celui des étages utilisés et de l'ef-

Ainsi, nous avons mis en lumière l'effet de la taille et de la technologie sur les choix. Ce ne sont toutefois pas les seuls, et les paramètres économiques ont un poids aussi déterminant que celui des paramètres techniques dans le résultat final.

Nous allons examiner en premier lieu l'influence de la tarification de l'électricité sur l'optimisation.

d) Tarification de l'électricité

C'est le facteur proportionnellement le plus influant sur le coût de l'unité de travail de séparation. L'électricité intervient sous deux aspects :

Tout d'abord, par l'intermédiaire de la consommation électrique spécifique, c'est-à-dire la quantité d'électricité exprimée en kWh nécessaire pour produire une unité de travail de séparation. Cette valeur dépend fortement des caractéristiques des barrières que l'on sait réaliser, qui conditionnent le niveau de pression de fonctionnement. On arrive actuellement à des valeurs voisines de 2 400 kWh/UTS pouvant descendre à environ 2 100 kWh/UTS si l'on fait de la récupération d'énergie. L'autre aspect, sous lequel intervient l'électricité, est colui de la tarification électrique. Une usine de diffusion gazeuse est une grosse consommatrice d'énergie, puisqu'une usine de 6 M.UTS/an nécessite en permanence une puissance de l'ordre de 1 700 Nd. On tombe donc dans le domaine des tarifications haute tension et longue durée. En outre, une usine de diffusion gazeuse est un consommateur susceptible de s'effacer lors d'incidents sur le reseau au bénéfice d'autres clients ne pouvant tolérer d'interruption de leur fourniture. Il y a 'onc là un bénéfice dont la diffusion gazeuse peut tirer quelques avantages. La part énergie représente environ 50 % du coût de l'unité de travail de séparation. Dans ces conditions, une variation de 10 % sur la part énergie, qu'elle résulte d'une modification de la tarification ou de la consommation spécifique ou d'une combinaison des deux. conduira donc à une modification de 5 % sur le coût de l'unité de travail de séparation.

La figure 5 montre graphiquement la sensibilité de la diffusion gazeuse à une telle variation. On a fait également figurer les autres facteurs, en particulier la partie amortissement du capital qui représente environ 45 % du total, et que nous allons examiner à présent.

e) Le taux d'intérêt

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D'une manière tout à fait similaire à ce que nous avons vu pour l'électricité, la part amortissement de l'unité de travail de séparation résulte de deux considérations, à savoir, d'une part, l'investissement spécifique, c'est-àdire l'investissement qu'il est nécessaire de faire pour produire une unité de travail de séparation par an, et, d'autre part, le taux d'intérêt et la durée de remboursement.

L'investissement spécifique dépend du niveau technologique atteint, en particulier, des performances des barrières qui définissent le niveau de pression de fonctionnement, donc ka dimension des appareillages et leur coût de réalisation.

En ce qui concerne la longévité, les performances connues des matériels dans le temps et leur durabilité permettent de fixer la durée d'amortissement sur laquelle il sera possible de rembourser les emprunts et reconstituer le capital. L'autre aspect est représenté par le taux d'intérêt pour, d'une part, la rémunération du capital et, d'autre part, les charges de l'emprunt. L'ensemble de ces deux facteurs joue de manière similaire à la consommation spécifique et à la tarification électrique, et l'on constate, toujours d'après la figure 5, que la sensibilité aux annuités est peu différente de celle de l'énergie.

Ici une remarque toutefois s'impose. En effet, pour la simplification de l'exposé, nous avons considéré séparément l'effet de la tarification électrique et l'effet du taux d'intérêt. En réalité les deux ne sont pas indérendants, puisque, comme nous l'avons fait remarquer, ils dépendent l'un et l'autre du niveau de technologie adopté, en particulier des performances de la barrière. Ils apparaissent, en fait, par le jeu de l'optimisation comme des frères siamois. En effet, un gain sur la part énergie, qui résulterait par exemple d'une diminution de la tarification pratiquée, conduirait également à une diminution de l'investissement spécifique par action sur les niveaux de pressions, donc les dimensions des appareillages. On constate que lorsque l'un augmente, l'autre croît également. On a donc tout intérêt à minimiser l'un et l'autre de ces facteurs pour le bénéfice de l'ensemble.

f) Autres paramètres

Leur importance est beaucoup plus faible puisqu'elle ne représente que quelques pourcents du coût de l'unité de travail de séparation. Il s'agit de l'ensemble des frais de fonctionnement que l'on peut décomposer en :

- personnel,
- frais d'exploitation divers, les fluides par exemple,
- l'entretien des matériels.

Une autre de leurs caractéristiques est que l'on ne dispose que de peu de moyens d'action sur ces facteurs. En effet, dans une usine de taille donnée, le personnel est celui strictement nécessaire, compte tenu d'une automatisation poussée de l'ensemble des tâches. Par ailleurs, les autres frais d'exploitation représentés par les fluides et les autres matières consommables ont été minimisés, et enfin les performances en endurance des matériels qui déterminent la part maintenance doivent nécessairement être supérieures à plusieurs années pour la garniture des compresseurs et à une vingtaine d'années pour les autres matériels du circuit, si l'on veut que l'exploitation soit rentable. Un gain audelà de ces valeurs, bien que restant intéressant, n'a plus d'effet déterminant sur le coût de l'unité de travail de séparation.

g) Premières conclusions

En résumé, nous avons passé en revue les différents paramètres qui influent sur la détermination et le dimensionnement des étages d'une usine pour une production donnée. Nous avons pu noter au passage plusieurs optimisations :

- tout d'abord dans la définition de la cascade où il s'agit de trouver un optimum entre les avantages de l'effet de série sur les étages et l'amélioration du rendement de la cascade ce qui conduit à choisir le nombre de tailles d'étages et le nombre d'étages de chaque taille.

- une optimisation dans la définition de l'étage où il s'agit de trouver un compromis entre la réduction du cont des appareillages du à l'augmentation de pression et la baisse du facteur d'enrichissement qui accompagne cette hausse de pression.
- une optimisation dans la définition des composants où il s'agit de choisir le meilleur compromis entre investissement initial et frais d'exploitation et d'entretien.
- enfin, une amélioration du couple technologie/environnement où un avantage sur l'un entraîne une amélioration sur son complément.

Il est évident que la définition d'un projet se fait par approximations successives, le résultat d'un cycle d'optimisations permettant de raffiner les données qui permettront d'aborder le cycle suivant.

Par ailleurs, les études et expérimentations menées en parallèle, ainsi que l'évolution de la conjoncture économique peuvent avoir des incidences sur les choix.

Mais nous allons examiner quelques autres éléments qu'il est nécessaire de prendre en compte lorsque l'usine est décidée. À savoir les problèmes liés à la construction et au démarrage des installations.

11.2 - <u>REALISATION DU PROJET</u> OPTIMISATION DE LA CONSTRUCTION ET DU DEMARRAGE

a) Problèmes liés à la construction

Tout d'abord, précisons que la durée optimale de construction s'établit aux alentours de 6 ans.

Cette durée comprend toutes les opérations, depuis les travaux sur le site, la mise en place des chaînes de fabrication des matériels spécifiques, ainsi que le montage des composants et leur mise en UF6.

L'ensemble de ces opérations est lié aux cadences de fabrication possibles, elles-mêmes résultant des chaînes de fabrication que l'on construit. Ceci a, bien entendu, une incidence sur le coût des composants, dont il a bien évidemment été tenu compte au moment de la conception.

b) Problèmes liés au démarrage

Plusieurs politiques peuvent être envisagées, chacune avec ses avantages et ses inconvénients. Par exemple, on pourrait en concevoir une dans laquelle on réaliserait la totalité de l'usine avant de la démarrer, et, dans ces conditions, il s'écoulerait une dizaine de jours entre la mise en service et le début de production au régime nominal, ces 10 jours étant nécessaire pour la constitution du hold up isotopique de l'installation.

Une autre politique consisterait à la mise en service des éléments au fur et à mesure de leur installation. Une telle stratégie aurait pour elle une bonne rentabilité de l'investissement et un espoir de diminution des intérêts intercalaires. Toutefois, elle est peu praticable sous cette forme extrême, en ce sens qu'une usine tronquée, telle que celle résultant de la mise à disposition de quelques centaines d'étages, ne permettrait pas une production à l'enrichissement souhaité. On ne peut donc en pratique commencer le démarrage que lorsque l'on a à sa disposition un nombre d'étages minimum. On sera néanmoins dans des conditions défavorables, ne serait-ce que par l'efficacité de l'usine. On est donc conduit à envisager d'autres stratégies, par exemple, de production à un enrichissement intermédiaire avec le premier assemblage réduit et réinjection de ce produit partiellement enrichi lorsqu'on disposera d'un assèmblage plus performant, permettant ainsi l'atteinte d'un débit de production plus important, ou à une date avancée.

La figure 6 est un exemple d'une stratégie avec production intermédiaire et réinjection de cette production au cours de l'étare suivante, afin d'atteindre une production importante à 3 % dès que possible.

c) Le marché

Les hypothèses généralement prises en compte supposent que la production est écoulée sans retard. Si ce n'était pas le cas, il faudrait prendre en compte les charges financières relatives à la durée de stockage. La non prise en compte d'une telle hypothèse suppose implicitement que l'on connaît bien le marché et que l'on a mis en place une politique commerciale permettant de trouver des débouchés aux productions visées. Un facteur intéressant provient du fait que la durée de construction d'une centrale nucléaire est un peu plus longue que la durée de construction d'une usine d'enrichissement. Ainsi, on devrait savoir au moment de la décision de construction de l'usine d'enrichissement quelles sont les centrales qu'elle sera susceptible d'alimenter. On pourra même prendre des options environ un an ou 18 mois avant la décision de construction, dans l'hypothèse où l'on envisage une production au cours de la période de démarrage. Par ailleurs, on aura certainement intérêt à viser les enrichissements inférieurs à la valeur nominale de production, ce qui est le cas de la filière des réacteurs bouillants.

c) Autres éléments de décision

La réalisation d'une production en période de démarrage doit avoir un bilan positif. En effet, la réalisation d'une telle production nécessite des dépenses, telles que le recrutement du personnel, l'énergie électrique consommée, l'uranium naturel de remplissage et l'uranium naturel d'alimentation. Ce dernier est en fait consommé à un rythme plus important que lors du fonctionnement de routine, compte tenu d'une teneur de rejet supérieure à la valeur nominale. Par ailleurs, la consommation électrique spécifique est accrue, compte tenu de l'efficacité plus mauvaise de l'assemblage dont on dispose, moyennant quoi il faut être assuré que les recettes provenant de la vente des produits restent supérieures aux dépenses que l'on encoure pour réaliser la production.

La décision résulte dans le cas général de l'examen des comptes prévisionnels d'exploitation, dans lesquels on détaille année par année ou fraction d'année par fraction d'année le bilan dépenses recettes pour différentes stratégies de production. Un tel examen a pour conséquence le renoncement à des productions précoces lorsque la faible dimension de l'assemblage conduit à des efficacités d'usine faibles et à des consommations prohibitives d'uranium naturel.

e) Conclusions

Deux nouvelles optimisations sont apparues dans la phase de construction et de démarrage :

- tout d'abord, dans le choix de la taille des chaînes de fabrication des composants où s'opposent les avantages que procurent l'étalement des fabrications et l'accroissement des intérêts intercalaires de construction de l'usine d'enrichissement.
- ensuite, une optimisation des régimes de démarrage dans laquelle interviennent des considérations de remplissage, de production et d'efficacité de cascade.

II.3 - EXPLOITATION

Une usine de diffusion gazeuse est conçue pour un régime de fonctionnement déterminé. Toutefois, au cours de la vingtaine d'années d'exploitation de routine, des évolutions peuvent se produire sur les besoins tant en teneur qu'en quantité : les conditions de fonctionnement des réacteurs peuvent évoluer modifiant l'enrichissement des produits à recycler. Par ailleurs, le passage en réacteur produit des isotopes nouveaux de l'uranium dont il faut tenir compte dans le fonctionnement de l'usine. Enfin, la connaissance cumulée du fonctionnement des matériels permet éventuellement d'en tirer meilleur parti (accroissement de la puissance).

Nous allons examiner les différents moyens permettant de modifier les conditions de fonctionnement de l'usine

a) Modulation des pressions

La modulation des pressions peut revêtir différents aspects, mais reste toujours soumise à la contrainte correspondant aux limites de fonctionnement possibles des matériels.

Une première possibilité consiste en la modulation monotone des pressions, c'est-à-dire d'un gonflage ou d'un dégonflage de l'ensemble des étages de l'usine. Ceci permet une évolution de la puissance de séparation. (voir figure 7) tout en maintenant sensiblement constant le coût du travail de séparation. Le fonctionnement de l'usine n'est plus optimal, mais on peut s'adapter dans une gamme assez large aux fluctuations du marché. L'efficacité de l'usine est peu changée, mais l'on assiste à une certaine modification de la teneur de rejet.

Une autre possibilité est offerte par une modulation différentielle des étages.

b) Flexibilité d'alimentation

Il s'agit d'une possibilité d'action sur l'usine en jouant sur le débit d'alimentation. L'accroissement de ce débit permet d'augmenter la production, avec, en contrepartie, l'augmentation de la teneur du soutirage pauvre.

La figure 8 montre la variation de la production et de la teneur de rejet en fonction du débit d'alimentation. Dans certaines conditions, le produit appauvri au-dessus de la teneur optimale peut être réinjecté soit dans la même usine en période de pléthore, soit dans une installation nouvelle temporairement surpuissante. La flexibilité en alimentation naturelle conduit à s'éloigner de l'optimum et pénalise quelque peu le coût du produit. Une telle stratégie pose des problèmes de surplus ou de déficit en uranium naturel dans le cas d'une politique d'enrichissement à façon.

c) Prix de l'uranium naturel

Le prix de l'uranium naturel est un élément susceptible de modifier les conditions de fonctionnement d'une usine de diffusion gazeuse, en ce sens que la teneur de rejet optimale résulte directement du rapport coût de l'alimentation au coût du travail de séparation. Si l'on s'attache à la variation du prix de marché de l'uranium naturel vis-à-vis d'une usine dont le coût du travail de séparation est fixé, on doit assister à une évolution de la teneur de rejet de l'usine d'enrichissement en fonction de l'évolution du coût de l'uranium naturel.

La figure 9 montre l'évolution de la teneur de rejet en fonction de ce paramètre. La figure 10 représente, dans certaines hypothèses, la variation de la teneur de rejet en fonction du coût de l'unité de travail de séparation pour une valeur déterminée du prix de l'uranium naturel d'alimentation.

d) Evolution de la teneur du produit

Une usine déterminée peut produire à une teneur isotopique différente de la teneur nominale ; ceci permet une adaptation vis-à-vis en particulier d'une évolution du marché résultant, par exemple, d'une modification des teneurs optimales nécessaire dans les réacteurs de puissance, ou d'une évolution de la répartition entre les différentes filières connues.

La figure 11 décrit les évolutions de production possibles dans une usine donnée en fonction de la teneur isotopique et l'effet sur les teneurs de rejet de fonctionnement.

e) Autre aspects

D'autres paramètres sont éventuellement à prendre en compte dans l'optimisation du fonctionnement d'une usine de diffusion gazeuse. Toutefois, ces éléments n'ont qu'une portée relativement faible dans les choix initiaux, c'est pourquoi nous nous contenterons simplement de les énumérer. Il s'agit en particulier de :

- L'évolution de la teneur isotopique des uraniums partiellement enrichis recyclés des réacteurs. Les teneurs prévues actuéllement sont comprises entre la teneur naturelle et un enrichissement d'environ 0,85 %. Des progrès accomplis sur les taux d'irradiation des combustibles en pile permettraient probablement une baisse de la teneur des produits réinjectés.
- Le recyclage d'uranium passé en pile a pour corollaire la présence de nouveaux isotopes, en particulier de l'uranium 236 provenant de captures neutroniques au rythme de 1 capture pour 5 fissions. On conçoit que le fonctionnement d'une usine de diffusion gazeuse alimentée principalement par des produits recyclés en provenance des réacteurs puisse conduire à un accroissement sensible de la teneur en uranium 236. Ceci peut conduire, dans la mesure où les produits recyclés représentent une part importante de l'alimentation et où leur teneur est voisine ou supérieure à la teneur naturelle, à une baisse d'efficacité dans la séparation U 235 - U 238. Bien que ceci n'ait que peu d'influence dans une optique de doublement accéléré de la puissance nucléaire installée, l'effet n'est pas nécessairement négligeable si une usine de diffusion gazeuse sert uniquement à assurer les recharges d'un certain nombre de réacteurs de puissance.

- Enfin, un autre élément susceptible d'influer sur l'optimisation d'ensemble résulte de l'introduction d'améliorations aux matériels existants. Un tel facteur est délicat à prendre en compte a priori, et il est de pratique habituelle de ne prendre les améliorations en compte que dans la mesure où elles apportent un gain substantiel sur la période d'exploitation restant à courir, compte tenu de leur installation dans une usine existante et de tout s les contraintes que cela implique.

III - MOYENS MATHEMATIQUES UTILISES

La variété des projets étudiés et le caractère répétitif des opérations d'optimisation ont conduit le CEA à mettre sur pied un outillage mathématique d'une ampleur considérable. Trois ensembles de programmes méritent plus particulièrement d'être cités :

- le modèle de flexibilité d'une usine
- le modèle d'optimisation d'une usine
- le modèle économique de projet d'usine.

III.1 - LE MODELE DE FLEXIBILITE

Il est utilisé aussi bien pour déterminer les régimes de démarrage que pour déterminer les régimes de production d'une usine d'enrichissement. Il permet de connaître les possibilités d'adaptation d'un assemblage d'étages d'enrichissement à une évolution des besoins en teneur et en quantité. On peut par exemple faire varier la production à teneur constante en agissant uniquement sur le débit d'alimentation en UF6 naturel. Bien entendu, dans ces conditions la teneur de rejet évolue. Les courbes correspondantes sont appelées courbes de flexibilité (voir la figure 12). Elles font apparaître une saturation lorsque l'alimentation devient très grande.

Avec une affinité et une translation, on peut déduire de la courbe de flexibilité P (F) une représentation de la variation du coût du produit enrichi en fonction de la production P, et de là obtenir une représentation géométrique des concepts économiques classiques : coût unitaire moyen, coût marginal et coût unitaire moyen minimum (voir figure 13).

Le modèle de base est complété par un jeu de modèles de flexibilité différentielle qui permet d'explorer l'incidence de n'importe quel paramètre sur les caractéristiques physiques et économiques de la production, et par une possibilité de prise en compte des données relatives au comportement des matériels.

III.2 - LE MODELE D'OPTIMISATION D'UNE USINE

Du fait du caractère répétitif des étages d'enrichissement, les usines de diffusion gazeuse se prêtent bien à une étude poussée de l'optimisation des appareillages installés et de l'investissement correspondant, le problème peut se résumer de la façon suivante : sachant que la clientèle demande une production caractérisée par des quantités d'uranium enrichi et les concentrations correspondantes, définir l'usine qui permet de les produire au meilleur prix et calculer ce prix.

Nous venons de voir que pour des valeurs de C_F et de C_{ab} données, le meilleur prix de l'uranium enrichi est obtenu pour une teneur de rejet définie. Il faudrait donc en théorie connaître déjà le coût du travail de séparation pour fixer la teneur de rejet, donc pour figer l'usine, ce qui nécessiterait une optimisation par itération.

En fait, on s'aperçoit que le coût du travail de séparation est très peu sensible aux variations de la teneur de rejet. Il suffit donc de donner à celle-ci une valeur plausible pour obtenir un Cotrès proche de l'optimum.

Une fois définis les objectifs de production et la teneur de rejet il faut entreprendre le calcul de la cascade réelle conduisant au coût minimal du travail de séparation. La méthode est la suivante :

- des formules sont établies donnant la variation du prix des équipements en fonction de leur taille.
- on relie la taille des équipements aux paramètres de procédé : débits, niveaux de pression, etc...
- on définit le coût unitaire du travail de séparation de l'étage en fonction de sa largeur. Inversement, on détermine pour chaque valeur de la largeur d'étage un jeu de valeurs des paramètres de procédé qui conduisent à un coût unitaire du travail de séparation d'étage qui soit minimal.
- on établit la variation du coût unitaire minimal du travail de séparation en fonction du niveau de remplissage.
- on détermine par ailleurs le nombre de tailles d'étages nécessaires et on définit pour chacune de ces tailles et le nombre d'étages nécessaires et leur niveau de remplissage.
- on calcule enfin le prix de revient moyen brut, puis en tenant compte de l'efficacité de l'assemblage, le prix de revient moyen net de l'unité de travail de séparation de l'usine.

La méthode ainsi décrite permet l'optimisation globale de la cascade et sa définition d'ensemble mais elle ne fournit encore qu'une estimation théorique dans la mesure où elle ne repose que sur des formules de prix. Le coût réel sera donné par l'étude complète du projet. Au cours de celle-ci, des optimisations partielles pourront être recherchées, selon une méthode plus fine, comparant à production constante le coût de réalisation et les dépenses d'exploitation d'un élément déterminé.

III.3 - LE MODELE ECONOFICUE DE PROJET D'USINE

C'est une exploitation de l'ensemble des éléments technico-économiques d'un projet d'usine, qui est destinée à dégager sous une forme synthétique et quantifiée la compétitivité de l'entreprise. La liste qui suit donne une idée du grand nombre de paramètres qui sont pris en compte dans ce modèle : citons parmi les données commerciales :

> - la capacité de l'usine, sa durée de vie, le facteur d'utilisation, les perspectives de prix de vente,

parmi les données techniques :

- l'investissement, la consommation en énergie, l'endurance, les frais d'entretien et d'exploitation, les progrès technologiques,

parmi les données économiques :

- le plan de financement, la rémunération du capital, le régime fiscal, le glissement des prix, l'investissement, le prix de l'énergie, le prix de la main d'oeuvre.

Faisant appel à toutes ces données, sont alors déterminés année par année, aussi bien pendant les périodes de construction, de démarrage et de production, les comptes prévisionnels de l'entreprise selon la méthode du cash flow actualisé.

Sont enfin tirées de ces exploitations, les caractéristiques d'ensemble du projet : durée de remboursement du financement, rendement du capital, etc...

IV - CONCLUSION D'ENSEMBLE

De l'examen de ces divers éléments, il résulte que le problème d'optimisation d'un projet d'enrichissement se situe principalement au cours de la conception, de la construction et du démarrage. Ce sont, en effet, des étapes qui conditionnent avec une certaine certitude l'ensemble de l'exploitation d'une usine de diffusion gazeuse, et qui peuvent, de ce fait, être pris en compte dans le modèle économique d'ensemble du projet.

Au stade de l'exploitation interviennent, outre l'action commerciale qui détermine le plan de charge de l'usine d'enrichissement, des éléments extérieurs : prix de l'UF6 naturel, recyclage de l'uranium des réacteurs, etc., qui appellent des optimisations touchant plusieurs maillons du cycle du combustible.



Fig: 1

BECOMPOSITION APPROXIMATIVE DU COUT DU KWA MUCLEAIRE



Fig: 2



COMPARAISON D'USINES





SENSIBILITE DU COUT DE L'UTS AUX PARTS EMERGIE, AMMUITES, TRAIS D'EXPLOITATION



DEMARAGE DUNE NOME DE DIFFUSION GAZEUSE

Production



ş










143

Tig: 11







Loût unitaire moyen du produit: Pente AM; Loût marginal: Pente tangente en M; Loût unitaire moyen minimun. Pente AM* (= Coût marginal correspondant)

Fig - 13

LE GRAPHITE NUCLEAIRE

DANS LES ELEMENTS COMBUSTIBLES

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ABSTRACT

1

Graphite was just as necessary as natural uranium for the development of the first nuclear reactors. It soon became less essential, but is now recovering some of its importance in connection with high-temperature reactors.

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A noteworthy characteristic of graphite is that its mechanical behaviour actually improves with temperature and irradiation, so that it becomes, in many different forms, an attractive structural element for certain types of fuel element.

The fabrication process also reveals an unusual property: to obtain the desired allotrophic transformation one raises the temperature of the product to 2800°.

The specifications are as diversified as the uses to which graphite is put, and are adapted to these uses. But graphite has two limitations: it is consumed rapidly at high temperature in the presence of oxygen or oxygen compounds, and its apparent density remains rather far from its true density; thus it is porous.

The relatively high cost of nuclear-quality graphite results essentially from the specifications dictated by each use, from the fact that the market, though important, is not large, and from the fact that unnecessarily stringent requirements are often made in view of the heterogeneous nature of the product.

The price could be lowered by close collaboration between producers and users, so that the latter get what they really need, without unnecessarily large safety margins.

Specialized production units are out of the question because the market is too narrow and orders, while important, are few and far between. The reserves of small or medium-sized clients could not sustain such a solution.

RESUME

Le graphite a été aussi nécessaire que l'uranium naturel dans le développement des premiers réacteurs nucléaires. Il a rapidement perdu de son importance, mais retrouve un regain d'intérêt dans les réacteurs haute température.

Sa caractéristique remarquable de voir sa tenue mécanique augmenter avec la température et l'irradiation en fait un élément de structure très intéressant pour certains éléments combustibles, et ceci sous des formes très variées.

Le procédé de fabrication fait également ressortir un élément assez unique : porter le produit à 2.800° pour obtenir la transformation allotropique souhaitée.

Les spécifications sont aussi diversifiées que les utilisations, et sont adaptées à ces utilisations, mais le graphite a deux limitations : il se consume rapidement à haute température en présence d'oxygène ou de composés oxygénés ; sa densité apparente reste assez éloignée de sa densité réelle, il est donc poreux.

Le coût plus élevé du graphite nucléaire résulte essentiellement de spécifications particulières à chaque emploi, de marchés importants mais exigus, d'exigences souvent inutiles mais liées à son aspect hétérogène.

Les prix peuvent être abaissés par une collaboration étroite entre producteur et utilisateur, pour que ce dernier obtienne effectivement ce dont il a vraiment besoin, sans prendre de marges de sécurité inutiles.

On ne peut pas concevoir d'unités de production spécialisées, car le marché reste trop étroit et sous forme de commandes importantes mais rares. Le fonds de roulement de petits ou moyens clients est inconcevable.

L'uranium naturel et le graphite ont eté les deux éléments essentiels qui ont permis la réalisation de la première réaction en chaîne le 2 Décembre 1942, il y a moins de trente ans. Mais, comme tous les précurseurs, ils n'ont pas eu longtemps raison et ont été plus ou moins

rapidement décriés. Il n'est toutefois pas impossible que le graphite ne retrouve une nouvelle jeunesse dans la production d'énergie nucléaire dans les réacteurs haute température.

Le graphite a donc constitué le modérateur des premiers réacteurs plutonigènes ou de puissance réalisés aux Etats-Unis et en U.R.S.S. et plusieurs dizaines de milliers de tonnes ont été utilisées ; il constitue le modérateur de la quasi totalité des réacteurs de puissance fonctionnant à ce jour en Grande Bretagne et en France.

Le graphite, matériau de structure

Mais le graphite possède une caractéristique uniqué qu'aucun autre matériau ne saurait lui disputer ; d'une manière générale, l'élévation de température et l'irradiation détruisent les caractéristiques mécaniques des matériaux, et ce dans des proportions souvent considérables ; pour le graphite, les effets sont rigoureusement inverses, les caractéristiques mécaniques augmentent de 40 à 50% quand la température croît jusqu'à 2500°C et peuvent facilement être doublées par l'irradiation. C'est pourquoi on a pu envisager de l'utiliser comme matériau de structure et comme support des éléments dont le comportement pouvait se révéler défaillant quand la température était élevée ou quand la dose d'irradiation augmentait ; ceci s'est révélé particulièrement intéressant pour les éléments combustibles.

C'est essentiellement en Grande Bretagne et en France que les études ont été activement poussées pour que, dans sa conception même, l'élément combustible fasse appel aux propriétés toutes particulières du graphite. D'importantes réalisations industrielles ont suivi dont nous allons dresser un panorama rapide.

Dans les éléments combustibles des réacteurs graphite-gaz construits en France, le graphite intervient trois fois, comme âme du barreau creux d'uranium, comme chemise du barreau gainé et comme selle supportant ce barreau gainé.

Dans la filière MAGNOX, il intervient comme barreau de centrage des éléments combustibles ou comme chemises des barreaux gainés.

Dans la filière A.G.R., deux barreaux gainés sont pris dans deux chemises intérieures, elles-mêmes glissées dans une chemise extérieure, le tout constituant l'élément combustible. Dans l'élément combustible EL 4, des grilles solidarisent entre eux les crayons combustibles et les lient à une chemise en graphite qui assure la rigidité de l'ensemble.

Si l'on passe aux réacteurs H.T.R., le graphite est vraiment le matériau indispensable sans lequel le réacteur n'est même pas concevable ; il constituait le gainage des premiers éléments combustibles Dragon et des premiers éléments combustibles Peach Bottom ; il contient les crayons combustibles des réacteurs type Gulf ; quant aux boulets combustibles des réacteurs allemands, si la pièce massive évidée n'a pu être réalisée de façon satisfaisante, le graphite en poudre reste un des éléments essentiels du choix actuel.

Fabrication du graphite

La diversité des applications n'a pu qu'entraîner la diversité des exigences qui ont été demandées, allant de l'extrême solidité des selles pour éléments E.D.F. jusqu'à l'imperméabilité presque totale, mais contrôlée, des éléments Dragon. Pour voir comment l'adaptation a pu se faire, il nous parait souhaitable de rappeler brièvement les principes mêmes de la fabrication du graphite.

La matière de corps solide (coke de pétrole, coke de brai, coke de Gilsonite, graphite naturel, noir de carbone) est broyée, tamisée, classée, puis malaxée avec un liant à haut rendement de cokéfaction tel que brai de houille, bakélite, résines fulfuryliques. Ensuite, le produit est mis en forme, et c'est un point sur lequel il faut tout particulièrement insister ; en effet, alors qu'une partie relativement faible des frais de fabrication a été engagée, le produit a reçu la forme sous laquelle il subira la suite des différents traitements, et il est toujours avantageux que cette forme soit très proche de celle sous laquelle il sera finalement utilisé. Le produit est cnsuite cuit vers 800 ou 1000°C pour cokéfier le liant ; la vitesse de cuisson doit rester lente et le cycle s'étale sur plusieurs semaines. On obtient, à ce stade, un produit poreux, dur, peu conducteur de la chaleur et de l'électricité. Des imprégnations avec des matières analogues aux liants du stade cru, suivies de recuisson apparaissent toujours nécessaires dans le domaine des graphites pour l'industrie nucléaire.

On passe ensuite à la graphitation qui consisté essentiellement à porter les produits vers 2800°C pour transformer le carbone amorphe en graphite ; on obtient ainsi un produit dont la densité apparente peut aller de 1,65 à 1,90 gr/cm³ suivant les qualités, alors que la densité

vraie est de 2,25 gr/cm³. C'est donc un produit poreux que l'on peut encore traiter par des imprégnations appropriées suivies de traitements thermiques. Mais, quoi qu'il en soit, on n'obtient jamais de produits totalement graphités qui soient réellement imperméables, car les traitements thermiques à 2800°C qui, seuls, permettent de passer au stade graphite, éliminent toujours des produits volatils qui créent des porosités ; nous ne voulons évidemment pas parler des monocristaux qui peuvent être produits mais n'ont rien à voir avec le graphite pour l'industrie nucléaire.

Puisque nous traitons du graphite nucléaire, expression à laquelle on donne habituellement le sens de graphite purifié, on ne peut passer sous silence la purification. Celle-ci est réalisée essentiellement par l'intermédiaire de produits halogénés, gazeux aux températures auxquelles ils sont actifs. Deux procédés sont utilisés :

- l'un où l'on épure pendant le stade même de la graphitation

- l'autre où le traitement épurant est fait après la graphitation sur les produits sortis du four.

Les spécifications

Du graphite utilisé dans les éléments combustibles, on attend que ses caractéristiques mécaniques ne soient pas défavorablement affectées par l'élévation de température et l'irradiation ; mais c'est la spécification que l'on n'exprime même pas car c'est une qualité intrinsèque du matériau. Mais la grande diversité des emplois eutraîne une grande diversité des propriétés particulières «xigées par chaque emploi spécifique ; et, pour prendre un exemple simple, nous retiendrons celui des dimensions des produits, en rappelant que la forme donnée dans les tout premiers stades de la fabrication est peu éloignée de la forme définitive.

Les selles des éléments E.D.F. ont un diamètre de 42mm, une épaisseur de 20mm. Les âmes graphite de ces mêmes éléments ont 25mm de diamètre et 600mm de long.

Les blocs combustibles H.T.R. sont des hexagones de 360mm sur plat, 800mm de long.

Les chemises EL 4 sont des tubes de 93mm de diamètre extérieur, 7mm d'épaisseur, 470mm de long.

Les chemises extérieures A.G.R. ont 240mm de diamètre extérieur, 16mm d'épaisseur, 1052mm de long.

TABLEAU DES SPECIFICATIONS

		<u>CHEMISES</u> POUR REACTEURS EDF (Chinon et St Laurent)	CHEMISES POUR AGR ("outer sleeves" pour Hunterston B)
1	Densité apparente Porosité ouverte effective	> 1,65	>1,78 <8 cm3 / 100 g
	PURETE		
	Teneur en cendres (ppm)	pour un lot 400 pour la campagne 250	
	Teneur en Bore (ppm)	pour un lot 0,6 pour la campagne 0,4	
	Teneur en Uranium (ppm)	0,1	
	Teneur en Cobalt (ppm)	0,2	
	Section de capture Réactivité à l'air, à 400°		pour un lot 4,5 " 3,95 en déduisant la con- tribution du bore $< 5 \cdot 10^{-6}$ g g ⁻¹ .h ⁻¹
	CARACTERISTIQUES PHYSIQUES Coeff. de dilatation moyen entre 20 et 120° Résistivité		$// > 2,0 . 10^{-6}$ $\downarrow < 5,0 . 10^{-6}$ $(10^{-6} \text{ ohm.cm}):<1200$
	CARACTERISTIQUES MECANIQUES Résistance à la compression Résistance à la traction Résistance à la flexion (4 points) Résilience Eclatement	$// > 430 \text{ daN.cm}^{-2}$ // > 150 daN.cm^{-2} // >0,5 daN.cm.cm^{-2} > 37 daN.cm^{-2}	//>207 daN.cm ⁻² L>155 daN.cm ⁻²

Par ébauchePar pièce usinéePerméabilité (poise microdarcies)5050

Les éléments combustibles Dragon ont 2 m de long.

Parallèlement à ces diversités de forme, se manifeste une diversité de spécifications, et nous voudrions donner l'exemple de deux produits classiques dont plusieurs dizaines et même plusieurs centaines de milliers d'exemplaires ont été élaborés : ce sont, d'une part, les chemises des réacteurs E.D.F., d'autre part les chemises extérieures A.G.R.

Tableau des spécifications (voir 4 bis)

Certaines spécifications sont parfois très difficiles à atteindre, et nous voudrions citer l'une de celles des premiers éléments combustibles Dragon où la perméabilaté devait être telle qu'elle permette de contenir les produits de fission vers l'intérieur grâce à une dépressior interne de l'hélium tout en évitant une rétrodiffusion des produits de fission à contre courant ; ceci avait conduit à des produits où la porosité devait être essentiellement composée de pores ayant des diamètres compris entre 0,1 et l micron avec élimination à la fois des pores de gros diamètres et de la porosité ultra-fine.

Pour en terminer avec ces aspects techniques ou technologiques, nous voudrions montrer sur un exemple que, lorsque les spécifications initiales sévères ont enfin pu être atteintes, il n'est pas impossible de voir apparaître un phénomène imprévu qui rend caducs tous les efforts qui ont été faits.

Lorsque le C.E.A. a envisagé d'armor l'intérieur des tubes d'uranium avec des âmes graphite, les études avaient conduit à imposer une spécification de porosité ouverte très sévère puisque celle-ci devait être inférieure à 6 cm³ pour 100 cm³; on voit tout de suite la difficulté, le produit formé contient des matières volatiles qui devront en sortir au cours des traitements thermiques et elles ne pourront se dégager qu'à travers des porosités que l'on voudrait en même temps pratiquement inexistantes. De longues études et mises au point avaient cependant permis de réaliser industriellement des produits qui contenaient des proportions importantes de noirs de carbone. L'évènement imprévu s'est présenté lors du traitement des combustibles irradiés ; en effet, pour attaquer chimiquement l'uranium, il fallait enlever l'âme graphite, et ceci ne pouvait guère se faire que par usinage dans une piscine ; or ces âmes à base de noirs donnaient, dans le cadre d'un usinage pourtant parfaitement étudié, une poudre qui transformait l'eau de

la piscine en encre de Chine. Ce type d'âme a dû être abandonné ; mais aucune autre formulation ne permettait de tenir la spécification initiale sur la porosité ouverte et il a donc fallu se résigner à élargir légèrement la spécification et la porter à ll cm³ pour 100 cm³.

Coût du graphite nucléaire

Passons maintenant aux questions relatives aux coûts de production et abordons le problème de front, comme certainement chacun le voit. Le graphite nucléaire coûte plus cher que le graphite ordinaire, c'est une affirmation. Pourquoi ? C'est une question. La réponse est simple, parce que l'on ne parle pas du tout des mêmes services à rendre, et donc des mêmes qualités de produits.

Le graphite nucléaire n'est pas toujours plus cher que le graphite ordinaire, et si nous considérons deux cas qui ont des similitudes, les empilements des réacteurs nucléaires et les échangeurs thermiques pour l'industrie chimique, on constate que les prix à la tonne du graphite livré sont du même ordre, avec généralement un prix plus bas pour le nucléaire.

Entre le nucléaire et l'ordinaire, existe aussi une différence fondamentale de réaction des clients ; l'aciériste veut que son électrode tienne bien sur le four, que la consommation soit réduite, mais il laisse le producteur libre de choisir ses matières premières, ses formules, ses traitements. Dans le domaine nucléaire, on est assujetti au respect de spécifications.

Il n'y a guère que pour le graphite nucléaire que se trouvent formulées des spécifications sur la pureté, et ceci non seulement par le biais d'un niveau de section de capture, mais également avec des limites sur la teneur en cendres et sur la teneur en certaines impuretés bien précisées. Un traitement supplémentaire est donc indispensable.

Ce n'est vraiment que pour les graphites nucléaires que l'on voit apparaitre des spécifications de porosité ouverte, de perméabilité, de résistance à l'oxydation à des températures bien définies, et ces spécifications se trouvent même souvent combinées entre elles, ce qui exige, soit des matières premières plus rares, soit des conditions de traitement plus élaborées et, dans tous les cas, plusieurs traitements supplémentaires.

Ensuite, et ce n'est pas la moindre des raisons d'augmentation du coût, figurent les spécifications d'aspect qui, malgré les efforts faits

tant par les producteurs que par les clients, restent toujours subjectives. On craint toujours que les défauts visibles sur les surfaces n'affectent grandement les caractéristiques mécaniques des produits ; il n'est pas très difficile de montrer que ce n'est que très rarement le cas ; et puis que dire des défauts cachés ? N'est-ce pas un peu la politique de l'autruche que de s'attacher à ce qu'on voit en oubliant l'intérieur ? Il serait assez curieux de connaître quelle pouvait être la résistance mécanique de la barre de graphite à l'intérieur de laquelle existait l'empreinte d'un bouchon de radiateur de Rolls Royce qui s'était évidemment volatilisé lors du traitement thermique ; un hasard a permis de le découvrir, mais on aurait très bien pu ne pas s'en rendre compte.

Il est généralement plus économique, et en tout cas plus efficace et plus sûr, de concevoir des tests mécaniques globaux, mettant à l'épreuve les produits dans les conditions les plus voisines de celles où ils seront utilisés.

Dans le cas des graphites utilisés pour les éléments combustibles intervient, si l'on considère les prix à la tonne, la forme même des pièces à fournir, généralement tubulaires. Nous avons vu que le premier stade est celui de la mise en forme, ici ce seront des tubes ; à ce stade les frais sont pratiquement proportionnels à la quantité de matière mise en oeuvre mais, pour tous les traitements ultérieurs, c'est le volume extérieur qui doit être pris en ligne de compte, car il faudra chauffer le tube et le granulé que l'on devra mettre à l'intérieur, et également il faut considérer que l'occupation du four est essentiellement volumique

D'autres raisons de l'augmentation du prix ne doivent pas être passées sous silence :

- les quantités commandées sont bien précises et l'on ne peut se permettre de livrer moins que nécessaire, car il est inconcevable qu'il manque une barre dans un empilement. Or un cycle de fabrication, sans compter l'usinage, dure facilement 5 ou 6 mois auxquels il faut parfois ajouter 2 ou 3 mois pour les contrôles. Pour ne pas être déficitaire sur la qualité fournie, il faut donc obligatoirement fabriquer plus que nécessaire et ce en n'ayant que très peu d'espoir de trouver un client pour les produits excédentaires.

- Dans certains cas, les quantités commandées correspondent à des tonnages très faibles ; or, en graphitation, l'unité de travail industrielle est le four d'au moins 30 t de capacité, et l'on se retrouve dans le cas précédent avec des produits excédentaires pour lesquels les débouchés n'existent pas.

Abaissement des prix

Comment peut-on réduire les prix ? Par la confrontation entre le coût de production et la valeur d'usage, c'est-à-dire par le dialogue entre le fabricant et l'utilisateur. L'utilisateur doit faire un examen de conscience ; les exigences qu'il a formulées, les spécifications qu'il a demandées sont-elles en raisor directe de ses besoins réels ? Car autant il faut avoir prévu les spécifications indispensables, autant il faut faire attention aux spécifications inutiles que l'on conserve pour la forme parce qu'elles sont toujours tenues ; ou l'utilisateur croit qu'elle n'est que pure forme, alors il faut la supprimer, car la maintenir c'est admettre qu'un jour elle ne sera pas remplie et que le produit sera rebuté, ce dont le producteur doit tenir compte s'il est un bon gestionnaire. Chacune des limitations imposée doit être examinée pour définir si la valeur d'usage est en rapport avec le coût de production, coût de production que fournira le fabricant.

Un climat de confiance réciproque doit présider au dialogue, à la confrontation, et il serait intéressant de rappeler un exemple de collaboration où des gains très substantiels ont pu être réalisés par le client sur des fournitures de 20 à 30 MF, sans que les intérêts du fournisseur aient été lésés. On a simplement évité le gaspillage.

Dans les réacteurs E.D.F., l'empilement consiste en colonnes de 10.200mm. A priori, il est demandé pour ces colonnes un découpage en 8 longueurs égales de 1.275mm ; mais une telle solution n'est absolument pas rationnelle. En effet, on demande également, et c'est normal, que 3% des barres soient échantillonnées par prélèvement en extrémité d'un troncon de 130mm ; on demande que les produits ne soient pas écornés en extrémités : on prévoit un descriptif des défauts locaux qui doivent entraîner le rebut des barres. Si on examine le problème dans son ensemble, on constate que toutes les barres où l'on a prélevé des échantillons sont à rebuter, quoique les 9/10 du produit restent bons ; le cas est identique pour les barres écornées ; on constate que toutes les barres présentant un défaut sur une longueur sont rebutées alors que la moitié au moins de la barre pourrait encore être bonne. Donc, sur la première hypothèse de 8 longueurs de 1.275mm, le producteur estime que 15% de ses produits seront rebutés et, bien évidemment, il en tient compte dans son offre. Mais ce pourcentage peut être réduit et ramené au niveau de 5% environ par une étude raisonnée du problème, et ceci en s'attachant naturellement à ce que la conception de l'empilement du réacteur ne soit pas modifiée ; on construit alors l'empilement sous forme de 6 lits d'une

longueur maximum à définir, plus 1 lit dont la longueur sera inférieure de 130mm pour tenir compte de l'échantillonnage et des barres écornées, plus 1 lit de longueur moitié pour tenir compte des défauts localises au hasard. Et, à ce moment-là, l'offre peut être présentée avec l'alternative : soit 8 lits de 1.275mm, soit 6 lits de 1.378mm, 1 lit de 1.248ana et 1 lit de 684mm ; mais, entre les deux termes de l'alternative, la différence de irix est de 10%.

Unités de production spécialisées

Et, pour terminer, on peut se poser la question : une unité de produc tion qui serait spécialisée pour le graphite nucléaire permettrait-elle de proposer des prix de vente satisfaisants tout en assurant une rentabilité correcte ?

Les contrats de fourniture de graphite nucléaire sont limités en nombre, irréguliers dans leur importance, quelque peu imprévisibles dans leurs dates de conclusion et également conditionnés par des prévisions évolutives. Nous pensons donc qu'à priori un producteur qui désirerait spécialiser un atelier pour cette production devrait accepter des charges d'amortissement élevées qui, supportées par les produits vendus, entraîneraient des coûts élevés ; une expérience a été faite en ce sens et l'atelier a souvent été en chômage. Il y a 7 ou 8 ans, les prévisions sur le marché français faisaient état, pour la période actuelle, de besoins de l'ordre du million annuel de chemises pour éléments combustibles or la réalité est bien différente, et il eût été imprudent de s'engager dans la création d'une unité spécialisée.

Le risque déjà grand pour un producteur établi deviendrait encore plus important pour un nouveau producteur ; en effet, les graphites nucléaires sont certainement ceux qui posent le plus de problèmes sur le plan de la recherche et de la mise au point. Bien souvent, dans le domaine nucléaire, on voudrait assimiler le graphite à un métal dont on souhaiterait qu'il ait toutes les qualités des métaux en plus, naturellement, de ses qualités propres. L'unité devrait donc avoir à soutenir les efforts constants d'une équipe de recherches bien étoffée, dynamique et compétente et, par conséquent, les frais de productions seraient lourdement grevés.

Quel est l'avenir du graphite nucléaire dans les éléments combustibles ? D'une euphorie un peu délirante il y a quelques années on a eu tendance à passer très récemment à un pessimisme noir. Nous pensons qu'il faut s'arrêter à un optimisme raisonné et raisonnable car la sagesse est toujours dans le juste milieu.

Quality demands on zirconium alloys

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Abstract

Zirconium base alloys have established themselves as structural materials for core components in water cooled nuclear reactors on the basis of low neutron absorption combined with satisfactory corrosion resistance and adequate strength. The corrosion properties required are primarily a function of the composition, implying the need for close control of impurities and contamination during fabrication of the base metal and manufacture of the finished components. The mechanical properties are basically related to the hexagonal structure of zirconium, giving rise to pronounced directionality of strength and ductility. While this may be of advantage on the user's part, the development of texture is also an important factor determining the route of manufacture, particularly in the case of tubing.

1. General background to the use of zirconium

The basic reason for the interest in zirconium as a structural material in nuclear reactors is its relatively low absorption of thermal neutrons:

	Cross section		
	barns/atom	equivalent thickness *)	
Carbon	0+0034	530	
Beryllium	0.010	180	
Magnesium	0.063	28	
Zirconium **)	0.180	10	
Aluminium	0.230	7.8	
Iron	2.53	0.71	
Stainless steel (18 % Cr, 18 % Ni)	2.9	0.62	
Nickel	4.8	0.37	

Arbitrary units indicating thicknesses which would yield the same absorption of thermal neutrons in all these materials.

**)

' A prerequisite for the relatively low neutron absorption coefficient of zirconium is the absence of hafnium, which has a very high neutron absorption. In nature hafnium occurs together with zirconium, and hence this element has to be removed. For use as structural materials in water cooled reactors beryllium and magnesium are ruled out because of their limited corrosion resistance. Aluminium does not have a sufficient (creep) strength at temperatures of interest in modern power reactors. Carbon or graphite have also been mentioned but only for use at temperatures above the present design limits.

Austenitic stainless steel is superior to zirconium base alloys with respect to general corrosion and strength as far as fuel element cladding is concerned. As will be discussed further in the lecture on special steels, the propensity of austenitic stainless steel to stress corrosion cracking excludes its use as cladding in boiling water reactors, BWR. While this is not necessarily true for pressurized water reactors, PWR, provided that they are operated with proper water chemistry, the cost penalty of the neutron absorption favours zirconium.

Apart from cladding, boxes (for BWR only) and spacers, zirconium base alloys are used as pressure tubes. These are the only components which will be considered in this lecture. Minor applications such as cladding and other core components in CO_2 -cooled, UO_2 fuel elements, as well as in thermal, sodium-cooled reactors, vessels in homogeneous or molten salt reactors, and hooks in Magnox fuel elements, will be excluded because of their minor economic importance from the production point of view.

2. Development of present alloys

There is one zirconium base alloy which dominates in all the major applications in water cooled reactors, namely Zircaloy-2, containing typically 1.5 % Sn, 0.10 % Fe, 0.05 % Ni and 0.10 % Cr. This alloy was introduced in the early 1950's for BWR and was later parallelled by Zircaloy-4, differing from Zircaloy-2 by a maximum nickel content of 0.01 % and an increased iron content, 0.15 %. The composition of the Zircaloys was selected primarily with regard to corrosion. Through the years considerable efforts have been made to find alternative alloying additions which might give similar or improved corrosion properties, in particular at higher temperatures than those common to present BWR and PWR. Only few elements or combinations of elements have yielded such a superiority over the Zircaloys as to warrant serious consideration for applications in commercial reactors.

At present there is only one alternative to the Zircaloys as cladding in water cooled power reactors, namely Zr-Nb alloys. In the Soviet Union Zr - 1 % Nb is used in PWR and the Canadians use Zr - 2.5 % Nb in pressure tubes.

Whether in the future new alloys will take the place of the Zircaloys and the Zr-Nb alloys is uncertain. Not only do new alloys have to pass extensive and lengthy tests both out-of-pile and in-pile; they may also require modifications to the manufacturing processes. To justify a change of alloy composition all of these economic factors have to be balanced by considerable advantages over the present alloys, for instance longer life under normal conditions or increased safety against premature failure of the component in question.

3. Corrosion behaviour

3.1 Mechanism of corrosion and effect of alloying elements

Water and steam attack zirconium, producing an oxide layer which grows in thickness at a rate which first decelerates parabolically and later becomes more or less constant, Fig 1. In the early stage the colour of the oxide is black, but later it turns white.

The mere reduction in load-carrying section of the metal by this oxidation process is usually of little importance, what matters is the simultaneous production of hydrogen $(2r + H_2^0 = 2rO_2 + H_2)$ which dissolves in the metal. If the metal becomes saturated with hydrogen, this results in the formation of another phase in the metal, zirconium hydride (approximately $2rH_{1.5}$), which precipitates in the form of plates. Fig 2. The amount of hydride that precipitates is determined by the solubility of hydrogen, which decreases with decreasing temperature to very low values at room temperature. The important effect of this is to lower the ductility of the material, in particular at lower temperatures.

As regards the mechanism of the effect of various elements on the corrosion and hydrogen absorption of zirconium base alloys in water and steam, relatively little is known. There are a number of factors which can be affected by compositional parameters: transport of the ingredients in the oxide, reactions in the metal-oxide and oxidewater phase boundaries, microstructure and mechanical integrity of the oxide. Transport processes in a bulk oxide phase are generally

known to be governed by the presence of defects in the arrangement of atoms in the crystal lattice. Common rules for the effect of alloying elements on the defect structure and the transport mechanism are not sufficient to explain even the major features of the corrosion process with and without irradiation. Fending a more complete understanding of the effect of practically important factors on the corrosion of zirconium alloys in a nuclear environment, it seems justifiable to assume that one has to take into account, not only the properties of the bulk oxide, but also its defect structure providing routes for the transport of oxidizing species, in addition to crystal lattice diffusion. These transport processes are likely to be affected by irradiation, via its effects on the structure and mechanical properties of the oxide and, possibly, also on the nature of the oxidizing environment. These effects differ from one alloy to another.

For most elements purposely added, such as tin, iron, nickel and chromium in the Zircaloys and niobium in the Zr-Nb type alloys, there is an optimum content with regard to their beneficial effect on corrosion. The total amount of alloy additions is, of course, restricted by the neutron absorption.

Concerning the alloying additions it should be noticed that their effect is beneficial except when they are segregated in the structure. A locally increased content of alloying elements, or excessive precipitation of them in the form of a high density of coarse particles of intermetallic phases, leads to local corrosion attack. Such agglomerates of precipitates may occur if the final melting process is carried out under argon, in which case argon bubbles are formed, accompanied by segregated zones. Another cause of areas with dense precipitates forming is forging in a temperature range where the phase equilibrium of the matrix is such that alloying elements become concentrated in certain regions.

3.2. Impurity effects, localized attack

Another compositional factor of importance from the corrosion point of view, is the presence of elements with a negative influence on corrosion. In this category belong not only non-metallic impurities, such as nitrogen, but also metals, such as aluminium, to

mention only two examples. A typical specification of Zircaloy-2, with regard to the beneficial and detrimental effect of different elements, is as follows:

A110	ying elements, %	Impu	riti es ,	max,	ppm
Sn	1.2 - 1.7	Cu	50	A1	75
Fe	0.07 - 0.20	с	270	Ti	50
Ni	0.03 - 0.08	N	80		
Cr	0.05 - 0.15	Si	200		

To limit the influence of impurities, precautious have to be taken during early stages in the production of zirconium. Absorption of nitrogen has to be carefully avoided during the handling of the primary zirconium sponge. To minimize absorption of oxygen and carbon, the melting and alloying of the metal is done by so-called consumable arc melting in a water-cooled copper crucible under an inert atmosphere or vacuum. Hydrogen is removed by remelting the metal under vacuum.

A further important aspect of the influence of different foreign elements on corrosion is the effect of surface contamination. This may, in principle, occur during heating or heat treatment of the material in the fabrication process, during the surface treatment of the finished product, and during welding. During heating the element of particular importance is nitrogen, which can in principle be avoided by using an inert atmosphere or vacuum. Welding has to be done under an inert atmosphere, usually argon, or preferably by electron beam welding in a vacuum.

The surface treatment, the purpose of which is to remove the contaminated surface layer or adjust the dimensions of the product, can be done either by chemical treatment or mechanical grinding. The latter normally presents no serious contamination problems. Chemical surface treatment involves pickling or etching in a mixture of hydrofluoric and nitric acids, followed by rinsing in water. If not properly rinsed, the zirconium may retain some fluorine-containing material on its surface. On subsequent exposure

^{*)} The same is true of excessive fluorine (and chlorine) in the UO, fuel, which makes the inside surface of cladding tubes sensitive to corrosion attack if the UO, contains moisture. Patches of white oxide are occasionally seen also on fuel elements with nominally satisfactory surface treatment. The cause of this type of accelerated corrosion is not known.

to water and steam such fluorine concentrations may give rise to accelerated corrosion, visible as white spots.

The last occasion for surface contamination to occur is during handling of tubes or finished fuel rods. Fingerprints on a zirconium metal surface will leave behind some chloride deposits which accelerate the corrosion attack showing up as white markings. Protection against such contamination may be obtained by applying an oxide layer to the surface by some kind of oxidizing treatment.

Finally there is another type of chemical attack which has attracted considerable interest lately, namely stress corrosion on the inside surface induced by iodine, which is produced by the fission of the fuel. So far the evidence of such a process in commercial power reactors is rather meagre. Our knowledge of how to cope with this problem is also very limited. A further phenomenon of concern these days is the formation of localized hydride patches on the inside of cladding, so-called blisters or sunbursts. This occurs with increasing frequency as the moisture and hydrogen contents of the fuel increases. The remedy, for lack of an understanding of the origin of such attacks in terms of the properties of the metal, is to keep the moisture content of the UO, as low as possible.

4. Mechanical properties

4.1. Deformation and texture

Stresses and strains in cladding arise from the weight of the fuel elements and the hydraulic forces of the coolant, the radial thermal gradients in the cladding wall, the pressure of the coolant and the fission gases respectively, and the geometrical changes of the UO₂ fuel during its life. It would take too long to give a detailed account of the effect of these different, often interacting types of mechanical conditions and their relative importance. For the purpose of this lecture only the strength and ductility requirements of core components in relation to the mechanical properties of zirconium base alloys will be dealt with. Accordingly this treatment will begin with a brief account of the principles of the deformation and fracture of zirconium.

The basic property of zirconium, from the deformation point of view is the arrangement of the atoms in the crystal structure. The zirconium crystal lattice has a so-called hexagonal symmetry (whilst the

iron crystal is cubic), Fig 3. When a metal crystal is stressed to the extent that it deforms plastically, it usually deforms by the relative slip of layers of atoms in the lattice, Fig 4. Intuitively one realizes that a metal with hexagonal symmetry must differ in deformation behaviour from cubic metals. In the latter the properties of the crystal are the same in all three directions of the coordinate system, so-called isotropy, but in the hexagonal metal crystal the properties differ in the different ortogonal directions, so-called anisotropy.

In a piece of polycrystalline zirconium, made up of a large number of crystals or grains, the orientation of the hexagonal axes of each grain differs from that of the other grains. The variation of the grain orientations within the piece of zirconium may be statistic or there may be some coordination to preferred orientations, so-called texture, Fig 5. Crystallographic textures occur when a piece of polycrystalline zirconium is deformed plastically, for instance during shaping by tube rolling. The reason is that the many differently oriented crystals or grains in the material respond differently to the stresses and strains imposed, because of the different orientations of their hexagonal crystal axis. As a result the distribution of orientations in the population of grains changes from statistical into some regular texture pattern, Fig 5. This in turn makes the further response to stresses and strain during use of the component dependent on the direction in which the stresses and strains are applied, Fig 6. The development of texture during shaping and fabrication of reactor components, tubing in particular, and the consequences of texture for the properties of the components are among the major areas of technical development in this field.

In principle the requirements of certain properties in different directions of, for instance, a cladding tube, imposed by the mechanical conditions mentioned in the introduction of this section, may be met by the adoption of a suitable route of deformation during fabrication. All the various demands of mechanical properties cannot be met by one texture. Therefore compromises often have to be made. A further complication to the texture problem is the occurence of hydride, as already described above. The hydride plates are usually not statistically oriented but show a parallel orientation

in a direction that is a function of the texture and/or the deformation process during fabrication of tubes or other shapes, Fig 2. A hydride plate which is oriented perpendicular to the stress decreases the ductility, i.e. the ability to be strained plastically before fracture occurs, to a larger extent than a plate parallell to the stress. In cladding tubes radially oriented hydrides must be avoided for this reason.

4.2. Irradiation effects

At this point it may be proper to bring up another complicating factor as regards deformation, namely the effect of neutron irradiation. The high energy neutrons in the reactor core knock out or displace zirconium atoms from their normal places in the lattice, leaving holes, or so-called vacancies, behind. In the structure the vacancies are found in the form of agglomerates, Fig 7, which impede the deformation process. Deformation by slip along crystallographic planes, Fig 3, requires that the planes contain a particular deviation from the regular atomic arrangement in the form of a slight shift in position of atoms along a certain direction in the lattice, a so-called dislocation, Fig 8. It is the movement of such dislocations that is retarded by the vacancy agglomerates induced by neutron irradiation.

On the other hand irradiation may enhance creep, i.e. slow, timedependent deformation in which obstacles are eventually overcome by activation processes. In this case the effect of irradiation is due either to the single vacancies which assist in the dislocation movement, or to changes in the internal stress distribution as a consequence of agglomeration of defects in particular configurations. Consequently the ability of the zirconium to deform plastically is reduced; the ductility or strain to fracture may be decreased by irradiation. Therefore the requirements of tensile or burst ductility of zirconium tubing as delivered contain a margin to account for the irradiation embrittlement. As shown by Fig 9 the relative reduction in tensile ductility differs with the amount of prior cold work.

4.3 Requirements on tensile and creep properties

From the designer's point of view the primary strength requirement is a minimum yield strength from room temperature to the

operating temperature, usually $300 - 320^{\circ}$ C for BWR and $340 - 400^{\circ}$ C for PWR. The strength of tubes is measured in the longitudinal direction by tensile tests as well as in the circumferential direction by burst tests. To a considerable extent the strength is determined by the oxygen content of the alloy, while the metallic additions do not play any very great role in this context, Fig 10. It should be noticed that the effect of oxygen decreases with increasing temperature.

As explained above the fabrication parameters determining these strength values in tubes, as well as the ductility, are the route of forming or shaping by extrusion, drawing, rolling or reduction. The means of achieving various combinations of strength and ductility combinations using different fabrication techniques will be described in another lecture. It may merely be mentioned that the required combination of strength and ductility is usually arrived at by adjusting the annealing of the cold work of the material, Fig 11. Deformation or cold work during forming makes the material stronger and less ductile, as a consequence of the increase in number of dislocations which interact with each other and restrict their movement. Annealing changes the dislocation pattern and causes the microstructure to recrystallize into grains with a low dislocation density, and hence a lower strength and higher ductility.^{*})

Ductility enters as a critical property as soon as deviations from primary operating conditions are considered. Although stresses and strains beyond the yield limit are not normally assumed in the design of fuel element cladding, experience from irradiation tests shows that some capacity for plastic straining is highly desirable. The implications of this point become particular evident when the effect of irradiation is considered, Fig 9. In a tensile test on

^{*)} For Zr - 2.5 % Nb there is an additional means of obtaining different strength and ductility levels which is not possible with Zircaloy. The niobium in Zr - 2.5 % Nb can be precipitated as small particles during cooling from the fabrication temperature, giving rise to relatively high strength by affecting the movement of dislocations. By changes in the cooling rate the number and size of the particles, and hence the strength, can be controlled. A common procedure is to heat the alloy to 875°C for 15-30 min, followed by quenching (80°C/sec) to room temperature, and finally tempering at 500°C for 24 h.

zirconium the maximum load is a conservative measure of the ability of the material to elongate without the development of local thinning, so-called plastic instability, which is the first stage in the fracture process. To avoid plastic instability after only very small strain after irradiation one has to establish a high ductility already in the unirradiated stage. This is one reason why ductility is considered important and has to be taken into account during fabrication.

Creep is of importance in particular for pressure tubes, which have to carry a substantial load from the internal pressure of the coolant. Again both the strength, i.e. the rate of creep deformation as a function of stress, and the ductility or strain to fracture are of interest, Fig 12. Broadly speaking, our knowledge of creep in zirconium alloys is relatively limited for reasons that have to do with the obvious lengthiness of experiments and the complexity of the mechanism, not to speak of the influence of irradiation. The mechanism depends on the stress, as indicated by Fig 13.

The conclusions from experience of the creep of zirconium as to specifications on creep properties are rather vague for the reasons indicated above. While, earlier, there was some preference for Zr - 2.5 % Nb in heat-treatment conditions, it now seems as if coldworked Zircaloy-2 is sometimes preferred.

5. Summary

There are certain demands on the quality of zirconium which can be satisfied by the adoption of suitable methods for the manufacture of the material and for the fabrication of finished products. First, to ensure a low neutron absorption coefficient, it is imperative to remove hafnium, and other highly absorbing ingredients, during the production of the metal. Corrosion resistance is mainly a function of composition, on the one hand permitting improvements by alloying and, on the other, requiring maximum limits for the contents of certain elements. The sensitivity of corrosion to certain impurities makes it necessary also to avoid surface contamination during the finishing operations, and during welding. Mechanical properties are governed by the particular crystallographic structure of zirconium metal, giving rise to anisotropy and texture. This provides a means of obtaining directionality of properties

by using different routes in shaping of the products with due regard to the influence also on other properties, such as hydride orientation.

Acknowledgement

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Fig. 1. Oxidation of Zircaloy-2

a) in principle,

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b) under PWR conditions.

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Fig. 2. Hydride plates in cladding tube of Zircaloy-2; Metallografic section perpendicular to tube axis, magnification 100x; a) tangential, b) radial and c) random orien-

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tation.



Fig. 3. Crystal structures of zirconium (right) and iron (left).



Fig. 4. Deformation of a crystal by slip of the upper portion relative to the lower portion (by movement of dislocations).



Fig. 5. Development of texture in zirconium tubing.

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Fig. 6. Influence of texture on deformation behaviour of Zircaloy-2 sheet.



Fig. 7. Vacancy agglomeration at the end of a displacement cascade, a so-called depleted or diluted zone.

Fig. 8. Formation of an edge dislocation (b) by insertion of an extra plane of atoms in a perfect crystal (a).



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Fig. 9. Change of tensile properties of Zircaloy-2 by irradiation.







Fig. 11. Change in strength and ductility of Zircaloy-2 after cold work by annealing at different temperatures.



Fig. 12. Creep of cold-worked Zircaloy-2 at 325[°]C as a function of stress and irradiation.

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Fig. 13. Stress dependence of the steady in-reactor creep rate for coldworked Zircaloy-2 at 300°C.
Quality demands on special steels

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Abstract

Special steels in this context are stainless steel and nickel base alloys for use as core components such as spacers in water cooled reactors and fuel element cladding in gas cooled reactors. The requirements of importance from the production point of view relate to corrosion, in particular localized corrosion of various types, and mechanical properties, in particular creep strength and creep ductility. All these properties are related, in different ways, to the composition and heat treatment of the steels and alloys. There are no simple rules to serve as a basis for selection of stainless steels or alloys for the applications of interest, but some guide lines can be distinguished. From the corrosion point of view an increased nickel content up to a certain level is helpful; the carbon content has to be balanced by stabilizing additions of titanium or niobium. The desired combination of creep properties, which are largely influenced by irradiation, can be accomplished by adjusting the nickel content and additions of alloying elements which can form secondary particles dispersed within the metal matrix. Examples are given of the choice of steels for some applications in present reactor systems.

1. Introduction

Special steels is a broad subject even if restricted to nuclear fuel. To cover even the most essential features of the metallurgical background of the manufacture of reactor components of special steels within one lecture, one has to confine the treatment to a few typical objects. Therefore this paper deals only with the following applications: spacer springs in water cooled fuel elements and cladding tubes for fuel elements in gas cooled reactors. By special steels in these components is meant austenitic stainless steels and nickel base alloys. The materials problems are essentially corrosion and high temperature strength and ductility.

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To begin with a brief account will be given of the metallurgical factors governing the use of different types of stainless steels and related nickel base alloys, in particular the effect of composition and heat treatment.

The denotation stainless steel implies in the first place resistance only to general corrosion or formation of a protective oxide layer on the surface. Broadly speaking most stainless steels are adequate for service in nuclear reactors with regard to general oxidation. A more selective view has to be taken when it comes to localized corrosion, i.e. stress corrosion and pitting.

Before discussing these various types of corrosion, a distinction has to be made between different types of stainless steels or alloys. The most common type is so-called 18/8, which is an iron base alloy containing 18% chromium and 8% nickel. Chromium is the alloying addition primarily responsible for the oxidation resistance, but nickel also contributes in this respect. In addition nickel has the effect of changing the symmetry of the arrangement of the atoms in iron base alloys from so-called body centered cubic (BCC) or ferritic into face centered cubic (FCC) or austenitic, Fig 1.*) The major technically important consequence of this change in crystallographic character is to improve the strength and ductility. Increasing nickel content of iron chromium-nickel steels brings the composition eventually to the range where the material is usually denoted a nickel base alloy, for instance an alloy with 75% Ni, 20% Cr and 5% Fe.

To complete this review of composition factors of importance for the corrosion resistance and the mechanical properties of stainless steel, the role of carbon should be mentioned. If present in sufficient amounts, which usually means a few hundredths per cent, this element forms precipitates or particles of carbides together with chromium and/or other alloying additions such as titanium or niobium, Fig 2. Precipitation of chromium carbides in grain boundaries may cause depletion of chromium in the surrounding regions, making them sensitive to localized corrosion attack.

^{*) &}quot;Austenitic" refers to the crystallographic symmetry of the arrangement of atoms. There is no obvious relationship between corrosion resistance and type of crystallographic structure. The fact that austenitic steels often have a higher resistance to general oxidation is mainly due to the addition of nickel, which is added in order to make these steels austenitic for the sake of strength.

2. Corrosion resistance

Corrosion of stainless steel in fuel element applications is a complex phenomenon which has to be treated with regard to differences between different reactor systems rather than in generalized terms. Corrosion in wate: -cooled reactors (BWR and PWR) is in many respects different from corrosion in gas-cooled reactors (AGR or HTGR) and sodium-cooled reactors (LMFBR). Common to all types of reactors, however, is the general oxidation process which produces a surface scale of oxide. Resistance to this kind of corrosion is promoted by high nickel and chromium contents in a stainless steel or alloy. These alloying elements promote the formation of a protective, slowly growing scale consisting of oxide(s) rich in chromium. Ine composition and structure of the oxide. as well as its rate of growth, differ not only with the alloy composition but also with the oxidizing conditions. The same is true of the adherence of the oxide and the tendency to locally accelerated growth of oxide, which sometimes may be more important than the overall rate of growth. Examples and illustrations of such effects will be given in the section below on applications.

Restrictions on the content of chromium and/or nickel are imposed by the increasing absorption of neutrons - valid, broadly speaking, only for thermal reactors (BWR, PWR and AGR) - and by possible disadvantages with regard to mechanical properties and dimensional stability. It is usually possible to select a composition of stainless steels for use in nuclear reactors with sufficient resistance to general corrosion to satisfy the requirement with regard to preservation of the section of metal in load carrying components. Apart from mechanical properties, however, problems may arise because of the propensity of stainless steel to corrode locally, leading to mechanical disintegration or chemical penetration.

Stress corrosion cracking in water or steam, Fig 3, is a phenomenon which usually occurs only in austenitic stainless steels and nickel base alloys, but rarely in ferritic, low nickel steel. As the name implies, the material cracks under the combined influence of mechanical stress and corrosion. The stress need not be very high; a relatively small fraction of the yield stress is necessary if the environment is corrosive. By the latter is usually meant that the water contains chloride and oxygen. However, in some instances stress corrosion cracking has occurred in nickel base alloys and ordinary stainless steel also in water with no detectable amounts of chloride or other impurities.

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There is no very simple relationship between the time to failure from stress corrosion and the critical contents of alloying elements in different steels, on the one nand, and the critical contents of chloride and oxygen on the other.^{*)} One reason for this unsatisfactory situation is the complexity of the corrosion process as related to the composition and microstructure of the material. With some simplification it may be said that the resistance of austenitic stainless steel to chloride induced stress corrosion increases with increasing nickel content. In so-called pure water there is conflicting evidence about the effect of nickel above 40 - 50%.

The microstructure, which is an important factor in this context, is extremely sensitive to variations in composition, in particular the content of minor elements such as carbon, nitrogen, sulphur and phosphorus, and also to the heat treatment of the alloy. One effect of minor elements is the increasing tendency to so-called intercrystalline corrosion attack - even in the absence of stress - by increasing content of carbon, precipitating in the form of particles of chromium carbide in grain boundaries, Fig 2. This precipitation of chromium carbide can take place during heat treatment of the steel between 400 and 900°C. Therefore, if possible, such treatments have to be avoided as far as possible for material that is to work in a medium known to enhance intercrystalline corrosion. A remedy to this is to add titanium or niobium, which ties up the carbon in the form of less harmful carbides whilst not causing depletion of chromium.

Pitting of the surface of austenicic stainless steels, Fig 4, occurs even without any mechanical stress in water containing chloride. Again there is not enough experience to suggest any rules for selection of a steel composition to be safe against this type of localized attack.

With the proper selection of alloy the corrosion effects dealt with in this review should not cause any alarm with the coolant chemistry normally prescribed. Serious corrosion attack may nevertheless occur

^{*)} Chloride (and oxygen) is not the only agent that can cause stress corrosion in stainless steels in reactor systems. Alkali, which is sometimes used in pressurized water reactors to keep up the pH to limit general corrosion, may give rise to stress corrosion, in particular in carbon and low alloy steels but also to some extent in stainless steels. The remedy appears to be an increase of chromium and nickel contents.

if the critical species in the environment become enriched, for instance at crevices, or if foreign matter (crud) is deposited, giving rise to contamination and increasing the surface temperature.

What has been said so far about general oxidation applies to all types of reactors, including helium cooled ones which normally contain some oxidizing species in the form of impurities. The same is true for grain boundary corrosion attack, usually due to the presence of carbide precipitates. In a later section of the lecture particular conclusions to be drawn in each case will be dealt with.

3. Mechanical properties

The requirements on mechanical properties of stainless steels and nickel base alloys naturally differ according to the application. Springs in spacers of fuel elements must have a high yield strength and low rate of stress relaxation or creep, i.e. high resistance to slow, timerdependent deformation, Fig 5. Cladding of fuel elements must have a high creep strength and creep life but also high ductility.

The mechanical properties of stainless steels, of course, vary with composition and heat treatment. Ferritic steels differ from austenitic steels basically in their different crystallographic structure. For the sake of simplicity this treatment shall be confined to austenitic steels and alloys owing to the fact that ferritic steels usually do not have sufficient corrosion resistance and/or high temperature strength and ductility (i.e. ability of the material to be deformed without fracture).

The mechanical properties of austenitic stainless steels are a function of their composition and heat treatment, in particular the presence of particles of second phases. The function of these particles is to control the elementary processes of plastic deformation or the movement of the so-called dislocations in the metal matrix. In stainless steels carbide particles (chromium, titanium or niobium carbides) are effective for these purposes. At higher nickel contents additions of aluminium can produce fine precipitates of nickel-aluminium phases to increase the resistance to high temperature creep.

^{&#}x27; The higher ductility of austenitic steels also implies that they are more suitable for production of thin gauges such as fuel element cladding tubes.

Strength is not the only requirement as regards mechanical properties. A complication in the selection of a suitable steel for use in reactor systems is the complementary requirement of ductility. It is true that the design of reactor components does not assume very much plastic deformation. Nevertheless some ductility is usually required - for no other reason than to accommodate unforeseen straining. This is particularly important for components in the core where neutron irradiation reduces the ductility of metallic materials. The effect of neutrons is to knock out atoms (iron, nickel or chromium atoms) from their normal places in the crystal lattice, leaving holes or vecancies behind. The vacancies interact in various ways with the metal during its deformation, usually reducing its ability to relieve stresses by slip within the grains of the metal. The net effect of this irradiation damage is to cause the metal to fracture after a smaller amount of deformation, or at an earlier stage, than normally. Often the fracture changes from running right through the grains or crystals making up the metal to intercrystalline fracture.

In addition to the embrittling effect of vacancy agglomerates neutron irradiation gives rise to transmutations which affect the mechanical properties. Neutrons produce helium by reactions with practically all constituents in stainless steel, so-called n- α reactions. In thermal reactors one source of helium is the small amount of boron present in most steels and alloys. In fast reactors the high doses render important the helium contributions also from nickel and chromium. The helium thus produced by transmutations embrittles stainless steels. There is no general theory for helium embrittlement, but a general feature seems to be the enhancement of fracture along grain boundaries. Whether this is generally due to the presence of the helium in the form of bubbles, in the grain boundaries as well as in the interior of the grains, or in other forms is still an open question. Some evidence indicates that irradiation-induced precipitates may be responsible, at least under some conditions.

Typical tensile strength and ductility values after irradiation of a stainless steel are shown in Fig. 6.

4. Selection of steels for different reactor types

To illustrate the general features of the behaviour of special steels for nuclear fuel, common to more than one reactor type, a series of

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examples will be given of the application of particular grades of steel in different reactors. The reactor types to be deal with are water-cooled reactors (PWR, BWR) and thermal gas cooled reactors (AGR).

A list of the steels with code numbers is shown in the Appendix enclosed. In the present generation of water cooled reactor fuel elements, special steels are used as spacer springs ^{*}) The selection of steel for this purpose involves a compromise between yield strength, resistance: to relaxation, resistance to stress corrosion and neutron absorption, which sets a restriction on the nickel content. A high yield strength is required to make possible a thin gauge to minimize the resistance to the flow of the coolant. The exposure to water and steam under conditions which enhance enrichment of chloride and crud in combination with the high stress constitutes a danger of stress corrosion cracking. The compromise between all these factors is usually a 304 type steel with 18% Cr, 8% Ni and low carbon content, but Inconel has in fact also been used in this application. If welding is used for joining, care must be taken to avoid sensitization.

In CO₂-cooled thermal reactors (AGR) the optimization of cladding material with regard to corrosion resistance and mechanical properties has resulted in the choice of an austenitic stainless steel with 20% Cr, 25% Ni and addition of niobium to stabilize the carbon content. Again this is a compromise; alloys with higher nickel content would impair the neutron economy.

Acknowledgement

I thank Dr. J. A. L. Robertson (AECL, Chalk River, Canada) and those of my colleagues who have read the manuscript and contributed on various points, in particular Mr. L. Dahl on corrosion and Dr. U. Bergenlid on void swelling.

^{*)} Stainless steel was earlier used for cladding in PWR but this was abandoned (in favour of Zircaloy) because of failure due to stress corrosion.

Denotations		Con	tents,	weigh	ht per	cent		
	Cr	Ni	Mo	Nb	Ti	С		
304	18	8	-		-	0.08		
316	17	11	2.4		-	0.05 -	0.10)
321	18	8	-	-	0.4	0.08		
347	18	8		0.8	-	0.08		
20/25 Nb	20	25	-	0.5	***	0.02		
12R72HV	15	15	1.2		0.4	0.10	B	0.006
Incoloy 800	21	32	-	~	0.3	0.10	A 1	0.3
Nimonic PE-16	16.5	43	3.0		1.2	0.02	A1	1.2
Inconel 600	16	75	-	-	-	0.15		

Typical composition of stainless steels and related alloys used in nuclear fuel elements

Appendix II

Stainless steels and nickel base alloys for fast reactor fuel elements

Void swelling 1.

In addition to the aforementioned problems of corrosion and embrittlement of stainless steel, which are common to all types of reactors, there is one phenomenon which exists only in fast reactors, namely so-called void swelling. As the name implies, it has to do with a volume increase associated with the occurrence of voids, in contrast to swelling due to gas bubbles. The origin of the voids, Fig A 1s the same as that of the defects causing ordinary neutron-induced embrittlement. Fast neutrons knock out atoms from their normal positions, leaving holes or vacancies behind. Due to the larger dose the amount of such vacancies is much greater than in thermal reactors, leading to supersaturation and subsequent precipitation in the form of voids.

Appendix II

This is the net result of competitive processes trying to eliminate the excess of vacancies. The simplest way of annihilating them is by reaction with interstitials. The reason why cancellation of all vacancies by interstitials does not occur is that the interstitials are preferentially attracted to relatively stable aggregates, so-called dislocation loops. Actually it is these interstitials that are responsible for the volume increase, because of the strains resulting from these configurations, while the corresponding vacancy voids do not exert any stress on the matrix.

Sinks for defects which may be effective in reducing the swelling are dislocations, grain and phase boundaries and phases with a capacity to absorp defects due to strains set up by differences between their crystal lattice dimensions and those of the matrix.

Of practical interest is the effect of dose, temperature and metallurgical conditions on the extent of the swelling. This is illustrated in Figs. B and C. The dose necessary to produce void swelling varies with irradiation temperature but as a rule of thumb 10^{22} n/cm² is a threshold value for measurable volume changes. The temperature dependence, Fig. B, shows a maximum, made up of a rising branch owing to the increased mobility of the vacancies (noticeable at 300 to 400°C) and a decrease above 600° C due to thermally activated movement of vacancies, eliminating void nuclei

In future sodium cooled fast breeders the temperature of interest will range between 400 and 600°C, while for gas cooled breeders the range will be 400 to 750°C. The point in mentioning the range is that there are temperature gradients which will cause swelling gradients, giving rise to bending stresses.

As regards metallurgical conditions, a number of experiments attempting to simulate fast reactor conditions have been made, indicating some trends which are considered of general validity. Cold work (20%) of type 316 steel decreases the swelling at low doses, but at high doses' this effect is diminished as the rate of swelling of cold-worked material increases with increasing doses. The effect of cold work is presumably to provide sinks for point defects. Second phases, like socalled γ' or Ni₃Al in Nimonic FE16, have proved to reduce swelling, Figs. B and C, presumably owing to the trapping of point defects at solute atoms or precipitates. The latter may also have the effect of restricting the expansion of dislocation loops.

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2. Selection of materials

The cladding of fast reactor fuel elements is subject to more demanding conditions than perhaps any other structural material in a nuclear reactor. From the corrosion point of view it has to be sufficiently resistant both to grain boundary attack from the coolant, corrosion on the inside surface and the dissolving action of the sodium coolant. Depending on the design, the requirements on creep properties may differ. The most common view is that not only a relatively high creep strength is necessary but also some creep ductility. This seems to be a modest requirement, but it should be recognized that irradiation lowers the creep ductility of most candidate steels by more than a factor of 10. On the top of this comes the swelling problem, to which there is no solution in principle at this time. With regard to these requirements and the need to be able to rely on consistent dimensional tolerances of tubing, the choice of cladding material for the present prototype reactors is the relatively well known cold worked 316 stainless steel. Increasing demands on composition stability against sodium may result in a preference for stabilized varieties such as 321. A few other alternatives (12R72, Nimonic PE 16 etc.) have also been considered for use in the future charges of fuel elements for prototype fast reactors.

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Fig. 1. Arrangement of atoms in the two types of cubic crystal symmetry; planes with the closest packing of atoms indicated.



Fig. 2. Grain boundary carbide of different degree of precipitation.



Fig. 3. Stress corrosion crack (transcrystalline).



Fig. 4. Pitting attack.



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Fig. 5. Creep:

- a) Typical creep curves at two temperatures, T₁ > T₂
- b) Effect of increasing temperature at constant stress; increasing stress at constant temperature has a similar effect.



Fig. 6. Effects of neutron irradiation [2.5x10²⁰n.cm⁻²(thermal), 8.0x10¹⁹n.cm⁻²(fission)at 40^oC] on the room and elevated temperature tensile properties of a 20%Cr:25%Ni:Nb stabilized austenitic steel.



Fig. A. Voids in type 316 stainless steel irradiated at 510° C to 8 x 10^{22} neutrons per cm².







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Fig. C. Swelling of stainless steel and a nickel base alloy as a function of dose.

CORRECTIONS OF PAPERS G. Ostberg, AB Atomenergi, Stockholm * 0 Quality demands on zirconium alloys p. 10, end of 1st paragraph : ...sterisk instead of B p. 11, line 2 : Reference to Fig. 12 after sentence ending ... of interest. p. 11, end of 1st paragraph : Fig. 13 instead of Fig. 12. 2° Quality demands on special steels p. 6, last paragraph, line 6 : Read : other reason than to p. 7, 1st line : ... effect of vacancy agglomerates p. 7, line 4 : stoel, so-called p. 7, line 10 : boundaries.

Mechanical factors limiting performance and life of fuel element cladding

Addendum to papers on quality demands on zirconium alloys and special steels.

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In addition to the primary design parameters there are some other mechanical conditions which may limit the performance and the life of fuel element cladding. These additional requirements from the mechanical point of view have to be taken into account already during the manufacture of cladding tubes.

The basic cause of stresses and strains in cladding in excess of those imposed by the weight of the fuel element and the forces exerted by the coolant is the change in dimensions and shape of the fuel as a consequence of burn-up and heat flow. This affects the cladding because it is brought in to close contact with the fuel during the rise to power due to the difference in thermal expansion between cladding and fuel materials (typically $6 \cdot 10^{-6} \, {}^{\circ}C^{-1}$ for Zircaloy-2 and $11 \cdot 10^{-6} \, {}^{\circ}C$ for UO₂; at high burn-up swelling of UO₂ becomes more important than thermal expansion). Fig 1. As the cladding is deformed to follow the changes in shape and dimensions of the fuel, ridges are formed at pellet interfaces. This leads to high, multiaxial stresses and strains in these regions.

In the longitudinal direction there is a successive stepwise elongation of the cladding due to the irreversibility of the straining action of the fuel. When the power is decreased, for instance during loadfollowing operations, the fuel contracts in the radial direction. The pellets eventually lose contact with the cladding and become free to move downwards. At a subsequent power increase they make new contact with the cladding at a lower position than before, con* sequently elongating the cladding a bit further. There is thus a stepwise increase in the length of a cladding tube, so-called ratchetting. This process may become particularly severe if the diameter of the fuel varies from one pellet to another. Problems due to such ratchetting growth of cladding tubes have been serious in some PWR's. Attempts to overcome or delay these problems have been made by increasing the gap between fuel and cladding and applying an over pressure of gas on the inside, so-called pre-pressurization, to increase the time to collapse of the cladding by creep deformation.

Power cycling may correspondingly give rise to mechanical cycling of the ridges, so-called low-cycle fatigue, eventually producing tangential cracks. Environmental factors, such as the presence of corrosive fission products might enhance the failure at such regions of high local stress and strain.

In addition to the effects of the changes in shape and dimension of the fuel one also has to consider the consequences of cracks in the UO_2 . Such cracks in the fuel occur due to the steep radial tem= perature gradient, causing thermal expansion stresses which the outer layer of UO_2 cannot accommodate without cracking. As the cladding is in contact with the fuel, its deformation at the circumference is governed by the expansion of the fuel via the friction between the two materials. When cracks in the fuel open up, the

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adjacent cladding becomes subjected to a local strain concentration. This might end up in plastic instability and cracking as illustrated in Fig 2. (The amount of such local strain to which the clad is subjected by ratchetting during a number of cycles can, in principle, be calculated in terms of the friction and the strain hardening coefficient of the cladding material.) Again, increased gap between fuel and clad, and pre-pressurization, may be used to diminish such effects. Lowering the friction by applying some coating on the inside of the cladding has also been suggested.

At present, due to additional mechanical conditions, there does not seem to be any simple means of coping with all these problems by adjusting the properties of the cladding tubes. This is even more true if the complex mechanical conditions at temperature transients during so-called post burn-out or loss of coolant are considered. Since cladding failure due to fuel interaction is usually related to the ductility of the material, improvements of this property should be helpful. However, in non pre-pressurized rods increased (creep) strength is also of value as it increases the time until the critical contact between fuel and cladding is established. The optimum combination between ductility and strength from the point of view of the types of failure dealt with above has not yet been assessed and consequently no particular manufacturing methods based on conclusions of failure analyses can be prescribed.

Contributions to this note have been made by Dr U Bergenlid. Fig 1 is from G Kjaerheim, E Rolstad, Nucl Appl Teck 7 (1969), p 347 and Fig 2 from G Östberg, Zırconium 68, Skoda Plzen, Czechoslovakia 1968, p 345.



FIGURE **1.** INTERACTION BETWEEN FUEL AND CLADDING



Fig. 2 Transverse section showing location of a crack in the cladding near a crack in the UO_2 , hydriding of Zircaloy crack walls and shear fracture.

Fabrication of zirconium sponge

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Abstract

The production of zirconium metal requires special techniques due, on the one hind, to the particular chemical properties of zirconium and its compunds and, on the other, the specific demands on the quality of the metal. The development of the process chemistry and the metallurgy of zirconium in the 1940's established a route for its production with the reduction of zirconium chloride by magnesium as the key reaction, proceded by purification from hafnium for reasons of nuclear quality.

1.- Introduction -

Zirconium was produced long before the nuclear age, but only limited amounts, and by methods which were not suited to economical, large scale production. 1) As a consequence of the discovery of the low neutron absorption of zirconium (after the removal of hafnium which occurs with zirconium in the minerals), in the 1940's development work on the extraction of the metal, by the so-called Kroll method, was intensified. The key feature of the Kroll process is reduction of zirconium chloride to metallic zirconium by magnesium. The following list the sequence of steps leading to this reduction process and the subsequent treatment of the product :

- Mining and concentration of the zirconium mineral, zircon or ZrSi04, orthosilicate containing theoretically 67,2 % of "ZrO2" and 32,8 % SiO2.
- 2. Chlorination of zircon into "ZrCl4" and SiCl4.
- 3. Removal of hafnium, yielding hafnium-free ZrO2.
- 4. Second chlorination into ZrCl4.
- 5. Reduction with magnesium, yielding zirconium raw sponge and MgCl2.
- 6. Removal of excess magnesium by vacuum distillation.
- 7. Crushing, grading, hand-picking and blending of the sponge.
- 8. Production of solid metal ingot by melting of compacted sponge.

Flow sheets of these operations is shown in Figs 1. and 2.

¹⁾ In the 1920's Van Arkel invented a process for the production of zirconium by the thermal decomposition of ZrI4 into metal and iodine. This principle is still the basis for the major part of the production of zirconium in the Soviet Union.

2.- Mining and concentration of zirconate ore.

Zirconium is found in a number of minerals of which only zircon or ZrSiO4 2) dominates industrially. 3) The majority of zircon is at present mined in Australia, where it exists in the form of beach sand as it does in India, Ceylon, Brazil and Florida.

3.- Chlorination of zircon -

Zirconium cannot be obtained from zircon by reduction with carbon or hydrogen, as can iron oxide ore. Before the major part of the zircon had reacted, such a process would yield large amounts of zirconium carbide or zirconium hydride, which would then have to be subjected to further treatment to produce zirconium metal. Even if this were thermodynamically possible, the product would be contaminated by carbon or hydrogen. Instead the route via halogenation and reduction with an alkaline earth metal was developed by Kroll in the 1930's.

The chlorination is carried out at 1000° C in a furnace charged with zircon and carbon (graphite) Fig.3, Chlorine is introduced at the bottom, while ZrCl4 vapor leaves at the top, and is condensed in conventional equipment. The more SiCl4 "volatile" is easily separeted.

4.- Separation of hafnium -

The chloride state offers a means of removing hafnium by techniques which have been developed for similar purposes in other systems. From the production of uranium, it is known that impurities which impair the neutron economy by absorption can eliminated by treatment of a compund of the metal with TBP, tributylphosphate. The function of the TBP is to combine preferentially with zirconium compounds leaving hafnium in the accompanying phase.

Accordingly the chloride is converted into nitrate solution, which is treated with TBP by the so-called liquidliquid separation. The resulting zirconium phase is nitrate, from which ZrO2 is obtained by precipitation. The hafnium content of the zirconium is below 200 p.p.m.

Another process, which is most common nowadays, is the thiocyanate separation, in which an aqueous solution of zirconyl an hafnyl thiocyanates is washed in columns by "MIBK" methyl isobutyl ketone or "HEXONE". The hafnium is preferentially

2) To distinguish it from zirconia, ZrO2.

5) Another mineral resource is baddeleyite, deposited in Brazil and sometimes called " favas de zirconio ".

The mining and concentration usually do not involve any particular difficulties, and the product is delivered in a relatively highly concentrated form. extracted with organic phase. The hafnium content of the zirconium can be less than 10 ppm and the zirconium content of the hafnium less than 0,5 %.

Since the product in this intermediate stage is a form of oxide, it has to be subjected to chlorination. This is done in the same way as described above.

5.- Reduction -

The choice of reducing agent among the alkaline or alkaline earth metals-sodium, magnesium and calcium in particular - is a compromise between cost, including ease of separation etc, and quality. In fact both sodium and magnesium have been used, although at present magnesium predominates.

The reaction

 $ZrCl_4 + 2 Mg = Zr + 2 Mg Cl_2$

takes place in vessels into which both reactants are charged under an inert atmosphere, followed by heating to 700° C (Fig 4). At this temperature the reaction starts and, because of its strongly exothermal character, the temperature rises to a maximum of 950° C.

The zirconium metal is formed in the crucible, Fig 5, growing in thickness towards the inside with a profile which is governed by the temperature distribution and the flow of reactants during the process. The bulk of the magnesium chloride is found on top of the zirconium, but it is also present in the cavities inside the sponge. In the latter it remains even after the separation of the bulk magnesium chloride.

6.- Removal of excess Magnesium -

There are two ways to remove the magnesium chloride, leaching with water, an vacuum distillation. Again the choice of method involves a compromise between cost and quality. While vacuum distillation is generally considered expensive it yields a zirconium metal product of a higher quality (ower oxygen content) than leaching. Another impurity of interest is nitrogen, which may become absorbed onto the surface of the sponge and later impair the corrosion behaviour of the metal.

7.- Handling the sponge -

The zirconium sponge is separated into first and second grades according to appearance and location in the crucible. These are then cut in large pieces, and a rotary crusher is used to reduce each batch to grains of 3 to 20 mm. Undesirable pieces are removed by hand-picking, a blender is used to mix the sponge, analytical samples are prepared, and a uniform quality sponge is made. Zirconium sponge 4s finally shipped in poliethylene bags filled inside with argon steel drums.

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Fig.1. (CEZUS) FLOW SHEET FOR " KHOLL PROCESS ". CHEMICAL PART



Fig.2. (CEZUS) FLOW SHEET FOR " KROLL PROCESS ". NUCLEAR SPONGE PART

Unité de Carbochlocation









THE MANUFACTURE OF ZIRCALOY AND STAINLESS STEEL CANNING TUBES

by Mr. B. Larsson, Sandvik, Sandviken, Sweden

ABSTRACT

Quality requirements on canning tubes for nuclear service are probably the most exacting which a tube producer may encounter today. A satisfactory quality level must be based on a carefully selected production sequence and a well--established quality control and assurance system. The lecture presents a survey of the normally used manufacturing methods from raw material in the form of sponge to finished canning tubes. The influence of production parameters on the properties of Zircaloy canning tubes is discussed and the methods of non-destructive and destructive testing are reviewed. The paper mainly deals with Zircaloy. However, most of what is said can also be applied to stainless steel.

INTRODUCTION

The demands on the integrity of the canning tubes in a thermal reactor are very high. The reasons for this are obvious. Leakage in the fuel elements may lead to serious disturbances in the operation of the reactor and the economic loss connected with a shutdown is very high.

Fig. 1 is a summary of requirements on the canning tubes with regard both to the manufacture of fuel elements and to their use in the reactor. As can be seen, some of these requirements appear in more than one place, e.g. low impurity levels in the material, good dimensional accuracy and cleanness. Very important for the use is of course freedom from defects and the combination of strength and ductility of the canning material.

In the following an attempt will be made to present the ways of fulfilling the demands in canning tube production. Aspects on production sequences and on facilities used in production, inspection and quality control are the main topics. The survey will in the first place deal with Zircaloy tubing but since the production routes and the equipment used in most cases are the same, most of what is said about Zircaloy is applicable also to stainless steel.

MANUFACTURE OF ZIRCALOY

The main operations in the production of canning tubes in Zircaloy from sponge are shown in the block diagram in Fig. 2. To be able to meet the specified analysis on the ingot, we must have sponge with very low contents of impurities, especially of those with high cross section for thermal neutrons, for instance Hf.

Uniformity in analysis within the sponge lots is also important. The alloying elements of which oxygen is an important one are added to the sponge.

The pressed briquettes are electron beam welded to form an electrode which is melted in a high-vacuum arc furnace at a pressure of about 10^{-3} mm Hg. This operation is done twice to ensure uniformity and homogeneity in the final ingot.

Zirconium has a hexagonal lattice at room temperature. In pure Zr this α phase is stable up to 862°C where it transforms to the face-centered cubic β -Zr. In the commercial alloys this transformation takes place over a temperature range where the material contains $\alpha + \beta$, Fig. 3.

Forging of Zircaloy is done in the $\alpha + \beta$ or the β -region. The equipment can be quite conventional. However, one must be sure that the heating conditions are such that no harmful diffusion of gases into the material takes place. The atmosphere should be slightly oxidizing. During cooling of the ingot and also after forging a secondary phase rich in the alloying elements Sn. Cr. Fe (and Ni for Zircaloy 2) will precipitate (Fig. 3). If this precipitation is not uniformly distributed in the material, it will have a negative influence on the corrosion properties. Therefore a β -quench is performed, i.e. heating to the β -region (~1050°C) and quenching.

The extrusion billets are prepared by drilling and turning. They are normally clad with copper to protect against surface contamination during heating and also to facilitate the lubrication during extrusion. The copper is removed after extrusion by pickling in nitric acid.

Extrusion is done in the α -region, i.e. at a temperature lower than $825^{\circ}C$. The microstructure of an extruded Zircaloy tube is shown in Fig. 4.

Cold working of Zircaloy tubes is normally performed as pilgering. Cold drawing is not used, because of the pronounced tendency of zirconium to gall and seize in the die which makes it difficult to find suitable lubricants. Besides, drawing is considered to give an undesirable texture in the tube wall, which will be discussed later. Pilgering can be performed with area reductions up to 80%.

All annealing operations must be carried out in vacuum lower than 10^{-3} mm Hg to prevent contamination of gases. The temperature of the intermediate annealings is around 650-700°C. To meet different requirements on mechanical properties the final annealing temperature may vary between 475-575°C.

In all stages of the production process the surface quality of the tubes must be carefully controlled. Surface conditioning of the finished tubes may involve inside vacuum blasting and acid flushing and outside belt grinding and pickling. The aim of all surface conditioning of finished 'ubes is to give them a surface with favourable properties with regard to corrosion resistance after autoclave treatment. Low surface standard will lead to a grey or white oxide layer instead of the desired black homogeneous surface.

Vacuum blasting and belt grinding are normally done with SiC since the use of Al_2O_3 may lead to the undesirable formation of zirconium oxide according to the formula

 $3 \operatorname{Zr} + 2 \operatorname{Al}_2 \operatorname{O}_3 \longrightarrow 3 \operatorname{ZrO}_2 + 4 \operatorname{Al}$

This is especially important if the mechanical surface treatment is not followed by any removal of material by pickling.

Pickling is done in a HF-HNO₃-bath where the HNO_3 to HF ratio is around 10:1. The HF content is normally 2-3%. The result of the pickling operation is very much affected by the time between the pickling and rinsing operations and also by the efficiency of the rinsing. Unsufficient rinsing may lead to white oxide spots due to fluoride residues on the surface.

MECHANICAL PROPERTIES OF ZIRCALOY CANNING TUBES

The mechanical properties of the canning tubes mainly depend on the following three variables: texture, area reduction in the final rolling, final annealing temperature.

Cold working of tubes by pilgering or drawing gives the material a texture where the c-axes of the crystals, i.e. the basal poles, are oriented in a direction perpendicular to the longitudinal axis of the tube. The direction of the basal poles within that plane can be affected by changing the so-called Q-value. This value is defined as the ratio between the reduction in wall thickness and the reduction in diameter.

$$Q = \frac{t_0 - t_1}{t_0} \times \frac{D_0}{D_0 - D_1}$$

where

a wall thickness before cold working
a wall thickness after cold working
a average diameter before cold working
a average diameter after cold working

A high Q-value, i.e. heavy reduction of the wall and little reduction of the diameter, will lead to a texture where the basal poles are oriented mainly in the radial direction, while a low Q-value (which is normal in drawing) will lead to tangentially oriented crystals (Fig. 5).

In practice, variations in the Q-value are obtained by rolling on mandrels with different conicity. Since the c-direction is the "hard direction" in the crystal, rolling with a high Q-value gives a tube with high hardness in the radial direction and rolling with a low Q-value gives high hardness in the tangential direction.

When deciding the Q-value one also has to take the hydride orientation into consideration. Hydrogen is absorbed ly the tube in service and precipitates during cooling in plate-like hydrides. These hydrides precipitate in the direction perpendicular to the main working direction in the last cold working operation. This means that cold working with a very high Q-value gives tangential hydrides and cold working with a very low Q-value gives radial hydrides. This is illustrated in Fig. 6. As the hydrides reduce ductility and may initiate cracks, the canning tube users normally specify a hydride orientation that must not differ too much from the tangential.

The other factors influencing the mechanical properties of the finished tubes were area reduction and final annealing temperature. Fig. 7 and 8 illustrate how yield strength and elongation are affected by these factors. For high area reductions the mechanical properties change very rapidly with annealing temperatures around 500°C. In some cases the requested mechanical properties necessitate annealing in this temperature range, giving a partially recrystallized material.

In such cases the exact annealing temperature must be determined individually for every lot of tubes. Furthermore, the temperature accuracy of $t'_{,:}$ annealing furnace must be very good, better than $\frac{1}{3}$ °C is necessary in some cases.

Very often transverse properties are specified. These are usually determined

by burst testing with an inside overpressure. Burst testing can be done with open or closed ends (Fig. 9) and it is important to define which method shall be used. The closed-end testing gives a stress ratio of 2:1 between the tangential and axial direction. This method gives substantially lower values on transverse elongation than the open-end method which gives a uniaxial tangential stress in the tube.

STAINLESS STEEL CANNING TUBES

The English AGR is practically the only type of commercial thermal reactor using stainless steel for cladding. The material used is a niobium-stabilized 20Cr/25Ni austenitic steel. The production sequence for canning tubes in stainless steel is very similar to that given for Zircaloy. In order to get a very pure steel free from harmful inclusions, the steel is melted twice, first in an induction furnace with very pure raw materials and then remelted in the same type of furnace as is used for Zircaloy, the vacuum arc furnace.

Although the hot working temperatures differ, the methods used are the same. This also applies to cold working where, however, drawing is used more than for Zircaloy. Canning tubes in stainless steels are often delivered in the cold worked condition (area reduction of 15-25%) to give the tubes a higher initial yield point than what is possible to reach with the annealed condition.

QUALITY CONTROL

General

The high quality demands on canning tubes are reflected in the specified limits for dimensional tolerances and defects. Tolerances are specified either for ID and OD with a minimum wall thickness or for ID and wall thickness.

Normal tolerances are:	for ID	± 0.04 mm
	for OD	\pm 0.05 mm
	for wall thickness	± 10%

The demands on freedom from surface defects, cracks, etc. are very stringent. The rejection level is often very close to what is possible to detect with reliability with the most advanced testing methods and equipment.

It is not possible to make a high-quality product only by very accurate final inspection. The quality of the finished product is a function of the processes by which it is produced. In other words, proper processing methods and an extensive in-process control are necessary means to establish good quality.

In-process control

Besides the chemical composition, which is controlled on samples from the top, middle and bottom of every vacuum melted ingot, there are many factors to control. In the forged bar stage, an ultrasonic examination is normally performed to ensure that the material is free from internal defects. Furthermore, in the preparation of extrusion billets the centre part of the bar is removed by drilling. In the production sequence for canning tubes, intermediate inspections with control of dimensions, surfaces, etc. are frequent operations. However, the process control shall not only inspect the tubes after operations like extrusion, cold rolling, annealing, etc. It is preferable to make spot checks in the course of the operations so that possible discrepancies are found and adjusted at an early stage.

Final inspection

Before being shipped to the customer, the canning tubes pass an extensive final inspection where tolerances, freedom from defects, specified properties etc. are checked (Fig. 10). Defective tubes are sorted out and, if possible, reconditioned and rechecked. If that is not possible, they are scrapped.

The tolerances on the inner and outer diameter are relatively close and it is necessary to control the diameters along the entire length of every tube.

The measuring of diameters is done using air gauges or electro-mechanical gauges. The air gauge is unexpensive and accurate but slow and gives a mean value over a relatively large section of the circumference. The electro--mechanical gauge is more expensive but fast and very flexible. It also measures a more true diameter as it measures the distance between two points in contact with the tube surface and can register very local variations in diameter. The result of the test is normally recorded on a chart.

For the detection of material defects a 100% ultrasonic testing is specified by all customers. The calibration of sensitivity and rejection level is done against in- and outside standard defects for which form, length and depth are prescribed by the customer.

Normally the rejection level is set equal to the amplitude of the signal from the standard defect (100%) but 75 and 50% of this amplitude are sometimes specified as the rejection level. Simultaneous testing for longitudinal and transverse defects is always requested. The standard defects can be down to 1 mm in length and the depth is normally 10% of the tube wall. They are made either by machining with a special tool in a lathe, or by the electrical discharge method. The fact that the standard defects are so small may lead to disturbances from a tube not perfectly cleaned or from an uneven grain size in the material. The latter case is something which especially must be taken into account when stainless steel tubes are tested. The testing is made in four directions, with two transducers each for longitudinal and transverse defects (Fig. 11).

The tube surface is scanned along a helical line with a pitch determined by a demand for a certain degree of overlapping. The signals from defects are recorded. To check the calibration, a 'ube containing the standard defects is run through the test unit at constant intervals, e.g. once every half-hour.

The wall thickness of the tubes is normally checked simultaneously with the defect control since it is performed by means of an ultrasonic resonance method. Calibration of this equipment is done by means of minimum and maximum wall thickness samples according to the specified tolerance limits.

The normally required destructive tests are also listed in Fig. 10. Check analyses for oxygen, hydrogen and nitrogen are requested to establish that no harmful contamination has occurred during the manufacture. Tensile testing is performed both at room temperature and at elevated temperature, normally the operation or design temperature. The burst testing for circumferential elongation has already been dealt with in connection with the mechanical properties. A standard autoclave test is always performed. It takes place in steam at 400° C for 3 days, and afterwards the weight gain of the samples is measured.

The hydride orientation is normally determined after loading the samples with hydrogen to a content of 100-200 ppm. Hydride orientation is usually specified by the F_n -number, defined as the ratio between the number of radial hydrides and the total number of hydrides observed in a transverse section of the tube.

The results of all tests are compiled in a certificate for every lot of tubes. This certificate is sent with the material to the customer.

FINAL REMARKS

A considerable part of the costs for the finished tubes lies in inspection and control. This is a natural consequence of the very high demands on the material which necessitate the use of technically very advanced methods for testing and inspection as well as for production.

Of course, there is a mutual interest from the supplier and the user to keep the total costs of the product down as much as possible. The best way to achieve this is an intimate contact and cooperation in such specification questions as properties and tolerances but also concerning methods and the extent of control.

- 1. Requirements with regard to the manufacture of the fuel elements
 - a) Dimensional accuracy (ID and OD)
 - b) Adequate straightness
 - c) Proper cleanness
- 2. Requirements with regard to the application
 - a) Low cross section for thermal neutrons Analysis - low impurity level Thin wall
 - b) Adequate strength and ductility, determined by
 - Oxygen content Tensile strength and ductility Transverse ductility Hydride orientation Freedom from defects
 - Dimensional accuracy (excentricity)
 - c) Good corrosion resistance to the coolant
 - Low impurity levels Favourable microstructure Adequate surface finish Proper cleanness (e.g. freedom from fluorides)

Fig. 1 Requirements on canning tubes



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TRANSFORMATION TEMPERATURES $\alpha \implies \beta$

Zirconium		862 ⁰ C
Zircaloy-2	Heating	825-985 ⁰ C
	Cooling	945-780 ⁰ C

APPROX. COMPOSITION OF SECONDARY PHASE IN ZIRCALOY-2

Fe	Ni	Cr	Sn	Zr	
7	7	1	1.5	83.5	Weight-%
ri. 2					

Fig. 3





Texture after reduction of



Fig. 5 Variation in texture with different Q-values



Fig. 6 Variation in hydride orientation with different Q-values











The internal pressure gives a tangential stress in the tube wall. The internal pressure gives a tangential stress and an axial stress with the ratio 2:1 in the tube wall.

Fig. 9

Principles of open-end burst testing and closed-end burst testing



Fig. 10 Final inspection and testing of canning tubes



F1g. 11

Principle arrangement for ultrasonic inspection and wall thickness test of cladding tubes

FABRICATION OF ZIRCONIUM-ALLOY CANNING MATERIALS

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ABSTRACT

The production of canning components for a nuclear reactor requires a high degree of quality since the failure to perform adequately results in costs far in excess of the cost of the components themselves. For most water reactors used for commercial power generation, the components consist of zirconium alloys, stainless steels, and nickel alloys. For the latter two materials, conventional practices apply.

Since zirconium is a new and uncommon metal to most fabricators, an education process in handling it is necessary. However, most operations common to other metals can be performed on zirconium alloys, provided due attention is paid to its few unique properties.

Components made of zirconium, other than fuel-clad tubing and control rod tubes, which are discussed elsewhere, include end plugs, spacer grids, and fluid-flow control channels. Use of zirconium for structural components of the fuel assembly is expected to increase as performance data become widely available, especially since matching of the physical properties of the fuel pin with the structural components of the assembly has been identified as an important factor in performance.

Fabrication methods for zirconium alloys must be tailored to incorporate several often conflicting criteria, including corrosion resistance, strength, ductility, and uniformity. And these are, in turn, related to many other effects. For example,

contamination encountered in fabrication can have a large adverse effect on ductility and corrosion resistance. Also, irradiation has effects on strength (or more precisely creep), ductility, etc., which cannot readily be predicted based on out-of-reactor testing.

Because of these complex effects, it is necessary for a fuel design engineer to involve himself intimately in the details of zirconium alloy fabrication. It is equally important that changes in fabrication procedures be considered carefully in order to assess these interrelated effects.

INTRODUCTION

Depending upon the reactor type, the zirconium used for components other than tubing is relatively small. However, the parts require attention in their design and fabrication and all the controls attendant on the tubing are equally necessary for these components. The use of zirconium alloys for these components is based on a compromise of properties. No single property of zirconium compells its use; however, good performance in all required areas makes it the leading material for a light water reactor. These performance areas include: good corrosion resistance, moderate strength, low neutron-capture cross section, low rate of embrittlement due to corrosion and irradiation damage, fabricability, and cost.

Dr. Ostberg has presented an excellent survey of the basis for the use of zirconium¹. While his discussion related primarily to tubular products, these principles apply to the other components, as we shall discuss. To do this, a pictorial tour

through a plant devoted primarily to the production of mill products is presented. While many of the processing steps described in this tour are located within a single facility, the principles and criteria employed at other facilities are quite similar. It should also be noted that it is possible, in general, to perform most working and fabrication operations on zirconium using equipment designed for such other metals as steel, copper and brass, and aluminum. In general, working stress requirements are less than those common to equivalent ferrous metal working. The controlling factor on the use of a given facility is primarily technology rather than equipment. Several specific exceptions will be pointed out as we proceed.

TECHNICAL CONSIDERATIONS

Two primary and closely related differences exist between zirconium and the ferrous metals: A need for levels of cleanliness and freedom from contamination well above most steel practice; and, significantly increased costs for surface conditioning. This latter need is based upon the fact that zirconium oxidizes rather rapidly at the temperatures employed for initial and intermediate breakdown operations. This oxide is very hard and abrasive; moreover, the oxygen-contaminated zone between it and the unaffected interior metal is extremely hard, abrasive, and chemically resistant. Failure to remove it can lead to cracking during subsequent fabrication or under the stress of operating conditions.

Mr. Larsson has discussed the need to properly homogenize the alloy constituents within the Zircaloys in order to achieve proper corrosion resistance². While the alloy additions are normally well dispersed on a macroscopic basis in the as-cast

ingot, segregation on a smaller scale accurs as a result of the relatively slow cooling rate during the arc melting operation. Also, segregation in the alloy takes place during the time intervals when the material is heated into the alpha + beta region (approximately 800°C-1000°C). Therefore, at some stage of the operation a homogenization heat treatment is necessary. This is normally achieved by reheating the material into the all-beta region (above 1000°C) and rapidly cooling the piece into the all-alpha region (less than 800°C). The maximum allowable time for this operation is approximately four minutes. This corresponds to a cooling rate of about 65°C per minute. Commercial practice consists of preheating in a large air furnace and guenching into water. The maximum section thickness which can effectively be quenched is approximately 0.25 meters. Subsequent to quenching, all operations must be performed below the two-phase region to prevent a significant deterioration of the corrosion properties.

Design engineers desire that the grain size of a metal be controlled within reasonably narrow limits. As a general rule, the grain size should be small with respect to the thickness of the component to give good ductility. On the other hand, coarser grain sizes give improved creep properties. For most Zircaloy products the former criteria, fine grain size, is most desirable. Moreover, the grain size should be uniform throughout the cross section of a component, and in practice, a large deviation in grain size signals a potential problem in Zircaloy.

Two closely related phenomena are active in the Zircaloy systems which can cause a serious grain coarsening; both have to do with critical amounts of strain within the crystallographic lattice of the metal. If a piece of recrystallized Zircaloy is

subjected to small amounts of cold work--on the order of 2 to 15%--and recrystallized, the number of nucleation sites available for recrystallization is very limited and an extremely coarse grain size will result (Figure 1). Also, the thermal strain induced during the beta quenching operation, or during welding, can result in the nucleation and growth of extremely large grains if the material is reheated into the high alpha region.

In both phenomena, the size of the grains produced is uncontrolled and sporatic, and may result in zones of relatively large grains in the immediate vicinity of more desirable finegrained Zircaloy (Figure 2). It is easy to see that subsequent stress on a material possessing this type of structure could result in non-uniform stress distribution at the intersection between the fine and coarse grain boundary components, resulting in fracture. This is seen in Figure 3.

The effect of texture on the mechanical properties of Zircaloy tubing is well documented in the literature. In products other than tubing, of course, the properties are similarly affected. Figure 4 shows graphically the effect of texture on the mechanical properties of fully recrystallized Zircaloy strip. This is the type used for fluid-flow channels in boiling water reactors. It can be seen that if fabrication techniques can be properly controlled, considerable control over final properties is possible. One difficulty existing in the current state of the art is that of relating simple mechanical test properties to actual service conditions which include long-term cyclic exposure to combinations of stress, corrosion and irradiation. Dominant among in-service conditions is the fact that the neutron irradiation has a marked effect upon the mechanical performance. Among the most active

programs currently underway are those attempting to relate longrange irradiation test data to properties which can be controlled in commercial fabrication of components.

The preferred orientation, or texture, development in zirconium creates one fabrication probl m: The metal has a memory. Any non-uniform processing will yield a component with varying texture, and consequently, varying properties which no amount of subsequent work will totally eliminate. Therefore, in order to ensure control of properties, deformation operations must be tailored to consider, even at an early stage such as forging, the ability of the metal to recall deformation variations in the final product.

The technology required to produce a sound, homogeneous casting is described elsewhere in this publication¹. It is possible to improperly fabricate zirconium alloys in such a manner that void formation may occur at grain boundaries (Figure 5), or at critical stress points within the material, such as the center of rod (shown in Figure 6). Using proper control, it is, of course, possible to avoid such phenomena. This will be discussed in the details of fabrication later.

Contamination can result if the material is subjected to heating in a reducing atmosphere where free hydrogen is available to contaminate the zirconium. Only 400 parts per million of hydrogen are necessary to severely decrease the ductility of Zircaloy in service. Fortunately, the heat transfer of zirconium is quite good, so times at temperature can be held low. For example, an ingot 0.5 meters in diameter and weighing five ton can be heated to 1050°C, ready for forging, in about four hours. Considerably less time is needed as the section size decreases. Even in this short period of time, an oxygen deficient

atmosphere can cause a large amount of hydrogen absorption into zirconium, especially if there is also direct flame impingement. Heating of smaller section sizes, of course, increases the possibility of contamination.

The final consideration in the fabrication of zirconium components that I will discuss today, and a major one, is that of economics. It is mandatory that the quality criteria discussed above be met or the resulting material will not be satisfactory for reactor application. However, there is still a concern that the process selected be commercially viable. As you will see later, there is a fairly broad choice of fabrication methods which can be used satisfactorily provided the critical aspects of processing are taken into account.

PROCESS DETAILS

Mill Products

The processing of zirconium alloys into nuclear components can be broken into two primary categories: mill products and components. It is not necessary that this distinction be made in a production one, although it is one which has developed in several facilities.

Let us look now at Figure 7 which shows schematically the fabrication of rod material such as would be used for end plug stock in nuclear reactors. It can be seen that methods available are essentially those available for many other metals. Primary ingot breakdown is normally press forging, although other methods may be used, provided they are compatible with the ingot starting size. The beta-quench stage is critical, as are subsequent reheats. Also, deformation operations--rod rolling, extrusion, etc.--should be controlled within limits such that the heat of

reactors. Again, the controls necessary to prevent contamination and maintain corrosion resistance are dominant. Variations in mechanical properties can be achieved primarily through variations in temperature of deformation, reduction sequences, and through use of tension, etc.

Let us now take our tour through a facility designed primarily for the working of reactive metals such as the Zircaloys. It is to be noted that such a special facility is not necessary to achieve high-quality product. However, the guidelines necessary for satisfactory production of high-quality product are more evident in such a plant. I will attempt to point these out in the slides and note how similar operations can be performed in plants which fabricate other metals, including titanium, steel, aluminum, brass, etc.

Figure 9 shows one basic feature of Zircaloy fabrication: The deformation stresses are very low compared to those encountered in steel or titanium. This 2000-ton forge press is a very old one used for many years in a tool steel facility.

The next step in the sequence is the beta-quench operation. Conditioning is necessary after the quench to remove the heavy layer of oxide and oxygen-contaminated metal resulting from being in the beta region--above 1000° C--for several hours. Figure 10 shows, at 100X magnification, both the oxide layer (on the left), and the oxygen-contaminated zone as indicated by the decreased size of the diamond hardness indentations. Conditioning is done by blasting with silicon carbide and local hand grinding followed by pickling in a HF-HNO; acid bath to thoroughly remove the contaminated metal as well as the blasting and grinding grit. This sequence is seen in Figures 11, 12, and 13. It is interesting to note that the pickling operation is of itself a good

control operation since any discontinuity on the slab will entrap acid and cause discoloration as it dries.

The next series, Figures 14 and 15, shows a slab of Zircaloy being fabricated, sequentially, from 100 mm to about 4 mm. At this stage, the material is air annealed, cond'tioned, and cold rolled. The final anneal, and sometimes the last intermediate anneal, is performed in vacuum. Vacuum annealing is chosen rather than air or protective gas in order to assure that surface contamination will not result from the heat treatment. Moreover, any hydrogen absorbed in previous heat treatments, or in pickling and handling, will be removed. Figure 16 shows individual sheets of channel material being cold rolled using no tension; Figure 17 shows the tension rolling of spacer strip.

It is not within the scope of this paper to become deeply involved in vacuum technology. Suffice it to note that vacuums achieved in fabrication of the less reactive metals, such as steel, nickel, and titanium, are generally not satisfactory for zirconium since zirconium's tendency to absorb hydrogen and nitrogen is greater than these other metals.

Figure 18 shows several units used for vacuum annealing of different sizes and shapes of zirconium alloys, including rod, tube sheels, strip, and sheet.

Note in the previous views the general cleanliness of the work area. While <u>general</u> cleanliness of a production facility need not be this absolute, it is mandatory that the metal be kept free of contamination: Oil and grease can decompose to cause surface hardening and hydrogen absorption; foreign metal particles may alloy with zirconium to cause local hard spots or cracks; blasting grit can be entrapped and reduce effective ductility, etc.

Components

As mentioned earlier, physical separation of mill product production from component production is not essential; however, many of the quality control functions must be performed on the mill product since inspectability and control will not be possible later. Therefore, the inspection and test procedures we have discussed for mill products must be completed and reviewed prior to component manufacture if reliability is to be achieved.

The manufacture of cladding components from Zircaloy mill products is considered by most manufacturers to be proprietary information, since critical design aspects are embodied in the process. However, several of the production aspects of the process will be discussed here. Figure 19 shows schematically the production of end plugs from rod stock--an apparently simple process, except for the requirement for rigid dimension control. This plug must fit the tubing very closely to minimize stress concentrations; some must also have controlled external dimensions since they are used to locate the tube within the spacer grid; the external surfaces must be smooth and streamlined to minimize fluid flow disturbance; they must be scrupulously clean since they will be welded to the tube and must also withstand the corrosion environment. Therefore, the inspection and test procedures are more involved than the basic machining-done either on a screw machine or on an automatic lathe.

Zirconium and its alloys can be readily machined by conventional methods. Three basic parameters should be used for all machining operations on zirconium and its alloys: 1) slow speeds, 2) heavy feeds, and 3) a heavy flood of coolant lubricant. Care should be taken to minimize very fine chips as they are pyrophoric in nature. Zirconium exhibits a marked tendency to gall and work

harden. Therefore, higher than normal clearance angles on tools are needed to penetrate the previously work-hardened surface and cut a clean, coarse chip.

Satisfactory results can be obtained with both cemented carbide and high speed tools; however, the carbide usually gives better finishes and higher productivity. Polishing or honing the cutting edges will give the tool added life. Zirconium and its alloys machine to an excellent finish, requiring relatively light horsepower compared to alloy steel. The tool forces required are also relatively low.

Spacer grids for light water reactors can be put in three major categories: Those employing no zirconium components; those employing only zirconium; and those employing zirconium as well as some other material. The differences are based on several somewhat conflicting facts: Zirconium has a low neutron cross section compared to most alternates; its use would yield an overall increase in available power and in fuel efficiency. However, the performance of Zircaloy at high stresses under irradiation is not defined clearly enough to ensure satisfactory performance. The critical aspect of the latter point is that of stress relaxation -- the decrease of load because of plastic strain at constant deflection. Such stresses must be relatively high on spacers in order to properly locate the fuel pins in a grid and hold them rigid. Release of such stresses causes vibration and "fretting"--a decomposition of the oxide film and subsequent erosion of the underlying metal. Figure 21 shows the magnitude and rapidity of this stress relaxation. Proper grid design can take advantage of this relaxation phenomenon.

Composite grids take advantage of the low neutron cross section but use a non-zirconium component to apply pressure

against the tube. Tests are underway at several test locations aimed at overcoming the limitations of the all-zirconium grid.

One problem which occurs with perculiar intensity in production of Zircaloy spacer grids is the complexity of establishing maximum allowable bendability. The textural effects described earlier complicate the evaluation of this factor. The high apparent tensile ductility--about 30 percent--coupled with good simple bend radius--three times the thickness--have to be evaluated conservatively when developing a design involving threedimensional forming operations. Minor variations in production methods cause lot-to-lot variations which make some designs marginal. Extensive experience in forming of such parts is necessary in order to avoid difficulties related to this variation. A typical fabrication sequence for spacer strips is shown in Figure 23.

Fluid flow channels are used primarily in BWR reactor types; however, their manufacture gives insight as to the care and cleanliness needed to handle Zircaloy materials. A schematic flow diagram of such production is shown in Figure 22. Within the framework of this outline, much variation exists including: Intermediate shape and its methods of formation; weld method; sizing method; nature and method of stress relief, if any; extent of autoclaving; and many lesser details.

Several key operations performed at one facility are shown in Figures 24 and 25. Again, note the cleanliness of the work area.

The use of zirconium alloys for all structural components in the core, rather than the more commonly used stainless steel and nickel alloys could result in a significant increase in neutron economy, and thus in overall economics. Uncertainty over stress-

relaxation effects has caused most designers to move cautiously in this area. However, a recent study⁴ has shown that use of zirconium structural components may reduce stress on the individual fuel pins. This is especially significant since stress on the fuel pin appears to be the major design criteria limiting fuel life.

Associated Facilities

Associated with the production of Zircaloy mill products and components is the need for testing and inspection facilities. For reasonable scheduling and material flow, some of these facilities must be located at the site of production. Such operations include ultrasonic inspection (Figures 26 and 27), liquid penetrant, dimensional inspection, and inert gas analysis. Other test facilities need not be intimately associated with the fabrication facility, although it is expeditious to do so. These include: Chemical analysis, including classical wet chemical, spectrophotometric, spectrographic, gas analysis, and specialized instrumental techniques; corrosion test facilities, metallography, and mechanical tests, including room and elevated-temperature tensile test, bend or other formability tests. Some of these facilities and tests are shown in Figures 28 through 31. It is not essential that these test facilities be reserved for only Zircaloy products, but considerable standardization on zirconium is necessary and frequent recalibration is requested.

Costs

The current level of zirconium production is about five million pounds annually. Much of it is processed in special facilities such as you have just seen. (Dr. Larsson has noted that

most zirconium fuel clad tubing is produced in facilities designed and operated either primarily or exclusively on zirconium alloys. Even organizations producing other tubular products have seen fit to segregate their zirconium product from most other products.) Such segregation is not so critical for flat and solid round product, nor is it so feasible until the production volume grows significantly.

A large portion of what is saved by sharing facilities with, say, a tool steel operation, will be consumed in higher costs incurred in maintaining physical segregation of material. It has already been mentioned that the vacuum furnaces needed are not generally available in existing facilities. Blasting to remove oxide can be best accomplished with silicon carbide grit--not too common in most other metal working facilities. Aluminum oxide is difficult to completely remove in subsequent operations; steel shots and grits remove the visible oxide, but not the underlying contaminated metal. The HF-HNO₃ acid is common to other metals, but the standards of cleanliness are more severe.

The low modulus of zirconium alloys requires modification of some pieces of equipment relying on this property--straighteners, roller levellers, etc. As mentioned earlier, primary working tools are in every way adequate.

Two components of cost are peculiar to nuclear component manufacture: First, the high cost of quality and quality control and second, the high capital cost of utilizing expensive machinery on items in only intermittent demand. This latter aspect will change as the nuclear market expands, but even the large facilities in the major, nuclear nations are subject to intermittent low utilization factors. Other use of this equipment is discouraged

by the requirements of cleanliness, radiation control, material accountability, and material segregation.

One aspect of QC which is used to facilitate removal of deficient material is that of material traceability. All zirconium alloy components are identified as to origin--in terms of manufacturer, ingot, and sometimes section number--at least until fabrication of core components is initiated; often identification is maintained throughout reactor life.

Cost of such a material accountability system is high; but its ability to root out deficient, or potentially deficient, material is deemed to be worth the investment in view of the high cost of defective fuel.

The price of zirconium products has held relatively steady during the past decade. Inflationary tendencies have generally been balanced by increased efficiencies and by the increase in volume. Table I tabulates the general price structure of Zircaloy and the common high volume zirconium alloys. Values are shown in U.S. dollars and are, for volume quantities, exclusive of transportation, duty, and product testing. Included in the cost base are purity and quality standards normally required by the nuclear industry--items such as three to five analyses per ingot; intermediate quality surveillance; customer access to critical processing steps (except those considered to be proprietary); and product control, except those items delineated below.

Table II shows typical testing costs. As noted, many considerations can affect the actual cost. The prices shown assum currently standardized methods and controls.

It takes only a few calculations to realize that a large portion of the cost of Zircaloy products lies in testing and controlling of the product. In view of the criticality of the

cladding application, and considering the relatively low percentage of cost of a nuclear core spent on the Zircaloy products, it becomes apparent that these costs of quality increments are well advised.

CONCLUSION

The processes currently used for component production have been described in brief. It is not possible to become involved in all the technical aspects of this production. Overemphasis of the complexity of the operations is not intended; however, several efforts at establishing zirconium facilities have proved unsuccessful, and at least a portion of each facility's demise was associated with either quality of the product or the flexibility of an operation in adapting to changed conditions and volumes. Many changes are certain to occur in coming years as increased volume and rising costs force out many of the laborintensive operations we have seen today. But the basic nature of the market appears to be well established.

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31. Wet Laboratory.

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Figure 2



Figure 3

FIGURE 4

Effect of texture on the mechanical properties of channel blank.







Figure 5



FIGURE 7 PRODUCTION OF ROD FOR END PLUGS AND STRUCTURAL MEMBERS

STARTING MATERIAL: MULTIPLE ARC-MELTED INGOT

PROCEDURE AND CRITERIA **PROCESS SEQUENCE** PRESS, HAMMER, OR ROTARY FORGE 1000°C HOT WORK REHEAT, BETA-QUENCH WATER TANK SANDBLAST, GRIND, PICKLE TO CONDITION REMOVE OXIDE, LAPS, SEAMS INTERMEDIATE WORK ROD ROLL, EXTRUDE ROTARY FORGE 775°C AND/OR ROTARY SWAGE SURFACE CONDITION SANDBLAST AND ACID PICKLE "COLD" WORK ROD ROLL OR ROTARY SWAGE CLEAN, CONDITION DEGREASE, SANDBLAST, ACID PICKLE, AND/OR CENTERLESS GRIND ANNEAL AIR OR VACUUM GRIND TO FINAL SIZE CENTERLESS GRIND ULTRASONIC INSPECT IMMERSION PULSE ECHO TRIM ENDS SAW OR SHEAR DIMENSIONS, END DEFECT REMOVAL: FINAL INSPECT [LIQUID PENETRANT] ON A SAMPLE BASIS: TENSILE TEST HARDNESS GRAIN SIZE AND STRUCTURE C, O, N, H ANALYSIS AND ſ 1 DENOTE OPTIONAL OPERATION, SEQUENCES

MAY ALSO VARY

FIGURE 8 PRODUCTION OF STRIP AND SHEET FOR CHANNEL AND SPACER APPLICATIONS

> STARTING MATERIAL: MULTIPLE ARC-MELTED INGOT

PROCESS SEQUENCE

PROCEDURE AND CRITERIA

HOT WORK 1000°C

REHEAT, BETA-QUENCH

CONDITION

ULTRASONIC TEST

INTERMEDIATE ROLLING 775°C

AIR ANNEAL

CONDITION

COLD ROLL

CONDITION

VACUUM ANNEAL

COLD ROLL!

CONDITION?

VACUUM ANNEAL

SEQUENCE MAY REPEAT

CONDITION

FINAL INSPECTION

PRESS OR HAMMER FORGE

WATER TANK

SAND BLAST, GRIND, PICKLE TO REMOVE OXIDE, LAPS, SEAMS

IMMERSION PULSE ECHO

HOT MILL

AIR FURNACE

SANDBLAST, GRIND, PICKLE

WITH OR WITHOUT COILERS

DEGREASE, PICKLE, SLIT AND/OR SHEAR

. da

VACUUM FURNACE-HORIZONTAL OR VERTICAL, COIL OR FLAT

WITH OR WITHOUT COILERS

ROLLER LEVEL, BELT SAND, PICKLE, SLIT OR MACHINE EDGES

DIMENSIONS, SURFACE QUALITY; ON A SAMPLE BASIS: TENSILE TEST BEND TEST AND/OR FORMABILITY TEST HARDNESS GRAIN SIZE AND STRUCTURE C, O, N, H ANALYSIS

DENOTES OPTIONAL OPERATION





Figure 10



Figure 11



Figure 12








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Figure 16



Figure 17



Figure 18

FIGURE 19 END PLUG MANUFACTURING SEQUENCE

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1

STARTING MATERIAL: ANNEALED AND GROUND ROD

RECEIV	ING INSPECTION	
		•
MACHINE	TO SIZE	
FINISH FINAL	SURFACE;	

PROCESS SEQUENCE

÷

1

....

PROCEDURE AND CRITERIA

REVIEW MATERIAL CERTIFICATE, DIMENSION, OVERTEST PRODUCER

SCREW MACHINE OR AUTOMATIC LATHE

2

TUMBLE OR ACID PICKLE

ON A SAMPLE BASIS: DIMENSION WITH DIAL INDICATORS, OPTICAL COMPARITORS, MICROMETERS, ETC., METALLOGRAPHY, AUTOCLAVE FOR OXIDE APPEARANCE CHECK

2

INDICATES OPTIONAL OPERATION

2 g - F

Figure 20

In-pile relaxation of Zircaloy - 4

Reference: R.H. Kreyns & M.W. Burkart Journal of Nuclear Materials 26 (1968) 87





Figure 21

In-pile relaxation of Zircaloy – 4

Reference: R.H. Kreyns & M.W. Burkart Journal of Nuclear Materials 26 (1968) 87





FIGURE 22 CHANNEL MANUFACTURING SEQUENCE

STARTING MATERIAL: COLD ROLLED AND ANNEALED SHEET EDGES SLIT OR MILLED

PROCESS SEQUENCE

PROCEDURE AND CRITERIA

RECEIVING INSPECTION REVIEW MATERIAL CERTIFICATE, MEASURE FLATNESS AND DIMENSIONS, OVERTEST PRODUCER

PRESS BRAKE

DRAW FILE

MAY MOVE

FORMING

EDGE PREPARATION

FIXTURE

WELD

SEAM LEVEL

ANNEAL

SIZE AND STRAIGHTEN

CLEAN AND AUTOCLAVE

WELD TABS, TRIM

FINAL INSPECT*

VERTICAL VACUUM FURNACE

TURK'S HEAD, DRAW DIE, AND/OR HAND-HYDRAULIC PRESS

EITHER WORK PIECE OR WELDER

TUNGSTEN INERT GAS, METAL INERT GAS, OR ELECTRON BEAM

1-3 DAYS, 350-400°C, 100-1500 psi (7-100 ATM.) STEAM

TIG, MIG

WELD QUALITY, OXIDE FILM UNI-FORMITY, DIMENSIONS, DESTRUCTIVE TESTS (SAMPLE)

* MANY IN-PROCESS INSPECTIONS AND TESTS ARE ALSO PERFORMED

INDICATES OPTIONAL OPERATION

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FIGURE 23 SPACER MANUFACTURING SEQUENCE

STARTING MATERIAL: COLD ROLLED AND ANNEALED STRIP

PROCESS SEQUENCE

PROCEDURE AND CRITERIA

RECEIVING INSPECTION

REVIEW MATERIAL CERTIFICATE CHECK DIMENSIONS OVERTEST PRODUCER

DEGREASE, ACID PICKLE

PUNCH PRESS

RIGID FIXTURE

FORM COMPONENTS

DE-BURR

CLEAN

ASSEMBLE FOR WELDING

WELD

CLEAN

AUTOCLAVE

SIZE

FINAL INSPECT*

TUNGSTEN INERT GAS, METAL INERT GAS; HAND OR AUTOMATIC SEQUENCE

DEGREASE, WIRE BRUSH, AND/OR ACID PICKLE

1-3 days, 350-400°C, 100-1500 psi (7-100 ATM.) STEAM

WELD COLOR AND APPEARANCE, DIMENSIONS, DESTRUCTIVE TESTS (SAMPLE)

* MANY IN-PROCESS INSPECTIONS AND TESTS ALSO PERFORMED

INDICATES OPTIONAL OPERATION





Figure 25











Figure 29





MANUFACTURE OF FUEL ELEMENTS

FROM ENRICHED UO2 POWDER IN THE UK-A REVIEW

by J. Doran

ABSTRACT

Springfields Works is the nuclear fuel centre of British Nuclear Fuels Limited, a company formed from the United Kingdom Atomic Energy Authority in 1971 and which undertakes all large-scale production of nuclear fuels in the United Kingdom. This Works has a long experience in the manufacture of fuels for gas and water-cooled reactors.

The lecture describes in detail the fuel fabrication plants at Springfields used for converting enriched uranium dioxide powder to fuel elements and assemblies having a wide range of specifications. The plants comprise:-

- (a) A pellet production plant of 300 tonnes uranium per annum easily extendable to 500 tonnes uranium per annum and capable of operating at up to 4% enrichment, with a wide range of pellet specifications.
- (b) A highly automated stainless steel fuel canning plant, for pins of 5 to 20 mm diameter and 900 to 1400 mm long, primarily designed for Advanced Gas-Cooled Reactor fuel, with a capacity of 500 tonnes uranium per annum and easily extendable to 1000 tonnes uranium per annum.
- (c) Canning and assembly plants for long zirconium and long stainless steel clad fuel with pin lengths up to 4500 mm, each plant handling of the order of 100 tonnes uranium per annum and used to make different types of watercooled reactor fuel and fast breeder reactor fuel.

The lecture recounts the operating procedures and the quality and production control systems in use. Reference is made to the Health Physics and Safety precautions necessary, the uranium accounting procedures established and the cost control method used in large-scale nuclear fuel production.

1. INTRODUCTION

Springfields Works is the nuclear fuel centre of British Nuclear Fuels Limited, a Company formed from the UK Atomic Energy Authority in April 1971. This Works has a long experience in the manufacture of fuels for gas and water-cooled reactors.

The fuel fabrication plants converting enriched UO₂ to finished fuel elements use a number of processes to make fuels with a wide range of fuel specifications. These plants comprise:

- (a) A pellet production plant of 300 tes U/annum capacity easily extendable to 500 tes U/annum and capable of operating with up to 4% enriched materials, with a wide range of pellet specifications.
- (b) A highly automated stainless steel fuel canning plant, designed primarily for Advanced Gas-Cooled Reactor fuel and capable of handling pins in the diameter range 5 to 20 mm, and in lengths of 900 to 1400 mm. The plant capacity is at present 500 tes U/annum with facilities for easy extension to 1000 tes U/annum.

(c) Canning and assembly plants for long zirconium and long stainless steel clad fuel elements with pin lengths up to 4500 mm. Each plant can handle of the order of 100 tes U/annum as finished fuel and they have been used for the UK Steam Generating Heavy Water Reactor (SGHWR) fuel, Boiling Water Reactor (BWR) and Pressurised Water Reactor (PWR) fuels, and Fast Reactor Breeder fuel.

The following account gives some details of the plant and processes used, together with operational features of interest.

2. MANUFACTURE OF SINTERED ENRICHED UO2 PELLETS

2.1 Introduction

The manufacture of sintered enriched UO_2 pellets is basically the same for any type of fuel element in that enriched UO_2 in a suitable form for feeding to a press is converted into "green" pellets which are subsequently sintered to achieve the desired density and in most cases ground to achieve the dimensional requirements. Finally, detailed inspection is carried out.

The reactor requirements and fuel element design determine the precise specification for the fuel pellet and this in turn determines the particular choice of process variants selected for the pellet manufacturing route. The Springfields Works pellet plant has of necessity been designed mainly to meet the high density, 10.6 to 10.7 g/cc, required for AGR fuel.

2.2 Ceramic UO₂ Powder Granulation

The UO_2 feed material to the presses must have good pour qualities, flowing freely and evenly into the press dies thus ensuring a constant die fill at high operating speeds.

Variations in the quantity of granules deposited in the die will affect the final product in some way depending upon the type of press being used, for example a mechanical press ensuring essentially constant pellet length or a hydraulic press ensuring constant green density.

In order to achieve these flow characteristics, the homogenised ceramic uranium dioxide powder is first granulated. This can be achieved via a so-called binderless route by pre-pressing the powder to briquettes and fracturing and sieving these to produce granules for pelleting. The preferred route in BNFL, however, uses a proprietary binding agent which amongst other advantages produces a green pellet of good mechanical strength. The first stage of the granulation process is to increase the specific surface area and activity of the powder by grinding it in a high pressure air fluid-energy mill. The mill used is a vertical toroid in which the uranium dioxide powder is violently circulated by jets of dry oil-free air. When the particles have been reduced in size by attrition, they are removed from the mill in the effluent air stream which is exhausted through a filter system.

The milled uranium dioxide powder from the filter system is continuously slurried with solvent and binder. To achieve the desired size distribution of the final granules, the slurry density is controlled within narrow limits. The slurry is pumped continuously to a spray drier through a swirt nozzle into the spray chamber where it is dried by preheated air. The effluent air with trichloroethylene vapour is filtered and the trichloroethylene recovered. The uranium dioxide granules are continuously discharged from the base of the spray drier and classified through a series of vibratory screens.

2.3 Performance Testing of UO₂ Granules

Batches of granules will vary in quality and it is considered essential to carry out a performance test under standard conditions on each granule batch before it is accepted for further processing. A small sample is pressed over a range of standard compaction pressures on a laboratory press and the resulting green pellets together with a standard green pellet for comparison are debonded and sintered. The sintered densities and diameters of the green pellets are measured and indicate if the batch can be accepted within the process limitations available. If so, the data is used to select the optimum pressing pressure and to determine the most economic die size.

2.4 The Conversion of Granules to Pellets

The granules are continuously fed to 60 tonne hydraulic, multi-punch pelleting presses, operating under closely controlled conditions of pressure, die fill, dwell time, etc. The pellets, which can be cored or solid, are then fed to debonding furnaces where binder removal takes place at 800°C. From there the pellets are fed through sintering furnaces where densification takes place at 1650°C in a hydrogen atmosphere. Following diameter grinding, the pellets are closely inspected for specification parameters. The important properties usually included in the specification for pellets are :

Uranium Content, Stoichiometry, Isotopic Abundance, Chemical Impurities, Neutron Capture Cross Section, Moisture Content, Gas Content, Sintered Density, Surface Defects, Structure and Dimensional Parameters. The chemical specification requirements are largely determined at earlier stages of manufacture although there is usually some removal of volatile impurities during sintering. It is, of course, essential to take precautions against contamination by, in particular, high neutron capture elements as well as the more obvious machinery contaminants such as oil and grease. Although isotopic abundance is not affected by the pelleting process, enrichment integrity is obviously vital and good house-keeping and satisfactory movement control are essential.

A wide range of pellet densities are currently specified ranging from about 90% theoretical density for some water reactor fuel to over 93% theoretical density for gas-cooled reactor fuel. A number of consequences follow from the particular density range quoted in the specification since the greater porosity of the lower density fuel makes the pellets more prone to high moisture and gas contents. Additionally, there is the problem of further pellet densification during irradiation and possible cladding collapse.

Small adjustments of pellet density can be achieved during pellet processing by varying press conditions and furnace temperatures and times but in the main, pellet density is dependent upon the inherent properties of the granules, which can be made from powder of high or low activity with or without binders and additions of various types. All these variants can affect the type and amount of porosity that is produced in the pellet. Specifications of pellet structural requirements are in general more qualitative than quantitative in nature and call for control of grain size, grain boundary porosity, within-grain porosity and macro porosity. All of these have an effect on the release of fission product gases and the swelling of the UO₂ under irradiation. The individual properties required for fuel in a particular reactor system are influenced by the working temperature, fuel rating and the fuel pin design concept.

The macro porosity requirements can be achieved by the controlled use of additives, the porosity varying within a matrix of standard structure. Finally, dimensional control is needed both for the finished product and process economy. For example, strict control and careful application of the results of granule performance tests combined with optimum choice of die size can markedly reduce the amount required to be ground off. As pellet chipping is the most serious cause for rejection in AGR fuel, much care has been given to the design and operating of process and handling equipment. However, chipping standards vary with different reactor requirements and in general are more severe for gas-cooled reactor systems.

2.5 Finishing of Pellets

Almost without exception, pellets are finished by grinding the cylindrical surface to dimensional tolerances that will meet the fuel pin specification. This operation is carried out on standard centreless grinders. Various claims are made of the advantages of either silicon carbide or diamond impregnated cutting wheels but the effect is marginal. The real aim is the achievement of the requisite surface finish and dimensions at the least cost in time and rejections. Skilful press control can yield dividends at this operation. There is an optimum stock removal per unit time consistent with machine utilisation and damage rejection rate.

Too much stock to be removed can imply double or even triple passes and excessive damage losses. Control of the press tools to match the sintering properties of a particular granule can make a large difference to the ease with which this operation can be performed.

2.6 Quality Control and Final Inspection

In the manufacture of UO_2 pellets, large numbers are being dealt with and each must comply with a tight specification. Thus for the CAGR fuel there are about 50,000 pellets per tonne U of fuel and higher quantities are more usual for water reactor fuel of smaller diameter.

Since the consequences of fuel failure during irradiation far outweigh fuel fabrication costs, it is imperative that adequate quality control and inspection procedures are employed.

Consequently, many features require 100% inspection, but some are amenable to statistical quality control schemes as serious deviations are unlikely to occur and by their nature would tend to be systematic and thus easy to detect. Quality control schemes used must give a very high confidence and are normally introduced at the most probable sources of rejection to assist in process control. Thus, for example, pellet length can be maintained by controlling the press and pellet diameter by controlling the grinding machine. These are simple examples but parameters such as density, chemical analysis and enrichment require special treatment and much more sophisticated control schemes. Some of the features requiring 100% inspection, e.g. surface condition, are not amenable to mechanical methods and inspectors become expert in detecting very low incidences of surface defects even when subjective standards are used. Other features, such as chipping, can be inspected mechanically and machines operating on the pneumatic pressure drop principle are used to inspect pellets for chipping. They work well on CAGR fuel pellet standards which are higher and more closely defined than water reactor pellet standards.

2.7 Production Control Procedures

Most reactor systems require enriched fuel, and large nuclear fuel fabrication plants producing fuel for several types of reactors handle a multiplicity of enrichments using feeds of granules of varying quality to meet different fuel specifications.

This raises several problems such as criticality control, enrichment integrity and inventory control which can only be overcome by sophisticated organisational and control procedures. The cost of this control is appreciable but the consequences of error are so severe both for fuel manufacturers and the reactor operators that there is no tenable alternative.

The over-riding danger of a possible criticality excursion sets the basic need for stringent formalisation of movement and storage control procedures and documentation. It is also necessary to establish proper warning systems and evacuation procedures which, to be effective, must be tested at reasonably frequent intervals.

On the maintenance of enrichment integrity, the consequences of a gross enrichment mix could be very serious for the reactor operator as well as the fuel manufacturer, and a serious incident would cause loss of public confidence in the industry as a whole. Consequently, close control and good house-keeping are essential. Further, the value of the enriched material being processed is often in the region of £100,000 per tonne U. Thus several tonnes of such material held as work in progress can incur very high interest charges. The difference between good and bad inventory control could, in the extreme, cost more than the profit margin of the fuel manufacture. Again, because of the high costs of individual pellets and their liability to chipping, there is considerable justification in spending time and money on careful handling procedures.

MANUFACTURE OF FUEL PINS

3.1 Introduction

Basically, fuel pin manufacture comprises the encapsulation of a stack of UO_2 pellets in a metallic sheath of stainless steel or zircaloy. The containment consists of a tubular can closed at each end with a cap which is welded into position. There must be some system for ensuring the integrity of the welds and the leak tightness of the whole pin. The degree of final inspection varies with individual fuel specifications.

Typically fuel pin specifications will include such items as pellet stack lengths and weight, dimensional parameters, helium content, leak tightness, absence of pellet gaps, surface condition and contamination.

3.2. Pellet Stack Build-up

There are significant differences between the pellet stack assemblies for water and for gas-cooled reactors. Most water reactors have a large gas plenum and the pellet stack length has a full pellet tolerance. Pellet stacks for gas-cooled reactors, however, present a more difficult problem as every fifth pellet is grooved and there are tolerances on the groove positions as well as the overall length. In fact a statistical method of pellet length control has been evolved by which it is possible to make up a stack of 64 pellets to a tolerance on length of $\pm .015$ ". It is most important in the case of gas-cooled reactor pins to inspect the pellets carefully at stack build-up to make sure that no chipping has occurred during the transit from the Pellet Plant or during stack build-up preparation, as any one chip larger than the permitted standard will result in a reject pin for indentation where the can wall has been forced into the chip during subsequent pressurisation. Pellet stack weight tolerances are usually determined by the pellet density specification and present little difficulty, the actual weights being required for accountancy purposes.

3.3 Fuel Pin Fabrication

All components used for canning are the result of long programmes of development and testing involving many design changes before a sealed reference design is achieved.

Materials have to be submitted to a reactor environment as well as the normal mechanical, physical, creep and other standard engineering tests and metallurgical proofs The end products of all this work are very closely specified from the metal refining stage to the finished component. They are subjected to extremely stringent inspection, much of which is double-checked by independent laboratories to ensure a high degree of confidence.

As fuel manufacturers, BNFL are closely involved in this work and act as the focal organisation in a large complex of Research Establishments, Design Offices and Manufacturers. Many of the manufacturers are completely dependent upon BNFL expertise in advising on methods of fabrication and control.

All components go through very thorough cleaning and degreasing treatment before being introduced to the production line.

The loading of fuel stacks into the can is a straightforward procedure but can result in high losses if care is not exercised, particularly with the thinner walled gas-cooled reactor cans. The pellets are loaded through orifices in a sealed partition which separates the "active" from the "nonactive" areas, thus minimising spread of contamination. The pellet loading operation is more difficult in the case of gas-cooled reactor fuel as the pellet/can clearances are usually much smaller because the can has to be pressurised onto the pellet stack without wrinkling. Automatic machine loading has been developed on the CAGR fuel line but manual attention is also required. High ventulation of loading orifices is advantageous from the Health Physics point of view.

The two most important succeeding stages are filling with helium and end closure by welding. Helium is used for two reasons, firstly it acts as a good heat transfer medium in the reactor and secondly it is a very satisfactory medium for leak testing by mass spectrometry.

It is essential to know that helium is present in the pins and either very elaborate self-checking filling techniques must be used or a radio-active tracer gas may be added.

Subsequent to helium filling, it is necessary to seal the closures and this is invariably done by one or another welding process. For stainless steel pins, TIG welding, resistance welding and electron beam welding are all fairly well established processes: It is well known, however, that the welding of zirconium especially by the TIG process is particularly prone to porosity and BNFL has adopted upset resistance welding with magnetic force, i.e. a variation of conventional upset resistance butt welding whereby a highly responsive electro-magnet is used to supplement the normal pneumatic or hydraulic force. By this means, the timing, duration and increase in magnitude of the force supplied during welding can be accurately controlled.

The welding process then becomes:

- (a) Heating the components to an interfacial temperature below the melting point but in the plastic region under the influence of a low applied force.
- (b) The progressive coalescing of the interfaces under an increasing force.
- (c) The completion of the weld by upset forging accompanied by the extrusion of plastic metal under a greater force.

The resulting weld exhibits a highly localised region of fine hot forged grains. The welding current, the mechanical force current and the upset can be monitored for every weld, ensuring consistent standards.

On stainless steel pins, X-radiography is used where practicable and on the automatic roller resistance spot welders, quality is controlled by monitors which halt production in the event of under-rotation or deviation in the number or amplitude of weld pulses.

In addition to these controls, a mass spectrometer test for helium leakage is invariably specified and undertaken.

Pins for the gas-cooled reactors are externally pressurised to lock the cans onto the pellets, every fifth of which carries a groove to provide an anti-stacking device and to even out any gaps along the can rather than accumulate them at the upper end. Water cooled reactor pins in zircaloy do not need this procedure because of the much better heat transfer properties of the coolant and the presence of a large gas plenum. Inde. d it is becoming more usual to pressurise the pins internally to prevent collapse of the can. Heat treatment and protective coatings are variously specified and are carried out in specially designed equipment. Zircaloy pins in particular are frequently specified with a black lustrous oxide film. This is produced by pickling the tubes in nitric/HF acid mixture followed by autoclaving in steam or CO₂ at high temperature. Autoclaving was originally introduced as a test for the purity of the zircaloy as well as to establish a stable oxide film before introducing the pin to the coolant, but the practice is tending to become obsolete as there is little proof of its effectiveness. It is not now currently being used for SGHWR fuel.

The finished pins must be as free as possible from external contamination and statistically sampled pins are measured in an ion chamber, for the presence of such contamination. Less than one millionth of one gram of uranium can be detected by this means A variety of techniques are used for the decontamination of pins ranging from simple acid pickles to electrolytic techniques. As in the case of pellet manufacture, much in-line control is exercised on the canning lines and a very thorough final inspection is carried out before a pin is released for assembly into a fuel element or cluster. This includes a 100% critical inspection for dimensional features and a 100% visual subjective examination for damage or contamination.

Non-destructive testing to prove the absence of inter-pellet gaps is performed and also an enrichment confirmation check by a Gamma rate meter device which distinguishes between the U_{235} energy levels and those of the daughter products which vary with age from chemical processing. The finally inspected fuel pin is then passed forward for assembly into fuel elements.

4. FUEL ELEMENT ASSEMBLY

The fuel element assembly, or cluster as it is sometimes called, is, of course, particular to the reactor concerned. All manner of pin arrays, circular, square and hexagonal have been used.

Assemblies must be designed such that build-up can take place using fairly simple engineering techniques giving a very high confidence that the specification can be met as any rejection at this stage is unacceptably expensive. The main essential is in the line-up and accuracy of the jigs which need to be of rigid construction. The normal procedure is to set up one or both end fittings and intermediate support grids very accurately in the jigs and to fasten them in some way, usually by riveting or welding, to support pins or tubes. Fuel pins are then inserted and these are fastened to or constrained by the end fittings.

In the case of the gas-cooled reactor assemblies, the pins and grids are supported by a graphite sheath, which also provides a proportion of the moderation. A double skin sheath is used to limit the outside temperature and thus the corrosion of the moderator.

Some reference has already been made to the nature and high cost of the majority of components involving lengthy development and testing and the aspect of time needs stressing as up to two years may be necessary to obtain the more sophisticated components. Because of this, the major parameters have to be fixed in good time if the final start-up date is to be met and detailed centralised planning is essential.

The packing of such assemblies for transport to the reactor site is expensive as the containers need to be very substantial to give adequate support and protection to the contents and must also be burst-proof, fire-proof and water-proof to meet criticality and uranium containment requirements under credible accident conditions.

5. HEALTH PHYSICS AND SAFETY

Originally all development and manufacture of nuclear materials in the UK was under Government control. Health Physics teams were set up at each station with advisory and inspection functions as well as departmental responsibilities for the up-keep and issue of protective equipment and clothing. They were combined with the departments set up to deal with the non-nuclear safety aspects. As these features began to pass from direct Government control, legislation was enacted to define acceptable operational limits and methods of control and a Nuclear Installations Inspectorate has been established. The Health Physics regulations are rigorously applied and add to the cost end complexity of nuclear fuel manufacture.

The prime danger with enriched uranic materials is always that of criticality and all working operations are carried out to strict rules in compliance with a comprehensive criticality clearance certificate, without which no work can be undertaken. Wherever possible, safe by shape designs are used but there are many operations where compliance with the regulations depend upon adequate training and supervisory control.

As mentioned in the section on pellet manufacture, criticality alarm systems are installed which respond to detection of neutron activity. Immediate evacuation is mandatory when such an alarm sounds. To avoid secondary incidents arising because automatic machinery has been left functioning but unattended, consideration has to be given to means of shutting down the plant from outside the danger zone.

Operations involving uranium powders are hazardous because of the possibility of ingestion and have to be strictly controlled by adequate containment and extraction, the use of protective equipment and clothing being necessary particularly on maintenance work. Large quantities of enriched material could also give rise to high radiation doses and these have to be regularly monitored to ensure safe working conditions.

Strict monitoring of all effluents, solid, liquid and gaseous is maintained to avoid pollution of the environment, and the standards set in the Works should meet any new regulations which might arise from current world wide discussions on this subject.

Air filtration out of uranium processing plants is the normal rule and for fuel element and pin manufacturing the incoming air is also filtered, minimising external contamination. Changerooms with showering facilities have to be provided in uranic processing plants, and periodic medical examinations are undertaken of each person employed in an active area.

6. URANIUM ACCOUNTING

Due to the very high intrinsic value of enriched uranium and its potentially dangerous nature, a material balance account is essential in any uranium processing plant.

Apart from detailed transfer documentation, analytical methods and assessments, weighing and metering precision and competent stock-taking all play their part. Recent international control agreements on safeguards require the disclosure to authonsed agency representatives of details and precision of control systems, materials balances and analytical techniques. These requirements are unique to the industry and involve a great deal of management time and effort as well as appreciable capital expenditure.

7. COST CONTROL

Manufacture of nuclear fuel elements is no different from any other manufacturing activity to the extent that profitability benefits from the application of normal cost control procedures. Capital and Operating budgets are produced and cost performance is regularly monitored.

On the shop floor it is essential to high-light the expensive features and, as might be expected, cost reduction is significantly greater by improvements in efficiency at the tail end of the manufacturerather than the early stages, thus each per cent improvement in uranic efficiency is twice as effective in cost reduction at pelleting and five times as effective at canning and assembly as it would be at the hex reconversion stage.

Comment has been made of the necessarily high integrity and consequently high cost of components and it may be of interest to note that component costs (not including the value of the uranium) amount to about 85% of the total canning and assembly costs of CAGR fuel.

Another indication of the importance of component costs is the fact that fuel element component costs amount to approximately one third of the Springfields Works operating budget.

Peculiar to the nuclear industry are the stringent quality control procedures necessary to meet fuel specifications and to meet nuclear health physics requirements. The costs incurred by so doing are estimated at Springfields to be approximately 6% to 7% of the Works Operating Costs.

NUCLEAR FUEL FABRICATION

by

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Abstract

Nuclear fuel fabrication is the first step in the fuel cycle which cannot prudently be treated as a commodity by the purchaser. Fabrication not only involves the manufacture of a complex product to very close tolerances but also requires the use of very sophisticated design tools, great emphasis on quality control, large inspection staffs, and very precise materials accounting and other data processing.

In design, a great deal of time and large expenditures are required to develop the necessary methods and tools. This involves not only development of the basic calculational procedures and computer programs but also feedback of operational data to improve the accuracy and precision of the design predictions. Five years or more may elapse from the time a fuel region is designed until complete data on its operation become available. We at Westinghouse have been involved in fuel design for more than fifteen years and have accumulated an extremely large body of operating data. We still find it necessary to fine tune our methods as additional data become available.

Because the fast breeder reactor will not be commercially available for more than a decade, the need for plutonium recycle in thermal reactors is obvious. This need adds an additional degree of complexity to fuel design and core management. Without a depth of design knowledge, the fuel supplier cannot hope to design fuel regions containing recycled plutonium which will meet the utility's power and operational requirements.

In a rapidly changing technology such as nuclear fuel, very large expenditures for product development are necessary. We at Westinghouse apply fully 20 percent of our total income to development, and we expect to maintain that level indefinitely. Without a very substantial sales volume, a fuel supplier cannot support development at the necessary level.

Like most manufacturing processes, nuclear fuel exhibits a significant economy of scale. Having operated both a relatively small plant at Cheswick and the world's largest water reactor fuel plant at Columbia for a number of years, we are in a very good position to evaluate the scale factor. This scale factor obviously applies

to the cost of the manufacturing facility and to direct production costs. Equally important, and probably more so, is the fact that the very large support staffs required can be used much more efficiently in a large volume operation. Fully half of our manufacturing operations staff and a significant proportion of our engineering staff are involved in support areas such as quality control, health physics, materials accountability. A smaller supplier would require a higher proportion of staff which would not be used as efficiently for these areas, or would not apply the effort in these areas we think is required. Other overhead costs are similarly reduced proportionally with larger volume.

In manufacturing, too, plutonium adds complexity. Because it is toxic, it must be handled in an enclosed space, for which reason increased mechanization is desirable. Because of its toxicity, environmental considerations have a much greater impact on the design of a plutonium manufacturing facility than on a uranium plant. All of these considerations result in a substantially higher cost for a plutonium plant and, to a lesser extent, in higher production costs than for uranium fuel.

This paper describes the evolution of the present fuel design, how that design is used to satisfy individual utility operating needs, and the manufacturing methods and processes in use today. Because of the many complex and interrelated factors cited, nuclear fuel supply by relatively few large facilities seems far preferable to a great many smaller facilities serving local markets. Many of the benefits of scale economy can be available to smaller satellite operations through joint ventures.

INTRODUCTION

Nuclear fuel fabrication is the first step in the fuel cycle which the purchaser cannot prudently treat as a commodity. Fabrication not only involves the manufacture of a complex product to very close tolerances, but also requires the use of very sophisticated design tools, great emphasis on quality control, large inspection staffs, and very precise materials accounting and other data processing.

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Figure 1 illustrates the processes involved in fuel fabrication and the relationships among them. As knowledge accumulates in each area, it provides useful inputs to the other processes. Research and development is the central process which provides a foundation for the others and integrates the knowledge and experience into a coherent body.

This synergistic process requires a great deal of time and very large expenditures. For example, five years or more may elapse from the time a fuel region is designed until complete operational data become available for feedback into the design and manufacturing processes. At the time of writing (June, 1972) sixteen reactors are operating using Westinghouse supplied fuel. The first of these has been operating since 1957. During those fifteen years, the fuel has generated some 100 billion kwhrs of electrical energy. The data accumulated from that operating experience have been continually used to improve our design and manufacturing methods and techniques. That process continues today at an accelerated rate. As a result, we are able to predict nuclear characteristics and power distributions with remarkable accuracy and expect continued improvement as additional experience accumulates.

DESIGN

The basic fuel assembly design integrates the diverse demands of the various engineering disciplines to provide economic and reliable fuel operation. The goal of the nuclear designer is to minimize non-fuel material in the core since it absorbs valuable neutrons. The mechanical designer must assure that the assembly maintains structural integrity throughout its lifetime. This is most easily done with large amounts of structural material. The thermal-hydraulic engineer specifies the addition of mixing vanes to the grids to assure good heat transfer. These, however, also absorb neutrons and require the mechanical designer to exercise great care in assuring that they reliably perform their intended function. The present design has evolved to meet, in the best way possible, these sometimes conflicting requirements.

In using the basic assembly design to provide a first core or reload region, the primary objective of the nuclear engineer is to satisfy economically the operating requirements the utility specifies. The utility needs to accommodate changing operating requirements while at the same time meeting shutdown schedules that avoid refueling during peak load periods. Accordingly, the first step in designing a core or region is to consider the utility's future operating requirements. These include the cycle length and the planned manner of plant operation.

Figure 2 shows a 'schedule of events for the design of a typical reload region. The utility tentatively sets operating requirements two years before the region is to be loaded. Final requirements must be set approximately one year before region loading to allow ample time for uranium procurement, conversion, and enrichment. Once the cycle operating requirements have been set, the nuclear designer has three major variables with which to work: number of assemblies, enrichment, and arrangement of the assemblies.

The number of new assemblies loaded is the primary mechanism for making major changes in the cycle length over a long period of time. The enrichment of the reload region can be varied to make small modifications in cycle lifetime. Finally, the arrangement of the fresh reload region and the partially irradiated assemblies remaining in the core can be varied to meet power peaking restraints. Because the core always contains some partially irradiated fuel, the designer must account for prior operating history in addition to future requirements.

All three variables affect the cycle lifetime and the power distribution in the core. Since it is possible to select a combination of the first two variables which would preclude any core loading; pattern from satisfying the power distribution criterion or other, safety-related criteria, all three variables must be considered before a final design specification is made.

The selection of the final loading pattern is essentially an iterative process. That is, the designer selects a pattern for detailed analysis and calculates the resulting power distribution. This process continues until a selected pattern falls within the criterion for maximum power peaking. Since the number of possible patterns is very large and each calculation requires a great deal of computer time, the designer must bring considerable experience and engineering judgment to bear in selecting the patterns for analysis.

By virtue of its accumulated fuel design and operating experience, Westinghouse is able to specify very precisely the number, enrichment, and arrangement of assemblies

necessary for a given cycle operating requirement. Because of this great body of experience, the Westinghouse designers have also been able to reduce substantially the number of iterations required to arrive at a satisfactory loading pattern. It is also possible, when necessary, to revise the selected pattern very quickly. When problems have arisen during first core loading or refueling, Westinghouse was able to resolve them expeditiously by providing new loading patterns, literally, overnight.

Reload regions must been in the relatively near future to incorporate plutonium recycle fuel. This is so because the fast breeder reactor will not be available commercially on a large scale before the latter part of the 1980's. At that time, the water reactors already installed will generate more than enough plutonium annually to supply the fueling requirements for any conceivable breeder installation rate. There is, therefore, no incentive for stockpiling plutonium in anticipation of breeder requirements. In fact, as plutonium becomes available in large quantity, there will be great economic pressure to use it as soon as possible after the spent fuel has been processed.

Including recycle fuel in a reload region adds greatly to the complexity of the nuclear analysis. The nuclear characteristics of plutonium in a thermal spectrum cause increased power peaking in a recycle region unless corrective design measures are taken. Of the alternative solutions available, Westinghouse has selected the discrete assembly concept as the most promising technical and economic choice. In this concept, a reload region is composed of two kinds of assemblies with identical mechanical designs. One type contains a uniform mixture of plutonium and natural uranium. These are interspersed with the second type, which contains only uranium of the same enrichment which would be used in a standard uranium reload region. In the assemblies containing plutonium, it is necessary to use multiple plutonium concentrations selectively distributed throughout the assembly to maintain acceptable power distribution. It is also necessary to exercise greater care in selecting the locations for the assemblies in a recycle region.

MANUFACTURING

The importance of having a sound, integrated manufacturing process and a comprehensive quality assurance program cannot be overstressed. Consider, for example, the core for a modern four loop plant. The fuel rods alone require approximately 80,000 welds, contain close to 10,000,000 UO₂ pellets, and use about 100 miles of Zircaloy tubing. Rod inspection must assure, with a high degree of confidence, that all the welds are acceptable, that the pellets are within specification, and that the tubing contains no flaw that might cause a leak. There are similar requirements for grids and nozzles, and for the assembly of these components into the finished product. In fact, the quality control checks made on each assembly number in the tens of thousands.

Westinghouse began its large scale manufacturing experience at its Cheswick, Pennsylvania facility. This plant was expanded as need dictated to a capacity of 300 metric tons a year. Late in 1969, a second plant was opened in Columbia, South Carolina. The Columbia plant has 210,000 square feet devoted to manufacturing and 50,000 square feet for office space. It has an ultimate capacity of 1000 metric tons a year, the equivalent of twelve four loop plant cores or thirty-six reload regions per year. The Columbia plant performs all operations necessary to manufacture finished nuclear fuel assemblies. It is the world's largest facility devoted to the manufacture of light water reactor fuel. Even though the Cheswick plant is the world's third largest, it is now being phased out of operation in favor of the Columbia plant, which yields still lower product cost.

The manufacturing operation begins with chemical conversion of uranium hexafluoride (UF_6) gas to uranium dioxide powder (UO_2) . UF₆ gas is withdrawn from the gaseous diffusion plant in the proper enrichment and shipped to Columbia in 4800-pound containers.

 UF_6 is a solid at normal temperatures. It is sublimed in a steam chest and fed into the process, Figure 3. Water is mixed with the gaseous UF_6 to convert the gas into a manageable form that permits precise process flow control. This reaction generates hydrofluoric acid and uranyl fluoride solution. The solution is pumped to a precipitation tank where it reacts with ammonium hydroxide. There, the uranium precipitates out as ammonium diuranate (ADU), a yellow solid.

The slurry is pumped into a horizontal-bowl centrifuge, Figure 4, where the solids are separated. The solids are then fed into a rotary calciner. The overflow solution still contains a small amount of solids. Therefore, it is channeled into a very-high-frequency centrifuge where the balance of the solids is removed.

Following this, the solutions are held in quarantine tanks. Each tank is individually checked for uranium content before it is released as waste. This check makes sure that none of the uranium has escaped the separation process. It is a quality check, a pollution control check, and an economic loss check.

The waste solution contains ammonium fluoride, which cannot be discharged into the waste stream. The waste solutions are reacted with lime to form calcium fluoride, which is insoluble, Figure 5. The calcium fluoride is precipitated out. Thick slurries from the precipitator are filtered off and buried in a land fill.

When UO_2 is discharged from the calciner, it is pulverized in a hammer mill to get the desired small final particle size. It goes through several quality control inspections to check the final material before it is released to the pellet area.

A direct-reading emission spectrometer is employed to analyze impurities in uranium dioxide as well as in the metals employed in fabricating nuclear fuel. Tests performed are capable of detecting concentrations of impurities as small as three parts in ten million.

A thermal-emission mass spectrometer is used in the detection of the isotopes of uranium, plutonium, lead or other elements with low-ionization potential. This instrument is capable of measuring samples as small as one hundred millionth of a gram.

Slugging (low-pressure pressing) shapes the powder into a form resembling a coin about the size of a United States quarter. The pressed "slugs" are granulated, or crushed into small particles, and passed through a screening operation, which provides a properly balanced particle distribution. This improves the flowability of the powder, which is a requirement in obtaining proper die feed for the pellet presses.

The screened granules are fed into high-speed presses, Figure 6, where the final fuel pellets are made. Control is maintained over the pressing operation so that pellets of uniform unsintered (green) density and size are manufactured. The green density determines the final size of the fuel pellet.

The green pellets are now sintered, Figure 7, at high temperature in electric furnaces; this process consolidates the aggregate of fine particles and results in shrinkage and densification of the pellet. Rigid process controls are maintained over the sintering process so that consistent final densities are achieved in the fuel pellets.

In the pressing and sintering operations, the fuel pellets are purposely formed slightly larger than the final size desired. To achieve this final size, each pellet must be ground to the exact dimensions required. Grinding is performed on precision centerless grinders. Tight process parameters are used to maintain control of pellet dimensions. After grinding, Figure 8, the pellets are transported to the quality control area for inspection. The material which is ground off the pellets is collected and reused.

Tubing is fabricated from Zircaloy ingots produced to Westinghouse specifications. These specifications go beyond those for standard tubes and place extremely stringent requirements on metallographic structure, physical and chemical properties, surface characteristics and dimensions. The tube is thoroughly inspected for integrity, using such nondestructive techniques as eddy current or ultrasonic tests. Westinghouse manufactures tubing made of Zircaloy at its Specialty Metals Division plant in Blairsville, Pennsylvania. Uranium fuel pellets are inserted into the Zircaloy tubes at the fuel rod loading station, Figure 9.

End plugs for fuel rods are precision machined of Zircaloy. After the tube is loaded, the end plugs are inserted in the tube. Magnified dimensional inspection of fuel rod end plugs and nondestructive testing assures the end product meets quality and reliability standards.

Prior to insertion of the top end plug, a spring is inserted into the fuel rod and the axial gap is checked against design requirements. The gap permits fission gas accumulation and axial thermal expansion of the fuel column.

The tubes are sealed by programmed TIG welding of the end plugs, Figure 10, to each end of the tube. A unique welding machine is used which performs the entire welding cycle automatically. After the electrode is set exactly at the proper location on the tube, the operator merely pushes a button and the welder does the rest. When it stops, the operator removes the rod.

After the rods are welded and leak-tested, they are conveyed to the quality control area, where they are X-rayed in a highly automated unit, Figure 11. After X-ray, the rods receive 100 percent dimensional and visual checks. The welds are checked for burrs, pits, gouges, discoloration. The rods are rolled on a flat-surfaced table, to make sure they are not bent or bowed. They are measured against a length standard. In addition, fuel rods are weighed to ascertain the amount of uranium contained as an accountability check.

The fuel assembly support structure consists of rod cluster control guide thimbles and spring-clip grids. The grids, Figure 12, are produced from individual grid straps which have been punched and slotted. Completed grids are inspected for proper spring tension and for dimensional accuracy. Grids are fastened to the control rod guide thimbles and the bottom end plate nozzle.

The fuel rods are supported axially along their length by the spring-clip grid assemblies. Several of the grids are mixing grids which increase the degree of intermixing coolant flow within an assembly and also between adjacent assemblies. The mixing action of these grids reduces the temperature gradient within and between fuel assemblies. Each fuel rod is supported in two perpendicular directions by formed spring clips. This support firmly restrains the fuel rod and prevents vibration. Perpendicularity and pitch dimensions of the grids are checked very carefully. Each grid is subjected to over eight hundred individual measurements.

An assembly of bottom nozzle, thimble tubes, grids, and adapter plate provides the framework which receives the loaded fuel rods and top nozzle — thus creating the final fuel assembly. Fuel rods are loaded by inserting them through the grids. During assembly operations, Figure 13, a fine spray is directed onto each rod as it is inserted into the grids. After rod insertion, the top nozzle is welded to the control rod guide thimbles to complete the fuel assembly. The top nozzle is engineered to support the entire fuel assembly as it is lifted and placed into the reactor.

The completed fuel assemblies are inspected and cleaned using an appropriate material. Except for the assembly and cleaning operations, all fuel assembly processing operations are performed dry. Completed fuel assemblies are handled and stored in a manner which provides a minimum 12-inch edge-to-edge spacing.

The Columbia plant required an initial capital investment of \$20,000,000. The initial investment was followed by additional capital investments to produce programmed increases in capacity for various processes, continued product improvement, enhanced manufacturing process development capability, further cost reduction potential, and continued compliance with ever more stringent standards for pollution abatement and other regulatory requirements. For example, additional chemical conversion capacity will go on stream this year, and other processes have been mechanized extensively. Increasingly stringent quality assurance criteria have also added appreciably to the capital investment.

The use and handling of a high cost, politically sensitive, potentially hazardous material like uranium demands elaborate safeguards, production control, and personnel protection. Thus, there is a continuing investment requirement to maintain and improve support services for nuclear materials safeguards systems and personnel; for health physics and criticality systems and personnel; and for physical plant protection systems and personnel. Westinghouse plans to incorporate enrichment control, over and above that already exercised, in the manufacturing cycle at various process points. These plans include installation of monitoring devices for fuel rods and UO₂ powder containers.

Investment in support costs also reach into the areas of tooling, dies, fixtures, and manufacturing process development and improvement. This rate of investment is influenced by the high rate of obsolescence related to the constant need for product design changes which improve or further guarantee reactor performance. All told, investments in the areas enumerated above represent more than \$10,000,000 additional funding since Columbia began operation.

The manufacture of high quality reactor fuel continually demands high support costs in areas such as process engineering, quality assurance, and laboratory services. These are only a few of the areas comprising the overhead burden which must be applied to production cost.

Quality assurance has always been a high cost item in the fuel fabrication business; fully 20 percent of the Westinghouse fuel manufacturing personnel are engaged in inspecting the product. Westinghouse also maintains a separate group to survey the activities and performance of our vendors at their manufacturing locations. Costs related to quality assurance have expanded significantly during the past two years. Westinghouse now provides utility customers with on-site surveillance service at the time of fuel receipt and during all handling operations. There has also been a decided growth in utility customer and third party audits of processes, written procedures, and product. These customer and third party audits naturally demand the time and involvement of Westinghouse personnel on a continuing basis. The United States Atomic Energy Commission has also stepped up its activities, and International Atomic Energy Agency inspections are also required from time to time. Inspections and audits by these outside groups can provide added assurance to the utility regarding the final fuel product, but they add appreciably to the support costs of fuel manufacturing.

As mentioned above, plutonium recycle in thermal reactors will be an economic necessity. Westinghouse has been developing the necessary technology during the past nine years. Because plutonium is toxic, it must be handled in an enclosed space. For the same reason, environmental considerations have a much greater impact on the design of a plutonium facility than on a uranium plant. The design must insure against plutonium escape in the event of earthquakes and extreme atmospheric disturbances such as tornadoes. These factors have a very substantial impact on the cost of the facility. We are now designing a full scale facility for operation beginning in 1977 when it will be needed to meet the increasing requirement for recycle fuel. This facility will be used to manufacture less than 15 percent of the fuel we will deliver in the late 1970's and early 1980's. For that small fraction of our total capacity, we estimate the facility will cost more than twice our investment in the Columbia plant.

Because recycle fuel manufacture will always be a relatively small scale operation, the production and overhead costs are much higher than for uranium. The need for working remotely increases the incentive for mechanization, and this increased mechanization reduces the production cost penalty somewhat. The manufacturer expects to recover an adequate return on investment. The net effect of these factors is substantially higher cost for plutonium fuel.

DEVELOPMENT

The present fuel design is the result of many years evolutionary development, including work in nuclear physics, heat transfer, and core performance under various conditions (including postulated accidents), as well as the more obvious types of effort such as mechanical design of the assembly.

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The rod cluster control (RCC) type fuel assembly, Figure 14, is the standard for all modern day pressurized water reactors. This concept represents neither the beginning nor the end of the technology. The earliest type of assemblies was that used in the Yankee Rowe reactor, located in the United States and owned by Yankee Atomic Electric Company, and in the BR-3 reactor, located in Belgium and owned by Centre d'Etudes de l'Energie Nucleaire. These reactors have rigid assemblies using brazed ferrules as a method of holding the fuel rods. The logical progression was to a more open lattice, although it was considered necessary to contain the assemblies in cans. The Chooz reactor in France, owned by Societe d' Energie Nucleaire Franco-Belge des Ardennes, and the Trino reactor in Italy, owned by Ente Nazionale per l'Energia Elettrica, use this design.

The RCC fuel assembly represented a significant step forward because it eliminated the fuel can, distributed the control rod worth, permitted higher power densities and allowed the reactor vessel to be shortened. Subsequent developments on the RCC have been confined to improvements in the basic design. Examples of this are the addition of mixing vanes, reduction in parasitic absorption material, and modified grid to rod supports. These improvements have been progressively incorporated into the newer fuel designs, such as those used in the Jose' de Cabrera reactor in Spain, owned by Union Electrica S.A.; the Beznau No. 1 and No. 2 reactors in Switzerland, owned by Nordostschweizerisch Kraftwerke AG, and the Mihama No. 1 and No. 2 reactors in Japan, owned by Kansai Electric Power Company.

The above process serves as a good ill'stration of the theme of development, which simply expressed is evolutionary improvement based upon prior design, experiment, and operational experience. Examination of any of the key parameters will show this to be true whether the parameters are hot channel factors, temperature, or power density. The present fuel design represents a very sizeable investment in technology. Investment in development continues at a very high level; Westinghouse assigns some twenty percent of its total fuel income to product development. We expect funding to remain at or near the present level indefinitely.

A rapidly expanding body of actual fuel performance data from operating plants has been factored into the design. This process continues today at an accelerated rate. Using this information, design methods are verified and improved. The efforts to improve the grid to reduce parasitic material still further, to optimize fuel rod support forces, and to achieve optimum flow distributions are continuing. While much of this effort has a basic objective of improving core performance, that objective, except where it is complementary, is secondary to the objective of maintaining a high degree of reliability. By far the majority of our resources are dedicated to extensive analytical and experimental programs directed toward designing and manufacturing fuel with a high level of reliability. These programs can be grouped into four major categories: (1) fuel performance verification, (2) structural and corrosion testing, (3) loop testing, and (4) general reliability activities.

One area of fuel performance verification involves both in-pile and out-of-pile verification of design codes and material specifications, relating principally to the mechanical properties and the corrosion resistance of Zircaloy cladding material. Another area involves analytical validation of design codes and fuel specifications related to fuel properties. A third large area of investigation involves the execution of irradiation tests in operating reactors. These are aimed at evaluating the influence of irradiation on mechanical properties, corrosion, and hydriding, and at determining the combination of power levels and burnup which may result in fuel rod failures.

Fuel assembly components are tested for strength, fatigue, and resistance to corrosion. In addition, prototype assemblies are constructed and subjected to various structural tests to determine the static and dynamic characteristics of the assembly under various operating and handling conditions. Tests include axial and lateral deflection tests, vibration, torsional, and handling and loading tests.

Prototype assemblies are subjected to a series of tests in a loop that simulates actual temperature, pressure, flow, and water chemistry conditions. Control rods are scrammed into the fuel assembly over 200 times as part of a typical 1000 hour test. The behavior of the assembly, i.e., stresses, strains, vibration, etc., is monitored throughout the test.

A number of general reliability activities to improve the designs have also been incorporated. Examples of these activities are (1) formal design reviews, (2) failure mode analysis, (3) analysis of manufacturing data, and (4) analysis of operating data. Another major segment of our resources is directed toward advanced fuel and process development, including plutonium recycle. Our recycle technology development has been progressing for nine years. Some of the programs undertaken are operation of the Saxton reactor with mixed oxide fuel, achieving peak burnup greater than 50,000 MWD/MT; analyses of various critical experiments; technical and economic feasibility studies; construction and operation of a pilot scale plutonium laboratory; and fabrication of mixed oxide fuel assemblies for use in demonstration programs in the San Onofre reactor in the United States, owned by Southern California Edison Company and San Diego Gas and Electric Company, and in the Mihama No. 1 reactor. The primary effort now involves process development and equipment evaluation to support design and construction of a full scale recycle fuel manufacturing facility.

VOLUME EFFECT ON COST

Experience at Cheswick and Columbia has shown that substantial base load is necessary to provide the cost improvement and overhead absorption to result in reasonable fabrication cost. Further increases in capacity and accumulated experience accentuate the trend. Scale economy is very appreciable below 500 metric tons a year, and attractive improvement potential remains available even at the 1000 metric ton a year level.

Investment in inventory also shows a marked decrease due to economy of scale because the raw material, in-process, and finished parts inventories necessary for efficient manufacture become proportionately lower as volume increases. A large volume also provides the flexibility to insure extensive and effective customer services. These customer services can be in response to plant construction delays or speed-ups, or unexpected events such as a fuel assembly damaged in shipping or handling. With its large volume, Westinghouse has and will continue to provide quick response to all customer problems.

Westinghouse has reached the level of scale economy wherein it can take advantage of the cost improvement related to volume, couple this advantage with experience and a continually improving engineered design, and now offer it to various fuel manufacturing partners. Such partnerships have been concluded in Italy (COREN) with Fiat and Breda, and in Japan (Mitsubishi Nuclear Fuel Company, Limited), with Mitsubishi Heavy Industries and Mitsubishi Metal Mining Company. In these arrangements, overhead support services such as the latest design data, latest manufacturing processes, current quality assurance procedures, and engineering assistance are provided to the smaller company at reasonable cost. This is made possible because the heavy expenditures necessary are spread over the large Westinghouse volume. The joint venture can, therefore, be viable at a far lower volume level than would be possible without the Westinghouse base.

Westinghouse also benefits from such arrangements by taking advantage of the expertise already available from programs in the countries involved and by accumulating and assimilating the additional experience available from the joint ventures.

CONCLUSION

Efficient nuclear fuel fabrication requires continual integration of the knowledge progressively derived from design, manufacturing, operation, and development. Accruing the basic knowledge and evolving the present design has taken a great deal of time, effort, and money. Even at the present level of technology, Westinghouse continues to apply twenty percent of its income to further development.

Economy of scale dictates large annual volume combined with high capacity, integrated manufacturing facilities. With large volume, the fuel fabrication is able to spread the heavy expenditures necessary to continue technological improvement. Substantial base volume is an absolute necessity for achieving economic unit fabrication cost. In market spheres which do not provide the necessary volume base, a considerable portion of the benefits accruing from scale economy can be realized by concluding a joint venture with a supplier already having the requisite volume.


FIGURE 1



FIGURE 2



FIGURE 3







FIGURE 5

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FIGURE 7



FIGURE 8





FIGURE 10

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FIGURE 14

THE MANUFACTURE OF FUEL ELEMENTS AND THEIR ASSEMBLY FOR PWR SYSTEMS

by

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ABSTRACT

Manufacturing fuel assemblies for pressurized water reactors is a developing technology. Methods and procedures are being upgraded constantly to improve the quality of, or remove inconsistencies from the product. The underlying philosophy governing these manufacturing operations is one of starting with only high quality raw materials and building quality into the product throughout the manufacturing sequence.

Quality control of fuel manufacturing is a continuous exercise of tests and inspections to assure that processes are under control and that only acceptable material is allowed to remain in the production stream. New and improved inspection methods are under development to function as on-line tests for product quality. The goal is to reach prompt feedback of deviations from the norm so that process adjustments can be made before rejectable material is produced.

The fuel manufacturing business is an enterprise that is more closely tied to financial considerations than many other manufacturing operations. Many of the raw materials used are high in cost. This fact makes inventory control vital and it supports capital investment decisions to increase manufacturing capacity so as to decrease processing time. Detailed financial analyses are necessary to arrive at the correct balance between capacity and throughput requirements. These analyses do indicate that relatively small throughput

requirements are adequate to support a fuels plant with a minimal scope of activities. As throughput requirements increase, integrating additional activities into the facility becomes justifiable.

Organizational structures in manufacturing facilities can be quite variable. In a fuel assembly plant, however, separate manufacturing and quality control organizations reporting to the same manager are mandatory. Manufacturing is responsible for making the quantity of fuel assemblies required at a level of quality that is acceptable. The Quality Control organization is responsible for assuring that only products of acceptable quality are delivered to the customer.

Selecting and training personnel for a fuel manufacturing operation is one of the more important activities in establishing a fuels plant. Management of the operation must be committed to manufacturing only high quality fuel assemblies. In all aspects of training and planning and developing procedures and equipment for fuel manufacture there must be no compromise with this philosophy.

Nuclear fuel manufacture, like most manufacturing operations, can be a fascinating and rewarding business. It must be approached as a business with full recognition of the fact that sacrifices in the level of product quality will lead to serious economic penalties. With this realization and the help of competent people, a developing nation with a minimum of operating reactors can enter the nuclear fuel manufacturing business on a modest scale. This base business can be expanded in manufacturing scope as competency is developed and the expansion is justified by the economics of additional throughput.

INTRODUCTION

In the design of nuclear fuel assemblies, fissionable material must be positioned and contained to provide for efficient removal of heat produced by the fission reaction and to maintain the fuel in a controllable geometry. Improper positioning can lead to failure of the fuel element, curtailme t of reactor oper tion and radioactive contamination of the reactor's primary coolant stream. Therefore, severe economic penalties, as well as potential loss of power generating capability can result from inadequately manufactured fuel assemblies.

While much engineering and scientific effort is expended in the design and testing of fuel assemblies, conservative design practices must also be followed so as to design into the fuel assembly an additional margin of safety. Conservatism is also required throughout the manufacture of fuel elements for this same reason. Manufacturing facilities must be designed and operated to build high quality fuel elements to the precision required by the designer. Compromises in manufacturing equipment and processes cannot be permitted if they will adversely affect reliability and performance of the fuel element.

Besides being technically necessary, quality and precision design and manufacture of fuel elements is the most economical approach to the nuclear fuel business. In the discussion of manufacturing methods, a philosophy of continuous attention to quality will be noted. Starting with an objective of manufacturing a high quality product, and constantly checking the product to assure that the objective will be met, results in a maximum amount of acceptable material. Material found to be rejectable is rejected in-process before additional effort is invested in it. In this way, only acceptable material continues in production. This practice also provides an early opportunity to adjust process equipment to preclude continuing production of off specification material. The combination of precise and timely process control to preclude production of rejectable material and rejection of any faulty material at its lowest value, yields the desired product at minimum cost.

The following discussions cover areas incumbent to the manufacture of nuclear fuel assemblies. Manufacturing methods and processes

are discussed. Quality Control is emphasized. Safeguards and Health Physics requirements are noted. The organization, management and training of staff is covered in some detail. Integration of the operational systems, controls and interfaces with peripheral and supporting organizations are presented and discussed as are the economics of fue' manufacturing. The economic discussion will also touch on the scope of the manufacturing operation and relate this to business volume.

MANUFACTURING METHODS AND PROCESSES

Fuel elements for modern pressurized and boiling water reactors consist of uranium dioxide fuel, zirconium alloy fuel cladding, spacing grids made from zirconium alloy, stainless steel or nickel alloy, stainless steel end fittings and nickel alloy hold-down springs. Manufacturing these components and integrating them into a fuel assembly requires various methods and processes. Most of these methods are conventional except that added attention is given to processing details to assure reliability and reproduceability.

Manufacture of oxide fuel pellets and zirconium alloy tubing are discussed at other sessions in these proceedings. These products will be treated as raw material for purposes of this presentation. Other raw materials include zirconium alloy, stainless steel and nickel alloy mill products and castings.

Manufacturing small hardware components, such as end caps for fuel rods and fastening devices for the fuel assembly, can be accomplished on conventional machine tools. Automatic or semi-automatic machines are preferred because of consistency of product and output. Hand-operated machinery is acceptable providing the operator's skill is adequate to produce consistently high quality parts. At The Babcock and Wilcox plant near Lynchburg, Virginia, automatic machine tools are used for most high production turning operations (Figure 1). However, an operator mans the machine to keep turnings from fouling the parts and to check periodically the dimensions and quality of the part being manufactured. Constant attention to output quality affords early detection and correction of trends that could lead to the production of unacceptable parts. Material received from outside sources, as well as small hardware components manufactured in-house, are quarantined until they have passed an incoming inspection (Figure 2). This over-inspection ranges from a statistical sample to a 100% examination of the parts for adherence to drawings and specifications. Once accepted, the parts are released to a storekeeper and held in inventory until needed for production. Manufacturing personnel requisition material from the storekeeper and the manufacturing organization is then held accountable for the quantity and quality of finished products produced.

At B&W's fuel plant, the cladding tubes are prepared by machining a welding configuration on each end at the same time the tube is cut to final length. The tubes are then cleaned and one end cap is welded in Place. Welding is accomplished in a small chamber slightly pressurized with inert gas (Figure 3). The end of the rod to be welded is inserted into the chamber through a sphincter-type sealing hole. The rod is held stationary while the tungsten inert gas welding head rotates about the weld joint. A programmed direct current pulsed arc welding current, with the tungsten electrode making several revolutions around the weld joint, completes the weld. The welding sequence is automatically controlled using a high frequency arc initiation, followed by a timed ramp increase in amperage to a constant pulsed welding current and finally a timed ramp decrease in amperage to terminate the arc. The operator positions the rod in the chamber, initiates welding by pushing a "start" button and removes the rod at the end of the welding cycle. Welding current and voltage are recorded on strip charts as an aid to process control. In this way, the operator and his supervisor can audit the welding operation and ussure that consistent welding parameters are being used. The operator also examines welds for appearance and final dimensions.

A unique number is next stamped on the end cap and is used as identification for that fuel rod and all subsequently incorporated materials. The pedigree of the individual fuel rod is retrievable through its identifying number.

Following identification, the end cap weld is inspected for dimensions and soundness by Quality Control personnel. An ultrasonic inspection is used for the soundness examination (Figure 4).

The ultrasonic inspection consists of rotating the rod under an ultrasonic search unit that is programmed to traverse the length of the weld zone. The ultrasonic transducer produces a pulsed and focused beam of sound entering the weld zone in a shear mode. Discontinuities reflect the sound back to the transducer causing a signal to be recorded on a strip chart as well as initiating an audible and visible alarm. The sound beam in the zone of inspection is 0.3 to 0.5 mm in diameter and it is translated axially 0.2 mm per revolution. Discontinuities on the order of 0.05 mm are detectable.

Cladding tubes with one end cap in place are inserted through a wall in the fuel loading room and into a loading machine for insertion of the fuel pellets (Figure 5). Only the open end of the clad tube is exposed to possible contamination by the fuel. The fuel pellets needed for one rod are weighed and their weight is recorded on the pedigree document for the fuel rod. The stack of fuel pellets is laid out to its full length and visually examined for adherence to pellet quality standards as well as length tolerance. Internal hardware, consisting of erd springs and insulators, is added to the stack and the entire column is inserted at one time into the waiting fuel rod. The rod is then removed from the loading machine, manually probed to check end clearance and the second end cap inserted. Second end welding is accomplished and inspected in the same manner as the first end weld.

The sealed fuel rod is then punctured by a laser drilling operation for vacuum outgassing and subsequent pressurizing with helium (Figure 6). The laser drilling operation is accomplished by inserting the rod into a small chamber, evacuating the chamber and discharging a small diameter laser beam at the end cap. The concentrated energy of the laser beam melts a small hole through the end cap. The atmospheric pressure in the fuel rod expels the molten Zircaloy into the evacuated chamber and the resulting pressure rise in the chamber is used to verify successful completion of the operation. Chamber pressure is recorded on a strip chart for auditing by the operator and his supervisor to assure that drilling has been accomplished.

Fuel rods with laser drilled end caps are accumulated and charged into a vacuum furnace in lots of approximately 200 rods (Figure 7). Sample rods, containing fuel pellets processed along with the pellets in the

fuel rods, are included in the vacuum furnace with end cap and laser drilled specimens intended for subsequent destructive examinations. The charged furnace is evacuated, back filled with inert gas and heated to expel excess moisture from the fuel rods. Repeated evacuation and back filling cycles are used to accelerate moisture removal. Moisture in the furnace atmosphere is monitored to control drying time.

Upon cooling, the rods are removed and inserted into another small chamber for pressurizing and sealing. Once in the chamber, the small laser drilled hole is aligned with a second laser used for fusing the hole. The rod is clamped and sealed in the chamber, chamber pressure raised with helium to the level desired and the laser discharged to fuse the hole in the end cap.

After pressurizing, the fuel rods are cleaned and subjected to alpha counting to assure there is no fissile material contamination on their outside surfaces (Figure 8). Each rod is given a visual and dimensional inspection at this point. The rods are then subjected to a total helium leak test to assure integrity of cladding, end welds and laser weld.

Grids for B&W fuel assemblies are fabricated from an age hardenable nickel alloy. Strips of alloy are punch pressed to shape the contact points and then assembled into the grid array. After assembly, the grid is welded together at the intersection of the strips. The grids are then placed in a heat treating fixture and age hardened. Grid inspection consists of measurements of cell size and outside dimensions as well as detailed visual examinations for defective welds and other imperfections in the materials. Each grid is numbered and materials pedigree and inspection data are recorded for each grid.

A set of six intermediate and two end grids is selected and the cells opened by a unique keying device. The keys hold the individual cells to an open dimension larger than the outside diameter of the fuel rods. Grid numbers are checked and listed in the record for the fuel assembly being fabricated. The keyed grids are then assembled into a horizontal assembly fixture that holds them in position

during assembly fabrication. The assembly fixture is mounted on a granite surface plate (Figure 9). When the manufacturing personnel have satisfied themselves that the grids are in proper position, Quality Control personnel inspect them for alignment using the granite surface plate as a plane of reference.

The assembly personnel insert fuel rods into the grids after checking the rod number and assuring themselves that each rod number is contained in the record for the fuel assembly being fabricated. Each fuel rod is hand-wiped with acetone as it is inserted into the grids. As the assembly is built up, a central instrument tube is added. Spacer sleeves are affixed to the instrument tube to act as positive separators between grids. After all of the fuel rods (208 in the B&W fuel assemblies) have been inserted, the grid keys are turned and removed to release the cells to grip the fuel rods.

The sixteen control rod guide tubes are inserted into the grids and the end fitting castings affixed to both ends. The guide tubes are mechanically fastened to the end fittings and the fasteners welded in place to prevent their becoming loose during subsequent handling and service. The end fittings are also attached to the end grids by welding small insert fasteners in place. A final check of the fuel assembly record is made to assure inclusion of all part numbers.

After being released by assembly operations personnel, the fuel assembly is placed on a transfer cart equipped with a hydraulic device for translating the assembly to the vertical position. Once vertical, the assembly fixture is removed. Subsequent handling is accomplished by lifting the assembly from the upper end fitting as it will be handled for loading into the reactor.

The fuel assembly envelope dimensions are checked in a vertical envelope gauge (Figure 10). Sensors travel over the full length of the assembly in the free standing and restrained positions to check for straightness twist, bow and envelope. Strain gauges are used to measure water channels at eight locations over the length of the assembly. Signals from the strain gauges are converted by computer to digital values for water channels. These data are then compared to standard values and, if acceptable, the assembly is released for further processing.

The hold-down spring and spider are added to the upper end fitting and inserts limiting spider motion are welded in place. The assembly is then transferred to a pit where a control rod with upper tolerance limits is inserted in the fuel assembly guide tubes. This insertion test is to measure guide tube alignment to assure control rod scram time is within tolerance.

Final visual inspection is performed by Quality Control personnel using high intensity lights and magnifying devices where necessary. Cleaning is accomplished by manufacturing personnel using acetone and compressed air. When satisfactorily cleaned, the assembly is sealed in a plastic sheath and either stored in the vertical position or placed in a shipping container for shipment to the reactor site.

Manufacturing of other components, such as burnable poison assemblies and control rod assemblies, is accomplished with the same attention to detail as described for fuel assembly manufacture. Minimal operator dependency is a general practice. Where dependence on the operator cannot be avoided, he is thoroughly trained and competency is established before he is permitted to work in manufacturing operations. An overcheck is scheduled in the manufacturing process after such an operation to monitor the operation and assure acceptability of the product.

Each operation in the manufacturing sequence is described by a separate written process procedure. Copies of these procedures are reviewed by the equipment operators and retained in a packet close to the operation for ear ' reference. Details of each operation to be performed are listed along with explanatory statements where necessary to improve the operator's understanding of his job.

A route card listing the operations to be performed in sequence is affixed to each group of components. These cards contain spaces for operators to initial and date the card signifying that the operations listed have been completed.

Internal audits an performed periodically to assure that the process procedures are being followed and the operations are being performed in their proper sequence. In this way, operators have an opportunity to suggest modifications to procedures that they feel would improve

the operations. Such suggestions are often found to be acceptable and, in such cases, the process procedure will be modified to incorporate the change. Until the procedure has been changed and approved, however, the previous issue remains in effect.

The manufacturing operations are fully described by formalized procedures and con olled by auditing the individual operations against these procedures. Individual operators are not forced to rely on their memory for details of their operations or for routing of materials through the production sequence. The operators are held responsible for the quality as well as the quantity of their work; however, over-checks are performed by Quality Control personnel to assure continued manufacture of high quality products.

QUALITY CONTROL

Control of the quality of fuel assemblies incorporates assurance of quality of raw materials, control of manufacturing and inspection processes to preclude production of deviated material, and finished product inspection to verify final product quality and adherence to drawings and specifications. It is standard practice to not only control the quality of products but to record and document inspection and test results. Thus a pedigree is available in the event problems are encountered during the operating life of the fuel assembly. Examination of a failed component's pedigree may provide information on the cause of .ailure. Design or manufacturing modifications can then be made to correct deficiencies that can lead to operating problems.

Raw material and component vendors are required to have formal quality control organizations and procedures. Frequent audits are performed to assure that the vendor is supplying quality material and that he is controlling his process so that unacceptable material is removed from the production stream.

Inspections of vendor supplied material may be carried out at the vendor's plant or on receipt of his product. Where special inspection equipment is required, the fuel manufacturer may choose to witness inspection of the product at the vendor's plant and also over-inspect the product on receipt, using conventional procedures. In any event,

the quality of purchased materials is verified by one or more inspections by the fuel manufacturer's quality control organization. The ultimate user of the fuel, the reactor operator, may choose to accompany the fuel manufacturer during his source inspections.

The degree of over-inspection is dictated by the importance of the purchas d component and by its alue. For example, zircaloy tubing and bar stock to be used for fuel cladding and end caps are subjected to extensive non-destructive inspections by the manufacturer. These include ultrasonic inspections covering 100% of the volume of the product and penetrant inspections over those areas where the manufacturing process can cause defects. Over-inspection might include the same type of ultrasonic and penetrant inspections using comparable equipment to that of the manufacturer but on a statistical sample rather than on 100% of the product.

Destructive testing of these particular raw materials is also common practice. Corrosion tests, mechanical property tests and check chemical analyses are performed on samples representing the material as it is received from the vendor.

In-house, the Quality Control organization assumes the role of auditor and over-inspector. The prime function of Quality Control is inspection of products and auditing of procedures. If faulty material is found, it is rejected along with the lot from which it came. It is not a quality control function to segregate acceptable from unacceptable materia", except in cases where 100% non-destructive inspections are practiced. In these cases, a normal reject rate is established and variations from the norm are used to effect process modifications as a form of process control.

There are continuing efforts being made to develop more exacting quality control and inspection procedures. Mainly, these efforts are directed towards development of on-line non-destructive testing procedures for continuous monitoring of the quality of the product. Results from these procedures can be fed back for continuous process control so that faulty products are not produced. The ultimate goal of Quality Control is just that -- preventing the production of rejectable products. Where 100% inspection is impractical or where destructive tests are required, samples are taken and tested as early as possible. The product may be quarantined until the test results indicate that the lot represented by the sample is acceptable, or a limited quantity of continued work on the product may be authorized depending on the risk involved. The principal objective, however, is to minimize the investment of additional effort in a lot of product until its acceptability has been verified.

As in the manufacturing operations, written procedures are prepared for all quality control operations. These procedures describe in detail the nature of the inspection or test to be performed. They provide such information as the types of instruments to be used, whether actual values of inspection results are to be recorded, the number of samples to be inspected and the disposition of rejectable material.

Another vital aspect of the quality control function is the assimilation and recording of data for the pedigree of the fuel assembly. At the conclusion of manufacture and final inspection of a fuel assembly, a conformance certification is prepared that lists the assembly number and all drawings and specifications applicable to the assembly. This document also lists the quantity of UO_2 and Uranium-235 contained in the fuel assembly.

The fuel assembly number is the key to tracing the pedigree of materials and components that were used in its manufacture. All of the inspection and manufacturing history for the assembly is retained in quality control files indexed to the fuel assembly number. The data and information available for each fuel assembly include:

- Final inspection results, including about 3400 water channel dimensions.
- Identification numbers for all components providing traceability to original material certifications.
- Individual fuel stack weights and Uranium-235 content.
- Route cards covering the manufacturing and inspection operations for all components in the assembly.

- Radiographs of end fitting components.
- Variations or deviations from standard operations and their disposition.

At the conclusion of manufacture of a core, or portion thereof that will be discharged as a unit from the reactor, all inspection records are compiled into a single file and retained until the core or batch has been reprocessed after service.

The principal quality control function is one of over-check and feedback. The quality control organization is responsible for checking the product to assure that quality standards are met and to feed back information on minor deviations that may indicate trends so that processing adjustments can be made before rejectable products are manufactured. Most of the materials making up fuel assemblies are costly, so an efficient quality control operation is an economic necessity. It signals the need for process corrections before the process reaches the point where scrap products will be produced and minimizes the expenditure of added effort on rejectable material.

To maintain this desired level of quality, approximately 35% of the total direct labor expended in the production of fuel assemblies is for quality control. Compared to other manufacturing operations, this level of expenditure for quality control may be considered high. However, such an effort is justified in view of the serious and costly operat.onal difficulties that _an be caused if faulty fuel assemblies are placed in service. A more quantifiable justification can be reached on the basis of manufacturing performance and low reject costs directly attributable to an effective quality control program.

ECONOMICS OF PLANT SIZE

There are many fixed cost functions associated with nuclear fuel manufacture. They include fissionable material accountability and safeguards, health physics and personnel monitoring, facility licensing and attention to regulatory requirements. In a free enterprise economy, the volume of business must be large enough so that the fixed costs are not a major factor affecting unit product costs. While not precise, the minimum size plant considered justifiable in the United States is one capable of assembling one fuel assembly per 8-hour shift. Such a plant will be primarily an assembly plant, purchasing most components of the fuel assembly in a form suitable for direct incorporation in the final fuel assembly.

The total cost of a one assembly per 8-hour shift plant is approximately 3.0 millin dollars. This covers a building of approximately 4000 m^2 area and equipment including:

- Cladding preparation equipment
- Degreasers, ultrasonic cleaners and associated cleaning equipment
- End closure welders
- Fuel rod pressurization equipment
- Fuel loaders
- Handling equipment
- Ultrasonic, helium leak, alpha counting and other nondestructive inspection equipment
- Fuel drying equipment
- Assembly fixtures
- Envelope and water channel inspection equipment
- Chemical and metallurgical laboratory equipment
- Finished fuel storage racks
- Fuel shipping containers
- Machine Shop
- Environmental monitoring equipment
- Fuel accountability control equipment
- Inert gas systems
- Compressed air system
- Electronic maintenance equipment
- Mechanical and electrical maintenance shop
- Demineralized water system and corrosion test autoclaves

In-house manufacture of some of the components of the fuel assembly is justified on the basis of volume. For example, a typical fuel assembly requires about 420 end caps, 20 guide tube upper and lower end fittings and fasteners, 32 end fitting to grid fastening inserts, 420 end void support devices, about 2000 insulator pellets and roughly 40,000 fuel pellets. The end caps and other small hardware components can be manufactured on relatively small and inexpensive automatic machine tools. In a one assembly per 8-hour shift plant, there is a sufficient quantity required to keep one or more machines occupied near full time in small hardware component production.

In-house manufacture of other components, even at low volume, can be justified on quality control and economic bases. For example, in-house manufacture of insulator pellets is feasible because the manufacturing equipment requirements are minimal and the costs of purchase of the small quantity of pellets needed to nuclear quality specifications are excessive. Equipment utilization in this case is low, yet justified because in-house manufacturing costs can be as much as a factor of 5 below outside procurement costs.

In-house manufacture of fuel pellets for a one fuel assembly per day production rate can be justified when quality considerations are included in the analysis. Purchase of fuel pellets entails a significant amount of over-inspection and testing on receipt to assure quality as well as accountability. Shipping damage to the pellets is also an expense factor. The delays encountered in vendor certification, shipment and receiver re-certification of the pellets represent a significant inventory carrying cost that can be eliminated by inhouse manufacture of pellets. These costs, in addition to the less quantifiable factor of enhanced control of the quality of the pellets, support a conclusion that in-house pellet manufacture is justifiable.

In suc a case, sintering and _rinding capacity would be required to meet a production rate of 40,000 pellets per day. On an 8-hour day basis, 5000 pellets per hour are needed. This is a reasonable production rate for a low-volume press. Associated equipment can be sized to match or slightly exceed this production rate without excessive costs. Thus a small pellet production line is feasible even for a "one assembly per day" fuel plant on the basis of one shift utilization of production equipment.

There are technology considerations involved in determining the scope of in-house manufacture. The technology required for small hardware manufacture is not sophisticated. However, manufacture of insulator and fuel pellets does require some ceramic or high temperature chemistry technology that may not be readily available. Access to such technology on a continuing basis is important also to modify and

improve processing and to understand and compensate for minor variations in the characteristics of incoming raw materials. For the initial start-up of a "one fuel assembly per day" plant, it is prudent to defer in-house manufacture of components requiring more sophisticated technology until the basic assembly plant is performing satisfactorily.

There are other omponents of the fuel a sembly that should not be considered for in-house manufacture until business volume is sufficient to justify a minimum sized production facility. For example, manufacture of fuel cladding requires a significant investment in machinery and equipment as do conversion facilities for the manufacture of UO_2 powder. These businesses are discussed elsewhere in these proceedings. As business volume grows, however, there will be a point at which powder conversion facilities, and at least finish tubing manufacturing equipment, can be justified. Integration into these businesses should be considered as incremental increases in production output are required.

Figure (11) shows a conceptual layout of a basic fuel assembly plant that has been expanded to include component manufacturing and finally UO_2 powder conversion capability.

There are economic considerations that may favor increasing the production capability of a fuel plant so as to reduce the processing time for high cost components of the fuel assemblies. Operating the basic "one fuel assembly per day" plant on a 3-shift, 7-day schedule will increase through ut by a factor of 4. A ditional equipment can also be added to increase throughput per shift. Increasing the "one assembly per day" plant to two assemblies per day will cost approximately 300,000 dollars for additional equipment. No increase in the 4000 m^2 plant should be required for this doubling of capacity.

A minimum sized pellet manufacturing facility will cost approximately 300,000 dollars when added to an existing fuel assembly plant, providing appropriate services have been included in the original assembly plant. The throughput of this pellet plant will be about one tonne of UO_2 per day with part of the equipment being operated on a 3-shift schedule.

Manufacturing grids and machining end fittings to support a two assembly per day throughput will require about 1,000,000 dollars for building,

machinery and tooling. This is about the minimum throughput facility that is considered economically attractive.

From the foregoing discussion, it can be seen that fuel assembly manufacturing can be initiated modestly and the scope of business expanded as warranted by throughput requirements. The phases of the business that require sophisticated technology and are highly capital intensive, such as UF_6 conversion to UO_2 powder, require detailed study and evaluation before a decision is made for incorporation into the business. It is advisable to forego entrance into the complicated phases of the fuel business until a fuel assembly plant has been well established and is operating efficiently.

Other considerations must include the variety of fuel assemblies to be manufactured. If a single assembly design is all that is required, equipment and tooling requirements will be minimal. However, if fuel assemblies of varying designs are required, the manufacturing facilities must be expanded and additional capital expenditures are required. It is prudent to limit the facilities scope to a single fuel design during the initial stages of operation. Expanding into multiple fuel designs can be considered and fabricating scope increased as manufacturing experience is gained and economic justification dictates.

APPROXIMATE U.S. CAPITAL INVESTMENT REQUIREMENTS FOR FUEL ASSEMBLY MANUFACTURE

		DOLLARS
Basic "one assembly per day"		
assembly plant:	Building Equipment	1,400,000 1,600,000
Added capacity to "two assemblies		
per day":	Equipment	300,000

Added scope - Grids and end fittings for two assemblies		
per day:	Building Equipment	100,000 900,000
Added scope - Fuel Pellets, 1 tonne per day (two assemblies		
per day):	Building Equipment	50,000 250,000

ORGANIZATION FOR FUEL MANUFACTURE

The principal philosophy to be followed in establishing an organization for the manufacture of fuel is that the product must be of exceptional quality. The personnel in the organization, therefore, must accept this philosophy and implement it at all times. Paramount to the implementation of such a philosophy is strong leadership dedicated to producing the best fuel assemblies possible. Besides dedication, however, the leadership must also be provided with the incentive to produce a high quality product and the authority to achieve this goal.

Preferably, fuel element design is included in the scope of the organization. I such a case, the desig and manufacturing groups should be maintained at the same level in the organization and made responsible to a general manager. The design group retains responsibility for review and approval of any deviation caused by manufacturing variables that the manufacturing group considers to be acceptable. The design group is also responsible for design modifications that might improve performance or quality of the fuel element or that might reduce the cost of the fuel element. Toward this end, suggestions from the manufacturing organization should be encouraged to enhance the "value engineering" aspect of design improvements.

The manager of the manufacturing group is responsible for the quantity as well as the quality of fuel assemblies produced. The quality control organization should be responsible for assuring the quality of incoming material as well as that produced in-house. In the former case,

vendor auditing, source inspection and receipt inspection are involved. In the latter case, in-process and component inspections should be utilized to assure process control and to assure conformance to specifications as the product is being manufactured. To meet these responsibilities, manufacturing and quality control groups must be separate and of equal authority.

The quality control organization performs a verification function over the products produced by the manufacturing organization. They should have no responsibility for repairing non-conforming materials or components or for separating acceptable from unacceptable material. In this way, the quality control organization is relieved of any responsibility to provide acceptable material. Their responsibility is simply to assure that only acceptable material is introduced into the manufacturing operation and that only finished products that meet the drawings and specifications are shipped to the customer.

This subtle definition of responsibility can account for the difference between an acceptable and unacceptable quality control program. When the quality control group is given responsibility for modification, selection or repair of material under inspection, there is some incentive to meet quantity goals even though such goals are not formalized. With no responsibility for modification or repair of the material being inspected and only responsibility for assuring that unacceptable material is not introduced to the production stream, the only incentive that the group has is to show its effectiveness in controlling quality. And this is precisely what they are responsible for.

The manufacturing group, on the other hand, has to be provided with incentive to manufacture a high quality product as well as meet its quantity goals. This can be accomplished by placing the responsibility for the quality as well as quantity of product on the manufacturing group. The fact that the product will be over-inspected by the quality control group should not be used as an excuse to relieve the manufacturing group of its responsibility for building quality into the product. If this philosophy is maintained, the manufacturing group will have an incentive to continually improve the quality of the product. If this philosophy is relaxed, the manufacturing group will have a tendency to place higher emphasis on quantity of production at the expense of quality.

The other principal line function, materials management, is also critical to the successful operation of a nuclear fuel manufacturing facility. Acceptable materials in the quantities required to sustain production must be available when needed to avoid expensive delays. Most of the materials used in fuel assemblies are expensive by virtue of their precision and quality requirements. While production can be delayed by the absence of a critical component, inflated inventories to preclude an "out of stock" situation is equally unacceptable because of the high cost of excess inventory. An efficient materials management organization is therefore a vital part of a fuels plant organization.

Because of the high value of nuclear materials, and international agreements to restrict proliferation of nuclear weaponry, nuclear fuel processors must have an effective safeguards organization. Briefly, safeguards is defined as the prevention or prompt detection of diversion of nuclear materials. An effective safeguards program must therefore ensure the security of operations and data from sabotage and theft, as well as ensuring that nuclear materials themselves do not fall into unauthorized hands.

In setting up a safeguards organization, personnel trained in statistics, nuclear physics, non-destructive assay, electronic data processing and accounting procedures need to be available. Knowledge in these fields is required to establish valid sampling systems and routine accounting and inventory procedures to assure accurate and timely control of the nuclear materials assigned to the facility.

Another part of the organization that must be established before the plant design is finalized is the health physics operation. Criteria for the maintenance of a nuclear and radiologically safe work place have been established by international organizations. Generally, regulatory controls imposed by individual countries conform to the restrictions suggested by such agencies. Interpretation and application of the guides requires both knowledge and applied experience in such disciplines as nuclear criticality safety, accident analysis, and health physics. Early in the planning stage for a fuel plant, consultation is required with nuclear engineers, physicists, biologists, and chemists for the primary purpose of assuring a plant design that is capable of safe operation. Involvement of these disciplines from

the earliest design phases through plant operation ensures that proper safety controls are initially established and designed into the plant and its associated equipment. This approach is advantageous for a number of reasons including the elimination of unnecessary conservatism in equipment design, the adaptability of plant safety systems to future expansion, and limitation of the necessity for "backfitting". Froper initial design, applying the recommendations of safety personnel, has been shown to reduce long range plant operating expense and to increase productivity by eliminating or reducing the use of safety related controls, such as respirators, which detract from employee efficiency.

SELECTION AND TRAINING OF PERSONNEL

Personnel experienced in the manufacture of nuclear fuel are not generally available. It is usually necessary to select individuals with appropriate education and experience for the more responsible positions and provide them with the training necessary for nuclear fuel manufacture.

Following is a list of key personnel required for a fuel assembly plant. The background requirements noted are not necessarily all-inclusive and it is advisable to consider additional training for car lidates for particular sitions on an ind vidual basis.

1. Position: Plant Manager

Background Recommended:	Background	Management of high quality light equip-
	ment manufacturing plant. Realization of	
		the nature and expense of nuclear fuel.
		Familiar with nuclear technology, con-
	sequences of fuel failures and effects of	
		the reactor environment on nuclear fuels
		and structural materials.

2. Position: Manager, Manufacturing

Background
Recommended:Manager of high quality metals fabrication
or light equipment manufacturing plant.

Familiar with nuclear technology and significance of quality in fuel components. Experience in high quality, precision welding of reactive metals.

3. Position: Manager, Quality Control

- Background Recommended: Quality Control management in high quality light equipment manufacturing plant. Familiar with non-destructive testing methods and equipment and with statistical quality control procedures.
- 4. Position: Manager, Manufacturing Engineering

Background Recommended: Graduate engineer with experience in process development and welding of reactive metals and stainless steels.

5. Position: Manager, Production Control

Background
Recommended:Metals fabrication, light equipment
manufacture, or chemical process
industry production control.

6. Position: Manager, Nuclear Materials Control

BackgroundEngineering degree with extensiveRecommended:mathematical statistics training.

7. Position: Radiation Safety Officer

Background Degree in Physical Sciences or Biology with training in radiation measurement and effects.

8. Position: Nuclear Safety Officer

Background
Recommended:Degree in Nuclear Physics with experience
in nuclear calculations.

9. Position: Manufacturing Foreman

Background
Recommended:Supervision in light equipment manu-
facturing plant. Quality oriented
and familiar with high quality manufacturing, welding and assembly methods.

10. Position: Inspection Foreman

Background Recommended: Supervision of inspection of high quality, precision equipment. Familiar with statistical quality control methods and with non-destructive testing methods.

If it is at all feasible, the most efficient way of further training personnel to fill these positions is to place them in an operating fuel assembly plant for varying periods of time. In this way, the individuals have an opportunity to participate in the routine of a fuels plant and to gain an understanding of the underlying philosophy of a high quality manufacturing organization.

SUMMARY

Manufacturing fuel assemblies for pressurized water reactors is a developing technology. Many of the methods and procedures used in fuel manufacturing operations were developed specifically for the production of this critical component of the nuclear reactor. Methodr and procedures are being upgraded constartly to improve the quality of, or remove inconsistencies from the product. The underlying philosophy governing these manufacturing operations is one of starting with only high quality raw materials and building quality into the product throughout the manufacturing sequence.

Quality control of fuel manufacturing is a continuous exercise of tests and inspections to assure that processes are under control and that only acceptable material is allowed to remain in the production stream. New and improved inspection methods are under development to function as on-line tests for product quality. The goal is to reach prompt feedback of deviations from the norm so that process adjustments can be made before rejectable material is produced. In the fuel manufacturing business, process control is essentially synonomous with quality control.

The fuel manufacturing business is an enterprise that is more closely tied to financial considerations than many other manufacturing operations. Many of the raw materials used are high in cost. This fact makes inventory control vital and it supports capital investment decisions to increase manufacturing capacity so as to decrease processing time. Detailed financial analyses are necessary to arrive at the correct balance between capacity and throughput requirements. These analyses do indicate that relatively small throughput requirements are adequate to support a fuels plant with a minimal scope of activities. As throughput requirements increase, integrating additional activities into the facility becomes justifiable. These points indicate that a gradual entrance into the fuel manufacturing business is not only desirable but financially attractive.

Organizational structures in manufacturing facilities can be quite variable. In a fuel assembly plant, however, separate manufacturing and quality control organizations reporting to the same manager are mandatory. Manufacturing is responsible for making the quantity of fuel assemblies required at a level of quality that is acceptable. The Quality Control organization is responsible for assuring that only products of acceptable quality are delivered to the customer. Any component that does not completely comply to drawings and specifications must be rejected unless the deviation is slight and has little effect on the performance of the component. The decision as to whether or not a deviation is acceptable must only be made by the organization responsible for the component's design. To facilitate analyses and evaluations of deviations, the manufacturing organization should have access to the responsible design organization.

Selecting and training personnel for a fuel manufacturing operation is one of the more important activities in establishing a fuels plant. Management of the operation must be committed to manufacturing only high quality fuel assemblies. In all aspects of planning and developing procedures and equipment for fuel manufacture there must be no compromise with this philosophy. Preferably personnel with nuclear backgrounds should be selected to staff a new plant. Individuals with experience in areas where the quality of the product was highly important can be trained to fill key positions, but they will require background training to establish an understanding of nuclear manufacturing technology. Nuclear fuel manufacture, like most manufacturing operations, can be a fascinating and rewarding business. It must be approached as a business with full recognition of the fact that sacrifices in the level of product quality will lead to serious economic penalties. With this realization and the help of competent people, a developing nation with a minimum of operating reactors can enter the nuclear fuel manufacturing business on a modest scale starting with assembly operations. This base business can be expanded in manufacturing scope as competency is developed and the expansion is justified by the economics of additional throughput.







FIGURE 4 ULTRASONIC TESTING OF END CLOSURE WELDS





FUEL LOADING STATION





FIGURE 7 VACUUM DRYING FURNACES







FIGURE 10 FINAL INSPECTION AREA





LOW BAY 14'(4M) HIGH HIGH BAY 25'(8M) HIGH

Boiling Water Nuclear Fuel Manufacture

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ABSTRACT

General Electric has a large and growing integrated manufacturing capability in the United States with licensees and jointly owned fuel manufacturing facilities in Europe and Japan.

Substantial research and development support is provided for all aspects of fuel design operation, manufacturing process, and quality methods.

The manufacturing efforts are backed by large scale development and testing at Vallecitos, California, design analysis at San Jose, California, and Process development in both San Jose, California and at Wilmington, North Carolina. Basic methods and material research is carried out at Vallecitos, California and the General Electric Research Center in Schenectady, New York. Specialized development process and pilot operations are conducted both at San Jose and Vallecitos.

Supporting technical evaluation of design, materials, methods, and product is conducted in various reactors including GETR at Vallecitos, the Halden reactor in Norway, and extensive field surveillance in operating reactors.

Reactivity control and power leveling in the fuel is accomplished by blending gadolinia in various concentrations with urania.

Segregation of multiple enrichments of U^{235} and the special combinations of gadolinia with urania in addition to the overall material, quality, and process control activities are maintained by a computer aided accounting and process system (MICS). The product produced is thoroughly tested by the latest quality information methods and equipment, including the scanning of completed fuel rods. The latter method can detect a single off-standard enrichment of pellet in a fuel rod.

Integration of the manufacturing process as described in the text extends from the reduction of tube shells of zirconium alloy and the conversion of uranium hexafluoride to the finished fuel. Limited capability for remelting of zirconium is available in the General Electric facilities.

The combination of vertical integration combined with multiple manufacturing facilities all backed by extensive technical support provides a broad base the rapid progression of fuel manufacturing technology and capability.

INTRODUCTION

The General Electric Company manufactures reactor fuel in large-scale integrated facilities in Wilmington, North Carolina, and under license arrangements with jointly owned facilities in Germany, Japan, and Italy. This manufacturing effort is supported by large-scale development and testing efforts at Vallecitos, California; design analysis and process development in San Jose, California, and in Wilmington; and basic materials and methods research at Vallecitos and in the GE Research Center in Schenectady, New York. Specialized developmental process facilities for UO₂ fuel and for plutonia-urania fuel are in operation in San Jose and Vallecitos.

Experience with the manufacture and use of zirconium-clad fuel in commercial boiling water power reactors now extends over 14 years. Continuing refinements in process, inspection methods, and manufacturing systems are required of the manufacturer. These changes are due in part to continually rising expectations and requirements made of the fuel. Typically these are: high production rates - well over 1000 tons/year; increases in fuel ratings; increases in burnup; and increases in statistical reliability assurance levels - as well as in actual fuel performance levels. Furthermore, the manufacturing plant is required to meet ever higher standards for health and safety, materials accountability and safeguards, waste and scrap inventory control, and nuclear safety. This combination of increasing expectations requires fundamental changes in the organization of manufacturing work relative to earlier methods which were adequate in small-scale production. Even in large-scale production operations, continuing flexibility is required to introduce changes as required by improved

design, improved processes, and improved inspection methods and statistical controls. A continuing iteration must occur between the production lines and fuel performance results, fuel development and testing results, and process and method development. To highlight the nature of the continuing feedback and dynamic adaptation required, this pr sentation will cover the general scope of BWR fuel manufacture -- the method and organization of the main process and production and quality control and support activities, and the organization of international and licensing activities. Following this will be discussion of the technical and developmental aspects which are a continuing influence on the manufacturing operation.

BWR Fuel Manufacturing - General Scope

Rapidly increasing demand for both initial and reload fuel cores for General Electric BWR's has necessitated substantial growth of manufacturing facilities. These have progressed from a modified laboratory type shop in the late fifties, to a job shop in the early sixties. In the mid-sixties, load increase made obvious the need for a substantially larger manufacturing facility with capacity, processes and systems to produce several projects simultaneously. Further expansion to capacity in excess of 1500 tons per year is in progress.

Genera' Electric has over 14 years of commercial zirconium clad fuel manufacturing experience and over 70 reactor years of cumulative operating experience using zirconium clad fuel in 13 BWR's now in full commercial operation.

Ever greater emphasis is being placed on quality control throughout the manufacturing cycle. The cost of the materials is high and the accountability demands severe. In addition it is necessary to operate safely with appropriate attention to health physics, criticality and with regard for increasingly restrictive environmental considerations.

A manufacturing facility today must provide substantial capacity and flexibility to accommodate multiple projects simultaneously, and to modify practice to apply design and process improvements. The latest and evolving techniques for advanced processes, inspection,

mechanization and automation are required to meet the demands for quality and productivity. Advanced material and production control techniques using data processing equipment are a basic necessity.

The trend of fuel manufacturing is moving rapidly toward high capital costs courled with highly skilled and experienced human resources.

<u>GE Fuel Facilities</u>

The Wilmington, North Carolina plant is well integrated starting with UF₆ for the fuel, extruded tube rounds for the cladding, bar stock for end plugs, semi-finished castings for tie plates and strip for channels and spacers. Rated capacity for completed fuel bundles is being expanded from 1000 tons contained uranium per year to meet load. The product mix requires precise control of operations to accommodate several different U^{235} enrichments and concentrations of gadolinia.

In addition to these production facilities, Figure 1 shows three current licensed manufacturing affiliates... Kernreaktorteile jointly owned with Allgemeine Elecktricitaets-Gesellschaft; Fabbricazioni Nucleari with Ansaldo Meccanico-Nucleare; and Japan Nuclear Fuel Company, with Hitachi, Ltd., and Tokyo Shibura Electric Co., Ltd.

Wilmington Manufacturing

The overall product scope of the Wilmington, North Carolina facility is the most comprehensive. The activities in the three affiliated companies will be described later.

The 2-1/2 square mile Wilmington site, as shown in Figure 2, is devoted to the manufacture of nuclear fuel and other reactor components. The large building in the foreground is the Fuel Manufacturing Operation. UF_6 is converted to ceramic grade UO_2 , made into pellets, loaded into cladding, degassed, welded, assembled into bundles and packed for shipment.

The Fuel Components Operation is devoted to the fabrication of zirconium components...end plugs, tubing, flow channels, and spacers for fuel bundles. Upper and lower tie plates are manufactured in the next building in addition to control blade assemblies, drives, and other stainless steel reactor internals.

The overall process is displayed in Figure 3. There are six major component groups which flow together to consummate a final fuel assembly. Each of these will be discussed in sequence.

Figure 4 shows detail of the nuclear fuel portion of the prior slide. The squares are operations, circles are inspection points, and the solid dots designate the generation of permanent records.

Uranium hexafluoride is received in 2200 kg cylinders. After weight verification and certification of the contents, the cylinder is heated, evaporating and driving the contents into the conversion process. A modified ammonium diuranate process produces a ceramic grade powder which is fed to pellet presses for the manufacture of green pellets.

A chemical recovery process is employed for reprocessing of excess and reject material. Solutions can be mixed to provide maximum assurance of homogeneity, permitting adjustment of enrichments for smaller quantities of material required for a specific project, thereb, reducing inventories.

A fundamental philosophy of the overall fuel operation is to minimize points at which different feed materials are introduced into the process, for example green scrap is dissolved, reprecipitated, and converted to ceramic grade oxide.

The product of the primary ADU lines is processed through one series of lines; and the product of the reprocessing operation is processed through separate lines. In this manner all residual products are reprocessed in a uniform manner. Throughout these operations there is strict control of process parameters, and intermediate product to assure appropriate quality. Enrichment is checked by a multiple channel enrichment analyzer, one of which is shown in Figure 7 utilizing direct gamma emission of the product.

To conserve space and assure controlled location in storage, a stacker crane and grid storage shown in Figure 5 are used for bulk containers, providing a critically safe array and easy access.

A new conversion process is being put on line which results in substantially less waste stream generation and attendant treatment.

Production of pellets is accomplished in multiple cavity hydraulic presses as shown in Figure 6. Prior to release of powder to the pressing operation, the enrichment is again checked by gamma counting in equipment shown in Figure 7. Where production volumes are high the more conventional presses can be augmented with high speed pressing equipment. An enrichment identification code is pressed into one end of each pellet.

Sintering of the green pellets is accomplished in a continuous furnace in a controlled atmosphere to achieve specified density and structure in the sintered pellets. Special control of temperature and atmosphere profiles is used to improve yields and to provide a more uniform product.

Subsequent to sintering, pellets are checked for chemical impurities, gas content and density.

Pellet grinding is accomplished on centerless grinders with mechanized loading and discharge across an inspection table ready for loading into pellet trays for drying prior to loading. After drying the pellets are checked for moisture content.

Pellets are then stored in critically safe cabinets with attendant identification and material control awaiting loading.

Tubing with the first end plug welded in place then enters the process. Pellets from the storage cabinets are released on a controlled basis and placed in mock-up channels weighed and inserted into the tubing as depicted in Figure 8.

After rod loading and weighing, rods are subjected to hot vacuum outgass ng in equipment shown i. Figure 9. The treatment removes moisture and other volatile material. Statistically selected rods are then checked in the laboratory to assure the outgassed rods conform to specification prior to release for bundle assembly. Final welding employs tungsten inert gas welding after a prior evacuation and backfill with helium.

In addition to fuel, the rods contain a plenum spring and a proprietary hydrogen getter. The getter provides preferential absorption of any traces of residual hydrogen or moisture.

The fuel rod is subjected to final physical inspection and scanning to assure that the pellets in the rod are of the proper enrichment. One method which has been used for lower production relies on the decay gamma of the fuel similar to the enrichment analyzer used for powder and pellets. The equipment is shown in Figure 10.

An active scanning technique shown in Figure 11 is being applied when required by high production rates, producing more rapid and sensitive scanning than previous methods. Transmission gamma scanning of the total rod measures pellet density, simultaneously detecting any gaps or defects in loading. The output information is integrated in data processing equipment providing a record and immediate on-line readout of acceptable and nonacceptable rods. Output data include integrated U^{235} content providing additional accountability information. Fuel rods are then stored in trays in a critically-safe array ready for fuel bundle assembly.

Manufacture of gadolinia-bearing urania fuel rods follows a similar process except for intensive blending required to assure homogeneity. Both the blended gadolinia-urania powder and resultant sintered pellet are checked for gadolinia content and for solid solution distribution of the gadolinia.

The integrated operation is carried out in a segregated facility - complete unto itself - from receipt of the UO_2 , Gd_2 , O_3 , the tubing and hardware, until the rod is final welded. This avoids contamination of the balance of the operation with Gd.

Tubing is manufact red in the Fuel Compone to Operation. Extruded tube rounds are received and inspected, then subjected to a series of tube reduction steps accomplished with intermediate cleaning and annealing, all with control of sensitive appropriate process variables.

After final annealing, straightening and polishing, the tubing is subjected to rigorous nondestructive testing for dimensions, surface and internal defects and mechanical properties. Typical ultrasonic equipment used is shown in Figure 12. After passing NDT inspection, tubing is cut to length and both the ID and OD are cleaned and autoclaved. Autoclaving is followed by inspection for uniformity of the autoclaved surface. Subsequent to this operation tubing is released for TIG welding of the first end plug.

End plugs are manufactured from bar stock by a process which precludes the possibility of porosity. After cleaning and inspection the plugs are packaged to preserve cleanliness, stored and released as necessary to the line for welding.

Bundle spacer assemblies are fabricated from zirconium strip and Inconel springs. Subassemblies are placed in suitable jigs and fixtures and the spacer welded by a sophisticated, sequencing, multiple head TIG welding apparatus. After inspection, the spacer is cleaned and autoclaved. The spacers are then inspected for cleanliness, packaged and stored ready for use in bundle assembly.

Upper and lower tie plate castings are machined by numerically controlled equipment in the stainless steel facility previously mentioned, cleaned, inspected, packaged and stored ready for assembly.

With the foregoing components, the assembly of a complete fuel bundle is accomplished by placing the lower tie plate and spacers in fixtures on a suitable table. The spacer capture rod, i.e., the rod that locates and secures the spacers axially, is placed in position. The remaining fuel rods are inserted and the upper tie plate positioned and secured.

The assembly table is then elevated and the bundle transferred, as shown in Figure 13, to leak checking. The bundle is then positioned in the final mechanical inspection fixture. Plastic spacers are inserted between rods to protect the bundle from mechanical damage during shipment and the bundle enclosed in a protective plastic envelope. Bundles ready for shipment are then stored in the rack pending final packing. On receipt of fuel at reactor site, fuel bundles are reinspected.

Flow channels are fabricated from strip, press formed, TIG welded, sized, cleaned, autoclaved and finally straightened. The channels are then recleaned, inspected, and packed for shipment to the reactor site for assembly in bundles.

Quality Assurance

Over 200 personnel are assigned directly to Quality Assurance. The overall function is organized as shown in Figure 14.

Increasing surveillance of fuel manufacturing processes and records by both customers and governmental agencies requires coordination of communications centralized in the Customer Relations function.

Quality Assurance Engineering is responsible for quality systems planning, advanced activities, purchased material control and quality data systems.

Quality Information Equipment Engineering, Figure 26, provides inspection and test systems engineering, nondestructive test, gage and instrumentation control and calibration.

Process Control Engineering, a key function, assures that quality is controlled at the earliest possible point in the process.

Quality verification is responsible for receiving, in-process, final factory and reactor site inspection. All data are reviewed to verify that specification requirements have been met.

In addition to the 18 requirements of 10CFR50, Appendix B, there are 20 additional mandatory criteria imposed by the Company.

Typical Quality-Information Equipment Engineering activities are the development and application of new inspection and nondestructive test methods and equipment.

Materials from which fuel is made are subjected to an overall reactivity check in a neutron test reactor prior to release for fabrication. This test assures that there are no inadvertent chemical constituents present which would act as neutron absorbers.

Manufacturing Information Control System (MICS)

Closely coupled to advanced material control, sophisticated queuing and production control, are the following: nuclear material management, inventory control, product quality data, process yields, traceability, and verification of product prior to release for further processing or for shipment. The Management Information Control System has been provided to fulfill these needs as shown in Figure 15.

Current operations are conducted with a FACTS data system which accumulates production and material control data. MICS being installed has substantially more capability. Pertinent data fed into the system are collated and matched against prior history including identity, weight, and quality. The system indicates whether a specific container or batch may be released for a specific operation.

The computer system provides redundancy to avoid production delays. Simple station input procedures identify the project, operator, the date and time, analytical and test information, and the content in normal production units. The latter is translated in parallel into nuclear material management data. Each input station also permits inquiry by the operator for specific information or rechecking purposes.

The system identifies containers by discrete numbers and locations, permitting tracking of material movement, providing a continuing upto-date inventory balance. Accountability information is carried by the same system. The system permits rapid detection and reporting of both product and uranium discrepancies to permit prompt corrective action. The inquiry-release system is coupled with movement tracking of container movement. This will effectively preclude enrichment mixing. The inputs of quality control data provide permanent information for the release of material for further fabrication and for accountability purposes.

The bank of data generated provides a permanent product history from raw materials, through the process operations, to the final product.

Station-by-station input permits the accurate and timely reporting of process yields, and measurement of personnel and equipment performance.

With increased throughputs in which numerous enrichments are processed, nearly two million transactions will be made on the computer system next year just to track the uranium material in various stages of manufacture. An estimated quarter of a million container movements will be involved. At any one time an estimated 60,000 containers may be in inventory.

Nuclear Safety

There are several aspects to nuclear safety including:

- 1. Shipment of nuclear material.
- Operation of the plant including criticality, accountability, and radiation control.
- 3. Control of airborne, liquid and solid releases from the plant.

Shipments are covered by the respective regulatory agencies in the involved countries.

Criticality control in manufacturing is obtained by a combination of area batch limit control and geometry. Nuclear absorbers and control of moderating agents may also be employed to conserve space and facilitate operations.

Monitoring of criticality as shown in Figure 16 is achieved by a detection system with gamma sensors located at strategic points in the plant.

The MICS system provides additional assurance of proper identification and content of all containers to prevent improper accumulation or discharge of radioactive material.

Airborne contamination is monitored continuously in the air filtration system. Personnel exposure to airborne radioactive material is regularly monitored through the use of in-plant area air sampling and personal lapel air samplers.

Surface radiation levels are controlled by routine monitoring with corrective action when control limits are exceeded.

Radioactive liquid waste systems are treated in various methods depending on their nature. Certain dissolved radioactive material is treated by ion exchange followed by precipitation of the depleted stream and subsequent centrifugation. Low level liquid effluents that carry particulate material are pH adjusted to assure precipitation of all the radioactive material and then centrifuged.

The underflow of the centrifuges is either reprocessed to recover uranium if the content is economic or contained and buried if not.

Solid wastes including packaging material, cleanup material, discarded equipment or portions thereof if combustible, are treated by incineration with appropriate filtration and reprocessing of the ash; or if noncombustible, disposed of by cont inment and burial.

Radiation exposure of personnel is continuously monitored with film badges. In addition, regular checks for lung burden are made by a whole body counting method. Operating personnel who may be exposed to highly soluble forms of radioactive material are regularly checked by the bioassay method. All personnel are alpha monitored when leaving a radioactive materials area.

Environmental Control

The nuclear industry and the regulatory bodies have established limits for radioactive discharge. Monitoring of the environs of the factory is routinely carried out to assure that in-plant control is effective.

With the spotlight on environmental control, all effluents of nuclear manufacturing facilities are coming under ever more rigorous control.

The Wilmington facility wastes include plating wastes containing Cu, Ni, Cr, with HNO_3 , H_2SO_4 , HNO_3 -HF, NH_3F , various oxides of nitrogen and NII_4NO_3 . Discharge of these materials in a noncontrolled manner is prohibited and controlled discharge is limited to extremely low levels.

Figure 17 shows the waste recovery plant just put into operation. These facilities have been added to augment previous facilities. Nitric and hydrofluoric acids are neutralized with lime, concentrated and stored. Nitric acids which may contain traces of radioactive material are neutralized, concentrated and settled. The supernatant liquor is centrifuged and the underflow of the centrifuge and the precipitated compounds set aside for recovery, if economical, or packaged for burial.

International Activities

A primary aspect of international joint ventures is the extent to which the overall resources are extended, both technical and material. The affiliate is provided with consultation to assist in business planning, facility design, preparation of necessary regulatory license documents, procurement and business systems. Initial and follow-on training of personnel is provided in facilities at Wilmington and San Jose. General Electric personnel assigned to the affiliate are responsible for continuation and refinement of the training and for initial operation of the plant.

As necessary, e.g., during plant startup, or if specific problem areas arise, qualified personnel provide support.

Regular refresher assignments of affiliate personnel at GE facilities are an integral part of the plan to provide maximum communication of the progress in manufacturing processes and systems.

Cross-checks on quality control tests are provided to assure agreement with national standards and correlation from plant to plant to assure consistency of measurement. The affiliates are visited periodically by a technical team to review operations in the affiliated plant providing direct in-plant communication.

Backing all of the foregoing is a licensee and support technical operation at San Jose providing pertinent manufacturing information routinely and on demand to assist in answering specific questions as they arise.

Fabbricazioni Nucleari located near Bosco Marengo in the Po Valley south of Milan is under construction at the present time and will start operations early in the Spring of 1973. Figure 18 shows an architectural rendering of the plant. The manufacturing product scope is light boiling water fuel starting with ceramic grade UO_2 and finishing with completed fuel bundles shipped to the site. The operation also provides for the assembly of mixed oxide recycle fuel.

Figure 19 shows a recent view of construction operations.

Japan Nuclear Fuel, just south of Yokohama, has been operating since 1971 with a product scope similar to that for FN. The facility is shown in Figure 20. A vacuum outgassing operation within the plant is shown in Figure 21.

Kernreaktorteile near Frankfurt commenced operation in 1966 with a somewhat broader product scope including the production of control blades. The overall facility is shown in Figure 22. The facility has recently been expanded doubling its capacity. Figure 23 shows a portion of the press room.

Figure 24 shows a fully automatic grinding station.

The production requirements for all three affiliated companies and Wilmington are growing rapidly. KRT is scheduled for a near term major expansion, JNF requires a multiplier of its capacity to meet firm order work load and Wilmington is being enlarged and improved. As these plants are expanded, the complexities of multiproject operation demand the application of the latest technology and flexibility to assure that quality products are produced. Concurrently, environmental requirements must be met and the product must be delivered on schedule.

TECHNICAL AND DEVELOPMENT - SCOPE RELATED TO FUEL MANUFACTURE

Fuel for commercial power light-water reactors has been under intensive development for over 14 years. Nevertheless, continuing pressure for increasing fuel performance, reliability, scale of operation, and economics of each element of the fuel cycle provide continuing increases in incentives for intensive development in manufacturing processes and specifications. Fuel for first-generation water-cooled reactors has behaved extremely well, generally exceeding the stated design capabilities. Occasional defects usually have been traceable to known and correctible deficiencies in manufacture, handling or operation.

Successive generations of design have made increasing demands on fuel performance. For example, nominal fuel lifetimes have nearly doubled for BWR's and nearly tripled for PWR's. Linear heat generation rates (kW/ft, or W/cm) have increased more than 50% for both BWR's and PWR's. Volumetric power density (kW/l) has increased about 60% for BWR's and almost 100% for PWR's. Other changes have included the standardization of use of zirconium cladding starting in the late 1950's for BWR's and in the late 1960's for PWR's, the elimination of axial segmentation for both, and an increase in active fuel length to 12 feet. Another performance environment change has been the increasing requirement and use of load following - initially only with BWR's and more recently with some PWR's.

For BWR's, the amount of energy generated using zirconium clad fuel which operates at the increased linear power conditions of a modern large plant (12 kW/ft or higher) has increased six-fold in the last three years. A similar increase in high-rated PWR cores with zirconium clad will occur in the period of 1972-1974.

With the increasing levels of fuel duty and sharply increasing volume of exposure statistics, there is more rapid and dramatic accumulation of evidence for life-limiting or "wear-out" mechanisms. In any instance there is always the interesting question of whether a given effect on fuel is due to inherent limitations in the design, in the environment, or in the manufacturing and inspection process. The nature of fuel reliability requirements is such that the effective "end-of-life" (for a given fuel type and manufacturing history) is reached when a few tenths of a percent of the fuel rods have failed. This level of rod failure corresponds to 25% failure on a fuel bundle basis for BWR's, and 100% failure on a bundle basis for PWR's. Except for creep-collapse in some light water reactors with nonfreestanding clad designs, most of the performance-limiting phenomena observed to date are the result of statistically infrequent combinations of factors. The interesting question of design vs. environment, vs. manufacturing and inspection process, must be re-evaluated again and again each time there is an increase in the size of the statistical population and whenever there are changes in design, environment or manufacturing techniques.

For General Electric Boiling Water Reactor fuel manufacture, the foregoing involves continuing interaction between development testing, diagnostic examination of fuel from power reactors (large-scale statistical base) together with intensive research and development efforts on the basic phenomena and materials properties which are perceived to be significant in life-limiting or "wear-out" processes. The interactions between the fuel and the reactor system are now increasingly important and lead to changes in design and manufacturing.

The following development areas are significant in that they contribute to the setting and revision of manufacturing and inspection tolerances:

Fuel and Core Development Areas

- (a) Nuclear methods development (reactivity and control, and over-all and local power distribution).
- (b) Thermal hydraulic methods and tests (critical heat flux margins, emergency core cooling margins, local void and temperature distributions).
- (c) Fuel modeling and stress analysis (including dimensional changes due to thermal and burnup effects; elastic, plastic, creep, and fuel redistribution effects).

- (d) Fission product distributions, relocation, and their effect.
- (e) Materials development (properties effects of composition and process variables, and effect of exposure variables).

Manufacturing Development Areas

The following areas of development are more directly in support of the manufacturing lines:

- (a) High capacity powder production (reduced downtime and improved yields).
- (b) Mechanization and/or automation process steps (increased sensing and automatic control of process variables and material handling).
- (c) Improved powder handling and pellet pressing for production rates up to 325,000 pellets per day per station.
- (d) Increased sensitivity of inspection procedures; high speed inline inspection; increased mechanization and/or automation of inspection; and computerized statistical control methods for process and material variables, materials flow and balance, accountability and safeguards methods.
- (e) Pre-testing of unit process improvements in pilot line and productionscale equipment both at Wilmington and San Jose.

Integral Testing

Design and process development is integrated in fuel testing development efforts. This includes the following activities:

- (a) Idealized individual parameter variation tests usually instrumented few-rod test basis.
- (b) Small-scale statistical tests with controlled environment.
- (c) Large-scale statistical measurements and correlations on power reactor fuels.

- (d) Special instrumentation and diagnostic measurements and observations on power reactor fuels.
- (e) Accelerated and overload performance tests of fuel in GETR.

Support Effort and Facilities

The manufacturing effort described here is supported by a massive, continuing effort in design and product and process development and testing. A well-coordinated effort is carried on in San Jose, the Vallecitos Nuclear Center, and the GE Research and Development Center in Schenectady.

Effect of Specific Fuel Performance Observation on Manufacturing

Various mechanisms of failure have been observed and analyzed, both in controlled tests and in large-scale statistical exposures in various water-cooled reactors. Some effects are common to several different reactor types; some are unique to one type.

- (1) Clad-splits due to tubing flaws.
- (2) "Hydriding" failures due to moisture and/or hydrogenous materials in fuel.
- (3) Fuel-clad interaction.
- (4) Increased surface temperature and corrosion rate resulting from film deposition.
- (5) Creep collapse of cladding.
- (6) Gaps in pellet column.
- (7) Clad fretting due to foreign material trapped at spacers.

It is evident that of these effects, the film ("crud") deposition and fretting by foreign materials are independent of fuel manufacturing process. Two of the items are primarily related to manufacturing [(1) and (2)]. Three of the items (fuel-clad interaction, creep collapse, and pellet gaps) involve a combination of design, materials

specifications, and manufacturing process variables and tolerances. For example, BWR's (and some PWR's) use free-standing clad, with relatively low working pressure and high clad strength. Creep collapse has not been observed in BWR Zircaloy clad fuel after more than 8 years' exposure.

Pellet column gaps have also not been observed in BWR's because of higher pellet density, larger initial fuel-to-clad clearances, and the absence of creep collapse.

Manufacturing Changes from Fuel Performance Results

With the exception of the fuel film deposition problem and fretting by foreign materials (which require system remedies), each of the known or suspected performance limitations in fuel has resulted in one or more changes in design and associated process and inspection specifications. Some of the more significant of the resulting changes corresponding to observations listed previously are:

(1) Tube flaws:

Improved tubing fabrication.

Redundant inspection which simultaneously checks and rechecks several characteristics.

(2) Hydriding:

Vacuum degassing and control of process conditions, acceptable statistical levels, incorporation of getters.

(3) Fuel-clad interaction effects:

Pellet L/D reduced and edges chamfered; dishing eliminated. Clad thickness and heat treatment changed, variability reduced, increased inspection sensitivity, smaller tolerances. Smaller rod diameters.

(4) Film deposition:

System changed.

(5) Creep collapse:

Lower working pressure; increased clad t/D; increased clad strength.

(6) Gaps in pellet column:

Increased pellet density; increased radial gap; freestanding clad.

(7) Fretting by foreign material:

System changed.

Continuing incentives for refinement of manufacturing specifications and quality levels result from the increased statistical levels of power reactor fuel experience. Large-scale statistical results (say 10,000 rods), and increasing levels of fuel duty serve to focus development and testing on more precisely defined material duty cycles. Such data, along with increasingly well-calibrated fuel modeling and stress analysis, result in progressive refinement and an improved basis for many specifications and tolerances. Statistical combination of material property distributions, environmental condition distributions, and manufacturing process variable distributions can now be determined through increasingly large-scale fuel manufacture, exposure, and diagnostic examination. Statistically infrequent effects which were hitherto inaccessible will increasingly determine the selection of manufacturing processes, tolerances, and inspection procedures.

Summary

Large-scale manufacture and compounding exposure data on fuel performance leads to increasing stringent demands on the fuel manufacture, and also to improved capability to meet these requirements. Continuing flexibility to exploit refinements in process and design is necessary to meet such demands. Considerable evolution in BWR fuel has occurred for over 14 years. Improvements, both in productivity and refinements to offset various lifelimiting effects have been developed and implemented. At the same time production levels have increased to over 1000 tons/year, and near term increases to levels exceeding 1500 tons/year are required. Increasingly reliable and economic fuel will result from this process of continuing iteration and refinement of process control and quality methods.



INTERNATION FUEL LICENSING

1. International Fuel Licensing



2. Aerial Photograph of Wilmington, N. C. Plant
FUEL BUNDLE PROCESS FLOW



3. Overall Fuel Fabrication Process Chart

NUCLEAR FUEL



4. Fuel Line Process Chart (with branches)

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5. Stacker Crane and Grid Storage



6. Pellet Pressing



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9. Vacuum Outgassing Operation



Passive Scanning



11. Active Scanning



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13. Fuel Bundles Testing, Storage, and Shipment



14. Overall Quality Assurance Organization

MICS CAPABILITIES



15. Overall MICS Functional Chart



Criticality Monitoring

16.



17. New Wilmington Waste Treatment Facility



18. Architectural Rendering of FN



19. FN Construction Operations



JNF Exterior

20.



21. JNF Vacuum Outgassing



22. KRT Exterior



23. KRT Press Room



24. KRT Grinder Station

Production of Fuel Material and Fuel Elements for High Temperature Reactors

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Abstract

For experimental and prototype high temperature reactors spherical as well as prismatic fuel elements are produced using the thorium-uranium fuel cycle.

In both types of elements the fuel material consists of coated particles which contain spherical ceramic fuel kernels. These fuel kernels which have a diameter of 200 - 600 um are either oxides or carbides. Powder metallurgical as well as chemical processes are used. Production plants have throughputs of several tons of kernels per year. In fluidized bed furnaces the kernels are coated with pyrolytic carbon layers. Coaters for 80 kg thorium kernels per day are in operation. Two production coaters working nearly fully automatically can be run by one operator.

Up to now more then 110.000 molded spherical fuel and absorber elements have been produced for the pebble bed reactors AVR and THTR. One pressing line for THTR elements delivers 500 spheres in 8 hours which then are heated up to 1900°C.

The most actual type of prismatic elements is the GGA design used for the Fort St. Vrain reactor, a hexagonal block of 790 mm height and 360 mm across flats. The production of the elements consists of three steps: machining the graphite block production of fuel sticks - assembly.

An alternate prismatic element is the molded Nukem block element being still in development. The whole element including fuel zones and cooling channels is fabricated by integral pressing, so obtaining a fuel element with low thermal resistance, capable to reach high fuel loading.

For quality control precise characterization procedures have been developed reaching a high degree of automation. The irradiation performance of the fuel elements has been shown in many reactor tests up to high neutron dose and fuel burn up.

1. Introduction

The special feature of fuel elements for high temperature reactors is that they contain the fuel in a graphite matrix. The graphite is structure material and moderator material as well and is also used as reflector material in the core.

Two types of fuel elements have been developed for the high temperature reactor, the prismatic element especially in the US and also in England, and the spherical element in the Federal Republic of Germany; in Germany, however, during the last years, intensive efforts have been made for the development of an advanced prismatic element.

2. Requirements

The requirements for the fuel elements are extremely high as to temperature, burn up, fast neutron dose and retention of fission products.

The residence time for the fuel elements in the core is approximately 4 years, and an increase of the residence time is desirable. Burn-up is nowadays between 80 000 and 700 000 MWd per ton, depending on the type of fuel cycle involved. The fast neutron dose, which essentially influences radiation behavior and residence time, reaches values of 8 x 10^{21} nvt at E greater than 0,18 MeV.

The maximum central temperature of the fuel material may reach 1350 centigrades, while the surface temperature is about 1000 centigrades.

The release of gaseous and solid fission products must be below about 10^{-4} , related to the total amount of fission products generated during the operation of the reactor. These requirements for the high temperature reactor fuel can be fulfilled today.

. Fuel cycle

Two different fuel cycles have been proposed for high temperature reactors:

- the thorium-uranium cycle and
- the uranium or so-called low-enrichment cycle

In the thorium-uranium fuel cycle a fuel element contains uranium fissile material and thorium fertile material in different coated particle types. Only the thorium containing fuel will be reprocessed in order to get the U-233 free from other uranium isotopes and this U-233 will be recycled after refabrication.

The ratio thorium to uranium varies from 10:1 to 20:1 depending on the reactor design. The fuel material will reach burn-up rates of approximately 700 000 MWd per ton for the fissile material and approximately 90 000 MWd per ton for the fertile material.

The thorium-uranium cycle is used in the AVR-Reactor in the Federal Republic of Germany and in the Peach Bottom Reactor in the United States and will be used for the Fort St. Vrain Reactor and the large power stations of 1160 MWe and also in the THTR-Reactor which is now under construction.

In the so-called low-enrichment cycle 5-9 % U-235 is used. In this cycle the spent fuel contains plutonium and this may be reused in high temperature reactors after reprocessing and refabrication. In this fuel cycle the uranium burn-up is about 80 000 MWd per ton. Recycling of plutonium however is not a specific criteria of the low-enrichment cycle because the plutonium can also be used in the thorium-cycle.

At present time there is no high temperature reactor with low enrichment cycle in operation or in construction.

- 4. Fuel materials
- 4.1 Description of the fuel materials

Independently from the type of fuel element and the type of fuel cycle the fuel material consists in any case of coated particles which contain spherical ceramic fuel kernels. These kernels are coated with different layers of pyrolytically deposited carbon. The coating ensures the retention of gaseous and solid fission products.

The ceramic fuel kernels have a diameter of 200 to 600 microns, depending on the requirements, and consist either of uranium or thorium carbide or of the corresponding oxides. <u>Figure 1</u>

The pyrolytically deposited coatings are multiple layers of carbon or silicon carbide. There are normally 2 or 4 of these layers. <u>Figures 2 and 3</u>

h.2 Production of fuel kernels

For the production of carbide or oxide fuel kernels powder metallurgical as well as chemical processes are used. In the powder metallurgical process the heavy metal oxide prowders are granulated to spherical kernels together with different types and amounts of binder. A sintering step at temperature of about 1500 to 1700 centigrades follows. For the production of carbide kernels according to this process carbon is added to the heavy metal oxides together with the binder before granulation. These kernels are converted into carbide at approximately 1800 centigrades. Melting at about 2400 centigrades gives dense spherical kernels. This can either be done in an inductively heated graphite powder bed or by a special arc melting method.

During the last years chemical processes have been developed especially for the production of oxide kernels. Here the feed material are metal solutions or metal soles which are dropped by means of a nozzle system into a solution where the precipitation and at the same time the solidification of the droplets takes place. Figures 4 and 5

After washing, drying and calcinating these kernels are sintered to oxide kernels in an usual pushing tray furnace. Figure 6

These wet processes can be used for the production of pure uranium oxide kernels, pure thorium oxide kernels and mixed oxide kernels.

The wet processes also are applicable for the production of carbide kernels. It is only necessary to suspend carbon into the solution. The kernels are reacted to carbides, or they are melted as mentioned before under the tropic of powder metallurgical processes.

It must be pointed out that the chemical processes are more elegant and more simple than the powder metallurgical processes, especially under the aspect of big productions. Today different wet processes are known and we can actually say that the production of large quantities of kernels under industrial conditions is solved. Such production plants have throughputs of several tons of oxide kernels per year.

4.3 Coating of fuel kernels

The deposition of pyrolytic carbon on fuel kernels is performed batchwise in fluidized beds at temperatures between 1500 and 1900 centigrades, depending on the deposition gas used and on the desired properties of the coating layers. As deposition gas acetylene, methane or propene are used.

The fluidized bed consists of a graphite tube heating element with an inside concentric reaction tube where the carbon layers are deposited on the kernels. The diameter of such a coater is approximately 240 mm.

The quality of the deposited carbon layers depends mainly on the deposition rate and the deposition temperature. The conditions for obtaining coating layers with specified characteristics are well known today. At the present time coaters with a capacity of approximately 10 kilograms per batch for thorium kernels are in operation. For uranium kernels the batch size is smaler due to criticality problems. At a first look 10 kilograms seems to be not a large quantity, however a throughput of up to 80 kilograms heavy metal within 24 hours seems to be possible. This is a capacity of 20 tons per year and the amount of thorium fuel for a 1160 MWe reactor can be produced in one year using two coaters.

It must be pointed out that the coating technology is nowadays on a high standard and many tons of coated particle fuel can be produced in an industrial scale. Figure 7 shows a nearly fully automatically working coating furnace. In the moment two of these furnaces can be run by one operator. Further development work is still necessary to improve process economy.

- 5. Fuel elements
- 5.1 Spherical elements

The present spherical element consists of a two component graphite matrix and fuel particles. The sphere has an inner core containing the fuel. The core has a diameter of approximately 50 mm. A surrounding fuel free shell of 5 mm thickness is pressed onto the core resulting in a homogeneous structure throughout the element. For the production of spherical fuel elements a special resinated graphite powder is mixed with fuel particles and molded in a rubber die to form the inner core. <u>Figure 8</u>. In a second step this fuel containing core is embedded in further resinated graphite powder to form the fuel free shell. After pressing the spherical element is treated at 900 centigrades, degased for a short period at 1800 to 1900 centigrades and machined to a diameter of 60 mm. <u>Figure 9</u> shoes a flowsheet of the production of the molded spherical fuel element. In the figures 10 and 11 we see a metallographic section and a x-ray picture of a spherical fuel element.

More than 60.000 fuel elements of this type have already been produced for the AVR reactor. At the present time fuel elements for the THTR are produced. 50.000 absorber elements and subsequently the fuel elements are produced with a rate of 1000 per day. <u>Figure 12</u> shows the pressing production line which has a throughput of 500 elements per 8 hours and is run by one operator. The above mentioned figures demonstrate that the spherical element is already produced in large quantities.

5.2 Prismatic elements

Concerning this type of element the most actual design is the GGA design. This fuel element will be used in the Fort St. Vrain reactor.

The GGA element consists of a prefabricated hexagonal graphite carrier block of 790 mm height and 360 mm across flats. <u>Figure 13</u> There are 102 cooling holes of 16 mm diameter as well as 210 fuel holes of 12,7 mm diameter in the block. The carrier block is machined out of an extruded cylindric graphite rod.

The fuel material is comprised in so-called fuel sticks of 50 mm length and 12,5 mm diameter. These fuel sticks are inserted into the fuel holes of the block.

For production of the fuel sticks the required quantity of fuel particles is vibrated into small aluminum tubes resulting in a dense packing. Then a fluid warm mixture of pitch binder and natural graphite powder is pressed into the aluminum tube. After cooling the fuel stick can easily be taken out of the die. The fuel sticks are heated up to about 800 centigrades and subcequently heated to a temperature of 1800 centigrades.

The fuel sticks produced in this way are inserted into the fuel holes of the block. The holes are then closed by a graphite plug. The essential steps of this production process are as follows:

- machining of graphite carrier blocks with hexagonal geometry and fuel holes as well as cooling holes
- production of fuel sticks
- assembly

It is possible today to produce more than 30.000 fuel sticks per 24 hours which corresponds to 15 block type fuel elements.

An alternate approach to block type fuel elements is the molded NUKEM block element. <u>Figure 14</u> The main advantages of this type of fuel element are:

- high fuel volume at high mechanical strenth
- low thermal resistance

The whole element including fuel zones and cooling channels is fabricated by integral pressing and subsequent heat treatment. Figure 15

Therefore the mechanical strength is homogeneous throughout the block and the whole block is serving as supporting structure. More over sufficient space is available for accomodating a cooling hole pattern, optimum with respect to low pressure drop and cooling efficiency. The low thermal resistance is obtained because of two things

- high thermal conductivity of the matrix which is homogeneous throughout the block
- efficient heat transition from the fuel particles into the matrix due to the absence of any gap.

This molded block type fuel element is still in the state of development.

6. Quality control

The control methods allow a precise characterisation of the fuel kernels as well as of the coatings and the graphite matrix. These methods have reached an advanced degree of automation which can be demonstrated for instance by the control method for coated particles.

In this case, the goal is to determine the kernel diameter, the kernel density, the thickness of the coating layers as well as the density of the layers. This is automatically performed by a particle counter in combination with a computer.

7. Irradiation performance

Parallel to the laboratory development the high temperature reactor fuel is tested in extensive irradiation programs sponsered by the German government and coordinated by the Kernforschungsanstalt Jülich.

Figure 16 gives a survey on the most important present irradiation test. The dose and burn-up requirements for THTR are already exceeded by some specimens. The irradiation of the others, marked by arrows, is continued.

The most important result is that the fuel particles and the surrounding matrix which is used for both the spherical and the prismatic elements are compatible with each other. <u>Figure 17</u> No irradiation damage can be observed after exposure to burn-up, temperature and dose values exceeding reactor operation conditions Satisfactory fission product retention by coating layers and graphite matrix has been found.

8. Summary

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It was the goal of this paper to demonstrate that the production of fuel material and fuel elements for high temperature reactors is now advanced so far that reactors of this type can be constructed under commercial conditions.



Figure 1



Figure 2



Figure 3



Figure 4



Figure 5



Figure 6



Figure 7



Figure 8







Figure 10



Figure 11



Figure 12

Figure 13



Figure 14



Figure 15



Figure 16



Unirradiated

Irradiated to full neutron dose

Fig.5: Metallographic sections of THTR fuel element showing one , or' embedded in matrix (Test R2-K3)

Figure 17

MANUFACTURING CANDU FUEL



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RÉSUMÉ - FABRICATION DU COMBUSTIBLE CANDU

La conception des combustibles CANDU est simple et des procédés de fabrication conventionnels ont été utilisés pour sa production. Ces procédés peuvent être mis sur pied soit par une main-d'oeuvre intensive ou par semiautomatisation. Il existe des méthodes de protection du personnel contre les dangers que présentent certains de ces procédés. Le personnel employé dans une usine de fabrication de combustibles CANDU doit posséder la compétence technique nécessaire pour interpréter la conception du combustible en langage de fabrication, spécifier et diriger les procédés, développer et mettre en place des méthodes de contrôle de la qualité appropriées. Le coût peu élevé des combustibles nucléaires est le résultat de la simplicité de conception du combustible CANDU qui a permis l'application de techniques de production conventionnelles.

ABSTRACT - MANUFACTURING CANDU FUEL

The design of the CANDU fuel bundle is simple, and conventional manufacturing processes have been modified for its production. These processes can be installed on either a labor intensive or a semi-automated basis. Procedures are available to protect personnel against the health hazards that are associated with some of the processes. Manufacturing organizations engaged in producing CANDU fuel require persons with the technical competence to interpret the fuel design into manufacturing language, to specify, install and operate the processes, and to develop and put in place appropriate quality control procedures. The simple design of CANDU fuel which has permitted the application of conventional manufacturing techniques in its fabrication has resulted in low cost fuel.

INTRODUCTION

By the end of 1971 more than 800 000 elements for CANDU fuel bundles had been manufactured in Canada. This paper draws on the experience that has been gained in the production of this fuel and presents a description of a typical CANDU fuel bundle, flow charts for the manufacture of CANDU fuel and a discussion of some of the parameters that are associated with the manufacturing processes and techniques, a discussion of some of the items that have to be considered for safety and for health reasons, and an outline of a manufacturing organization for producing replacement fuel for two 540 MW CANDU generating stations.





CANDU FUEL BUNDLE

For the purposes of this paper the nuclear fuel bundle for Pickering Generating Station is used as the reference CANDU fuel bundle. This fuel bundle, shown in Figure 1, consists of 28 closely spaced elements each of which contains high density natural UO_2 in thin Zircaloy sheath. The elements are held together by plates welded to the ends and are separated by spacers brazed to the sheaths. The bundle is 50 cm long and 10 cm in diameter and contains approximately 19.7 kg of U and 2.1 kg of Zircaloy.

Because the CANDU reactor system uses natural uranium as fuel, the fuel bundle has been designed to keep materials other than UO₂ to a minimum in order to achieve good neutron economy. The original designers of CANDU fuel achieved this goal by using an element that has a nominal wall thickness of 0.42 mm and by designing all other parts of the bundle as near as possible to the minimum practical dimensions. This thin element sheath collapses onto the pellets when the fuel bundle is exposed to the operating pressure in the reactor. Because of this the clearance between the pellets and the sheath must be kept small enough so that the sheath will collapse without forming a sharp axial ridge. At the same time the clearance between the pellets and the sheath must be large enough to maintain the sheath strain caused by thermal expansion of the fuel at an acceptable level. This close control on clearance between the pellets and the sheath has placed a demanding requirement on the control of the diameter of the pellet. Similarly the fact that the sheath collapses onto the pellets in the reactor has made it necessary to specify that the surface of the pellets will be free of cracks and chips for the reason that, if a chip becomes lodged between a pellet and the sheath, additional strain will be applied to the sheath and it may fail. It has been found that, when the fuel pellets fragment in the reactor, this collapsible sheath holds the pieces together so that small chips cannot move to a position between the pellet and sheath.

Space has been provided for the thermal expansion of the UO_2 in each element by putting a dished end on each pellet and by leaving a space at the end of the stack of pellets. Both the dish and the space at the end of the stack of pellets are closely controlled to ensure that there is enough space for the thermal expansion of the UO_2 and, at the same time, to ensure that as much fuel as possible is placed in the element. As a matter of fact, as experience has been gained in the performance of CANDU fuel, customers for this fuel in Canada have been interested in getting as much fuel as possible into an element. To satisfy this desire, manufacturers have gradually increased the amount of fuel in an element by finding ways of controlling dimensions to closer tolerances than specified in the design and by achieving high densities in the pellets. Densities with values over 10.5 gms per cc are being achieved consistently. CANDU FUEL BUNDLE (Cont'd)

The design parameters for the Pickering fuel bundle are shown in Table 1.

TABLE 1. PICKERING FUEL BUNDLE DESIGN PARAMETERS (1)

Nominal bundle diameter	100 mm
Bundle length	500 mm
Weight of uranium	19.7 Kg
Number of elements	28
Element sheath O.D.	15.20 mm
Element sheath wall	0.42 mm
Fuel pellet O.D.	14.3 mm
Fuel pellet length	20.9 mm
Fuel - UO ₂	
Metal - Zircaloy - 4 only	

The six components that make up the Pickering fuel bundle are shown in Table 2.

TABLE 2. COMPONENTS OF PICKERING FUEL BUNDLE

Uranium dioxide pellet Zircaloy spacer Zircaloy bearing pad Zircaloy fuel sheath Zircaloy end cap Zircaloy end support

MANUFACTURING PROCESS FLOW CHARTS AND DESCRIPTION

Flow charts and a brief description of how the components of the fuel bundle are made and how the fuel is assembled are now presented.

UO2 PELLETS

Uranium dioxide powder is received from the supplier in lot quantities that are suitable for planned production rates. Prior to using a specific lot an evaluation sample is processed to pellet form and the pellets are examined for conformance to specification. For some lots the parameters of the processes may have to be altered to ensure that the pellets conform to the specification. If the pellets made from the evaluation sample meet the specification, the lot is accepted.

The UO₂ powder is compacted on a press at pressures of 550 kg/cm² to 850 kg/cm² into wafers or slugs. The "slugs" are broken up and passed through a fixed-size sieve.

The granular material is fed to a pill press (mechanical or hydraulic) where it is pressed, at pressures of 2800 kg/cm² to 5600 kg/cm², into "green" pellets to pre-determined weight, diameter and height. A statistical sampling plan is used to confirm that the process is producing "green" pellets that conform to the specification.

The "green" pellets are placed in containers or boats which are continuously stoked through a sintering furnace. The atmosphere in this furnace is hydrogen or cracked ammonia and the peak temperature is 1650° C to 1700° C.

Samples of the sintered pellets are checked for diameter, height, density and physical defects. Sample pellets are also checked for chemical analysis and a metallographic examination is carried out.

The sintered pellets are ground to the required diameter on a centerless grinder, are washed to remove the grinding coolant and sludge, are dried, and are inspected dimensionally and for surface imperfections.



Figure 2. UO₂ Pellets Flow Chart

SPACERS AND BEARING PADS

When the zircaloy strip for spacers and bearing pads is received it is inspected for thickness, width, flatness and surface defects and a mill certificate provided by the vendor is checked for compliance with the material specification.

The strip is cut into lengths which are suitable for loading into the beryllium coating machine and for being blanked into spacers and bearing pads. These strips are inspected visually and dimensionally.

The strips are degreased, pickled, washed, vapor blasted, rinsed, and dried.

The strips are loaded into the beryllium coater where a thin layer of beryllium is deposited on one side. The coating process is "vacuum deposition".

Spacers and bearing pads are blanked from these strips. Normally the bearing pads require some straightening.



Figure 3. Spacer and Bearing Pad Flow Chart

END CAPS

Zircaloy bar for end caps is received and is inspected visually, dimensionally and a mill certificate that is provided by the vendor is checked for compliance with the specification.

The bar is ultrasonically tested for defects, the most likely of which may be a longitudinal hole in the centre. Ultrasonic standards have been developed that enable such holes to be identified even if they are .025 mm in diameter.



Figure 4. End Cap Flow Chart

END CAPS (Cont'd)

The end caps are then machined, normally on an automatic screw machine.

A visual and dimensional inspection is then carried out after which the caps are degreased, pickled, rinsed and dried. The end caps are now ready for use at the Closure Welding.



Figure 4. End Cap Flow Chart (Con't)

END SUPPORTS

The strip for end supports is inspected visually and dimensionally and the mill certificate provided by the vendor is checked for compliance with the specification.

The end supports are blanked from the strip.

A dimensional and visual inspection is carried out.

Degreasing, pickling, and rinsing operations are carried out.



Figure 5. End Support Flow Chart

ELEMENT AND BUNDLE

The Zircaloy sheaths for the fuel are inspected on a statistical basis on receipt in the shop both visually and dimensionally including straightness and the mill certificate from the vendor is checked against the specification.

The sheaths are then ultrasonically tested for transverse and longitudinal defects, are sorted by inside diameter size range, and degreased.



Figure 6. Element and Bundle Flow Chart



Figure 6. Element and Bundle Flow Chart (Con't)

ELEMENT AND BUNDLE (Cont'd)

each element to the end supports.



and dimensionally and are packed. Also after assembly, as a control procedure, a

in Figure 1. A spot welder is used to weld

percentage of bundles are autoclaved to confirm that the surface of the bundle has not been contaminated with some foreign material.

In addition a percentage of bundles are monitored for radio-activity to confirm that the surface of the bundles are free of such contamination.

Figure 6. Element and Bundle Flow Chart (Con't)

COMMENTS ON PROCESSES

The areas of expertise that the Canadian nuclear fuel industry has developed in manufacturing CANDU fuel cover the complete range of manufacturing.

PELLETIZING

In the Pelletizing Operation the processes have been developed so as to produce defect free, high density pellets which meet tight tolerances on diameter, length. end squareness, land width, and dish depth. This high level of quality is attained with a minimum of scrap.

INCOMING INSPECTION

At the incoming inspection of sheathing, procedures have been developed for the segregation of sheathing into a minimum number of ranges of inside diameter. Consequently the matching of sheathing and pellets and the control of lots of material through the shop are relatively simple processes. The use of ultra-sonic equipment for locating very small defects in the bar stock has been so effective that less than ten elements have been rejected for leaking end caps during the manufacture of the first charge of fuel for Pickering Generating Station. All of the elements that were rejected met the specification. (There are more than 1 000 000 end caps in this first charge of fuel.)

TACKING AND BRAZING

The tacking and brazing operations are controlled so that the sheaths are not deformed and so that the metallurgical and physical properties are still satisfactory after the sheaths have been exposed to the high brazing temperatures. In addition the requirements for completeness of braze and maximum amount of free beryllium are met.

DIAMETRAL CLEARANCE

As mentioned early in the paper the diametral clearance between pellets and element sheath must be closely controlled and the stack length must also be closely controlled. These items require a system of control which ensures that the diameter of pellets which are loaded into the fuel sheaths is that which gives the designed diametral clearance and that the stack length leaves the required amount of space in the element.

CLOSURE WELDING

When a closure welder is being prepared for production, whether it is a new machine just being commissioned or an existing machine that has been maintained, the methods personnel make welds with the process until they are satisfied that the welds meet the design requirements. A statistical sample of welds is then made during which the machine parameters are varied the way they might vary during a normal production run. All of the welds in this sample are examined to determine if they meet the specification from the viewpoints of integrity and geometry. If they do meet the specification, the machine is turned over to the Shop for production. If they do not meet the specification, corrective work is continued on the machine.

Control parameters have been developed to indicate when the welding process is beginning to deteriorate so that equipment maintenance can be carried out before poor welds are made. To the best of our knowledge no elements with poor welds have been assembled into fuel bundles. In addition to weld integrity, the amount of upset in the weld must be controlled within close tolerances in order to ensure that the finished elements will be of a length that can be assembled into a bundle.

BUNDLE ASSEMBLY

Each bundle must be assembled so that it meets the specification with respect to squareness of the end supports, diameter, minimum spacing of elements, radial distance from the outer surface of the outer element to the end support, location of spacers and overall size. This last check is performed on a "GO-NO-GO" gauge which represents the smallest opening that a bundle has to pass through in entering and leaving a fuel channel in the reactor.

SAFETY AND HFALTH

CRITICALITY

There is no criticality hazard in a plant that is manufacturing natural uranium fuel because efficient neutron reflectors such as heavy water, graphite or beryllium are never present in sufficient quantities to create a problem.

HEALTH

Uranium

Natural uranium has a very low level of radiation. As a result the dosimetry levels established by the International Commission on Radiological Properties are not exceeded on a 40 hour per week exposure in a Canadian natural uranium fuel manufacturing plant. Never-the-less personnel in these Canadian fuel plants are classified as Atomic Energy Workers and wear film badge dosimeters which are monitored regularly.

Inhalation, ingestion and absorption are hazards which exist in the processing of natural uranium. Appropriate safety measures have to be taken in the different areas of the fuel manufacturing plants to ensure that these hazards are kept under control. For instance in the powder preparation area of the Pelletizing Shop the operator wears a respirator and gloves, and clothing which is laundered in the plant. In addition appropriate ventilation and dust collecting equipment are provided. In the same manner appropriate measures are taken in other parts of the Fuel Manufacturing facility to protect the health of the personnel. In addition persons who handle exposed uranium dioxide are checked by regular urine analysis and medical examinations.

Beryllium

Beryllium is more toxic than natural uranium and the precautions that are taken to protect the health of the persons who work with it include adequate ventilation, appropriate protective clothing, separate lunch rooms, gloves, laundry, personal cleanliness and appropriate housekeeping.

ACCOUNTABILITY

Accounting for uranium which contains a fissionable isotope is a requirement of the International Atomic Energy Agency. A fuel manufacturing plant which is processing only CANDU fuel has a relatively simple accountability job compared to fuel manufacturing plants where fuel with several enrichments is being processed. As a matter of fact a fuel manufacturing firm making CANDU fuel which has an inventory accounting and control system that is satisfactory from a cost point of view, will meet the IAEA requirements. The only extra work that the IAEA regulations involve is making it possible for the IAEA Inspector to witness the taking of inventory and to provide the total input, work-in-process, and output information to him in a way that is meaningful for his work.

APPLICATION OF MANUFACTURING TECHNIQUES WITH RESPECT TO USE OF LABOUR

As has been shown the fabrication of a CANDU fuel bundle is carried out with simple manufacturing processes and techniques. These lend themselves either to the intensive use of labour or to the use of semi-automatic equipment.

In the pelletizing process either relatively inexpensive toggle presses with single cavity molds or more expensive hydraulic presses with multi-cavity molds and many added features may be used.

Several low-capacity muffle-type sintering furnaces or one large capacity sintering furnace (many types are available) can be used.

At pellet grinding and pellet inspection there is a wide range of choices between the use of a high number of operators and simple equipment to do the work and the use of semi-automatic equipment with a proportionately lower number of operators.

In the preparation of spacers and bearing pads if the flow charts shown in Figure 3 are followed, the use of persons can be intensive. On the other hand one or two of the processes can be changed so that they will be at least semi-automated with the resultant removal of labour.

These examples for pelletizing and for spacers and bearing pads are indicative of the range of choices that are available throughout the total CANDU fuel manufacture for using labour intensively or for using semi-automatic and automatic equipment. The choice in many cases can be made in the light of economic circumstances.

USE OF OTHER MANUFACTURING FACILITIES

There are many operations that lend themselves to the use of other manufacturing facilities or the sharing of equipment from other manufacturing facilities by the fuel manufacturing facility. Examples are:

- (a) Pellet presses, sintering furnaces and grinders used in tungsten carbide work can be modified for use in UO₂ pelletizing.
- (b) Simple punch presses that are used in many processes can be used for blanking the spacers, bearing pads and end supports.
- (c) An automatic screw machine in some other operation can be used to machine the end caps from the bar stock.

The criteria that must be met if other facilities are used are those of appropriate quality control and assurance that there will not be contamination from such materials as halogens, hydrocarbons, hydrogen, boron or cadmium. In addition, if a fuel manufacturing facility is located near a metal working factory which has chemical and metallurgical laboratories, the chemical and metallurgical work that has to be done for fuel manufacturing can probably be performed with the equipment that exists in that plant or with the addition of one or two pieces of equipment.

STAFF REQUIREMENTS FOR A FUEL MANUFACTURING FACILITY

The replacement natural uranium fuel that would be required on an annual basis for a Nuclear Generating Station consisting of two 540 MW Reactors similar to those at Pickering, would be approximately 125 Mg U at approximately 90% availability and between 75% and 80% utilization. Based on Canadian experience - and the Canadian degree of semi-automation, the staffing for such a plant might be the following:

1. PLANT MANAGER

2. DESIGN ENGINEERING

This function would provide interpretation of the design parameters, metallurgical engineering, skills in stress analysis, heat transfer, and nuclear physics. Personnel in this function would probably also provide primary liaison with the Licensor and the customer(s). The staffing would probably be two engineers with Bachelor or Master Degrees.

3. MANUFACTURING ENGINEERING AND QUALITY CONTROL

3.1 Manufacturing Engineering

This function would look after the shop equipment with respect to specifying parameters and providing equipment for processes such as pressing pellets, sintering pellets, beryllium coating, brazing, closure welding, pickling, bundle assembly, providing day-to-day tool proving and methods work, day-to-day shop problem solving, productivity improvement work, and issuing manufacturing instructions.

In addition it could administer the plant engineering work, laundry, janitoring, waste disposal, maintenance and similar work. Staffing would probably be two engineers, four or five methods men and five or six maintenance persons and janitorial persons. The engineers and methods men would need a wide range of knowledge in metallurgical, mechanical and electrical technologies.

3.2 Quality Control

This function would be responsible for designing the Quality Control System and issuing the Quality Control Instructions to the appropriate areas. In addition it would evaluate the incoming inspection information and the in-process inspection information to determine whether items are out of specification or are tending to go out of specification.

The skills that are required for specifying and maintaining the non-destructive testing equipment and measuring equipment can be made available in the personnel in Quality Control or in Manufacturing Engineering. It is a matter of preference and/or specific capability. If we assume it is made available in the Quality Control area, the staffing of the total Quality Control function will probably be one quality control engineer, with experience in non-destructive testing techniques, two technologists or engineering assistants, and two clerks.

3.3 Laboratory

The laboratory would do metallurgical examinations of pellets, brazed joints, welded joints and similar items, and chemical analysis of beryllium contamination in the beryllium work areas, of uranium contamination in the uranium work areas, of the pellets, of the pickling solutions, for moistuin the completed elements and such other work as may be assigned. Three to four persons could handle this work. This is one of the services that can be purchased from a metal working plant, if there is one located near the fuel plan

4. MATERIALS UNIT

This function would provide the following activities - purchasing, scheduling, ordering, inventory control, dispatching, storing, receiving and shipping. Between 7 and 10 persons would be required to handle this work.

5. SHOPS

This function would do the actual manufacturing of the pellets and bundles. The pellet shop would operate three-shifts for five days and the balance of the shop would operate two-shifts for five days. Three to four foremen, four to five area technicians and dispatchers and 70 to 80 direct employees including inspection personnel would be required in this shop for the first 12 to 18 months. After this period a learning curve would apply. The amoun of learning and the cost improvements that can be achieved are directly proportional to the desire of the personnel in the manufacturing organization to set targets and achieve them.

6. OTHER SERVICES

This function would provide such services as employee relations, payroll, cost accounting, other accounting, plant hospital, safety work and clerical pool. Probably 12 persons would be able to do the required work.

SIZE OF BUILDING FOR FUEL MANUFACTURING FACILITY

A building 80 metres long and 30 metres wide would provide the space for the staff, equipment and auxiliary areas needed for manufacturing 125 Mg U of fuel per year. An approximate breakdown for the use of this space is shown in Table 3.

TABLE 3.SPACE REQUIRED FOR MANUFACTURING125 Mg U CANDU FUEL PER YEAR

Office - Plant Manager, Engineering, Manufacturing Engineering Quality Control Materials		
and other personnel.	500	m ²
Pelletizing Shop	450	m ²
UO ₂ Stores	50	m ²
Fabrication Shop	750	m ²
Laundry, Washrooms and Lunch room	150	m ²
Other Stores	100	m ²
Maintenance	100	m ²
Chemical and Metallurgical Laboratories	150	m ²
Receiving and Shipping	150	2
Total Area	2400	m ²

The services required for this plant are the normal services that are required for a light manufacturing plant with the exception of the large power supply for the sintering furnaces, the high intermittent power demands for the resistance welder, and the stringent requirements for ventilation and filtering of exhaust air from the UO_2 and beryllium areas.

PACKAGING CANDU FUEL FOR SHIPPING

Because the CANDU fuel bundle is small, light, and easy to handle, a method of packaging it for shipping has been developed which performs the functions that are required of a shipping package and which is inexpensive. A sketch of such a package is shown in Figure 7. For domestic shipments the outer covering of the package is cardboard and for overseas shipments the outer covering of the package is a wooden box. This package has a capacity of 36 fuel bundles of the reference design and its total weight is less than 1000 kg.



40"×45"36-BUNDLE PALLET

Figure 7 CANDU Fuel Shipping Package

COST OF CANDU FUEL

The engineers who did the early work on designing and developing CANDU Fuel conceived a fuel bundle which was inexpensive to manufacture. In his paper "Estimating CANDU Fuel Costs", A.W.L. Segel shows the distribution of fuel manufacturing costs for CANDU fuel of the reference design at volumes of one Mg U per day. These costs are representative of what a customer pays for this fuel at these volumes. When this fuel is used in a 540 MW Pickering Reactor and an average burn-up of 215 MW hrs/kg is achieved, the fuelling cost is 0.8 mils per kW hr.

SUMMARY

A description of manufacturing CANDU fuel has been presented which includes a verbal description of the steps followed in making the fuel together with the applicable flow charts, comments on the process used, particularly with respect to quality control, an outline of the health precautions that are necessary, and an indication of the flexibility that is available for using labor intensively or for using semi-automatic equipment as well as for sharing other manufacturing facilities. In addition the staffing and space requirements for a plant capable of manufacturing 125 Mg U CANDU fuel per year are described. Finally the manufacturing of CANDU fuel results in costs which are represented by those discussed in recent Canadian technical papers. (1)(2)

References

- (1) A.W.L. Segel, "Estimating CANDU Fuel Costs", AECL-4273, August 1972.
- (2) L.R. Haywood et al, "Fuel for Canadian Power Reactors", AECL-3979, September 1971.

by

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ABSTRACT

The HTR is characterised by the use of graphite as moderator and fuel element structure, helium as coolant and ceramic coated fuel microspheres (coated particles) for fission product containment. The flexibility this gives, in the choice of fuel cycle, is high-lighted and the consequent range of fuel element designs, varying from the pebble-bed to the several types of graphite block structures, is described.

Following an outline of the philosophy underlying the design of the coated particles, the processes required in manufacturing HTR fuel elements are described in sections on graphite manufacture and machining, fuel particle manufacturing processes, fuel body fabrication, assembly, reprocessing, refabrication and quality control. Existing HTR fuel element manufacturing plants are indicated.

The main features in HTR fuel element performance are outlined with particular emphasis on the parameters having most impact on the fabrication processes.

It is concluded that the most neglected area of technology is refabrication and that the implications on the fuel element design and on the choice of fuel cycle should not be overlooked.

1. THE HTR SYSTEM

The High Temperature Gas-Cooled Reactor (HTR) is characterised by the use of graphite as the moderator and Juel structure, helium as coolant and pyrolytic carbon and silicon carbide for fission product retention. These refractory materials and the inert coolant are used to lift the constraints on burn-up, rating and temperature imposed by metal cladding, to minimise thereby both the parasitic absorption of neutrons by the core structure as well as interaction of the coolant with the core and primary circuit materials.

In consequence the HTR is potentially the most efficient way of burning fissile material and the fuel cycles range from the possibility of thermal breeding in a U-235/Th-232/U-233 thorium cycle to a U-235/U-238/Pu-239 low enriched cycle whose high burn-up eliminates the need for plutonium recovery $[1-12]_{\circ}$

The homogeneity of the heat production in the core leads to the ability to achieve high power densities and the lack of constraints allows wide variation in the disposition of fuel, coolant channels, moderator and control rods. The numerous possible core designs include:

- (a) hollow rod, as used in Peach Bottom [13, 14] and the Dragon Reactor Experiment [15],
- (b) the pin in block concept, with tubular [16, 17] or teledial pins [18] or directly cooled fuel,

- (c) the pebble-bed as used in the AVR [19, 20] and to be used in the 300 MW(e) THTR [20, 21, 22].
- (d) the multi-hole graphite block as in an early De Havilland concept [23], in Fort St. Vrain [24, 25] and in the large Gulf General Atomic (GGA) HTRs [26],
- (e) the integral block fuel element proposed by Nukem [27].

Hexagonal, pentagonal and triagonal graphite moderator blocks have all been considered for particular designs. Methods of optimisation of the thermal design of the fuel cell geometry have been reported [28].

Present design assessments and economics have led to core power densities of $\sim 8 \text{ MW/m}^3$, to the use of the high enriched U-235/Th-232/U-233 or the low enriched U-235/U-238/Pu-239 fuel cycles and to three major variants of core design, the pin-in-block, the pebble-bed and the multi-hole graphite block with the common feature of a replaceable core. Each solution represents a summation of relatively small differences in the margins set by design constraints, safety considerations and economics.

In the pebble-bed system, fuelled graphite spheres are randomly packed in a cylindrical enclosure or reflector also of graphite. The individual fuel element spheres, ~6 cm diameter, are recycled by removal from the base of the core and re-introduced on top of the packed bed. In contrast, the prismatic core [17, 29] is made up of stacked graphite blocks, usually hexagonal, ~1 m long and ~50 cm wide containing holes drilled lengthwise for coolant and fuel. Charge and discharge are achieved by a skewer passed down a handling hole in the block. Two variants, which may be termed 'Hot' or 'Cold' block designs, have been developed. In the latter the fuel holes are large, say 75 mm diameter and consequently relatively few in number (~18). These holes contain graphite tubes or pins (~50 cm long) and coolant flows over the fuelled pins which are centrally located in the holes. This configuration leads to a block temperature close to that of the helium coolant. The coolant and fuel holes are much smaller (~1 cm) and larger in number (200-300) in the alternative 'Hot Block' design. The heat generated in the fuel is transported across the webs to the coolant holes leading to a more complex thermal stress pattern and a generally higher block temperature.

The multi-hole hot block design has been used in Fort St. Vrain [29] and is standard for the large GGA reactors while the pebble-bed concept will be used in the THTR reactor under construction at Schmehausen. The pin-in-block or cold block design [29] is that which was offered for the 750 MW(e) low-enriched uranium cycle Oldbury reactor [30, 31].

With the inherent flexibility of the HTR, the fuel particles can be varied to suit the particular design, fuel cycle, performance and economic situation. Thus changes in the chemical composition of the fuel kernel, in the size of the particles and in the thicknesses and types of coating are easily made and do not represent fundamental changes in the concept.

Two extreme cases illustrating the variations adopted in commercial prismatic reactor designs may be cited. In the large reactors, sold by GGA in the USA, the coated particles are desgined within the frame of thorium utilisation and are essentially of two types; fissile and fertile [32, 33]. The fissile particles contain pure, highly enriched uranium dicarbide or dioxide kernels (~200 μ m diam.) surrounded by a porous buffer coating of pyrolytic carbon. The thickness of this coating is designed to provide free space for the accommodation of fission products. Outside this coating is a composite pressure vessel (about 200 μ m thick) of pyrolytic carbon with a pyrolytic silicon carbide interlayer. This coating contains the heavy burden of fission products associated with the 80% burn-up of the uranium.

A larger fertile particle, used to breed U-233 from thorium, has a kernel about 400 μ m diameter and is composed of thorium oxide or carbide coated with pyrolytic carbon. To operate the fuel cycle, this fuel is discharged after about 7% heavy metal burn-up and treated to extract the bred U-233. The two types of particle are separated in the reprocessing to prevent significant cross-contamination of the U-233 with U-236 from the fissile particle.

In contrast, only a single particle was foreseen in the low-enriched uranium cycle offered for the Oldbury HTR [34] Here the fuel kernel was of UO₂ with a

uranium enrichment of $\sim 5\%$. A marginally different philosophy, stemming from the Dragon Project, is adopted to accommodate fission products in that $\sim 20\%$ free space is built into the kernel as porosity. The buffer coating surrounding the particle is primarily to absorb fission fragment damage and is surrounded by the structural pyrolytic carbon/silicon carbide/pyrolytic carbon coating. Here again the coating thickness is chosen to withstand fission gas pressure and is 200 µm for an 800 µm diameter kernel undergoing a peak fuel lifetime heavy atom burn-up of about 10%. The much larger particle employed in this system is advantageous in the fuel compact which consequently has about 35% volume packing of particles. By contrast, a packing fraction up to about 60% follows from the GGA design. In the latter case, which represents close random packing, pressing of the fuel into compacts is precluded. At the lower volume packing of 35%, however, processes have been developed [35] to allow a near perfect distribution of fuel particles in a high density graphite matrix whose thermal conductivity is exploited in the fuel element design.

A fuel particle intermediate between the fissile and fertile types is used in the 300 MW(e) THTR [21], such that the thorium and uranium are contained in the same particle. This is similar to the quasi-fissile particle used in the Fort St. Vrain reactor where a Th:U ratio of 4.25:1 is used to produce a fuel for a heavy atom burn-up of 20%. The same principles of particle design are used, i.e., provision of internal free space for accommodation of fission products; buffer coating to resist fission fragment damage; a structural and diffusion barrier for fission products. In the pebble-bed system, where separate feed and breed elements can also be foreseen [21] the fuel particles are contained in a sphere made from graphite (as in the initial charge of the AVR [19]) or moulded within a pressed graphite body.

This variety of approaches illustrated in Fig. 1 underlines the range of choice now available to the reactor designer to exploit particular local circumstances to the maximum. All of the above types of fuel are commercial manufacturing realities.

Sales of commercial HTRs began in August 1971 when Gulf General Atomic sold two 1,100 MW(e) stations to Philadelphia Electric. Up to the time of writing a total, of 5,400 MW(e) has been sold [36, 37, 38] and further stations are on option (see Table 1).

These massive sales clearly demonstrate that the HTR is a commercial proposition, provided it is backed by a strong enough company. Furthermore the future capabilities of the HTR for tasks other than steam-raising connot be ignored. These include:

- direct-cycle gas turbine (net station efficiency pessimistically 44.4% [39] but capable of stretch beyond 50%),
- process heat for chemical and metallurgical processes [40, 41],
- rocket propulsion [42].

Table 1 Commercial HTR Orders (up to 23rd June, 1972)						
Item	Utility	Planned Output MW(e)	Date of Order	Date for Commercial Operation		
1	Philadelphia Electric	1,160	August, 1971	1979		
2	Philadelphia Electric	1,160	August, 1971	1981		
3	Delmarva Power & Light Co.	770	December, 1971	1979		
4	Delmarva Power & Light Co.	770	December, 1971	1982		
5	Southern California Edison	770	May, 1972	1981		
6	Southern California Edison	770 5,400	May, 1972	Not yet decided		

Permutations of fuel cycle, core design and the method of heat disposal lead to an extensive range of possible HTRs. The presence of several running alternatives is not necessarily bad, since competition within the species will strengthen the concept. One must therefore foresee continuous development over the next quarter century such that, at any one time, some particular type of HTR will be under development whilst at the same time some other type is being sold commercially.

2. THE DESIGN OF THE COATED PARTICLE

We have seen that HTR core design is extremely flexible and that, although several different types of fuel element have evolved, it is common ground that the fuel is incorporated in the form of coated microspheres. The general philosophy of coated fuel particle design has been discussed by Huddle [43] and Graham [44].

The function of the coated particle is to generate heat and to retain fission products whilst achieving the target fuel burn-up. Coated particles (Fig. 2) consist of a fuel <u>kernel</u> surrounded by a <u>coating</u>. The fissile and/or fertile material is incorporated into the kernel generally in oxide or carbide form. Because the kernel is the substrate on which the coating layers are deposited, it is desirable that the fuel kernel should be spherical.

As pointed out in the previous section, space is provided to accommodate fission products and kernel swelling either by incorporating the majority of the porosity in the kernel or in the first layer of the coating.

The coatings must, in combination, constitute a highly efficient barrier to the release of fission products, both from the viewpoint of diffusion and mechanical integrity in service. They may be composed of pyrolytic carbon alone or with an interlayer of silicon carbide. In the latter case the functions of the coatings are as follows:

(a) Pyrolytic Carbon Buffer Layer

The buffer layer is an absorber of fission recoils and together with the porosity of the kernel gives free volume sufficient to accommodate fission products and kernel swelling. It also gives a mechanical separation between the kernel and the structural layers of the coating.

(b) Inner Pyrocarbon Coating

This is the first of the three coatings which make up the structural layers. It is required to have specific properties to minimise the effects of fast neutron irradiation which lead to dimensional changes and consequent stressing. This layer also serves as a barrier to gross diffusion of kernel material (fissile, fertile or fission products) which may affect the integrity of the silicon carbide. With uranium dioxide kernels an additional manufacturing layer of pyrolytic carbon may be required outside the buffer to seal off the kernel and avoid conversion to carbide if subsequent coating operations are to be carried out at high temperature.

(c) Silicon Carbide Interlayer

The interlayer is the main diffusion barrier to metallic fission products. It also plays an important role in the mechanical performance of the particle [45] and because of its stability under fast neutron irradiation [46, 47] imparts dimensional stability to the whole particle.

(d) Outer Pyrolytic Carbon Coating

This coating completes the particle and is the third of the structural layers. It aids in the mechanical performance of the particle by putting the SiC layer into compression at an early stage of operation and also provides chemical protection for the silicon carbide.

3. HTR FUEL ELEMENT FABRICATION PROCESSES

3.1 Graphite Manufacture

The manufacture of nuclear graphite for use in HTRs is essentially conventional in that coke powder is bonded with pitch, formed to shape by extrusion or pressing and then baked and graphitised. Additional impregnation treatments may be necessary to achieve the required level of apparent density. The only major difference in processing is that, either a nuclear purity coke must be used as raw material and processed under clean conditions or, more usually, because of the paucity of pure coke sources, a less pure coke source is used and neutron absorbing impurities are removed as volatile halides by a purification process during the graphitisation stage [48, 49].

The quality of graphite to be used in HTRs may be specified using the following tests, carried out at room temperature:

- 1. Dimensions
- 2. Thermal neutron absorption cross section
- 3. Electrical Resistivity

4. Apparent Density

on whole blocks

- 5. Thermal Expansion Coefficient, parallel (11) and perpendicular (1) to the axis of pressing or extrusion
- 6. Ultimate Tensile Strength and Modulus of Elasticity, 11 and L
- 7. Chemical Reactivity

The obvious importance attached to irradiation behaviour and quality when using graphite in a structural role in HTRs normally implies a period of irradiation testing to select the most suitable graphite and some long term guarantee of quality. This latter requirement is primarily related to the stability of the coke source. Since the coke is often a by-product it must be expected that the quality of the coke may change over a period of, say, 10 years. Fortunately the peak fast neutron dose levels in HTRs are not high and can be achieved in modern Materials Testing Reactors in about 1 year.

3.2 Graphite Machining

The machining of graphite for HTRs is accomplished by conventional machine-shop techniques. The only differences from ordinary commercial practice are that care must be taken to avoid adventitious contamination [50] with neutron absorbers or oil and that close tolerances are often specified [51].

3.3 Fuel Fabrication Processes (see Fig. 3)

3.3.1 Kernel Fabrication

The kernel is an intermediate product whose chief requirement is to be a suitable feed to the next process - coating. Near-spherical kernels can be made by agglomeration of powders or by utilising the surface tension of a liquid to form droplets.

The powder agglomeration process involves the growth of kernels from fine powder mixed with a binder [52-54] followed by heat treatment. The alternative kernel fabrication methods have the attraction of utilising liquids to form spherical droplets. Two general categories of surface tension method that have been studied are:

- (i) spraying of liquid droplets into a second phase (gaseous or liquid) chosen to solidify and retain the spherical shape produced [21, 55, 56, 57, 58, 59, 60]. Thus a sol or solution containing the appropriate fuel can be injected into a second liquid under conditions which yield discrete droplets which gel by chemical reaction. Hence the terms 'Sol-Gel' or 'Gel-Precipitation' used to describe these processes*. The 'green' spheres so formed are heat treated to give them adequate strength,
- (ii) formation of fuel agglomerates or 'crumbs', each of which is the volume of the desired kernel, followed by fusion [61]. Again a second phase is necessary to avoid coalescence of the liquid droplets.

Kernels of given properties can be made using any of these routes, although the Sol-Gel and Gel-Precipitation processes are sometimes considered more amenable to control by automatic methods. On the other hand, in some cases, porosity can be incorporated more easily using the powder agglomeration route.

3.3.2 Coating

The original suggestion for coating nuclear fuel particles with pyrolytic carbon (PyC) was made by Huddle [62, 63] and some preliminary experiments were carried out at AERE, Harwell where the coating was applied in a rotary tumbling bed by pyrolysis of a hydrocarbon. The kernels must be agitated while the deposition is occurring otherwise agglomeration and inhomogeneous deposition may occur. Of possible methods of achieving the desired degree of agitation, only fluidisation and tumbling have been studied in any detail. The tumbling bed avoids the use of fluidising gas but has technical and economic limitations and virtually all coating of fuel kernels has been carried out as a batch process in high temperature fluidised beds.

Because it is the most "glamorous" of the unit operations in HTR fuel manufacture, it has been the subject of extensive study [64-68]. The fluidising reactor is normally constructed of graphite which may be resistance or induction heated to temperatures in the range 1200- $2000^{\circ}C$. For the deposition of pyrolytic carbon a hydrocarbon, such

^{*}Suggested definitions of the terms 'Sol-Gel' and 'Gel-Precipitation were given by Hardy at an IAEA Panel held in 1968 [57].

as methane, mixed with an inert carrier, is introduced into the reaction chamber using a single nozzle, multi-nozzle, or porous plate gas entry system. Other, higher, hydrocarbons, such as butane or propylene [66, 69, 70, 71], may be used to impart desired properties to the pyrolytic carbon. To lay down a coating of silicon carbide (SiC) [72-75] the particles are fluidised in hydrogen or inert gas/hydrogen mixture, containing a low concentration of methyltrichlorosilane which is generally preferred to other sources. Layers may be deposited in sequence, such as PyC/SiC/PyC in the same fluidising unit: alternatively the large volume increase (typically a factor of 3 for the Dragon Reference Particle) and density decrease which occurs during coating can lead operators to split the coating process between more than one furnace. Deposition rates are ~1 μ m per minute.

3.3.3 Consolidation

The coated particles are either dispersed in a graphite matrix or bonded together to define their position in the reactor and incorporate them into a thermally conducting, mechanical unit able to withstand the temperatures as well as the thermal and irradiation induced stresses arising in service.

In the first of these processes [35, 76] the coated particles are overcoated with a graphite matrix powder, pretreated with thermosetting resin, and the desired shape is formed by warm pressing, followed by heat treatment and degassing. Apart from complying with the requirements listed above this process leads to a fuel body which is relatively isotropic and the overcoating of individual particles provides a mechanical and chemical interparticle barrier which isolates the effect of an individual broken particle.

The alternative bonding processes for consolidating fuel bodies are particularly applicable when the volume fraction occupied by the particles is high. In one case an injection moulding process [76] is used in which close-packed beds of particles are bonded with a viscous mixture of pitch and graphite powder which is injected into the packed bed prior to carbonisation and degassing [32]. An alternative technique is to bond and densify the bed of coated particles with pyrolytic carbon in situ, by gaseous impregnation with natural gas at <1000°C [78-80].

3.3.4 Fuel Element Assembly

In the case of HTR fuel, the assembly of the fuel element is an extremely simple procedure. The fuel body is a mechanical unit which can be tested and approved as such. The graphite components are supplied as fully machined and inspected items. It only remains to marry the components together.

Except in cases where the fuel cycle involves refabrication (see below), assembly can therefore be carried out on an open bench, the main precaution being to prevent accidental contamination of the graphite with neutron absorbers or oxidation catalysts.

3.3.5 Reprocessing and Refabrication

The spent fuel can merely be stored but to obtain low fuel cycle costs, it is necessary to reprocess the fuel to extract the fissile component (U-233 or Pu). In some cases, especially those using a fully enriched uranium feed, the fuel cycle economy leads to a requirement to refabricate this fuel for further use in HTRs.

Reprocessing consists of the following steps:

- (i) head end processing,
- (ii) dissolution and solvent extraction to obtain selective recovery of the desired stream,
- (iii) waste disposal.

Retabrication comprises the following operations:

- (i) kernel preparation,
- (ii) coating,
- (iii) consolidation,
- (iv) assembly,
 - (v) scrap recovery,
- (vi) waste disposal.

Reprocessing

An important characteristic of the HTR fuel element, bearing on reprocessing, is the incorporation of the carbon moderator in the fuel element. The carbon: heavy metal atom ratio is typically 200-350. For some fuel element designs it may not be possible to guarantee separation of the spent fuel from the majority of the carbon and therefore the present schemes for reprocessing envisage treating the complete spent elements. Typically the elements are subjected first to stage crushing followed by burning and leaching.

Low enriched fuel will contain uranium and plutonium isotopes which are easily separable to recover the plutonium for refabrication or sale.

In the case of the thorium ycle, it is neces any to achieve selective recovery of the bred U-233 without significant crosscontamination from the partially burned-up U-235. Imperfect separation of the fissile and fertile particles is not crippling. Thus Stewart et al., [32] quote a penalty of only 0.01 mills/kWh for a 4% loss of U-233 to the spent U-235 stream or a 20% back mixing of U-235 and U-236 into the bred fuel stream. Complete mixing of the two streams would lead to an undesirable build-up of U-236 in multiple recycle.

There are advantages in achieving the U-233/U-235 separation by having different types of particle within a fuel element. Such particles can be designed to differ in several respects any one of which can be employed for separation:

 (i) the fertile particles could have all pyrolytic carbon coatings. Then the fissile particles should have a silicon carbide interlayer. In this way the U-233 can be dissolved in the leach solution whilst the fissile particles remain unattacked,

- (ii) the particles could be significantly different in size. This will normally mean that the fertile particle is the larger in diameter.
- (iii) the particles could be significantly different in density. The use of carbon dilution in the fissile particle may be advantageous in increasing high temperature operating capability while at the same time reducing the overall particle density.

Although one may commence by assuming that the importance of reprocessing is to recover the U-233, the recycle of the discharged U-235 is well worth investigation, having been estimated as equivalent to 0.05-0.06 mlll/kWh [33].

The overall pattern of recycle is as illustrated in Fig. 4.

Refabrication

Apart from the fact that the operations are remote or semiremote the refabrication processes presently envisaged parallel those of direct fabrication. It is not evident that this trend will continue but it would ensure that the refabrication plant could be worked up economically.

3.3.6 Quality Control

One of the factors which has contributed to the advance of coated particle fuel, from being an interesting idea to the production of power reactor fuel on the tonne scale, has been the parallel development of suitable quality control and evaluation techniques. A great advantage with HTR fuel is its subdivision into a very large population of microspheres. Even with relatively large kernels of UO, 800 μ m diameter, 80% dense there are ~4.8 x 10⁸ particles per tonne U. Consequently classical statistical methods can be applied.

The release from coated particle fuel at the start-of-life derives from three sources:

- (a) heavy metal contamination in the outer pyrolytic carbon coating,
- (b) the proportion of particles without a silicon carbide interlayer or with a defected silicon carbide layer (characterised by the equivalent defective fraction O'),
- (c) the proportion of particles with all coatings broken (characterised by the equivalent broken fraction \emptyset).

In normal operation a minute but predictable release will occur: this is in marked contrast with the customary situation with other reactor types where an essentially zero fission product release is unpredictably interrupted by large releases from failed cans. To guarantee that the above source terms do not change in service it is necessary to control:

- (i) the amount of free space to accommodate fission products both in the kernel and the buffer layer,
- (ii) the properties of the pyrocarbon layers,
- (iii) the properties of the silicon carbide layer.

The operational function of Quality Control must therefore be to control and define the source terms at the commencement of irradiation and to control those parameters which affect particle endurance. The following parameters are used to estimate these two requirements:

Coated Particles

- kernel diameter*
- kernel porosity*
- O/U atom ratio in the kernel*
- fissile isotope abundance*
- thickness of the individual coating layers
- density of the buffer and structural coating layers
- isotropy of the individual pyrocarbon coatings
- contamination of the outer pyrocarbon layer
- ceramographic examination to check coating integrity, presence of flaws, diffusion of kernel etc.

(*May more conveniently be carried out at an earlier stage but the results are strictly applicable to the coated particle).

Fuel Bodies

- dimensions
- heavy metal loading
- heavy metal dispersion
- $\begin{bmatrix} & \emptyset \\ & \emptyset \end{bmatrix}$ as defined above
- matrix density

The majority of the techniques for testing of coated particle fuel have been surveyed at a recent IAEA panel [81] whilst the analytical methods were described in a parallel IAEA symposium [82]. Reference [81] was a co-ordinated paper which included information on test methods developed by AERE Harwell, UK, Belgonucleaire SA, Belgium, CEGB, Berkeley Nuclear Laboratories, UK, OSGAE Seibersdorf, Austria as well as by the Dragon Project. Within the regional grouping of signatories to the Dragon Project, a Quality Control Working Party exists which provides a forum for discussion of the philosophy of quality management, statistical techniques and detailed test methods as applied to HTR fuel.

4. HTR FUEL ELEMENT MANUFACTURING FACILITIES

4.1 Fuel

In this Section an attempt is made to indicate the known HTR fuel element manufacturing facilities. This is an area which impinges on commercial practice and our information is likely to be incomplete. In one case we are using information which has been specially made available for this paper.

United States of America

In 1959 a group of 58 US Utilities called High Temperature Reactor Development Associates, the USAEC and General Atomic (now Gulf General Atomic) entered into a commitment to build the 40 MW(e) Peach Bottom HTR. In parallel, development work continued at General Atomic and Oak Ridge National Laboratory with support from the USAEC and utilities. This led to the construction of the 330 MW(e) Fort St. Vrain HTR [26].

It is evident therefore that Gulf General Atomic, the supplier of the fuel for Peach Bottom, Fort St. Vrain (and UHTREX), has considerable manufacturing capability and experience on thorium cycle fuel.

Coated particle fuel with and without a silicon carbide interlayer has been investigated (termed TRISO and BISO respectively) [77]. The tendency has been to use relatively small coated particles and the consequential close packed beds of coated particles have been bonded using an injection moulding process (see Section 3.3.3) [32, 77].

The USAEC is sponsoring programmes at Gulf General Atomic and Oak Ridge National Laboratory on U-233 refabrication and recycle. The form of the development programme has been detailed [83]. In ORNL-4702 the total USAEC budget is given as 56.8 \times 10⁶ US dollars over a ten year period notionally starting in 1970. In practice it appears that this programme is running a little later than that time scale. Although work is proceeding on all phases of recycle development [32], at present the main emphasis (involving over 75% of the expenditure during 1972 and 1973) is on Head End Development (by Gulf General Atomic) and Refabrication Process Development (by Oak Ridge National Laboratory).

Other firms within the USA have worked on coated particle fuel. In particular mention should be made of the fabrication of the first charge for the AVR by Union Carbide Corporation [19] at their Lawrenceburg plant.

Belgium

The Belgian Nuclear Research Centre (CEN/SCK Mol) and Belgonucléaire have co-operated on coated particle fuel for many years. The powder agglomeration process developed by Gorlé and Huet [52] led to close collaboration with the Dragon Project. The programme first centred on the thorium cycle and the utilisation of plutonium [10, 84, 85, 86] but since 1966 has been modified towards the low enrichment cycle [87]. Within and between batch variability have been studied in detail and results [87] are given in Table 2.

The coating units presently available have a capacity of 1.8 tonnes U/y [87].

France

The HTR programme in France began somewhat later than in other countries although, as a member country of the Dragon Agreement, there has always been access to the extensive programme undertaken by the Dragon Project. The French approach as elaborated by Tanguy et al., [80] is to be extremely flexible. Work on kernels is being conducted by SFEC (Société de Fabrication d'Element Catalytiques) using two powder agglomeration processes [80]. Coating has been developed by CERCA (Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques) and it has a production capacity at the present time of 2.5 tonnes coated particles/y [88].

Special attention has been paid to the consolidation of the coated particles into fuel bodies and preparative work has been carried out by Société le Carbone Lorraine with the powders being evaluated by the CEA [80]. A significantly different approach to consolidation using a gas-cracking process has been made [79, 80] (termed the BB5 process) and compacts made in this way are being irradiated in OSIRIS as well as the Dragon Reactor.

The industrial work is supported by extensive irradiation facilities as well as by research and development carried out by the CEA at Saclay and Grenoble.

United Kingdom [30, 31, 34, 54, 57, 72]

The following information has been supplied by Mr. J. Stephenson, UKAEA Risley:

"A manufacturing plant has been developed, constructed and operated at UKAEA Springfields to make fuel for zero energy reactors and large scale irradiation trials, some 80,000 fuel compacts having been made. Its other major use has been for confirmation of plant throughput and reliability and statistical assessment of typical production fuel.

The plant comprised the fabrication of: kernels by the powder agglomeration route, coated particles using large fluidised beds, graphite overcoated particles and finally, the consolidation of these into compacts or rods.

	Table 2 Statistical Data on Coated Particles (from [87])						
Characteristics		Nominal Unit Design Valuc	Nominal Design Valuc	Statistical Results			
					Standard Deviation		
			Average	Within a Batc	Batch to Batch		
1.	UO Kernel						
	Density		% Theoretical Density	82-93	90	4	2
	Diameter		j tm	780	780	19	60
	0 / U		-	2.000	2.003	-	0+005
	Max./Min. Diameter		-	≰ 1.1	1.03	0.024	0.020
2.	Inner PyC						
1	2.1 Porous	Density	g/cm ³	1.05	1.0	-	0.05
		Thickness	μm	33	30	-	3
[2.2 Transition	Density	g/cm ³	1.5	1.6	-	0,05
		Thickness	11m	30	30	-	5
1	2.3 Inner HD	Density	g/cm ³	1.8	1.8	-	0,05
		BAF	-	1.08	1.04	-	0.04
[Thickness	<u>7710</u>	27	29		5
	2.4 Total	Thickness	μm	90	90	7	13
3.	SIC						
		Density	g/cm ³	3.2	3.2	-	0.02
		Thickness	μm	35	35	2.4	3.4
4.	Outer PyC		-				
		Density	g/cm ³	1.8	1.9	-	0.05
		BAF	-	1.08	1.04	-	0_04
		Thickness	arth .	30	30	2.7	5
5.	Total Coating	Thickness	p t Ⅲ	160	160	8	10
6.	Coated Particle						
		U Content	w/o	64	63	-	0,6
		Contamination	Free Uranium Fraction	<10 ⁻⁴	6 x 10 ⁻⁷	-	2 x 10 ⁻⁶

f

Included in these stages of operation were methods of on line quality control. The latter was augmented in the development stage by statistical sampling methods for the parameters of interest for irradiation performance assessment, e.g., density, sphericity, layer thicknesses, isotropy, damage fraction of coated particles in fuel bodies etc.

"It should be stressed that the manufacturing process was designed to give a high quality product rather than, for example, to achieve this by inspection and selection. The process produced kernels with a narrow size distribution which further allowed tight control of coating thicknes as in the coating step.

"The overall plant throughput was a minimum of 5 tonnes U/y. The stage throughputs were not matched because this was not essential to meet the work objectives. Matching would be attained in the ultimate plant by replication, unit size being dictated by geometry and uranium enrichment. Therefore in some areas stage throughputs were at least 10 tonnes U/y.

"Although the plant was optimised for the production of the UK Mark 3 HTR Reference Design Fuel, it has sufficient flexibility to make a range of HTR fuels. The most recent addition to this relating to fuel pins with integral cladding the directly cooled fuel pin concept".

Federal Republic of Germany

The recent Geneva conference was the occasion on which a review of production experience and irradiation testing in the Federal Republic of Germany was given [21]. The preparation of spherical fuel and fuel elements for the AVR and THTR reactors has been concentrated at Nukem GmbH, Wolfgang bei Hanau backed by research work and irradiation testing organised by the Kernforschungsanlage, Jülich [66]. The flow sheet of the Nukem production line for THTR spherical elements is given in Fig. 5 reproduced from [21]. Since Dr. K. G. Hackstein is also presenting a paper to this study group it would be inappropriate to go into greater detail except to point out that:

- (i) KFA, Julich is carrying out work on thorium recycle [89] particularly on the head end processing of the graphite, dissolution and extraction.
- (ii) Nucem GmbH have, more recently, been engaged in the development of prismatic fuel components including the integral block element mentioned earlier.

Other Countries

In various countries including Italy, (Agip Nucleare and SNAM Projetti), Japan [90], the Netherlands (KEMA) [91] and Switzerland (Eirwurenlingen) [59] there has been research and development on the fabrication of ceramic fuel microspheres suitable for use in HTRs, though the work has often been related to other possible reactor applications. The SNAM gel-supported precipitation process for microsphere preparation has been described in sufficient detail to enable comparison with Sol-Gel processes [92]. Three pilot plants employing the SNAM process have been built. One is intalled at the Casaccia Research Centre of CNEN to produce 500-900 μ m diam. mixed urania-plutonia [93]. The other plants are devoted to urania production and typically each has a throughput capability of 2 tonnes U/y. In Austria some early work on phase diagrams and on the development of manufacturing techniques was carried out by Metallwerk Plansee, Reutte for the Dragon Project. The work as OSGAE, Seibersdorf has concentrated on the development of fabrication methods [55] and on specialised aspects of preand post-irradiation testing such as the measurement of optical anisotropy in pyrocarbon coatings [94] and the measurement of internal gas pressure in irradiated particles at their operating temperature. The programme of OSGAE has been closely linked to the Dragon Project and to the national programme of the Federal Republic of Germany.

At KEMA in the Netherlands internal gellation using hexamethylenetetramine has been studied for the preparation of UO_2 spheres [91] and the process developed is being used in particle model experiments carried out jointly by KEMA/KFA Julich/Dragon Project.

Dragon Project [10, 15, 35, 43, 44, 58, 69, 70, 73, 74, 76, 81, 82, 85, 86]

The OECD High Temperature Reactor Project (Dragon) formed in 1959 is based at AEE, Winfrith, United Kingdom and is funded by Signatories representing 12 Countries in Western Europe. The role of the Project is to foster the HTR. It has had a large R & D programme of fuel and has the manufacturing facilities to support the operation of the 20 MW(th) Dragon Reactor Experiment which has now operated for over 1,100 full power days. The Dragon Project's programme has been broadly based and over the last 12 years has covered virtually every type of coated particle fuel. Extensive compaigns have been undertaken, in facilities capable of processing 2 tonnes U/y, to provide fuel for the UKAEA, the CEGB and the CEA for physics experiments.

The Project has a further role to play in helping to train staff. Thus many scientists from institutes and producers in Signatory Countries have had a period of secondment to the Project.

The largest size of bed which has been operated is 127 mm diameter, the upper limit contemplated for highly enriched fuel and at least 1,200 coating runs have been carried out in furnaces of this size. At least 100,000 fuel bodies of all types have been manufactured by the Project.

5. HTR FUEL ELEMENT PERFORMANCE

No discussion on the fabrication technology required for nuclear fuel elements would be complete without some reference to performance aspects. This is particularly so for the HTR as its unique features may be used to advantage when one considers performance in relation to the fabrication. Generally speaking, the favourable features in performance spring from an all-ceramic core and the subdivision of the fuel within the core in a way which leads to the maximum benefit on operational and safety grounds.

Performance data on HTR fuel elements is very extensive, accumulating from the parallel projects in Europe [19-22, 95] and the USA [96] over the past decade and the successful operation of the AVR Reactor in Germany [97], the Peach Bottom Reactor in the USA [98] and the OECD Dragon Reactor [95]. Fuel element performance may be conveniently divided into the areas of fission product release and mechanical behaviour.

The unique feature of the HTR in terms of fission product release behaviour is the number of barriers which lie between the source of the fission and the environment. After the primary barrier of the fuel particle coating, important delaying features exist in the fuel matrix, transport across gaps, slow diffusion through graphite and surface adsorption/evaporation phenomena: all these have to be overcome before the fission product enters the helium coolant.

Concerning the fuel particle, the function of the various layers in terms of performance has been described from the viewpoint of both fission product migration and mechanical endurance. Due to the very low migration rates of rare gases in pyrocarbon coatings [99, 100] the release of these species by diffusion from within the particle is zero. Very small, predictable levels of rare gas activity derive from fuel contamination near the surface of the PyC coating: this level is only of the order of 1×10^{-6} fraction of the kernel weight and leads to negligible releases into the primary circuit [101]. Although the rate of migration of particularly important metallic fission products such as caesium and strontium is much higher in pyrocarbon coatings, the use of a silicon carbide interlayer provides complete retention of these fission products also [102, 103].

In this situation, therefore, the dominant source term for fission product release is the fraction of particles with broken coatings which are present at the start of life as an outcome of manufacture or which occur during service. Control methods relating to fabrication effects have been discussed in Section 3.3.6. The mechanisms by which coatings are ruptured during operation are:

- (i) Bursting particles arising from burn-up related internal pressure.
- (ii) Crackling due to stresses generated from the effects of fast neutron dose.
- (iii) Chemical interaction between kernel and coating.

According to the requirements of the fuel cycle and the core design fuel particles have been designed to cover a burn-up as high as 750,000 MWd/t in the fissile particles of thorium cycle reactors, to the lower level of about 75,000 MWd/t in fertile particles of the thorium cycle reactors and in the low enriched uranium versions of the HTR. This range is achieved not by fundamental variations in the materials used but merely by geometrical adjustments in the design of the particle which alter the amount of internal voidage available. It also illustrates that considerable flexibility is possible which can be used to increase the operating margins if required or to ease critical areas in the fabrication technology. A further factor assisting in this flexibility is the relative insensitivity of the fuel cycle to specific heavy metal investment which leads to the adoption of a realistic and practicable approach to fabrication rather than the extreme in which theoretical economic considerations push the fuel requirements beyond reasonable limits.

The mechanical performance of coated particles has been subject to much study and apart from burn-up dependent phenomena coated particle life can be affected by fast neutron induced dimensional changes and irradiation creep in pyrolytic carbon. This has led to comprehensive experimental work on the parameters involved and to the development of numerical methods to evaluate mechanical performance [104-106]. For example, where the PyC coating is restrained by the more stable and stronger silicon carbide, the irradiation induced shrinkage is prevented but results in a tensile stress being developed in the pyrolytic carbon. The level of the stress is dependent on the characteristics of the pyrolytic carbon leading to specification requirements in the fabrication. Generally speaking these involve specifying the source gas, the density of the product and its crystallographic isotropy.

Chemical interactions in fuel particles are not limiting in current designs of power reactors which require peak operating temperatures of about 1250°C. Furthermore scope still exists for further developments to make way for higher temperatures of operation in more advanced systems [107].

Summarising, one can see in the coated particle the ideal form for a nuclear fuel. Unlike other systems one is not faced with large step functions in release due to the failure of a can which contains a large quantity of fuel. Rather one has a statistical array of microscopic ceramic fuel elements which will not suffer simultaneous failure since there is no sharp cut-off from anything analagous to a metallic can melt-out. Furthermore, due to the statistical spread any deterioration is progressive and the absence of fundamental limits still allows scope for yet further enhancement in performance.

Effects in the fuel matrix and graphite which control fission product release can vary enormously depending on the fuel element design and on the operating temperatures. Of the two materials much less attention has been paid to the matrix than the graphite. This, in general arises from the fact that the operating temperature of the matrix is higher than that of the graphite. Thus migration rates of the more important long-lived fission products are too high to be of great benefit when compared to the coated particle retention. On the other hand there is evidence to show that useful retention effects can occur for caesium by adsorption in the matrix carbon. It has also been shown that gaps between matrix and graphite can lead to strong retention at the matrix surface due to the low vapour pressure of metals adsorbed on the matrix. Although these phenomena exist they have not been exploited in designs to date but will become more important with the development of directly cooled fuel designs.

Significant fission product retention effects are present in the graphite and in the non-fuelled zone of directly cooled fuel elements which may be treated as graphite. Benefit results both from the time-delay in fission products migrating across the graphite and the rate of evaporation from the graphite surface. Computer codes have been developed to take these into account when assessing release phenomena in HTRs [108]. Generally speaking it is the behaviour of the metals particularly caesium and strontium which is of most interest in the fission product performance of the graphite. The migration rate of strontium in particular is sufficiently low in graphite that even at reasonably high temperatures the graphite affords such a delay that a considerable source of strontium release from the fuel could be tolerated [109-110].

This is not so true for caesium, however, and for this isotope greater reliance must be placed on the primary barriers in the coated particle and on evaporation phenomena at the fuel tube surface. From the fabrication viewpoint there seems to be relatively small variations in different graphites with respect to fission product behaviour and the graphite specification for HTR contains no points specific to fission product behaviour: instead the question of graphites for the HTR has been always approached in the main context of the structural integrity of the fuel element and core.

The uniqueness of graphite as a structural material lies in its increasing strength with temperature (the UTS at 2600°C is twice the room temperature value) coupled with its relatively high thermal conductivity and low coefficient of thermal expansion. This combination leads to it being the ideal material for withstanding high levels of heat flux such as are experienced in High Temperature

Reactors operating at high specific power densities. The alternative designs of HTR have been outlined earlier. The common features relating to mechanical performance have been studied [29, 111, 112] and relate to:

- (i) thermal stresses and strains,
- (ii) irradiation induced stress and strains,
- (iii) effects arising from mechanical interaction between different . components in the fuel element and core

To obtain design information programmes of physical and mechanical characterisation of graphite and matrix fuel bodies have been conducted and these have been based around extensive irradiation programmes [113]. It is indeed the irradiation behaviour which is the prime component in dictating the type of specification for material supply [113].

In the case of graphite this is also very much related to the commercial availability. Fortunately the basic HTR feature of a core replacement at intervals of about 1,000 days leads once again to a flexibility of approach since the resulting peak fast neutron damage to the graphite is relatively modest. Thus the designer finds it possible to work within the known behaviour of the material and the specification for its supply relates mainly to maintaining consistency of product.

6. CONCLUSIONS

- 1. The underlying features of the HTR in the use of coated particles, graphite and helium lead to a system which is elegant and highly flexible both in terms of choice of fuel cycle and fuel element design.
- 2. Although coated particle technology is relatively new, manufacture has not posed difficult problems and the unit equipment used in the manufacture is basically simple.
- 3. Much of the HTR fuel element rests on conventional graphite technology and the life-time requirement in power reactors is such that requirements are not stringent.

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4. The most undeveloped area in the fuel element technology is that related to remote refabrication. Although the processes in manufacturing HTR fuel elements are ideally suited to this, further costly work is needed to lead eventually to a production capability and there may then be interactions on fuel element design. Because this is essentially a problem of cash flow for the commercial organisation(s) involved, alleviated to a greater or lesser extent by governmental funding, this is certain to lead to a geographical concentration of the refabrication. It also leads to the view that, in the short term, fuel cycles not depending on refabrication should not be lightly abandoned.

7. ACKNOWLEDGMENTS

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In these references the following abbreviations are used:

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FIG. | HTR FUEL ELEMENT DESIGNS

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OECDFIG. 2DRAGONPHOTOMICROGRAPH OF EQUATORIALDRAGONSECTION OF TYPICAL UO2 COATEDPARTICLE AND DESIGNATION OF THE.COMPONENTS OF THE PARTICLE.







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FIGURE 4 - THORIUM RECYCLE



THE REPROCESSING OF IRRADIATED FUELS

TECHNÍCAL AND ECONOMIC CONSIDERATIONS - A REVIEW

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ABSTRACT

The main emphasis is laid on the processing of light water reactor fuels, as this is the only kind which has reached full industrial scale today.

All the chemical processes used at present are derived from the "Purex Process", developped in the United States around 1954. The main characteristics of the chemical process and the main aspects of plant operation are reviewed.

The discharge of radioactive gases and effluents being one of the major problems facing the reprocessing industry, a brief statement of the envisaged solutions is given.

Economic considerations involving investments and operation costs, taking into account the main parameters such as the size of the plant and its load factor, are also discussed.

Some indications of the R and D efforts currently being made to develop reprocessing of the high temperature gas-cooled reactor are given, since no production facility is in existance as yet.

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1. INTRODUCTION

This paper presents a survey of the technical and economic aspects of the reprocessing of spent reactor fuels. The main emphasis will be laid on the reprocessing of the light water reactor fuels (LWR), which until today is the only process to have reached full industrial scale.

The processes universally adopted are aqueous processes all of which are derived from the "Purex Process", developed in the United States around 1954.

The routine discharge of radioactive gases and effluents (one of the major problems facing the reprocessing industry) is also discussed and the possible answers to these problems are analysed.

Economic considerations on the investments and operation are presented to complete the review on light water reactor fuel reprocessing.

Some information on the Research and Development efforts currently being made to enable the reprocessing of high temperature gas-cooled reactor fuels (HTGR) are reviewed, no production facility at present existing to reprocess this type of fuel.

2. THE REPROCESSING OF LIGHT WATER REACTOR FUELS

2.1. Description of the process

Figure 1 gives an outline of the reprocessing process.

2.1.1. S, ont fuel trans; rtati n and storage

The fuel is received in shielded casks transported either by road or rail. Up till now transports in Europe are performed in dry casks, cooled by natural convection. However, in the United States transport of power reactor fuels is generally made with watercooled containers.

Unloading of the casks and storage of the fuel is generally performed under-water in various interconnected pools. The storage racks are geometrically arranged to ensure that they are critically safe.

Fuels are normally allowed to cool for at least 150 days prior to handling.

2.1.2. Mechanical treatment and dissolution

Mechanical treatment has now been adopted overall as the preparation step prior to the fuel dissolution.

The Eurochemic plant at Mol is the only facility applying chemical decanning to LYR fuels. The main drawback of chemical decanning is the high volume of intermediate level wastes: $4 - 5 \text{ m}^3/\text{ton}$ of uranium for stainless steel cled fucl and $8 - 10 \text{ m}^3/\text{ton}$ of uranium for zircaloy cled fuel.

Mechanical treatments applied or envisaged can be sub-divided into two main categories: shearing without any previous dismantling, or initial partial dismantling by cutting off the end portion of each fuel element and removing any existing casings and/or separating the bundles of pins into sub-assemblies.

With both techniques the pins are cut into pieces, ranging in length from 1.5 to 5 cm. The cut pieces generally fall into a chute and are collected in fuel canisters.

The shear is usually actuated hydraulically.

The fuel canisters, made of stainless steel perforated sheets eventually containing boron as a neutron poison, are leached with boiling nitric acid.

Complete dissolution of oxide fuels can be achieved in about 2 to 4 hours; metal fuels require about 24 hours.

The treatment of the off-gas implies :

- the reconversion of the nitrogen oxides resulting from the chemical reaction between the nitric acid and the core of fuel, into nitric acid which is recycled to the process ;
- various treatments to remove or allow safe discharge via the stack of radioactive species such as iodine, tritium and noble gases (treated in more detail under § 2...2.);
- filtration through high efficiency filters to retain solid active or activated particules prior to discharge of exhaust through the stack.

The fuel solution is transferred to feed adjustment and accountability tanks, where the volume is measured accurately and multiple samples of the solution are taken for analysis. This analysis represents the feed input to the system and constitutes the financial accountability of the plant. It is also important from the standpoint of criticality control for the whole process, since the parameter usually adopted to assure the safety of the operation is the limitation of the fissionable material. Indeed, fully geometrically safe equipment is not practical for large production plants, both from an investment and from an operation point of view.

After the transfer of the feed solution the leached hulls are rinsed and dried. The removed hulls are usually checked to guarantee the efficiency of the core dissolution. This check can be performed by visual inspection on hulls randomly selected from the dissolver canisters, or by O-scanning, by passing the canisters through a neutron beam produced by an auxiliary neutron source.

The hulls are finally sealed into drums, transported under shielding and stored as radioactive solid waste.

Since the radiation level of these hulls is very high (10,000 R/hr and above) these operations must be performed by remote control.

Figure 2 schematises the various steps of the mechanical treatment and dissolution operations.

2.1.3. Purification by solvent extraction

The decontamination and recovery of uranium and plutonium from fuel solutions is universally accomplished today by solvent extraction processes, using tributyl phosphate (T3P) dissolved in saturated hydrocarbons of the kerosene or n-dedecane types.

All these processes are derived from the Purex Process applied industrially for the first time in the United States in 1954.

Since then all attempts made to substitute other processes - such as the so-called "3ry methods" for instance, based on the distillation of fluorides - for solvent extraction have never reached production stage and have even been withdrawn from research and development programmes in many countries.

Indeed, solvent extraction presents several major advantages for the reprecessing of the spent nuclear fuels, as summarised in the following table (table 1).

The main advantage of using TBP as extractent, is the possibility it provides of using nitric acid as salting agent. Indeed, this nitric acid being easily purified by distillation, it is possible to recycle it into the process and to reduce the amount of radioactive effluent to be discharged by consequence. Figure 3 summarises an extraction flowsheet of the Purex type. However, several versions of Purex flowsheets are in existance, but all are based on the same general features.

In the first extraction cycle or "co-decontamination cycle" the nitric acid solution containing the uranium, plutonium, fission products and various impurities, is contacted with the solvent (30 vol % T3P in kerosene). The organic phase extracts, in order of preference, the uranium and the plutonium and leaves elmost all the fission products in the aqueous phase, which subsequently becomes the high level radioactive aqueous waste from the process.

The organic phase, which contains the uranium and the plutonium, is scrubbed with nitric acid to remove additional impurities and is then transferred into a second contactor, where it is contacted with water which strips the uranium and the plutonium back to an aqueous phase.

In the second extraction cycle or "U-Pu Partitioning Cycle" the aqueous phase leaving the first cycle is after concentration in an evaporator - fed again to an extraction-scrubbing contactor. The loaded phase leaving this contactor is routed to another, where the uranium is separated from the plutonium by contacting the organic phase with an aqueous solution that contains a reducing agent, such as uranous nitrate or ferrous sulphamate. The plutonium which has been tetravalent up to this point of the process is now reduced to the less extractable trivalent state, so it can be removed from the contactor in the aqueous phase. The uranium is then stripped again from organic phase, in a third contactor.

One or two additional cycles of solvent extraction are used to complete the uranium product. The plutonium is processed through one additional solvent extraction cycle, followed by anion-exchange treatment to achieve final purification and concentrations.

The acid waste streams are evaporated:

- to concentrate them for storage ;
- to decontaminate the excess of liquid ;
- to recover nitric acid for recycling to the process.

The solvent streams leaving the extraction streams are purified from the remaining uranium, plutonium and fission products and from the degradation products of the diluent and the TBP resulting from the chemical and rediochemical damage to the organic phase. The cleaning processes involve generally caustic and acid washes of the solvent. After cleaning the solvent is recycled back to the process.

The extraction cycles of the reprocessing plant generally allow the recovery of 98.5 to 99.5 % of the uranium and plutonium fed into them.

Table 2 hereafter cutlines the decontamination factors usually achieved with a Purex flowsheet of the type described in figure 3.

2.2. Plent design

2.2.1. General

Considerations for the design of a plant are in some respects similar to those facing the designer of any complex chemical plant, involving widely differing unit operations and unit processes, and handling valuable materials.

One of the major design problems of radiochemical plants is the control of radioactive and fissile materials being handled. The presence of large amounts of radioactive compounds requires that precautions be taken for the protection of personnel, equipment components and processing reagents. Such protection must also be extended to the surrounding areas beyond the plant limits. Radioactivity also requires remote operation of the plant and presents problems with regard to maintenance of the equipment.

Nuclear safety also constitutes a major design problem for radiochemical plants handling large amounts of fissionable isotopes. This problem is countered by the appropriate choice of a combination of process conditions, plant layout and equipment sizing, along with proper plant supervision and intensive personnel training.

2.2.2. Protection against radioactivity

Shielding is the most practical and commonly used method of ensuring personnel protection. In addition provision must be made for strict control of the working time of personnel under certain circumstances.

Although there are more effective shielding materials than concrete, it is comparatively cheap and is the most common material used for shielding in reprocessing plants. It serves not only as a shield but also as a structural material and helps to make the plant resistant to external blasts. As decontamination is achieved by the various process steps, less shielding is required along the production line.

Shielding economies are made by dividing the building into zones of various radiation levles. Table 3 below gives an example of such a segregation of areas.

2.2.3. Protection against contamination

Ventilation is the safest means of preventing the spread of airborne contamination in the working areas and plant environs. It constitutes a protection against the ingestion of radioactive materials.

In practice the ventilation system is broken down into areas by the level of radiation. The direction of air flow must always be from the area of lowest to the area of highest radioactivity. Differential air pressures are used to ensure this control or air movement.

It is necessary, within the cells themselves, to operate process equipment at a somewhat lower pressure than the cell ventulating system, in order to avoid the possibility of exhausting highly radioactive gases or vapours into the cell ventulating system.

Before release to the stack, all the exhausted air is passed through batteries of filters which normally remove 99.99% of the particulate matter. The stack must be of sufficient height to minimize recirculation or contamination of immediate surrounding areas under essentially all atmospheric conditions.

2.2.4. Maintenance

Because of high radiation levels present in reprocessing plants, the maintenance is a problem of great magnitude. There are two basic approaches to the problem, namely remote or direct maintenance.

It is obvious that maintenance is a major consideration in plant operating time. The basic requirement in design of any type of plant is that the equipment and layout be such that an absolute minimum of maintenance and resulting idle-time is required.

The remote maintenance system adds considerably to the initial cost of the plant. This increment is not materially altered by the size of the plant, since the cost of the various devices required remains essentially the same. It is difficult therefore to justify remote maintenance for a small plant. On the other hand, the direct maintenance concept requires periodic shut-down and decontamination before direct maintenance can be performed on plant equipment.

For production plant a suitable compromise between the two types of maintenance has to be found.

2.2.5. Criticality

The possibilities of an atomic explosion in a reprocessing plant are negligible. However, a critical incident can and will produce lethal radiation and can be expected to produce physical damage in the immediate area, heavily contaminating the nearest surroundings. Furthermore, distant contamination might occur as a result of airborne fission products or dissipated fissionable materials. It is obvious that the plant and equipment must be designed with suitable controls to prevent such critical incidents.

Criticality is a complicated problem, since reprocessing operations handle fissionable isotopes in numerous forms and under a variety of conditions.

Moreover it is normally impossible to isclate the operations to such a degree that the human element can be ignored. Therefore, management must provide adequate control in the design and operation of the plant and must also educate operating personnel to prevent criticality incidents.

In the plant design stages a decision must be made as to what type or types of control are to be used. Unce the control philosophy has been defined, all equipment or operations used must conform to the limits established.

For any of the fissionable isotopes to become critical certain conditions must be met, which can be described by the following variables :

- a) mass of fissionable material
- b) concentration of fissionable material
- c) isotopic composition of uranium and plutonium
- d) components present along with fissionable material (moderators)
- e) poisons (scluble or not)
- f) volume

- g) geometry (shape)
- h) reflection
- i) interaction
- j) primary dimensions (cylinder diameter, slab thickness, etc...)

Safe conditions can be prescribed by selecting limiting values for one or better two independent variables amongst those above.

2.3 Equipment design

2.3.1. General

There are several factors requiring special consideration in equipment design for radiochemical plants. The unique factors that must be considered are avoidance of criticality and the special hezerds associated with the handling of radioactive materials.

In addition the high product recovery and purity require special design consideration. Other than the limits imposed by these special problems, equipment design follows standard practics.

2.3.2. Radiation aspects

a) Tightness

Because of the radiation problems it is essential that leaks be virtually non-existant. Equipment and piping specifications demand absolute leak-tightness as far as possible, both for internal and external joints. Added expenses are esteemed necessary to ensure that these specifications are met. Such rigorous conditions are justified in view of the fact that any type of maintenance operation in a radiochemical plant is many times more expensive and timeconsuming than in any conventional plant. Thus any reasonable effort and expense expended in the initial design, fabrication and installation of equipment are more than repaid during the life of the plant in terms of reduced maintenance and operating costs.

Mechanical design of plant vessels and piping in radioactive areas generally requires "all welded" construction. The welding of stainless steel equipment is accomplished primarily by using inert gas welding techniques followed by various very severe checkings. b) Radiation damage

The radiation levels have little effect on the metal commonly used in fabricating equipment.

dadiation effects on various non-metallic materials limit their usofulness in "hot" areas. Local shielding is used to protect equipment containing nonmetallic components such as gaskets, electrical insulation, packings, lubricating oils or greases etc..

Equipment with components susceptible to radiation damage and not handling "hot" solutions directly have to be located outside the cells.

c) Maintenance

Because the personnel will come into contect with the quipment to some degree, either in replacement or maintenance (which will become necessary after a certain time lapse, inspite of all efforts), design features must be incorporated to allow decontamination of the equipment while it is still in place.

To facilitate decontamination operations, specifications for all equipment designs and fabrication should be stringent. Equipment should be designed so that accessory lines and surfaces will drain free to minimize the hold-up of active solutions. Both internal and external surfaces should be reachable by decontamination solutions.

2.3.3. Criticality aspect

In the design of equipment, the type of criticality control to be used must be considered. Once the control methods are established, the process and equipment must be designed to meet the limiting values of the control methods used. Basically, plants may be designed using mass, concentration, volume, geometry control or combinations of these latter.

Nuclear safety may be incorporated into process equipment through the application of one or more of the variables enumerated in § 2.3.5. It is impossible for example to design a reprocessing plant using geometry as a control, in which all the equipment is so shaped that no critical accumulation can occur, regardless of other factors such as the quantity of material in process, its chemical concentration or the proximity of neutron-reflecting bodies. At the other extreme the total mass in process at any one time can be maintained below the minimum that can support a chain reaction, with all other contributing factors optimized.

Neither of these extremes is entirely workable. Instead the practical approach to nuclear safety is the simultaneous application of two or more criteria, so that uneconomic extremes are avoided.

2.3.4. Materials of construction

Equipment utilised in spent fuel processing is subjected to varying degrees of radiation, chemical corrosion, mechanical stresses and high temperature conditions.

The worst corrosive conditions occur in dissolvers, concentrators, nitric acid absorbers or fractionators, waste evaporators and storage tanks.

In choosing a material to resist corrosion the following factors must be considered :

- a) the production of high purity product limits the permissible pick-up of contaminants from the construction materials;
- b) the extensive problems posed by maintenance require that materials of construction be chosen for long life;
- c) the materials of construction must resist decontaminating solutions used.

The construction material generally used is stainless steel of the 300 series, and most specifically 304 L, stabilized toward intergranular corrosion either by extra low carbon content or by a niobium (columbium) additive. (See table 4)

2.3.5. Chemical Process Equipment - Extraction Contactors

It is impossible, within the limits of this review, to describe all the various types of chemical process equipment used in radiochemical plants. However, the heart of the reprocessing chemical plant being the extraction section, special attention will be paid here to the liquid-liquid extraction contactors used in the various "Purex" plants.

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Extraction equipment used in reprocessing operations has been developed over the past 20 years to give good separations, decontamination and high recovery on a continuous basis. Of equal importance has been the desire to make the equipment as small and compact as possible, since the shielded buildings are quite expensive. Moreover, the permanent increase of the specific activity of the solutions processed has enhanced the necessity to minimize the contact time between the organic and the aqueous phases, in order to reduce the risks of radiolysis of the former.

The ideal contactor requires a minimum investment for both associated building and equipment. It maintains a high efficiency over a wide range of operating conditions, has a high capacity with a low hold-up, is reliable in operation with simple controls and minimum maintenance requirements, is flexible for process variations and reduces the organic-aqueous phases contact time.

The choice of contactor for a particular process is governod by the desire to include as many of these characteristics as possible.

a) Packed columns

represent the simplest type of contactor, as they have no moving parts. They have been adopted in the early stages of the reprecessing technology but are now almost abandoned due to their poor performances compared with pulsed columns and mixersettlers of comparable size.

Indeed, in these two types of extractors, mechanical agitation is used to enhance the efficiency of the extraction processes.

b) Pulsed columns (figure 4)

use hydraulic pulsation of the liquid phases through a series of perforated plates, to enhance the rate of mass transfer.

Due to their greater efficiency, pulsed columns are more compact than packed columns and thus require less shielding. They also have a smaller liquid hold-up and therefore the risk of solvent degradation by radiation is decreased. The hydraulic pulse is generated by application of air pressure by a mechanical pulser located outside the shielded cells housing the columns.

fhe capacity and efficiency of the pulsed columns are influenced by a large number of variables such as: pulse frequency and amplitude, perforated plate geometry, column dimensions, etc...

c) Mixer-settlers (figure 5)

The "pump-mix" mixer settlers, an horizontal contactor in which the mixing intensity, the interface control and the hydraulic head for the aqueous phase flow are provided by the impeller, is the type of mixer-settler most commonly used in the reprocessing plant.

A contactor consists of any desired number of stages (one stage being formed by a mixing chamber associated with a settling chamber where the aqueous and the organic phase separate), arranged horizcutally.

In spite of the fact that each stage requires an individual motor for lriving the impeller of the mixing chamber, mixer-settlers have been used in many reprocessing plants.

However, the accumulation of radioactive cruds at the interfaces are often observed during the processing of fairly radioactive solutions. This phenomenon, due to the rather long contact time between the phases at each stage, reduces to some extent the efficiency of the decontamination. Consequently, the use of mixer-settlers to process highly irradiated fuels such as those discharged from light water power reactors is rather questionable.

d) ^vast contactors (figure 6)

A large part of the efforts devoted to the adaptation of the Purex Process to the higher specific activities of the spent fuels is coviously related to the minimization of the radiation damage to the solvent phase.

Attempts to develop other extractants with a better resistance towards the radiation than FBP have been made without great success.

It appears now that the best improvement has to be found in the contactor technology. Because the radiation damage to the organic compounds varies proportionally with the time that these compounds are exposed to the ionizing radiation, and because this exposure time is primarily determined by the time required to separate the aqueous and organic phases, the main efforts have been devoted to designing means for shortening the time required to separate the phases.

Centrifugal force is the most commonly considered means.

For the past few years reliable equipment known as "fast contactors" or "centrifugal contactors" has been developed in the United States, France and Germany.

A hot pilot unit, of industrial scale, is in operation at the Savannah River Plant in the U.S.L.

The contactor is made of 18 stages of the type shown in figure 6. It operates at about 1750 revolutions per minute.

New reprocessing plants, or existing ones to be adapted for the processing of fuels from coming power reactors, must most probably be equipped with fast contectors, at least for the first contactors of the first cycle.

This requirement is essential if the plant uses mixer-settlers, since the overall contact-time between the aqueous and organic phases is by far longer than in pulsed columns.

2.4. The waste problem

2.4.1. General

The treatment, storage and disposal of radioactive effluents from reprocessing plants is one of the most important and difficult technical problems to be solved in the nuclear power industry.

In current fuel reprocessing schemes, the volatile or gaseous radionuclides present in the spent fuels (e.g. 1^{129} , 1^{131} , Kr^{85} , λe^{131m} , λe^{133}) are volatilized and the resulting gaseous stream is treated for removal of some of these nuclides to meet emission limits.

Tritium is usually included in this category of volatile radionuclides, in spite of the fact that a large portion of it generally follows the water streams.

Approximately 99.9% of the remaining fission products are collected in the wasto from the first extraction cycle (see figure 3).

Figure 7 gives a general scheme of the origin of the radioactive wastes from reprocessing plants.

The analysis of the problems of waste management is beyond the scope of this review. However, due to the importance of the subject some general considerations are given hereafter, concerning the treatment of the volatile and gaseous wastes and high level liquid wastes, which are of major concern for reprocessing plant operators and environmentalists.

2.4.2. Vclatile and gaseous wastes

It is present practice to reprocess fuels about 150 days after discharge from the reactor. In these conditions volatile or gaseous radionuclides to be taken into consideration are Kr^{85} , I^{131} , I^{129} .

Today it is considered that the best way to dispose of tritium - present as tritiated water in all the distillates of the various waste streams (see figure 7) is to vaporise it through the chimney.

This is the reason why tritium is associated with the gaseous and volatile radionuclides mentioned above.

To illustrate the magnitude of the discharge of these effluents, data from an American origin are given in table 5 below.

Actually the required decontamination factors for Kr⁸⁵ are very low, for the commercial plants envisaged at present. However, in connection with the future processing of short cooled fuel from fast breeder reactors several studies are underway, for removing noble gases from off-gas streams. These include room-temperature or low-temperature adsorption on charcoal, molecular sieves or silicagel, cryogenic distillation, extraction by liquids, trapping in clathrates and separation using permselective membranes.

All these processes can remove more than 90% of the noble gases.

Simultaneously useful techniques for the control of iodine volatilization and its release to the atmosphere are being developed. They include: the use of nitric acid/mercuric nitrate (D.F. $10^2 - 10^3$) or caustic (D.7. $10^3 - 10^4$) scrubbers and the use of charcoel adsorbers (D.F. $10^4 - 10^5$).

These indications show that the present techniques allow the requirements defined in table 5 to be met.

2.4.3. High level wastes

The high level wastes $(0.5 \text{ to } 1.2 \text{ m}^3/\text{ton of uranium}, after concentration)$ are those which contain at least 1 Ci/liter of liquid or 70 Ci/kg of solid. These values are high enough to require thick radiation shielding and cooling of the significant heat evolution from decay of radioactive fission products.

This waste must be stored for centuries to permit decay of the major long-lived radicactive fission products (primarily Cs¹³⁷ and Si⁹⁰) to non toxic levels; the small amount of plutonium present as a waste loss (0.1 to 0.5 % of the original amount in the fuel) and some long-lived radionuclides (transplutonium elements) require thousands of centuries to decay to non toxic levels.

At present the only practical means of preventing the radioactivity from entering our environment is to store the material under carefully controlled conlitions.

at this time, major technological advances are required for development of new techniques.

a) Liquid storage

Interim storage of those high level wastes as liquids has been done safely from the early stage of the atomic power industry.

The storage tanks vary in capacity from 40 m³ to 4000 m³. The tanks are housed in concrete vaults, underground or above ground. They are generally equipped with cooling coils for heat removal and airlift circulators to prevent deposition of solids.

Generally speaking the high level wastes are stored as acid solutions; however, some American facilities neutralize their waste to alkaline conditions, prior to storage. Stainless steel (304 L) is now generally used to store acid waste; carbon steel has been adopted for alkaline waste storage.

No leakage has been reported today for stainless steel tanks, but fifteen carbon steel tank failures have occurred in U.S. facilities.

The storage of acidic wastes in stainless steel tanks is today regarded as preferable to alkaline storage, based on :

- lower volumes
- smaller volume of precipitated solids
- easier control of heat removal
- smaller total salt content, giving less volume after solidification
- less tendency to leak
- comparable costs.

As the nuclear industry expands, storage problems become increasingly complex. In the last years it has been generally recognized that storage as solids versus liquids would be considerably safer.

The proposed policy for all U.S. commercial fuel processing plants stipulates that :

- 1) the inventory of high level wastes in liquid form will be limited to the quantity produced in 5 years
- 2) the wastes in excess of this inventory must be converted to an approved solid form
- 3) that the solidified wastes must be transported to a federal repository as soon as practicable, but in no event later than 10 years following the fission products separation from the irradiatec fuels.

It is likely that a similar approach will be adoptel in other countries, making the liquid storage only an interim solution to the problem.

b) Solidification

Processes for the solidification and permanent disposal of high level wastes are being developed or considered in countries that have or plan to have reprocessing plants.

The general objectives of solidification are :

1) to reduce the mobility, solubility and volume of wastes

2) to maximize the thermal conductivity and thermal stabilities of the final product.

Table 6 summarizes the characteristics of the main high level waste solidification processes.

As shown by this table none of these processes, with the exception of the fluid bel calcination process developed at Ilaho Chemical Plant, has reached industrial application yet.

However, the calcine obtained by the Idaho process cannot be considered as a product suitable for perpetual storage.

c) Long term disposal or storage

Long term disposal or storage is a feature common to all currently practicable methods for disposal. The methods which have received attention are :

- disposal as solids in salt mines
- storage as solids in deep underground caverns
- storage as solids in man-made vaults
- disposal as solids in ocean depths
- disposal as liquids in deep wells
- perpetual storage as liquids in deep underground caverns
- perpetual storage as liquids in tanks.

To many waste disposal experts the schemes considered to have the better potential for long-term safety are those whereby wastes are placed in certain deep geological formations.

Placing the wastes in these locations should keep the wastes out of man's immediate environment for the required length of time, the wastes would be less affected by future unknown conditions on the surface, caused by either man or nature, and reliance on human surveillance should be minimized.

One source of concern involved with any storage or disposal technique is whether or not enough is known to be certain that stored wastes will remain "eternally" out of the biosphere. Systems offering retrievability allow for a problem that may not have been foreseen. On the other hand, retrievability in a storage system would normally require more surveillance, more control records and more legislation to be enforced. Today no system for final storage has been definitively adopted and consequently there is no system in actual operation.

The Unite? States and the Federal depublic of Germany are studying the storage in a salt mine. If the German project (Asse, Lower Saxony) is pursued according to a strict planning, the U.S. programme seems delayed due to local conditions. The USAEC, which still considers storage in a salt mine to be an acceptable solution, nevertheless decided to adopt "engineered storage facilities" for the immediate needs of the country.

France and the United Kingdom have not yet fully defined or announced the lines they want to follow in this matter.

Ubviously, the management of the radioactive wastes to prevent them from entering man's biosphere in significant quantities is a formidable task which requires international cooperation.

2.5. Nuclear Material Accountability

The nuclear material accountability is needed for plant operators to follow the flow of material through the plant, to prevent losses, to prevent undetected accumulations of fissionable materials which could lead to criticality accidents, etc...

The same information is also basically required for safeguards control, which must keep track of the fis-sile material.

To facilitate these operational and safeguards controls the reprocessing facility is generally subdivided into <u>Material Balance Areas</u>. The in and out flows through each of these areas are checked as carefully as possible.

Special attention must be paid to these requirements right from plant design' stages, to provide each material balance area with the necessary equipment to allow efficient measurements: sampling loops, accurate volume measurements, calibrated vessels, etc... In addition, during plant operation periodical (at least yearly) determination of material hold-up in the plant is needed, to allow a better compilation of the accountability data accumulated during operation.

On the basis of the present technology an overall material recovery (uranium and plutonium) of minimum 98% is generally considered. The 2% losses cover the losses in liquid and solid wastes and the unavoidable discrepancies in the material balance.

2.6. Economic Considerations

2.6.1. Introductory lemark

Since the most complete economic data at the author's disposal dates from the end of 1969, 1970, the figures presented below refer to the US & of that time. No attempt has been made to actualize these figures due to the financial events which have occurred in the meantime.

2.6.2. Reprocessing cost and energy cost

It can be estimated that a 1000 dV(1) light water reactor discharges 35 tons of spent uranium per year. (1 kg uranium producing 2.10^5 kwh).

Assuming 4.5 mills/kwh as the total electricity cost, the fuel cycle total cost would represent about 1.5 mills/kwh or a third of the electricity cost. As the reprocessing part in the fuel cycle is generally fixed at 5 to 10% the reprocessing cost may be estimated at 0.07 to 0.15 mills/kwh or 15 to 30 % per kg uranium.

2.6.3. Investment costs

A characteristic of reprocessing plants is their high base cost. This is mainly due to the heavy shielding, the requirements for remote operation and maintenance free operation of equipment inside the cells.

Items which are essentially independent of capacity incluse civil engineering, ventilation, instrumentation, the analytical facility and general services.

The general curve for upsealing the investment cost for reprocessing plants follows a law of the type :

$$\frac{\mathbf{I}}{\mathbf{I}_0} = \mathbf{A}^{\mathbf{X}}$$

where $I_{\alpha} =$ reference plant cost

- I = plant cost
- A = capacity factor
- x = lies between 0.24 and 0.4, according to local contingencies.

In 1969 the investment cost of a 1 ton/day plant could be assumed at 40 - 45 MS and that of a 5 t/d plant at 80 - 85 MS, including all ancillary costs prior to start-up and assuming a construction time of 4 - 5years with an interest rate of 10%.

In reality the cost of a new plant may differ considerably from these amounts. A precise estimate can only be made when the size and location, which has a marked impact on the waste storage and discharge possibilities, are fully defined.

Table 7 illustrates a possible breakdown of the investment costs. These figures are purely indicative; they have been compiled from Eurochemic data and various external publications.

Economic calculations usually adopt a depreciation time of 15 years, but obviously the technical life-time of a plant is generally longer.

2.6.4. Operating costs

The plant operating costs are usually sub-divided into constant costs, variable costs and maintenance costs.

a) Constant costs

Constant costs include:

1) Personnel costs - Personnel requirements depend essentially on plant lay-out, degree of automation, and availability of external assistance during maintenance periods for instance.

It can be assumed that of the order of 250 and 350 persons are required to operate plants of 1 T/day and 5 t/day respectively. ii) Operating costs - These costs include plant services which run independently of the throughput (water, steam, electricity for ventilation, air), consumable materials (apart from processing chemicals), meintenance and spare parts, edministration and insurance.

These expenses remain practically unchanged, whether the plant is in operation, maintenance or temporarily shut-down.

The constant operating costs are generally estimated to amount to approximately 2/3 of the personnel costs.

b) Variable costs

These expenses depend basically on the quantities of fuel handled; they mainly cover plant services (steam and electricity), process chemicals in plant waste handling and interim storage on site. Waste final conditioning and storage are not considered here.

Table 8 gives an indication of the distribution of these variable costs among their constituting items.

c) Maintenance costs

On one hand, maintenance costs are lower than in conventional chemical plants due to :

- high investment in civil engineering (shielding)
- built-in safety and strict specifications.

On the other hand, repairs are more expensive because of :

- decontamination prior to access, which often means long shut-downs and costly reagents;
- difficult access and restricted working conditions, which slow down the repair work.

Maintenance generally accounts for 7% of the yearly operating expenses; man-power represents more than 80%. Maintenance costs have been included in item a) above.

2.6.5. Overall reprocessing costs

Due to the high level of investment and constant costs, high load factors and plant capacities are required in order to achieve reprocessing costs in the range of 15 to 30 %/kg U as indicated in [2.6.2.

Table 9 hereafter illustrates these facts.

2.6.6. Additional costs

Besides the operating costs discussed above, some additional costs have to be taken into consideration if a total view of the spent fuel processing economy is wanted. For instance :

- the transport of the irradiated fuels ;
- the final conditioning and storage of the high level wastes ;
- the fissile material accountability and safeguards;
- the general safety of the plant operation, including permanent personnel training.
- a) Transport costs

The transport costs of irradiated fuels are not very sensitive to distance; the most influential factors are the utilisation coefficient of the transport containers (rotation speed) and their load factor (ratio between container weight and fuel weight).

In Europe the transport costs may be evaluated at 5 to 9 ω/kg uranium.

b) High level waste conditioning and final storage

<u>all</u> estimations in this field <u>are purely indicative</u> as no operation of this type has yet been carried into effect, as seen in § 2.4.3.

A number of cost estimate studies have been made and published, especially by Blomeke and coworkers at ORNL.

These studies show that for management comprising :

- interim liquid storage, solidification by pot calcination ;
- interim storage of the solids ;
- transportation of the solids (2000 miles round-trup);
- storage in salt ;

the estimated cost is of the order of 9.5 to 14.5 3/kg or uranium processed.

Comparable costs for perpetual storage as liquid vary from 7.8 up to 9.1 ϕ/kg U.

Consequently high level waste management <u>could</u> represent up to 30% of the actual reprocessing cost if it is estimated around 30 δ/kg (see table 9). 3. THE REPROCESSING OF HIGH TEMPERATURE GAS-COOLED REACTOR FUELS

> At present no reprocessing plant exists to process fuels from high temperature reactors. However, important 2 & D progremmes are devoted to this problem, namely in the U.S.A. (Oak Ridge National Laboratory and Idaho Chemical Plant) and the Federal Republic of Germany (Julich).

> The methods for processing HTGR fuels aim to recover the U-233 and the thorium from the fertile and recycled particles and to separate the highly burned U-235 particles. The high U-236 content of the latter makes the U-235 undesirable for recycling.

It can be said today that the burn-leach process is preferred to the grind-leach process, at least in the United States. The major factor in this choice is the necessity for physically separating the particles.

The principal processing problems encountered in the burn-leach methol are :

- 1) mechanical operation prior to burning ;
- 2) control of the fluid-bed burner ;
- 3) removal of radioactive particles and gases from the off-gas ;
- 4) separation of the U-233 containing particles from the U-235 particles.

The ORNL tentative flowsheet is schematised in figure 8.

Initial sawing, crushing and burning tests have been successfully performed with unirradiated fuels and long cooled Peach Buttom fuels irradiated up to 41,500 MVD/t.

Preliminary tests have indicated the effectiveness of the screening for separating U-233, U-235 and Al_2O_3 particles.

A roll grinder seems satisfactory for crushing particles coated with silicon carbide and allows an efficient exposure of the core to the thorex leach solution made of 13 M HNO₃, 0.05 M Al³⁺ and 0.05 M F⁻.

If losses of uranium and therium to the inner graphite particles are significant, burning of the carbon residue may be envisaged before leaching.
The thorium and uranium can be separated from each other by the thorex solvent extraction process which uses tributyl phosphate as extractant and has been applied many years ago at Oak Ridge National Laboratory.

An HGTR reprocessing plant requires a newly designed head-end system, but the solvent extraction equipment existing for light water reactor fuel processing can be adjusted for use with the thorex process.

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Reprocessing requires :	Solvent extraction allows ;
1. Remote handling behind shielding	- Compact, reliable mechanically simple equipment
2. Extremely great separation and recovery factors for the macro- components (U, Fu) from the microcomponents (7.F.)	- Extraction systems with high selectivity for the macro- components - Countercurrent techniques
3. minimisation of the waste volumes containing radioactive materials	- Use of salting agents and sol- vents readily recoverable and decontaminable in view of " recycling

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,	Decontamination Factor					
Cycle	Specific t	o the cycle	Over	all		
	U	Pu	U	Pu		
Codecontamination	$10^3 - 10^4$	$10^2 - 10^3$	$10^3 - 10^4$	$10^2 - 10^3$		
Partitioning	$50 - 10^2$	$10 - 10^2$	5.104 - 106	104 - 105		
V Purification	$10 - 10^{2}$	-	5.10 ⁶ - 10 ⁷			
Pu Purification	-	$10^3 - 10^4$		$10^7 - 10^8$		

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TABLE 2 - DECONTAMINATION FACTORS FROM GROSS FISSION PRODUCTS

TABLE 3 - SEGREGATION OF AREAS IN A RADIOCHEMICAL PLANT

Lone	Type of activity	Access requirements for personnel
Hot cells	Mechanical head-end, dissolvers, 1st extraction cycle, high level waste concentration and storage	No admittance, except after extensive decontamination
Warm gel ls	2nd and 3rd extraction cycles, solvent recovery and piping	idem
Intervention galeries	Sampling, pumps, pulsating devices, motors, access to cells	ndmittance under care- fully prescribed and monitored conditions
Service galeries	Control rcon, workshops, chemical make-up, fission product storage	Admittance at any time. Regular monitoring.

TABLE 4 - CHEMICAL COMPOSITION OF 304 L STAINLESS STEEL

Weight
< 0.03 %
≈18.5 %
9 - 10 %
remainder

TABLE 5 - REQUIRED DECONTAMINATION FACTORS FOR 30C T/YAND 1500 T/Y PLANTS TO ALLOW ROUTINE DISCHARGE OFVOLATILE AND GASEOUS FISSION PRODUCTS (1)

El em.	Ci/T (2)	Ci/y (3)	Annual plant capa- city above which decontamination is	Required D?s for plant capacity (tons/y)		
			(tons)	306	1500	
Kr ⁸⁵	10.600	10x10 ⁶	935	-	1.6	
Н _З	680	7x10 ⁶	10,300	-	-	
I131 I129	1.95 0.032	} 5 }	2.5	120	600	

- Data taken from D.E. Ferguson and R.E. Blanco General Survey of Reprocessing of Thermal Reactor Fuels. Proceedings A.N.S. Conference Nov. 10-15 1968, Washington D.C. p.p. 245-264.
- (2) Fuel burn-up 33,000 NVd/t; specific power 24.5 NV/t; cooling time 150 days
- (3) Annual allowable discharge rates from a 100 m high stack to achieve the maximum acceptable concentrations at 2 to 3 km distance in the direction of the prevailing winds and with average annual meteorological conditions at the CRNL site.

The maximum acceptable concentrations are 1/3 of the limit as stipulated by the Code of Federal Regulations, Title 10, Fart 20 "Standards for Protection Against Radiation.

TABLE 6 - MAIN HIGH LEVEL WASTE SOLIDIFICATION PHOCESSES

	ladioactive runs			Ohemaaal	Status of
Process and	Lab. Pilot		Product	Chemical	work
country	Scile	Scale		and tryes	(1972)
<u>Pot</u> <u>Calcination</u> U.S.4.	yes	yes	calcine	calcıum, suiphate	in progress
Spray Calcination					
U.S.A.	yes	yes	ceramic, glass	phosphate, borcphosphate	in progress
U.S.S.R.	?	?	calcine, glass	borosilicate	in progress
<u>Phosphate</u> glass U.S.A.	yes	yes	glass	phosphate	in progress
Fluid Bed					
U.S.A.	yes	yes	granules	none	industrial scale
U.S.S.R.	?	?	granules, "lass	borosilicate	in progress
Pot glass					
U.K. France	yes 11	yes 11	glass "	borosilicate "	done 1n progress PIVER
U.S.#.	none	none	semi-glass	phosphate, borophosphate	done
Spray glass F.R.G.	yes	none	glass	borosilicate	ın progress
Rotary kiln U.S.A.	none	none	homger	none	done
France	Ħ	17	glass	phosphosilicate borosilicate	ın progress

TABLE 7 - BREAKDOWN OF INVESTMENT COSTS

(INDICATIO , OF AVERAGE VALUE)

Item	% of investment cost
Site development	>
Process buildings	23
Fuel storage and handling	5
Chemical equipment	15
Instrumentation	10
Piping	20
Utilities	5
Ventilation	10
Analytical facility	7
	100

Engineering costs are equal to 20 to 25% of investment.

Waste interim storage, conditioning and perpetual care are not included.

TABLE 8 - VARIABLE OPERATING COSTS

TENTATIVE DISTRIBUTION

Item	\$/kg uranıum
Plant services, process chemicals	1.0
Low and intermediate level liquid waste (1)	1.0
High level waste (1)	2.5
Solid waste storage (2)	0.5

(1) assuming on site liquid storage

(2) assuming on site storage

)

TABLE 9- ILLUSTRATIVE REPROCESSING COSTS VERSUSDIFFERENT PLANT LOAD AND CAPACITY

	[
Plant capacity	1 T/d or 300 T/y			5 T/d or 1500 F/y			
Plant Load %	50	80	100	20	50	86	100
Financial charge (1)	90	56	45	85	34	21	17
Constant operating costs	22	17.5	11	16	6	4	3
Variable operating costs	5	5	5	5	5	5	5
Total costs	117	78.5	61	106	45	30	25

(in US \$/kg U, 1969 value)

(1) Taken yearly as 3C% of the investment, to cover : the cost of money (10%), the depreciation over 15 years on a linear basis, the return on capital.
Investment costs : 45 n \$ for 1 T/d, 85 No for 5 T/d.



FIG 1 OUTLINE OF REPROCESSING PROCESS FOR LWR FUELS



FIG 2 MECHANICAL TREATMENT AND DISSOLUTION OPERATIONS













FIG.7 ORIGIN OF RADIGACTIVE WASTES FROM REPROCESSING PLANTS



FIG 8 ORNL TENTATIVE CHEMICAL FLOWSHEET FOR HTGR FUELS

Implication of Reprocessing on Fabrication Techniques

YASUJI NAKAMURA SHINRO ABIKO

Abstract

In general, fuel material composition, cladding and other structural materials and fissile atom enrichment as well as fuel element and assembly configurations, fabrication processes and irradiation conditions have serious effects on the specific processes chosen for reprocessing, and on the other hand, species of heavy elements, chemical form and quality specifications of reprocessing products affect the type of refabrication technique, design of recycle fuel element and choice of unit processes in the refabrication.

Therefore, it is indeed necessary to consider in detail the requirements and impacts on fuel element design and fabrication technique from the standpoint of spent fuel reprocessing and also to endeavor to establish both efficient uses and economical fabrication techniques through technology development and industrialization, when a reactor system and its associated fuel cycle are planned for the national power generating system.

Depending on its final enrichment, the depleted uranium may be either discarded, or re-used as blending material or as source material for reenrichment. While plutonium may be used for recycling in thermal reactors, it can be used most effectively as the fast breeder reactor fuels. And whether it will be stored until the commercial fast breeder age is realized or used for thermal recycling depends on the level of technological developments and economic conditions in a country. However, the latter case seems to be more preferred one in many countries.

Following discussions are mainly concentrated on the fuel cycling in thermal power reactor systems of uranium and plutonium fuels with some references to our experiences on the plutonium fuels development as well as to general reviews of fuel recycling technology in advanced countries.

1. Introduction

The fact that the fuel element designs and fabrication techniques are quite different with various reactor systems requires a specific fuel cycle to be selected for a particular reactor system adopted, because the selection of a fuel element design involves not only the consideration of geometry, performance and fabricability of the fuel, but also the overall optimization of the fuel cycle cost, where the actual values of recovered fuel materials from reprocessing may also be taken into account.

In general, fuel material composition, cladding and other structural materials as well as fuel element or assembly configuration have serious effects on the specific processes chosen for reprocessing on one hand, and the chemical form and quality specifications of recovered materials affects the tactics of re-use and thus re-conversion and re-fabrication techniques on the other hand.

Therefore, it is indeed, necessary to consider in detail the requirements on fuel element design and fabrication technique from the standpoint of reprocessing, when the reactor system and the associated fuel cycle are selected for the national power system.

The aqueous reprocessing processes, which are the only ones to be applied on an industrial scale presently, seem to be generally versatile and capable of accomodating large variations in fuel composition without or with some additional installation of head-end processing equipments. But the economy of reprocessing is largely affected by its capacity scale and plant load factor, owing to its high capital investment.

Contrary to the reprocessing plant, the establishment of fuel fabrication plants does comparatively not require such a high capital investment as to make it difficult on a national level, except for remote refabrication facility of highly radioactive recycled fuel materials.

Depending on its final enrichment, the depleted uranium may be discarded or re-used as blending material or as source material of reenrichment. While plutonium may be used for recycling in thermal reactors either as the direct substitutes of enriched uranium fuel element needed for refueling for the same reactor or as the burning fuel in some particular reactors of the same or diffent type. Needless to say, plutonium is most effectively used as the fast breeder fuel, and might be stored until the commercial fast breeder age is realized

As for the U-233 and thorium fuel-cycling, the system is still in very premature stage compared with uranium and plutonium fuel-cycling, and seems to be developed independently.

Following discussions are mainly concentrated for the fuel-recycling in thermal power reactor systems of uranium and plutonium fuels.

2. Considerations on unit-process economy in fuel cycle stages.

2.1 General

The demand for nuclear fuel processing increases year by year with a gradual growth rate, while the process capacity is increased stepwisely as the result of additional installations. The unit-process cost is affected more or less by its plant capacity, reducing the unit cost with the increasing capacity. On the other hand, the gross economic demerit is much greater in case of low plant factor having a large process capacity.

The economic character of each processing in the nuclear fuel cycle industry is widely different. The unit cost of reprocessing involves big fractions of the fixed capital and operational costs, and minor fractions of variable operational costs such as chemicals and utilities. While the main fraction in the unit cost of fresh fuel fabrication is the procurement and processing of structural hardwares including inspection of them, and the working capital charges and costs related to fuel material loss and salvaging, and carrying charge interest are the other main fraction. In the case of plutonium fuel fabrication for thermal recycling, larger capital charge for the installations and higher maintenance labor costs are required than uranium fuel fabrication.

The gaseous diffusion enrichment has a quite different economic character where the predominant fraction of unit cost is attributed to the electric power, and the capital charge accounts for a medium fraction. And the plant operational flexibility seems to be limited in a single cascade plant, in order to meet the variable requirements on amount and enrichment of the product and feed materials.

2.2 Reprocessing

A well analysed information is available in "The Future of Reprocessing in Europe" [1].

The plant construction cost is referred in Fig. 1, where the curve A corresponds to the announced costs and the curve B contains all ancillary costs prior to start-up. The ancillary costs will be needed more and more in the future because the problems of environment protection become more serious.

The capital investment is so high that the unit process cost is largely affected by the plant load factor. The unit process cost is expressed by the following formula in the USAEC report WASH-1082, [2]:

$$\frac{1}{L} = \frac{37.2 + 11.2L^{0.33}}{L} M^{-0.67} + 0.15B$$

where M is the plant capacity in MT/Day, L is the load factor and B is the burnup level of the fuel in MWD/KgU.

In order to have a higher plant load factor, the plant design is desirable to have a high operational flexibility. Both the PWR and BWR fuels can be processed in a common reprocessing plant with some special installation of handling equipments and chopping-machine attachments for each type of fuel assembly, whose costs are estimated to be several tenths of million dollars. It is believed to be possible that the fuels containing burnable posion materials are processed without any change of the process flow sheet, but the metallic foreign material built-in the assembly might cause some mechanical troubles during the chopping and disolving operations. Some advanced fuel design of light water reactors will increase difficulties in accountancy of plutonium because of increased calculational uncertainty. The fuel element discharged from the HWR's may be processed without serious technical difficulties. In order to accommodate for the capability of processing of the HTR uranium fuel and the FBR fuel in a common Purex plant, some additional installations to the head-end processing are needed, whose costs are estimated as several million dollars. Irradiated thorium fuel containing U-233 would be processed in a separate facility because of the cross contamination of uranium.

2.3 Fuel fabrication

A comprehensive description on the thermal reactor fuel manufacture in the United States was presented in the last Geneva Conference, (3) in which the calculated fabrication costs of LWR fuel by means of the FABCØST 9 computer code are shown in Fig. 2. Projected fabrication costs in the US economy were reported in the WASH-1082 as following,

$$\frac{1}{1000} = 31.6M^{-0.425}/L + 28.7M^{-0.097}$$
 for BWR
= 40.0M^{-0.414}/L + 30.2M^{-0.097} for PWR

The BNWL-273 is a very useful information to understand the fuel fabrication cost-comp nents and their differences in cases of U 235 enriched fuel, low exposure plutonium enriched facel, and high exposure plutonium enriched fuel for which automatic semi-remote fabrication equipments and local gamma shielding are necessary. [4]

3. Re-use of the recovered fuel material and the associated technical problems

Unless the re-use of recovered fuel materials is economically favorable, there will be no incentive to reprocess the spent fuels from power reactors. Since the industry for nuclear fuel cycle services is still at a developing stage in a national system, or even in the international systems and, in addition, the nuclear fuel designs are still being advanced even for some proven type power reactors, it is extremely difficult to forcast the optimum fuel cycle management, which may vary from country to country, and from time to time owing to the economic, technical and geographical situations.

One of the serious problems is the availability of uranium enrichment services in the near future. Recently the USAEC announced that the tail assay in the diffusion plant will be changed in order to meet the forseen shortage of supply capability of enriched uranium, and published a report (WASH-1139 [5]).

3.1 Depleted uranium

The U-235 content in recovered uranium from the irradiated natural or very low enriched uranium is less than the natural abundance, so that they are almost useless in the near future. The depleted uranium from LWR and AGR has a higher U-235 content than the natural uranium, though it might be reduced below that level by the more advanced fuel designs and core fuel management techniques.

- Possible ways of re-use of these depleted uranium are,
- (1) re-enrichment following to re-conversion to UF6,
- (2) blending with higher enriched uranium, and
- (3) enriching with plutonium.

Considerations must be taken in the case (1) for the economic disadvantage of re-conversion and re-enrichment of smaller batches with varied U-235 contents, and for the technical penalties which come from the existence of U-236 that will be more accumulated in the product stream of the enriching cascade. In the case (2), the unit cost of blended uranium is somewhat higher than the ordinary cascade products because of the non-linear price curve of enriched uranium versus U-235 content. Considerations must be taken for the U-236 content and for the surplus of blended material whose amount is dependent on the U-235 content in the high enriched uranium used for blending. As for the case (3), an interesting investigation was reported in the IAEA panel on "Plutonium Recycling in Thermal Power Reactors" held in Vienna last year, $\{6\}$,

3.2 Plutonium

The plutonium isotopic compositions and their contents in the discharged fuel are widely varied with the neutron energy spectrum in the reactor core and the exposure level of the fuel. The recovery cost of plutonium from the irradiated natural or very low enriched uranium is higher than that from the irradiated LWR uranium fuels, because of the fact that the plutonium content is lower and all the processing expenses must be practically borne by the plutonium in the former case. [7]

Plutonium has a higher value as a fast breeder fuel than a thermal power reactor fuel. But the annual production rate of plutonium will by far exceed over the demand for the fast breeder development in the coming ten or fifteen years. Stock-piling of the plutonium causes some economic problems: hoarding of valuable material, the capital charge interest for its potential value as well as the actual expenses for recovery and storage, and the deterioration due to the decay of fissile Pu-241 into non-fissile Am-241. The safeguarding requires something an additional expense.

Plutonium is a highly toxic alpha-emitter and the dose rates of gamma-rays and spontaneous-fission-neutrons are significant in case of recycled plutonium owing to the higher contents of heavier isotopes. The criticality mass is so much smaller than U-235 that the accountability controls and the nuclear safety considerations ane very important in the plutonium handling operations. Alpha-tight glove-boxes with local gamma shielding equipped with mechanized or automated processmachines are necessary for the plutonium fuel fabracation in large quantities. So the incremental fabrication cost of the plutonium fuel over that of the corresponding enriched uranium fuel is significant as shown in Fig. 2.

Theoretical feasibilities of the plutonium-recycling in AGR, HWR and LWR have been studied; however, the commercial utilization seems to be carried out only for LWR in the near future from the view points of economic feasibility and development stage of the associated technologies. Plutonium-recycling in large PWRs and BWRs has been demonstrated in the United States and Western Europe, and is being studied in Japan, too. Technical information exchanges and discussions have been we'l done in the LAEA sporsored technical panels. [8] A recent review on the plutonium fuel fabrication was reported in a journal. [9]

Present installations of plutonium fuel fabrication facilities are limited in their numbers and capacities, and the factual incremental fabrication cost seems to be higher than the forcast. Even if a large scale commercial plutonium recycling becomes in reality, the demands for plutonium fabrication are less than one fifth of those for uranium fuel, so the actual cost differences will be larger than the one calculated for the same production capacity. The actual plutonium fuel fabrication cost penalty will affect the plutonium prices in the free market if it should exist.

3.3 LWR fuel cycle

Future cost projections for each stage of the LWR fuel recycling are given in the WASH-1082. The projected data generally agree with the recent cost trends, one exception of which is the revision of uranium enrichment service charge of the USAEC.

An economic consideration of fuel-recycling based on these costs data and the material balance calculations was reported by the present author in the last IAEA Plutonium Recycle Panel. [10] Though the calculations did not include the transportation costs of fresh fuel material and the carrying charge interests, thereby it gave only the economic trends, the results seemed to be suggestive for the understanding of the economic implication of reprocessing and recovered-fuel recycling in the LWR system which would be taken place in the near future. The fact that a significant saving of the requirement for enrichment service is possible by the plutonium recycling, should be stressed.

As for the fabrication techniques of plutonium fuel for LWR recycling. almost same process-techniques as UO2 peilet fabrication are applied, that is, cold pressing, pre-sintering, sintering and centerless grinding if necessary. The differences are the powder preparation and its characterization. In the powder mechanical blending process, which is a predominant process used in various facilities, fine particles of PuO_2 is mechanically blended with UO_2 powder by means of ball mills or kneiders, followed by V-blender-mixing according to the requirement on plutonium homogeneity. The plutonium oxide is usually derived from oxalate precipitation by calcination at high temperature. As the PuO₂ forms a solid-solution with UO2 at the higher temperature while the fuel pellets are loaded in the reactor core, this process gives a satisfactory irradiation performances for thermal reactor conditions. unless the size of PuO2 particle agglomerates is not so big as a few hundred microns. The co-precipitation process in which the plutonium hydro-oxide precipitates together with the ammonium di-uranate from uranium-plutonium blended nitrate solution by the addition of ammonium hydroxide, gives a better homogeneity of plutonium in the finished pellets. However, the process seems to be a little more expensive than mechanically blending process, because the handling limit of plutonium in solution state is as low as 220 grams due to the criticality consideration.

Since the Pu-241 decays into Am-241 with a half-life of about 13 years, and the gamma emission from the daughter of Am-241 is fairly strong, and the thereby the external radiation dose increases as the elapse of time after the reprocessing, a chemical purfication process prior to the plutonium fuel fabrication is advisable.

The fuel rod fabrication process and final assembling are the same as those of enriched uranium fuel fabrication, though more careful operations are required for the decontamination and radiation monitoring.

The alternate techniques such as vibratory compaction and associated Sol-Gel processing have been developed, and seem to be economically promising. However, the potential cost savings by these processes for the thermal reactor fuel do not seem to justify the further development and expensive assessments including irradiation tests, from the view points of the total economy of fuel fabrication as well as the conservative attitude of the users. [11]

3.4 HWR fuel cycle

The Standard CANDU type heavy water reactors are fueled with natural uranium and the once-through use of the fuel has been projected. A Canadian report was presented in the IAEA Pu Recycle Panel in 1971 in which a fuel costs evaluation including plutonium recycling was made for the CANDU-PHW type reactor. [12] It was reported that the plutonium recycling in the system will be profitable if the plutonium cannot be sold at \$8/gram or more. As the only one valuable material in the reprocessed products is plutonium, some modifications of the process flow-sheet of reprocessing and waste management would be necessary. Since the fuel rod diameter is larger, its length is shorter and fuel assembly design is simpler than those of LWR fuel, the automated refabrication techniques would be possible to achieve lower fabrication cost expressed as \$/KgM.

A heavy water moderated and light water cooled power reactor system is being developed in several countries. This type of the reactor is said to have a high operation flexibility, that is, the reactor could be operated not only with slightly enriched uranium oxide fuel but also with uranium-plutonium mixed oxide fuel without any modification of fuel assembly design, giving almost optimum condition of power generating cost for both cases. Thus this reactor will present a favorable selection of fuels for users, from time to time, depending upon the economic situation. Should one does care to use or to feed natural uranium only, "Plutonium Self-Sustaining Cycle" would then be proposed, in which the natural uranium enriched with plutonium (recovered from its own spent fuel) is to be fueled. [13] As the average conversion ratio of HWR is around 0.7--0.8 for U-Pu cycle while that of LWR is 0.6 or less, the natural uranium consumption per unit power generated in LWR is 1.5 to 2 times as much as the one in HWR. The power generating cost would not be so affected by the price changes of uranium and enriched uranium. The fuel cycle cost in (NU+Pu) fuel system is affected by plutonium cost, enrichment and mixed oxide fabrication cost as shown in [13].

.5 HTR fuel cycle

Several investigations on the fuel cycling in thorium fueled HTRs were reported in the TAEA Symposium on Thorium Utilization in Power Reactors, Vienna, June 1965 [14], and the International Conference on Fuel Cycles of High Temperature Gas Cooled Reactors [15], [16]. The USAEC report WASH-1085 contains many comprehensive informations. [17]

As the coated-particle $(U, Th)C_2$ fuel has shown good irradiation behaviors, more HTRs are expected to be introduce in the nuclear power generating systems in future. Though many elaborate developmental works have been carried out on both reprocessing of the irradiated fuel and re-fabrication, there remain many technical aspects to be developed in these fields of the fuel cycle technologies and above all, no commercial facilities to process large quantities of the material are existing at present.

A special head-end processing prior to solvent extraction is necessary such as mechanical crushing and screening of irradiated fuel elements followed by burn-and-leaching.

Irradiated fissile particles contain residual U-235, bred U-233, remaining Th-232 and minor fractions of other heavy metal atoms and fission products. Irradiated fertile particles contain similar kinds of heavy elements except for U-235. The FORATOM report, referred above (1) says as follows. "The cost of such a facility capable of handling 2t/d of the HTR fuel may add 10-15% to the investment cost of a 5t/d LWR fuel reprocessing plant excluding the costs of development work. It is doubtful if the HTR fuel containing thorium should be processed in a plant normally handling uranium, becuase the crosscontamination of uranium isotopes will be hard to avoid. Other specific problems are due to the higher gamma radiation and longer decay time encountered in the thorium cycle." Irradiated thorium is highly radiactive owing to the co-existence of Th-228, Th-224 and Th-231. And the high energy alpha particles of the recycled U-233 and its decay products can produce significant fast neutron dose rates as a result of (α, n) threshold reactions with light elements. These high radioactivities make the re-fabrication necessarily to be performed remotely in a heavy shielding, so that the re-fabrication of recycled fuel in the Th-U cycle requires both more advanced technology and more expensive facilities than the refabrication of recycled fuel in the case of U-Pu cycle. However, owing to the high burn-up of virgin as well as recycled fuels, the fuel cycle costs in the U-Th cycle are expected to be reasonable.

An interesting analysis of the HTGR recycle economy as a function of the time when recycling is started in the projected growth of the reactor system in the United States is reported in a ORNL publication, [18]

4. Conclusion

Introduction of a large nuclear power generating capacity in a national system as well as the choice of the reactor types has a very close interrelation-ship with the technological development and industrialization of the associated nuclear fuel cycle on the national basis.

Since different reactor systems require different fuel element designs, if multiple reactor types are adopted in a national nuclear power system, it will be necessary to install multiple fuel fabrication plants in the country, though some flexibilities are possible for uranium oxide fuels fabrication for PWR, BWR and HWR capable of carrying out in a single plant.

Spent fuels reprocessing from multiple reactors will be carried out in a single plant with or without additional installations of head-end facility depending upon economic point of views. More detail investigations and development works seem to be necessary in order to decide whether the irradiated thorium bearing-fuels should be processed in a separate reprocessing facility or not.

Since the economy of spent fuel reprocessing largely depends on the scale of plant capacity and its load factor, the installation of reprocessing plant should be considered both nationally and internationally, based on the forcast of growth rate of nuclear power generation in the nation.

In the field of re-fabrication of recycle fuels, the plutoniumuranium mixed oxide fuels for LWR, HWR and AGR, if planned, can be handled in a same facility in the early stage of the development. The manufacturing of hardware components and final assembling may be conducted in the existing UO_2 fuel fabrication facility.

It seems to be unpractical that the mixed carbide fuel fabrication is conducted in a same facility used for the mixed oxide fabrication, because separate glove-boxes with inert gas atomosphere are necessary for the former operation.

The thorium-U-233 mixed oxide or mixed carbide fabrication must be done in a separate facility with the heavy shieldings and remote fabrication equipments.

Major questions to be answered in planning the extension of fuel cycle schemes as well as the development efforts of various reactor systems are as follows:

- (1) What is the proper timing for commercial construction of fuel fabrication plant, reprocessing plant, and refabrication facility?
- (2) Will the plant load factor be kept at sufficient high levels?
- (3) Can the quality of recycle fuel be consistently high for power reactor use under the commercial fuel warranty system?

- (4) How will the fuel value trends affect, and be affected by economic improvements in over-all recycle technology?
- (5) What will be the market situation of nuclear full supply and possible savings of resources and service capacities at the time considered?

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Fig. 2 Fabrication Cost for Light Water Reactor Fuel. (A/conE cohE 49/P/062)





Fig. 5 Costs of HTGR Recycle Economy as a Function of Time When Recycle is Started.



Fig.- 4 Fuel Cycle Cost vs Pu Concentration in Japanese HWR Concept

ECONOMIC CRITERIA FOR FUEL FABRICATION IN PAKISTAN

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ABSTRACT

A CANDU type pressurized heavy water reactor is in operation in Pakistan. In order to ensure a continuous supply of fuel without dependence on a foreign fuel manufacturer, a medium size fabrication plant is considered viable for Pakistan. It will initially meet the requirements of 137 MWe KANUPP reactor, and will develop potential to fulfil the future fuel needs of the country for its nuclear power programme. The technical, economic and social criteria for the establishment of a fabrication plant are discussed, with particular reference to Pakistan. The estimate of unit capital cost of a fabrication plant producing KANUPP type fuel in Pakistan, has been made. This amounts to \$ 104 per kilogram of natural uranium fuel. The production costs for the reference fuel have been calculated for different plant lives and at variable interest rates. For a plant life of 20 years, the production cost per kilogram of fuel will be \$ 52. In comparison, the cost of the imported fuel is considered to range between \$58-63 or even higher depending on the conditions attached to the purchase of the fuel. The economic appraisal thus shows a considerable profitability by producing fuel in Pakistan, with an additional benefit of savings in foreign exchange.

1. INTRODUCTION

This paper attempts to give national views about the importance, programmes and criteria for the establishment of a fuel fabrication plant with particular reference to the economic aspects of the project. Pakistan has recently entered the field of nuclear power production by putting up a 137 MWe pressurized heavy water reactor of CANDU type in Karachi. We further have a modest nuclear power programme of producing cver 1000 MWe by the end of the decade by installing a 500 MWe power plant in the northern areas of Pakistan, as well as a 400 MWe dual purpose plant on the Makran Coast.

In order to ensure a smooth running of the power programme and to avoid continuous foreign dependence, it is planned to establish different nuclear fuel cycle industries in Pakistan. The fuel fabrication plant has been chosen as the first industry out of the fuel cycle programme because,

- a) the fuel fabrication occupies an important position in a nuclear fuel cycle and the costs of fuel fabrication influence significantly the fuel cycle economics of the nuclear power plants.
- b) ensuring regular annual supply of fuel for Karachi Plant(KANUPP) is important, in order to avoid dependence on a foreign fuel supplier.

It has, therefore, been decided to set up a medium sized fuel fabrication plant of capacity less than 100 tons/year. This plant would initially be used exclusively for the manufacture of KANUPP fuel and later it could be adapted to other types of fuel for the reactors to be established in the country.

2. PROGRAMME AND CAPABILITIES

In the beginning, the natural uranium-zircaloy clad fuel for heavy water reactors will be manufactured under licence from some foreign fuel manufacturer and later on capabilities will be developed to fabricate other types of fuels. The KANUPP fuel bundle is of split spacer design as shown in figure I. It has already been adopted for replacement fuel

TABLE I-PARAMETERS OF THE KANUPP FUEL DESIGN

۵	Bundle	
	- Type of assembly	19Cylinderical element in circula array with brazed appendages
	- Dimensions	
	Length	495.30 mm
	0.D	81 . 30 mm
b	Fuel element	
	 Length of an element 	494.53mm
	- No. of pellets/element	23
С	Fuel	
	– Material	Natural UO ₂
	- Form	Sintered and ground pellets
	– Density	95-98% T.D.
	– Pellet Diameter (nominal)	14.236 mm
	– Hight / Diameter ratio	1. 5
	- Dishing	Both ends
d	Cladding	
	- Material	Zr - 4
	- Inside Dia. (nominal)	14. 380mm
	– Wall thickness (average)	0.419mm
	— Surface	Non-autoclaved
e	Fuel components	
	- Material (all components)	Zr - 4
	- Spacers	
	- No. per bundle	144
	- Bearing pads	
	– No per bundle	36

for NPD and Douglas Point, as well as for the initial charges for the Pickering reactors. It differs from the old generation wire-wrap bundle design. It is a nineteen element bundle with over-all dimensions of 81 mm in diameter and 495 mm long, which are supported by two end plates and separated by spacers with bearing pads on the outer periphery of the 12 outside elements. Some parameters of the KANUPP fuel design data are presented in table 1.

2.1 Method of Production

The technology involved in the manufacture of the KANUPP type fuel mentioned above can be divided into two broad categories and is briefly explained for recapitulation as follows :

- Pellet Fabrication
- Bundle Fabrication

2.1.1 <u>Pellet Fabrication</u>. The uranium dioxide powder is compacted to 'green' pellets by double pressing and granulation technique. The . pressed green compacts are sintered in hydrogen atmosphere. The sintered pellets, after being discharged from the sintering furnace, are ground to the required diameter in a centreless grinding operation, washed and dried for final inspection before loading.

2.1.2 Bundle Fabrication. The zircaloy fuel sheaths after degreasing are put to induction brazing. It consists of coating the zircaloy strips with beryllium vapours, and then blanking out the bearing pads and spacers out of the zircaloy strip. Bearing pads are tack welded to 12 elements. and the spacers to all the 19 elements. The elements are then brazed under vacuum and cooled. The tubes are cut to exact length and welded at its one end with prepared end-plugs by magnetic welding. The tubes are then loaded menually in the upright position and the second ends-plugs arc welded once again on the magnetic welding machine. The completed elements are placed in the bundle assembly fixture where the end-plates and the elements are assembled and the complete fixture is placed in the resistance welding machine on a pneumatic indexing table. The indexing table helps to locate each weld or element site under the welding electrode and also rotates the assembly on different pitch circles. After completing this operation on both end-plates, the bundle is subjected to final inspection.

2.2. Research & Development Efforts

In order to back up a successful commercial operation of this nature, and to carry out the developmental work, the Pakistan Atomic

Energy Commission (PAEC) has decided to set-up well established materials laboratories at its research Institute, PINSTECH. The laboratories related with fuels have already been set-up, with facilities for the treatment of yellow cake and manufacture of uranium dioxide, the pelletization and sintering of ceramic fuels, the welding and surface treatment operations, measurement of powder properties, and different facilities for the non-destructive and destructive testing of fuel clements etc. Efforts are being made to expand the scope of nondestructive testing facilities in order to be able to check up and control the specifications of the incoming materials and the produce. For the fuels development programme, the post-irradiction examination facilities are under construction and in due course of time more elaborate metallurgical research tools will be available in these laboratories.

3. CRITERIA FOR ESTABLISHMENT OF PLANTS

The national criteria of Pakistan for establishing a plant is perhaps not much different from the international criteria, except that certain factors are to be given more importance than in a developed country. The main factors taken into consideration are, the economic feasibility, technical know-how, the manpower availability, the resources, the availability of raw material and a sense of self reliance. The solution to most of these factors and a compromise to some extent with the remaining factors have led to the final decision of establishing a fuel fabrication plant in Pakistan.

3.1 Economic Feasibility

This factor is of prime importance internationally, and becomes a starting point for all other factors for consideration. However, the importance of 'foreign exchange' for a developing nation gives a different perspective to this factor. Contrary to the general principle of producing at a price competitive to the imported cost and thereby making a saving; a certain saving in foreign exchange is of significance even if the over-all production cost in terms of local currency is higher than the imported cost. The fuel fabrication plant has been found to be economically feasible in Pakistan and the detailed economics of the fabrication plant will be dealt in the later part of the paper.

3.2 Technical Know-how

The technical know-how would be achieved by fabricating the fuel under license from a competent foreign fabricator. In addition, as mentioned earlier the materials isboratories at PINSTECH will provide requisite base knowledge as well as a nucleus for know-how advancement.

3.3 Manpower

Pakistan Atomic Energy Commission has over the years made efforts to build a basic trained technical manpower, which would sustain the national power programme. In this regard, PAEC has been able to train over 500 scientists and engineers in different nuclear fields. In order to meet the particular requirements of the fuel cycle programme, a Nuclear Materials School has been considered to be established at PINSTECH for the fresh professionals and technicians coming out of the universities and plytechnic schools. The orientation courses in nuclear materials and practical training, will further help to build up teams who could take care of the different projects.

3.4 Resources

Even the economic feasibility has to be reviewed in the light of country's over-all national development programme, which considers the merits of a project in the light of the availability of foreign loans, assistance etc. The nature of the loans not only effects the economic feasibility due to different interest rates, but also limits the choice and manner of utilization of the loans in certain cases. Therefore, even the technical merits of a project have to be compromised at times, because of its being tied up with the source of aid/assistance.

3.5 Raw Material

The nature of a project depending on indigenous or imported raw material determines its continuous dependence on a foreign country. Hence a care has to be taken in assuring a continuous supply of raw material and the foreign exchange requirements thereof. For the fuel fabrication requirements, it is planned to import continuously the cladding material. However, for the fuel material encouraging deposits of uranium have been found in Dera Ghazi Khan district along the Indus River. The work is being carried out to determine the extent of the deposits and to exploit them. In the meantime the work

has been started to concentrate the local ore; a pilot plant of 1000 lbs/day capacity has been built at Lahore and the Yellow cake has successfully been produced therein. However, till such time that the indigenous uranium is available, the fuel material will be purchased from the international market, and the plans are in hand to assure its availability.

3.6 Self Reliance

Even in the case of some unfavourable economic aspects of a project, the factor of self reliance becomes worth consideration. The policies of a developing nation rely to a large extent on international politics and, therefore, self sufficiency to the greatest extent is desirable.

4. ECONOMIC ASPECTS OF FUEL FABRICATION

The economics of fuel fabrication in Pakistan has been studied for a medium size plant of capacity less than 100 tons/year based on concrete offers. The fabrication cost has been worked out on job shop basis for KANUPP type natural uranium fuel and is represented, in 1972 dollars.

4.1 Capital Cost

The unit capital cost of a plant in \$/kg U is represented in table 2. The Plant would cost \$ 104/kg U with 90 per cent in foreign exchange, showing a great deal of dependence on foreign loans and assistance. The unjor expenditure incurred is under the head of direct costs amounting to 87% of the total capital, of which the plant equipment end hardware is the most significant, being entirely in foreign exchange. The plant equipment head is inclusive of the cost of air conditioning equipment in addition to the normal process fabrication equipment. The climatic conditions of the site necessitate summer airconditioning, costing nearly 10% of the total equipment cost. The fee for technical know-how and consultation to the tune of 10%, mostly in foreign exchange, is also a significant expenditure. This adds a great strain on the total foreign exchange liabilities vis-à-vis the quantum of the information and the help extended in addition to the restrictions imposed on the utilization of the information for purposes other than specified.

Of the indirect costs, the interest on the capital during the construction period of little over 2 years is also 10% of the capital .

The interest rate of 6% on the foreign exchange and 8% on the local currency has been considered. In case softer long term foreign loans were available to the government, it would make a great difference on the fabrication cost due to lower capital cost as well as its amortization.

4.2 Annual Operating Cost

The annual operating cost of the fabrication plant in \$/kg U/year is given in table III. The total operating cost inclusive of the material cost is \$ 44.60 of which 82% is in foreign exchange.

The material cost constitutes 78% of the total operating cost, with 90% in foreign exchange and 10% for the gases etc. available in the local market. The calculations of the material cost have been based on UO₂ price of \$ 19/kg. The cost of all other material calculated is based on the actual plant experience of the licensor and is known in terms of \$/kg of UO₂. The electricity cost has been considered as 5 mills/unit.

The royalty fee is to be paid on a fixed production extended over 10 initial years of production. The royalty fee is nearly 8% of the total operating cost, entirely in foreign exchange and is the second largest cost factor after the materials cost.

4.3 Cost of Production

Different fuel production costs have been calculated by considering the amortization of the capital for plant life of 10-30 years and at different interest rates. The interest on the local currency component has been taken at a fixed rate of 8%, while the foreign exchange component has been discounted at 6% rate as is generally the case, to most optimistic rate of 2%. The results have been shown in table IV. The two extreme limits of fuel production cost including material are \$ 49.9 to 58.7/kg U, with a foreign exchange component of around 80%. This would readily indicate a foreign exchange saving of the order of 20% per kg of the fuel, even if the production cost would have been just competitive to the imported fuel. It may be noted that the net fabrication cost without material of the type of fuel under consideration varies between \$ 24 - \$ 15. This range of fabrication costs compares well with those reported (1), (2), (3) & (4).

'A comparison of the cost of the imported fuel with that manufactured in Pakistan is illustrated in figure 2. The cost of the foreign manufactured fuel is considered to be ranging between \$ 58 - 63/kg U.It is apparent that only at 10 years plant life the fuel fabricated in Pakistan is not competitive with the imported fuel at its minimum price

TABLE II-UNIT CAPITAL COST OF A FUEL FABRICATION PLANT IN PAKISTAN (\$/kg U)

ITEM	COST, \$	% F.E. COMPONENT	% OF TOTAL CAPITAL COST.
A. DIRECT COSTS			
1.PLANT EQUIPMENT, HARD WARE etc. AT PAKISTANI PORT.	6940	99.20	66.50
2. CONSTRUCTION AND SERVICES	2 30	4.40	2.20
3. INSTALLATION & COMMISSIONING	9 00	51 10	8.62
4-LICENCING AND CONSULTATION	10 70	9 3 40	10.30
SUB-TOTAL	91.00	91.00	87 62
B. INDIRECT COSTS			
5. CONTINGENCIES & OVERHEADS	225	66.5	2.28
6 INTEREST DURING CONSTRUCTON	10 43	86 5	10 - 10
SUB TOTAL	12 - 68	83	12. 38
TOTAL	104 08	90	100

TABLE	III – UNIT	ANNUAL	OPERATI	NG COS	ST OF	Α	FUEL
	FABF	RICATION	PLANT I	N PAKI	STAN		
		(\$ / kg \	J/ YEAR)			

NO	ITEM	COST, \$	% F.E.	% .OF TOTAL OPER. COST
ţ	MATERIAL COST (FO.B)	35.10	89.70	78 55
2	OPERATION AND MAINTENANCE	3.00	50 00	6.75
3	OVERHEADS	2 50		5.70
4	ELECTRIC POWER	0 50		1 . 15
5	ROYALTY (FOR 10 YEARS ONLY)	3 50	100 00	7.85
	TOTAL	44.60	82.00	100.00

TABLE (V-	PRODU	ICTION	COST	OF	NA1	URAL	URANIUM
		FUEL	MANU	FACTUR	ED	IN	PAKIS	TAN

PLANT LIFE,	ANNUAL OPER- -ATING COST \$/KgU		ANNUAL OPER- AMORTIZATION OF CAPITAL,		FUEL COST, \$/KgU							
				F.E		LOCAL		INTER	EST RA	NTE, %		
VEADE			INTEREST, RATE,		%	2		4		6		
TEARS	F.E	LOCAL	2	4	6	8	TOTAL	F.E %	TOTAL	F.E %	TOTAL	F.E %
10	36.50	8.10	10.5	11.6	12.7	1.50	56.50	82.6	57.60	83.5	58.70	84.0
15	35.30	8.10	7.25	8.4	9.7	1.16	51.81	82.4	52.96	82.8	54.26	83.Q
20	34.75	8.10 .	5.75	6.9	8.2	1.00	49.60	81.8	50.75	82.2	52.05	82.5
25	34.40	8.10	4.80	6.0	7.3	0.90	48.20	81.4	49.40	81 .8	50.70	82.2
30	34 15	8.10	4.15	5.4	5.8	0.85	.47.25	80.9	48.50	81.5	49.90	61.9

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F.E. Foreign exchange component # The decrease in the annual cost is due to the royalty fee which is to be paid for initial 10 years only

TABLE	V -	UNIT N	ET P	RESENT	r i	WORTH	OF	FUEL
		FABRIC	ATION	PLANT	IN	PAKISTA	٨N	
			(\$	/ kg U)				

Time, Years	Capital Investment	Annual Operating cost	Net Production value	Gross Production value	Present value	Net Present value(NPV)	N P V/ Year
0	104.08		-104.08		-114.51		
1-10		44 . 60	+ 14 . 10	58.70	+ 108.40	- 6.11	0.60
1-20		42.85	+ 9.20	52.05	+ 175.90	+61.40	3 .07
1-30		42 25	+7.65	49.90	+ 329.00	+214.49	7.15
of \$ 58/kg U and considering 6% rate of interest on the foreign exchange. At plant lives beyond that period, there is a net gain in fabricating fuel in Pakistan. The net profit per kg of the fuel is indicative from area between the curves and the shaded part on the figure.

When the demand arises, the plant is operated at higher capacity with increased shifts, it is estimated that the unit fabrication cost will further decrease by \$ 4/kg U.

4.4 Economic Appraisal and Evaluation

Different profitability criteria have been applied on the fuel fabrication economic aspects, in order to quantify the cost and benefits of the investment on the plant from the point of view of the private returns.

4.4.1 <u>Financial Profitability.</u> At market price basis, as given in table V, the present worth of all costs-excluding interest and depreciation discounted at the minimum required rate of return of 8% for the local component of the capital cost and 6% on the foreign exchange component, work out to be \$ 114.5/kg U at the period of completion of the plant. Correspondingly the present worth of the benefits (revenue) are estimated to be \$ 108.4, \$ 174.0 and \$ 329.0/kg U for the periods of 10,20 and 30 years of plant life respectively. This implies that in 10 years the benefits fail to recover the invostment by \$0.61/kg U and hence does not render the project viable from economic point of view. However, for higher plant lives of 20 and 30 years, there is respectively a turn over of \$ 3.07/kg U and \$ 7.15/kg U over and above the investment, thus rendering the project viable. These findings conform to those indicated earlier in figure 2.

4.4.2 <u>Social Profitability</u>. In order to estimate the social profitability, the adjustments are made to both the cost and benefits streams, according to certain criteria of shadowing and increasing the prices as prevalent in the country. For estimating the present worth of social costs and social benefits, usually a higher rate of interest of 12% on the local investment is taken. It is revealed that the present worth of social costs exceed the present worth of social benefits by a substantial margin if the plant life is taken as 10 years. Whereas, for 20 and 30 years of plant life, the present worth of social benefits exceed the social cost, thereby making the project viable for the two later cases even taking into account the social factors.

4.4.3 <u>Rupees Cost Per Dollar Earned</u>. This criteria has been applied in order to give an idea of the measure of the efficiency in resources¹use per unit of foreign exchange earned or soved. The local currency components of the fuel production cost as indicated in table IV are estimated at rupees 102.3, 100.2 and 100.1/kg U at the rate of rupees 11.0 to one dollar at plant life of 10,20 and 30 years respectively. The corresponding foreign exchange component of costs of production are \$ 49.3, 42.94 and 40.8. Therefore, the rupee cost of each dollar earned, considering the cost of imported fuel as \$ 58/kg U, is as follows :

Plant Life	10	20	30
Rupee Cost per Dollar Rs/\$	11.80	6.65	5.85

This implies that & 11.8 are employed to earn one dollar of foreign exchange as against official rate of & 11.0/\$ when the plant life is taken as 10 years. Clearly, the domestic resources employed per unit of foreign exchange reflects uneconomic trade off between the two. On the other hand the trade off between rupee and dollar becomes economic when a higher plant life is taken.

5. CONCLUSIONS

In order to manufacture fuel for KANUPP and later to meet the fuel demands of increased power programme of the country, it is desirable to establish a medium sized fuel fabrication plant in Pakistan. The economic aspects of such a fuel fabrication plant reveal that by spending \$ 104.08/kg of fuel and with an operating cost of \$ 44.6/kg U, the project is feasible, if the plant life is more than 10 years. The establishment of such a plant in the country would result in a substantial saving of foreign exchange in addition to attaining self sufficiency.

ACKNOWLEDGFMENT

The authors would like to acknowledge Dr. M. Yunus, Head, Nuclear Materials Division, for his assistance and useful suggestions in the preparation of this paper. (1) Hulka, I., "Some Economic Aspects of Fuel Cycle in the CSSR." (Proc.Sym.Gottwaldov, 1968), IAEA Vienna (1968)55.

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Fig.2. Dependence Of Fuel Cost On Plant Life At Different Interest Rates On Foreign Exchange

A. Le Duigou

ABSTRACT

The French Nuclear fuel manufacture developed according to the needs of the national programme and, up to now, has been the responsibility of the Commissariat à l'Energie Atomique (C.E.A.).

The C.E.A. continuous policy consists of three phases:

- 1. Design and development, eventually first fabrications within the C.E.A.
- 2. Entrusting one private firm with the first industrial production together with part of R. and D.
- 3. Call for competition of another private firm when the need becomes large enough.

Gas-Graphite Reactors

Manufacture began in 1955, in Annecy, where the works of S.I.C.N. reached progressively a maximum output of 1200 t/y of metallic natural uranium. S.I.C.N. has also a research and production centre in Veurey near Grenoble.

C.E.R.C.A., choosen to compete in 1961, established a research and production complex in Romans-sur-Isère. The capacity of the metallic fuel workshop has grown up to about 800 t/y of uranium.

The effects of that commetition were very profitable (see Table II: fuel failures).

The decision to cut back the reactor programme decreased the fuel requirements and unfortunately compelled us to come back to the second phase with one single manufacturer since the beginning of 1972.

Heavy Water Reactors

Not very significant, fuel manufacture is performed by C.E.R.C.A. in Romans.

Fast Breeder Reactors

The production of mixed U.Pu fuel is still in the hands of C.E.A. (Cadarache Centre). Conventional assemblies (fertile blankets and shields) are made by both industrial firms.

Light Water Reactors

The decision favouring this type is recent enough in France; the organization in fuel manufacture is not yet established and is still under negotiations.

Through its experience in submarine reactors fuel, in other lines, in fuel management, in quality inspection on new and spent fuel, C.E.A. is ready to display a consistent part, whatever scheme shall be the final one.

RESUME

L'industrie française des combustibles nucléaires s'est développée en fonction des besoins du programme national et relève jusqu'à présent de la responsabilité du Commissariat à l'énergie atomique (C.E.A.).

La politique suivie par le C.E.A. comporte trois phases :

- 1. Le C.E.A. procède seul aux études techniques et théoriques et réalise éventuellement les premiers éléments;
- 2. Il confie à une seule société privée les premières fabrications industrielles, ainsi qu'une partie de la recherche et du développement;
- 3. Il accepte la concurrence d'une deuxième société privée lorsque les besoins prennent une ampleur suffisante.

Filière graphite-gaz

La fabrication a commencé en 1955, à Annecy où la production de la Société industrielle de combustible nucléaire (S.I.C.N.) a progressivement atteint une capacité maximale de 1200 t d'uranium métal par an. La S.I.C.N. dispose également d'un centre de recherche et de production à Veurey, près de Grenoble.

La Compagnie pour l'étude et la réalisation de combustibles atomiques (C.E.R.C.A.), admise en 1961 à entrer en concurrence, a créé un ensemble production-recherche à Romans-sur-Isère. La capacité maximale de l'atelier de production des combustibles graphite-gaz s'est progressivement élevée jusqu'à 800 t d'uranium par an.

Ce régime de concurrence a donné des résultats remarquables (voir tableau II intitulé "ruptures de gaines").

La décision de restreindre le programme des réacteurs a diminué les besoins en combustibles et a malheureusement contraint le C.E.A. à revenir dès le début de 1972 à la deuxième phase, c'est-à-dire à la collaboration avec un seul fabricant.

Filière à eau lourde

Cette filière n'est pas très importante; la fabrication des combustibles est assurée par l'usine de Romans de la C.E.R.C.A.

Filière à neutrons rapides

La fabrication des combustibles mixtes d'uranium-plutonium est toujours assurée par le C.E.A. (au Centre de Cadarache). Les éléments classiques (couvertures fertiles et blindages) sont produits par les deux sociétés industrielles déjà mentionnées.

Filières à eau naturelle

La décision d'orienter le programme français vers ces filières a été prise à une date assez récente; l'organisation des fabrications de combustibles n'est pas encore clarifiée et fait l'objet de négociations.

Fort de son expérience en matière de combustibles pour sous-marins atomiques, d'autres réalisations, de gestion des combustibles et de contrôle de la qualité des combustibles nouveaux et usés, le C.E.A. se prépare à jouer un rôle important, quel que soit le schéma industriel qui sera finalement retenu.

INTRODUCTION

L'industrie française des combustibles nucléaires est née et s'est développée en fonction des besoins propres du programme atomique national. Comme on le sait, ce programme a, dans sa première phase, été axé vers l'utilisation de l'uranium naturel dans les réacteurs de la filière "graphite-gaz".

Le Commissariat à l'Energie Atomique a toujours été responsable pour cette filière de la conception, du développement et de la production de tous les combustibles.

Simultanément, il s'est préoccupé du développement d'autres filières : HWR, HTR, surrégénérateurs. Enfin il assure pour la Marine Nationale la réalisation et le combustible des réacteurs de propulsion nucléaire des sous-marins (type PWR).

Le tableau I donne la situation actuelle du programme françai et de ses besoins en combustibles.

S'y ajoutent les réacteurs électrogènes à eau ordinaire, totalement français ou en participation multinationale. Nous y reviendrons.

POLITIQUE SUIVIE PAR LE C.E.A.

Que ce soit pour la production d'uranium ou pour la fabrication des éléments combustibles, la politique générale du C.E.A. a été et reste la suivante : dans un premier temps, étudier les procédés de base, puis, dans un deuxième temps, transférer à l'industrie pri ée l développement et l'exploitation industrielle de ces procédés. On peut ainsi distinguer schématiquement trois phases dans le développement d'une fabrication donnée :

<u>l ère phase</u> : le C.E.A. étudie <u>seul</u> les procédés de base et, éventuellement, réalise seul les premiers éléments.

<u>2 ème phase</u>: Le C.E.A. confie à <u>une</u> société privée d'une part les premières fabrications industrielles, d'autre part une partie du développement des procédés utilisés. Cette part de développement, qui complète les études que le C.E.A. continue à poursuivre dans ses propres établissements, est faite sous contrat d'étude de telle sorte que les résultats obtenus soient la propriété du C.E.A. Un groupe de travail assure la coordination indispensable entre fabrication, études C.E.A., études sous contrat.

<u>3 ème phase</u> : Lorsque les fabrications atteignent un volume et un chiffre d'affaires suffisants, le C.E.A. accepte la concurrence. Il confie alors en général à une deuxième société privée une partie de ses besoins, en la faisant bénéficier comme la première, du résultat des études qui lui appartiennent.

FILIERE GRAPHITE-GAZ

C'est en 1 955 que le C.E.A. a recherché le support de l'Industrie privée pour la fabrication du coeur de Gl, premier réacteur plutonigène de Marcoule, aujourd'hui déclassé.

C'est la Société Alsacienne de Constructions Mécaniques, devenue en 1 957 la Société Industrielle de Combustible Nucléaire (S.I.C.N.), qui s'est lancée dans cette opération en s'implantant dans une usine existante de mécanique et d'électricité à Annecy (Haute-Savoie).

Cette unité de production s'est développée jusqu'à atteindre une capacité maximum de l 200 T/an d'uranium métal. En même temps la SICN a créé à Veurey près de Grenoble un centre de recherches important, dont l'atelier pilote s'est orienté progressivement vers certaines fabrications de séries d'importance modeste. En 1 961, le C.E.A. a estimé que le programme français autorisait le passage à la 3 ème phase et a suscité la concurrence de la Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques (C.E.R.C.A.) qui produisait déjà pour le C.E.A. des combustibles à plaques pour les piles type M.T.R. Elle décida de créer un ensemble important (productions et recherches) à Romans-sur-Isère (Drôme).

La capacité maximale de l'atelier de production des combustibles graphite-gaz a cru progressivement jusqu'à 800 T/ an environ de métal.

Ce régime de concurrence s'est maintenu depuis 1 963, année des premières fabrications significatives de C.E.R.C.A. jusqu'en 1 971. Les résultats en ont été remarquables.

L'émulation technico-commerciale conduisit les fabricants à rechercher, seuls ou en collaboration avec le C.E.A., tout ce qui permettait d'alléger les conditions de production :

- adaptation des équipements (exemple des fours de fusion).
- suggestions d'aménagements des spécifications.
- gestion plus stricte des matières ou des demi-produits de grande valeur.
- standardisation et automatisation.
- levée progressive de certaines sujétions édictées à l'origine dans un souci de sécurité.
- etc...

Les améliorations ainsi obtenues ne l'ont jamais été au détriment de la qualité, ainsi qu'en témoigne le faible nombre de ruptures de gaine constatées dans les réacteurs électrogènes (cf. tableau 2).

La décision formelle de l'abandon de la filière graphite-gaz annoncée fin l 969, devait mettre un terme à cette période. Une fois les premiers coeurs achevés (SL 2 - Bugey I - Vandellos) le volume de travail en régime ne pouvait justifier le maintien de deux producteurs, essentiellement parce qu'il s'agit d'une industrie assez lourde, devant satisfaire à des exigences spéciales de propreté, de protection, etc...

Aussi, depuis le début de la présente année, un seul fournisseur assure le façonnage des cartouches de la filière graphite-gaz. Dans ce cas particulier, la conjoncture a amené exceptionnellement le C.E.A. a rétrograder de la 3 ème phase à la 2 ème phase exposées au paragraphe précédent.

EVOLUTION DES PRIX DE REVIENT

Cette évolution a été très favorable depuis l'origine des fabrications pour EDF, et la décision que nous venons d'évoquer, aussi délicate et regrettable qu'elle ait été à certains titres, pouvait seule en conserver le bénéfice.

Elle est concrétisée par le tableau N° 3 qui retrace, en francs courants, la variation du coût (hors uranium métal) de l'élément combustible EDF type 3 depuis 1 963. Cet élément alimente indifféremment les réacteurs Chinon 2 et 3, Saint-Laurent 1 et 2 et Vandellos.

Nous avons fait figurer sur ce tableau un graphique (en pointillé) qui retrace pour la même période l'évolution relative des tonnages totaux produits pour la filière. Ce dernier graphique, étant tracé en ordonnées décroissantes, permet immédiatement de juger de l'importance de l'effet de masse. On constate en effet, pour ce qui est du façonnage proprement dit, que les frais fixes qui représentaient à l'origine plus des deux tiers des charges des usines en constituent encore environ la moitié. En ce qui concerne les matériaux de structures (gaines, bouchons, chemises de graphite) cet effet est beaucoup moins sensible car leur élaboration s'effectue dans des ensembles dont l'activité globale est très grande par rapport à cette activité propre (frais fixes de l'ordre de 10 à 15%).

Il n'en reste pas moins que les effets concurrentiels, joints aux allègements des spécifications, ont été un facteur important de baisse des coûts. Il faut préciser en effet que l'évolution relative des prix est donnée en francs courants : or dans le même laps de temps, la variation des conditions économiques aurait conduit, toutes choses égales par ailleurs, à les accroître d'environ 60 %.

CRITERES D'IMPLANTATION DES USINES

Nous avons vu que pour Annecy, la motivation du fondateur à l'origine était de faire profiter les nouveaux ateliers des supports général et technique d'une installation existante.

Le site de Veurey a été choisi pour sa position au bord de l'Isère et surtout la proximité de Grenoble : zone industrielle, à forte densité de population, et scientifique avec le Centre d'Etudes Nucléaires, une Université importante et plusieurs grandes écoles techniques.

La C.E.R.C.A. a retenu Romans-sur-Isère pour son extension et sa décentralisation (sa première usine est à Bonneuil en région parisienne). Romans se trouve à proximité des Centres Nucléaires de Grenoble, Marcoule, Cadarache et du complexe de Pierrelatte. L'installation a été édifiée sur un terrain en bordure de l'Isère, mais non inondable, et est reliée à une station hydroélectrique très proche et à un réseau d'effluents se rejetant dans l'Isère en aval de la ville pour plus de sécurité. Enfin, une main-d'oeuvre courante relativement abondante a permis de ne déplacer que les techniciens qualifiés.

En règle générale, dans les limites de la France Métropolitaine, il n'y a pas de graves problèmes d'approvisionnement en eau ou en énergie électrique. Le coût des transports reste faible dans le prix global des éléments combustibles neufs ; la provimité des Centrer d'Etudes Nucléaires est plutôt recherchée pour faciliter les échanges scientifiques et techniques. La main-d'oeuvre peut poser des problèmes assez souvent catégoriels.

FILIERE A EAU LOURDE

Les fabrications de combustibles pour cette filière ont été jusqu'à ce jour assez peu significatives, se bornant à l'approvisionne ment du réacteur EL 4, en plus des piles de recherche du C.E.A.

Elles sont assurées jusqu'à présent par l'usine de Romans de la C.E.R.C.A. ; un appel à la concurrence n'est pas exclu.

FILIERE A NEUTRONS RAPIDES

Compte tenu des sujétions particulières, la fabrication des combustibles contenant de l'oxyde de plutonium (besoins de Rapsodie et ceux du coeur initial de Phénix, dont la réalisation est en voie d'achèvement) est assurée dans les ateliers spécialisés du C.E.A. à Cadarache. Des équipes extérieures ont déjà participé aux travaux de ces ateliers ; le transfert de ce type de production au secteur privé est donc envisagé, mais l'échéance de ce transfert dépendra naturellement de celle de l'avènement des réacteurs surrégénérateurs au stade industriel.

Par contre, les éléments plus conventionnels : couvertures fertiles, protection neutroniques etc... et même des aiguilles fissiles à l'oxyde d'uranium enrichi, sont produits depuis l'origine par les sociétés privées (S.I.C.N. à Veurey et C.E.R.C.A. à Romans) pour le compte du C.E.A.

REACTEURS POUR SOUS-MARINS

Leurs éléments combustibles sont conçus et développés au C.E.A. La fabrication fait cependant intervenir l'industrie privée dans une certaine mesure.

FILIERES A EAU NATURELLE

Pour tout ce qui précède, le C.E.A. a jusqu'à ce jour conservé la maitrise d'oeuvre.

La décision ayant été prise assez récemment d'orienter le programme français des prochaines années vers les filières à eau légère, l'organisation des fabrications de combustibles pour ces réacteurs n'est pas encore clarifiée et fait l'objet de réflexions et négociations. Il n'est possible actuellement de faire aucune projection sur l'avenir.

La C.E.R.C.A. produit déjà des éléments de ce type pour le réacteur Franco-Belge de Chooz, sous licence Westinghouse, et a aussi fabriqué des aiguilles de combustible pour le navire marchand nucléaire allemand OTTO HAHN. Elle est responsable de la fabrication de la première charge du réacteur de Tihange et se prépare à la produire. Le Commissariat à l'Energie Atomique occupe une position qui s'appuie sur ses acquisitions dans plusieurs catégories d'activités:

- Le Prototype à Terre de Cadarache, décidé en 1 959, opérationnel en 1 964, a donné naissance aux propulseurs nucléaires de sous-marins à l'entière satisfaction des utilisateurs. Il a permis depuis d'effectuer des essais importants et couronnés de succès sur des combustibles adaptables aux réacteurs électronucléaires à eau. Il sera renforcé dans un proche avenir par la Chaufferie Avancée Prototype (C.A.P.) à double objectif civil et militaire qui permettra les tests les plus sévères sur des éléments des filières à eau naturelle, pratiquement en vraie grandeur, Ainsi sera obtenue, par le C.E.A. lui-même (ou par des irradiations délà très avancées dans des réacteurs extérieurs de recherche ou de puissance). la qualification des combustibles qu'il a développés et qu'il peut d'ores et déjà produire à une capacité significative. De plus, la capacité de l'Atclier de Technologie du Plutonium de Cadarache, se basant sur l'expérience des combustibles de Rapsodie et Phénix, peut s'appliquer à la production d'aiguilles pour le recyclage du Plutonium dans les réacteurs thermiques à eau ordinaire, recyclage dont l'intérêt est indéniable.
- L'expérience acquise par le C.E.A. pour la transmission vers l'industrie privée de ses connaissances en matière de réalisations de combustibles a été exposée ci-dessus au sujet des autres filières et son efficacité est démontrée. Il en est de même pour la gestion de ce genre de productions.
- Les dispositifs de contrêle de qualité dans tous les domaines sont parfaitement rodés au C.E.A. : inspections en usines, analyses chimiques et isotopiques, essais mécaniques, dynamiques ou thermiques, etc... Les Départements Scientifiques peuvent de plus prendre rapidement en mains tout problème nouveau. Cet ensemble s'applique aussi bien à la fabrication qu'au suivi d'utilisation grâce aux équipements et au personnel qualifié des Laboratoirs de Haute ou très Haute Activité implantés dans les différents centres.

C'est pourquoi le $C_{+}E_{+}A_{+}$, fort de son expérience et de son potentiel tant humain que matériel, se prépare à un rôle important, quel que soit le schéma industriel qui sera finalement retenu sur le plan national.

NOTA

Il n'a été traité, dans le présent exposé, que de la fabrication proprement dite des éléments combustibles.

En ce qui concerne les matières nucléaires, la politique du C.E.A. n'a pas été différente :

- <u>Uranium métal naturel</u> : l'élaboration des lingots a été mise au point au C.E.A., pratiquée en première phase à l'usine du Bouchet (ancienne poudrerie nationale), partagée ensuite avec la Société de Raffinage de l'Uranium exploitant l'usine de Malvési près de Narbonne, puis confiée entièrement à cette dernière lors de la chute des besoins. (La S.R.U est devenue partie de COMURHEX par fusion).
- Oxyde d'uranium frittable : La production, mise au point au C.E.A., est partagée entre le C.E.A. et la Société des Usines Chimiques de Pierrelatte (S.U.C.P.).
- Fabrication des pastilles frittées : la mise au point a été faite en commun avec la Compagnie Industrielle des Combustibles Atomiques frittés (CICAF), centre de recherches de Corbeville près de Saclay. La production est actuellement assurée par la Société Française d'Fléments catalytiques (S.F.E.C.) dans l'usine de Bollène (voisine du complexe de Pierrelatte).

En ce qui concerne les matériaux de structure (graphite nucléaire, gaines et accessoires en alliages de Magnésium ou de Zirconium et en aciers inoxydables, ...) nous ne croyons pas qu'un exposé à ce sujet présente ici de l'intérêt ; la forte position française est connue dans les domaines du zirconium et de ses alliages ainsi que du graphite et du magnésium nucléaires.

Tableau I

REACTFURS DF PUTSSANCE EN SERVICE EN FRANCE

			(GAS-0	RAPHITE)		
Nom Name	Lieu Location	Puissance élec- trique (nette) MV	Date de divergence Critical	Production principale	Consormation Annuelle (t.U) Yearly need	Observations
G 2 G 3	Marcoule CFA	40 40	1959 1959	Plutonium	Non disponible	
Chinor T	FDF Chinon	Bo	1962	Electricité	30	Arrêt définitif prévu
Chinon II	n	200	1964		60	
Chinon III		480	1966	Ħ	100	•
St Laurent I	FDF St-Laurent des Eaux	480	1969	W	100	
St Laurent II	••	515	1971	Ħ	100	
Pugey I	FDF St-Vulbas	540	1971	#	150	Combustible annulaire à double refroidissement -
Hifrensa	Vandellos (Fspagne)	515	1972		100	FDF : 25 % - Combustible fourni par le CFA
		II - FI	LIERF FAU LO	PDF-U02 PFU	FNRTCHT (HWR)	
τ <u>τ</u> μ	Brennilis	1 73	1 1966	Flectricité	1 5	
		III - <u>FI</u>	TTEPE A NEUT	RONS RAPIDES-	U02-PU02 (FBR)	
Phénix Papsodie	Marcoule Cadarache	250	1973 1967	Flectricité Néant	17	Fxpérimental
		IV - PF	ACTEURS DE SO	DUS-MARINS-U	FNRICHT - (PWR)	- SUBMARINES
P.A.T Prot C.A.P Chau "LF FFDOUTABL "LF TFPRIPLE" "LF FOUDROYAN "L'INDOMPTABL "LE TONNANT"	otype à terre à liferie avancée p E ⁿ - Divergence - Divergence en E ⁿ - Divergence E ⁿ - En construc - En projet	Cadarache - Diverge rototype à Cadarach en 1969 1971 en 1972 tion	nce en 1964 le - Divergen	ce prévue en	1975	

I - FILIFRY GRAPHITF-GAZ-URANIUM NATUREL

FILIERE GRAPHITE-GAZ

RUPTURES DE GAINES (FUEL FAILURES)

(Cartouches nominales)

(nominal elements)

REACTEURS	СН Ј	CH 5	СН 3	SL 1	SL 2	(VANDELLOS)	(BUGEY)	TOTAL
Nombre de cartouches chargées 1 Loaded elements	47 000	79 000	79 000	52 500	43 000	(40 000)	(12 800)	300 000
Nombre total de ruptures - Failures (2)	12	2	10	10	٥			34
Nombre de ruptures dues à incidents d'exploi- tation 3 Running accidents Différence 4	6 6	5 0	1 9	8	0 0			15 19
Intervalle moyen entre ruptures (en cartouches) 5 = 1/4 elements	7 800	39 500	8 800	26 700				15 800
(1) JEPP cumulés 6	1 553	1 744	681	450	236	(40)	(20)	4 664
Intervalle moyen entre ruptures (en JEPP) (7) = (6) / (4)	259	872	76	225				245

(1) JEPP = jour équivalent de pleine puissance Equivalent day at nominal power

TABLEAU Nº III EVOLUTION RELATIVE DU COUT DU COMBUSTIBLE EDF TYPE 3 TONNAGE (HORS LINGOTS D'URANIUM, EN FRANCS COURANTS) En tireté : évolution relative du tonnage global de la filière. 1 <u>.</u> .::. ----...... _____ **.** i., •..: ---------**r** ! /-----٠... 601 1 Ī 1: ::: . ----. / 0,8 Ess. 0,75

3. Ŧ -. ---------------..... ·..... Ŧ 1 1..... -----÷.... N 45 in -----. N 1,00 0,6 ••• N. ·Ŀ 7 _____ ____ į., i -----..... <u>=--</u> ----------..... -**E** ----_____ : -:: -----PACONNAGE (y compris pertes et retraitements de d'uranium). déchets 4 ------••• F 0.4 :: . i. : · 1,25 ____ • • • • • 1::: .----------<u>i</u>::::: : ----..... ----ľ-<u>...</u> ... • . • • [---=-: ÷ **7**17 : • ... • :• -**;** ------Ţ -------------..... Ī 0;2 -•• **-**.... ----..... -----Ч.: ..**:** · ··· /----.---. : 1 -----Matériaux de structure ----------. B.... •••• • ÷..... : ::-:: • • • • • • ------··:: ---------. ---------------. ····· <u>-----</u> 1

07.09.72

1972

1970 1971

1967

1965 1966

• •

1968

·. . .

63 ...

1964

.....

Technoeconomic Prospect of fuel fabrication in support of Korean Nuclear Power Program

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Abstract

In order to discuss the technoeconomic feasibility of light water reactor fuel fabrication in Kores, the overall picture of Korean Nuclear Power Program up to the year 1990 along with the justification of nuclear power generation is presented. As of June, 1972, nuclear power generation cost was estimated to be 6.62 mills/Kwh in Korea whereas conventional oil fired thermal power generation cost was 8.05 mills/kwh.

Brief summary of R & D effort in support of fuel fabrication is made. The results of activated sinterring of UO₂ pellet by TiO₂ addition has a potential for commercial exploitation.

As a relevant data for nuclear fuel fabrication; the estimated capital investment, personnel requirement, technical training requirements and cost of labour in Kores are summarized.

Being a labour intensive nature of fuel fabrication process, this portion of fuel cycle industry can very well be commercialized in near future.

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In discussing the technoeconomic prospect of fuel fabrication in Korea, I would like to touch briefly on Korean Nuclear Power Program in order to provide a overall pictures of fuel fabrication as a part of fuel cycle in support of nuclear power program up to 1990.

Then, I will describe the role of Korean Atomic Energy Research Institute in related R & D effort. Finally, I will come to the technoeconomic prospect of fuel fabrication.

Korean Nuclear Power Program

Being a natural resource deficient country, Korea has to rely its major energy supply on import. The dependency on imported energy would become more pronounced in future eventhough there are signs of economically mineable oil, uranium and thorium deposit.

In view of the large amount of energy to be imported, the imported nuclear power has a definite advantage over the imported oil fired thermal power in following respects.

1. Imported nuclear power is cheaper than the imported oil fired thermal power. The unit power generating cost of 600 Mwe PWR(AORI Nuclear Power Plant) can be compared to the imported oil fired 600 Mwe thermal plant as shown in table 1.

Because of these advantages, the future demand increase on electricity would be likely to be met by mainly nuclear power. The demand in nuclear power and fuel fabrication requirements are as shown in table 2.

Table 1. Unit Cost and Power Generation Cost comparison between Nuclear Power Plant and Oil fired Thermal Power Plant.

.

	(as of June, 1972)				
Cost	Nuclear	Oil Fired			
Construction (\$/Kw)	308	181			
Fixed Charge (%)	10,16	10.16			
Load Factor (%)	80	80			
Generation (mills/Kwh)					
Fixed charge	4.46	2.62			
Fuel.	1.83	5.18			
Operation	0.3	0.25			
& Maintenance					
Nuclear Insurance	0.05				
Total(mills/Kwh)	6.64	8.05			

Remarks : annual saving of \$ 6 millions.

2. Ease of transportation and storage

3. Less air pollution and public hazard

4. Prospect of stable nuclear fuel supply

Table 2. Estimated demand of nuclear power and fuel fabrication requirements

Year	1975	1977	1980	1985	1990
Nuclear Power (Mwe)		600	1,200	3 , 800	5,800
Fuel Fabrication Requirements (M/T-u)	48	16	64	210	310

R & D effort in support of Fuel Fabrication

As shown in table 1, nuclear power generation cost is cheaper than the oil fired thermal power generation cost. Especially, the fuel cost has an outstanding advantage in nuclear power.

Because of Korean industrial capability, it would be some time before Korea will be capable of manufacturing the major hardware of nuclear power plant economically. On the other hand, the fabrication of fuel is estimated to be technically feasible at present state of art. Moreover, fuel fabrication is the most labor intensive portion of the fuel cycle. Therefore, this portion of the fuel cycle can well be industrialized in Korea.

With these philosophy behind, R & D program in fuel fabrication was started in 1968 at Korea Atomic Energy Research Institute. The related R & D programs can be grouped as follows.

- 1) Fuel Fabrication
- 2) Fuel Reprocessing
- 3) Fuel Cycle Analysis
- 4) Fuel Economic Analysis
- 5) Treatment of Fission Products Waste (Sr, Cs) by Ion-Exchange Process.
- 6) Crystal Imperfections in the Reduction Process of U_3O_8 by Electron Microscopy.

Since it would be time consuming to go into every details of related R & D programs, I will pick up few high-lights in fuel fabrication program. 1. Activated Sintering of UO_2 :

The mechanisms involved in activated sintering of UO_2 powder were studied by high temperature X-ray diffractometry. As shown in fig. 1, TiO₂ addition would give rise to the increase in lattice constant at high temperature. Hence, the self diffusion of UO_2 is expediated resulting in faster densifications.



By activated sintering of UO_2 green compact, the sintering temperature can be lowered appreciably resulting in not only the savings in power but maintenance cost too. While there is no sign of adverse effect of TiO₂ additions on mechanical properties of UO_2 pellet, the variations in thermal conductivity or melting point due to additives under irradiation conditions have to be determined further.

2. Encapsulation :

TIG welding chamber was designed and made by machine shop. When the material was chemically cleaned prior to the welding and the vacuum in welding chamber by pumping was good before flushing with helium and argon mixture gas, the quality of welding Zircaloy 4 end plug on tube was excellent.

3. Feasibility study on fuel fabrication pilot plant :

The preliminary feasibility study on fuel fabrication pilot plant was made with the limited amount of technical informations such as

WASH-1082 Current Status and Future Technical and Economic Potential of Light Water Reactors.
WASH-1099 Reactor Fuel Cycle G sts for Nuclear Power Evaluation. published by the Division of Reactor Development & Technology, USANC.

Technoeconomic Feasibility of Fuel Fabrication

Because of follow up research and the continued effort on development in fuel fabrication, the pelletization and encapsulation techniques are fairly well established in laboratory scale at Korea Atomic Energy Research Institute. The inspection, nondestructive testing and helium leak

detection facilities have to be set up further along the line with the fuel assembling engineering developments. In order to evaluate the fuel performance, the modest irradiation testing and post irradiation examination program is being planned mainly through technical cooperations with advanced countries. In this respect, the regional colaborations or technical assistance of IAEA would be most desirable. Along with the ceramic fuel development, metallic fuel fabrication techniques are being studied in order to set for the second nuclear power plant which is under considerations.

Therefore, no foreseeable major difficulties would exist in the technical aspects of fuel fabrication. However, the economic aspects of fuel fabrication have quite different picture. For this end, the plant scale factor, fabrication cost analysis and marketability have to be studied.

Since the domestic market is very limited as shown in table 2, the feasibility study on commercial fuel faorication plant would be prematured unless the marketability is assured in regional or world wide scale. Therefore, the commercial scale operation was excluded from the feasibility studies.

The preliminary study was limited to the fuel fabrication pilot plants to be located at the Atomic Amergy Research Institute. The following assumptions are made;

- UO₂ powder or uranium metal of nuclear fuel grade and zircaloy tuings and rods of appropriate dimensions are imported as a feed materials to the plant.
- Being a pilot plant ; the start up operation, tost fuel assembly and irradiation performance evaluation cost are included.

Because of the nature of plant operation, the optimum plant capacity and the economy of scale factor was not considered, but the required investment only. Since the pilot plant has to be operated by Korea Atomic Energy Research Institute, the operating cost was not included.

1. Required Investment :

The total investment for fuel fabrication pilot plant of 10-15 tons per annum is estimated as follows, as shown in table 3.

Table 3. Required Investment

Building Cost	\$ 900,000
	(260,000)
Equipment Cost	618,000
o Pelletization facility	124,000
	(120,000)
o Encapsulation and assembly facil	ity (340,000)
o Analytical, health physical	
& supporting facility	(154,000)
Start up operation and test fuel	
assembly cost	(491,000)
Irradiation performance evaluation	(208,000)
Total	\$2,217,000
	(1,573,000)

Note: Figures in parentheses are the investment required as the foreign currency. The pilot plant is for the oxide fuel with contact operation only. 2. Personnel requirement:

Because of the developmental work responsibility of the pilot plant, the priority was placed in research than the operating staff. For altogether, 12 graduate scientists and engineers, 19 technicians and 15 supporting workers would be employed.

3. Technical training requirements:

In order to operate the pilot plant and carry out the developmental work, all the graduate scientists and engineers have to be trained abroad. However, technicians and supporting workers can be trained at fuel fabrication laboratory of AERI prior to the operation start up of pilot plant.

4. Cost of labour:

The estimated wage rate of staff personnel and workers as of June, 1972 are as follows, as shown in table 4.

Concluding Remarks

We believe that our un'versities and Korra Atomic Energy Research Institute can train and provide the technical manpower required for the fuel fabrication. However, we do not have material testing reactor and post irradiation examination facilities. Therefore, the fuel irradiation and performance evalutions are the area where we would most like to have a tMechnical assistances from advanced countries or IALA on top of the other phases of fuel fabrication development.

Being a labour intensive nature of fuel fabrication processes, we feel that this portion of fuel cycle industry can very well be industrialized in near future.

Table 4. Wage rates

a. Engineers ; \$/month College graduates (manager-engineers)..... 500

b. Office workers ;

c. Laborers ;

Skilled and experienced laborers are readily available in Korea.

- (1) Skilled laborers (including semi-skilled laborers); High school graduates (ten-year experience).....145 High school graduates five-year experience)....100 High school graduates (three-year experience)....90 Junior high school graduates(five-year experience)....90 Junior high school graduates(three-year experience)....90 Primary school graduates (f² ve-year experience)....70 Primary school graduates (three-year experience).....70
- (2) Unskilled laborers ;

High school graduates5	0
Junior high school graduates	2
Primary school graduates	5
Female laborers2	7

Note : Most companies pay employees bonuses every four months that are equivalent to three months' salaries yearly.

Session E. Economic Criteria for Fuel Fabrication Prospects of Domestic Fabrication of Nuclear Fuel in Japan

T. Haginoya*

Abstract

Nuclear power generation in Japan will reach 60,000MWe by 1985 and 100,000MWe in 1990. In keeping with this, active measures must be introduced to ensure safety, preserve the environment, provide stable supplies and effective utilization of nuclear fuel and proper management of nuclear wastes.

For the present, light water reactors are the mainstay of nuclear power generation in Japan, their demand for enriched uranium is expected to be 5,000 tons-SWU in 1980 and 11,000 tons-SWU in 1990. Therefore uranium needed for this program will be 8,000 S/T U308 in 1980 rising to 15,000 S/T U308 in 1990, c: mulative requirement up to 1990 would be 170,000 S/T U308. The requirement for fuel fabrication will be 1,400 tons in 1980 and 2,800 tons in 1990. Plutonium production in Japan is estimated to be 13 tons cumulative in 1980 and dumulative amount of that would be 45 tons by 1985.

Environment and safety matters are great importance to develop and utilize nuclear energy, therefore Japan is exercising strict control and regulation of radiation, giving much greater consideration to such safety than to other branches of industry.

Japan Atomic Industrial Forum

Various industrial groups in Japan incorporated nuclear fuel manufacturing corporation either by themselves or jointly with foreign companies, as the results Japan has fabrication capacity of about 500 tons in this year and about 1,000 tons in 1974.

1. Introduction

According to the Long-Range Program which has recently revised by Japan Atomic Energy Commission. The basic policies of the program are as follow: (1) while strongly promoting Japan's own development efforts, this should be done in such a way as not to disrupt international collaboration; (2) more emphasis should be placed on basic research; (3) the implementation of the program should be carried out effectively and efficiently from an overall point of view.

Nuclear power supply will have to reach 60,000 MWe by 1985 and 100,000 MWe by 1990. In keeping with this, active measures must be introduced to ensure safety, preserve the environment, provide for stable supplies and effective utilization of nuclear fuel and proper management of nuclear wastes.

It is essential that stable supplies of nuclear fuel be secured and effectively used in order to achieve the scale-up envisaged in the nuclear power generation program.

The construction of conventional types of reactors must continue for the time being, but in order to solve the problem of stabilized fuel supplies and efficient utilization of fuel resources, appropriate new types of power reactors must be

developed to make use of the full merits of nuclear power generation. This means that the ATR and the FBR now being developed in Japan should come into commercial use between 1975 and 1984 (ATR), and between 1985 and 1994 (FRR).

The Advanced Thermal Reactor being developed in Japan is a heavy water-moderated and boiling light water-cooled type reactor. Since this type has a high neutron economy factor, its uranium consumption, especially consumption of enriched uranium, can be drastically reduced compared with the LWR.

For the present, however, the LWR will be the mainstay of nuclear power generation, and the demand for enriched uranium, is expected to reach 5,000 tons-SWU by 1980 and 11,000 tons-SWU by 1990.

Essential to the attainment of the foregoing, is the provision of a stabilized large volume supply of nuclear fuel and its effective use. Japan must secure uranium resources and enriched uranium, and, to meet the needs of the future, the fabrication of fuel and reprocessing of spent fuel must be done. This calls for the `stablishment of a complete nuclear fuel cycle to ensure economic feasibility and independence. In reaching the stage of a proper fuel cycle, private corporations, on principle, are expected to take the responsibility, but as nuclear energy is an important state responsibility, the Government has a big part to play in working for this end.

2. Fuel Supply

The uranium needed under the revised program will be 8,000 S/T annually by 1980 rising to 15,000 S/T by 1990. Japan has only poor uranium resources, and will have to

depend almost entirely on foreign sources. The electric utilities are expected to secure the uranium they need on short and long term purchase contracts. Long range, the ratio of imports from sources developed by Japan should be increased to about one third of the annual need. Prospecting and the development of uranium mines overseas must be strongly undertaken. The exploration being done by PNC need to be expanded and financial provisions made for private corporations engaged in prospecting, making available loans to be repaid when the projects are successful.

The existing supply capacity of the United States, now supplying enriched uranium to the western world on a commercial basis, will reach its limit about 1980.

Japan must therefore see that supplies of enriched uranium are available from the United States for power plants that will be commissioned up till 1980, and the necessary R&D must be done by Japan itself to make sure that domestic production can supply the further needs beyond 1980, as well as Japan participating in international enrichment projects.

Uranium	Requirement i	n	Japan	(Short	tons	U308)
	Annual		C	umulati	/e	
1975	4,000			16,000		
1980	8,000			48,000		
1985	12,000			99,000		
1990	15,000			170,000		

In nuclear fuel fabrication, domestic makers are consolidating their basis for local fabrication, but the foundation is not yet firm enough. Technological development must be further promoted. The requirement for fuel fabrication will be 1,400 tons in 1980, 2,800 tons in 1990. Conversion to uranium hexaflouride from natural or depleted uranium is performed abroad for the time being, however, our own capability shall be developed, in particular for the purpose of domestic production of enriched uranium in future.

Plutonium production in Japan is estimated to be 13 tons cumulative in 1980 and cumulative amount of that would be 45 tons by 1985, in contrary, its requirement up to 1985 would be only several tons for research purposes then there must be 40 tons or so excess plutonium by that time.

Plutonium can be most effectively used in the FBR, but pending its coming into commercial use, there are plans for Pu use as fuel in the LWR, as a means of reducing the consumption of uranium and enriched uranium. This is considered feasible in view of the cost of storing plutonium. The Government must take proper steps to help private corporations engaged in technological development on plutonium to use the facilities of PNC and JAERI.

Requirement for reprocessing will reach to 700 tons in 1980, 2,600 tons in 1990 in Japan.

For the reprocessing of spent fuel, PNC is building the first reprocessing plant, to be ready by 1974. The second and subsequent plants are expected to be built and operated by private corporations, on the principle that reprocessing of spent fuel is to be done in Japan.

But as a spent fuel reprocessing plant involves many as yet unsolved problems, the Government is required to adopt a proper siting policy and provide low-interest, longterm loans, as well as energetically promoting R & D on reducing to the lowest practicable level the ratic active nuclear wastes discharged into the environment.

3. Environment and Safety

Proper development and utilization of nuclear energy is possible only when radiation is safely controlled. Japan is so far exercising strict control and regulation of radiation, giving much greater consideration to such safety than to other branches of industry.

But large quantities of nuclear waste will certainly be produced when more nuclear energy for power and other large scale developments are carried out. The amount of radioactive matter emitted into the environment is expected to increase accordingly. Further, multi-lateral application of radiation, expected to expand greatly, will increase the possibility of more people and nuclear-energy handling staff being exposed to radiation.

To meet this situation, and to maintain the present high level of safety and safeguard factors, even greater efforts are needed and more consideration given to questions of safety and preservation of the environment. This requires proper steps on the siting of plants, safety measures covering all facilities and equipment, and radiation control is necessary; studies are needed on background radiation and its effects as well as on the

disposal of nuclear wastes.

On these questions of safety and preservation of the environment, private corporations engaged in development and utilization of nuclear energy are naturally called on to meet their full responsibilities. For its part, the Government must exercise strict control to ensure the safety to the people. Only by so doing, can the Government expect to develop and utilize nuclear energy properly and improve the people's welfare.

4. Requirements for Production Capacity and Investments of Nuclear Fuel Industry

The following tables have been calculated by Japan Atomic Industrial Forum in relation to "Nuclear Vision for the Year 2000".

	Conversion		Fab	rication	Reprocessing				
	(Ton-UO2/Yr)		Den	land	Requi Produc	red tion	(Ton-U/Yr)		
Year	Case I	Case II -(c)	Case I	Case II -(c)	Case I	Case II -(c)	Case 1	Case II -(c)	
1980	2,100	1,700	1,200	1,400	1,800	2,100	500	500	
1990	5,400	3,200	3,400 (100)	4,700 (100)	5,100 (150)	7,100 (150)	2,100 (40)	3,300 (40)	
2000	5,700	3,300	4,700 (700)	6,300 (700)	7,100 (1,100)	9,500 (1,100)	4,000 (450)	5,600 (450)	

Required Production Capacity of Nuclear Fuels

Note: (1) Figures in parentheses are cores for FBR (Pu-U ton/yr) and are not included in the major figure.

Investment Requirements

(in million dollars at 1970 price level)

	Year	1971	- 1981	1981 - 1990		1991 - 1		
		Case I	Case II -(c)	Case I	Case I ~(c)	Case I	Case II -(c)	- Nemarks
F	Conversion	25.0	16.7	47.2	28.8	47.2	28.8	Excluding plants
e	Fabrication	69.5	83.3	222	306	667	695	to be
s	Reprocessing	97.2	97.2	306	422	611	695	by PNC

Conversion: plant life - 10 years/plant capacity - 300 ton U02/yr till 1980; 500 ton U02/yr after 1980. LWR fuel fabrication: plant life - 15 years/plant capacity - 300 ton U/yr till 1980; 500 ton U/yr Note: (1)

(2) after 1980.

LWR fuel reprocessing: plant life - 15 years/plant (3) capacity - 3 ton U/day.

Note for both tables:

Case I: LWR plus FBR

Introduction of FBR is to begin in 1986, and its shares is to increase gradually, so that by 2000. FBR will account for all the new installations and replacements.

Case II: LWR plus ATR plus F3R

Conditions the same as case I will hold for FBR. ATR introduction is to begin in 1981. ATR will be introduced in three alternate subcases where it accounts for (a) 10%, (b) 30% and (c) 50% of all the new installations and replacements other than FBR.

In both cases, LWR will account for the remaining portion.

As indicated above, amounts of various elements in the nuclear fuel cycle were estimated for different combinations
of reactor types. For computation of fuel requirements, case I and case II-(c), representing maximum and minimum were taken into account to estimate yield.

5. Present Structure of Fuel Fabrication Industry

The Mitsui group centering on Tokyo Shibaura Electric and Ishikawajima-Harima Heavy Industries and the Tokyo Atomic Industrial Consortium, built around Hitachi Limited jointly with General Electric, has set up a joint venture JNF (40 per cent of the capital shared put up by GE, 30 per cent by Toshiba and 30 per cent by Hitachi). The company has been fabricating BWR fuel at Yokosuka factory since 1970 on the basis of a technological tieup contract with GE. Its fabricating capacity is now 210 tons-U a year, but the company obtained Government permission to expand its facilities to 490 tons a year by the spring of 1974. JNF has so far delivered the first and second replacement fuel for the Tsuruga power plant of Japan Atomic Power Co. and the first fuel for the Fukushima No.2 plant of Tokyo Electric Power Co. It has contracted for the delivery of the first fuel to Shimane No.1 plant and the third replacement fuel for the Tsuruga plant. The company is thus prepared for the fabrication of fuel for all BWR plants built by its parent companies GE, Toshiba and Hitachi.

The Mitsubishi group is building PWR on technology it introduced from Westinghouse Electric. Last year in December, the group incorporated Mitsubishi Nuclear Fuel Company. The shares of MNF's capital are 51 per cent from Mitsubishi Metal Corp., 15 per cent from Mitsubishi Heavy Industries and 34 per

cent from Westinghouse. MNF has 100 tons capaerty and plans to expand the factory to produce 280 tons within this year, its capacity must be increased to 420 tons by 1974 for which government permission is being sought.

Furukawa Electric Co., and Sumitomo Electric Industries in July 1972 jointly set up Nuclear Fuel Industries for the fabrication of nuclear fuel. The new company will be the third nuclear fuel company, following Japan Nuclear Fuel and Mitsubishi Nuclear Fuel. This now means that all five nuclear industry groups, either independently or jointly, will have their own nuclear fuel fabrication companies bringing the fuel fabrication business to full status in Japan. The new company will be assisted by Gulf United Nuclear Fuel Co. on fueledesign, fabrication and management technology. The formation of this company which is able to manufacture fuel for both BWR and PWR, must be considered a very important step in promoting commercial competition in the future nuclear fuel market.

In the field of conversion of UF6 to UO2 (enriched uranium), Sumitomo Metal Mining, of the Sumitomo group, is building a plant with a conversion capacity of 240 tons a year at Tokai-mura, Ibaraki Prefecture, to be completed in October this year. MNF will have a production capacity of 240 tons a year, is scheduled to be commissioned in January next.

In the manufacture of zirconium cladding tubes, demand for which is expected to increase with the growth of the fuel conversion and fabrication business, Kobe Steel, Sumitomo Metal Industries and Mitsubishi Metal Corporation have already begun production. Kobe Steel has a factory

producing 130,000 meters of tube at Chofu, Yamaguchi Prefecture, and capacity is planned to be increased to 400,000 meters by the end of 1973.

Sumitomo Metal Industries has facilities in its central research institute for producing 50,000 meters of cladding tubes a year. Capacity is to be increased to 150,000 meters by 1974. After that, a new factory will be built in Shiga Prefecture for annual production of 500,000 meters.

Mitsubishi Metal has completed facilities at Okegawa, producing 50,000 meters. The capacity will be increased to 150,000 meters within this year and to 300,000 meters by the end of 1975.

Zirconium Industry Company was founded last year as the first specialized maker of zircalloy tube shells in Japan. The company is a joint venture, 25 per cent of whose capital is provided by Mitsui & Co. and 75 per cent by the Ishizuka Research Institute Ltd. Technology was imported from the Teledyne Wahchang Albany Company of the United States. Using these techniques as well as techniques developed by Ishizuka Research Institute, the company is going to undertake the full process of manufacture from zirconium sands to tube shells. Nippon Mining Co. which has been making zirconium sponges, is also reported to be founding a new company jointly with Sumitomo Metal Industries.

INDIA'S POLICY WITH REGARD TO DOMESTIC FABRICATION OF NUCLEAR FUEL

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Abstract

In planning a sustained nuclear power programme the need for self sufficiency in fuel fabrication is imperative. This paper describes the manner in which nuclear fuel technology has been developed in India. Natural uranium metal fuel production plants with rated capacities of 30 tonnes per annum have been made operative as early as 1959 to meet the fuel requirements of the research reactors, CIRUS and ZERLINA. Since then these plants are catering to the replacement fuel requirements of these reactors. As a next step, development is carried out on uranium oxide-based power reactor fuel. Aluminium-clad UO, fuel clusters for Zerlina and Zircaloy-clad UO, fuel bundles for the CANDU-type power reactor, RAPP-1, have been produced; to the extent of 40 tonnes of contained UO2. Stainless steel-clad plutonium oxide fuel pins have been produced from plutonium separated by the reprocessing plant for the fast critical test facility at Trombay. Thorium oxide, Thorium metal and its alloys have been fabricated in small quantities for use in the research reactors and for collaborative efforts with Sweden and West Germany. Separation of U²³³ has been carried out on a laboratory scale.

For meeting the large scale fuel demands a uranium mine and mill have been commissioned in 1968 at Jaduguda and large scale fuel production facilities are being commissioned at Hydorabad. The mill at Jaduguda processes about 1000 tonnes of ore per day. Facilities at Nuclear Fuel Complex at Hyderabad are planned to produce 100 tonnes of finished fuel elements of RAPP type starting from zircon sands from Kerala and uranium concentrates from Jaduguda. Plants are being put up to produce enriched uranium oxide fuel assemblies towards reload fuel for Tarapur Power Station from the enriched uranium hexafluoride procured from U.S.A.

The Nuclear Power Programme has been formulated in the "Profile for the Decade 1970-80" with a targeted installed capacity of 2700 MW(e)

by 1980 mainly based upon natural uranium oxide - heavy water moderated and cooled reactor systems. Design studies and test programme are being undertaken for fuel for the advanced large size reactors. As a preliminary to launching a fast breeder reactor programme based on thorium cycle, provision is included for a 40 MW(Th) Fast Breeder Test Reactor, FBTR, at Kalpakkam near Madras. Facilities are being planned for production of fast reactor fuel; while the hot laboratories at Trombay would fabricate the initial fuel charge for FBTR.

Thus, with a clear foresight, fuel production facilities are made operative well in time for the implementation of the reactor construction programme.

The per capita power consumption in India is very low compared to the advanced countries. As published in UN Statistical Year Book 1967, the per expita power consumption in India was 75 KWh as against 6345 KWh im U.S.A., 7907 KWm in Camada and 2171 KWh in Japan. For the envisaged progress in the country it is very essential that the power generation has to be considerably increased. India's resources for such a large scale power generation are advargely distributed in specific locations and not uniformly distributed throughout the country. For instance, the major coal reserves of the country are in the North-Eastern zone of Bengal-Bihar coal helt with very small reserves in the central part of the country. The major hydro power resources are in the not-easily approachable hill ranges and in other available limited zones are very much dependent on the vagaries of rainfall. The established oil resources are very limited. Taking the above into consideration the Atomic Energy Commission of the Government of India has stressed that nuclear power programme has a very important role to play in meeting the power meeds of the country. Here again the established reserves of uranium are of limited nature and are not sufficient to sustain any significant installed nuclear

power capacity for a long time based on natural uranium reactors. However India is one of the foremost countries in its thorium reserves. As such India's nuclear power programme has to be launched towards utilisation of thorium in the long range planning. It has also been recognized that to launch a power programme based on enriched uranium fuel is also difficult because of the large resources that will be required for such a programme. Heavy water moderated and cooled natural uranium fuelled power reactors offer a very advantageous system for the first phase of the power generation programme leading to plutonium fuel based power reactor programme in the second phase and finally the thorium bresder systems. The nuclear power strategy drawn up by the Indian Atomic Energy Commission is shown in Figure 1. To implement such a programme the industrial infra-structure within the country is measure. While in the conventional areas such as generators, pumps, heat-exchangers etc. public sector and private sector industry is coming up for eatering to the meeds of the country, it has been realised that the nuclear fuel production in all its aspects has to be arranged as a governmental activity by the Department of Atomic Energy. Such a programme will bring not only indigenous capability of fuel production with greater flexibility for catering to research and power reactor construction programme but also be ag in self-reliancy in this field. The estimates have indicated that the fabrication costs are comparatively low against the landed costs for the imported fuel. The foreign exchange saving effected by the indigenous operations is also of great importance to a developing country. A mention need also be made of the technological development that comes up within the country because of carrying out such a highly technological and sophisticated operation as fuel fabrication and the employment potentialities arising thereby.

2. Since 1950 onwards three plants of a public sector company, Indian Rare Earths Ltd., viz. the Mineral sands industry in Manavalakurichi and

Chavara, Rare Earths Plant at Alwaye and Thorium Plant at Trombay are being operated for the Department of Atomic Energy for processing monazite sands on the West coast of Kerala for the recovery of thorium salts and zircon and other minerals. The main sources of uranium are located at Jaduguda in the Singhibum district of Bibar State. A mine and mill with a rated capacity of 1000 tonnes of ore per day is being operated by the Uranium Corporation of India Ltd., a company under the Department of Atomic Energy formed in October 1967.

As for the reactors, the construction of a 380 NW(e) Boiling Water 3. Reactor station comprising of two units each of 190 MW(e) at Tarapur (TAPS) has been completed by International General Electric Co. of U.S.A. in the year 1969 and has been in operation since then. Two heavy water reactors, each of 200 MW(e) of CANDU type are under construction at Ranapratapsagar near Kota, Rajasthan (RAPP I, II) and the first reactor is scheduled for criticality in July/August 1972. Construction work is progressing on the two 200 MW(e) reactors at Kalpakkam near Madras (MAFP I, II) to become critical by 1976. A site has been chosen at Narora in Uttar Pradesh for the construction of two 200 MW(e) reactor station. In its profile for the decade 1970-1960 "Atomic Energy and Space Research", the Department of Atomic Emergy has proposed a total installed capacity of 2700 MW(e). While the Tarapur reactor is based upon a slightly enriched uranium oxide fuel cled in zircaloy, other stations are based upon natural uranium oxide fuel clad in zircaloy.

4. The fabrication of fuel has been taken up on a firm basis with a commitment to supply uranium metal fuel cled in aluminium for the initial load of 40 MW(Th) CIRUS research reactor at Trombay. A uranium metal production plant and a fuel fabrication plant were set up by 1959. The major operations for metal production from the magnesium diuranate concentrate comprise of dissolution in mitric acid, purification by TBP solvent extraction and calcio-thermic reduction of the uranium tetrafluoride. Taking

into consideration the annual production rate of 30 tonnes the ingot size is kept to 50 Kgms. In the fabrication plant the metal is vacuum melted and cast, hot rolled in the alpha range, beta heat treated, machined and clad in aluminium tubes. The detailed fabrication flow sheet is given in Figure 2. The initial investment in the fuel fabrication plant is of the order of five million rupees (1959 prices) with a foreign exchange component of three million rupees. It may not be out of place to mention here that basic design of the fuel element has been supplied by Atomic Energy of Canada Limited and opportunities were provided for the scientific staff to visit some of the fabrication operations. The whole technical know-how had to be entirely developed locally and the choice of the equipment and its layout, commissioning and putting into operation has been entirely handled by the Indian personnels. From the initiative and drive put into the programme it has been possible to produce more than half the initial core and a total requirement of components for the initial core loading of CIRUS. The replacement requirements are being completely met since then out of indigenous manufacture. The behaviour of the fuel in the reactor has been entirely satisfactory all these years. Having the facilities for the fabrication of uranium metal, the fuel for ZERLINA (Z ro energy reactor for lettice investigation and new assemblies) based on uranium metal rod clad in aluminium has been fabricated and supplied. 5. By this time it has become apparent that for power reactors uranium oxide fuel has considerable advantages over metal fuel. In conformity with the decision to go in for uranium oxide fuelled heavy water reactor development work has been taken up with regard to the technology of oride fuel fabrication. After considerable study of the production parameters with regard to the pressing and sintering of ceramic grade uranium oxide powder, aluminium clad high density uranium oxide fuel has been produced for a full core of Zerlina reactor. This has given sufficient insight and experience into the technology of production of high density uranium oxide pellets. By this time a commitment was made to produce half the

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initial core loading and stand-by fuel for the Rajasthan-I reactor while half the initial core loading was being imported from Canadian Westinghouse Company through AECL. During the period of fuel production in Canada, a few of the engineers had the opportunity to watch the operations in the Canadian Westinghouse and with the assistance of AECL, specialized equipments for the fuel bundle assembly have been procured from the Canadian Westinghouse as a commercial purchase. Taking the time factor into consideration facilities were improvised at the uranium metal fuel fabrication plant itself for the production of the above requirement. Even though considerable difficulties have been experienced in the early stages of fabrication the full commitment of 2500 fuel bundles with about 40 tonnes of contained uranium oxide has been produced and delivered well in time. In addition to the inspection and quality control supervision of the department. AECL provided quality surveillance and the fuel produced has net all the quality requirements in full. Details of the Rajasthan fuel bundle are given in Annexure I and the flow sheet is shown in Figure 3. 6. To eater to the full scale fuel requirements of the reactor power programme it is felt essential to put up a large scale production plant. Planning has been done not only for the production of fuel but also the sirvaloy and other components required for fuel manufacture and also for the structural members of the reactor such as coolant tubes, calandria tubes etc. These activities are carried out at the Nuclear Fuel Complex set up in Hyderabad which is scheduled for commissioning and operation before the end of 1972. The complex will essentially have the following constituent units : Zirconium Plant (ZP) comprising of Zirconium oxide plant, Zirconium sponge plant, and Zircaloy fabrication plant; Natural Uranium Oride Plant (UCP), Ceramic Fuel Fabrication Plant (CFFP), Enriched Uranium Oxide Plant (EDOP), Enriched Uranium Fuel Pabrication Flant (EFFP), and a quality control laboratory (CPF). The locations of the Nuclear Fuel Complex and the Reactor Systems are shown in Figure 4 and the activities at the Nuclear

Fuel Complex in Figure 5. As for the starting materials for the Nuclear Fuel Complex, zircon comes from the mineral beach sands plant in Kerala of the Indian Hare Earth Ltd.; uranium concentrates from the Jaduguda mill of Uranium Corporation of India Ltd.; enriched uranium in the form of uranium hexafluoride from U.S. The total capital outlay for the Nuclear Fuel Complex is around rupees 140 millions with a foreign exchange component of rupees 47 millions and the details are :

	<u>2</u> P	UOP	CFFP	EUOP	EFFP	CPF	Total	
Annual production capacity Tonnes/yr	50	125	100	30	24	-		
Equipment and machinery R. millions	58.68	9-52	12.20	5.51	3.82	13.82	103.55	
Bldg.and utilities Bs. millions	12-38	2.33	3+1	2.87	0.35	10.13	31.16	

The complex will provide employment for around 1230 personnel including 80 engineers/scientists, 200 supervisors, 600 technicians and 350 semi-skilled persons. This does not include personnel for administration, purchase and accounts staff.

8. The provisions at the Ceramic Fuel Fabrication Plant (CFFP) and Enriched Uranium Fuel Fabrication Plant (EFFP) are dealt with in more detail.

The capacity of CFFP is initially fixed at 100 tonnes per year based on the initial fuel inventory and sustained reload requirements of four 200 MW(e) CANDU-type reactors with fuel burn ups of the order of 8000 MWD/TeU. As per the present indications, the RAPP-I reactor is scheduled for criticality in July/August 1972 and the other reactors by 1976. Depending on the schedule of installation of additional reactors of this type, the capacity of the CFFP can be increased up to 400 tonnes per year by adding the necessary equipment and increasing the working shifts. Important particulars of the fuel and the quantities of the UO₂ required for one reactor charge are presented in Annexure I.

The manufacturing sequence is given in the flow sheet as shown in Figure 3. The area required to house the plant including the component stores, switch gear room and change room for personnel is estimated at 4275 Sq.M. This is proposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building. The front bay accomposed in a 90 x 47.5 M single story building through filters.

The capital outlay for the plant is estimated at rupees 15.28 million (foreign exchange component is rupees 7.15 million) out of which buildings and services are expected to cost rupees 4.01 million. This cost does not include the cost of roads, area services and other ancilliaries like workshop, analytical facilities, cooling tower, canteen, stores etc. which will be provided under the Cosmon Plant Facilities at the Nuclear Fuel Complex.

The total number of personnel required for the plant is estimated at 147 comprising of 8 engineers, 19 supervisors, 101 technicians and 19 19 19 supporting staff. The plant will be fully operational by the end of 4this year.

The cost of operating the plant for 100 tonnes of fuel is estimated to be rupees 7.47 million annually including provisions for process losses ' and 6% interest on 6 months' fuel inventories. The cost of fuel works out to rupees 480 per Kg. of contained UO_2 . This cost includes the cost of about 20 % UO₂ recycling and 10 % excess sirceloy. The landed cost of

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similar fuel works out to rupees 512 per Kg. of contained UO₂. The net fabrication costs starting from U-concentrates to finished products (excluding zircaloy) comes out to rupees 13.20 million in the total value of the product of mpees 48.00 million. This fuel if imported would cost rupees 5..20 million. The savings are only marginal due to the prices of Indian Uranium Concentrates being much higher compared to international prices. Indigenous manufacture of the fuel results in a foreign exchange savings to the extent of rupees 36.9 million which is a major consideration in undertaking this project.

The Enriched Uranium Fuel Fabrication Plant meets the reload requirement of Tarapur Atomic Power Station (TAPS) a 580 MW(e) Boiling Water Reactor. The capacity of the plant is fixed at 24 tonnes per year based on the annual fuel replacements upto 20 % of the core with fuel burn ups of the order of 23,000 MWD/TeU. The enriched uranium hexafluoride imported from U.S. is converted to enriched UO₂ powder at the Enriched Uranium Oride Plant.

The TAPS was commissioned in May 1969 and the first refuelling was done with imported fuel in 1971/72. The second refuelling scheduled for 1972/73 would be also from imported fuel. Nuclear Fuel Complex will meet the refuelling requirements commencing from the third reload.

Important particulars of the fuel and the quantities of the enriched UO_2 and zircaloy required for one reactor charge are presented in the Annexure II. The manufacturing sequence is given in the flow sheet as shown in Figure 7. In view of the limited requirements of this type of fuel, and the similarity in fabrication to that of natural uranium oxide fuel the facility for TAPS fuel fabrication is located in one of the bags of the Ceramic Fuel Fabrication Plant as shown in Figure 6. This resulted in considerable savings in building costs and the services. The main fabrication area is provided in a bay measuring 90 N x 10 N. The layout of the fabrication facility has been designed keeping in view the

requirements of the external inspection agency on the imported enriched uranium. For alkali cleaning, antoclaving and for leak inspection the handling provisions of the 4 meter long assembly dictate a clear 11 meter head space requirement. This is proposed in a separate 6 meter deep pit as an extension to the existing 3.0 meter deep pit of the CFFP. The total number of personnel required for the plant is estimated at 80 comprising of 5 engineers, 17 supervisors, 52 technicians, and 6 supporting staff. The total capital outlay is estimated at rupees 4.17 million with a foreign exchange provision of rupees 1.55 million.

The cost of fabrication of the fuel works out to rupees 562 per Kg. of UO_2 . This cost includes cost of about 20 % UO_2 recycling and 10 % excess sizealoy. The landed cost of similar fuel works out to rupees 2417 per Kg. of UO_2 as against the local cost of rupees 2215 per Kg. of UO_2 including enriched uranium. The net value of the fabrication costs starting from U hexafluoride to finished fuel (excluding sircaloy and hardware) comes out to rupees 8.68 million in the total value of the product of rupees 44.30 million. The landed cost of imported fuel would be around rupees 46.34 million.

9. A comparison of the breakdown of fabrication/production costs for the natural uranium oxide fuel (RAPP fuel) and enriched uranium oxide fuel (TAPS fuel) with the corresponding imported cost is presented below :

		RAFP Fuel		TAPS Fuel		
		MFC Cost	Ex-works cost	NFC Cost	Ex-works cost	
		Rse	Canada Rs.	Rs.	U.S.A. 85.	
1)	Chemical conversion per Kg. UO, in finished fuel.	57.24	}	184.1	140	
2)	Cost of pelletising and assembly per Kg. UO ₂ in the finished fuel.	74.72	} 139 }	134-2)	500	
3)	Cost of Zirc. and other components per Kg. UO in the finished fuel. ²	78.50	92	200.0		
		210.46	231	518.3	640	

10. Thorium fuel cycle is of great significance to countries like India which possess large reserves of thorium. In the nuclear power generation programme in India, the third generation reactors will be $Th-U^{235}$ breader systems. Development work was taken up on thorium fuels as early as 1958 in the fuel fabrication facility at Trombay. Thorium and thorium oxide pelletised fuels canned in aluminium have been made for Conversion studies in CIRUS. There are in all 90 positions available in the CIR, J-rod annulus for irradiation purposes. Thorium metal pewder is produced by calcio-thermic reduction of ThO_2 . The flow sheet for the fabrication of thorium and thorium oxide fuel shapes is given in Figure 8. Under a collaboration agreement with Sweden about 550 Kg. of sintered pellets of ThO_2 were supplied to A.B.Atomenergi.

Under Ingo-German Technical Collaboration programme, an advanced fuel concept is to be developed for sodium cooled fast breeders which could lead to high breeding gain and low doubling time. The basic idea is to use thorium metal as a matrix for the fissionable material (Plutonium or enriched uranium) with a thin cladding. For preliminary investigations Th, Th-U (10%) alloy strips and blocks have been supplied to KFA, Julich. These blocks are produced by extruding the are melted sintered thorium.

The fuel reprocessing plant at Trombay has successfully separated U²³³ from the irradiated thorium rods.

11. Plutonium fuel development work was initiated with the commissioning in 1964 of the fuel reprocessing plant at Trombay to reprocess the irradiated fuel from CIRUS. Pure PuC₂ sintered pellets of about 90 % TD have been produced and fabricated into stainless steel clad fuel pins for the Zero Energy Pulsed Fast Critical Assembly. Flow sheet for the fabrication of these fuel pins is given in Figure 9. About 200 such fuel pins containing about 24 Kg. of PuO₂ have been fabricated for the critical facility.

A 40 MW(Th) fast breeder test reactor basically similar to the French RAPSODIE-FORTISSIMO is being constructed at the Reactor Research Centre, Madras under an agreement with CEA, France. FBTK will have (C.7 Pu.3)0,

uranium oxide enriched to about 90 % in U^{235} , as fuel. According to the agreement. the responsibility for the whole project including the construction of the reactor will lie with the Indian Atomic Energy Commission. Indian industry will have a maximum share in the engineering of the project and supply of materials and equipment. Investigations are being carried out at Trombay on U0,-Pu0, fuels for developing a fabrication process for the fast reactor fuel. Details of the FBTR fuel are given in Annexure III. The general flow sheet for the fabrication of fissile fuel assembly for FBTR is given in Figure 10. The initial fuel charge for FBTE will be fabricated at Trombay. It may be mentioned here that the drawings for a similar fuel were supplied by CEA, France and the whole technical know how starting from the drawing up of the specifications, design of the equipment, building up of the fabrication facility and production of the fuel assemblies has to be entirely developed indigenously. Planning is in progress for the setting up of a plutonium fuel fabrication facility at Reactor Research Centre, Madras. Concluding, the Department of Atomic Energy has acquired during the 12. last decade a capability to process and manufacture different types of uranium, thorium and plutonium fuels indigenously. The commissioning of the large scale fuel fabrication plants well in time for the implementation of the reactor construction programme, is yet another milestone in our endeavour to achieve self-sufficiency in the field of nuclear power technology.

Annexure I

DESCRIPTION OF RAPP FUEL BUNDLE

The fuel, uranium oxide is in the form of sintered pellets, 14.26 mm (0.56") in diameter and about 20 mm in height. One end of the pellet is made slightly dished. About 24 numbers of such pellets are encased in a zircaloy tube, 15.22 mm (0.6") o.d., 0.38 mm (0.015") wall thickness, and 49.5 mm (19.5") in length, both ends of which are sealed by welding zircaloy end plugs. Nineteen such tubes filled with UO2, called "elements" are assembled between two end plates in two concentric rings of six and twelve "elements" respectively around a central element. The six elements of inner ring and the six alternate elements of the outer ring are wrapped with 0.05" dismeter sircaloy-2 wire in a helical manner to promote inter-sub-channel mixing of the coolant. The other six elements of the outer ring have spacer wires welded on to them. In addition, all the twelve outer elements are provided with wrapped bearing pads of 0.065" dia. zircaloy-2 wire, which space the bundle from the coolant tube and reduce the wear on the coolant tube during the fuel charging operation. This 19-element assembly is called a "fuel bundle". The details of the RAPP fuel are given in the enclosure.

Enclosure to Annexure I

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RAPF FUEL DETAILS

I. Uranium Oride

1.	Number of	oxide pellets/tube	1	24 ±	2	(hei	ight	20 mm)
2.	Number of	pellets/bundle	:	24 x	19	样	456		
3.	Weight of	pellets/tube	:	24 x	33	*	79 5	# 800	g •
4.	Weight of	pellets/bundle	ŧ	800 ;	x 19) ==	15+1	2 Kg.	

II. Zircaloy components per bundle (finished weight)

1.	Tubes	(19 Nos.)	\$	1079•2 s •
2.	End plags	(38 Nos.)	ŧ	159.6 g.
3.	End plates	(2 Nos.)	:	33.2 S.
4.	Spacer wire	(0.050" dia.)	:	93.0 S .
5.	Bearing pad wire	(0.065" dia.)	\$	37.0 5.

III. Reactor charge

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1.	Weight of the finished fuel bundle	:	16.7 Kg.
2.	Number of bundles/channel	:	12
3.	Number of channels in the reactor	:	306
4.	Number of bundles in the reactor	:	3672
5.	Weight of UO in the reactor/ full loading	1	56 tonnes
6.	Total number of fuel bundles required before start up	3	4000
7.	Total fuel inventory	\$	60 tonnes
8.	Annual replacement (average burn-up @ 8000 MWD/t U)	\$	191C bundles
9.	Finished UO ₂ required for replacement	1	30 tonnes

Annexure II

DESCRIPTION OF TAPS FUEL ASSEMBLY

The fuel, uranium oxide slightly enriched in U^{235} , is in the form of sintered pellets 12.24 mm (0.482") in diameter and about 20 mm in height. About 200 Nos. of such pellets are encased in an autoclaved zircaloy tube 14.28 mm (0.5625") o.d., 0.69 mm (0.035") wall thickness and 3886.2 mm (153") in length, both ends of which are sealed by arewelding zircaloy end plugs. Thirty six such tubes filled with UO_2 , called elements, are arranged in a 6 x 6 square array in between two stainless steel tie plates. The spacing between the individual elements of this long assembly is maintained by means of spacers, seven in each assembly. One of the thirty six elements, called the segmented element, locks the spacers in position. The entire assembly is inserted in a four meter long zircaloy channel of square cross section. Each 36-element assembly contains about 160 Kg. of UO_2 and there are 284 such assemblies in each reactor. The details of the TAPS fuel are given in the enclosure.

TAPS FUEL DETAILS

1.	Uranium Oxide	Number/Weight
	 i) No. of pellets/tube ii) No. f pellets/assembly iii) Weight of pellets/tube iv) Weight of pellets/assembly 	200 7200 4.3 to 4.45 Kg. 155.0 to 160.0 Kg.
2.	Zircaloy components per assembly	· , , , , , , , , , , , , , , , , , , ,
	i) Tube - 35 each 153" long) 8 each 18.6" long)	35 Kg.
	ii) End plugs 72) connector plugs - 7)	1.07 Kg.
	iii) Channel	17.44 Kg.
3.	Other components for assembly	
	i) Upper and lower tie plates ii) Guard iii) Plenum springs iv) Fuel assembly springs	Stainless steel 1 + 1 Stainless steel 1 Music wire steel 43 Inconel 1
	v) Fuel rod outer springs	Inconel 36
	vi) Hexagonal nuts	Stainless steel 5
	vil) Locking Teb washers	Stainless steel 4
	VIII) Vap screw	STRINICSS STOCL 1
	1X) LOCK Washer	Inconel 1
	X) retaining ring	
	rt) obscel staamottes	income avaiant 7
	wid) Bus late for environters	Stainlass at all 7
	viii) Wafera for connectors	Stainless steel 14
	Reactor about for 100 Me station	
4 +	AGACTOR CHARGE IVE 170 BEE STATION	
	i) Weight of UO ₂ per finished fuel assembly	155 to 160 Kg.
	* ii) No. of assemblies in the reactor	264
	*iii) No. of channels in the reactor	284
	* iv) Weight of 00 in full reactor loading	47 tonnes
5.	Average replacements (for two reactors)	
	i) Average annual replacement (advanced design)	20 tonnes
	11) Finished assemblies required for	105 No
	replacement (average)	727 108. 28 Noo
	ii) No. of sircaloy square chamels if 70% are reused	JO NOE.
	* Data for the station as a whole is t figure given under this item	wice the

Annexure III

DESCRIPTION OF FETR FUEL BUNDLE

The fuel for FHTR core is mixed oxide of uranium (85 % enriched) and plutonium. The mixed oxides sintered pellets, 4.36 mm dis. are canned in a 5.1 mm. o.d. stainless steel tube. The total length of the fissile pin with end plugs welded is 53.5 mm and has a spacer wire (0.76 mm dis.) helically wrapped around it. A fuel element assembly consists of 61 fissile fuel pins in the middle section whereas the top and bottom sections are axial blanket pins containing sintered thorium oxide pellets canned in stainless steel.tubes. The top and bottom blanket sub-assemblies are 7-pin clusters mounted on the inner hexagonal tubes. The fuel pins and the top and bottom sub-assemblies are finally clad with the external hexagonal tube to form the fissile fuel element assembly. There will be about 65 such fuel assemblies in the FETR core. Amendments to be incorporated in the Text of the paper "INDIA'S FOLICY WITH REGARD TO DOMESTIC FARRICATION OF NUCLEAR FUEL" by Pr. N.Kondal Rao Bhabha Atomic Research Centre, Trombay, India.

-x-x-

Reference	As appeared in the text	To be amended as
1. Page 3 Para 3 4th line	"Two heavy water reactors each of 200 NW(e) criticality in July/August 1972"	"Of the two heavy water reactors, 200 Md(e) each, under installation at Rana Fratap Segar near Kota, Hajasthan (HAFF 1 & II), HAFF I has gone critical on 1972 and HAFF II is in an advanced stage of construction".
2. Page 7 Fara 8 6th line	"As per the present indi- cation criticality in July/August 1972"	"hAFP I reactor has gone critical on 11th August 1972 and as per the present indications the other reactors will be on line by 1976".
3. Page 12 7th line	" enriched to about 90% in U^{235} as fuel"	" enriched to about 85% in U^{235} as fuel".
4. Enclosure to Annexu II - 3(vi:	"Cap Screw - S. Steel" re ii)	"Cap screw - Stainless Steel"





FULL ASSEMBLY

FIG.2 URANIUM FUEL ELEMENT FABRICATION SCHEME



FIG. 3 FLOW SHEET FOR FABRICATION AND TESTING OF RAPP FUEL BUNDLE.





FIG. 5 ACTIVITIES OF NUCLEAR FUEL COMPLEX





FIG. 7 FLOWSHEET FOR THE FABRICATION OF TAPS FUEL ASSEMBLY





PLUTONIUN (IV) NITRATE SOLUTION S.S. TUBE PREPARATION (6~10 g/LIT. N 3M) TUBE INSPECTION, DIMENSIONS CHECK UP, PLUTONIUN (IV) OXALATE PRECIPITATION RADIOGRAPHIC EXAMINATION (50°C, 2 NR.) FIRST END PLUS WELDING FILTRATION, WASH AND AND WELD EXAMINATION DRYING (150 °C) LOADING OF PELLETS CALCINATION 658 C PUOP EVACUATION, HELIUM DEAGGLOMERATION- 200 NESH FILLING, INSERTION OF 2ND END PLUS PRECOMPACTION 2 TE/CHT AUTOMATIC WELDING OF IND END PLUS GRANULATION -10 TO +20 MESH SURFACE CONTAMINATION CHECK FINAL COMPACTION, 3-8 Te/GINE GREEN DENSITY 46 5 T D. CLEANING IN ALCOHOL SINTERING IN \$3 HZ-A GAS WIXTURE HELIUM LEAK DETECTION AT 1480-1820°C. 4H. HEATING 4°C/MIN. COOLING SC/MIN. RADIOGRAPHIC EXAMINATION PELLET INSPECTION, NEASUREMENT OF GEONETRIC AND BULK DENSITIES FINAL INSPECTION, DIMENSIONAL CHECK UP ULTRASONIC CLEANING OF FINISHED FUEL ELEMENTS PELLETS IN ALCOHOL STORAGE IN MRDCAGE FOR SHIPMENT DRYING UNDER INFRA-RED LANP FISSILE COLUMN LENGTH AND WEIGHT CHECK UP J ه هې همه به مه به مه

FIG. 10 FLOWSHEET FOR THE FABRICATION OF FISSILE FUEL ASSEMBLY FOR FBTR

FISSILE FUEL PINS

LOWER SUB-ASSEMBLY

UPPER SUB-ASSEMBLY



ESTIMATING CANDU FUEL COSTS

by

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ABSTRACT

A method of estimating CANDU fuel costs is described The equations are sufficiently generalized that the method can be applied to a new fabrication industry where the fabrication techniques, costs of buildings, equipment and labour, and financial practices differ from those in Canada. Once a reference design cost has been established, the method can be used to estimate the effect on fuel cost due to expansion of the industry, learning and design modifications.

When applied to Canadian industry the method estimates the known fuel costs with adequate accuracy. It is also apparent that the low cost for CANDU fuel is due largely to the simple design and small number of components. Even when enriched, the cost of CANDU fuel is significantly lower than the cost of light-water reactor fuel.

Résumé

L'auteur décrit une méthode permettant d'estimer le coût du combustible CANDU. Les équations sont assez généralisées pour que cette méthode puisse s'appliquer à une nouvelle industrie manufacturière où les techniques de fabrication, les dépenses engagées pour les bâtiments, l'équipement et la main-d'oeuvre, ainsi que les pratiques financières diffèrent de celles du Canada. A partir d'un coût de réféference calculé, la méthode envisagée permet d'estimer l'effet, sur le coût du combustible, de l'expansion de la nouvelle industrie, de l'expérience acquise et des modifications apportées aux concepts originaux.

Appliquée à l'industrie canadienne, cette méthode permet d'estimer les coûts connus du combustible, avec une bonne precision. Il est, par ailleurs, évident que le faible coût du combustible CANDU découle en grande partie de son concept simple et du petit nombre de scs compos nis. Même lorsqu'il est enrichi, le combustible CANDU est nettement moins coûteux que le combustible destiné aux réactours à eau légère.

e.

NOMENCLATURE

8	fractional irrecoverable loss allowance	ℓ _R	reference design pellet length — mm
a'	tubing cost equation constant	۹ ۳	modified design pellet length — mm
a″	particular value of a'	L	bundle length — m
A	annual labour cost	m	ratio of capacities of new to reference
b	fractional annual charge rate on buildings		piant
b'	constant in end cap cost equation	м	annuai plant capacity — Mg U/yr.
b″	particular value of b'	Mo	reference plant integrated production for doubling Mg
B	cost of buildings	ΣM	industry integrated production for
c	constant in other metal cost equation	20142	doubling Mg
c _F	feed component for extra U-235 enrich-	n	number of doublings
đ_	element diameter — mm	N	number of elements in bundle
-0 đ.,	reference design pellet diameter mm	NR	number of elements in reference design.
°К д.,	modified design policy diameter - mm	NM	number of elements in modified design
M D	hundle diameter - mm	0	fractional overhead charge rate
e	fractional annual charge rate on equip-	Pj	generalized fabrication cost per kg U for j th plant
E	cost of equipment	$(\mathbf{p}_{\mathbf{R}})_{\mathbf{j}}$	reference fabrication cost per kg U for j th plant
f	fraction in pelletization cost equation	(pp);	base fabrication cost per kg U for i th
f(L)	bundle length function	(* <i>B</i> /j	plant
F	fixed cost per kg U	PR	reference fuel cost per kg U
g	fraction in encapsulation and assembly cost equation	PB	base fuel cost per kg U
h _o	operator purchased material cost per kg U (without losses) for i th plant	(p _P) _R	reference design pelletization cost per kg U
h	fabricator purchased material cost per kg	(p _E + p _A) ₁	Rreference design encapsulation and assembly cost per kg U
H _j	generalized fabricator-purchased material	q	plant capacity equation constant exponent
-	cost per kg U for jui plant	<u>ک</u>	plant capacity equation constant ex-
(H _R)j	reference design reference fabricator-		ponent
	plant	8 C	cost of supprises as fraction of A
(H _B);	reference design base fabricator-pur-	OUTT	
· <i>w</i> .y	chased material cost per kg U for jth	340	separative work units
	plant.	t v >	processing time — yrs
(H _{OR}) _j	reference design reference operator-pur- chased material cost per kg U for j th	(l _{oo})j	time from start of fabrication in jur plant to delivery of fuel bundle – yrs
	plant	ι _w	tube wall thickness - mm
(H _{OB}) _j	reference design base operator-purchased	т _ј	shipping cost per kg U from j th plant
	manchal competing of the picture	۷	fraction of costs that are variable
л Т	internet observe nor by II accondited with	V	variable costs per kg U
'n	jth plant	W	fractional annual charge rate on working capital
j	plant index		

W	weight of uranium in a fuel bundle — kg	Z	fractional plant load factor
W _M	weight of uranium in modified design bundle – kg	$\rho_{\rm R}$	effective density of reference pellet — kg U/mm ³ UO ₂
W _{R.}	weight of uranium in reference design bundie kg	۴ _M	effective density of modified design pellet — kg U/mm ³ UO ₂
x	constant in learning equation	CANDU	Canada Deuterium Uranium
У	constant in learning equation	PHW	Pressurized heavy water

1. INTRODUCTION

This paper presents a set of rules, based on Canadian industry's experience in fabricating several hundred tonnes, for estimating CANDU fuel costs. By suitably modifying these rules as dictated by the financial practices, labour rates and material costs in other nations, it is possible to estimate the cost of fabricating CANDU fuel elsewhere than in Canada. The rules are not limited to today's market and design but permit assessment of future costs as a function of demand, learning and design changes.

The current Canadian price for natural uranium, 28 element, 100 mm diameter CANDU fuel is about $45/kg \ U \ (1972 \ Canadian \ dollars)^{(1)^1}$. The reason for this low price for nuclear fuel need not be mysterious; it is due to factors such as these.

- 1. The design is simple; the fuel bundle has a minimum number of components and is not encumbered with parts for other functions such as control rods or channel delineators. Consequently the investment in plant and equipment, the plant operating costs and the material costs are all reduced relative to those for fabricating more complex fuel assemblies.
- 2. The fuel bundle is short allowing more repetitive operations per unit mass of fuel which yields the advantages of mass production with a smaller throughput. An additional benefit is that the elapsed time for fabrication is shortened which reduces storage and inventory costs.
- 3. The fuel material is natural uranium. Hence the incremental costs associated with handling enriched uranium are avoided. Even with enriched uranium the fabrication cost for CANDU fuel is significantly less than for more complex fuel assemblies.

Comparisons of CANDU fuel with the light-water reactor fuel by numbers and kinds of components to be processed and inspected, and by quantities of materials per unit weight of uranium are given in Table 1⁽²⁾. Table 2⁽¹⁾ demonstrates that the benefits of mass production can be quickly gained.

TABLE 1 FABRICATION DATA

item	Approx. Ratio of Quantities PWR CANDU
INSPECTION	
Parts to be inspected/kg U	2
Kinds of parts/kg U	6
Length of tubing/kg U	2
PROCESS	
Kinds of parts/assembly	6
Kinds of process activilies	2
Circumference of pellet for grinding	2
MATERIALS	
Length of tubing/kg U	>2
Weight of end plugs/pair	>2
Weight of other Zr/kg U	3
Weight of stainless steel/kg U	cc

TABLE 2. PRODUCTION EXPERIENCE DATA (Canadian & U.S. Production to end of 1970)

	CANDU	BWR ⁴	PWR*
No. of assemblies	30,000	7,800	3,800
No. of cap welds (x 10^3)	1,400	760	1,340
No. of rods (x 10 ³)	700	380	670

⁸ Assumes equal production of BWRs and PWRs

2. THE COMPONENTS OF FUEL COSTS

Fuel cost is the total cost to the operator of the nuclear reactor for the fuel delivered to the reactor. The components of fuel cost include

- the fabrication cost encompassing the depreciation and carrying charges for the plant buildings and equipment, the plant operating costs, and plant overhead costs,
- the cost of material in the fuel bundle (uranium and zirconium) with an allowance for irrecoverable losses,
- the charge for financing materials used in production,
- shipping costs.

¹ Reference (1) states the price is less than \$44/kg U in 1967 Canadian dollars. As will be seen in this text the effect of inflation of the dollar from 1967 to 1972 has been largely offset by a reduction in the constant dollar fabrication cost.

To analyze and estimate fuel costs these component categories are subdivided as necessary to correspond to the separate process operations and material purchases.

2.1 Fabrication Plant Costs

The plant – the buildings, equipment, supplies² and labour force – required to fabricate fuel is adaptable. It can be planned so that the entire fue. fabrication process is under one roof, or the component processes can be geographically separate. The processing method can range from fully automated to highly labour intensive. Whatever the detail of the plant, a fuel fabrication cost can be estimated Once this cost for a reference plant and fuel design is available, then the effect on cost of extrapolations in plant capacity and load factor (utilization of eapacity) can be analyzed.

The fundamental approach is to categorize the fixed (F) and variable (V) cost components. By fixed is meant those costs that remain the same per unit of product; that is, they vary linearly with plant capacity. For example, the direct labour in a manual process operation such as visual inspection of a component, or equipment that must be increased in numbers to increase capacity represent constant costs per unit of product. The variable cost components are those that have a non-linear relationship to plant capacity and load, and so are not constant per unit product. Examples of this type are indurect labour such as office staff or labour to operate equipment that can be scaled in size to suit plant capacity. Generally the costs of the building, equipment, indirect labour and supplies for office and maintenance are variable, while most, but not necessarily all, direct labour costs are fixed per unit product.

2.2 Material Costs

Materials used in fuel fabrication $-UO_2$ and zirconium alloys are the only materials in CANDU fuel – can be acquired by the fabrication plant at different stages of preparation. The plant might be associated with a uranium mine, or purchase U_3O_8 which it would convert to UO_2 or purchase the UO_2 powder. For the purpose of this analysis it is assumed the materials are acquired as U_3O_8 for natural uranium fuel, UF₆ for enriched uranium fuel and the zirconium alloys are bought in the form of tubing, bar stock and sheet. It should be apparent that if the plant carries out more or less of the materials processing, the principles of estimating costs enunciated here can be applied to the materials processing.

Some material, usually uranium, is purchased by the reactor operator and supplied to the fabricator for further processing. Other material, usually zirconium alloys, is purchased by the fabricator. In either case, the cost of materials in the form purchased is treated as a separate component of fuel cost. The cost of processing the material is a part of the fabrication cost component. Due allowance must be made for processing losses; some losses are irrecoverable, others are recovered as scrap which can be recycled.

The cost of uranium per unit product does not appear to vary with plant capacity and so is a fixed cost per unit product Zirconium alloy costs vary slightly with plant capacity, perhaps reflecting the benefit of large order processing by the material supplier.

2.3 Financing Charges

2.3.1 Capital Charges

The fabricator's investment in building and equipment, composed of some proportion of equity and debt must be amortized over their lifetimes. This annual depreciation is a charge on fuel fabrication. In addition, an annual return on the investment which could be taxable, the insurance on these assets and property taxes are costs of fabrication. The total of these, the capital charge rate, will likely differ as between buildings and equipment since the latter generally has a shorter useful life. As components of the fuel fabrication cost, the capital charges are variable

2.3.2 Working Capital

A supply of capital must be available to the fabricator to meet the operating and purchased³ material costs. Generally it is adequate to have sufficient to cover these costs for the time required to process the product. As this capital is cycled rather than expended, the contribution to fabrication cost is the charge for using it. The charge is apportioned between fixed and variable cost components in the same ratio as the operating costs are divided.

2.3.3 Overhead

In this category are the fabricator's costs of re-

² Supplies refers to the materials consumed in the plant offices and shops. It does not include the uranium and zirconium materials used in the fuel,

³ Material purchased by the fabricator for processing. Material purchased by the reactor operator and supplied to the fabricator for further processing does not require working capital.
search and development, warranties, administration and marketing. It is adequate for estimating the contribution to fabrication costs to allow some percentage increase on all costs, the allowance will be a function of local practices, as for example, whether the fabrication plant is publicly or privately owned, whether the warranties are on workmanship only or include fuel performance, and so on.

2.3.4 Interest on Component Inventory

As materials purchased by the reactor operator are acquired and parts of the fabrication process are completed, payments will be made to suppliers. But until the fabrication is complete and the fuel is available for the reactor these payments represent an investment by the reactor operator, which will carry some charge that contributes to the cost of the fuel. The charge, a rate of interest appropriate to the economy, is for the period from acquisition of the material or component until the fuel fabrication is complete. The charge on the inventory of completed fuel being held for the reactor is not included as a fabrication cost.

3. ESTIMATING FUEL COSTS

3.1 The Reference Industry

Assuming a fuel fabrication industry is to be established, the first step in estimating fabrication cost is to select the fuel fabrication route for a reference design⁴, that is, what the annual capacity of the industry should be, in what form materials will be purchased from other suppliers, what processes will be separated or integrated, where plants will be established, how each process will be carried out, and what space, equipment and labour will be needed.

The fabrication cost component of fuel cost is estimated for each plant performing a basic operation. The annual cost in the ith plant is the total of

 the annual capital charge on buildings, a variable cost bB

where B is the cost of buildings,

b is the fractional annual charge rate

- the annual capital charge on equipment, a variable cost eE
 - where E is the cost of the equipment
 - e is the fractional annual charge rate

TABLE 3. PICKERING FUEL BUNDLE DESIGN PARAMETERS

Nominal bun	dle dia	imeter	100 mm
Bundle lengt	h		0.5 m
Weight urani	um		197 kg
sumber of e	ement	*	28
Element shea	ih OD	ı.	15 20 mm
Element shea	ith wal	ii	0 42 mm
Fuel pellet O	D		14 3 mm
Fuel pellet le	ngth		20,9 mm
Fuel		UO2	
Metal	-	Zircaioy-4 only	

- the annual operating cost consisting of the variable cost v(1 + s) A (1 + wt) and the fixed cost (1-v) (1 + s) A (1 + wt)
 - where v is the fraction of costs that are variable 1-v is the fraction of costs that are fixed
 - A is the annual labour cost

 - s is the annual cost of supplies as a fraction of A
 - w is the fractional annual charge rate on working capital
 - t is the processing time as a fraction of a year.

The fabrication cost per unit product in the jth plant is then

$$(\mathbf{P}_{\mathbf{R}})_{\mathbf{i}} = (\mathbf{V}_{\mathbf{i}} + \mathbf{F}_{\mathbf{i}}) \tag{1a}$$

where V_1 is the variable cost per unit product

F_j is the fixed cost per unit product
and V_j =
$$\frac{1}{M}$$
 [bB + eE + vA(1 + s)(1 + wt)](1 + o)
F_j = $\frac{1}{M}$ (1 - v)A(1 + s)(1 + wt) (1 + o)

where M is the annual plant capacity - Mg U/yr

o is the overhead fractional charge rate

The component of fuel costs for materials purchased by the jth fabricator is

$$(H_R)_j = h(1 + wt)(1 + o)(1 + a)$$
 (1b)

⁴The reference design for the CANDU system in the Pickering fuel bundle The design parameters are listed in Table 3.

where h is the cost to the jth fabricator for material per unit product

a is the fractional irrecoverable loss allowance

The component in fuel costs for materials purchased by the reactor operator and supplied to the jth fabricator for further processing per unit product, is

$$(H_{OR})_{j} = h_{O} (1 + a)$$
 (16)

where h_0 is the cost to the reactor operator of material per unit product.

The interest charge on the payments made by the reactor operator for materials or completed component inventory supplied to the *s*th fabrication plant is

$$I_j = i \left[(H_{OR})_j + ((H_R)_{j-1} + (p_R)_{j-1}) \right] (t_{\infty})_j$$
 (1d)

where i is the fractional annual interest rate

- (t_{cc})₁ denotes the time as a fraction of a year from delivery of the material or component to the jth plant until delivery of the finished fuel to the reactor
- $(H_R)_{j-1}$ is the cost component for material from the (j-1)th fabricator
- $(p_R)_{j-1}$ is the cost component for fabrication by the $(j-1)^{th}$ fabricator.

The component for the cost of shipping from the j^{th} fabricator or material supplier is T_j . (1e)

The reference fuel cost is the total of these components

$$P_{R} = \sum_{j} \left[(p_{R})_{j} + (H_{R})_{j} + (H_{OR})_{j} + I_{j} + T_{j} \right]$$
(2)

3.2 The Effect of Growth

As the nuclear power system grows, the fuel fabrication industry will expand With additional experience new fabrication techniques will evolve. Even the fuel design will likely be modified. This growth will alter the cost of fabrication.

3.2.1 Plant Capacity and Load Factor

For each fabrication plant in the reference industry operating at its full annual capacity, i.e., 100^{cr} load factor, the cost per unit product was given by equation (1a) Should the annual capacity of the plant⁵ be changed or should it operate at less than full capacity, then the fabriction cost becomes

$$P_j = \frac{V_j(m)^{-q}}{Z} + F_j$$
 (3a)

where m is the ratio of the annual capacities of the new plant to the reference plant,

Z is the load factor expressed as a fraction,

q and r are constant exponents. Their particular values are functions of the general fuel design.

The fabricator-purchased inaterial costs expressed by equation (1b) are also affected by plant capacity. As a function of capacity the cost is

$$H_{i} = (H_{R})_{i} (m)^{-r}$$
 (3b)

3.2.2 Learning

As the industry gams experience, it will discover less costly ways of doing things and become more efficient. This is the learning process. The cost components that are reduced by learning, viz: fabrication and fabricator-purchased materials, are multiplied by a learning factor, $S \leq 1$, to estimate the cost reduction due to learning. The relationship between experience and learning is not likely to be linear since potential additional benefits from learning decrease with learning The learning factor can be expressed by an equation of the form

$$S = 1 \cdot x [1 \cdot exp(-yn)]$$
 (4)

where n is the number of production doublings, that is, the number of times the total production of the industry, ΣM , has doubled referenced to the total production, M_0 , of the largest capacity plant in the industry up to and including the first order processed that exceeds its annual capacity. By definition,

$$\frac{\ln \left[\frac{\sum M}{M_0}\right]}{\ln 2}$$
x and y are constants

By combining equations (3a) and (3b) with (4) the base costs for fabrication and fabricator-purchased materials in the jth fabrication plant are:

$$(p_{B})_{j} = S \left[\frac{V_{j}(m)^{\cdot q}}{Z} + F_{j} \right]$$

 $(H_{B})_{j} = S (H_{R})_{j} (m)^{\cdot r}$ (5)

The base fuel cost for the reference design is

$$P_{B} = \sum_{j} \left[(P_{B})_{j} + (H_{B})_{j} + (H_{OB})_{j} + I_{j} + T_{j} \right]$$
 (6)

⁵ Plant in this context can be either the expansion of an existing plant or the addition of a new plant producing the same product. Where there are competing plants fabrication cost estimates are based on the capacity of the largest plant.

3.2.3 Design

Modification of the basic CANDU design can be anticipated. There are several reasons, viz: each member of the "family" of CANDU reactors imposes different operating conditions upon the fuel; new fuel cycles may require modified design parameters; experience in fuel performance might suggest improvements; or changing fabrication costs might be optimized by altering the design Provided the "rodded" bundle remains as the basic concept, and the materials are UO2 (with or without enrichment by extra U-235 or Pu additions) and zirconium alloy metal only, the effect of design modifications on fabrication cost can be reasonably well estimated. Some minor design modifications are not accounted for in what follows, as for example, the number of inter-element spacers, they generally do not change the fabrication cost significantly, certainly less than the probable error in the fuel cost estimate.

To assess the fabrication cost when the design is modified, it is first necessary to establish a base cost for the reference Pickering design for the year of the new estimate. This is done by applying the rules set out above.

Note that if the design change involves a change of the fuel material, that is, if enriched uranium or plutonium is used instead of natural uranium, then a new reference fabrication cost must be computed. The change in fuel material alters not only the fuel material cost but all fabrication plant requirements. This new reference fabrication cost becomes the base cost for factoring in the effects of other design changes.

3.2.3.1 Materials

Uranium costs up to and including the cost of preparing UO_2 powder are not affected by design changes.

The zirconium alloy material supply costs are modified by design changes as follows.

a) Tubing

The cost of tubing per unit length is a function of its dimensions and is expressed $as^{(4)}$.

$$Cost/m = a' [1.12 (d_0 - t_w)t_w + d_0]$$
(7)

where a' is a constant

 $\mathbf{d}_{\mathbf{n}}$ is the outside diameter of the tube – mm

The constant a' can be evaluated from the reference design dimensions and base cost. For the modified design then, the cost of tubing per unit weight of uranium is

$$\operatorname{Cost/kg} U = \mathbf{a}'' \quad \frac{\mathrm{LN}}{\mathrm{W}} \left[1.12 \, (\mathbf{d}_{\mathrm{O}} \cdot \mathbf{t}_{\mathrm{W}}) \mathbf{t}_{\mathrm{W}} + \mathbf{d}_{\mathrm{O}} \right]$$
(8)

where a" is the particular value of a' for the base cost reference design

), is the length of an element in the modified $a_{estgn} = n_{e}$

N is the number of elements in the modified design

W is the weight of uranium in the modified design bundle -kg U.

When more than one element size (diameter) is used in a bundle, the formula should be applied to each using the number of elements of a size, and weight of uranium in those elements.

b) End Caps

The length of the end cap is not likely to be altered. The cost is proportional to the diameter,

$$Cost/end cap = b'd_0$$
(9)

The constant b' can be evaluated from the reference design dimensions and base cost. For the modified design, the cost of end caps is

$$\operatorname{Cost/kg} U = \frac{2b'' N}{W} d_0 \tag{10}$$

where b'' is the particular value for b' from the base cost reference design

When there is more than one element size in the fuel bundle, equation (10) should be applied to each using the number of elements of a size and the weight of uranium in those elements.

c) Other Metal

Other metal includes the end plates, element spacers and bearing pads. The cost is basically a function of the bundle dimensions, that is,

$$\operatorname{Cost/kg} U = c \frac{D^2 L}{W}$$
(11)

Where D is the bundle diameter in mm

L is the bundle length in m

The constant c is evaluated from the reference design base cost

3.2.3.2 Pelletization

The cost for pelletization is a function of both the number of pellets per unit weight of uranium and the weight of uranium in the fuel bundle. That is, the cost of pelletization consists of

- the fraction that is constant per kg U, $f(p_p)_R$

where f is a fraction less than 1

 $(p_p)_R$ is the reference cost for pelletization per r_g U

- plus the fraction that is constant per pellet. Since the number of pellets per unit weight of manium is inversely proportional to the density and volume of a pellet, then this fraction of the cost is (1-f) $(p_D)_B \rho_B R_B d_B^2$

where ρ_R is the effective uranium density,⁶ kg U/mm³ UO₂

 $R_{\rm R}$ is the length of a pellet, mm

d_R is the diameter of a pellet.

For a modified design the cost of pelletization is

Cost/kg U =
$$(p_p)_R \left[f + (1-f) \frac{\rho_R \ell_R d_R^2}{\rho_M \ell_M d^2_M} \right]$$
 (12)

where the subscript R refers to the reference design pellet

the subscript M refers to the modified design pellet.

The fraction, f, is evaluated either from experience or by doing a detailed engineering cost assessment of the pelletization process.

When there is more than one element size, the formula should be applied to each size on an element basis then summed over all the elements.

3,2.3.3 Encapsulation and Assembly

As the effect of design changes is much the same for encapsulation as for assembly they can be treated together. For a constant bundle length most of the cost is a function of the number of elements; the remainder is fixed per bundle. A change in bundle length would alter all encapsulation and assembly costs⁷. That is, the cost of encapsulation and assembly conasts of

- the fraction that is constant per bundle and is

$$g(p_E + p_A)_R W_R$$

where g is a fraction less than 1

 $(p_{\rm E} + p_{\rm A})_{\rm R}$ is the reference design cost for encapsulation and assembly per kg U

W is the weight of uranium in the reference design bundle, kg U

 plus the fraction that is constant per element and is

$$(1 g)(p_E + p_A) = \frac{W_R}{N_R}$$

where N_R is the number of elements in the reference design bundle.

For a modified design the cost of encapsulation and assembly per kg U is

$$(p_{E} + p_{A})_{R} = \frac{W_{R}}{W_{M}} \left[g + (1-g) \frac{N_{M}}{N_{R}} \right] f(L)$$
 (13)

where the subscript R refers to the reference design

the subscript M refers to the modified design

f(L) is a function expressing the effect of bundle length on encapsulation and assembly cost.

The fraction, g, is evaluated either from experience or by doing a detailed engineering cost assessment of the encapsulation and assembly processes.

3.2.3.4 Multiple Element Sizes

The effect on cost of more than one element size in a bundle, over and above the direct perturbations accounted for in sections 3.2.3.1 and 3.2.3.2, should not be large. Estimates — there is no large-scale Canadian production experience — vary up to 5%of the pelletization, encapsulation, assembly and zirconium alloy material costs. An increment of 3% in these costs is thought to be reasonable.

3 2 1 Order Size

Generally fabrication costs estimates are needed to assess the cost of energy from a reactor system. Hence, the fuel order assumed is likely to be large, probably not less than 25% of the capacity of the targest plant, i.e., about 3 months' production. The direct overhead cost/kg U for such an order should not be significantly different from that for a much larger order. Therefore for estimating purposes it is

⁶The weight of uranium in the right cylinder volume of a pellet.

⁷All CANDU bundles have been 0.5 m long; hence there is no Canadian experience to assess the relationship between bundle length and costs for encapsulation and assembly. As the length is increased, the cost of assembly jigs and handling per bundle must increase; the term f(L) in equation (13) is likely of the form $a + b \frac{L_M}{L_R}$ where a + b = 1

arbitrarily assumed that for any order greater than 25% of the annual plant capacity, the cost is independent of order size.

For order sizes less than 25% of the annual plant capacity, it is recommended that the pelletization, encapsulation, assembly and zirconium alloy cost components be incremented according to the relationship

$$\Delta(\text{Cost})/\text{kg U} = \underbrace{0.25 - \frac{\text{Order Size}}{\text{Annual Plant Capacity}}} (14)$$

where $\Delta(\cos t)/kg$ U is the incremental cost

(cost) is the cost for pelletization, encapsulation, assembly and zirconium alloy materials for a large order (>25% of plant capacity)

For pelletization, encapsulation, assembly and zirconium alloy materials the small order size cost is then $[(cost) + \Delta(cost)]/kg U$. Other cost components are unaffected by order size.

4. FUEL COSTS IN CANADA

To illustrate, the method is applied to estimate the cost of fuel fabricated in Canada. It is not claimed that the numbers are accurate but they are broadly representative of the Canadian situation using fabrication techniques as described by W.C. $Durant^{(6)}$ for which most practical production experience exists.

4.1 Pickering Natural UO₂ Design

As a substantial portion of CANDU fuel produced to date has been for the Pickering reactors, the Pickering natural UO_2 fuel bundle is selected as the reference design. The reference costs are expressed in 1972 Canadian dollars and derived for plants with capacities of 1 Mg U/d operating 250 days per year, 2 shifts per day at 100% load factor.⁸

The components of fuel cost which reflect the organization of the fabrication process in Canada are:

- 1. Purchase of U_3O_8 by the reactor operator.
- 2. Conversion of U_3O_8 to UO_2 by a powder preparation (PP) plant.
- 3. Purchase of zirconium alloy tubing, bar stock and sheet by a pelletization, encapsulation and assembly (PEA) plant.

- 4. Pelletization, encapsulation and assembly in a PEA plant.
- 5. Shipping.

The financial rules that reasonably reflect Canadian practice and current rates are summarized in Table 4. Production times and irrecoverable loss allowances appropriate to Canadian fabrication plants are itemized in Tables 5 and 6^9 .

The estimated costs of buildings, equipment, and annual operation for the PP and PEA plants are listed in Table 7 and 8.

The Zircaloy-4 material costs itemized in Table 9 are the costs to the fabricator for these materials. The Canadian PEA plants do some further processing on them; the labour for this is included in the PEA plant operating cost. The PEA plant adds an overhead charge for the materials it purchases from other suppliers.

From these data, the fabrication cost for the Pickering natural UO_2 fuel is estimated as demonstrated in Table 10.

TABLE 4. FINANCIAL ASSUMPTIONS FOR CANADA

1 CAPITAL CHARGES (% per sonum	,		
		Buildings	Equipment
Return on Investment before Yau	n	12.3	12,4
Departation		6.7	13,6
Property Tax		1.0	1,4
Insurance		1.0	1.0
	Total	21.0	27.0
Capitalization is assumed to be 30	17 debt and 7	Drie combing	
Depreciation is linear ^a			
Tax on revenue is 50% of grass			
Internation debt 6 %			
Reluin on equity 12 %			
2 WORKING LAPITAL CHARGES			
264 p maura			
3. OVERHEAD ALLOWANCE			
Research and development	s % of cu.	la l	
Warranty	1 % ut cm	ı	
Administration and marketing	10 % of cu	1	
Overhead	16 % of ca	1	
4 INTEREST			
7.5 yer annum			
4 % Depreciation per sumum - Expo	in	* n yes.*	

b Gut to reactor operator and includes fabricator-purchased materials and fabrication custs

⁹The enriched uranium route data, included for information, are based on Canadian industry subcontracting some of the fabrication processes to US plants (note Appendix 1).

⁸ The reference cost is not necessarily equivalent to the estimated 1972 fuel cost in Canada as the plant capacities, load factors and learning factors applicable .o the industry in Canada in 1972 differ from those .ised for the reference cost.

TABLE 5. PROCESSING TIME

Process	Elapsed Time Weeks
U ₂ O ₈ to natural UO ₂	6
U308 to UF6	6
Enrichment of UF6	13
Ship UF ₆ to converter	4
Enriched UF ₆ to UO ₂	6
Fabricate fuel (from UO ₂ powder to finished assembly) and deliver to customer's inventory	12

TABLE 6. IRRECOVERABLE MATERIAL LOSS ALLOWANCES

	Percent
Metal during fabrication	5
U3O6 during conversion to UO2	9.5
U ₃ O ₈ during conversion to UF ₆	0.5
UF ₆ during conversion to UO ₂	0.5
UO2 during labrication	0.2

TABLE 7. COSTS FOR 1 MgU/J NATURAL UO, FOWDER PREPARATION PLANT IN CANADA

	(10) ³ \$
Buildings	706
Equipment	944
Annual Operating	265

TABLE 8. COSTS FOR 1 MgU/d NATURAL UO2 PEA PLANT IN CANADA

	(10) ³ \$	
Buildings	9 50	
Equipment	1,950	
Annual Operating	1,920	

TABLE 9 COST OF ZIRGALOY-4 MATERIALS (for Fickering Reference D), at a Production Rate of 1 MgU/d)

	\$/kgU	We been the stars .
Tubing	7.24	
End Cap Bar Stock	1.44	
Other Zircaloy Sheet	1.08	

NOTE. Irrecoverable loss allowance and overhead allowance in FEA plant are not included

4.2 CANDU Fuel with Enriched UO₂

In the introduction to this paper it was stated that the relatively low cost for CANDU fuel is primarily attributable to its design, it was acknowledged that there is some cost saving associated with not having to deal with enriched UO_2 . The cost of enriched CANDU fuel can be estimated by the method presented here.

Several hundred CANDU bundles with enriched UO_2^{10} have been made in the Canadian fuel fabrication plants. This provides adequate experience for estimating the cost of fabrication plants producing large quantities of enriched UO_2 CANDU fuel. The estimated costs for plants producing 1 Mg U/d are listed in Tables 11 and 12.

The, cost of enriched UF₆ is based on supplying Canadian U_3O_8 to a plant for conversion to UF₆, paying the USAEC toll enrichment (note Appendix I) and shipping the enriched UF₆ to a Canadian PP plant for conversion to UO₂.

The calculation for the reference Pickering bundle containing 1.5% enriched uranium is detailed in Table 13.

By comparing Tables 10 and 13 it can readily be seen that the bulk of the cost increment for enriched over natural fuel is in the costs for enriched uranium and for powder preparation. The cost components that are truly a function of design, i.e., zirconium materials, PEA plant fabrication and shipping, actually increase only 7%%.

By comparing the cost of the enriched CANDU bundle with the cost of the light-water reactor enriched fuel bundle⁽⁴⁾, the cost reduction attributable to the CANDU fuel bundle design becomes apparent. Table 14 is reproduced from reference (2). The numbers for the CANDU bundles are slightly different from those listed here in Tables 10 and 13. The differences are due in part to escalation of the Canadian dollar from 1970 to 1972 and in part to the

¹⁰Up to about 3% U-235 in total U.

additional experience gained in Canada in the intervening time.

The method could also be applied to estimate the costs for UO_2 -PuO₂ and ThO₂-UO₂ fuels when the fabrication cost increments for these fuels are known

4 3 Modified Design Forecast

A more highly subdivided fuel bundle has been proposed for an improved CANDU-PHW reactor. Assuming the design is adopted it might be produced about 1976. The fabrication cost in 1972 Canadian dollars can be estimated by extrapolation from the reference costs for the reference design.

The reference design reference costs of Table 10 must be extrapolated to a 1976 base cost for the reference design. Table 15 lists the plant capacity and

load factors, production doublings and hence learning factors forecast in Canada⁽²⁾ The F and V costs in T able 10 are regrouped in Table 16, some additional subdivision has been included in order to deal with the subsequent design modifications. By applying equation (5) the base costs in Table 17 for 1976 production of the reference design in Canada are derived. The exponents q and r in equation (5) have values of 0.32 and 0.05 respectively⁽²⁾. The U₃O₈ cost is based on reported US market trends⁽³⁾ modified by assuming a dollar inflation of $3\frac{1}{2}$ per annum.

The parameters of the modified design are listed in Table 18. The effects of the design modification are summarized in Table 19; Table 20 gives the resulting fuel cost for the modified design from the assumptions stated.

TABLE 10.	CANDU REFERENCE DESIGN	NATURAL UO ₂ 1	FUEL COSTS (1972 Canadian dollars)

Component	Calculation	Cost: \$/kg U
U ₃ O ₈	Market price delivered to converter 15 \$15.75/kg U ^a	15 86
	Allow subsequent losses of 0.5 and 0.2%	
PP	Building charge is 21% of \$706,000 (V)	
(convert U ₃ O ₈	Equipment charge is 27% of \$944,000 (V)	
to UO ₂ powder)	Operating cost is \$265,000 { (V)	
	Working capital for operating is 26%/yr for 6 wk \$8,000	
	Apply overhead charge of 16% to total, and divide by annual production	
	of 2.5 (10) ^s kg U	3.22
Zircaloy-4	To costs listed in Table 9, add 5% loss allowance, apply PEA plant overhead	
Material Supply	charge and charge for working capital for 12 weeks at 26%/yr	
	Cost is $[7\ 24 + 1\ 44 + 1.08](1.05)[1 + \frac{12}{52}(.26)]/.84$	12.92
PEA	Building charge is 21% of \$950,000 (V)	
	Equipment charge 1s 27% of \$1,950,000 (V)	
	Operating cost is \$1,920,000 (about 90% F	
	plus working capital charge for 12 wks. \$115,000 and 10% V)	
	Total, add overhead charge and divide by annual production	13.15
Interest Charges	7% for 6 weeks on value of U_1O_8 during conversion to UO ₂ (on 15.86)	
•	7% for 12 weeks on value of UO ₂ during PEA (abrication (on 15.86 plus	
	3 22)	0.44
Shipping	Based on Canadian experience for distances up to 500 km	0.12
Total		45 71

^a This is the approximate US market price from reference (3).

TABLE 11 COSTS FOR 1 MgU/d ENRICHED DO, POWDER PREPARATION PLANT

	ан, анта арай-алардаар са арандаа, ала антану улуунунунунун д
	(10) ² \$
	and a set - hours and a set of the set of th
Buildings	1,930
Byuijanent	4,500
Annual Operating	1,235

TABLE 12. COSTS FOR 1 MgU/d ENRICHED UO2 PEA PLANT

	(10) ³ \$
Buildings	950
Bquipment	2,020
Annual Operating	2,280

TABLE 13. CANDU REFERENCE DESIGN, 1 5% ENRICHED UO2 FUEL COSTS (1972 Canadian Dollars)

Component	Calculation		Cost: \$/kg U
Enriched UFs	U ₃ O ₈ cost is \$15.75/kg U. From USAEC data (Appendix I), feed component is 2.544. Subsequent loss allowances are 0.5, 0.5 and 0.2%. Cost is (15.75)(2.544)(1.005)(1.005)(1.002)	~ 40 ,55	
	Convert U ₃ O ₆ to UF ₆ costs \$2.61/kg U(4). Increment by leed	i 12	
	component and subsequent losses of 0.5 and 0.3%	= 6,69	
	Toll enrichment. Cost is \$32,00 (1972 US) per SWU, 1 227 SWU		
	for 1.5%, subsequent losses of 0.5 and 0.2%	× 39,54°	
	Withdrawal, insurance, ship to PP (approximate)	- 0.89	87.67
PP (convert	Building charge is 21% of \$1,930,000 (V)		
UF ₆ to UO ₂)	Equipment charge is 27% of \$4,000,000 (V)		
	Operating plus working capital charge is 103% of \$1,235,000 (V)		
	Adding overhead and dividing by production of 2.5 $(10)^3$ kg U		13.78
Zircaloy-4 Material Supply	No change from table 10		12.92
рел	Building charge is 21% of \$950,000 (V)		
	Equipment charge is 27% of \$2,020,000 (V)		
	Operating plus working capital charge is 106% of \$2,280,090 (90% F, 10% V)		
	Apply overhead and total production		15.06
Interest	7% on \$40.55 for 6 weeks, \$47,24 for 13 weeks, \$87,67 for 10 weeks and		
	\$101.45 for 12 weeks		3,98
Shipping	Based on Canadian experience for distances up to 500 km		0.17
Total			133.65

" The 1972 Canadian and US dollars have been token at equal value.

TABLE 14. FUEL FABRICAT	YON PRICES
-------------------------	-------------------

Item	Amount 1970 \$/kg U				
	PWRb	Enriched-uranium CANDU ^b	Natural-uranium CANDU ^C		
UO2-powder preparation ^d	13 02	13.02	4.00		
Pelletization	15.10	8.80	8.00		
Materials other than UO ₂	29.65	13.58	10.05		
Encapsulation and assembly	17 62	7.97	19 00		
Shipping	0.72	0.72	0.11		
TOTAL	7 6 .11	44.09	27.76		

^a All for plants of 1 MgU/d capacity assumed to be operating in 1970 at 100% load. Amounts include allowance for working capital and overheads.

- b Costs are calculated in accordance with the methods of reference (4) except as noted later and are escalated from 1967\$ at 5%/year with the exception of materials costs which appear to have remained nearly constant since 1967. In arriving at the estimate for Encapsulation and Assembly of enriched CANDU bundles, the costs for buildings and operations have been substantially reduced below the figures given in reference (4) to allow for effects of mass production.
- ^c Estimated for 28-element 100 mm-diameter bundles on basis of Canadian experience and includes 5% allowance for research and development.
- d The conversion from enriched UF_6 to UO_2 is not required for natural-uranium CANDU fuel.

Year	Maxumum Plant Capacity (PP and PEA) Mg U/d	Plant Load	Number of ^a Doublings	Learning ^b Factor
1970	07	0 83	0	1.0
1975	16	0 67	2,5	0 78
1980	2.9	0.67	4.2	0.71
1985	5.0	0.67	5.7	0.67
1990	89	0.67	70	0 66

TABLE 15. FORECAST OF CANADIAN PRODUCTION PARAMETERS

^a The doublings (as integers) occur in intervening years. The fractions are a result of interpolation.

b The data are taken from reference (2). Equation (4) yields a reasonable fit when x = 0.4 and y = 0.3

	v	F	н	Total
U ₃ O ₈				15.86
PP	3.22		-	3.22
Zircaloy-4				
Tubing			9.58	
End Caps	-		1,91	
Other	-		1.43	12.92
Pelletization	1,95	3.66		
Encapsulation and Assembly	2.05	5.49		13.15
Interest				0.44
Shipping				0.12
				45.71

TABLE 16. DISTRIBUTION OF REFERENCE COSTS FOR NATURAL UO1 REFERENCE DESIGN (1972 Canadian Dollars/kg U)

TABLE 17. DISTRIBUTION OF BASE COSTS FOR 1976 PRODUCTION OF THE REFERENCE DESIGN (in 1972 Canadian Dollars/kg U)

	v	F	Н	Total	
U ₃ O ₈				17 40	
PP	3.18	•••		3 18	
Zircaloy-4					
Tubing		~	7.20		
End Caps	-	~	1,44		
Other		-	1.08	9.72	
Pelletization	1.93	2.82	 .		
Encapsulation and Assembly	2,03	4 23		11.01	
Interest				0 48	
Shipping				0.12	
				41.91	

Data used for equation (5)

Maximum plant capacity is 1.6 Mg U/d Load factor is 0.67 Learning factor is 0.77 q = 0.32 and r = 0.05

TABLE 18. MODIFIED FUEL DESIGN PARAMETERS

Number of elements	37
Nominal bundle diameter	100 mm
Bundle length	0.5 m
Weight uranium	18.8 kg
Element sheath OD	13.08 mm
Element sheath wall	0.42 mm
Fuel pellet OD	12.15 mm
Fuel pellet length	17.8 mm

TABLE 19.	EFFECT OF	DESIGN	MODIFICATIONS	ON COST	COMPONENTS
-----------	-----------	--------	---------------	---------	------------

ltem	Equation	Valu	e O	l Constant	Cosi \$/kg U
Tubing	8	a"	*	0.229	8.58
End Caps	10	ь"	×	0.033	171
Other Metal	11	C	٠	4 255 (10) ³	1.13
Pelletization	12	f	*	0.5	6.24
		1-1	=	0.5	
Encapsulation	13	£	=	0.15	7.97
and Assembly		ŀg		0.85	
		f(L)		10	

5. SUMMARY

A method of estimating current and future CANDU fuel costs has been described. The equations developed are sufficiently generalized that the method can be applied to a new fuel fabrication industry where the fabrication techniques, the costs of buildings and equipment, the cost of labour and the financial practices are quite different from those in Canada. Once the constants in the equations have been evaluated for a new fuel fabrication industry producing a reference CANDU fuel design, the method can be used to estimate the effect on fuel cost due to expansion of the industry, additional experience and fuel design modifications.

When applied to Canadian industry the method estimates the known fuel costs with adequate accuracy. It is also apparent that the low fabrication cost for CANDU fuel is due largely to the simple design and small number of components. Even when enriched, the cost of CANDU fuel is significantly lower than the cost of light-water reactor fuel.

TABLE 20. ESTIMATED FABRICATION COST FOR 37-ELEMENT CANDU-PHW FUEL BUNDLE PRODUCED IN CANADA IN 1976 (1972 Canadian Dollars)

		\$/kg U
U308		17.40
۲P		3.18
Zircaloy-4		
Tubing	8 58	
End Caps	171	
Other	1 13	11.42
Pelletization		6.24
Encapsulation and Assembly		7.97
Interest		0.48
Shipping		0.19
		46.81

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Cost of Enriched UF₆

Enriched UO_2 is prepared by converting natural U_3O_8 to UF_6 , enriching it, then converting the enriched UF_6 to enriched UO_2 . The Canadian nuclear reactor operator (or the fuel fabricator) can acquire the enriched UF_6 by either

- (A) entering into separate contracts for
 - (i) the supply of U_3O_8 ,
 - (ii) the conversion of U_3O_8 to UF_6 .
 - (iii) the enrichment of UF_6 by the USAEC using its toll enrichment service.⁽⁵⁾

or

(B) contracting with the USAEC to supply enriched UF_6 .

If arrangement A is chosen, then the cost of the enriched UF_{ϕ} is computed as follows:

- A1 The weight ratio of natural U feed to enriched U product is given by the feed component, $C_{\rm F}$, for the desired enrichment level as listed in Table I-1. This must be increased to allow for subsequent fabrication losses as itemized in Table 6. The cost of U_3O_5 per kg of enriched U required is then [cost of U_3O_5/kg natural U] $C_{\rm F}$ (1.005)² (1.002).
- A2 The cost of converting U_3O_8 to UF₆ from reference (4), is \$2.61/kg natural U. Hence, in terms of the enriched U required, the cost is \$2.61 C_F (1.005) (1.002)/kg enriched U
- A3 The cost of toll enrichment is currently \$32.00 (US) per separative work unit (SWU)⁽⁵⁾. The number of SWU's for the required enrichment level is listed in Table 1-1. The cost of enrichment is then \$32.00 SWU (1.005)(1.002)/kg enriched U in 1972 US dollars.
- A4 There are charges for withdrawal, container rental, insurance and transport to the UF_6 to UO_2 conversion plant. These can be accurately assessed, but all except the first are dependent on the location of the UO_2 powder plant. For the Canadian market, the cost of these items can be adequately estimated by applying the relationship

Cost, $\frac{1}{2} \log \frac{1}{2}$ Cost, $\frac{1}{2} \log \frac{1}{2} \log \frac{1}{2}$

If arrangement B is selected, then the cost of enriched UF_6 is computed from:

B1 The USAEC scheduled base charge for the supply of natural UF₆ per kg natural U required is currently \$23 46 (1972 US dollars)⁽⁵⁾. The cost of natural UF₆ per kg enriched U required is then $23.46 C_F (1.005)(1.002)$.

- B2 The cost of enrichment as in A3.
- B3 Miscellaneous costs as in A4.

TABLE I-1. STANDARD TABLE OF ENRICHING SERVICES

Product Enrichment	Feed Component Factor, C _F	Separative Work Units
wt% U ^{2 3 5}	kg U natural	SWU/kg U product
	kg U product	
0 20	0	0
0.25	0.098	-0.100
0.30	0,196	-0.158
0.35	0.294	-0.189
0.38	0.352	-0,197
0 40	0,391	-0.198
0.42	0.431	-0,197
0.44	0.470	-0,194
0.46	0.509	-0.189
0.48	0.548	-0.182
0.50	0.587	-0.173
0.52	0,626	-0.163
0.54	0.665	-0,151
0 56	0 705	-0.137
0.58	0.744	-0.123
0.60	0.783	-0,107
0.65	0.881	-0.062
0.70	0.978	-0 012
0711	1,000	0.000
0.75	1.076	0.044
0.80	1,174	0.104
0.85	1 272	0.168
0.90	1.370	0.236
0,95	1.468	0.307
1.00	1.566	0.380
1 10	1 761	0.535
i 20	1 957	0.698
1.30	2.153	0.868
1.40	2.348	1 045
1.50	2.544	1.227
1 60	2.740	1.413
1.70	2.935	1 603
1.80	3.131	1.797
1 90	3.827	1 994
2,00	3.523	2,194
2 20	3.914	2.602
8,40	4 305	3.018

^a This table has been extracted from reference (5).

Product Enrichment	Feed Component Factor, C _F	Separative Work Units
wt% U ²³⁵	kg U natural	SWU/kg U product
	kg U product	
A 6A	4 400	9 4 4 1
2.00	2 002 [,]	9 873
2,80	0.VAQ 5.470	3.071
3,00	0,479 E 671	4.000
3.20	D.871	5,790
3.4U 9.60	0.30Z 6.654	J 171 K 698
0.9V 9 8A	0.004 7 D45	6 0900
0.0V 4.00	1 VIJ 7 198	6.000 8 514
4.UU 4.50	(,930 9.416	0.041 7 see
1.00	0 202 0,202	(.03V 8 851
0.VU 5.50	7,573 10 979	0.091 10.099
9.0U 6.00	10,372	11 903
0.00	11.304	11 200
7,00 8.00	18.307	15,007
0.00	10,207	18 492
9,00 10.00	17,241	20,444
10.00	00 400 0 1 1.41	95 789
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18.00	34.834	40.124 Ac NAT
20.00	38.746	40./4/ 69.960
20.VV	47,032 50 91 <i>4</i>	71 084
84.34 35.00	00,317 49 100	11,UV1 82 816
40.0U	00.IV6	00.919 02 A14
4V.UU 50.00	11,001	100 944
00,00 60.00	77.400 117 A95	144.077 148.025
100.00 70.00	117,VZQ 196 505	140.800 177 900
10.00	100.080 186 1 <i>84</i>	114.0VA 900 405
95,00	100,109	474.QUU 918 809
53,UU 55,UU	100 242	410,074 997 941
90.00	170,734	841 391 999 70c
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TABLE I-1. (continued)

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DECISION CRITERIA FOR BUILDING FUEL FABRICATION PLANTS

E TRAUWAERT J. VERRÄVER Rapporteur E JONCKHEERE

ABSTRACT

The decision to build a fuel fabrication plant must depend on many criteria including economic ones.

In a first part, the fuel fabrication cost for a reference plant is evaluated and an analysis is given of the influence of various parameters such as the size of the plant and its expected load factor, the rate of the financial charges, the warranties of the fuel elements, etc ... Possible external economies are also taken into account.

In a second part, the world market conditions for the years to come are estimated through the description of a most probable "average" fabrication plant.

By comparison of the world market conditions with the local ones, it is then possible to bring out a few criteria measuring the economic and political interest to build a nuclear fuel fabrication plant.

1. INTRODUCTION.

In its "Forecast of the growth of Nuclear Power" $\begin{bmatrix} 1 \end{bmatrix}$, the USAEC estimates the cumulative capacity of nuclear power in the western world (i.e. excluding USSR, P.R. China and Eastern Europe) to reach 95 GWe by 1975, 250 GWe by 1980 and more than twice this figure by 1985. This large increase in nuclear power over the next decade will call for a similar increase in volume of fuel fabrication. The questions thus arise of where, when and which fuel fabrication plant to build.

As fast reactors are not expected to get any significant share of the market before approximately 1985 [2], the above power requirements will mainly have to be met by LWR.s, HWR.s and HTR.s. Hence a subsidiary question : should the fabrication plants be specialized in fuel elements for one or more types of reactor or should they be completely polyvalent ? In this paper, we have chosen the solution of fully specialized fabrication plants, not only because this approach was easier to handle, thus giving more meaningful results, but mainly for the following reasons ·

- 1° The present fabrication market is such that most fuel suppliers are closely linked with the reactor suppliers, so that a specialization in one type of fuel is almost natural.
- 2° HTR fuel is so different from the other types of fuel that it is almost certain that it will be manufactured in a separate fabrication unit.

Hence there are good indications that most future fabrication plants will be specialized in the fuel for one type of reactor. The fact that the fabrication plants have some flexibility, that enables them to produce other types of fuel, with some performance penalty, will be disregarded in this model as a minor effect.

This leaves us with our first question about the size of fabrication plants, as a function of time and location, and this will be the object of this paper.

Although we want to put this question in an international context, it is not our ambition to develop an overall optimization model for all the fabrication plants to be built over the world.

We will rather tackle this problem from the point of view of the marginal investor : given well defined world market conditions for nuclear fuel fabrication, what must be the policy of a potential investor ?

To define further the problem, we will assume that the investor wants to do a benefit of some kind.

Thus given a certain volume of fuel fabrication one might supply, we shall investigate whether it is beneficial to invest in a fabrication plant rather than to buy on the world market. More precisely we will indicate, under certain assumptions, some minimum demand levels under which no benefit can be expected from a fabrication plant investment decision, given the offer on the world market.

A more advanced approach of the problem would be to try to calculate the maximum benefit one can expect from an investment decision. This is a very complex problem that lies outside the frame of this work.

For the definition of the benefit, many options are possible. At one end of the spectrum, we have the accountants definition in which the total balance of both solutions (plant construction or fuel import) is made. At the other end, we find the point of view of someone anxious about the balance of payment, taking only those expenses into account that are to be paid in foreign currency. We can also imagine a local development agency who is concerned with the economic and social development of a district and is prepared to pay some subsidy considering the contribution of the firm to the local welfare (external economies). These three options will lead to different investment criteria.

The aim of this paper is more to illustrate the problem than to quantify it with precision and to every detail. Therefore, we shall restrict the discussions to the fuel for one type of reactor and focus the analysis on the variations of the costs with various parameters and not on its absolute value.

2. DEFINITION OF THE PRODUCT.

2.1. SCOPE OF SUPPLY.

A fuel manufacturer may offer, in addition to fuel fabrication, a large spectrum of services covering the whole fuel cycle from the procurement of base nuclear material to the disposal of spent fuel. For example, G.E. will offer three kinds of fuel services packages [3] :

- 1. fuel fabrication service, including fuel design and conversion of UF_6 to UO_2 ;
- 2. initial fuel service, i.e. fabrication service plus procurement of yellow cake conversion to UF_6 , procurement of enrichment services and fuel management engineering;
- 3. fuel cycle service, that is initial fuel service plus spent fuel transportation and reprocessing and recycle or resale of recovered uranium and plutonium.

Of course, the warranty offered by the fuel manufacturer will be proportionate to the scope of fuel services. The most comprehensive fuel cycle service will practically carry with it the warranty of the fuel cycle cost, calculated under nominal plant operating conditions and with a conventional financial charges rate. At the other end, the warranty associated to the fuel fabrication service will be limited to the mechanical integrity of the fuel and possibly its heat output.

For the purpose of this study, we shall adopt a still more restricted definition of fuel fabrication by taking away the fuel design from the scope of supply and the corresponding charge from the fuel fabrication cost.

We define therefore fuel fabrication as covering the following steps :

- acceptance of enriched UF_6 .t the proper enrichment ;
- procurement and acceptance of rod cladding ;
- ~ fabrication or procurement of other rod and element hardware items (end caps, grids, heads and feet, ...);
- conversion of UF₆ to UO_2 ;
- pellet fabrication (fuel pelletizing, sintering, grinding);
- rod fabrication (fuel loading and end cap welding);
- element fabrication (element assembling and packaging);
- recycle or recovery of fabrication scrap ;
- controls required at each fabrication step to ensure the quality of the final product.

The <u>fuel warranty</u> associated with the product will cover good workmanship and materials exclusively. The mechanical design of the fuel required for its fabrication may have been made available through a licence agreement or by a separate department of the same company which performed the necessary research.development and qualification work. Similarly, the nuclear design or fuel management engineering, may have been performed by an external engineering company or a separate department.

The associated charges have to be added to the fabrication cost as defined here before comparison with standard market prices We preferred to leave them out because :

- the charge for fuel design is very difficult to evaluate and will depend greatly on commercial practices and on the amount of fuel previously fabricated;
- 2. the fuel management can be contracted to an engineering firm rather than to a fuel manufacturer ; besides the associated charge will depend mainly on the reactor characteristics.

2.2. FUEL CHARACTERISTICS.

The reference fuel element considered in this study is intended for a PWR of Westinghouse design in the 1000 MWe range $\begin{bmatrix} 2 \end{bmatrix}$. This element is of cluster control rod design and has the following characteristics :

Fuel characteristics

fuel

composition	^{UO} 2		
form	pellets		
outer diameter	9.32 mm		
active length	3650 mm		
smeared density	93.5 %		
rods :			
cladding material	Zircaloy		
outer diameter	10.72 mm		
wall thickness	0.61 mm		
overall length	4064 mm		
fuel rods/element	204.		

3. REFERENCE FUEL FABRICATION COST DATA.

3.1. BASIC ECONOMIC DATA.

For this study, we did not evolve an evaluation of our own for the cost of fabrication of PWR uranium fuel. Such an evaluation, to be meaningful, implies in fact a complete design of the fabrication plant and this design, even if preliminary and coarse is costly and time consuming.

Instead we have resorted to some of the many studies on the subject which are available. Table I shows a comparison of three studies very similar in context ; they are all :

- relevant to the United States ;
- made by non fuel suppliers ;
- expressed in 1967 dollars on average ;
- centered on 1 t/day plant.

The first reference, FABOCST $\begin{bmatrix} 6 \end{bmatrix}$, is in fact a computer code written by ORNL, which we have adapted on our computer and in which we have entered the data of § 2.2.

The second reference is a study made by Jackson and Moreland and Stoller Associates on technical and economic aspects of LWR's [2].

The third set of data comes from a comparative study of the costs of oxide fuel elements performed by BNWL in 1966 $\lceil 9 \rceil$.

In spite of their general identity of context, the three studies show quite large discrepancies in results. The difference between the first two sets of data seems to be chargeable to the way of reporting the results, the final figures being quite similar.

The low level of the figures in the third set of data can presumably be attributed to an underestimate of the cost of equipment design and of the indirect charges. Indeed, the A.D. Little study on the competition in the nuclear power supply industry [3], which uses these data, quotes a cost of "entry" - covering research, development and qualification as well as design - of 5 to 10 M\$ and part of this should be added to the capital cost in Table I to make its context coherent with the other two data sets. The same A.D. Little study adds 2 M\$/year to the operating charges given by BNWL to account for General and Administration.

Taking these points into account, it seems that the FABCOST data may be accepted as the reference ones, and we have used this code for the parametric study reported below.

It should be noted that the capital costs and operating charges of Table I are exclusive of the conversion unit for which no data, except the \$/kg cost, are given in FABCOST or in the RNWL study. According to WASH 1082, the conversion unit would add some 4.2 M\$ to the capital expenditures and 1 M\$ to the operating charges. Also to be noted is that the operating charges are exclusive of any cost or charge relative to uranium and of the cost of direct materials (hardware). The latter is evaluated at about \$ 28/kg U in FABCOST and at about \$ 22/kg U in the other two studies. Since a large part of this cost is associated with element hardware (grids, head, feet, .), part of this discrepancy may certainly be attributed to differences in reference fuel design on which little information is generally given.

3.2. PARAMETRIC SURVEY OF FABRICATION COSTS.

3.2.1. Main parameters.

In the cost survey, the following were kept constant :

- type of fuel (see § 2.2);
- type of fabrication plant (hooded);
- type of equipment ;
- mode of operation : 260 operating days per year ;
- overall reject rate : 5 %.

In order to investigate the febrication costs for a potential investor in a given country and the possible market conditions he will have to face, we have varied the plant size, its load factor, the rate of fixed capital charges and the level of operating charges.

1. Plant size.

The plant capacity has been varied from 0.2 to 5 tU/day; the latter corresponds to the highest capacity in the near future and the former is close to the smallest size which could be considered in practice.

2. Plant load factor.

The load factor has been varied from 0.8 down to 0.6; this low value could occur if one has preferred, for practical or for tactical reasons, to build a plant larger than strictly needed at the beginning of its operation.

3, Rate of fixed capital charges.

It includes amortization, return on investments, taxes and property insurance. In most American studies, the rate taken is between 25 and 30 %. In order to reflect the various helps that a potential investor could get from national as well as international authorities to foster the introduction of a nuclear fuel industry in a new country, this rate has been varied from 25 down to 10 %.

4. Level of operating charges.

Typically about two thirds of the operating charges are salaries, wages and related social expenses, the rest being power, utilities and other consumptions. To account for the difference in wages between the US and other countries, we have introduced a coefficient multiplying the operating charges. This coefficient has been varied between 0.5 and 1.0.

3.2.2. Results.

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'A 'The results of the survéy are shown in Figures 1 to 5 for the different fixed charge rates. In each figure, the shaded areas represent plants of equal size but with different load factors and different operating charges coefficient; the lines sloping downwards from left to right and connecting together the shaded areas represent plants with equal load factor and operating charge coefficient but with different sizes.

All the costs on these graphs are expressed in relative units, the reference cost (100) being the cost of fabrication in a 1000 kg/day plant operating at 0.8 load factor under a 25 % fixed charge rate and nominal operating charges (coefficient = 1.0). The absolute value of this fabrication cost is \$ 73/kg U. This figure reflects 1967 conditiona; to adjust it for today conditions, one should account for $\{2, 5\}$:

- a) escalation of labour and materials prices at 5 6 % per year on average;
 - b) improvements in production techniques ;
- c) changes in quality control requirements ;
- d) changes in safety and safeguards requirements ;

e) differences in the scope of supply (§ 2.1).

If the net price increases until 1972 is taken as 25 % (normal price escalation minus technical progress), the reference fuel fabrication cost is \$ 91/kg U in today value. Points c) and d) above are difficult to evaluate. As to point e), it seems that an allowance of about \$ 10/kg U might have to be added for R & D, design, warranty and working capital [2, 3]. This gives a real price of \$ 100/kg U, in round numbers (without fuel management engineering).

3.2.3, Analysis.

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From the Figures 1 to 4, one may see that, in the range considered, the most important factor is the plant capacity. Thus, even with very good operating and financial conditions (e.g. 10 % fixed charge, 0.8 load factor and 0.5 operating charge coefficient), a 200 kg/day plant cannot compete with a 1 t/day plant under nominal or even bad conditions (25 % fixed charge, 0.6 load factor, 1.0 operating charge coefficient).

This last figure is given for the sake of illustration; it does not correspond to the reference scope of supply and will not be used in the following analysis.

The relative influence of the load factor, operating charges coefficient and fixed charge rate decreases with increasing plant size as shown in the table below :

Fabrication cost structure

<u>plant size</u>	<u>capital</u>	operating	hardware and conversion		
(t/day)	(%)	(%)	(%)		
0.2	28	32	40		
1.0	24	27	49		
5.0	20	22	58		

This effect is largely due to the fact that tubing cost, which is a large part of the hardware and conversion cost, is practically independant of fabrication plant size since it is an external supply.

In this respect, it should be noted that FABCOST assumes for the other hardware cost (end caps, springs, element hardware) and for the conversion cost, a significant variation with plant capacity. This would not be applicable if these items are purchased on the market or supplied by the licensor of fuel design ; in this case, the tilt in relative fabrication cost structure from small to big plants would be even more pronounced.

For the purpose of interpolating in the graphs of Figures 1 to 4, one may use the following fitting formula :

 $K = 35.9 + [12.6 (1 + 1/L) OC + 71.1 A/L + 13.6] C^{-0.53}$

where :

- K = relative fuel fabrication cost (%)
- L = plant load factor
- OC = operating charges coefficient
- A = rate of fixed capital charges
- C = plant capacity (tU/day).

The standard deviation is 0.75 %.

The values of the coefficients entering the formula depend on the form which was entered in the fitting programme and on the range of variation of the parameters. One should, therefore, be very careful when trying to interpret it in terms of hardware cost, investment cost and operating charges.

3.3. EXTERNAL ECONOMIES OF FUEL FABRICATIONS.

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3.3.1. Fuel supply reliability and independence.

It is a well known fact that the nuclear power system provides a nation with a considerable degree of power supply reliability [4], not only because a nuclear reactor has an important working autonomy, but also because the storage of nuclear fuel is cheaper than the storage of any other conventional energy sources (oil, coal, ...). It is also true that this advantage is often paid, at least for some time, by an increased dependence from foreign nuclear plant suppliers, fuel suppliers and reprocessing facilities. Does the construction of a local fuel fabrication plant bring any change in this situation?

From the point of view of supply reliability, the storage of enriched uranium will be cheaper than that of fabricated fuel, if only because of the lower added value. However, this consideration would only play a significant part if one needs a lot more working autonomy than what is already built in the nuclear reactor system.

The dependence on foreign fuel suppliers will be limited to purchase orders to suppliers of natural uranium and of enrichment services and, possibly, to license agreements with suppliers of fuel design instead of purchase orders for fuel fabrication services. It does not seem that this will change significantly the degree of dependence from foreign countries.

The only case in which the building of a fabrication plant could change considerably the dependence picture of a nation is that of a natural uranium reactor type : indeed there would be many potential suppliers of natural uranium on the international market, and, in some cases, this material could even been produced locally.

Thus, the external economics resulting from the increased supply reliability and the reduced foreign dependence are not likely to be important except when the adopted reactor type uses natural uranium.

3.3.2. Environmental aspects.

The implantation of a new fabrication plant on a site may have environmental effects of three types : it will change the landscape of the site, it may be a source of radiation and it produces radioactive waste.

The changes to the landscape are almost inevitable but they are similar to those produced by any other industry. It is the job of the architects to try to integrate the plant in their matural environment and nuclear fuel fabrication plants do not, seem to give rise to special difficulties. On the contrary, the high standards of cleanliness adopted throughout the nuclear industry is found ' practically in no other industry.

From radiation point of view, it will be appreciated that unirradiated low enriched uranium has a very low radiation intensity. Moreover, the safety requirements are, nowadays, well established and fairly easy to fulfil. This explains the very low level of radiation hazards produced by the nuclear industry in the past and all indications are that it will remain the same in the future.

The waste disposal is presumably the only tricky environment problem of nuclear industry. However, this problem is certainly far less severe at the head end of the fuel cycle that at the tail end where one has to dispose of all the radioactive fission products. Furthermore, it is only the "perpetual care" which gives rise to difficulties and this problem has to be tackled on a worldwide basis. The waste disposal problem is therefore relatively minor for fuel fabrication and it will be but marginally affected by the location and sizes of the plants.

3.3.3. Impulse for scientific and economic development.

It sounds almost trivial to emphasize the scientific and economic secondary effects of the so-called spear-head techniques. This idea is so broadly accepted that organizations like the NASA tend to base the justification of their whole program on the external economies of this type it is supposed to produce.

The impulse on the scientific development associated with the construction and operation of a fuel fabrication plant depends strongly on the origin of the fuel and plants designs. If these are developed locally, the impulse will be very significant since it implies very qualified engineers and physicists working on an extensive research, development and qualification programme.

- If these are of foreign origin and obtained through licence or association agreements, the impact will be practically limited to the formation of highly qualified technicians and workers and this is even more so if the fuel hardware is wholly or partly purchased abroad, for instance, from the licensor.
- From economic point of view, a fuel fabrication plant can, of course, play a part in the industrialization of a disadvantaged district, provided enough qualified personnel is available. However, this economic benefit must be compared with the one resulting from building any other industry of similar importance on the same site. It may then become clear that a more labour intensive industry is more advisable for the economic development of that district.

3.3.4. Concluding comments.

From the above remarks, it is seen that the external economies, although very much dependent of local conditions, are probably more limited than one could expect. They are also very difficult to evaluate with some accuracy.

Obviously this problem is not unique to the nuclear industry and does not prevent people from taking decisions on the basis of such effects whether they are correctly evaluated or not. It is, therefore, better to consider and to analyse them seriously with the help of experts in this field rather than to just forget about them. 4. WORLD MARKET CONDITIONS.

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It is a very hazardous business to try to predict the future world market conditions. However, on the basis of some realistic assumptions, one can propose estimates of different economic parameters.

We shall start from the assumption that the world market price will be governed by the price from an American supplier working under nominal economic and commercial conditions.

We have therefore to look at the fuel fabrication demand for US suppliers at the time considered (say about 1980), the sizes of the plants satisfying this demand and the corresponding fabrication cost.

Estimates of the cumulative capacity of nuclear power plants for the United States and for the Western world are given in Table II $\begin{bmatrix} 1 \end{bmatrix}$. It will mainly correspond to thermal reactors as the fast reactors are not likely to become commercial before 1985.

Let us suppose, as an illustration, that all reactors are of the PWR type, with a typical heavy metal inventory of 80 metric tons per 1000 MWe, which takes three calendar years to burn to the nominal discharge irradiation. Thus, according to the estimated power capacities for the year 1980 (Table VI), the total refuelling capacities of plants working 260 days per year with a 0.8 load factor would be 19 t // day for the USA and 36 t // day for the Western world. If we further suppose that the yearly increase in installed power calls for an increased fabrication capacity one year in advance, we will have, in addition, the following first core fabrication capacities : 9 $t^{\#}/day$ for the USA alone and 16 t^{\pm}/day for the Western world.

Other authors $\begin{bmatrix} 3 \end{bmatrix}$ and $\begin{bmatrix} 5 \end{bmatrix}$, are quoting estimates of the fabrication demand for the next few years :

Yeaŕ	Babcock and Wilcox [®] reactors	Combustion Engineering reactors	General Electric reactors	Westinghouse reactors	
1970	0.1	0.3	1.3	0.7	
1975	1.2	0.8 [°]	4.0	2.8	
1980	3.1	2.2	9.2	7.1	

DEMAND [metric tons per day]

These demands are based on 250 full production days per year. Today's latest facilities have an estimated capacity between 2 and 5 L'day [3], [5] although these plants may be far from saturated.

equivalent PWR fuel,

From the above considerations it seems realistic to accept that the average fabrication plant in the United States around 1980 will produce the equivalent of 5 t/day of PWR fuel during 208 equivalent full production days per year (or 260 days at 0.8 load factor). This figure is corroborated by a non to far off 4.3 t/day capacity in 1980 found by an optimization study based on constraints of estimated demand, share of the market between existing manufacturers, intitial load factors and specific variations with time. This result was published in $\begin{bmatrix} 2 \\ 2 \end{bmatrix}$.

According to Figure 1 with 25 % of fixed charge rate and unit operating factor, the corresponding cost would be $\underline{64}$ % of our reference cost (1 t/day) or $\underline{58}$ \$/kg U in 1972 currency. We will accept this as the reference world market price.

Although this price was calculated as a production cost, we will not consider any supplementary profit margin because remuneration of the capital is already included in the rate of fixed capital charges.

Moreover, to be representative of a market price on any place of the world, this USA price should be increased by proper transport cost. The error resulting from neglecting this factor is of the order of 1-2 $\frac{1}{2} \log U$ (to be compared with [2] quoting 0.5 $\frac{1}{2} U$ for shipping within the USA).

However, we will also neglect the transport cost of enriched uranium, hardware and all other imported materials when considering the local production plants (cfr. infra). In the comparison between these two solutions, both types of transport costs will tend to counter-balance each other and the remaining error is believed to be very small indeed.

5. DECISION CRITERIA AND PLANT SIZE.

5.1. STATING THE PROBLEM.

In the preceding chapters, we have described the variation of the cost of fuel fabrication, as defined in this study, with various local parameters and the price of fuel fabricatic 1 on the world market.

The question that now arises is : given these data, is it worthwhile to build a local fabrication plant rather than to buy fuel fabrication abroad ?

In this chapter, we shall consider this question only in terms of minimum plant size and we shall leave the problem of the corresponding fuel demand to the next chapter.

The answer to this question depends on the particular aspects of the fabrication plant which are regarded as most important by the manufacturer as well as by the national authorities.

The following aspects may be considered :

- the fabrication cost level relative to market price ;
- the effect on foreign currency balance ;
- the external economies from the point of view of national economic development.

5.2. FABRICATION COST LEVEL.

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When looking at the Figures 1 to 5, one sees easily that the 64 % level does not necessarily correspond to a 5 t/day plant wherever in the world. Indeed, apart from the size of the plant, the fuel fabrication cost depends on a lot of other factors and the quoted 58 \$/kg U corresponds to conditions in the United States, which are probably quite different from anywhere else in the world. This is obvious for the United States manpower which is highly qualified but also very expensive. The lower salaries of another country, if not offset by the increased number of workers needed, will result in lower <u>operating cost</u>. The extent to which the operating cost could be reduced depends on local conditions, bearing in mind that charges for energy and other consumptions, which are probably lass sensitive to local conditions, are also part of the operating cost.

In many cases, one may also expect to get lower <u>capital</u> charges. Indeed, if the investor is the national government or a public organization, the state may forgo, in some cases, the taxes and the remuneration of the state-owned capital.

The manufacturer might also hope to get loans from international banks of organizations. Mr. Khan [8] has quoted a few examples of nuclear investment for which the interest rate did not exceed 6 %. Some saving on capital cost may also be expected from lower building charges due to local supply of the majority of the manpower.

However, the production equipment will probably have to be imported to a large extent and it will be penalized by transport costs.

- A third parameter to be specified for the determination of the local fabrication cost is the <u>loed factor</u>, the value selected for the characterization of the world market price being 0.8.
- It seems improbable that a higher value than 0.8 could be reached by a new fabrication plant as this value represents practically the working level allowing only for halts between batches and for normal unprogrammed breakdown of production.

Furthermore, one should bear in mind that starting with a new technique has always yielded a lot of unforeseen difficulties and temporary setbacks so that it seems realistic to adopt in this case a load factor somewhat lower than for normal mature equilibrium production.

Let us now try to apply these considerations in a more specific example. Let us imagine that the operating charges are 50 % of those valid in the USA, that the capital fixed charge rate is 10 % and that, for a first fabrication plant, the load factor is 0.6. The question is now how big a plant do we have to build in order to break-even with the assumed world market price of 58 \$/kg U or 64 % of the reference 91 \$/kg U ? As shown on Figure 4, a break-even price occurs for a plant of 2 t/day.

This capacity may look already very important for a new production plant However, it should be realized that this plant is 2.5 times smaller than the average world market plant and that it, nevertheless, competes in cost thanks to assumed local conditions.

FORBIGN CURRENCY BALANCE.

When having to choose between buying foreign fuel elements or building their own fuel fabrication plant, most countries will carefully consider the consequences of both solutions on the foreign currency balance.

Let us assume for a moment that the only problem is to avoid deterioration of this balance. Hence any investment decision that would ensure the same fuel supply at lower cost to the foreign currency balance than the import of the fuel from a foreign country would be adopted.

This would mean that we have to look for an investment in fabrication plant of such a size that the part of the unit fuel fabrication cost to be paid in foreign currency does not exceed the 64 % level (corresponding to the world market conditions) of the reference cost. The problem is thus to evaluate that part of the fuel cost to be paid in foreign currency.

We shall first recall that the total cost is the addition of the capital charges, the operating charges and the hardware and fuel preparation cost.

Capital charges :

let us suppose for example that 50 % of the financial charges are related to the buildings and the rest to the equipments. As approximation, we assume that all equipment will have to be paid in foreign currency whereas the buildings are paid with national currency. If we go back to the parameter values of the competitive price criterion, it would mean that the part of the fixed capital charges to be paid in foreign currency amounts to 10 % on half of the capital cost, i.e. 5 % of the total capital cost.

Operating charges :

the operating charges consisting mainly of salaries and partly of charges for energy, utilities, etc ...; as a first approximation, we may consider that all operating costs are paid in national currency. Hence for the evaluation of the balance of foreign currency, we put the operating coefficient equal to zero.

Hardware and fuel preparation :

as another approximation - that will however partly compensate the previous one - we may assume that this part of the total cost is entirely paid with foreign currency. It is a matter of judgment for the possible manufacturer to evaluate which part of the hardware and fuel preparation could be provided locally and how far the above assumption would thus be invalidated.

Summarizing the above arguments, we take the following parameter values for reading our graphs :

rate of capital fixed charges	:	5 %
operating charge coefficient	:	0.0
load factor	;.	0.6

The first two values are exceptionally low, but one must bear in mind that they do not reflect the actual cost but only the foreign currency part of the total cost. For these very low values, a special graph was established (Figure 5). It is characterized by the absence of any operating charges and by a set of low rates for financial charges varying from 5 to 15 %.

Comparing the above parameter values with the 64 % level of the reference cost, we find from Figure 5 that, in order to break-even with the assumed world market conditions on basis of foreign currency alone, the capacity must be 0.45 t/day. With this plant size, the fuel fabrication would cost 58 \$/kg U in foreign currency and 30 \$/kg U in equivalent home currency. If, for the hardware, we had assumed that only the tubes and about half the element hardware were imported, the corresponding minimum plant size would have been 0.2 t/day and would have yielded an equivalent national currency cost of 63 \$/kg U in addition to the 58 \$/kg U in foreign currency.

5.4. EXTERNAL ECONOMIES.

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As we have seen by the survey of some of the external economies, the comparison of the local fabrication cost with the fuel price on the world market does not express the total interest for a country to build its own fabrication plant. Nor is this aspect fully taken into account by the consideration of the foreign currency balance.

Indeed, generally these approaches take into account neither the external economies nor the economic national planning aspects. The difficulty of evaluating the impact of these factors arises because our quantitative information is poor and also because it supposes an exact definition of what is good for the society. Clearly this ethical problems cannot be solved by engineers and economists alone. But we may assume that it can be solved by governments and that the estimated extra plant economies of an activity will result in the grant of governmental subsidies.

The subsidies may take many forms : the government may offer a low interest lean for part of the construction cost, for instance that associated with local manpower and supplies, or take over the interest charges of a bank loan, or pay for the research and development work. These forms of governmental help can be treated in our model by decreasing the fixed charge rate of the operating charge coefficient as we have done above (cfr. § 5.2).

One may also notice that the balance of foreign currency criterion (§ 5.3) normally leads to total fabrication cost above the market price ; if the customer is not to be penalized, the government will have to pay the difference.

To avoid duplication, we may simply assume for the external economies aspect that the government agrees to pay a percentage of the total fabrication cost. Let this percentage be 10 %; this means that the imported fuel at 64 % of the reference cost will break-even with the locally fabricated fuel at a cost of 71 % (of which the government pays 7 %) of the reference cost. If we take the other parameters to be unchanged (load factor : 0.6 ; rate of financial charges : 10 % ; operating charge coefficient : 0.5), we find from Figure 4 a corresponding plant size of 1.3 t/day as the minimum above which it pays to build a local fabrication plant rather than to buy the fuel from abroad.

5.5. INTEGRATED DECISION.

If we summarize the results obtained with the three above criteria, we get the following table :

type of criterion	minimum plant size [t/day]	Unit fabrication cost [\$/kg U local currency foreign currency s		U] subsidy
Cost level	2	17	41	
foreign currency	0.45 (0.2)	30 (63)	58 (58)	
externsl economies	1.3	20	- 44	- 6

Each of the above criteria has the same short coming : it considers only one aspect of a very complex situation. Clearly a final decision should rely on the combination of several criteria, including some which we did not consider or could not even imagine. There is no magic formula for this combination and it will be a matter of individual judgement for the investor to give the proper weights to each criterion.

6. FABRICATION DEMAND AND INVESTMENT STRATEGIES.

6.1. NUCLEAR POWER PROGRAMME AND FUEL DEMAND.

In the previous chapter, we have considered the minimum fabrication plant size which could be envisaged in a country on the basis of various criteria as to its profitability and under the assumption of a 60 % load factor.

It is now time to compare these results with the effective demand for fabrication. In order to relate this demand to the associated nuclear power programme, let us recall that 1000 MWe of nuclear power (PWR) require 80 t as initial core and one third of this, or 26.7 t for annual replacement (under normal operating conditions).

A fabrication plant of 1 t/day capacity, working at 60 % load factor, could thus supply the first cores for about 2000 MWe or, alternatively, the replacement fuel for 6000 MWe. If we consider simultaneously both types of demands and assume a nuclear power doubling time of 5 years, the same demand level occurs when the installed nuclear power reaches 4000 MWe; this last figure does not account for the times and delays associated with fuel fabrication nor for the possible delays in power plant operation.

6.2. EFFECT OF BUYING PRACTICES.

It is a common practice when ordering a nuclear power plant to give to the supplier of the nuclear steam supply system (NSSS) also the order for the first reactor core and forward commitments for a few reloads. This is done for reasons not only of convenience but also of supplier's warranty.

The extent to which this will be applicable in a particular case depends strongly on the ties between the NSSS supplier and the local manufacturer. If there is a licence or an association agreement between them, the latter could supply already the fabrication of the first core ; otherwise, it would probably have to wait for the second or even the third core to really enter the picture.

6.3. INVESTMENT STRATEGIES.

Figure 6 shows how one could place a fabrication plant in a nuclear power programme, i.e. in a forecast of fabrication demand versus time : time t is defined as the time when the fabrication plant of the minimum capacity under the prevailing investment criteria and working at the assumed 60 % load level exactly covers the possible fabrication demand.

In fact, the definition of size and time of fabrication plant investment should result from a careful optimization study, accounting for the numerous parameters involved.

In particular, one ought to account for the following arguments :

1. load factor.

For the sake of simplicity, we have used, in the foregoing, a constant load factor of 0.6. Obviously, the load factor is not constant and this value should be considered as a present worth average over a given period. The real load factor will certainly vary with time, starting from a rather low figure and increasing to the equilibrium level over a few years.

The fabrication plant could thus start production 2 or 3 years before t.

2. Capacity extension.

So far we have always considered the fabrication plant as a whole. As a matter of fact, a plant of reasonable size, say with a design capacity of 1 t/day, normally consists of a number of presses, sintering furnaces, loading units, etc The design capacity of the plant needs not to be implemented right from the beginning; one could start with just one of each necessary piece of equipment and duplicate these as demand increases. This would result in a somewhat lowed initial investment cost for a given ultimate design capacity.

On the other hand, when all the equipment is installed and when the design capacity is reached, there is still room for an increase in effective capacity thanks to improved plant operation or to replacement of some production units by new and more efficient ones.

3. Scale effect.

Since the fabrication cost for a given load factor decreases significantly with plant size increase, it could be interesting to build a larger plant. The plant size will have to be selected by making the balance of the extra investment cost, of the extra operating charges during early years and of the extra fuel fabrication during later years. If the increased size goes with delayed construction, one should also take into account the extra amount of fuel to order abroad before plant start-up.

7. CONCLUDING REMARKS.

For local manufacturers in countries where the nuclear power programme is still at a low level, the application of the above considerations will normally results in not very bright prospects if the possible market is limited to the national one.

This could be an overpessimistic assumption and various ways of enlarging the potential market are to be found.

One way would be to build a bigger plant than necessary from the national point of view and to rely on the exportation of part of the production on a competitive price basis, possibly with the help of national authorities.

Another way would be to pool the markets of several countries in a similar position through agreements between companies or between governments. Such an agreement might involve a programme for building future fabrication plants in the other countries or might cover other aspects of the fuel cycle.

One could also consider a commercial agreement with an already established fuel supplier and fabricate the fuel under licence. In addition to the economies already mentioned for research and development, one could spare much of the design work for the new plant and also much of its teething problems.

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TABLE I

COMPARISON OF FABRICATION COST DATA (EXCLUDING CONVERSION)

<u>Reference</u>	FABCOST	WASH 1082	BRAVL
a) <u>Capital cost</u> (M\$).			
1. Equipment	3.64	4.64	2.12
Building	2,41	5.30	2.54
Total l	6.05	11,94	4.66
2. Equipment design	1.96	n.a. (1)	n.a. (2)
Building design	0.38	n.a.	0.68 (2)
Total 1 + 2	8.39	11.94	5.34
3. Contingency	1.68	n. a.	0.34 (3)
Preoperation	3.33	n. a.	0.27 (3)
Grand total	13.40	11.94	5.95
b) Operating charges (M\$/year) (excluding	hardware)	•	
1. Direct labour		2.84	1.11
2. Other operating costs	5.0		1.91
3. General & administration		1.81	0.17 (2)
4. Working capital		0.66	0.13
Total	5.0	5.31	3.32

Notes : 1. n.a. = not directly available ; included in previous figures.

- 2. See comments in the text.
- 3. Contingency and preoperation relative to building only; for equipment, not available.

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TABLE II

CUMULATIVE CAPACITY OF NUCLEAR POWER PLANTS

Year	U.S.A.	Western World outside U.S.A.	Total
1975	59	37	96
1976	73	50	123
1977	89	64	153
1978	103	85	188
1979	128	104	232
1980	150	127	277
1981	173	146	319
1982	199	174	373
1983	230	203	433
1984	263	238	501
1985	299	271	570

[Thousands of Electrical Megawatts at End of Calendar Year]

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FIG. 5



Nuclear power plants supplied with fuel fabricated by :





local manufacturer (with 60 % average load factor)

INSTALLED NUCLEAR POWER AND PLANT SIZE

Study Group on the Facilities and Technology needed for Nuclear Fuel Manufacture

Grenoble, 4 - 15 September 1972

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