Fuel Processing [and Discussion]

R. H. Allardice, R. S. Pease, R. Lallement, H. Bailly and J. D. Lewins

Phil. Trans. R. Soc. Lond. A 1990 **331**, 381-393
doi: 10.1098/rsta.1990.0076

Phil. Trans. R. Soc. Lond. A **331**, 381–393 (1990)
Printed in Great Britain

Fuel processing

BY R. H. ALLARDICE
*British Nuclear Fuels plc, Risley, Warrington, Cheshire WA*3 6*AS, U.K.*

The technical and economic viability of the fast breeder reactor as an electricity generating system depends not only upon the reactor performance but also on a capability to recycle plutonium efficiently, reliably and economically through the reactor and fuel cycle facilities. Thus the fuel cycle is an integral and essential part of the system. Fuel cycle research and development has focused on demonstrating that the challenging technical requirements of processing plutonium fuel could be met and that the sometimes conflicting requirements of the fuel developer, fuel fabricator and fuel reprocessor could be reconciled. Pilot plant operation and development and design studies have established both the technical and economic feasibility of the fuel cycle but scope for further improvement exists through process intensification and flowsheet optimization. These objectives and the increasing processing demands made by the continuing improvement to fuel design and irradiation performance provide an incentive for continuing fuel cycle development work.

1. INTRODUCTION

The fast breeder reactor (FBR) has been developed to meet a future need for major and rapid expansion in the nuclear component of the electricity generating network, which could otherwise be significantly constrained by the increasing scarcity and cost of natural uranium to fuel thermal reactors. To fulfil this role the FBR system has to be shown to be capable of rapid commercial-scale deployment which requires that the plutonium bred within the FBR and that available in stock from thermal fuel reprocessing be utilized with maximum efficiency. Accordingly, FBR fuel and core designs have been targetted on minimizing plutonium inventory in the reactor core and the external fuel cycle and on maximizing plutonium breeding in the reactor.

It was recognized at the outset of the British fast reactor project that the commercial viability of the system would only be demonstrated to the electricity utilities and regulatory authorities if the reactor was supported by an equally well-developed, reliable, efficient and economic fuel cycle. Consequently, the development of processes for the fabrication of fuel, transport before and after irradiation and the reprocessing of the spent fuel and breeder was treated as an integral and important part in the overall system development. The diverse and sometimes conflicting requirements of the fuel and reactor designer, the reactor operator and the various fuel cycle process stages has required close coordination of all R&D activities.

In this paper some of the fundamental issues which have required attention by the fuel processing development teams are identified. Particular reference is made to the practical demonstration of the ability to recycle plutonium rapidly and efficiently through the reactor and fuel cycle facilities since this is essential to the adoption of the system. The importance of this is illustrated by figure 1, which shows the significant influence of fuel cycle duration and plutonium recovery efficiency on the FBR system expansion capability.

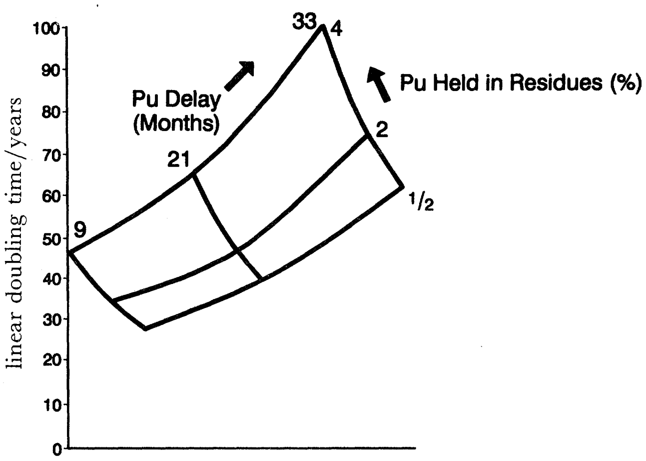

Figure 1. Effect of fuel cycle parameters on breeder linear doubling time. 'Doubling time' provides a measure of the plutonium utilization efficiency of the system. It is the time taken for the amount of plutonium associated with a particular reactor and its fuel cycle to double and hence enable the launch of a further reactor and fuel cycle.

2. OVERVIEW OF THE FUEL CYCLE

The external fuel cycles of the thermal and fast reactor systems are similar and differ mainly in the flows of the plant feed and reprocessing product materials (figure 2). For both fuel cycles the processing plant unit operations are also largely the same. Fuel fabrication involves the manufacture of ceramic fuel pellets by the pressing of powders and subsequent assembly into fuel pins and fuel elements. Spent fuel reprocessing involves fuel dismantling and the cropping of fuel pins, dissolution of the fuel in nitric acid followed by a highly selective solvent extraction process to separate the plutonium and uranium from each other and the unwanted fission products and actinides. The uranium and plutonium are converted into high-purity dioxide powders for reuse in fuel fabrication plants. The process stages of the FBR cycle are well documented (Salmon & Allardice 1979; Allardice 1986).

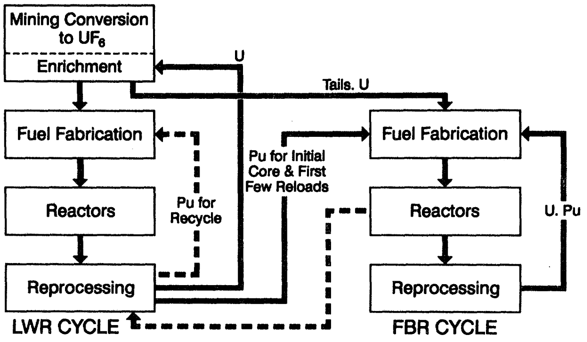

Figure 2. The LWR and FBR fuel cycle.

The decision to base the processing of FBR fuel on the thermal fuel cycle technology was a logical extension of long and successful experience with thermal fuel processing, which has spanned over 35 years. Even so, significant additional development and demonstration of the FBR fuel cycle has been necessary because of the unique and enhanced technical problems of processing plutonium fuel and because of the fundamental differences in the fuel design, the reactor operational régime and the special requirements of the fuel cycle mentioned earlier. Compared with the thermal fuel cycle, which uses uranium dioxide (UO_2) for fuel, the characteristics of the FBR fuel that provide additional fuel processing problems are the following:

(i) the fissile material being plutonium rather than ^{235}U;
(ii) the higher fissile content of the fuel;
(iii) the fuel consisting of two phases mixing PuO_2 in UO_2;
(iv) the FBR fuel structural complexity;
(v) the higher burn-up performance of the FBR fuel;
(vi) a shorter fuel 'cooling' period before reprocessing.

These differences and the consequences are quantified in table 1 and are briefly discussed below.

TABLE 1. TYPICAL FBR–PWR FUEL PARAMETER COMPARISON

(High breeding efficiency FBR, e.g. the U.K. CDFR.)

parameter	per assembly PWR	FBR	ratio FBR:PWR per kg of fuel[a]
core fuel type	UO_2	($0.8\ UO_2$) ($0.2\ PuO_2$)	—
breeder fuel type	—	UO_2	—
total assembly weight/kg	665	276	—
total (UO_2+PuO_2)/kg	520	108	—
fuel fraction	0.8	0.4	—
irradiated fuel data			
burn-up/(GWd t^{-1})	33	130	—
cooling time/years	5	0.5	—
radiation/(10^{15} MeV s^{-1})	3	30	50
neutron emission/(10^8 s^{-1})	2	5	12
heat output/kW	1	6	30
irradiated Pu content/kg	3	7	12

[a] Based on smeared value for FBR, i.e. core fuel and axial breeder fuel combined for fuel dissolution.

Compared with uranium as a fuel the use of plutonium creates increased radiological control problems in two respects, by the lower critical mass and by the higher radioactivity. All the isotopes of plutonium undergo spontaneous decay and spontaneous fission and consequently they emit a wide spectra of α-, β-, γ-rays and neutrons. The penetrating γ and neutron emissions require that occupational workers are isolated and shielded from the source while the biological effects of α particles when taken into the body (by inhalation, ingestion or through wounds) requires containment of the source. As a consequence plant designs provide for the minimization of radiological exposure of operating staff by the inclusion of high-integrity containment and appropriate shielding and by the provision of automated process stages, product transfers and maintenance procedures.

In addition to the above fundamental radiological protection principles plutonium fuel fabrication and reprocessing plant designs incorporate the use of geometrically safe equipment to prevent accidental criticality. The development of both these features has required the deployment of substantial engineering resources.

The FBR core fuel is a two-phase mixture of PuO_2 in a fertile UO_2 matrix, which allows potential for the non-homogeneous distribution of the fissile material. Since pure PuO_2 is insoluble in nitric acid the efficiency of the fuel dissolution is closely related to the homogeneity of the two constituents achieved during fuel fabrication. This has required careful development and control of the powder preparation processes and pellet-sintering parameters.

The engineering complexity of the FBR fuel assembly is high. The fissile fuel is highly concentrated into small annular fuel pellets which are enclosed, along with the axial breeder UO_2 pellets, in sealed, high-integrity cladding tubes (fuel pins). The fuel pins are supported by a system of either grids or wire wrapping spacers within a hexagonal wrapper tube. The top and bottom end features of the assembly include the reactor location devices, coolant flow gags and mixers and neutron shielding. The materials of construction are mainly high-quality stainless steels but for the future, alternatives such as nimonic and oxide dispersion strengthened ferritic alloys are being developed. This mechanical complexity makes the fabrication plant design objective of remote automated fuel assembly more difficult to achieve and the reprocessor has similarly to adopt procedures for dealing with increased hardware at the fuel dismantling and dissolution stages.

The high burn-up performance of the FBR fuel compared with thermal fuel and the prolonged irradiation in an intense fast neutron flux leads to some unique problems. The fuel assembly components become distorted due to neutron-induced voidage (NIV) swelling and thermal creep dilation effects which exacerbate the problems of mechanical dismantling. The levels of fission products and actinides in the spent fuel are significantly increased which results in high specific activity and decay heat (table 1). The high γ and neutron emissions require increased radiological protection in the reprocessing plant and enhance the potential for radiological damage and serious degradation to the organic solvent used in the reprocessing separation stage. The high decay heat output of the fuel requires special attention during fuel transport, handling, dismantling and dissolution. The formation of noble metal fission product alloys in the irradiated fuel create a particular problem because they have a high specific heat output at short cooling times (1 W g^{-1} at 180 days) and they are insoluble in nitric acid.

3. FUEL FABRICATION

The achievement of a sustained high level of performance by the fuel in the reactor and the ease of subsequent reprocessing is significantly influenced by the ability of the fuel fabricator to conform to the stringent product specification while maintaining a high-volume plant throughput on a routine basis.

Taking the fuel pellet as an example, this is a high-quality sintered ceramic product of typically 7 mm diameter and 10 mm length. During its operational period measured in years it will continuously produce about 0.4 kW (heat), experience a centre temperature of 2300 °C and a temperature gradient of 550 °C mm^{-1}. It will be subjected to thermal cycling, volume swelling due to fission product generation and a high neutron damage dose. Thermal diffusion and vapour phase transport can adjust the local concentrations of constituent materials and fission product gases will be released into the fuel pin void. To cope with these extreme conditions, therefore, the following chemical and physical properties of the fuel pellet must be controlled to narrow tolerances to eliminate or minimize any interaction between the pellets and the fuel pin cladding.

1. Plutonium:uranium ratio and isotopic composition, which determines the reactivity value of the fuel.

2. Chemical composition with respect to impurity limits, which influences cladding corrosion behaviour.

3. Stoichiometry: a low oxygen:metal ratio is desirable for minimizing fuel cladding internal corrosion, but it can adversely affect fuel pellet solubility.

4. Fuel density: economic and performance considerations require high density (about 95% of theoretical).

5. Micro-homogeneity and phase composition: to ensure the homogeneous distribution of the fissile plutonium in the pellet for performance and fuel solubility reasons.

6. Solubility in nitric acid: to ensure that the amount of insoluble fuel is minimal and more than 99.5% of the plutonium entering the fuel cycle can be recovered.

The processes for the manufacture of fuel pellets have been the subject of significant development work. The objectives of making homogeneous fuel while complying with the fundamental radiological protection principles and the need to minimize waste arisings led to the development of 'direct wet routes' in which the mixed uranium–plutonium-nitrate stream would be converted directly and continuously to a solid fuel form without any 'dusty' powder handling stages.

The route studied in the U.K. was gel precipitation in which droplets of mixed uranium–plutonium-nitrate containing a gelling agent when suspended in ammonia were converted into spherical particles of ammonium diuranate and plutonium hydroxide. The spheres were washed, dried, debonded and sintered in reducing conditions to produce high-quality ceramic grade mixed-oxide spheres. The original intention was to use two size fractions of spheres and to fill the fuel pins by vibro compaction; however, the irradiation testing programme for this fuel showed that it did not perform as well as pelleted material. Therefore, although the gel–vibro compaction route was an elegant continuous and efficient fuel fabrication process, it was not pursued further.

An extension to this work assessed the feasibility of pressing pellets by using the gel spheres as a press feed. This retained the potential advantages of the gel fuel preparation process and the annular pellet fuel form. However, the physical properties of the gel spheres had to be modified to allow pellets to be formed and the resulting pellets did not meet the specification for density and physical dimensions. This development plus supporting comparative engineering design studies and cost estimates led the U.K. to conclude that there was no significant advantage in moving from the traditional powder pellet route. The powder pellet route operated for the Dounreay Prototype Fast Reactor (PFR) fuel fabrication is shown schematically in figure 3. Similar routes are used on a world-wide basis.

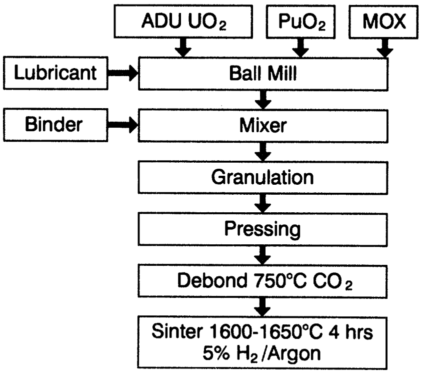

FIGURE 3. The PFR fuel fabrication route.

As mentioned earlier, the main difficulties associated with the powder pellet route lie with ensuring that the sintered pellet product is of adequate homogeneity. The processes that control this are the mechanical blending and the sintering cycle. The efficiency of the mechanical blending process depends not only on the milling system used but also on the properties of the starting materials provided by the reprocessing plant.

Development work has shown that the most homogeneous oxides are prepared by using PuO_2 powders with a high surface area which breaks down readily during milling. Particle size analysis and electron microscopy has shown that the particles making up these powders should be less than about 30 μm diameter and with an open microstructure. In the past, conventional roller mills have been used to blend the powders and parameters such as the ball to powder charge ratio and milling time have been shown to play an important role in ensuring a homogeneous product. Recently, a high-energy stirred ball mill has been developed as an alternative blending process; the advantage of using this system is that milling times can be reduced from several hours to only a few minutes and thus throughputs in the plant are increased. These features, together with an important design feature that avoids the use of make-and-break connections for filling and emptying (thus avoiding the dispersion of dusty Pu active material throughout the glove box), have made this mill a very attractive development.

One of the most important processes in fuel pellet manufacturing is the sintering cycle in which material is heated under precisely controlled atmosphere conditions, to temperatures of 1650 °C. Atmosphere control is important to control the density of the material. If material sinters rapidly as it would in an atmosphere with a high oxygen content, gases become entrapped in pores and as the temperature increases and the material becomes more plastic in nature, it can deform or expand in an unacceptable manner thus reducing the density. The rate of homogenization of fuel during sintering depends on the temperature used and the oxygen content of the sintering atmosphere. High temperatures and high oxygen content result in a fast rate of homogenization. However, the requirement to control the sintering rate to maintain density means that very precisely controlled conditions have to be selected for manufacturing fuel of acceptable density and homogeneity. Typical conditions used are temperature of 1650 °C and atmosphere of $4\%\ H_2 : 96\%$ Ar gas mixture. The oxygen potential of the sintering atmosphere is dependent on the exact composition of the start gas and the furnace loading and generally is of the order of 400 kJ per mol O_2.

In addition to the successful development of the pellet fabrication route an engineering development programme has been aimed at achieving the fundamental operational and radiological control principles of the plant. By the end of 1988 work had been completed on automated equipment for metering and dispensing of powder feed, high-speed dimensional gauging of pellets, loading of pellets into fuel tubes and transfer of pins into and between work stations.

Future fuel fabrication development will need to keep pace with changes brought about as the fuel developer extends the bounds of the fuel performance. Fabrication of pins incorporating new cladding alloys is already part of the programme and, in the more distant future, mixed-oxide fuel may be replaced by other ceramic compounds such as high-density monocarbides or nitrides.

4. FUEL REPROCESSING

The reprocessing of fuel commences with the receipt of the highly active fuel assembly. The complex structure will be distorted and emitting high α, β, γ and neutron radiation, giving rise to (decay) heat (table 1). The reprocessing task is to recover the plutonium and uranium in a highly pure form for return to the fabrication plant and to put the radioactive wastes into a suitable form for treatment, storage and disposal.

The specification for the reprocessing products, i.e. the plutonium and the uranium, requires that the reprocessor achieves a decontamination factor of greater than 10^7 for fission product activity in the uranium and plutonium and for plutonium in uranium. Inactive impurities, which may influence the ability to fabricate fuel pellets or affect the performance of the fuel in the reactor, are reduced to typically less than 50 p.p.m. (by mass) in the products.

Figure 4 shows the main process stages in the reprocessing plant. The central processes of dismantling, fuel dissolution and solvent extraction are linked by the common requirement to ensure high plutonium recovery efficiency by minimizing the losses to solid and liquid waste streams.

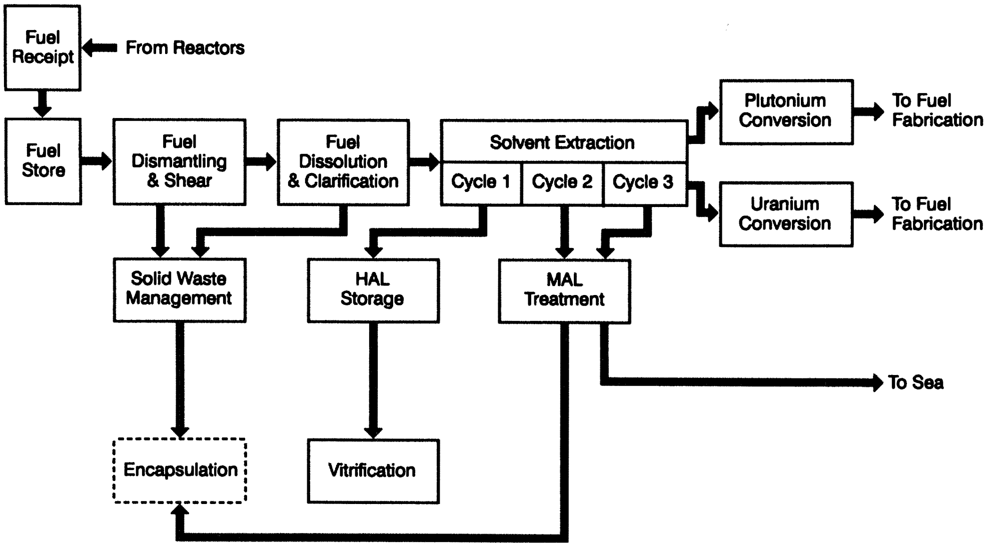

FIGURE 4. HAL: highly active liquor; MAL: medium active liquor.

(a) *Fuel dismantling*

The dismantling of the irradiated fuel assembly was anticipated to be a potential major problem. The possibility of swollen distorted components has been mentioned and the concern was that the forces required to withdraw pins from the wrapper and grids might be high enough to cause pin failures particularly if irradiation induced cladding embrittlement occurred. Extensive engineering development allied to theoretical predictions and post-irradiation observation of dimensional changes has provided reassurance on this point and appropriate dismantling techniques incorporating laser cutting and automated pin pulling procedures have been developed. The future introduction of swelling-resistant alloys and modified 'compliant' component designs into the fuel assembly will enable the dismantling of fuel at even higher burn-up levels.

(*b*) *Fuel dissolution*

Following fuel element dismantling and pin shearing the next stage in reprocessing is to dissolve fuel in nitric acid. The economic and strategic importance of high plutonium recovery efficiency has been shown earlier together with the potential problem of insoluble PuO_2 fuel. Solubility testing of as manufactured fuel has shown that fuel pellets manufactured with microscopic PuO_2-rich regions leave fine powder insolubles with a $Pu:(U+Pu)$ ratio higher than the parent fuel. However, homogeneous, high-density, large-grain-size fuel pellets (greater than 8 μm diameter) sometimes give residues in the form of pellet skeletons with the same $Pu:(U+Pu)$ ratio as the parent fuel. This is believed to be because of the low surface area available for attack by the acid. There is no evidence that the skeletal type residues persist after irradiation.

The solubility of irradiated fuel has been studied theoretically and experimentally. A computer code has been developed to predict the solubility of fuel during reprocessing from a knowledge of the as-manufactured homogeneity and the subsequent irradiation history. During irradiation thermal diffusion and vapour phase transport mechanisms can change local plutonium concentrations. The burn-down of fissile atom content with irradiation will reduce plutonium concentration and the thermal cracking of pellets will facilitate subsequent acid ingress to the fuel. The results of these studies suggest that irradiation will enhance the 'as manufactured' solubility of fuel by a factor of at least 2–3.

The PFR plant uses a batch dissolution process but for future commercial scale plants in the U.K. a continuous pulsed-vee dissolver is being developed in which the fuel–cladding is moved counter-current to the dissolver acid by liquor pulsing. Consideration is also being given to the use of alternative dissolution reagents. One concept being examined for the longer term is the total dissolution of cladding and fuel. Initial work suggests that solvent extraction from the resultant solutions is feasible; however, considerable work is required on waste treatment processes and on identifying dissolver construction materials compatible with an aggressive dissolution reagent.

(*c*) *Chemical separation*

Uranium and plutonium are recovered from the dissolver liquor, and separated and purified to very clean solutions of uranyl and plutonium nitrates by a solvent extraction process. The extractant is tri *n*-butyl phosphate in odourless kerosene or dodecane diluent. Current plant designs use pulsed column contractors (Jenkins 1987) in which the aqueous phase (dissolver liquor) and organic phase (solvent) are fed counter-current, contacted in a section of column fitted with perforated plates and then disengaged in settlers at either end before transfer to further operations.

The uranium and plutonium are selectively extracted into the organic phase while the highly active fission products and unwanted higher actinides remain in the aqueous phase. The (U–Pu) loaded organic product is scrubbed with fresh nitric acid to enhance the removal of fission products before being fed to further columns where U–Pu is stripped into dilute acid and chemically conditioned before separation of U and Pu and further purification in subsequent solvent extraction cycles. U–Pu separation may be achieved by Pu valency adjustment (to form non-extractable Pu^{III}) or by forming a non-extractable Pu sulphate complex or by adjustment of acidity to promote differential U–Pu extraction.

Experimental work on pulsed plate column performance (Jenkins 1987) has shown that product purity can be adversely affected by entrainment of very small drops of the aqueous phase in the organic product. This could result in the carry-over of non-extractable fission products and higher actinides such as caesium, strontium, americium and curium with the uranium and plutonium, which would put an undesirable burden on the downstream purification and waste treatment processes. Experiments have established that two mechanisms are responsible for this entrainment; drop shear during mixing and chemical precipitation of dissolved water during uranium extraction. Further work is in hand to study how the two mechanisms determine carry-over of non-extractable impurities and to develop methods by which entrainment can be reduced.

A wide-ranging review of options for providing more complete, 'intensified' solvent extraction identified the centrifugal contactor as one possibility for longer-term development. Contact times are typically a few seconds per theoretical stage compared with minutes for the pulsed column and this has the important benefit of reducing the radiation dose received by the solvent by a factor of 2–10. Studies have shown that the adoption of centrifugal contactors would also result in substantial savings on capital, operational and waste disposal costs.

The centrifugal contactor is one example of process intensification through the use of small-volume high-throughput equipment, other opportunities exist, e.g. through the use of power fluidic devices. These opportunities plus the optimization of process stages to reduce capital and operating costs are the subject of future development programmes.

5. Plant operational experience

Practical experience of the FBR fuel cycle has been obtained in European facilities over the past three decades within coherent programmes supporting firstly the engineering test reactors DFR (U.K.), Rapsodie (France) and KNK II (West Germany) and later the larger prototype reactors, PFR (U.K.) and Phénix (France) commissioned in the early 1970s. There has also been significant development in Japan (Joyo), U.S.A. (FFTF) and the Soviet Union (BOR 60, BN 600).

Experience with FBR mixed-oxide (MOX) fuel fabrication has been gained in a number of facilities (Megy *et al.* 1987). In the U.K. the PFR fuel fabrication plant located at Sellafield and operated by BNFL has produced over 75000 finished fuel pins incorporating 17 t of MOX fuel (about 12 million pellets) at plutonium concentrations of up to 33 % for irradiation in the PFR. The CEA plant at Cadarache has made about 146000 pins for Phénix and over 100000 pins for Superphénix. Both the PFR and Phénix fuel cycles have been closed by refabricating fuel with plutonium recovered by reprocessing. The excellent irradiation performance of this fuel (outlined in another paper in this symposium by A. Brandstetter, A. M. Broomfield and & B. Saitcevsky) is testament to the consistent high-quality product achieved by the fuel fabrication plants.

The technical feasibility of reprocessing high burn-up FBR MOX fuel has been extensively demonstrated in both the U.K. and France (Megy *et al.* 1987). In the U.K. the reprocessing of irradiated PFR fuel at Dounreay commenced in 1980; the PFR fuel cycle was closed in 1981. Since then the plant has reprocessed about 14.5 t of heavy metal (16.3 t of oxide) at a maximum burn-up of 15.9 % of heavy atoms (135 GWd t^{-1}) (see table 2). The plant chemical flowsheet has proved to be very successful being tolerant to feed variations (±10%) and giving high overall fission product decontamination (greater than 10^7) and a once-through plutonium efficiency of greater than 99.5%.

TABLE 2. PFR REPROCESSING PLANT EXPERIENCE TO FEBRUARY 1989

total oxide reprocessed/t	16.3
plutonium recovered/t	2.6
no. of assemblies	158
max. burn-up/(GW d t^{-1})	135
(% HA)	(15.9)
clad displacement dose (DPA)	116
min. cooling time/d	136
max. heat rating/kW	2
Pu recovery efficiency (%)	> 99.5

A similar quantity of MOX fuel has been reprocessed in France in the APM pilot plant at Marcoule and the UP2 thermal reprocessing plant at La Hague. In the latter, over 10 t of Phénix fuel was reprocessed by dissolution in the presence of the neutron poison gadolinium, followed by dilution and blending with a solution of gas-cooled, graphite-moderated thermal reactor oxide fuel. This demonstrated the feasibility of reprocessing FBR fuel in a thermal oxide plant.

These pilot scale plants have provided invaluable experience of engineering design and capital and operating costs. Their operation enabled newly developed equipment and processes to be tested under fully radioactive plant scale conditions. Overall they have provided a convincing practical demonstration of the technological capability for reprocessing FBR fuel in the U.K. and France.

6. FUEL TRANSPORT

The precise scale of fresh and irradiated fuel transport movements will be determined by the location of reactors, fuel cycle plants and waste repositories and the requirements of the fuel cycle. Transport of irradiated fuel assemblies as originally envisaged at short cooling periods, with high decay heat, may require the use of sodium as a heat transfer medium. The more recent acceptance of a longer cooling period, e.g. two years for the European Fast Reactor (EFR) design, reduces the decay heat (less than 2 kW) and hence the demand on the reprocessing system and permits the transport of fuel assemblies free of sodium.

Studies in the early 1980s relating to CDFR and the requirement to minimize plutonium out-of-pile time, indicated that two shipment procedures were practical. The first procedure transported up to seven intact fuel assemblies in individual sodium-filled canisters within a multiposition flask. The second procedure involved the dismantling of the fuel assemblies at the reactor site, cleaning them free of sodium, and then enclosing them in small gas-filled canisters holding 50–60 pins. The equivalent of 10 fuel assemblies per flask could be transported by this method. In both cases the facilities required at the power station and the reprocessing plant, as well as the flasks themselves, have been evaluated in sufficient detail to demonstrate technical feasibility and allow meaningful cost comparisons to be made.

The flask studies considered thermal performance, shielding and criticality requirements and their resistance to impacts and fires. They confirmed that an irradiated fuel transport flask capable of meeting the requirements of the (IAEA) regulations for the Safe Transport of

Irradiated Materials could be manufactured and transported between the reactor and reprocessing plant.

System studies have also shown that the transport of irradiated fast reactor fuel from a power station to a distant reprocessing plant is technically feasible; there are no technical reasons requiring nuclear parks containing several reactors together with fuel cycle plants. Additionally, it has been shown that the transport costs form only a small part of the total generating costs (not greater than about 1%), so there are no financial constraints to prevent power station and reprocessing plant being widely separated geographically.

7. Fuel cycle costs

On the assumption that the utilities already own the necessary plutonium and depleted uranium, the costs of the FBR fuel cycle arise equally from the fuel fabrication and spent fuel management operations. The fabrication and reprocessing plant capital and operating costs thus have a significant effect on fuel cycle economics and on the overall electricity generating cost since current estimates indicate that the fuel cycle contributes about 20% to the total. This is unlike the thermal fuel cycle in which costs are dominated by the front end operations of uranium ore purchase and enrichment.

During 1987 a joint BNFL–CEGB–UKAEA economic assessment of the FBR and PWR fuel cycle costs was completed (Allardice *et al.* 1988). The assessment was firmly based on detailed engineering design studies in the U.K. for fabrication and reprocessing plants. The reference fuel cycles chosen for the study were typical rather than of a 'one-off' nature and appropriate to commissioning in the shorter term (year 2000) and the longer term (year 2020). The study included a sensitivity analysis of the basic assumptions.

This comprehensive study by the British nuclear industry showed that competitive fuel cycle costs were achievable even for early fast reactors. By the year 2000, FBR and PWR fuel cycle costs were shown to be similar and associated uncertainty ranges were comparable. In the longer term performance improvements favour the FBR with the result that the FBR fuel cycle costs will be less than 75% of those for the PWR. Assessment of the use of plutonium recycle in the PWR did not alter this conclusion.

Table 3. Lifetime fuel cycle costs for the fast reactor and the PWR

(p kWh^{-1}, March 1986 money values.)

	shorter term, commissioned in year 2000		longer term, commissioned in year 2020	
	fast reactor burn-up 15% peak	PWR, burn-up 49 GWd t^{-1}	fast reactor burn-up 20% peak	PWR, burn-up 49 GWd t^{-1} MOX
U ore	—	0.22	—	0.23
conversion	—	0.02	—	0.01
enrichment	—	0.14	—	0.09
fuel fabrication	0.24	0.05	0.16	0.06
spent fuel management	0.25	0.05	0.16	0.05
U credits	—	(0.01)	—	(0.01)
total	0.49	0.47	0.32	0.43

8. Conclusions

Efficient and economic fuel processing is an essential and integral component of the fast breeder system.

Despite the more challenging technical requirements of processing fuels of high plutonium content and high irradiation, the basic technologies used to process thermal reactor fuels have been successfully extended and improved to cope with fast reactor oxide fuels.

The technical feasibility of fabricating and processing high-quality FBR fuel efficiently has been convincingly demonstrated at pilot plant scale notably in the U.K. and France.

This practical experience coupled with supporting design and development programmes has allowed the economics of the future fast reactor fuel cycle to be estimated with considerable confidence. Compared with the PWR, the FBR fuel cycle can be cost competitive even in the short term.

Design led development programmes aimed at the construction of small-volume high-throughput plants based on process intensification and flowsheet optimization should lead to further improvements in fast reactor fuel cycle technology and reduced unit costs.

References

Allardice, R. H. 1986 General description of fast reactor fuel reprocessing. Submission to public inquiry on European Demonstration Reprocessing Plant, Thurso, U.K.

Allardice, R. H., Hall, R. S. & Pilling, R. L. 1988 FBR fuel cycle can be competitive. *Nucl. Engng Int.* **33** (404), 13–16.

Jenkins, J. 1987 Pulsed columns in the nuclear industry. *Chem. Engr* **441**, 16–19.

Megy, J., Allardice, R. H., Ebert, K., Morelle, J. M. & Vendetti, P. 1987 Fast breeder fuel cycle in Europe: present status and prospects. *Proc. Int. Conf. fast breeder systems*, Richland, Washington, U.S.A., vol. 1, pp. 5.2–1 to 5.2–7, September, 1987.

Salmon, A. & Allardice, R. H. 1979 The uranium–plutonium breeder reactor fuel cycle. *Proc. Conf. breeder reactor and Europe*, Lucerne, Switzerland, October 1979.

Discussion

R. S. Pease, F.R.S. (*Pease Partners, Newbury, U.K.*). Mr Allardice has rightly mentioned the importance of the costs of the fuel cycle in the assessment of the overall competitiveness of the FBR, but his paper quotes only theoretical values of this quantity. Speaking as an experimentalist, it would be very helpful to see some experimental values. Both in France and in the U.K., we have been told in earlier papers, some tonnes of recycled fuel have been used in PFR and Phénix: what were the observed costs of the fuel cycle in these cases?

R. H. Allardice. It is difficult to give actual costs that would be meaningful in terms of the commercial fuel cycle. For example the reprocessing plant at Dounreay was built with a capacity for 8–10 t a^{-1}. The main purpose of this plant was to demonstrate the technology. It has been operated at full capacity only for short periods, amounting to an overall load factor of about 25%. However, the experience has been used in our cost comparison study, which gave a total cost of £600 per kilogram of U and Pu.

R. Lallement (*CEA, France*). Mr Allardice indicated striking differences between LWR and FBR fuels. What therefore is his opinion on the necessity for a separate reprocessing plant for FBR fuel. Does he think that a flexible, multipurpose plant is feasible?

R. H. Allardice. Our experience over the past 25 years has shown that separate fast-reactor fuel reprocessing is technically and economically feasible. When commercial fast reactors are established, and our customers then decide what is wanted, it can be done. For example, substantial thermal-reactor fuel reprocessing may not then be required. In the short term, if existing plant capacity is available, it will be used to process fast-reactor fuel to defer capital investment in new plant.

H. Bailly (*CEA*, *Cadarache*, *France*). Mr Allardice said that reprocessing in Europe is well established, based on the PUREX process, and that it would be difficult to bring about substantial changes to this. However, we note that the U.S. are following a different route based on a pyrometallurgical process. The economic projections on this given earlier by Dr Griffith appear very favourable. What are Mr Allardice's personal opinions on this?

R. H. Allardice. I do have some reservations about the U.S. cost projections for the electro-refining process. We have some experience with metal fuels in the U.K. The concept was abandoned because we could not make the reactors economic, not because we could not reprocess the fuel. The solvent extraction process is very efficient and very flexible. I would make the point that reprocessing technology must relate to the fuel requirements. It should be possible to optimize the system as a whole. For example I believe that the present specifications for oxide fuel are too stringent. Some relaxation of this could lower the costs of both fuel fabrication and reprocessing.

J. D. Lewins (*Cambridge University*, *U.K.*). The cost estimates Mr Allardice showed looked quite favourable to the fast reactor. What uranium price was used in calculating PWR generating costs.

R. H. Allardice. The figures we used were projections of world market prices for readily available uranium and were based on advice from the British uranium procurement organization. They were taken for two dates: 2000 and 2020. It is very difficult to construct a level playing field when making comparisons between future fast and thermal reactor fuel cycle costs. For example, we must also take into account exchange rate fluctuations between the U.S. dollar and Sterling over this period. We made the comparison on the basis of a relatively small number of fast reactors against a tranche of PWRs with deferred reprocessing. No allowance was made for technological improvements in the fast reactor fuel cycle. However, the sensitivity analysis carried out in our study showed that if the uranium price remained constant from 1995 onwards, the cost comparison would become 8–9 % more favourable to the PWR fuel cycle, and if the uranium prices remained constant after 2020, costs would move 25 % in favour of the PWR fuel cycle. In other words even if uranium prices remain constant for the next 50 years, the fast reactor fuel cycle cost would still be only 70 % of that of the PWR fuel cycle when commercial-scale fast reactor fuel cycle plants are in operation.