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Development, Operational Experience and Implications for Future Design of Fast Reactors in Western Europe [and Discussion]

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Development, operational experience and implications for future design of fast reactors in Western Europe

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Over the past 30 years, the partners now collaborating in Europe on fast-reactor development have taken the technology from a theoretical possibility to an engineering reality. In that time there has been a progression from experimental zero-energy facilities followed by small power-producing engineering test reactors, to prototype reactors and a large commercial-size reactor. The paper describes the highlights of the reactor programmes in the partner countries and by example illustrates the experience gained from reactor operation and some of the principal activities in the supporting development programme. These include such topics as fuel performance, fast-neutron physics, liquid-metal thermal hydraulics, sodium chemistry, instrumentation and safety aspects. The paper concludes by summarizing some of the main objectives of the current development programme, which is directed to the support of the European Fast Reactor design being prepared by the European design and construction companies.

1. Introduction

Fast-neutron reactors were first studied in Europe in the late 1940s, when their potential for using uranium more efficiently than thermal reactors was recognized. Development efforts intensified in the following decade, with the result that challenging engineering problems of fuel, coolant, reactor control and safety were overcome and beginning in the late 1950s test reactors with core outputs up to 60 MW_t were commissioned. Larger, 'prototype' reactors with electrical power outputs in the range 250–280 MW_e were built in France and the U.K. in the 1970s and somewhat later in the F.R.G. Their aim was to prove design concepts suitable for application to commercial plant and to test on a substantial scale advanced mixed plutonium–uranium oxide fuels. More recently Superphénix, at 1200 MW_e the world's largest fast reactor, was built in France and taken to full power in 1986.

Since 1984, France, Italy, the U.K. and the DeBeNe grouping of countries (see table 1) have been collaborating on the design and development of fast reactors against the background of nearly 30 years of operating such reactors. Collaboration has the aim of sharing design and development costs between the partners by rationalizing the use of teams and facilities.

The essential importance of the fuel cycle to the fast reactor system has resulted in a substantial amount of development effort also being invested in demonstrating fuel performance at high burn-up level (to minimize fuel cycle costs) and in establishing the processes of fabrication, reprocessing and waste management. The experience from prototype plants has provided a firm base for the design of commercial-scale fuel cycle plants. Development of such

TABLE 1. ORGANIZATIONS INVOLVED IN EUROPEAN FAST REACTOR COLLABORATION

country	utilities	design and construction companies	R & D organizations
France	Electricité de France (EdF)	Novatome	Commissariat à L'Énergie Atomique (CEA)
F.R.G.	Schnell Brüter Kernkraft-werksgesellschaft (SBK)	Internationale Natrium Brutreactorbau (INB)	Interatom, Kernforschungszentrum Karlsruhe (KfK)
Belgium	Electro-nucléaire/SBK	Belgonucleaire/INB	CEN/SCK Mol
Netherlands	SBK	Neratoom/INB	Energieonderzoek Centrum Nederland (ECN)
Italy	Ente Nazionale per l'Energie Elettrica (ENEL)	Ansaldo	Ente Nazionale per l'Energie Alternative (ENEA)
U.K.	Central Electricity Generating Board (CEGB)	National Nuclear Corporation (NNC)	U.K. Atomic Energy Authority (UKAEA)

plants, including collaborative arrangements in Europe, are dealt with in a separate paper in this Symposium.

This paper summarizes the position on reactor development in Europe by reviewing the experience gained from reactor design, construction and operation, and outlines the scope of the associated development programme. The latter has as its aim the reduction of the capital cost of a fast-reactor plant, without compromising safety or reliability, sufficiently to bring the cost of its electricity down to the level of that generated by the more intensively developed pressurized water reactors (PWRS). This is being tackled by carrying out work on an advanced design of commercial demonstration reactor, known as the European Fast Reactor (EFR) with an electrical output of 1450 MW_e, to a specification defined by the European Fast Reactor Utilities Group (EFRUG). The design and construction companies and R&D organizations in the partner countries are collaborating on the EFR project, which is currently in the conceptual design phase. The plan is to be in a position to begin construction of this demonstration plant in the mid 1990s. The paper concludes by summarizing some of the main features of the EFR programme.

2. Progress of fast-reactor development in Western Europe

The development of fast reactors proceeded in Europe by the progression, following zero-energy studies of basic reactor physics, through engineering test reactors to the more sophisticated power plants with outputs up to 1200 MW_e as shown in table 2. The roles and achievements of the various types of reactor are outlined below, with a more detailed review of the experience gained in §3.

(a) *Test reactors*

Test reactors, DFR (U.K.), Rapsodie (France), and KNK II (F.R.G.), with powers in the range 40–60 MW_t were constructed to study the following aspects of fast-reactor technology: core design and operation; irradiation performance of fuel to progressively higher burn-up; use and handling of liquid metal coolant; power operation and electricity generation; safety and control. In addition, first experience of fast-reactor construction was obtained.

DFR and Rapsodie are now both shut down and are contributing usefully to decommissioning studies. KNK II continues in operation as a fuel test reactor.

Table 2. Progression of fast reactors in Western Europe

country	test	type of reactor prototype	commercial-size
France	Rapsodie (1967–82) 40 MW_t	Phénix (1973–) 250 MW_e	Superphénix (1986–) 1200 MW_e
U.K.	DFR (1959–77) 60 MW_t/15 MW_e	PFR (1974–) 250 MW_e	—
F.R.G.	KNK II (1978–) 58 MW_t/20 MW_e	SNR 300 280 MW_e	—

All three test reactors were of loop construction with separate primary vessel, intermediate heat exchangers and pumps. DFR used a 70:30 sodium–potassium alloy coolant, which remains liquid down to 38 °C, whereas Rapsodie, KNK II and the later reactors used sodium, which freezes at 98 °C. Operation of these reactors was instrumental in proving the large-scale use of liquid metals as heat transfer media and the ability to control impurities (oxygen, hydrogen and carbon) in the coolant and eliminate entrainment of the blanket gas. The cores were proven to be stable in operation and temperature coefficients of reactivity were confirmed.

Fuel irradiation in the test reactors confirmed the high burn-up potential of oxide fuel and set the design targets for the prototype stage. The important phenomenon of neutron-induced voidage swelling was also discovered.

(b) *Prototype reactors*

Three prototype reactors (Phénix, PFR and SNR300) with electrical outputs in the range 250–280 MW_e were built in Western Europe, as the next stage of development. The object was to prove the technology of fast reactors on a generating plant of intermediate scale between the test reactors and future commercial reactors and to continue with the process of developing improved fuel. A major requirement was to demonstrate the technology of sodium-heated steam generators. For these a variety of designs were chosen, ranging from the Phénix modular type, to the large units chosen for PFR. Phénix and PFR, both pool-type reactors, started operation in 1973 and 1974, respectively. The operation of SNR300 in the F.R.G., by contrast a loop-type reactor, has been delayed pending licensing approval, which is now expected in 1991.

The construction of the prototype reactors was important in giving experience of detailed design and supporting analysis, manufacture of components and site construction, licensing and commissioning procedures. It also resulted in the formation of design and R&D teams with the broad experience necessary to advance the fast reactor system.

Phénix has had a very successful operating history, generating over 20 TW h of electricity in 15 years at a lifetime load factor of around 62%, despite the fact that it has been necessary to remove and repair all intermediate heat exchangers (IHXs) in the primary circuit and to deal with leaks in the reheater modules of the steam generators. While the IHXs in PFR have operated normally, steam generator leaks have had a greater impact on operational availability, with the result that PFR had generated a total of 5.5 TW h of electricity by early 1989.

(*c*) *Superphénix*

The culmination of fast reactor development in Europe to date is the 1200 MW_e Superphénix constructed in France by NERSA. The object was to gain experience in constructing and operating a large fast reactor incorporating commercial-scale components and with an electrical output comparable with contemporary PWRs. Design and construction of Superphénix proved the practicability of site fabrication of reactor structures and large components in the primary circuit.

Commissioning of Superphénix revealed various plant problems, particularly vibrations associated with the primary vessel cooling system, which were successfully overcome. Full power was first achieved in December 1986. In March 1987 sodium leaked from the vessel of the external fuel storage tank into the interspace between the vessel and its surrounding guard vessel. Although the fuel storage tank is completely separate from the primary vessel, the French safety authorities decided that the plant should be shut down for location of the leak site and re-evaluation of the safety case covering the integrity of sodium vessels (see §3 (*d*) for further details). This process occupied some 18 months. Approval for return to operation was given in early 1989 with power operation being progressively increased from April 1989.

3. Experience gained from operation

The experience gained from reactor operation to date is summarized below with comment on those topics where the prototypes have contributed most to the development of the fast reactor, and those where operating problems have pointed out the areas for attention and improvements.

(*a*) *Fuel cycle*

The prototype reactors are fulfilling their role in fuel development by demonstrating performance and by probing the limits of designs and materials at high burn-up and damage doses. The close integration of post-irradiation examination facilities and reactors has been beneficial in this endeavour. The lead subassemblies in PFR, Phénix and KNK II have reached 15–20% burn-up and clad damage doses above 130 displacements per atom (DPA). This may be compared with the target values for EFR fuel of 20% burn-up (170000 MW d t^{-1} oxide) and 180 DPA damage dose. To the end of 1988 Phénix had irradiated around 155000 pins, PFR some 83000 and the smaller KNK II around 8000. DFR and Rapsodie irradiated some 28000 oxide fuel pins. The first core of Superphénix contains nearly 100000 pins.

Irradiation performance of fuel pin and wrapper is being improved by the replacement of the original stainless steels, with their tendency for high rates of void swelling and irradiation creep, by improved low-swelling materials. Those now used as fuel pin clad include improved austenitic stainless steels, high-nickel alloys (PE16 in the U.K., Inconel 706 in France). The potential of oxide dispersion strengthened (ODS) ferritic steels is being investigated. Ferritic steel is the main candidate for the wrapper surrounding the fuel subassembly pin bundle. Fuel performance issues are dealt with in more detail in a later paper in this Symposium.

Despite the high void swelling experienced by some of the original subassemblies using stainless steel clad and wrapper, fuel handling machines have been remarkably tolerant in dealing with the resulting bowing, axial growth and dilation of wrappers without resort to special recovery procedures. Improved low-swelling materials progressively introduced during the fuel irradiation programme are considerably reducing the demands on fuel handling.

The fabrication of high-quality fuel has been successfully developed. The number of failures in service (a few tens of pins) has been remarkably small for the number of pins irradiated and these failures have, in most cases, occurred in the process of probing the limits of experimental designs of fuel. Fuel failure events have been shown to develop benignly. Typically failure of the pin clad is characterized by a gas release followed by exposure of fuel to the sodium coolant. The exposed fuel releases delayed neutron precursors into the coolant, the neutrons from which can be readily detected. The area of fuel exposed to the coolant increases with time as the breach in the clad grows. In PFR and Phénix, operation can be continued (for up to 10 weeks in some cases) until the delayed neutron detection signal reaches a limit which maintains an acceptable degree of reactor protection. The result is that valuable information has been gained on the behaviour and development of fuel failures.

The removal of sodium from irradiated fuel subassemblies before reprocessing has been successfully developed. The processes are in routine use for PFR and Phénix fuel.

Reprocessing of fuel from Phénix and PFR has been satisfactorily demonstrated, with over 99% recovery of plutonium. The fuel cycles for Phénix and PFR have been closed by fabrication of new fuel from plutonium recovered through reprocessing. A total of some 21.9 t (oxide) of Phénix fuel has been reprocessed to date. By February 1989 some 16.3 t (oxide) of PFR fuel had been reprocessed, including one subassembly at a peak burn-up of 15.9% and clad dose of 116 DPA. No special problems were experienced with the breakdown or chemical processing of this subassembly, through the loads needed to remove pins were higher than normal as a result of neutron induced voidage effects in the pins and grids.

(*b*) *Reactor and sodium systems*

The cores and primary circuits of these reactors have in general been exceptionally fault-free. The mechanical sodium pumps, in particular, have been very reliable in operation, with Phénix and PFR together logging over 1.3 million pump operating hours.

Neutron absorber performance has been continually monitored in the operating plants. The data confirm the fundamental reliability and good performance of this engineered safety system. Cleaning and exercising of absorbers has been found necessary to counter effects of sodium aerosol deposition on extension rods or magnet faces. This experience is an important guide for future absorber design and for other mechanisms working in the argon cover-gas region above the primary sodium.

For Phénix, the major source of forced outage has been the intermediate heat exchangers. Leaks of secondary sodium were detected in the annular interspace between the cold inlet and hot outlet ducts in two separate instances in July and October 1976. The cause of the leaks was found to be mechanical constraint leading to overstressing of welds in part of the secondary sodium duct. Repair to a revised design was necessary for each IHX. Removal and modification of an IHX has always been recognized as one of the most difficult potential maintenance tasks on pool-type reactors. The accomplishment of this task on all six IHXs at Phénix with the minimum of reactor downtime and radiation dose to plant operators, represents a conspicuous success and demonstrates the importance of foreseeing events at the design stage, making adequate provision of equipment and space for the repair, planning and learning from repeated operations. The importance of having spares of certain major components available has also been recognized.

In Superphénix vibration of part of the reactor internal structure occurred at certain core flow rates, caused by waves on the sodium surface exciting one of its resonant modes. It was found that the sodium level at the main vessel cooling weir (deversoir) was a critical factor. By changing the flow adjusting gags in some 20 fuel subassemblies and thus increasing the flow rate over the weir, it was possible to reduce the fall over the weir, taking vibrations into the quiescent region for all normal operating conditions.

(c) *Steam generator performance*

Steam generator designs differ significantly among the five European plants as follows:

KNK II	concentric tube;
Phénix	modular serpentine tube;
PFR	U tube;
SNR300	both straight-tube and helical;
Superphénix	large helical.

The performance of the steam generators in Phénix, PFR and KNK II to the end of 1987 is summarized in table 3.

TABLE 3. STEAM GENERATOR PERFORMANCE TO 31 DECEMBER 1988

plant	steam generator circuit	operating hours	number of leaks
Phénix	1	88500	2
	2	81900	1
	3	84900	1
PFR	1	66000	9
	2	45000	22
	3	50000	9
KNK II	1 and 2	26000	1

Tube-to-tubeplate weld leaks in evaporator units have been a severe source of loss of availability on PFR with the bulk of leaks occurring before 1980. The main cause was hardness of the welds due to lack of post-weld heat treatment and a combination of sodium, caustic and other impurities. Fitting of explosive-welded plugs proved effective for returning the units to operation following early leaks. However, sleeving of all 3000 tubes to bypass the tube-to-tubeplate welds in the three evaporators has, since completion of the task in 1984, eliminated the problem entirely. In early operation and in recent years the austenitic superheaters and reheaters have also given trouble. After the early superheater and reheater failures, a complete set of replacements was ordered to a different design eliminating the hard highly stressed tube-to-tubeplate weld and in 9Cr–1Mo ferritic steel rather than austenitic stainless steel. With the exception of one reheater, it was possible to repair all the units and keep them in service. Operation on three circuits with one reheater missing caused a small reduction in station performance, but its duty was essentially shared between the reheaters in the other two circuits. As soon as the first new unit became available it was installed allowing routine full power operation. Subsequently, there have been leaks on two of the original superheaters and the opportunity has been taken to replace all of the original superheaters and reheaters with new units.

The superheater leak in February 1987 occurred during full power operation. It was the first under-sodium leak in the plant's history and was both larger and developed more rapidly than previous events. The automatic protection system operated as designed, tripping the plant, dumping steam and sodium and isolating the IHXs on secondary circuit 2. In the subsequent shutdown careful attention was given to removing all reaction products from the sodium in this circuit.

Phénix has experienced four sodium–water reactions in steam generators, with the first taking place in April 1982 and the last in August 1983. The failures were all located at the upper bend of a reheater module and were due to faults at the heat affected zone of a butt weld in a tube. Only on the first occasion were there complications, with sodium contaminating the 12 steam circuit reheater modules in the circuit affected. There were no consequences for plant safety. These systematic failures, occurring after nine years of operation resulted in a decision to replace all the reheater modules and this has now been done. In addition, immediate modifications were implemented to improve steam generator surveillance.

Experience with the KNK II ferritic (8 Cr Mo Bn 910) steam generator has been satisfactorily. To date only one minor sodium–water reaction has occurred and that was in early operation. The cause was faulty welding in the vicinity of a spacer. There were no consequences for plant safety.

These experiences have influenced design thinking and affected the direction of future R&D work on steam generator leaks. In particular greater emphasis is being placed on the early detection of leaks and the removal of the reacting fluids to minimize the consequences for the steam generator. The importance of condition monitoring during operation (e.g. acoustic monitoring to characterize vibration and warn of changes) and in-service inspection (e.g. inspection of tubes and welds for defects by ultrasonic or eddy-current methods) has also been highlighted. R&D is being carried out to improve the understanding of the characteristics of sodium–water reactions and to improve leak detection methods. Nevertheless, it is clear that, even in the event of a severe sodium–water reaction, the system design will meet plant protection requirements.

(*d*) *Integrity of sodium circuits*

The designs have been entirely successful in preventing leaks from primary circuits. The simple, high integrity, double-walled vessels fabricated in austenitic stainless steel provide a strong guarantee that sodium coolant cannot be lost. In the more complex secondary circuit pipework, leaks have occurred with consequences for plant operation but not for plant safety. Such leaks, mainly from cracked welds, developed slowly and were easily detected.

The leak in the carbon steel (15D3 type) fuel storage vessel of Superphénix caused a major interruption in the early operation of the plant, although the reactor itself was unaffected. Investigation indicates that the cracking in the inner vessel wall was caused by a combination of stress and a hydrogen-rich environment before power raising which caused embrittlement of welds. The sodium storage tanks in SNR300 fabricated in a similar material suffered weld cracking failures which were also attributed to hydrogen embrittlement. In the latter case the 10 large tanks were rewelded *in situ*. For Superphénix the existing fuel storage vessel will be removed and replaced by an austenitic stainless steel vessel containing a simple fuel transfer device.

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On two occasions in 1987 and 1988 small leakages of sodium occurred through cracks in welds in the PFR austenitic stainless steel steam generator vessels. Ultrasonic inspection revealed a number of other defects in the welds. The defects were either repaired, instrumented, or identified for future in-service inspection. Investigation has shown that each major defect is associated with weld repair during fabrication of the vessels, emphasizing the importance for the future of setting clear specifications and improving fabrication and inspection standards.

(e) *Personnel doses and environmental aspects*

Radiation doses for fast-reactor operating teams have been low in comparison with other reactor systems. For Phénix, PFR and KNK II doses have generally been in the range of 60–300 man mSv per year compared with PWR operator doses varying between 2000 and 5000 man mSv per year for 1000 MW reactors. Analysis shows that the major component of fast reactor operator dose was due to special active maintenance operations, such as the IHX repair programme in Phénix, charge machine refurbishment in PFR or modifications to the KNK II core to eliminate gas entrainment. Gaseous and liquid discharges from the operating reactors have been low and well within the limits set by site licences.

4. Future designs of fast reactors in Western Europe

(a) *Economic perspectives*

Since completion of the prototype reactors in the 1970s, design work in Europe has continued on the forerunners of the large commercial fast reactors expected to be the most economical means of generating electricity in the next century. Superphénix was built to advance the technology of fast reactors a further stage, but its design is not regarded as being economically optimized. Superphénix thus carries a high capital cost penalty, especially when compared with multiple-unit PWR stations in France (specific capital cost 3.3 times greater). Nevertheless, from the experience gained in its design and construction, further steps towards the reduction of capital cost can be made.

During the first phase of the European collaboration starting in 1984, the three national reactor projects, Superphénix 2 in France, SNR 2 in Germany and the Commercial Demonstration Fast Reactor (CDFR) in the U.K., were retained. R&D was harmonized to a large extent during this period. However, in 1987 the major European electricity utilities with an interest in fast reactors proposed that, instead of proceeding with national projects, a single design, to be known as the European Fast Reactor (EFR), should be prepared with the aim of taking a step beyond the national designs, combining the best ideas from each and making a further capital cost reduction. For example in France the Superphénix 2 design had already been estimated to have a capital cost some 38 % less than that of Superphénix. The objective set for EFR, on which work began in 1988, is to achieve a further 15–20 % reduction in nuclear steam supply system costs to demonstrate a design which, in series installation, would have a total capital cost within 20–25 % of that of an N4 PWR (French figures). Cost comparison across national boundaries is difficult, but broadly speaking assessments in the partner countries show similar conclusions for the cost ratio between fast reactors and PWRS.

The European partners have gained substantial experience in the fabrication, irradiation to high burn-up and reprocessing of fast reactor fuels in both dedicated prototype (10 t a^{-1}) plants and thermal fuel plants (by dilution). Designs have been prepared for both intermediate

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(60–$80\ \mathrm{t\ a^{-1}}$) and commercial scale ($120\ \mathrm{t\ a^{-1}}$) reprocessing plants. This work has provided a firm basis for the assessment of the fuel cycle cost component of overall generating cost of future fast reactors and has allowed comparison with the established fuel cycle costs for the PWR. The most recent assessment undertaken in the U.K. indicates that around the year 2000 fast reactor and PWR fuel cycle costs would be similar. By the year 2020 relative improvements favouring the fast reactor are predicted due to improved fuel performance, the introduction of commercial scale fuel cycle plants and the expected increase in uranium ore prices, with the result that fast-reactor fuel cycle costs would be around 75% of those for the PWR. While the achievement of these savings requires dedicated fast-reactor fuel reprocessing plants to be built, whether it will be economically sensible to reprocess the fuel from early reactors in this way will depend on the rate of installation and the view taken on the early proving of commercial-scale plant operation. Alternative means exist, at a time of plutonium abundance, for managing spent fuel in the early stages of commercial development. Storage with deferred reprocessing, use of small dedicated plants or coprocessing with thermal reactor fuel have therefore also been assessed and shown to be comparable in cost.

Allowing for design improvements considered practicable and economies through replication, all cost assessments carried out by the partners indicate that the generating cost differential between future commercial fast reactors and PWRs will be eroded and eventually eliminated as demand for uranium increases and uranium prices rise.

(*b*) *Forward strategy*

The partners agree broadly that there is a need to be ready by 2010 to make a decision on the building of commercial fast reactors as PWRs are retired. The object of the European programme is therefore to have a proven commercial fast reactor design available and to have constructed and operated EFR for a sufficient time, say five years, before this date to confirm claims for economic competitiveness. The design of EFR aims

to reduce capital costs without reducing current safety margins;

to improve operational reliability;

to reduce fuel cycle costs; and

to meet the evolving safety standards set by the licensing criteria of the countries in the European partnership and to seek to maximize the benefits to be gained from passive safety features.

The design features necessary to meet these aims are currently being evaluated in the two-year EFR conceptual design phase, now over half complete. The main design parameters for EFR are given in table 4. The following sections give an indication of some of the design solutions being considered for EFR together with the supporting development activities.

(i) *Reduced capital cost*

The design intent for EFR is to reduce costs as compared with previous reactors. The specific capital cost can be reduced most readily by making the plant simpler and more compact without reducing its power output. Making the primary circuit more compact and reducing the number of secondary circuits lowers costs by reducing the quantity of material required, but presents design challenges as a result of increased mechanical and thermal loadings on the structural components. Reducing the diameter of the primary vessels means that the primary circuit thermal hydraulic performance has to be re-examined. The aim is to

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TABLE 4. MAJOR EFR PERFORMANCE PARAMETERS

reactor thermal output	3600 MW_t
gross electrical output	1520 MW_e
primary vessel diameter	17 m
number of primary sodium pumps	3
number of IHXS	6
primary sodium flow rate	19293 kg s^{-1}
core inlet temperature	395 °C
core outlet temperature	545 °C
number of core fuel subassemblies	346
refuelling interval	2 years
fuel residence time	6 years
fuel burn-up/dose targets	20%/180 DPA
number of steam generators	6

determine the hydraulic and thermal loadings on the major structures of the primary circuit such as the vessel, core support structure, redan, IHXS, above-core structure and roof. The hydraulic loadings are influenced by the behaviour of the free surface and the turbulent flow in the hot pool. Temperatures in the IHX are affected by irregularities in the flow distribution at the IHX inlets.

The important thermal loadings are determined by the mixing of hot and cold sodium, operating transients in reactor power or coolant flow, and by motion of thermally stratified coolant at low flow rates when decay heat is being removed. The most sensitive components are the above-core structure and the redan.

The reduced dimensions of the primary circuit place particular demands on the reduction of the uncertainties in calculation of shielding effects to minimize the induced activation in the secondary circuit sodium. Experiments using representative arrangements of neutron sources, sodium and steel are being prepared.

An established base of materials, as used in existing plants, is available for construction of future reactors. Particularly in the case of 316-type stainless steels, the only need has been to refine the specification. However, for the steam generators, the opportunity exists for significant cost reductions through the use of improved strength materials, such as an advanced variant (T91) of 9Cr–1Mo ferritic steel.

Because of safety and licensing requirements, the cost of nuclear-grade equipment is much higher than that of its conventional counterpart. A design objective is therefore to explore the potential for deploying natural processes, such as natural circulation cooling for decay heat removal, to reduce safety demands on the conventional steam raising plant and the amount of emergency electrical power needed, thereby satisfying licensing requirements and reducing costs.

(ii) *Improved operational reliability*

The considerable operating experience gained from the existing reactors and the experience of dealing with the problems that have arisen provide a firm basis for the design of reliable components and structures. Structural faults, especially in certain of the steam generators, have had a particularly adverse affect on operational reliability and have prompted improvement in design and manufacture.

Early detection of steam generator leaks and rapid shutdown to limit the consequences have been recognized as key requirements for commercial plants. Techniques for repair and replacement of steam generators have been effectively demonstrated should failures occur. The two main aims of the supporting R&D are to validate the material selected for the EFR steam generator units and to gain a better understanding of sodium–water reactions and in particular of the mechanisms by which a leak in one tube might propagate damage to other tubes. Subsidiary aims are to confirm the thermal hydraulic performance, the margins against tube buckling and vibration, and the selection of materials to minimize fretting wear of the tubes against the support grids.

It is important to be able to detect steam generator leaks very quickly so that damage can be minimized by shutting down the affected unit. Leaks may be detected by acoustic or chemical means. Instruments that detect the hydrogen formed by the reaction between sodium and water are very sensitive but cannot be made to respond very quickly. Acoustic leak detectors are faster of response, but are rendered less sensitive by the background noise caused by the normal flow of steam and sodium. Development is in hand to ensure that improved acoustic and hydrogen detection methods are available.

In general it is very important to have validated design rules which will assure the reliability of all the reactor structures, of both primary and secondary circuits. Apart from the obvious safety reasons, this is also necessary so that the incidence of sodium leaks will be very low. The problems posed by the structure of a fast reactor, consisting as it does of relatively thin-walled components in which the primary stresses are low but the thermal stress may be very high, are different from those of water reactors. They are discussed in another paper in this Symposium.

(iii) *Lower fuel cycle costs*

Burn-up limits have been progressively raised in the prototype reactors from original targets of around 7.5% resulting in a significant reduction in predicted fuel cycle costs. The priority is to irradiate pins and candidate clad and wrapper materials to the 20% burn-up and 180 DPA neutron damage dose targets for EFR's later fuel charges. The programme includes fuel pins with the reference austenitic stainless steel clad (15/15/Ti) and the alternative PE16 clad. 15/15/Ti is expected to reach a dose of 150 DPA in early 1990 and PE16 180 DPA during 1991. The pins to be irradiated will be to the conventional homogenous core design. In parallel, core mechanics codes, which describe the mechanical interactions between subassemblies, are being validated by experimental data from other fast reactors and from test rigs. These codes will be used for the calculation of the static behaviour of the EFR core and the forces involved in removing and replacing fuel. Core dynamics codes are also being validated against experiments.

Axially heterogeneous core designs, in which a section of fissile material at the core centre is replaced by fertile material, may also be advantageous in increasing the average fuel burn-up for a given neutron damage dose and reducing control rod reactivity requirements. Irradiation of axially heterogeneous fuel pins in Phénix is planned to achieve a dose of 180 DPA by 1996. Absorbers of advanced design which can remain in the reactor for longer periods are also being irradiated in PFR and Phénix to confirm endurance limits.

(iv) *Safety*

The inherent design features of the liquid-metal-cooled fast reactor (low-pressure circuit, large temperature margin to boiling, large heat sink) contribute strongly to the system's overall safety. Operation of test and prototype reactors has shown in addition that some of the

TABLE 5. EXAMPLES OF NATIONAL R&D MAJOR ROLES

country	plant item	R&D area	R&D objective	facility/test section
France	core	zero-energy physics	validate theoretical methods	Masurca zero-energy facility (CEA Cadarache)
		in-pile safety experiments	confirm fuel safety margins under overpower/undercooling conditions	CABRI reactor (CEA Cadarache)
	reactor vessel/structures	thermal hydraulics	determine temperature profiles for structures and components	Colchix and Cormoran rigs (CEA Cadarache)
	intermediate heat exchanger	structural integrity	assess consequential stress conditions and equipment lifetime	–
	steam generators	major component testing in sodium	component proving	
U.K.	core restraint	core mechanics	confirm loadings in a distorted core during normal operation and fuel handling	Chardis test rig (UKAEA Risley)
	shielding	shielding studies	validate theoretical methods	NESTOR reactor (UKAEA Winfrith)
	core and primary circuit	component testing in water	ensure optimum heat transfer under normal flow and natural circulation conditions	large-scale hydraulic test facility (UKAEA Risley)
	above-core structure	thermal striping	ensure adequate life of ACS against thermally induced cracking	Supersomite sodium test rig (UKAEA Risley)
	steam generators	sodium–water reactions	confirm reliable detection of leaks determine consequences of leaks	Supernoah sodium–water reaction test facility (UKAEA Dounreay)
DeBeNe	reactor roof	cover gas studies inflatable seals	confirm designs of components and especially insulation in the presence of sodium aerosols	RSB test rig (I A Bensberg) NACOWA test rig (KfK Karlsruhe)
	decay heat removal system	component testing thermal hydraulics	ensure optimum heat transfer under decay heat removal conditions	Ramona, Neptun test rigs (KfK Karlsruhe) Ilona sodium test facility (I A Bensberg)
	secondary circuit	leak before break sodium–water reactions sodium fire and aerosol studies	—	—

theoretical safety concerns (fuel melting following coolant blockage or overpower transient, loss of coolant flow due to pump failure, sodium–water reactions) are mitigated in practice. Natural circulation cooling of the core to remove decay heat has been demonstrated. Further experiments are planned in the reactors, to extend the understanding of circulation patterns and allow predictive computer codes to be tested, and in rigs to prove the performance of the heat rejection loops. The programme of work to study the performance under severe transient conditions of fuel pins irradiated to increasingly high burn-up is continuing.

(v) *Other areas of design and R&D attention*

Instrumentation: exploitation of acoustic methods of detecting sodium boiling and other operating abnormalities, of measuring sodium temperature remotely and of inspecting structures under sodium are under development in the U.K. and elsewhere.

In-service inspection and repair: techniques for reliable detection of sodium or cover gas leakage and for volumetric inspection of vessels are under development.

(*c*) *Organization of research and development*

More than 30 years' development of fast reactors in Western Europe has resulted in a well-established technological basis for design, construction, and operation. In the past the European Fast Reactor R&D programmes have supported the operating plants as well as looking ahead to the needs of new reactors. With the maturity of the prototype reactors and the reducing need to support the commissioning of Superphénix, the collaborative R&D programme is now centred on the EFR. The scope of the programme is illustrated by table 5, which relates the components to the relevant R&D areas and objectives and the main European test facilities. The table also shows how the rationalization of areas of expertise and major facilities used in each country, carried out over the past few years, has resulted in the allocation of lead R&D roles among the major partners in the collaboration.

5. Conclusions

The development of the fast reactor in Western Europe over the past 30 years has followed a progression from small-scale engineering test reactors to larger sophisticated power plants. The continuing operation of the prototype reactors, Phénix and PFR, and the test reactor KNK II, has demonstrated the viability of the fast reactor in large-scale engineering terms. These reactors have also been instrumental in proving better fuel performance and together with the experience from associated fabrication and reprocessing plants have shown the way forward to a successful and economic fuel cycle.

The 1200 MW_e Superphénix has a leading role in providing the first experience of operation of a commercial-scale plant. Experience gained from its design and construction indicates that capital cost reductions must be made in future plants if generating cost comparability with advanced PWRs is to be achieved early in the next century. The European partners are now collaborating on the design and development of the European Fast Reactor with the object of being ready to construct the plant in the mid 1990s. This timing will allow sufficient operating experience to be gained by around 2010, at which time decisions will have to be made by European utilities on the type of reactor to replace the existing thermal reactor power plants.

Discussion

B. T. Price (*Beaconsfield, U.K.*). Mr Broomfield noted that future uranium price rises would improve the comparative economics between fast reactors and PWRs. As uranium costs make up only some 10% of the PWR generating cost and rises in uranium prices would undoubtedly reactivate the exploitation of relatively cheap reserves, should he rely on this factor for economic competitiveness?

A. M. Broomfield. The illustration I gave was that in the year 2000 fuel cycle costs for fast reactor and PWR would be 0.49 and 0.47 pence per kWh respectively. For the PWR, uranium ore cost was 0.22 pence per kWh, roughly 10% of overall generating cost. In the longer-term projection to the year 2020, we show that fuel cycle costs for the fast reactor will decrease to 0.32 pence per kWh because of improved fuel performance and use of larger, purpose-built fuel plants, whereas PWR fuel cycle costs, even for an advanced mixed-oxide fuel would remain relatively static at 0.43 pence per kWh. In this case uranium ore costs are estimated as little changed in absolute terms at 0.23 pence per kWh.

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