REVIEW OF THE UNITED KINGDOM FAST REACTOR PROGRAMME March 1986*

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ABBREVIATIONS USED IN THIS REVIEW

ACR	Above-Core Restraint
AEA	United Kingdom Atomic Energy Authority
AEEW	Atomic Energy Establishment, Winfrith
AERE	Atomic Energy Research Establishment, Harwell
AGR	Advanced Gas Cooled Reactor
AKUFVE	Apparatus for continuous study of distribution factors in liquid/liquid extraction (Swedish acronym)
ASD	Alternative Shutdown Device
ASME	American Society of Mechanical Engineers
BCD	Burst Can Detection
BNFL	British Nuclear Fuels PLC
BNL	Berkeley Nuclear Laboratories (CEGB), Gloucester
CDFR	Commercial Demonstration Fast Reactor
CEA	Commissariat à l'Énergie Atomique
CEGB	Central Electricity Generating Board
CERL	CEGB Central Electricity Research Laboratory
CFR	Commercial Fast Reactor (The future series of LMFBR following CDFR)
COGEMA	Compagnie Generale des Matières Nucléaires
CSNI	Committee on the Safety of Nuclear Installations
CTS	Central Technical Services, RNE
c.w.	Cold Worked
DFR	Dounreay Fast Reactor
DMSA	Demountable Sub-Assembly
DNE	Dounreay Nuclear Power Development Establishment
DPA	Displacements per Atom
DRDR	*Development Reference Design for Reprocessing*
EdF	Electricite de France
EDRP	European Demonstration Reprocessing Plant
EDTA	Ethylene Diamene Tetra Acetate
efpd	Effective full power days
EIR	Swiss Federal Reactor Research Establishment
ESR	*Electro-slag refined*
FRCC	Fast Reactor Co-ordinating Committee, EEC
HAZ	Heat Affected Zone
HCDA	Hypothetical Core Disruptive Accident

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(Cont'd)

HM	Heavy Metal
HT	Tritiated Hydrogen
HTO	Tritiated Water
HTSL	High Temperature Sodium Loop
HTU	Heights of Theoretical Transfer Units
IHX	Intermediate Heat Exchanger(s)
IRD	International Research and Development Co., Newcastle
IRS	Inner Rotating Shield
ISAT	Individual Sub-assembly Temperature Monitor
ISI	In-service inspection
JEF	Joint Evaluated File
KfK	Kernforschungszentrum, Karlsruhe
LBB	Leak-before-break
LOF	Loss of Flow
LSAS	Links Sweep Arm Scanner
LWR	Light Water Reactor
MBFP	*Main Boiler Feed Pump*
MEL	Marchwood Engineering Laboratory, CEGB
MFCI	Molten Fuel/Coolant Interaction
MFTF	*Molten Fuel Test Facility, Winfrith*
MI	Mineral Insulated
MIG	Manual Inert Gas (Welding)
MMA	*Manual Metal Arc (Welding)*
NDT	Non-destructive testing
NII	*Nuclear Installations Inspectorate*
NNC	National Nuclear Corporation
NRT	Norgett-Robinson-Torrens (model of neutron displacement cross-sections)
N+T	Normalized and tempered
OK	Odourless Kerosene
PFR	Prototype Fast Reactor, Dounreay
PIE	Post-Irradiation Examination
PTE	Post Test Examination
PWHT	Post Weld Heat Treatment
PWR	Pressurized Water Reactor
Q+T	Quenched and Tempered
R and D	Research and Development
RNL	Risley Nuclear Power Development Laboratories
RT	Room Temperature
SFTF	Structural Features Test Facility
SNIP	Sodium Intermediate Plenum Rig
SNL	Springfields Nuclear Power Development Laboratories
SOBOB	Sodium Boiling Bundle
SRD	Safety and Reliability Directorate, UKAEA
SSEB	South of Scotland Electricity Board
ST&A	Solution Treated and Annealed
TBK-OK	Tributyl Phosphate in Odourless Kerosene
TEM	Transmission Electron Microscopy
TIG	Tungsten-Arc Inert Gas (Welding)
TOFT	Time-of-Flight Technique
TOP	Transient Over-power Accident
TREAT	Transient REActor Test Facility, Idaho
UK	United Kingdom
USV	Under Sodium Viewing
ZEBRA	Zero Energy Breeder Reactor Assembly, Winfrith

1. BACKGROUND TO THE 1986 REVIEW

In 1985, nuclear power produced about 20% of the electricity in the United Kingdom (UK), with some of the stations recording very high load factors. The Hinkley Point A Magnox station (430 MW(e)), first connected to the National Grid in May 1965, recorded an annual load factor of 94% and the Hunterston B advanced gas-cooled reactors (1150 MW(e)) recorded an annual load factor of 82%. The latest advanced gas-cooled reactor stations under construction, Torness (1250 MW(e)) and Heysham II (1250 MW(e)), are close to completion within the originally estimated costs and the planned time schedules. Both are expected to be in operation within the next year, bringing the nuclear component to about 20% of UK generating capacity. A particularly noteworthy feature of the UK's gas-cooled reactors has been the low doses experienced by the operators; the Hunterston AGR station recorded a total annual dose (integrated over all staff) of 80 rem, which is much smaller than that experienced by the operators of some light water reactors.

The national perception of the acceptability of nuclear power in the UK improved after the nuclear stations' invaluable rôle was demonstrated during the 1984-85 dispute in the coal industry. But there have been some ill-informed criticisms about fuel reprocessing activities following a number of well-publicized incidents at the British Nuclear Fuel plc (BNFL) Sellafield plant in recent years. BNFL has been making considerable efforts to improve its operations at Sellafield and the past year has seen the commissioning of a £315M fuel receipt, storage and decanning facility for Magnox and AGR reactor fuels, (the building is large enough to hold a 40000 tonne cruise liner), and of an ion exchange effluent plant (SIXEP) to reduce liquid discharges.

The European partners have accepted the concept of a joint commercial-scale reprocessing plant to service the European commercial demonstration reactors envisaged in their collaborative agreement. It could be sited in France or in the UK. BNFL and the UKAEA are planning a plant for reprocessing oxide fuel with an annual capacity of 60-80 tonnes. If all partners agreed, this plant would be located at Dounreay and be designated the European Demonstration Reprocessing Plant (EDRP), providing fuel reprocessing services for the three demonstration fast reactor power stations expected to be built in Europe before the end of the century. An application to the local government for planning permission was submitted in June 1985 and a local planning inquiry opens in Thurso on 7 April 1986.

The outcome is still awaited of the public inquiry into the proposed construction of a Pressurized Water Reactor (PWR) at Sizewell. The inquiry ended in March 1985 after over two years' hearings. The Central Electricity Generating Board (CEGB) has indicated that a successful outcome will prompt the construction of a small series of stations to the same design. Sites under consideration include Hinkley Point, Winfrith, Dungeness, Druridge Bay, Trawsfynydd and Wylfa.

The Nuclear Industry Radioactive Waste Executive (NIREX) was incorporated as a company, UK NIREX Ltd, at the end of 1985.

The examination of potential sites for the storage of radioactive wastes has continued and a statement of intentions is expected soon.

The CEGB and Electricite de France (EdF) completed installation of the first stage of a link of 2000 MW(e) capacity under the English Channel to couple the two National Grid systems, and electricity was first transmitted in November 1985.

Following repairs to its evaporators in 1984, the PFR has operated reliably at or near full power, except in mid-1985 when an output transformer failed causing a delay of eight weeks of power operation, see Section 2.2.

Some of the fuel in PFR has now reached, without failure, burn-ups close to 15%, the preliminary target set for future commercial stations. A main objective of the PFR is to achieve high burn-up with substantial quantities of advanced fuel in order to demonstrate lower fuel cycle costs and improved load factors for future commercial stations.

Some substantial improvements have been made in the design of commercial-size fast reactors. Capital and operating costs have been reduced, both of which have been prime objectives of the UK and European programme. During 1985, the National Nuclear Corporation (NNC) produced new estimates for the capital cost of a first demonstration fast reactor in the UK. These show reductions of around 20% in capital costs compared with figures used during the major previous review of fast reactor policy, and further reductions may be expected for later series-ordered fast reactors. The design work has helped to re-focus the Research and Development (R&D) programme onto the aspects of commercial fast reactor designs where there is potential for further reductions in capital costs and for increased lifetimes and reliabilities.

Recent work has confirmed that total fuel cycle costs for future fast reactors will be substantially less than those for thermal reactors. The total fuel cycle costs for these fast reactors should represent some 15% of the total generating costs, i.e. the cost of generating electricity from fast reactors is not very dependent on the cost of its fuel cycle. Assessments have also shown that in terms of capital costs, joint European fuel fabrication, reprocessing and waste management plants will represent less than 10% of the capital cost of the three European demonstration reactors they could serve (see remarks on EDRP above).

The number of professionally-qualified staff employed for fast reactors in 1985/86 was about 930. Of these, 800 were in the UKAEA with the remainder in NNC, BNFL, CEGB and in other organizations with an interest in fast reactor development. Expenditure in 1985/86 by all organizations was about £116M of which £106M was spent by the UKAEA (£116M less £10M for the sale of electricity from the PFR).

2. PROGRESS WITH THE PROTOTYPE FAST REACTOR (PFR)

2.1 Highlights

1985/86 saw a continuation of the progress reported last year; together, Runs 9 and 10 accounted for 165 efpd at an availability of just short of 70%, excluding refuelling periods. Both major outages, Reloads 9 and 10, were completed to schedule. The principal operating statistics for 1985 were:

Core irradiation	169 efpd
Electricity generated	893210 MWh
Electricity exported	825630 MWh
Load factor (1 January 1985 to 31 December 1985)	40.8%

The most notable aspect of operation was the continued availability of the full set of steam generators, with the achievement of design power, 250 MW(e) on 4 March 1985, and a steady 255 MW(e) in January 1986. The station availability has improved markedly consequent upon the ability to retain water in the steam drums following automatic trips. During Runs 9 and 10 the station provided important data on operation with failed fuel pins. The frequency of trips and unscheduled outages is less than half that for the period 1976-1979. A further achievement was the completion of the steam pipework inspection programme started some five years ago.

The main factors affecting load factor were failed fuel, the replacement of the main generator transformer, seaweed ingress (which 'cost' an estimated £1.8M in lost revenue in 1985), problems with charge machines, the removal from service of the west crane, condenser leaks, seizure of the main boiler feed pump (MBFP), and alternator cooling gas humidity levels.

The statistics given in Section 2.3 summarize operations in 1985/86. Further information on the PFR is given in Sections 3.1, 8, 9.2 and 11.2

2.2 Operating history

The year under review started with the continuation of Run 9 which was terminated in mid-May. After Reload 9 and a transformer replacement, Run 10 commenced in early July and ran, with a couple of significant interruptions, until late November. After a rapid reload, Reload 10, operation at Run 11 commenced immediately after Christmas 1985, is still in progress and is programmed to end in May 1986.

At the beginning of the financial year (April 1985) the reactor was shut down to remove a sub-assembly containing a failed fuel pin. Because of heavy seaweed ingress at the seawater channel to the condenser, power operation was not resumed until 7 April. On 26 April the plant was shut down to clean the condenser and replace the moving seals on both drum screens and part of the static seal on drum screen No. 2. During a routine inspection the cross travel wheels on the reactor hall west crane were found to be cracked and the crane was removed from service until the wheels and their rails could be replaced.

After power operation was resumed on 7 May, the main station generator transformer failed (11 May 1985) and Run 9 was curtailed after 65.5 efpd and Reload 9 began. During Reload 9 a replacement transformer was identified (from CEGB West Thurrock), shipped to DNE, installed (after substantial civil work), tested and commissioned whilst the failed PFR unit was sent for repair.

Shortly after full-power operation was resumed on 14 July, unambiguous signals were received from the gas blanket detectors and the bulk delayed neutron detectors, showing a failed fuel pin in the core. The failure was located in a DMSA cluster. The delayed neutron signals rose steadily and power was decreased to 200 MW(e) and held steady until 23 July when a failure of the condensate extraction pump (CEP) caused a trip of the plant. Power operation at 200 MW(e) was resumed on 29 July but it was apparent that the failure signal had increased further. On 1 August after approximately 15 efpd a controlled shutdown was begun to remove the failure. Apart from the failure it was decided to remove another similar DMSA and five sub-assemblies which were judged to be close to operating limits, the objective being to minimize the risk of further fuel failures during the remainder of the run.

After some difficulties with the main boiler feed pump, normal operation was resumed on 9 September and continued at full power until 16 October, when there was a malfunction of the cell 3 feed water regulator valve. Later, full power operation was maintained until the planned shutdown for Reload 10 on 22 November. Run 10 had extended for 135 days, with a core irradiation equivalent to 100 efpd, and a net generation of 486 x 10^6 kWh and a turbine load factor of 65%.

The significant features of Reload 10 were the smooth operation of the charge machine and, in particular, the reduced time for replacing a control rod guide-tube to one day; the inspection of the deaerator and its clean bill of health; and the return to power operation one day ahead of schedule on 26 December.

Subsequent operation was interrupted by a very heavy and sudden ingress of seaweed (30 December 1985) and limited thereafter to 160 MW(e) by high moisture levels in the alternator until the plant had to be shut down on 7 January to dry the unit. Power operation was resumed on 14 January. This outage as a result of seaweed brought the total lost generation from this cause in 1985/86 to nearly 125000 MWh.

On 23 January there was a large condenser leak, the largest ever experienced in the PFR. The leak was located in the bellows and repaired but the subsequent clean up of the water conditions required the station to be out of operation for nearly seven days. The turbine was synchronized again on 30 January.

2.3 Station statistics

Run 9	In Run	Total from 1 January 1985
Start (turbine synchronization)	25 January 1985	
Shutdown	11 May 1985	
Electrical generation in period (gross), MWh	349910	349910
Electrical generation in period (net), MWh	323460	323460
Electrical consumption during outages, MWh	5200	9120
Maximum electrical power level, MWh	253	253
Maximum station efficiency	40.3%	
Generation load factor at 250 MW(e)	55.3%	44.9%
Equivalent full power days (600 MW(th))	65.5	65.5
Maximum burn-up increment	2.0%	1.5%
Maximum dose increment (dpa)	19.1	
Fuel sub-assemblies loaded (at beginning of run)	23	
Fuel sub-assemblies discharged (at end of run)	–	–
Mean burn-up of discharged sub-assemblies	–	–
Maximum burn-up in core (sub-assembly)	10.3%	
Mean burn-up in core (sub-assembly)	8.5%	
Maximum burn-up in core (cluster)	13.6%	
Maximum burn-up in blanket (sub-assembly)	1.26%	

Run 10		
Start (turbine synchronization)	10 July 1985	
Shutdown	22 November 1985	
Electrical generation in period (gross) MWh	525540	875450
Electrical generation in period (net) MWh	485860	809320
Electrical consumption during outages MWh	9048	23050
Maximum electrical power level MW(e)	248	253
Maximum station efficiency	39.1%	
Generator load factor at 250 MW(e)	64.3%	44.8%
Equivalent full power days (600 MW(th))	99.6	165.1
Maximum burn-up increment	3.1%	
Maximum dose increment (dpa)	18	
Fuel sub-assemblies loaded (at beginning of run)	Reload 9 None	
Fuel sub-assemblies discharged (at end of run)	Reload 10 19	
Mean burn-up of discharged sub-assemblies (Average mean centre plane)	7.8%	
Maximum burn-up in core (sub-assembly)	12.7%	
Mean burn-up in core (sub-assembly)	10.8%	
Maximum burn-up in core (cluster)	13.6%	
Maximum burn-up in blanket sub-assembly	1.8%	

Run 11 to 28 January 1986	In Run	From 1 January 1986
Start (turbine synchronization)	26 December 1985	
Electrical generation in period (gross) MWh	95720	77960
Electrical generation in period (net) MWh	88510	72200
Electrical consumption during outages MWh	3100	2880
Maximum electrical power level	257	257
Maximum station efficiency	39.1%	
Generator load factor at 250 MW(e)	48.3%	48.1%
Equivalent full power days (600 MW(th))	18.9	15.3
Maximum burn-up increment	approx 0.5% to 31 January 1986	
Dose	approx 5 dpa to 31 January 1986	
Fuel sub-assemblies loaded (at beginning of run)	17	

Aggregated achievements, 1985

Gross electrical generation	1985 (calendar year)	37217.08 MWd
Generator load factor	1985 (calendar year)	40.8%

3. REPROCESSING

3.1 PFR fuel reprocessing

During 1985, 2.5 tonnes of heavy metal were reprocessed containing 350 kg of plutonium. The peak burn-up was 10.7% and the shortest cooling time before reprocessing was eleven months. The fuel dismantling equipment, dissolver and solvent extraction plant have operated satisfactorily at design throughput. The majority of the fuel processed was manufactured by the co-precipitation route and good plutonium recovery was maintained. The first physically blended (U, Pu) oxide sub-assembly discharged from PFR was also processed and the dissolution behaviour was similar to that of the co-precipitated material.

3.2 Aims of the reprocessing development programme

The main aims of the programme during 1985 were to:

(i) review, consolidate and re-direct the programme where necessary in order to support a UK design for a joint European Demonstration Reprocessing Plant (EDRP) assumed to be situated at Dounreay,

(ii) continue with experimental work to demonstrate the generic sulphate flowsheet and process and equipment choices selected as a result of a series of options studies,

(iii) define the experimental facilities required to implement the future development programme,

(iv) prepare proposals for a joint UK-French reprocessing development programme.

3.3 Spent fuel transport

A study on wet versus dry receipt storage of irradiated fuel at the EDRP concluded that a dry storage system had advantages and proposed gas cooling as the reference transport system. The main research and development task identified was the behaviour of defective irradiated fuel during flask transportation. Following consideration of the practicability of utilizing current Light Water Reactor (LWR) flask designs it was concluded that two existing flasks, if suitably modified, would have the capability of transporting irradiated fast reactor sub-assemblies with end fittings removed, to the EDRP from within the UK and Europe during the early stages of plant operation. The study also concluded that in the longer term a purpose-designed flask would offer substantial payload and cost advantages, and that the necessary development, design and supply could be completed within ten years.

3.4 Removal of sodium from sub-assemblies

Responsibility for examining the requirements to permit the steam cleaning of a gridded sub-assembly in 2-3 h has been transferred to NNC who will include it in a broader scheme to examine the cleaning of sub-assemblies, with higher decay heat outputs, using a direct water immersion technique.

3.5 Fuel storage in water

The experiment to examine the storage in the PFR water-filled buffer store of two sub-assemblies irradiated in the PFR has continued. One sub-assembly has been immersed in the PFR pond water for 37 months with no sign of pin failure. Another sub-assembly which contains at least one failed pin was removed from the PFR pond, placed in a sealed water-filled can and replaced in the PFR pond to await destructive examination in late 1986.

3.6 Fuel sub-assembly dismantling and shearing

Work to date has increased confidence in the viability of the Development Reference Design for Reprocessing (DRDR) route for sub-assembly dismantling. Further features are being introduced into the sub-assembly designs which will lower pin extraction loads and improve operations. An automated single pin pulling machine is being developed as an alternative to the DRDR route and the first prototype machine has been ordered.

The prototype laser beam delivery system and horizontal cutting head which should confirm the feasibility of consistent wrapper cutting without damage to the underlying pins was delivered by Culham to the Springfields Nuclear Laboratories (SNL) in mid-1985. Selected guidance methods are being examined.

3.7 Batch dissolver development

The commissioning of the near full-scale uranium dioxide prototype batch dissolver has started with the first of a series of 10 kg (U) dissolutions in nitric acid. Confirmation of dissolution performance using the full 100 kg (U) design batch loading is planned for early 1986.

The inactive coupon corrosion studies to examine candidate dissolver fabrication materials are proceeding. Further experiments using small-scale batch dissolvers are planned to examine certain fabrication design features in both 310-NAG stainless steel and Inconel-690.

3.8 Post-dissolution accountancy and chemical adjustment

A full-scale accountancy tank weighing facility has recently been constructed at SNL and is being commissioned. A detailed work programme now formulated will be implemented over the next 12 months.

Studies using the PFR fuel reprocessing plant accountancy tank have demonstrated the successful conditioning of the dissolver solution from Pu(VI) to Pu(IV) using sodium nitrite additions. Currently, sodium nitrite is the preferred conditioning agent for the EDRP accountancy tank because of its inherently high efficiency.

3.9 Separation chemistry

The effect of trace components in the odourless kerosene (OK) diluent on the partition coefficients of uranium and plutonium in the U-Pu-HNO_3 – TBP/OK system and specifically for the sulphuric acid flowsheet, is being studied using the AKUFVE automated mixer centrifuge unit.

Mini-mixer settler studies simulating various parts of the EDRP flowsheet have included the Cycle I extract/scrub and strip, Cycle II(Pu) extract/scrub and strip at two different plutonium feed concentrations and Cycle III(U) extract/scrub and strip. Preliminary results confirm the data used for the design study flowsheet.

Mass transfer measurements have been made on the Dounreay uranium-active pulsed column rig in which a depleted uranium solute was transferred between nitric acid and tributyl phosphate diluted with odourless kerosene. Uranium concentration profiles in both phases were measured using a new design of sample collector. Heights of theoretical transfer units (HTU) and mass transfer coefficients were calculated under a variety of pulsing conditions for both forward and back extraction. Under extract conditions the HTU was substantially constant over the range of pulse velocities investigated, whereas under strip conditions, the value of the HTU decreased with increasing pulse velocity.

3.10 Solvent extraction chemistry

The six-column plutonium-active pulsed column rig has been constructed and the glovebox containment has been successfully leak-tested. Commissioning is in progress and the rig has operated under two-phase conditions (nitric acid – 30% TBP/OK). The data logger and the instrumentation are being commissioned. It is expected that the rig will be made plutonium-active in early 1986.

3.11 Waste treatment – aqueous streams

Combined treatment processes are being considered for the medium and low active waste streams from the EDRP, embracing precipitation, ion-exchange, ultra filtration and use of chemical additions in order to maximize radioactivity removal and minimize the bulk of secondary waste retained for disposal. An extensive R&D programme is under way to confirm and optimize process decontamination factors.

3.12 Waste treatment – non-aqueous wastes

Two techniques are being investigated for solvent destruction. The first involves the use of a special purpose-built incinerator in which phosphoric acid corrosion will be overcome by making replaceable those components most at risk. The second technique, which could be used for much larger-scale solvent arisings, is alkaline hydrolysis followed by acid hydrolysis to complete the conversion to inorganic phosphate. Microbiological digestion is also being examined as a potentially simpler alternative.

3.13 Waste treatment – gases

A small rig has been installed in the reprocessing plant to examine the off-gases arising from the nitric acid dissolution of irradiated mixed (U, Pu) oxide fuel pins. This primarily concerns the carbon-14 and tritium content of the gas stream. Initial results from the latest reprocessing campaign indicate that 99% of the tritium is evolved as tritiated water (HTO), 1% as tritiated hydrogen (HT) and about 0.1% in organic form.

3.14 Solid waste – volume reduction and decontamination of sub-assembly components

The device for the compaction of fuel pin hulls has been successfully tested in-cell during which inactive hulls with ragged ends were compacted. Some slight modifications will be needed to permit manipulator operation and maintenance of the equipment under active conditions.

4. COMMERCIAL DESIGN STUDIES

4.1 Introduction

During 1985, increasing attention was paid by the National Nuclear Corporation (NNC) to design work in support of Superphénix 2 and SNR2 by agreement with Novatome and Interatom. The work is carried out in the expectation that an 'Industrial Agreement' (referred to in last year's review) will be signed in due course by the NNC, Novatome, Interatom and NIRA. Consolidation of the design of the CDFR was continued, with good progress on reducing capital costs. Discussions were started on a European common model design incorporating the best features of the designs of each European partner.

4.2 Core and fuel

A programme of restrained core calculations, to give a preliminary comparison of the CDFR and Superphénix core styles, continues. Damage dose and temperature data maps have been prepared for the current CDFR core plan. Wrapper temperature estimates take account of interstitial flows. The size of sector studies has been increased for CDFR from 62 to 103 sub-assemblies, in order to include the shield elements. Results have been obtained for the CDFR core with barrel restraint. Generally, the results are in line with previous calculations although wrapper dilation predictions at core mid-plane have fallen considerably due to a combination of lower internal pressures in the input data, a hardening of the Nimonic PE16 creep rules and changes to the CRAMP coding. The results confirm that Nimonic PE16, behaving as the mean rule recommends for CDFR, will be a suitable wrapper material approaching 15% burn-up.

A major item of work is to evaluate the differences between the CDFR and Superphénix 2 core styles in order to make an informed choice for the common model.

Generally, the structural integrity of the wrapper is endorsed by the irradiation of sub-assemblies under suitable conditions in the PFR. CDFR conditions exceed those in PFR in two respects. The restrained core contacts in CDFR give rise to significant wrapper bending moments and local pad stresses; and the CDFR low baffle in the above-core structure deflects hot pool sodium back into the interwrapper spaces.

Estimates have been made of the combined stresses arising in wrappers from the currently identified loadings. In addition to the stress arising locally at the pads from restrained core interactions and the stresses from coolant internal pressure, the stresses arising from temperature gradients in the walls of the wrapper, associated with hot pool re-entrant flow, make a significant contribution. Guidance has been given to fracture mechanics analysts to help identify the mechanical properties of irradiated materials required most urgently.

More detailed finite element stress analysis of wrappers is being done to determine the stress conditions and distortions which will result from a variety of expected pad contact patterns.

An exercise is underway with BNFL to identify those features of sub-assembly design which could be omitted, simplified or replaced by cheaper alternatives in a future UK advanced fuel sub-assembly. The costs of various existing features have been identified but some of these cannot be omitted without considerable development work.

4.3 Rotating shield and above-core structure

The current phase of analysis work on the inner rotating shield (IRS) has been completed and a report has been issued by the contractor, Vickers Shipbuilding and Engineering Limited, on the results of the 3-D finite element study of the design incorporating reinforcing webs and 35 mm thick plates. This covers the effect of rearranging the shear webs and refined analysis on the penetration tubes and the highly loaded ligaments. The results show that the IRS has stresses well within the ASME code limits for the majority of the structure and only a few very localized areas remain outside the limits under the simulated incident loading conditions. Further analysis and/or minor design changes are expected to bring these areas within allowable limits.

NNC have carried out a stress analysis and code comparison of the flow baffle and shroud tube arrangement during a reactor trip transient, based on the PHOENICS code thermal hydraulic assessment. The ABAQUS finite element program was used to calculate the temperature gradients and the stresses. Several cases of the basic reference design of double baffle were run, with different material thicknesses, to investigate the effect on predicted life. The conclusion is that the double baffle can meet the code requirements, although the thickness of the primary baffle might be slightly thinner than the 10 mm previously assumed.

A proposal to examine the PFR convection barrier for evidence of how it is functioning and to measure its temperature, both at shutdown and during reactor operation, is being implemented. This information will support the case for a similar arrangement of barrier, mercury dipseal and shield bearing in the CDFR design and as a proposal for the common model.

4.4 Absorbers and mechanisms

The primary absorber rod test programme has been completed, the final stages having been concerned with stability. With the rod raised to various positions within its stroke, the lateral movements of the rods were observed first with flow in the channel itself, and then full flow was introduced in surrounding sub-assemblies. The performance of the rod was noted in the first instance with guidance as per the reference design and then with a large (6 mm) radial clearance. Lateral movement of the rod occurred and was limited by the close fit, but in the larger clearance condition the radial movement of the rod was about 3 mm.

4.5 Primary containment

An assessment of alternative support configurations for the Superphénix 2 roof is also underway. The objective is to develop a design which will ease any sodium fire hazard on the roof by allowing any sodium leaking from IHXs/secondary circuit pipework to be drained off the roof space. A number of solutions utilizing variants of the CDFR conical support concept have been drawn out and are being evaluated.

A significant amount of work has been done in support of the common model design particularly with regard to advanced design concepts, including a design layout which is aimed at developing an advanced low cost primary circuit incorporating a syphon feed to the IHXs. Two alternative primary circuit arrangements are being worked up, one with a small hot pool having a cylindrical inner vessel to replace the redan without thermal insulation, and the other with a large hot pool like the CDFR intermediate plenum design.

A detailed 3-D finite element analysis of the 19.2 m dia. roof of the CDFR has been completed. Both thermal and mechanical aspects are covered. Load cases include design, shut down and 0.7 MPa post HCDA pressure. No stress or deflection problems have been found under normal operating conditions. The thermal analysis of the roof has been completed for design conditions. The results show that maximum ΔT within the roof structure never exceeds 46°C and this is found only at the fuel loading penetration. This has enabled the roof insulation system to be simplified and only IHX penetrations now require insulation to achieve temperature gradients.

A number of analyses have been completed to study the effects of total loss of roof cooling. These have been done by different organizations using different computer models. Not surprisingly, there are differences but the general trend indicates a roof structure temperature of about 100°C within the first few days. There is sufficient time for repair and the introduction of emergency cooling.

A preliminary investigation has revealed that a capital cost saving of £600k could be achieved if Type 304 stainless steel is used instead of Type 316 for the cooler parts of the primary circuit (less than 400°C).

4.6 Steam generators

Work has continued on the development of the once-through J-tube design and on examination of important features of European straight tube designs. The effect on capital cost of using different ferritic steels, different design rules and fabrication and development requirements has been assessed and the potential for a 5 to 10% reduction in capital cost is indicated as compared with the present UK design. Having established that there are no feasibility issues in the production of 36 m long tubes (previously 30 m) a revised tube bundle design has been derived with a 35% reduction in the number of tubes, with improved heat transfer performance and a possible reduction in capital cost of around 10%. There is also a strong cost incentive to eliminate the alumina Inconel grid bushes.

4.7 Layout

Layout work during 1985 has concentrated mainly on the arrangement of plant on the nuclear island within a 'vented low pressure' containment building. Initial studies were based on the more conventional rectangular building similar to the PFR. Reconsideration of fuel handling storage and transportation and also relocation of certain facilities off the reactor 'island' led to a radical change of layout. Siting of the steam generators in satellite configuration with simplified secondary circuits and pumps inside the containment building, resulted in a more compact and circular building. A surrounding circular service building also accommodates smaller steam generator cells.

The 46 m dia containment building, about the same as a potential UK PWR building, with a 75 m dia surrounding service building, represents an estimated 20% reduction in volume compared with Superphénix 2 and 45% compared with Superphénix.

4.8 In-service inspection (ISI)

The overall CDFR ISI approach for reactor internals has been reviewed and is providing a basis for the future evolution of the UK approach to periodic and continuous ISI. The periodic inspection approach is being actively pursued by means of the interspace robot, links manipulator and articulated joint development. The EEC has placed a contract with CEGB/NNC to study the various European approaches to LMFBR ISI; the NNC brief is to examine under-sodium ISI and this should provide a valuable insight into identifying common themes for mutually-beneficial UK collaboration.

It is planned to use a links sweep arm scanner (LSAS) for the deployment of an under-sodium viewing (USV) system in the gap between the core and the above-core structure of the PFR to endorse the larger generic 'links manipulator' for sodium viewing applications envisaged for CDFR and other commercial LMFBRs.

4.9 Fuel handling outside the reactor

The study of alternative concepts aimed at eliminating the large sodium-filled core component store and the implications on fuel handling outside the reactor has continued. One proposal concerns the storage of spent fuel in water and work has continued on the preparation of detailed designs and building layouts for the plant and facilities to carry out two possible schemes:

(i) to receive spent fuel sub-assemblies directly from the reactor core at decay powers up to 35 kW, to clean free of sodium, and store in water prior to despatch to the reprocessing plant at ratings of up to 2 kW, or

(ii) to receive spent fuel sub-assemblies directly from an in-vessel store at decay powers up to 5 kW, to clean free of sodium, and store in water prior to despatch to the reprocessing plant at ratings up to 2 kW.

The study will also consider the possibility of immersing sodium contaminated sub-assemblies (after draining) directly into water, as a means of sodium decontamination.

4.10 Secondary containment

The concept of a low pressure containment necessitates the inclusion of a highly efficient filtration system for the removal of sodium combustion aerosol in the unlikely event of a sodium fire. NNC has continued with the testing of a Buffalo Forge gas scrubber unit which was identified as best fitting the design requirements, having high efficiency filtration, high loading capability, low pressure drop and low power requirements. Experiments have been carried out at Berkeley Nuclear Laboratories with AEE Winfrith making the aerosol measurements. To date, the test results have been disappointing as the efficiency of the scrubber unit has been below that expected. One of the reasons may have been the choice of particle size of the aerosol; a further series of tests carried out with a modified experimental procedure produced larger diameter particles, the results of which showed a significant improvement in removal efficiency. In any accident in a low pressure containment it would be expected that particle sizes would be large owing to the long transit time from the fire to the scrubber.

5. STRUCTURAL INTEGRITY

5.1 Introduction

A commercially viable fast reactor must be capable of achieving more than 30 years service. Premature shutdown because irreplaceable fixed structures have become unserviceable, or cannot be shown to be capable of safe operation, is unacceptable. Demonstration that primary and secondary circuit structures can survive for this period is therefore an essential part of the design process.

Many fast reactor structures are subjected to service conditions which are beyond the range of present design experience. Existing design procedures are inadequate to provide a degree of assurance of the required long-term life and this is particularly true of the high temperature structures subject to thermal shock and thermal fatigue. As a result of research already carried out, the factors which control Structural Integrity are beginning to be understood.

In some instances the work is sufficiently far advanced to enable tentative design and assessment procedures to be established for use on a trial basis. Refinement and improvement of the procedures will continue in parallel with experiments on a large scale to demonstrate their validity. In a few areas, work of a more fundamental nature must be completed before the factors that govern long-term life of the component can be established. This mainly applies to components that operate at core outlet temperature, one of the main uncertainties being the performance of welds. Until this research has been completed it will not be possible to provide design procedures giving the necessary assurance that the required working life can be achieved.

5.2 Structural analysis

This work comprises simplified methods of analysis, ratcheting rules and the development and use of computer programs.

The development of simplified methods to establish structural shakedown has proceeded sufficiently far to provide an interim procedure, which will become generally available early in 1986. Work is also in hand on the complementary computer code to establish optimum residual stress distributions.

The use in Code case N47 of a Design Stress, S_o, in addition to a high temperature Design Stress, S_t, leads to unnecessary complication. A recommendation that S_o be deleted has been examined and incorporated in Design Procedure documents.

The enhancement of peak strains which results from plasticity and creep has been examined for two typical geometries, a fillet weld and a tubeplate ligament. Elastic-plastic-creep calculations have been completed which show only a small degree of strain enhancement when the loading is within shakedown. Consideration is being given to generalizing the results in simplified design formulae.

Investigations by Professor Ponter and his colleagues at Leicester University have shown that the ratcheting clauses in Code case N47, and the accompanying diagrams, are inadequate.

Experimental verification of shakedown and ratcheting predictions in the UK have so far been restricted to two-bar tests at Leicester University, and work below the creep range at Liverpool University. These will be supplemented by tests on more realistic structures using a new rig at SNL Springfields due to come into operation in 1987.

For the computation of stresses and strains in the high temperature structures, the ABAQUS finite element analysis program, developed in the United States, continues to be used extensively by NNC and the UKAEA. It has been employed in a series of benchmark calculations organized by the CEC FRCC Codes and Standards Working Group. ABAQUS has performed as well as the programs used for the design of fast reactors elsewhere in Europe. An important point to note is that the main discrepancies between experiment and calculation occur as a result of inadequate description of material behaviour, not because of numerical inadequacy.

A considerable amount of high temperature deformation data has been accumulated, partly as a by-product of materials tests and partly from specific investigations. A start has been made on interpreting these data for structural analysis. Initially, emphasis will be on the simple descriptions of materials behaviour which adequately represent the main effects. This will be in preference to the derivation of complete constitutive equations, favoured by some of the European partners.

5.3 Defect assessment

The overall aim of this work is to provide a validated procedure for assessing the effects of defects on the integrity and safety of fast reactor components. An important step in the safety case for many components is to demonstrate that, if a defect were to propagate, observable leakage would occur well in advance of failure (leak-before-break).

For those parts of the fast reactor, including the main vessel and the core support structure, that operate at temperatures below the creep range, considerable progress has been made towards establishing procedures. The full validation of these procedures and the provision of adequate computerized techniques form an essential part of the programme.

At the higher temperatures, where time-dependent materials properties are significant, the way forward is less clear. Before assessment procedures are established, a better basic understanding is being sought of the phenomena involved.

During the past year, considerable progress has been made in the extension of the CEGB R6 defect assessment procedure to cater for strongly strain-hardening materials, including the austenitic steels which are almost exclusively used in fast reactor structures. Revision 3 of R6, which includes these developments, will be issued early in 1986.

The so-called wide-plate tests at Chapelcross have been used to validate the R6 procedure. Initial discrepancies between predictions and experimental observations are now being resolved. The wide-plate tests have also been used to check assumptions about the rôle of residual stresses in fracture, and the additional margins of safety introduced by stable tearing of the defect. Further work on both these topics is to be undertaken in 1986/87.

During the year, NNC have completed a study of leak-before-break as applied to the safety case for the primary vessel. This shows that defects of the size and shape most likely to occur in the welds will, if they were ever to propagate, give rise to leakage without failure. Further investigations are needed in certain areas of the vessel; also confirmatory tests are required. Nevertheless, leak-before-break seems to provide a powerful safety argument, and a formal assessment procedure is being drawn up.

On computational matters, good progress has been made by CEGB on methods of calculating fracture parameters in components where thermal and residual stresses are present, both key factors in fast reactor structures.

Final validation of fracture assessment and leak-before-break procedures must be obtained from tests at a scale representative of full sized reactor structures. Two large-scale facilities have been proposed for this – the *20 MN bi-axial wide plate rig*, and the *100 MN Structural Features Test Facility* (SFTF). Work is well in hand on the 20 MN rig, and commissioning is forecast for early 1988. Authorization to proceed with the SFTF has been obtained, with completion scheduled for 1989.

5.4 Life assessment

This work is primarily concerned with establishing the design and assessment criteria to be applied to structures which operate at high temperature, where time-dependent creep effects are significant. For the fast reactor, thermal shock and thermal striping are of prime importance, compared with the more usual steady-load creep conditions. The programme is oriented towards the failure criteria which are to be applied. This calls for an integration of the work of structural analysts and physical metallurgists.

Work in the UKAEA and CEGB has confirmed earlier doubts about the validity of the rules for combined creep and fatigue, which originated in Code case N47, which are also repeated in the French RCC-MR and the Japanese design guide. It has therefore been decided, for UK purposes, to replace these empirical procedures with new rules based on ductility exhaustion. The background documentation is currently being compiled, with the intent of introducing a new procedure during 1987.

Experimental work on small components, which simulate some of the principal features of fast reactors, has continued in the UKAEA, NNC and CEGB. Such tests are necessarily expensive and time consuming; moreover they require detailed inelastic analysis, which stretches computing expertise and knowledge of constitutive equations to the limit. Nevertheless, some worthwhile progress has already been made, and the results of the experiments are seen as an essential step in validation of the new assessment procedures which are being developed.

The eventual aim must be to consider the integrity of a fast reactor structure subjected to both thermal striping and thermal shock. New investigations are in hand to study thermal striping in isolation, both theoretically and experimentally. Test work so far has been confined to relatively small specimens in the SOMITE rig, which can subject specimens up to 12 mm diameter to highly incoherent striping. This will be supplemented by SUPER SOMITE, which can subject a specimen of 72 mm diameter to conditions similar to those encountered in the above-core structure on a fast reactor. SUPER SOMITE is due to be commissioned in 1987.

The theoretical work, centred on the CLOUDBURST computer program, is intended to evaluate crack growth and crack arrest under thermal striping conditions.

5.5 Crack growth

As already noted, much more research is needed before the fundamentals of defect growth at high temperature are properly understood, particularly under combined creep and fatigue conditions. The position is very different from that at low temperatures, where the fundamentals of fracture are sufficiently well understood to provide a starting point for the development of assessment rules.

The propagation of cracks under steady creep conditions has been investigated in CEGB over the past decade. The emphasis has been on steels used in fossil-fired power plants. During the past year, a procedure has been developed to assess structural integrity under these conditions. The next step must be to decide how far this procedure can be applied to the fast reactor structures which are subject to steady loading at high temperature.

More important for fast reactors are the factors which control crack propagation and the cyclic loading at high temperature. Preliminary tests suggest that the problem is quite complex, and that much further research will be required, which brings together physical metallurgy and continuum mechanics concepts.

6. ENGINEERING AND COMPONENTS

6.1 Primary circuit

6.1.1 Hot pool thermohydraulics

The NNC 1/15 scale water model of the CDFR has been used as the main tool for investigating the behaviour of the hot pool under a range of operating conditions, with the object of confirming or improving the design.

Experimental work has finished and a broad data base for the hot pool and its structures has been built up. Pressure measurements from the water model have been used in the calculation of flow interaction with the intermediate plenum. It is necessary to know the amount of flow passing through the intermediate plenum in order to assess its thermal resistance.

The 1/15 scale model has also been used as part of a joint NNC-RNL study of the behaviour of the free surface. The formation of vortices and the entrainment of gas are particularly important and difficult to predict, the more so as the relationship between scale models and the full-size plant is not known. A sequence of water models is therefore proposed, of which the 1/15 scale model is the first step. A geometrically similar 1/8 scale model has been approved as the next stage, and the final step would be a 1/2 scale model, which would be incorporated with a rig for studying flow between the fuel sub-assemblies. The 1/15 and 1/8 scale models take in the full 360° of the hot pool, but the 1/2 scale model can cover only 90°, so part of the experimental programme for the two smaller models involves the insertion of baffles at 90° to determine the effect. The main part of the experimental work is at the planning stage.

6.1.2 Thermal striping

The avoidance of thermal striping damage is a significant constraint on the design of a number of components, in both primary and secondary circuits, but the most severe problems arise in the hot pool of the primary circuit where streams of sodium from core and radial blanket mix at markedly different temperatures. The above-core structure, which supports the core outlet instrumentation and the control rod guides, is the component most at risk if suitable design measures are not taken.

The basic experimental study of thermal striping has been conducted by RNL and has two aspects; temperature fluctuations in the fluid and heat transfer between fluid and structure. The AKB3 experiment, conducted by RNL in the AKB sodium loop at Interatom, Bensburg, has been the main source of data on fluid temperature fluctuations; and analysis of the measured data continues.

So far, the analysis has concentrated on determining the detailed distribution of peak-to-peak amplitude fluctuations. It is necessary to compare sodium and air data for validating the use of air models to study thermal striping in liquid metals. The general impression gained from the sodium-to-air comparison at the 3:1 flow split and the higher flow rates (roughly equivalent to reactor conditions) is that the agreement is good. At the lower flow rates, the effects of the higher thermal conductivity of sodium are apparent.

6.1.3 Cover gas

The main outstanding technical problems concern heat transfer through the cover gas, which has to be understood if thermal stresses in the roof structure are to be predicted accurately and thermal insulation is to be designed. Heat transfer is difficult to calculate because the effects of sodium aerosols in the gas are not understood and because data on the emissivity of sodium and sodium-contaminated surfaces are not available.

Currently, work on aerosols is concentrated at Harwell. A theoretical model of the behaviour of aerosols, concerning particularly their effect on heat transfer, has been constructed and is being compared with experiment. A sodium heat and mass transfer rig, called the '60 cm rig', has been constructed for this purpose. Dry commissioning tests on the rig have been completed. Tests have been carried out with the floor and roof of the chamber at a variety of temperatures. For each configuration of roof and floor temperature, the side walls have been maintained initially at the floor temperature, and then at the roof temperature. This considerable body of convective heat transfer data has been analysed, and a reasonable correlation for the heat transfer coefficient, applicable to all cases, has been produced.

Sodium was introduced into the rig towards the end of July. First observations of the pool surface indicated the presence of sodium oxide patches. On running the rig for some weeks at temperature, these have disappeared and observations indicate a clean, mirror-like surface to the pool. Tests with sodium have started.

A simpler air-water aerosol rig specifically intended to assist in refining and validating the theory has been proposed.

The work on measuring emissivities is the responsibility of WNL and is done under an extra-mural contract at Manchester University. The emissivity of clean sodium surfaces has been measured over a range of temperatures. In addition, measurements have been made in an argon atmosphere of the emissivities of various steels which have been contaminated with sodium and then drained. These latter measurements are important in the context of fuel transfer machines (see below).

6.1.4 Intermediate plenum

Work continues on the concept of the intermediate plenum, which separates the hot and cold sodium pools. The main outstanding problems are the response of the roof of the plenum to operational temperature transients, and the effect of gaps in the roof, necessary to allow differential expansion, but which also allow a flow of sodium through the plenum, which tends to disrupt the temperature stratification in it.

The OTTER and TIGER rigs at Harwell are providing data, under forced and natural convection conditions for various intermediate plenum arrangements. In particular, the effects of roof porosity are being investigated. In RNL a 1/4 scale water model has provided data on the effects of flows injected below the roof of the intermediate plenum and a model using mercury (MITRE) has given heat transfer data under natural convection in a liquid metal to aid in extrapolation of results from model to reactor. Within NNC, the PIP model has been used to compare the performance of the chimney system with that of the simple sealed flat roof. The permeable designs have shown much better transient response but with some loss of performance in the steady state. The double roof with chimney design has given the best overall performance.

Further experimental work will involve the use of a large sodium-filled rig, SNIP, to measure natural convection effects, a forced-convection version of the MITRE rig, and a 1/25 scale water model at Lucas Aerospace Ltd to help with extrapolation.

Analytical work involves the use of codes such as PHOENICS at NNC, and FLOW3D and ENTWIFE at Harwell. PHOENICS is being used for flow interactions between the hot pool and intermediate plenum. A steady-state run has been made and transient calculations are in progress.

6.1.5 Cold pool, diagrid and core thermohydraulics

Again, the problem is to predict thermal stresses arising from reactor transients, and to provide support for modifications to the design intended to ameliorate these stresses. All experimental work for cold pool development is taking place at Lucas Aerospace Ltd. Two water models are in use generating flow and temperature data under steady-state and transient conditions. Various devices for improving the mixing of hot and cold streams have been tested, including nozzles and baffles on the IHX outlets. A comprehensive set of temperature data has been accumulated, some of which has already been used in stress analysis of cold pool structures.

Some work on heat transfer from the floor of the intermediate plenum into the cold pool has been done, and further work using a direct heat transfer modelling technique in place of the methanol bleed flow method used to date, is intended. The technique is to be proved in a small-scale test before commitment to the 1/11.5 scale model.

A 1/4 scale water model of the CDFR diagrid is being provided by NNC for operation in the RNL small water facility, and is being assembled. It will be used to study the flow distribution and the coolant supply to individual sub-assemblies.

6.1.6 Roof and rotating shields

A series of tests on multi-plate thermal insulation for the underside of the vessel roof has been in progress for some time. The most important question is the effect of sodium aerosols, which tend to penetrate between the plates, on the insulation performance. NNC have mounted a series of tests in an insulation rig, SARI, in the RNL sodium facility SCTR.

The objective of the first SARI test was to determine whether sodium freezes in the roof insulation. It is also intended to provide some data on heat transfer performance although this is a secondary requirement because of the rig limitations.

Careful attention to design has to be given to the sealing of the rotating shield. The PFR mercury dip seal performs very well, but similar seals on other reactors have been plagued by the deposition of sodium in the mercury, which causes it to solidify. Results from a dip seal test rig have shown that the dip seals are very effective against heat transfer and the operational experience with PFR indicates that they also prevent mass transfer. However, there is still no clear understanding of the basic principles involved. A better understanding of how the dip seal convection barrier works would be very useful in connection with European commercial demonstration reactors.

The dip seal rig has been slightly modified to enable mass transfer experiments (using water and air) to be performed. An attempt has been made to predict the dip seal performance in the rig using the TAU code. These calculations used the effective thermal conducticity concept to account for the convection effects within the dip seal.

To contain the pressure pulse which might result from a severe accidental reactivity transient, the dip seal has to be backed by a flap seal to provide positive pressure containment. Throughout the normal life of the plant, however, the rubber flap rubs against its sealing ring whenever the shield is rotated. Tests on the ability of the seal to withstand this rubbing have been completed. Nothing of importance was noted during the tests. On disassembly the contacting lips of the seal were in good condition.

6.2 Component development

6.2.1 IHX thermohydraulics

Work continues to validate the ANTHEA computer code, which is used to predict temperatures in an IHX, and to use the code to assist design. The crucial questions concern the temperature distribution in the tubes, because temperature differences between different tubes cause thermal strains, and it has to be shown that these are acceptable under all operating conditions.

Detailed calculation of the sodium flow distributions using the ANTHEA code has continued in order to study the effects of design changes associated with the edge gap grid resistance, buoyancy, and the effects of reactor trips on the structural response of the unit. The simulations of reactor trips are being performed on the AERE CRAY computer.

6.2.2 *Tube bundle cross-flow and vibration*

Another important problem in IHX design is to avoid vibration of the tubes caused by cross-flow at the inlets and outlets, where the shell-side sodium enters and leaves the tube bundle. Very similar problems arise in the design of the reactor diagrid, as the sodium flows between the sub-assembly support tubes, and in the steam generators.

To address all these problems a programme of basic work on cross-flow in tube bundles has been undertaken at RNL.

A basic tube bundle rig is being operated to obtain pressure drop and force coefficients and flow-dependent excitation frequencies (Strouhal numbers) for cross-flow perpendicular to tubes at Reynolds numbers up to $\sim 5 \times 10^5$ in water. Test sections have pitch-to-diameter ratio (p/d) of 1.3 (to represent CFR IHXs), 1.65 (the CFR diagrid), and 2.0 (steam generators).

All the test bundles are of triangular pitch and can be operated with flows either parallel or perpendicular to rows of tubes, viz. rotated and normal triangular arrays respectively.

6.2.3 *Fuel handling*

If an irradiated sub-assembly in its transfer bucket gets delayed or stuck for some reason in the transfer mechanism, overheating is possible. To be certain that there will be no release of radioactive material it is necessary to demonstrate acceptable fuel pin temperatures during fault conditions. There are two main aspects: heat transfer within the transfer bucket by natural convection in the sodium, and heat transfer from the outer surface of the bucket to the walls of the transfer tube by radiation and convection in the gas, complicated by the presence of sodium aerosols. The work on heat and mass transfer in the reactor cover gas, mentioned above, is important for the latter.

A computer code ALOOP has been written to calculate the natural convective heat transfer within the bucket. To validate this code a 1/3 scale model using mercury as the modelling fluid has been constructed and commissioned at RNL. It is now being used to determine heat transfer rates and temperature distributions and to ascertain the flow patterns. Preliminary results show that the natural convection flow is turbulent (this is an important prerequisite of accurate modelling), and enhances the heat transfer as expected. Detailed comparisons with ALOOP are in progress.

6.3 Steam generators

6.3.1 *Tube-to-tubeplate welds*

Development of the explosive welding technique has been carried out by IRD Ltd under contract. The activities have concentrated on eliminating cracking problems at the beginning and end of the explosive welds.

Wire-fed fusion welding techniques are being developed at Babcock Power Ltd. The feasibility of an orbital wire fed weld has been demonstrated. However, the process is complex with the wire being a source of problems.

6.3.2 *Thermohydraulics*

An experimental study of shell-side flow distributions has been made at NEI Ltd in a 60° sector air model.

Measured flow distributions have been used to validate the hydraulic part of the SGU code BESANT. Analysis of the effects of the perforated grid plate across the outlet window is continuing.

Major computational developments have been associated with generating a 3-D steam generator code based upon the INCA code. The waterside is not yet complete but the representation of the hydraulics is working very well and the effects of the single entry and exit pipes can be clearly demonstrated. A module capable of solving general problems has been written, as well as a sophisticated graphics package for post-processing of results. The INCA code is being compared with experimental results for earlier designs of SGU.

An experiment to investigate flow and temperature distributions in the "J-bend" region is being commissioned. In parallel with the experimental programme, analytical work is proceeding with the aim firstly of correctly predicting the rig results and then of calculating the full size SGU CASE. Initially PHEONICS was used but there is some concern over the result because of possible inadequate turbulence modelling at low Reynolds numbers and a laminar calculation failed to converge. Work, however, continues with this code with calculations of various experimental test cases.

6.3.3 *Sodium/water reactions*

Work on intermediate leaks has continued in the Super Noah Rig. Refurbishment work on the rig was completed. The water pressurizer, which had reached the end of its life, was replaced by a larger capacity vessel and the effluent lines were replaced by ones more closely modelling the PFR design. Following completion of this work, two tests have been performed. These tests simulated the progression of initial small leaks, 6 g/s, and 20 g/s respectively, in a PFR evaporator. Both tests have produced valuable data for validating the leak progression code being developed at Dounreay.

The corrosion occurring when steam reacts with a sodium pool surface has been examined in the small water leak rig and in a glovebox at Dounreay. Results so far have shown that localized impingement-type corrosion damage can occur from horizontal steam jets just above the sodium surface. In other tests, the steam has been introduced into the gas space so as not to disturb the sodium surface. Results have shown that corrosion rates at the surface are lower.

6.4 Instrumentation

6.4.1 *Under-sodium inspection*

Under-sodium viewing is part of the wider field of under-sodium ultrasonics technology. The importance of this work follows an examination of the ASME code (section XI division 3), which is still the only published standard for fast reactor inspection. It defines a rôle for ultrasonics in inspection as providing a pseudo-visualization of potential loose parts and debris under sodium. The same document also defines a need for dimensional measurements under sodium to detect failure of redundant structures. Interest in under-sodium ultrasonics has not only been for inspection, however. Ultrasonic sweep arms, core component identifiers, and alternative shut-down position indicators are also in the development list.

6.4.2 *Boiling noise detection*

Techniques for discriminating the signal for boiling based on filtering to monitor high frequencies or an amplitude analysis to detect an increase in the pulse rate have been developed over the years on the detection system in PFR.

Current work on this topic is directed towards the development of a location capability by treating the transducers available in the reactor as an array. This promises two benefits. Firstly, by protecting the trip instrumentation from signals which arise outside the fuel region the spurious trip rate can be reduced or, in practice, the trip level can be reduced without increasing the spurious trip rate. Secondly, by enabling an acoustic map of the core to be drawn, the emergence of a new acoustic source can be detected at an early stage and monitored. This should provide the basis of a useful warning system for local boiling as mentioned above.

Preliminary experiments in a water tank using a two-dimensional array of transducers above a half-scale model of the PFR core have been completed. An assessment of the delay and sum beam-forming technique has been made for a range of acoustic signals.

6.4.3 High temperature transducers and cables

Ultrasonic transducers capable of surviving and operating at reactor temperature are required. The minimum specification is for a sodium-proof transducer capable of operating at 250°C. A suitable transducer developed at RNL was used in the PFR core top viewer in 1982. The main thrust of the work recently has been towards the provision of suitable transducers and cables for the improved articulated viewer being designed by NNC for PFR. This requires flexible cables which can be bent many times, which will be a feature of any future application of under-sodium viewing.

7. MATERIALS

7.1 Mechanical properties in air of smooth specimens of Type 316 steel and associated weld metal

7.1.1 Type 316L(N) steel

This steel has been selected as the primary circuit construction material for the three reactors in the European demonstration reactor programme.

The mechanical properties of Type 316L(N) steel have been compared with those of standard Type 316 steel. On the evidence available, it has been concluded that the Type 316L(N) steel has similar or slightly improved properties, although there is a lack of long term creep/fatigue and stress rupture data. Plates from two casts of Type 316L(N) steel have been obtained from Interatom for testing, and a plate from a further cast is expected soon from EdF. During 1986 the UK expect to order a large cast of Type 316L(N) steel to a mutually acceptable specification which will be utilized for both mechanical and structural tests.

7.1.2 High cycle fatigue

Strain-controlled high cycle fatigue data are being generated to provide information against which the resistance of structures to thermal striping damage can be judged. Constant strain amplitude data have already been generated but questions had been raised regarding the applicability of these types of data to plant conditions in which the temperature fluctuations and hence the repeated surface strains are generally random. To answer this question, random strain-controlled high cycle fatigue tests have been performed on Type 316 steel using a random trace which is believed to be relevant to an above-core structure. The results show that using a method of cycle counting known as "Rainflow" the cumulative linear damage calculated at failure is about 0.7%; this reduction in endurance from the ideally expected value of 1.0 is small enough to accommodate it within the design factor normally applied to constant amplitude data.

Some special tests have been performed in which rapid constant-strain amplitude cycles were applied during tensile hold periods of creep/fatigue cycles which resulted in the rapid cycles being applied at high mean stress. This type of test resulted in an abnormally low endurance which could best be rationalized on the basis that fatigue damage was caused by the tensile strain range rather than the total strain range. The implications of this finding to the design assessment of structures are still being evaluated.

7.1.3 Irradiation effects

Short term stress rupture tests ($\approx$ 1000 h) on Type 316 steel following irradiation to relatively low helium generation levels, have shown significant reductions in strength and ductility. The change in properties is believed to be due to the helium on the grain boundaries facilitating internal grain boundary cracking, leading to premature failure under slow strain rate conditions. These findings may be relevant to structures subjected to irradiation damage at temperatures in the creep range such as the above-core structure. Longer term tests more relevant to the design conditions are necessary in this area, consequently a programme has been agreed involving the irradiation of Type 316 steel and associated weld metal with varying contents and distribution of boron to vary the helium generation, followed by stress rupture tests to 10000 h. This work has been in progress for about one year. Irradiations have been completed on most specimens and stress rupture tests will commence in 1986.

7.2 Mechanical properties of ferritic steel

7.2.1 Thick section 9%Cr1%Mo steel

A tubeplate forging in the quenched and tempered condition has been obtained from Kobe Steel for evaluation purposes. Since in practice the forging will be subjected to a number of post weld heat treatments (PWHT) various tests have been performed to establish the likely heat treatment condition in which such forgings will be put into service. Microstructure and hardness evaluations have been made after various potential heat treatments in the temperature range 675-775°C. The changes in tensile properties have also been evaluated, and stress relaxation tests were also performed over the previously mentioned heat treatment temperature range. From a consideration of the various results and consideration of practicable heat treatments, it was concluded that a treatment of 8 h at 725°C was most appropriate. Accordingly, blocks of the forging were subsequently given such a simulated PWHT prior to specimen manufacture for the main test programme.

An evaluation of the tensile and stress rupture tests performed to date suggests that for design purposes the strength values for thick section material may need to be reduced by 10% relative to the established strength values for thin section material.

7.2.2 Effect of thermal ageing on fracture properties

The relative effects of thermal ageing on the fracture properties of 9%Cr1%Mo steel and on strengthened 9%Cr1%Mo(VNb) steel have been investigated. Thermal ageing at temperatures in the range 438-650°C reduces the fracture resistance, the effect being much the same for both steel compositions. The fracture resistance is of particular importance in relation to thick sections and future work will involve fracture tests on specimens taken from the 9%Cr1%Mo steel Kobe forging before and after thermal ageing, in order to establish the practical significance of the earlier findings which were based on small specimens taken from thin section material.

7.2.3 *Properties of strengthened 9%Cr1%Mo(VNb) steel*

An evaluation has been made of the available properties of 9%Cr1%Mo(VNb) steel; in general these have been good, and it has been concluded that this steel has a good long term potential, particularly for steam generator applications. There is, however, a lack of long term stress rupture and combined creep/fatigue data; additionally, the weld joint properties require further evaluation. The composition of this steel can lead to complex long term microstructural changes with concomitant changes in properties which have still to be evaluated.

7.3 Mechanical properties in sodium environments

Stress rupture tests in high purity sodium to ≈ 15000 h (2 years) have been completed on wrought Type 316 steel and thin section 9%Cr1%Mo steel. In general, the results show that high purity sodium is not detrimental to the stress rupture properties, and a recommendation for these steels is that their behaviour in sodium can be taken to be the same as that in air. It is therefore no longer necessary to make an additional allowance for sodium in arriving at allowable design stress values for steels with section thicknesses typical of real structures.

7.4 Sodium compatibility

Studies of the mass transfer behaviour of Types 316 and 321 stainless steels in a pumped loop (ΔT 650-450°C) at oxygen levels below 5 ppm have proceeded satisfactorily throughout the year. The aim is to investigate corrosion and deposition behaviour in sodium systems at low oxygen levels, characteristic of those obtained during long term operation of reactor plant. Two more tests, each of six months duration, are required to finish the programme. Corrosion losses recorded on specimens removed from the high temperature part of the loop (650°C) are still in reasonable agreement with predictions based on the existing RNL design equation, and particulate material, similar to that observed in earlier tests at oxygen levels of ≈ 25 and ≈ 10 ppm, has been identified in deposits in the cooler parts of the system. Part of these deposits consists of particles ≈ 2 μm dia. which originated from oxidized steel surfaces downstream of the high temperature region of the loop. Metallography has also revealed sub-surface oxidation in some high temperature regions with the loss of alloying elements.

Investigations to establish the effects of oxide film formation and deposited corrosion products on the heat transfer coefficient of stainless steel tubing are complete. Heat balance measurements were taken over a three-year period at oxygen levels in the range 10 to 25 ppm, using a simple regenerative heat exchanger, operating with inlet and outlet temperatures of 640 and 480°C respectively. These did not reveal any significant changes in the heat transfer coefficient of the steel.

7.5 Nitriding studies

Nitriding studies have demonstrated that the room temperature uniform ductility and ultimate tensile strength of Types 321 and 316 stainless steels are unaffected by exposure to argon/sodium vapour cover gas environments with nitrogen contents of 2000 and 20000 vpm heated to temperatures in the range 520°-620°C, although pure nitrogen environments cause some loss of uniform ductility due to cracking of the nitrided surface. Diffusion (or penetration) coefficients for nitrogen in austenite measured at 620°C give a value of ≈ 3×10^{-12} mm/s. At low nitrogen concentrations (2000 vpm) denitriding of Type 316 steels has been observed in the sodium at 620°C and proposals have been made that denitriding of Type 316L(N) stainless steel should be investigated as part of this programme.

7.6 Waterside corrosion

Post-corrosion examination of a test section designed to study synergistic corrosion effects between chloride and sulphate has been completed. The 9%Cr1%Mo test section containing spark eroded defects had previously been exposed to a heat flux of 860 kW/m² for 4000 h under off-normal water chemistry conditions (3 ppm sodium chloride with 2 ppm sodium bisulphate). A survey of on-load corrosion in the spark eroded defects has shown the presence of intergranular attack to a depth of ≈ 300 μm. This is ≈ 10 times higher than was found in tests with sodium bisulphate at a lower heat flux (660 kW/m²). Transgranular pitting attack was also observed, but to a much smaller depth. Intergranular attack along grain boundaries could be highly deleterious if it occurred at the dryout zone in a once-through steam generator, the points of attack acting as potential stress raisers for the initiation of corrosion fatigue.

7.7 Caustic environments

Slow strain rate tests have been carried out on special casts with low (0.003 w/o) and high (0.061 w/o) phosphorus contents, in both the unaged and aged (1000 h at 550°C) conditions, at a potential of − 400 mV (Hg/HgO). Ageing of the high phosphorus material substantially increased the susceptibility to cracking; times to failure were 94 h for aged material compared with 152 h and 125 h for unaged material.

Ageing, which has been shown to cause segregation of phosphorus and nitrogen at grain boundaries, can therefore result in enhanced stress corrosion cracking, though the susceptibility to cracking is not entirely eliminated by reducing the phosphorus level in the steel to 0.003 w/o.

The effect of ageing under stress was also investigated. Its significance is that plant components usually contain surface defects from manufacture or subsequent ill-usage and the possibility that these will act as initiators for stress corrosion cracking under adverse operational environments, might be increased by segregation of impurities at the 'crack' tip during heat treatment. Evidence of such effects was found, which means that pre-cracked or notched specimens should be included in the future programme.

7.8 Susceptibility of explosive welds to stress corrosion

To evaluate the environmental crack susceptibility of explosively welded tube/tubeplate joints, a peel test has been developed in which the components are separated by slowly straining in the environment of interest; mechanical parameters and post-test examination are used to judge whether such separation has been environmentally assisted. Tests have been performed at 340°C in pressurized water, and also in caustic and sodium-based environments. The programme is not yet complete, but no evidence has yet been obtained to indicate that these media are likely to affect significantly the integrity of such joints.

Peel tests in sodium hydroxide-based environments have been performed at various levels of hydrogen overpressure and moisture contents. Other parameters which have been varied are displacement rate and the addition of zinc contamination to basic melts at low moisture contents. The environmental influence in these tests has been assessed by comparison of the crack growth and interfacial crack tip ripple size with that obtained for a test in argon and from load-displacement behaviour. None of the specimens tested has suffered from severe stress corrosion cracking.

7.9 Inspection development

Future fast reactors will involve relatively thick section austenitic welds and the large plant investment will require extremely high levels of component integrity. This will necessitate rigorous inspection at the fabrication stage although it is hoped that adoption of a leak-before-break (LBB) philosophy will minimize the need for volumetric in-service inspection (ISI). Even so, it does not appear that existing examination procedures will be adequate to provide the required degree of assurance of structural integrity for the future and further development, particularly of ultrasonic methods, continues to be necessary.

Theoretical work at Harwell on ray tracing in inhomogeneous media has been extended to include a model for beam shape and arrival times in a model representing the grain structure in simple manual metal arc welds. The path of ultrasonic waves of arbitrary polarization can be traced through such a system and the amplitudes of rays reflected and refracted at each interface can be collected together with phase information. In addition, studies have been undertaken of the effect of defect roughness on the scattering of ultrasound.

Work has started at Harwell on a study of the potential of horizontally polarized shear (SH) waves for the ultrasonic examination of austenitic welds. Preliminary results indicate that, as predicted, the defect signal to grain noise is significantly higher than when compressional or vertically polarized (conventional) shear waves are used for the examination of welds in plate material. Also at Harwell, work is in progress on theoretical studies of the influence of wave mode on the reflection and skewing of ultrasound beams in austenite. Other theoretical studies have involved a reconciliation of theory and experiment for the behaviour of glycerine-filled smooth cracks and the further development of the theory for roughened crack surfaces.

Virtually all the previous development programme on weld inspection has been concerned with straight butt welds but current proposals for future fast reactors include tee-butt and cruciform welds. Detailed consideration of certain specific design features of tee-butt welds proposed for the CDFR primary circuit has drawn attention to the uncertainties as to the formal fabrication inspection requirements and to the potential problems of applying even a high sensitivity surface inspection. Radiography is not generally acceptable for volumetric inspection of such welds and the occurrence of any delamination will hamper ultrasonic examination even if the extent of the delamination is not itself sufficient reason to reject the weld.

A series of four tee-butt and cruciform welds containing deliberately induced fabrication flaws have been designed and are being fabricated at RNL in thick section austenitic steel plate for test purposes.

7.10 Fabrication techniques

The occurrence of heat affected zone (HAZ) liquation cracking has been studied in thirteen butt welds in Type 316 steel. Only two showed cracking; these were in the steel with the highest boron content and that with the lowest, suggesting that boron is not a major cause. Bead-on-plate comparisons of two 50 mm Type 316 steel plates of similar composition and manufacturing history showed that one was highly susceptible to cracking. A cracking threshold of 1.6-1.7 kJ/mm was found with increasing amounts of cracking as linear arc energy increased. The other showed no susceptibility at heat inputs up to 6 kJ/mm. These plates also showed similarly marked differences in both weld and HAZ cracking in tungsten-arc inert gas (TIG) welded varestraint tests. Sensitivity to liquation cracking appears closely related to HAZ grain coarsening due to increasing heat input which concentrates grain boundary impurities such as sulphur and phosphorus. In the steels which were not susceptible, the presence of ferrite or formation of ferrite in the HAZ probably provides a sink for these impurities.

8. *SODIUM CHEMISTRY*

8.1 Activity transfer in sodium circuits

8.1.1 Basic radionuclide transport studies

A major investigation on the Active Mass Transfer Loop at Harwell into the transport and deposition behaviour of elemental caesium (traced with ^{134}Cs) has just been completed. Initial results suggest that caesium did not deposit so readily on loop surfaces as did zinc, considerably smaller amounts having to be added during the run to maintain the caesium-in-sodium concentration. While caesium showed some preference for deposition in the colder loop regions this was not as strongly marked as for zinc, and it is unlikely that as high a proportion of the total caesium deposited will be found in the cold trap as the 45% recorded for zinc.

8.1.2 Radionuclide distribution in the PFR

Further vault scans were made during shutdowns in 1985 as part of the continuing investigation into the deposition of active species (notably ^{54}Mn, ^{60}Co, ^{137}Cs) below the diagrid in the primary circuit. A preliminary assessment indicates that all nuclide levels increased between shutdowns in November 1983 (when last measured) and June 1985. In particular, ^{137}Cs and ^{134}Cs increased following a fuel failure in the Spring of 1985. In contrast, measurements during the shutdown in December 1985 showed no significant increases from the previous scan, apart from ^{54}Mn, despite further fuel failures. This may reflect the early removal of the failures after detection compared with the previous failure which was left in the reactor for some two and a half months.

8.2 Manufacture, development and performance of oxygen meters for sodium circuits

A study of the electrochemical performance of a large number of Harwell oxygen meters over several years has shown that the accuracy with which the oxygen concentration can be determined is much greater if a calibrated sensor is used.

The Mk IIa oxygen meter sensor (with a long ceramic thimble) installed on the Secondary Cold Trap Loop (SCTL) of the PFR in December 1984, failed in August 1985 after completing 6200 h operation in the temperature range 380-410°C. As on previous occasions, failure was believed to result from thermal shock. During its time in service a calibration check was carried out (April 1985). The SCTL was isolated and the oxygen meter outputs measured at various cold trap bottom basket temperatures. The results indicated excellent agreement with bottom basket temperatures over the range 5-20 ppm. The meters continue to provide valuable information on oxygen ingress into the secondary circuits and its control by cold trapping. Apart from one meter achieving a lifetime of 14000 h in the temperature range 380-410°C, the average lifetime of the meters at temperature is around 5400 h.

8.3 Sodium impurity control

8.3.1 PFR cold trap loops

A limited inspection of a PFR secondary cold trap basket has been carried out. The basket, installed in October 1983, was removed in June 1984 when the cold trap loading had reached 18 kg sodium hydride and 59 kg sodium oxide. Impurity deposition in the cold trap vessel was confined to a narrow ring at the bottom of the vessel. Deposition within the basket itself was similar to that previously observed. Deposition on the experimental support stool which replaced the bottom doughnut was also similar to that seen previously, i.e. there had been significant deposition on the radial plates of the support where the sodium flow had passed directly over them. There were no obvious signs of blockage within the basket. The measured tritium release from the trap on cleaning (4.2 Ci) was close to the predicted cold trap loading based on the short period of operation experienced by the trap at relatively low power (6 Ci).

A further basket was removed from the secondary cold trap in November 1985 with a final cold trap loading of 15.8 kg sodium oxide and 47.3 kg sodium hydride. The much higher proportion of sodium hydride in the trap on this occasion reflected the continued operation of the secondary circuits with the main source of impurity ingress being hydrogen diffusing into the circuits from the waterside of the steam generator units. Oxygen contamination resulted from circuit maintenance and ingresses from the dump tank during circuit filling. In-service performance measurements on the basket indicated efficiencies of over 80% for hydrogen and 80-90% for oxygen. These efficiencies were maintained throughout the life of the trap.

8.3.2 Cold trap development

The small cold trap development programme for NNC on the Risley Sodium Technology Loop has continued. The initial design of a meshless surface for impurity collection, consisting of a straightforward concentric tube, has been successful in removing oxide and hydride impurities from the bulk sodium. The first run, in which approximately 100 g of sodium hydride was added to the rig over a period of five months, has been completed. The experiment was terminated when efficiencies decreased stepwise to about 10% coupled with a change in the temperature profiles within the trap due to leakage flow around the thermal barrier separating the inlet and outlet sodium streams. Examination of the trap revealed crystalline deposits on the outer cooling wall of the annulus and both walls of the thermal barrier, in particular at the base of the thermal barrier where flow reversal occurs. Analysis of the deposits gave approximately 30% sodium oxide, 35% sodium hydride and 35% undrained sodium. Samples were examined under a scanning electron microscope at Risley. The deposits were highly crystalline, with octahedral crystals typically 100 μm in size.

8.4 General sodium technology

8.4.1 DFR decommissioning

The objective is to proceed to 'Stage 1' of the decommissioning programme which is the safe disposal of all easily removable activity.

Schemes are being prepared for the requalification and automation of the NaK disposal plant and for the removal of radioactive contaminants from the neutralized aqueous effluent by ion exchange.

The last two plant items outside the primary circuit formerly wetted with Nak, Nos 10 and 11 dump tanks, have been despatched for disposal as low active solid waste. The last remaining traces of low level radioactive contamination on the site of these and similar plant are being cleaned up.

8.4.2 Detection of oil ingress into sodium circuits of the PFR

Experiments are being undertaken at Risley to establish methods of detecting the presence of carbon bearing species in the primary sodium and cover gas of PFR, following adventitious ingress of oil from components such as the mechanical pumps. The instruments being used in the investigation are the Harwell Carbon Meter, which measures the level of chemically active carbon in the sodium, and gas phase detectors, which analyse the cover gas for the presence of hydrocarbons (notably methane) derived from the interaction between oil and sodium. The aim is to simulate (quantitatively) the situation in PFR where, in the event of spillage, oil or its reaction products leave the pump and pass up through the hot core (about 650°C) before their presence is detected by the reactor carbon meter situated in the top of the IHX, downstream of the core (temperature about 520°C).

8.4.3 Effects of carbon and oil ingress in sodium circuits

Work at CEGB Berkeley, relating to the carburization of stainless steel in sodium environments, as a result of oil ingress, has continued. In laboratory-scale tests at 550°C, surface carburization has been shown to be more rapid, and carbon penetration profiles more complex, than predicted by simple theory. To aid elucidation of these phenomena, control tests have been performed using a chemically simpler graphite source of carbon, but with conditions otherwise unchanged. The rate of surface carburization was significantly slower, and the carbon profiles more uniform than with oil. The sodium carbon activity produced, as monitored by nickel foil analysis, was unity on the graphite scale, consistent with the sodium-carbon solution being saturated with respect to the graphite source. The carbon activity produced by oil ingress, however, was some two-fold higher ($a_c \approx 2$).

Carbonaceous sodium-oil reaction products, thermally aged for some two years in sodium, have also been used as a carburizing source. The steel carburization observed was characteristic of that corresponding to fresh oil ingress rather than graphite. This result is at variance with the theory that carbonaceous residues should pyrolyse to inertness, and it is significant that such material retains a higher carburization potential than that corresponding to graphite carbon, which has itself been shown to be an effective steel carburizer.

The possibility that surface carburization in cracks, crevices or narrow coolant channels can differ from that experienced by surfaces exposed to bulk sodium is also being investigated. Preliminary results indicate that a given steel surface can be 'shielded' from carburization by other carburizing surfaces in its vicinity, in spite of free sodium access and a carbon source in large excess.

8.4.4 Gas blanket monitoring in the PFR

Argon gas blanket monitoring for hydrogen, oxygen, helium and nitrogen using the Hewlett-Packard gas chromatograph continued throughout 1985. In line with previous observations, helium levels have increased transiently during plant trips. The increases are thought to be associated with gas venting from boron carbide control rods. Calibration of the instrument for methane in the range 0-500 vpm has indicated a lower limit of detection of (currently) 50 vpm and a linear relationship between peak area and concentration.

Gamma radionuclides, in particular ^{133}Xe and ^{135}Xe, have been monitored throughout the period at the argon gas blanket monitoring station. The installed equipment has operated satisfactorily and allowed successful early detection of fuel failures during 1985.

Plans are in hand for improving the monitoring station to allow better control of gas flows and include for the first time an in-situ calibration chamber for the installed gamma spectrometry instrumentation.

8.4.5 Hydrogen and tritium levels in the PFR

Primary circuit hydrogen levels continued to be monitored using the spare Harwell Carbon Meter membrane when circuit temperatures were above 500°C. During the period up to September 1985, when the primary cold trap was on line, hydrogen stayed in the range 0.40 to 0.55 ppm.

Primary circuit tritium levels have also been measured using the spare Harwell Carbon Meter membrane coupled to a proportional counter. Measured primary circuit tritium levels have increased throughout the period from 780 nCi/g (March 1985) to 1150 nCi/g (November). The net increase in tritium levels has been estimated for different plant conditions. With the primary cold trap in line, the net increase is around 4 Ci per efpd. With the cold trap isolated this increases to 5-6 Ci per efpd. In addition, a comprehensive series of tritium measurements has been made throughout the plant and the results are being assessed. The preliminary view is that the current theoretical estimates of the tritium production rate is high by a factor of about 2. Further tests are in hand to confirm the results.

8.4.6 *Hydrogen ingress to the secondary circuits of the PFR*

Hydrogen ingress rates into the secondary circuits from the waterside of the steam generator units continued to be monitored throughout the year using the under-sodium nickel membrane/katherometer hydrogen detectors. Following the commissioning of Reheater 3, with its replacement tube bundle, in Circuit 3 in January 1985, hydrogen ingress rates were 0.2 g hydrogen/h compared with 0.1 g/h for Circuits 1 and 2 during the same period. These observations were consistent with previous experience and reflected higher waterside corrosion rates in the new reheater unit. As expected, these ingress rates decreased throughout the year until November when they were 0.09-0.10 g/h compared with 0.08 g/h in Circuit 1 and 0.085 g/h in Circuit 2.

8.5 Cleaning and decontamination of the PFR fuel charge machine

During July and August 1985 the Mk I and Mk II charge machines were sodium cleaned and decontaminated using SDG3 solution. Black deposits, similar to those found on the Mk II machine in 1980, were observed on both. Swabs of the deposits were taken for oil/carbon analysis.

Dose rates following the SDG3 solution treatment in the decontamination vessel for the Mk I machine were 50-300 mR/h ($\beta+\gamma$), 50 mR/h (γ) with hot spots of up to 2 R/h ($\beta+\gamma$), 150 mR/h (γ) in the area of the chute, and for the Mk II machine were 30-150 mR/h ($\beta+\gamma$) with hot spots of up to 5 R/h ($\beta+\gamma$), 1 R/h (γ). Activity levels were higher than previously measured, consistent with the small number of pin failures in 1985. Further scrubbing with SDG3 solution of the Mk I machine and some stripping and scrubbing with SDG3 of the Mk II machine, were required before acceptable activity levels were achieved.

9. CORE AND FUEL

9.1 Development objectives

The primary objectives of the fuel programme, which are to develop and endorse core, blanket and absorber designs capable of meeting CDFR exposure conditions and targets, remain as last year although a number of factors have contributed to changes in tempo and emphasis. Continuing assessments of the features affecting fuel cycle cost have reinforced the conclusion that high burn-up and reliability are paramount and must not be put at risk in striving for lesser benefits from other sources. On the basis of informed extrapolation of current performance potential, burn-up targets have, therefore, been raised from 15% to 20% as a step towards an optimum level which may be substantially higher. The associated dose target is about 170 dpa NRT(Fe) compared with maxima achieved to date of ≈ 100 dpa, underlining the key role of materials development in providing clad and wrapper materials capable of retaining adequate physical and mechanical properties at extremely high damage dose levels. Since less importance is now attached to improvements in intrinsic breeding, except insofar as it affects reactivity control, interest in increased smear density or reduction in clad/wrapper thickness has diminished. Minimization of plutonium inventory and doubling time, which determines the timescale on which fast reactors can supersede thermal reactors, are ensured by the UK insistence on high fuel mass rating as a design criteria. Reduction in fabrication and dismantling costs are still pursued but only in areas where they are not capable of being over-ridden by minor performance penalties.

The implications of higher core burn-ups on blanket sub-assemblies and control rods are not yet clear but development programmes are being modified accordingly. Relevant considerations affecting these two items and the equally important restrained core endorsement task are outlined below.

(i) *Radial blanket.* In principle, increase in burn-up target from 2% to 3·4% should be the aim for maintaining the pattern of blanket and core discharges originally conceived. Endurance levels in blanket sub-assemblies, however, are dictated by considerations of increasing linear rating and clad temperature, as burn-up proceeds, as well as by concerns about resistance to irradiation damage and rising internal pressure. Blanket sub-assemblies designed for high burn-up are likely to require radical redesign, including thinner pins and increased coolant flow, the latter exacerbating thermal striping damage potential. Re-appraisal of blanket fuel management economics is therefore to be undertaken to establish optimum burn-up targets before changes in current designs are proposed.

(ii) *Absorber rods.* The exposure target for CDFR control rods is currently 300-400 efpd, equivalent to ≈ 10% burn-up of boron atoms and closely comparable to core fuel sub-assembly exposures required for 10% burn-up. Control rod residence time must be increased to ≈ 800 efpd if parity with core sub-assembly residence times is to be maintained. It is judged that the burn-up corresponding to this exposure, i.e. 20% burn-up of boron atoms, could be achieved and studies of burn-up limits have been initiated to confirm this view. However, both increase in core burn-up target and extension of the interval between reloads necessitate boron enrichment above existing levels, proportionately increasing the rate of boron burn-up and reducing residence time. It follows that core fuel and control rod management must be reviewed in concert.

(iii) *Restrained core.* Preference for the restrained core style specified for CDFR (see Section 4.2) is based on the twin objectives of limiting core component distortion associated with differential swelling and protecting against the consequences of seismic activity. The reliance on low, or nil, swelling wrapper materials to achieve high burn-up weakens, but does not entirely dissipate, the requirement to constrain sub-assembly distortion. The difficulties in establishing the characteristics of a restrained core without resort to the expensive exercise of operating PFR in this mode, encourages exploration of alternative means of protection against earthquakes. Meantime, the restrained core design remains the favoured option and studies aimed at endorsement of the concept continues using laboratory models and the limited irradiation facilities available.

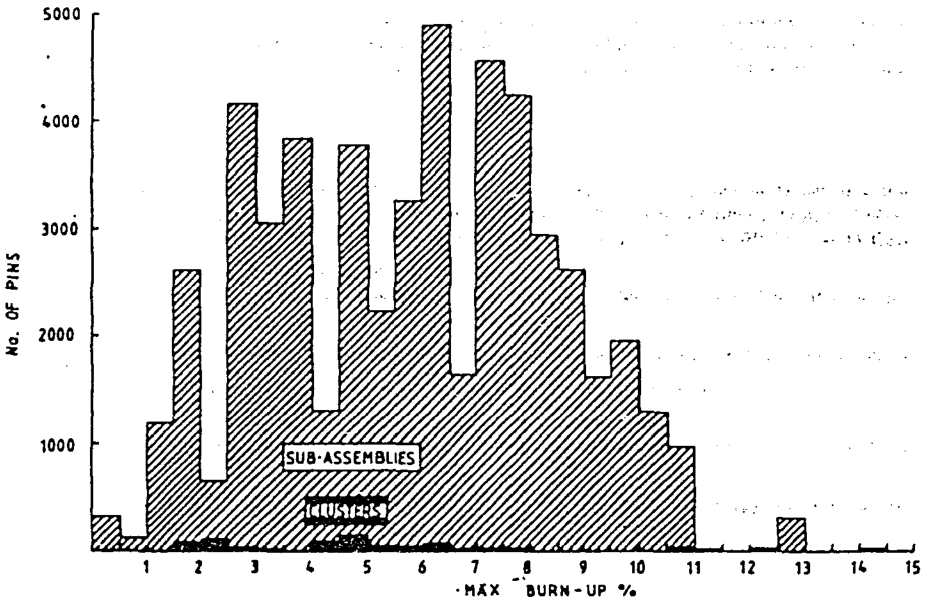
The evolution of sub-assemblies guaranteed to attain reasonably high burn-up targets with vanishingly small failure incidence for PFR driver charge applications has formerly been regarded as complementary to CDFR fuel development. This view has finally been discarded because of the strengthening awareness that the CDFR must be endorsed in PFR by full core loadings over an extended period. Consequently, CDFR designs ultimately become the PFR driver charge, and the emphasis in irradiation programmes has been changed to allow the high burn-up capability of the CDFR designs to be demonstrated much earlier than hitherto.

9.2 PFR experience

9.2.1 *Irradiation progress*

PFR Run 9, 65.5 efpd, and Run 10, 99.6 efpd, were completed in 1985, adding an exposure increment of 165 efpd and bringing the cumulative total for the reactor to 677 efpd.

Overall burn-up statistics for the 54000 pins irradiated up to the end of Run 10 are depicted in Fig. 1. Maximum dose and burn-up levels achieved are given in Table 1.



Total pins 54154

Fig. 1 Burn-up statistics of PFR fuel pins up to the end of Run 10

TABLE 1

Highest burn-up and dose levels attained in PFR at Reload 10
(c.w. = cold worked: S/T/A = solution treated)

Material	Sub-assemblies Burn-up (%)	Sub-assemblies dpa (NRT)	Clusters Burn-up (%)	Clusters dpa (NRT)
c.w. M316 clad pins	10.7	79	14.2	50
S/T/A PE16 clad pins	12.8	90	-	-
Pins containing recycled Pu	8.1	59	-	-
c.w. En58B wrappers	-	57	-	-
S/T/A PE16 wrappers	-	90	-	-
Ferritic/martensitic wrappers	-	43	-	-
Radial blanket pins	1.44	29	-	-

The outstanding advance in 1985 has been the achievement of 12.8% burn-up, 90 dpa by Nimonic PE16 clad pins, 90 dpa by the associated Nimonic PE16 wrappers, which are new highs for sub-assembly components irradiated in PFR. No progress has been made in raising burn-up levels in PFR MkI driver charge assemblies incorporating Type M316 clad, annular pellet fuel pins, although 12.5% burn-up was predicted to be a feasible target for 1985. The explanation lies in that the exposure of pins at high reactor power for 30 efpd at the end of Run 8 had produced a dramatic increase in Type 316 and En58B swelling rates.

9.2.2 *Failed fuel experience*

Three fuel failures occurred during 1985. Early in Run 9, a fuel pin failure was indicated by a cover gas activity transient. A delayed neutron signal was detected twenty days later. This failure was located in a sub-assembly containing Type M316 clad vibro fuel pins; it was discharged at 10.7% burn-up, 58 dpa, at Reload 9a.

The second failure was signalled early in Run 10 by a cover gas activity transient. This failure quickly progressed from the 'gas leaker' phase to yielding delayed neutron signals. The failure was located in a lead pin cluster operating at a maximum linear rating of 60 kW/m.

Fission gas activity transients shortly after the re-start of Run 10 marked the onset of the third failure. These signals continued erratically throughout the remainder of the run but since no delayed neutron signals were detected, positive identification of the failed component was not possible. The probable identity of the failure was inferred from the behaviour of the gas activity transients, and a pin cluster containing experimental wire-wrapped pins of increased diameter was discharged at Reload 10.

The situation continues that pin failures in PFR are confined to experimental sub-assemblies and clusters. No reference PFR sub-assembly pin design, comprising annular pellet fuel in c.w. M316 clad, has yet been seen to fail at burn-up levels to 10.7%. Moreover, evidence from lead pin and pilot pin clusters suggests that, clad swelling considerations apart, reference driver charge pins would be capable of achieving the 12.5% burn-up target originally cited.

9.2.3 *Post-irradiation examination*

Sub-assemblies discharged at Reloads 8, 9a and 10a have been examined during the year, raising the maximum burn-up and dose levels at which quantitative data on irradiation effects are available to 10.7% and 79 dpa NRT(Fe) respectively. These figures compare with upper limits of 9.5% and 57 dpa from earlier reloads, so that the later data would be expected to illustrate any effects developing with increase in dose level. The 1985 PIE results, however, were chiefly of interest in that components discharged at Reload 8 were the first to have experienced a sustained period of steady reactor operation at full power. Evaluation of their behaviour would, therefore, help to resolve long-standing uncertainties concerning irradiation performance at full power and the effect of prior exposure at low power levels.

The key feature emerging from Reload 8 examinations was the significant acceleration in the neutron-induced void swelling rates for c.w. M316 clad.

Length and bow measurements on c.w. En58B wrappers also suggested substantial increase in swelling rates since high power operation began.

Data on FV548 clad pins and Nimonic PE16 wrappers indicate that average swelling rates have been relatively unaffected by the transition to high power.

Destructive examination of standard driver charge pins also disclosed significant differences in the condition of pins discharged before and after the start of Run 8. Pins at burn-up levels to 9.5%, 57 dpa, discharged prior to Run 8 showed few signs of restructuring, apart from equi-axial grain growth. Fission gas release fractions for annular pellet pins were in the range 30-40%.

In contrast, pins subject to high power operation showed significant restructuring. A pin from the lead sub-assembly discharged at Reload 8 (10.7% burn-up, 79 dpa) showed extensive columnar grain formation, variation in centre hole diameter, multiple pellet cracking and fragmentation of the fuel at the pellet bore, and the collection of debris on the fuel support platform. Fission gas release was measured as 58%.

Examination of a blanket sub-assembly discharged at Reload 8 at 1.22% burn-up, end of life rating 390 W/cm, showed the assembly to be in good condition. Selected pins were withdrawn without difficulty. The maximum clad diameter increase was 0.4%. Destructive examination showed the fuel to be in excellent condition with no evidence of fuel/clad contact at any point, suggesting that pin diameter increase had resulted from void swelling rather than fuel/clad mechanical interaction. It was concluded that the pin design would be capable of achieving the specified 2% burn-up target.

9.2.4 *Fuel modelling*

Recent advances in the development of the TRAFIC code, which has now effectively superseded FRUMP in pin design applications, have been made in the following areas:

(i) the addition of a clad corrosion model which simulates progressive weakening of the clad as a function of temperature and burn-up;

(ii) extension of the plutonium redistribution model to incorporate the effect of vapour phase transfer, significantly affecting plutonium redistribution in a transient;

(iii) development and testing of a coupled mechanics/heat transfer version;

(iv) simulation of vibro-fuel behaviour; and

(v) modelling of pin failure development, introducing realistic pin failure criteria, improved constitutive equations for the clad, and modelling of the chemical behaviour of the pin before and after clad rupture.

In parallel with the evolution of more versatile models, a simplified version of the TRAFIC code is being developed which can more readily be used as a design tool.

9.3 Restrained core studies

The development of the restrained core distortion calculation code CRAMP has continued. The points of key interest are:

(i) The ACR/CRAMP code, which includes the effect of absorbers and other features entering the core from above, is being tested in a stand-alone mode.

(ii) Robust friction algorithms cán now be made available to users.

(iii) The appliction of CRAMP to simulate CDFR conditions in the CHARDIS II rig at SNL is substantially achieved.

(iv) A model to simulate the complete withdrawal and loading of a sub-assembly from the charge machine is under development.

(v) A core restraint verification exercise proposed by the UK was accepted by the IWGFR in June 1985. Solutions to the reference problems submitted by the UK (NNC), France, Germany, Italy, India, Japan, the USSR and possibly the USA will be assessed in the UK at the end of 1986.

9.4 Absorber pin development

The principal objectives of the neutron absorber development programme have been, first to endorse the target burn-up specified for the CDFR reference pin design, later to extend service life without sacrifice of reliability by appropriate manipulation of design or operating parameters. Information on B_4C pellet and absorber pin performance, which is available from two sources, viz experiments in DMSA clusters and standard control and shut-off rods in PFR, is added to the data base against which the predictive code BORCON may be validated and new sub-routines tested as they evolve.

The initial aim of the capsule and model pin irradiations in PFR is to underwrite absorber pins for 8% burn-up in CDFR by establishing the behaviour of boron carbide beyond this level and by demonstrating the successful performance of model pins with the appropriate pellet/cladding gap.

Four model pins, simulating the PFR Mk IV control rod pin design with a bottom vent, have completed their irradiation. The experimental features examined were:

Experiment No.	*Cladding*	B_4C	pellet/clad gap %	efpd	Maximum burn-up %
13/08/01	PE16	Nat	1	113	2.75
02	PE16	40% enriched	4	165	7.50
03	M316	Nat	4	319	7.50
04	M316	10% enriched	4	154	7.00

The three 13/08 series experiments with 4% p-c gaps have achieved burn-up levels close to the target value of 8%. The M316 clad pin in 13/08/04 is now undergoing PIE at Culcheth and shows no sign of external damage at a clad strain of ≈ 0.2%.

Only one experiment remains in PFR at present, viz a model CDFR control rod pin, loaded at Reload 10 for 360 efpd exposure to achieve a target burn-up of 10%. The burn-up goal in pin and capsule experiments has now been increased to ≈ 20%, to match improved fuel performance targets, requiring corresponding increases in p-c gap. A capsule experiment, being prepared for loading at Reload 12, aims to achieve 18% burn-up (equivalent to a fuel cycle of two years) in enriched pellets, and will also yield data on creep, restrained growth and the influence of pellet manufacturing and density variants on helium release.

A PFR Mk III control rod is being dismantled for preliminary examination at DNE after ≈ 270 efpd exposure. Three control rods in PFR have been irradiated for 256 efpd and will reach 356-380 efpd at the end of Run 11. This compares with the CDFR target of 400 efpd.

9.5 Clad and wrapper alloy development

9.5.1 *General*

The formal specification of higher burn-up targets for CDFR fuel elements has further accelerated the trend, evident in last year's Review, towards concentration of resources on the development of a small number of technically superior alloy types. Programmes in all areas of materials endeavour have been critically examined and reshaped to give greater prominence to the most promising CDFR candidate alloys at the expense of materials with less potential. These deliberations have led, essentially, to the demise of austenitic stainless steels as clad or wrapper materials on the grounds of inadequate microstructural stability, strength and swelling resistance. However, the best of the group, HL548, has been retained in longer term programmes as a fallback in the event of unsatisfactory performance of the nominally superior alloys.

It is now more clearly recognized that alloys must ultimately be developed independently for clad or wrapper service although, fortuitously, the simpler high strength alloys under consideration retain a role in both contexts. The pre-eminent requirement for clad alloys is seen to be assured high temperature creep resistance during long exposure to high neutron flux; a limited amount of swelling and irradiation creep are judged to be beneficial in deferring clad failure due to fuel/clad mechanical interaction. In contrast, the lowest possible swelling and irradiation creep are called for to minimize dimensional instability and distortion in wrappers whereas the demands on high temperature creep strength are small.

The policy of including pins filled with void swelling specimens in place of fuel in core sub-assemblies, in order to aggregate dose on the shortest possible timescale, has been pursued by the preparation of four replacement pins. The first began irradiation in Run 9 (January 1985) and is due to be discharged in October 1986 after irradiation to a peak dose of 75 dpa in the temperature range 420°C to 560°C. The second pin will achieve a similar dose three cycles later. The third and fourth pins were loaded into CFR Mk II sub-assemblies scheduled for charging into PFR in May 1986 and should receive a peak dose of 140 dpa before discharge in 1989. The last pair of pins are sealed and sodium filled, allowing transfer to another vehicle for further irradiation.

Two experiments began irradiation in EBR2 in January 1985. UK1/4 is the final phase of a void swelling study in which specimens are irradiated at eight temperature intervals between 400°C and 650°C. The experiment is to be discharged after seven reactor cycles in September 1986, when the highest dose achieved will be 95 dpa, NRT(Fe). Further irradiation of selected specimens in PFR vehicles is planned. The second experiment, UK-8, is designed primarily to study fracture toughness changes in ferritic and PE16 specimens irradiated at temperatures in the range 370°C to 410°C, a temperature range which is relevant to CDFR but cannot be investigated in PFR. Void swelling specimens are also included in the rig. The experiment will be discharged after twelve reactor cycles in September 1987 (39 dpa peak) for re-encapsulation of specimens in a second vehicle proceeding to 78 dpa peak by the middle of 1990.

9.5.2 Void swelling studies

Neutron-induced void swelling data have been obtained on a number of CDFR candidate alloys irradiated in EBR2 at doses up to 70 dpa NRT. No significant swelling has been detected in the 12% Cr ferritic steels (FV448, FV607, FI) or in 20% c.w.&A Nimonic PE16. Out-of-reactor studies on the microstructural stability of c.w.&A Nimonic PE16, however, suggest that a cold work level less than 20% may need to be specified to avoid recrystallization during long term irradiation. The swelling of STA Nimonic PE16 is low, even under some controlled non-isothermal conditions imposed during the experiment. Thus the high and erratic rates of swelling displayed by STA Nimonic PE16 components in PFR remain an anomaly for which the more logical explanations are being progressively refuted.

Similar data on PFR oriented materials suggests that the effect of carbide stabilization in the Ti-modified Type M316 variant, P316, has been to extend incubation by 20 dpa NRT compared with c.w. M316. The other two stabilized steels examined, c.w. En58B and c.w. FV548, both showed evidence of high temperature swelling associated with recovery/recrystallization and both appear to be adversely affected by single step temperature changes during irradiation. These observations reinforce previous indications that the ability of near commercial austenitic steels to sustain doses significantly beyond 100 dpa NRT without unacceptable swelling or degradation of mechanical properties is seriously in doubt. It remains to be seen whether the improvements claimed from tailoring alloy composition, as in the specification of HL54B, will be realized.

Density/TEM studies have been completed on cladding from a PFR fuel pin irradiated to 10.7% burn-up, 79 dpa NRT (max). The results show that virtually all the diametral strain measured on this pin is due to void swelling of the c.w. M316 clad.

9.5.3 Irradiation creep

The first measurements of irradiation creep specimens exposed in PFR were carried out during the year. Post-irradiation examination indicated that the irradiation vehicles had achieved their design performances and measurement techniques worked well. The results on helical springs irradiated at 500°C to about 12 dpa NRT(Fe) yielded creep strain rankings which were generally consistent with those observed in PFR. FV 448 and Nimonic PE16 show low and comparable creep strains, FV 548 and Type M316 straining at a very much higher rate. The ferritic alloy FI, however, crept considerably faster than did Type M316, which is not a promising result for the alloy and effectively excludes it from consideration as a wrapper material.

The first pressurized tube samples were examined after irradiation to ≈ 13 dpa at reactor inlet temperature. Broadly, the materials could be separated into two groups: FV607, Nimonic PE16 and P316 showing low creep, M316 and FV548 creeping appreciably more. This pattern reflects that seen in the spring creep tests, although the finding of low creep in P316 was not anticipated. Where several casts of one alloy had been irradiated, cast-to-cast variations in creep rate were clearly apparent.

9.5.4 Thermal creep studies

Uniaxial stress ramp tests have continued on current and developmental alloys. A thermal creep equation has been derived for STA Nimonic PE16 and is undergoing final validation. Derivation of an equation defining thermal creep in the ferritic/martensitic alloy FV448 as a function of temperature from the limited data available, is also being attempted to assist prediction of FV448 clad pin irradiation performance.

The need to relate creep correlations based on unirradiated data with in-reactor creep behaviour is being met by the irradiation of pre-pressurized tubes in isothermal heat pipe rigs at temperatures between 650°C and 730°C. The first heat pipe operated at 700°C for ≈ 5000 h without incident and was discharged at ≈ 18 dpa in November 1986. These heat pipe experiments, and pressurized tube irradiations at lower temperatures, have benefitted from the development at SNL of a new technique for manufacturing pre-pressurized tubes using electron beam and laser welding.

10. SAFETY

10.1 Fuel failure

10.1.1 PFR-TREAT programme

Following the major policy change in the USA to reorient fast reactor development towards small, inherently safe cores, the PFR/TREAT programme has been reviewed and it has been agreed to reduce the number of fuel pin failure tests. The experimental programme was suspended during 1985 and is expected to resume in late 1986.

Analysis of the data from the thirteen tests already completed on UK pins has continued, under the PFR-TREAT agreement in the US and UK, and under the CAPT agreement in France and Germany. In the UK, the methods for analysis of pin failure have been largely validated, and attention is now concentrated on analyses of fuel dispersal as evidenced by the bundle tests.

10.1.2 The CABRI programme

In the CABRI programme for 1985 and early 1986, two transient overpower (TOP) tests and three loss-of-flow (LOF) tests were performed. Three of the tests were on 5% burn-up fuel pins pre-irradiated in PHENIX and two of the tests were the first of a series on 3% burn-up fuel pins pre-irradiated in the PFR. The UK has contributed to this programme by pre- and post-test analysis of the test fuel pins irradiated in PFR and by carrying out PIE of CABRI test rigs. Data from these tests will permit comparisons with fresh and intermediate irradiated fuel and are an important part of the information used to validate the FRAX whole-core accident code. Completion of the testing is expected at the end of 1986.

10.1.3 *Analysis of test results*

With the progression of work from failure analysis to post-failure analysis, the TRAFIC, PINEX-AR, and NASLIP computer codes have become less relevant to the work at Winfrith. The SABRE and SIMMER-II codes have continued in use as the main codes for test analysis in the thermal-hydraulic and post-failure fields respectively, whilst FRAX, with the COMCYL and EPIX modules, is assuming increasing importance.

In the PFR/TREAT programme, SIMMER-II calculations have been completed on two further 7 pin bundle tests. One, a fast TOP on irradiated fuel, showed only minor differences in fuel dispersal from the corresponding test on fresh fuel, a result confirmed by the PTE. The other, a TUCOP test on fresh fuel, showed the importance of the hottest sectors of the fuel pins in calculating dispersal consistent with the experiments. Three further TUCOP tests on 7-pin bundles, which investigate the effects of irradiation and power/flow ratio, remain to be analysed with SIMMER-II.

10.1.4 *The SCARABEE-N programme*

The UK is collaborating in a programme of tests operated by the CEA to gain information on the behaviour of large (19 and 37-pin) clusters over a wide range of extreme sub-assembly accident conditions in order to validate codes and methods for more prototypical bundle sizes.

Six of these large cluster experiments have so far been carried out. They are: three of slow flow run-down to failure at full power; two of instantaneous inlet blockage at full power; and one molten pool test. Three PTEs are complete. One example of each of these three types has been selected for an independent and detailed analysis at Winfrith.

For the pre-failure stage the transient boiling sub-channel code SABRE is being validated for its subsequent use in extrapolation to reactor-size geometries. Experience to date suggests that even at this small size of cluster the wire wraps are probably playing an important rôle in the redistribution of the coolant and thence in the timing and location of failure.

10.2 Sodium boiling

10.2.1 *Boiling codes and analysis*

The major effort in this field is the development and application of the SABRE code at Winfrith for the calculation of single and two-phase flow in rod bundles. The ability to represent geometrical details and distortions of the bundle makes the code a powerful tool in the analysis of both operational and safety problems.

The SABRE development work has been carried out in close collaboration with the team analysing the SCARABEE-N experiments and has, largely, been influenced by the requirements of this programme. In particular, the effect of choice of boiling model on predicted transient behaviour and the application of SABRE to boiling pools have been studied.

10.2.2 *Low power sodium boiling experiment*

The Sodium Boiling Bundle (SOBOB) test is concerned with sodium boiling phenomena at decay heat power levels. Tests will be made on a 19-pin bundle, prototypical of the CDFR design, in a test section on the High Temperature Sodium Loop (HTSL) at Risley. Measurements will start in the Autumn, 1986.

10.2.3 *Ispra 12-pin sodium boiling experiments*

The first stage of an analysis of the data from the gridded 12-pin test section has been completed at Winfrith. A number of problems with the self-consistency of the data have been identified and resolved. The single phase pressure drop due to the grids and the tube bundle have been determined from experimental measurement. The distribution of heat generation along the pin and the heat losses from the different zones of the test section have also been calculated. A SABRE model of the bundle including the heat loss calculations for the various sections has been constructed and calculations on two-phase pressure drop and dryout have been started.

10.3 Fault detection

10.3.1 *Pattern recognition for temperature noise signals*

A study at CEGB, Berkeley of the application of different pattern recognition methods to temperature noise signals has been completed under an EEC contract. A method for comparing the relative merits of the different techniques has been formulated. The work has been performed in conjunction with the University of Compeigne and it is now proposed to produce a joint European report covering all aspects of the study.

10.3.2 *Acoustic detection of boiling*

Data from the multi-pin boiling experiment carried out in the KNK II reactor have been obtained from KfK for analysis at RNL, Risley as part of the European collaboration. The experiment is significant since the boiling source is realistic (a blockage in a 37-pin bundle of electrically heated pins mounted within the core) and the detecting transducers are mounted in the upper plenum in positions similar to those which would be available in CDFR.

In a similar European collaboration on the Siffleur experiment in Superphénix, an assessment is being made of the effectiveness of the acoustic waveguide system by use of a cavitation noise source. This experiment should take place in March or April 1986.

Studies of signal processing have continued at RNL, Risley with an investigation of Array Beamforming Techniques using a half-scale model of the PFR core. The gain in signal/noise ratio achieved has been close to the value predicted by theory. Attention is now being directed to adaptive methods and in particular the Eigenvector Analyser or Orthogonal Beamformer. This method takes advantage of the existing knowledge of the normal background to achieve an improvement in the sensitivity of detection of an emerging new noise source.

CEA have invited the European partner countries to participate in the analysis of the temperature noise data being obtained from Superphenix. The intrinsic thermocouples in Superphénix are monitored by the ANABEL computer to produce records of temperature noise. These records are being offered to CEGB, Berkeley and RNL, Risley for analysis.

10.4 Energetic Molten Fuel-Coolant Interactions (MFCI)

10.4.1 *Studies at Winfrith using thermite generated molten UO_2*

The experimental programme in the Molten Fuel Test Facility (MFTF) has continued with the study of molten fuel-coolant mixing in the development of UO_2/water MFCI. Operations were suspended from January 1985 for rig modifications which enabled the high pressure and temperature experiments in this series to proceed. As a result, the programme of mixing experiments has been extended to May 1986. Preliminary results indicate that at low system pressures with saturated coolant, little liquid coolant was present within the mixture envelope and, correspondingly, no MFCIs were observed. Also, penetration of the melt into the coolant deviated from existing correlations, particularly in narrow mixing vessel geometries.

Following completion of the mixing experiments, the MFTF will be converted to sodium-based operations which are expected to begin in August 1986. Planning has continued for a programme of out-of-reactor studies of mechanical and thermal modes of propagation of fast reactor sub-assembly faults.

10.4.2 Collaboration between Winfrith and CEN Grenoble

It has been agreed that the UKAEA will provide U/MoO_3 thermite charges to be used in the French CORRECT-2 rig at Grenoble to extend the investigation of MFCI effects between uranium and sodium. The special transport clearances obtained from the UK and French authorities for these charges have expired and application for their renewal has been made. It is planned to send the first pair of charges in March 1986 with the possibility remaining of sending a second pair some 4 to 6 months later.

10.4.3 Film boiling investigations at Winfrith

The film boiling collapse experiments are complete and fully reported. A separate steady-state calculation for the thickness of the vapour film and experimental pressure thresholds has confirmed the predictions of the model that a triggered collapse is largely unresisted. Significant pressure pulses are observed as a result of the collapse. They are considered to arise from coherent explosive boiling on direct contact and, as such, could start the initial fragmentation necessary to launch an MFCI. In a limited series of isothermal experiments with mercury drops in water, trigger pulses up to 3 MPa did not cause fragmentation, thereby affirming that thermal processes are indeed involved.

It is believed that these investigations are leading to an improved understanding of the response of boiling films to an imposed pressure transient. Further experiments with the high pressure thermite rig will provide new opportunities for comparisons with the model.

10.5 Sub-assembly faults

10.5.1 Static loading of sub-assemblies

A programme of work has been carried out at AWRE Foulness to study the structural response of an irradiation-embrittled core to the dynamic forces resulting from an MFCI in a single sub-assembly. Some supporting static crushing tests were carried out at Foulness to provide load/deflection characteristics, but the mode of failure in these static tests was inconsistent with that sustained by the sub-assemblies in the dynamic cluster tests. This was due to the method of applying the load to the wrapper assembly, where a sub-assembly supported on three sides was crushed between two thick steel plates, causing the unsupported sides to buckle outwards.

A new rig has therefore been designed and built at Winfrith to overcome the problems encountered with the Foulness static tests. The Winfrith rig simulates the conditions of the dynamic tests by applying a hydraulic load to the top face of the wrapper using a rubber diaphragm.

The improved loading arrangement in the Winfrith tests produced a crushed wrapper shape which differs markedly from the static Foulness tests but agrees with the dynamic loading test, leading to a more representative load/deflection characteristic. The data produced have been used by CTS Risley to improve the modelling of the Foulness dynamic tests (see below).

10.5.2 Analysis of data from damage tets

The SPOKE model, simulating relative motion of one sub-assembly to another, forces on contact, fluid expulsion as sub-assemblies approach and with generalized loading characteristics, has been applied by CTS Risley to predict the response of a single line of up to four CDFR sub-assemblies to an MFCI in one overheated sub-assembly. Unirradiated Nimonic PE16 wrapper characteristics (as determined in a static hydraulic crushing test at Winfrith) were used for peak forces up to 1.2 MN, corresponding to a pressure of 30 MPa for a load length of 0.5 m.

With the crushing of the first row (row 1) limited to the theoretical maximum value of 65 mm, row 2 can undergo severe residual crushing (30 - 40 mm) but rows 3 and 4 will have very little crushing. Defining the energy expended as the integral of the dynamic force-deflection response for row 1, the energy per sub-assembly for maximum crushing is about 40 kJ. Three-row crushing (if ever achievable) would require 1.5 MJ. These results will not increase if a different load length is considered.

It is probably necessary to extend the calculations to cover a wider range of loading characteristics. It would also be desirable to confirm that irradiated Nimonic PE16 wrapper characteristics are not worse than those used for this unirradiated wrapper.

10.5.3 Wrapper fracture toughness

The implications of experimentally determined levels of irradiated toughness for Type 321 stainless steel and Nimonic PE16 on fast reactor wrapper integrity have been explored at Harwell, based on a defect assessment route which incorporates both unstable crack growth and plastic collapse. A starting defect geometry is assumed comprising a long shallow surface-breaking crack at the wrapper mid-flat position under combined bending and hoop stresses generated by uniform internal pressurization.

Preliminary assessments indicate that as thin-walled components, wrappers will fail by fast fracture only for severely embrittled materials/conditions, whereas for modest levels of toughness degradation an entirely ductile, plastic collapse failure mode will prevail. These conclusions will need to be re-examined for postulated cracks located at wrapper corners.

10.5.4 Natural convection in sub-assemblies

Discussions have been held to review the status and future direction of the SONACO project at EIR Würenlingen (an experimental investigation into natural convection in blocked sub-assemblies). Representatives from EIR (The Swiss Federal Reactor Research Establishment), UKAEA Dounreay and Risley, CEA Grenoble and the ETH (Technical University Zürich) were present. It was concluded that there was a strong case for SONACO since it addressed aspects of natural convection in LMFBR sub-assemblies not covered by either the UK or French programmes, i.e. core-plenum interactions via thermal-syphon convective coupling. The use of SONACO for simulating forced-flow fault conditions was also identified and, in particular, the conversion by mixing of abnormal core temperature profiles to outlet temperature noise. Several INCA computer code simulations of the axial (thermal-syphon) cooling mode have been performed at Dounreay for comparison with the SONACO tests.

10.5.5 Support work at Winfrith for the SCARABEE-N programme

Two rigs to study the penetration of molten UO_2 through sections of a fast reactor sub-assembly have been manufactured. One rig (the Small Melt Penetration Rig) is operational and the first test was carried out in December 1985. In this test, molten UO_2 from a 5 kg U/MoO_3 thermite charge was injected into a small test array about 7 pins wide by 19 rows deep to study melt penetration in 1-dimensional geometry. The injection pressure was about 0.6 MPa and this gave penetration through 17 rows of pins. It is planned to carry out either three or four further tests in this rig (varying the injection pressure and the temperature of the test assembly) during the period up to about October 1986 after which the second rig (the Large Melt Penetration Rig) will be commissioned. This will enable measurements to be made on full size sub-assemblies using up to 24 kg of thermite.

The data generated in these tests will be used to assist in the validation of the computer code PROPAGEL, under development at Cadarache, and the UK SABRE code.

As a check on the behaviour of the thermite material whilst it is freezing, a limited number of experiments of its penetration along steel tubes will be made. The first is planned for February 1986.

10.6 Whole-core accident analysis

10.6.1 Whole-core accident modelling

Preliminary work has been completed concerning the assessment of the limiting reactivity addition required to produce any fuel pin failures for a fast TOP fault on CDFR. The results showed that the previous 1.0$ limit determined using the CRAFT code was very conservative. In fact, FRAX indicated that fast reactivity additions of about 2.3$ could be tolerated without fuel pin break-up. This upper limit cannot currently be fully justified on experimental grounds, but indicates the degree of conservatism in the CRAFT results. Work is continuing on assessing the influence of fuel pin burn-up and trip action.

The latest version of FRAX-4 is being applied to two accident types: slow TOP (0.03 $/s) and slow LOF. Calculations indicate rapid shutdown for the slow TOP and low energetic release for the slow LOF using the best calculated reactivity coefficients.

10.6.2 Equation-of-state and properties of materials

The principal objective of this programme at Harwell is to build up a collection of critically evaluated data for the chemical and physical properties of reactor materials for conditions appropriate to accident analysis. An important part of the programme is to assess experimental data from overseas on the equation-of-state of oxide fuel.

Data have been collected and recommendations made for many properties of uranium dioxide and uranium-plutonium oxides. Among the properties which have been assessed are thermal expansion, temperatures of fusion, vapour pressures of the liquids, enthalpy content and specific heat of the solids and liquids, emissivity, viscosity and diffusion. Some detailed attention has been given to the mechanisms of cation diffusion in UO_2; possible mechanisms involving defect clusters have been invoked. Defect clusters are of significance in the trapping of fission product gases within oxide fuel.

An assessment has also been made of the chemical speciation of the volatile fission products caesium, iodine and tellurium during the conditions of hypothetical core disruptive accidents; such information will be required for the estimation of the release of fission products in these accidents.

Experimental studies on the reactions between liquid sodium and UO_2, urania-plutonia oxide and simulated irradiated oxide have been carried out up to 1600°C. This work, financed by the JRC Ispra as part of the European post-accident heat removal programme, has extended the knowledge of these reactions to temperatures up to 800°C higher than previous investigations.

10.7 Containment loading and response

10.7.1 Studies at Winfrith

Further experiments in the WINCON series have investigated the response of increasingly representative models of the CDFR roof to fluid impact loadings arising from a simulated disruptive whole-core accident. Analysis of recent experiments with the containment codes SEURBNUK-EURDYN and EURDYN-03 has indicated that reliable estimates of the mode and level of roof displacement can be obtained provided that collapse or buckling modes are not dominant; it appears that the stiffening effect of the penetration cylinders acts against such response modes. The latest WINCON experiment has provided an initial indication of the response of the IHXs and pumps and the final experiment, now being manufactured, will study the interaction of these components with the reactor roof. Earlier roof response studies were reported in a paper presented to the Knoxville Conference on Fast Reactor Safety and a comprehensive review of UK containment studies was presented in an invited paper at the SMIRT-8 post-conference seminar.

SEURBNUK-EURDYN is now the most widely used containment code in Europe, having benefitted from the extensive validation against Winfrith containment experiments. EIR, Würenlingen continue to support the code development programme under a collaboration agreement extended to 31 December 1986. There have been major improvements to the fluid impact algorithm and a treatment of structure-structure sliding has been introduced. A comprehensive revision of the SEURBNUK-EURDYN input manual is almost complete.

10.7.2 Q and energy source studies at Winfrith*

A new 1-dimensional model of sodium vapour bubble growth during a hypothetical core-disruptive accident has been developed. This includes the compression of the cover gas, the inertia of the liquid coolant, and the absorption of latent heat by a well-stirred zone that is just a fraction of the total mass of coolant. Uncertainties in heat transfer rates are avoided by taking the maximum possible values and, by intentionally minimizing the condensation rate, a plausible upper-bound for pressurization of the reactor vessel is derived. Even when the entire fuel inventory degrades at 4 tonnes per second to particles of 1 mm radius with a saturated sodium pool at 1153K, the predicted peak pressure of 6.5 bar is unlikely to damage the roof seals. Higher pressurizations would require degradation rates and particle sizes totally beyond existing experimental observations with urania-water systems and computer simulations of TUCOP accidents. The principal assumption in the model is that the well-stirred zone of liquid surrounding a bubble equals only the volume displaced by bubble growth. Although this is an entirely reasonable conjecture, validation against experiments is clearly necessary before Q*events can be dismissed as innocuous.

10.7.3 Code development in CTS, Risley

A SEURBNUK code calculation using a 200 MJ energy HCDA bubble has been run for 390 ms. The primary vessel deformation phase was complete at 320 ms, resulting in 0.15% strains. The roof loading data from this calculation will now be transferred to the Risley 2982 computer where they will be processed to provide input data for the ROSHOD and EURDYN codes.

The new ROSHOD model for primary vessel roof deflections with the vault hold-down has been modified by CTS Risley to include a representation of the conical skirt. At present, a linear response of the skirt is used and the parameters for the vault hold-down have been obtained from CFRSIM code modelling. A series of calculations with this updated ROSHOD model has been completed. The base calculations with the new loading calculated by the SEURBNUK code now produces a peak roof deflection of approximately 320 mm. The value calculated previously without moveable skirt and hold-down was approximately 280 mm. Other calculations have investigated features such as a detached hold-down from bed rock, and instantaneous detaching of the skirt from the vault in compression or tension.

The CDFR roof model using the EURDYN3 dynamic stress strain code has been run with missing elements to assess the importance of buckling in various regions under HCDA loads. New EURDYN3 models of a section of the primary vessel have been run on the Risley VAX computer. Two models have been considered: one is a simple shape with constant thickness walls whilst the other has both variations in wall thickness and a change in curvature in the lower section. The second model is based on a new proposed NNC design. Initially, the deflections, stresses and strains for corresponding nodes in the two models were compared under conditions of a constant pressure of 0.5 MPa. EURDYN cases using the same models but under conditions of an HCDA pressure transient have also been run. Calculations for the primary vessel and core support system have shown no impact between the inner cylinder of the core support and the primary vessel for this calculation.

10.8 Debris studies and post-accident heat removal

Experiments on boiling dry-out in volume heated particle beds have continued at Winfrith. The previous upper limit of 5 mm on particle diameter has been extended to 12.7 mm, resulting in some interesting conclusions concerning the different ways of heating a bed.

Fibre-optic and infra-red radiation sensors have been fully tested in the Winfrith dielectrically heated particle beds and they work extremely well. The dielectrically heated bed experiments have been held up by difficulties in manufacturing 12.7 mm diameter 'Ferrite' spheres. These difficulties have now been overcome.

Some useful advances have been made in analysing experimental data. Two dimensionless groups have been identified as important and they have been shown to correlate data for different fluids from many sources. These dimensionless groups indicate that in beds of particles greater than 1 mm diameter, the dominant mechanism governing dry-out is the balance between vapour drag forces and gravitational forces. It also appears that an increase in surface tension lowers the dry-out heat flux. This effect is not represented in existing theoretical models of dry-out.

10.9 Structural response and missile impact studies

An initial series of tests at Winfrith of the perforation resistance of pre-stressed concrete targets to impacts by hard missiles, has indicated that pre-stress in itself has no measurable influence, but the tendons themselves contribute directly by acting as additional reinforcing steel.

Experiments with hemispherically nosed, hard missiles have shown that they require about 35% more kinetic energy to perforate a reinforced concrete target than flat-faced missiles of the same weight and diameter. The effects of missile cross-section on the perforation performance of reinforced concrete have been investigated using hard missiles with square, rectangular, triangular and semi-circular cross sections. Experiments are continuing, but initial indications are that missile perimeter is a major influence, and that equivalent behaviour would result from circular cross-section missiles having the same circumference as the non-circular missile perimeter.

Tests of the effects of heavy dropped load impacts on concrete floors are in progress. Initial results indicate that perforation energies at impact velocities down to 5 m/s are closely in agreement with predictions based on high velocity missile behaviour.

Development of the DRASTIC finite element computer code has continued to the stage that good representations have been obtained of the behaviour of reinforced concrete slabs, with span: thickness ratios of 15:1 or greater, impacted by soft missiles.

The collaboration agreement with France (CEA-EdF-GDF) continues to provide mutual advantages for both parties, as does the information exchange agreement with Denmark and Sweden.

10.10 Secondary containment studies: sodium fire aerosols

The 9 m^3 sodium fire facility at Winfrith has been modified and re-commissioned following refurbishment of the aerosol laboratory. A safety case has been written covering the initial series of experiments to study gravitational agglomeration. These experiments are expected to start early in 1986.

A third series of tests has been carried out on the Buffalo Forge fume scrubber installed at CEGB, Berkeley. The tests were carried out on sodium pool fires with surface areas of up to 625 cm^2. Some of the fires were positioned in the burn chamber away from the scrubber intake to allow time for the particles to agglomerate. Two different fan speeds were used to determine the effect of this factor on scrubber efficiency. Some increase in mean particle size and in the overall scrubber efficiency was observed in these tests.

11. PLANT PERFORMANCE STUDIES

11.1 Neutronics data

11.1.1 Nuclear data libraries

The JEF1 neutron interaction cross-section library was distributed to NEA data bank participating countries in mid-1985 and the fission product yield and radioactive decay data libraries will be distributed shortly. In order to complete one benchmark test of JEF1, calculations have been made for a series of zero leakage pin lattices built in the PROTEUS facility at Würenlingen. These contained mixed oxide (PuO_2 + UO_2) pins together with either UO_2, iron or stainless steel pins; the calculations used a quarter-lethargy group library derived from JEF1 and the MURAL code. k_∞ was underestimated by up to 1%, except for the assembly with the highest proportion of iron, for which k_∞ was overestimated by 1.8%.

Some differences have been found between calculations made using this quarter-lethargy group library at the NEA data bank and at Winfrith for fast critical assemblies. A number of additional assemblies have been calculated to try to find the reason for the differences. It seems that differences in treatment of resonance shielding in ^{238}U could be the explanation, these being associated with the method of interpolation between the tabular shielding factors which are too widely spaced in this library (being given only at decade intervals) for accurate interpolation. A more accurate library is to be produced for the MURAL code and this will be used for the more accurate benchmark calculations planned for 1986/87.

11.1.2 Cross-section processing codes

A group at Saclay has developed a database system, THEMIS, for managing the NJOY suite of programs and the data generated by NJOY. It is proposed to adopt this version of the Los Alamos cross-section processing system NJOY for the European collaboration.

Studies in the UK have concentrated on methods for treating resolved and unresolved resonance regions and for deriving sub-group data to represent them in calculations.

Winfrith has made further contributions to the IAEA study comparing cross-section processing codes using SIGAR, and the code REMO which accepts resonance data in the Reich-Moore formalism. Studies comparing SIGAR with NJOY have highlighted serious errors in NJOY when processing resonance parameters with non-physical spin state values; several of the JEF files fall into this category.

11.1.3 *Fission product data*

The new fission yield libraries (the unadjusted set C4U and the set C4A of yields adjusted to fit physical constraints) have been used in decay heat calculations and the results compared with those using the older library C3I and the US library ENDF/B5.

Calculations of the sensitivity of decay heat to data uncertainites have been made using FISPIN-SENS and the latest data libraries. From these, estimates of the accuracy of decay heat predictions have been made. Similar studies have been made on the neutron and gamma emission rates from spent fuel.

In collaboration with Birmingham University, the evaluation of fission yields is being continued. A revised library has been produced, incorporating more recent measurements, and fission products for which experimental data are greatly discrepant have been identified for further detailed study.

An evaluation of data on tritium and other light nuclide yields from fission is in progress; it will almost certainly emphasize the need for further measurements (or at least the completion of the measurements on samples already irradiated at Dounreay) for ^{239}Pu fission.

11.1.4 *HELIOS — The Harwell electron linear accelerator*

Routine operation of the HELIOS linac has continued for most of the period on the fast neutron, condensed matter and low energy cells. The change of the condensed matter target from tantalum to uranium gave 70% increase in neutron output.

Good progress continues to be made on identifying the sources of discrepancies, improving the analyses and obtaining the necessary new data for ^{56}Fe and ^{238}U, the nuclei being considered by the two international Task Forces set up by the NEANDC (Nuclear Energy Agency Nuclear Data Committee). A paper describing the progress of the ^{238}U Task Force was presented at the International Conference on Nuclear Data for Basic and Applied Sciences at Santa Fe in May.

Experimental work on HELIOS has mainly been on Fe though a variety of other measurements have been made. High resolution transmission measurements have been made on natural iron samples 2 mm and 15 mm thick and analysis of these data to give resonance parameters is in progress. Measurements have also been made with a new capture detector of the capture cross-section of a pure sample of natural iron over the energy range 190 eV to 120 keV. These data have an excellent signal-to-background ratio and should enable good resonance parameters to be obtained when the performance of the detector has been verified.

11.2 Experimental neutronics

11.2.1 *ZEBRA*

The zero power reactor ZEBRA has remained shut down. Maintenance has continued on preventing deterioration of the plant and the fuel inventory is checked at monthly intervals by EURATOM inspectors. To compare the relative merits of ZEBRA and the corresponding French system, MASURCA, for future fast reactor experiments, work is in progress to establish the cost and time required to recommission ZEBRA.

11.2.2 *Pin-plate reactivity discrepancies*

There are still no definite indications of the cause of the pin-plate reactivity discrepancy between calculation and experiment. The majority of participants in the international comparison of calculations show overpredictions of 0.003 to 0.006 of the k-value for the pin core relative to the plate core using their best available methods.

11.2.3 *MASURCA reaction rate comparison*

Provisional results of the reaction-rate measurements carried out by seven teams (UK, USA, Belgium, Switzerland, Italy and two teams from France) in MASURCA were compared at a meeting at Cadarache in July 1985. The results showed a relatively wide dispersion which was inconsistent with the claimed accuracies, but similar to that obtained in previous comparisons. Further investigation has eliminated a number of erroneous results.

11.2.4 *Reactivity feedback in the PFR*

A transient experiment was carried out to enable a separation of the core (fast acting) and out-of-core (slow acting) components of the reactor power coefficient. The test involved the rapid (200 ms) insertion of about 15 cents of reactivity with the reactor operating at 585 MW(th) and the analysis of the subsequent transient. The experiment was successfully completed with only minimal disturbance to the steam plant. Use of best estimate data gave a fast component of reactivity feedback about 20% high; a best fit was obtained by reducing the Doppler constant by about 20%. To maintain agreement between calculation and measurement at longer times after the transient, the long term feedback was increased (i.e. made more negative) to maintain the total power coefficient in agreement with static measurements.

11.2.5 *Reactivity effects of flow changes in the PFR*

When primary circuit flow is first raised following a reload, a small amount of reactivity is lost. Measurements are now carried out at the end of each reload and, in addition, where possible, at the beginning of each reload. At the end of Reload 10B (December 1985) a reactivity loss of 9 cents was observed, of which only 3 cents were recovered on reducing flows. This result is in line with measurements at earlier reloads.

At the end of Run 9 (June 1985) a reversible loss was observed of about 2 cents on increasing primary flow from 25% to 100%. At the start of Run 10 (July 1985) the reactivity loss in raising flow was 7 cents of which 2 cents were regained on reducing flow.

At the end of Reload 10B (December 1985) a reactivity loss of 9 cents was observed, of which only 3 cents were recovered on reducing flows. This result is in line with measurements at earlier reloads, when it was concluded that the combination of a several weeks shut down and vibrations due to sub-assembly rotations was sufficient to allow a substantial number of pins to fall from a levitated position. The maximum movement is about 30 mm.

11.3 Neutronics methods development

11.3.1 *COSMOS modular code scheme*

A number of improvements have been made to CODIP, the system which provides automatic management and guarding of data produced in calculations.

A computer program, the COSMOS Information Scheme, has been developed enabling easy access to relevant information and documentation concerning COSMOS. Approximately 650 reports and 200 COSMOS Modules exist and by working interactively with a subject key word system, required references to particular papers or codes can readily be obtained from the new program.

11.3.2 *European common code system*

Four code schemes, COSMOS (UK), KAPROS (Germany), IANUS (Interatom) and CCRR (France) have been evaluated in the past year by a representative European group. Preliminary reports from workshops have been prepared for each scheme and a final paper comparing the systems and specifying the requirements for a common European scheme is in preparation.

11.3.3 *Nodal methods for solving the neutron diffusion and transport theory equations*

Nodal methods are more accurate than the usual finite difference methods. Consequently, accurate solutions can often be obtained using a coarser mesh than is required with a finite difference program and the computing cost can then be significantly lower.

The three-dimensional hexagonal geometry nodal diffusion theory code HEXNEC produces the average flux in each node (i.e. coarse mesh volume) together with average incoming and outgoing currents at the node surfaces. A method has been developed for deriving from this information, the flux shape within any node, thus providing a more detailed solution where this is required.

Work has continued on the development of a nodal transport theory method. The method has been implemented in rectangular and hexagonal geometry. In reactengular geometry, the solution is only two to three times slower than nodal diffusion theory. Further work is required to examine possible improvements to the accuracy of the methods. For the hexagonal geometry formulation, better reference solutions are needed to assess its accuracy.

11.3.4 *Fuel management codes*

A computer code, MISSIONARY, has been written to produce data libraries for CRACKLE from the COSMOS system. Modifications made to the CRACKLE code now permit simulation of differential control rod insertion.

11.4 Neutronics performance calculations

11.4.1 *Subcritical monitoring*

Anomalies in predictions of sub-critical monitor readings with blanket sub-assemblies in the PFR core were largely resolved by incorporating reactivity adjustments derived at Winfrith into the calculations to compensate for a sodium-filled channel. These adjustments use improved methods for calculating effective diffusion coefficients for low density channels, together with an allowance for the effects of the presence of such channels on the axial buckling.

11.4.2 *Value to designers of further physics R&D*

As part of a European study, an assessment was made of the economic benefits to CDFR of reduced uncertainty in prediction of fuel enrichment, power distribution, control rod worth, decay heat and shield performance. Easily the largest benefit is associated with improvements in the accuracy of decay heat because of improved availability consequent on shorter cooling time.

11.5 Energy deposition and shielding

11.5.1 *PFR neutron shielding studies*

The comparison of measured reaction rates throughout the primary sodium pool of PFR with values calculated using the RZ removal diffusion calculations performed with the SCORMA code, has been extended to a number of locations in the inner and outer pools. Within the inner sodium pool, agreement between calculation and measurement for the (n,γ) reaction rate profiles is good. The exceptions are in the upper axial blanket and near the sodium-argon interface. The agreement in the axial profile for ^{58}Ni (n,p) is good but the absolute values calculated are lower than those measured.

11.5.2 *Updating of computational techniques*

A review of the status of UK calculational methods has been undertaken. The first task was to develop the new code SNAPSH to replace the obsolete 2-D SCOREM code which was still being used for CDFR shield design. SNAPSH is a special shielding version of the 3-D reactor physics code SNAP which provides the Hex-Z and Rθ geometries required for a proper treatment of pool reactor designs.

The modifications to SNAPSH include:

(i) New convergence criteria for fixed source attenuation problems.

(ii) A library of adjusted diffusion coefficients for ADC calculations in steel, sodium and graphite systems.

(iii) Coupled neutron-gamma data in multigroup form.

(iv) Post-flux edits to generate important information from adjoint ADC calculations.

(v) A new user guide.

In parallel with the work on SNAPSH, a study of the complementary Monte Carlo route has been undertaken using a simplified model of the CDFR radial shield and IHX design. The conventional Monte Carlo code MCBEND, which is used for thermal reactor shield design, is inefficient for fast reactors (in common with all other Monte Carlo codes) because large numbers of collisions are required to moderate intermediate (keV) neutrons in sodium. The new code, FEDRAN, tracks particles directly in a finite-element (FE) mesh. The preliminary runs in the CDFR heat-exchanger problem have shown an increase in speed of about 30% compared with MCBEND but it is clear that the tracking efficiency is still limited. It is accordingly proposed to produce a new fast reactor set of diffusion coefficients specially adjusted for sodium-steel systems. This will improve the accuracy of both forward and adjoint calculations for use in shield design studies and Monte Carlo accelerations.

11.6 Reactor assessments

11.6.1 *Work with the FRESCO whole-station parametric study code*

The new Mk IV version of the FRESCO code has been brought into use and a wide range of parametric studies has been carried out on core and pin optimization, using a range of pin diameter, core length, axial blanket length, and core pressure drop for Nimonic PE16 clad designs within the UK reference sub-assembly across-flat dimensions. For all of these the levelized fuel cycle component of generating cost has been determined using the best available fuel fabrication and reprocessing costs.

11.6.2 *Calculation of damage dose rates from FRESCO data*

It has been suggested that maximum fuel clad damage dose rate could be a more important limiting parameter than maximum burn-up when comparing fast reactor designs. Earlier work has shown that damage dose rate per % burn-up is roughly proportional to the reciprocal of core fuel enrichment. As a result, designs with fatter pins and lower fuel mass ratings would tend to be restricted to a relatively lower burn-up than the reference 325-pin design if a damage dose rate limit were to be applied. Some preliminary calculations have been done to establish a calculational route for clad damage dose rate using flux distributions obtained from FRESCO de-bug print outs in conjunction with FD5 data for damage displacement cross-sections.

11.6.3 ENS paper on fast reactor assessments

A paper on advanced fast reactor fuel was written and published in the May 1985 issue of Nuclear Europe, the journal of the European Nuclear Society. This paper briefly described the methods used at Risley to assess fast reactor performance and economics using the FRESCO, MARS and NIMROD programs to calculate both single station and system costs. A sample selection of data extracted from current work was included to show the effects of varying core fuel pin diameter and maximum core fuel burn-up. These results indicate that the potential economic gains available from optimizing fuel pin diameter by varying the number of pins in a standard UK sub-assembly are relatively small compared with the gains which could be achieved by suitable increases in burn-up.

11.6.4 Lower capital costs and the need for fuel-efficient stations

Whilst the gains in availability with lower-rated designs are undoubtedly attractive on a single station basis, the lower ratings lead to larger plutonium inventories and this feature may well limit the penetration of fast reactors into an electrical generating system, and may also, depending on the relative costs of fast and thermal reactors, lead to higher system costs than a more highly rated design with slightly lower availability and marginally higher single station generating costs. In recent years the conventional wisdom has assumed low electricity demand growth into the far future and a relatively late introduction date for series ordered fast reactors with only a low penetration rate of fast reactors thereafter. Studies at Risley are showing that new reduced capital costs make earlier introduction of fast reactors economically attractive and a higher penetration rate of fast reactors into the system, within a defined nuclear sector, again becomes desirable on economic grounds. In these circumstances, even with low electricity growth, a re-examination of the plutonium logistics and system costs of a generating system with a higher fast reactor penetration of the nuclear sector than assumed in recent studies has been considered worthwhile.

11.7 Control and dynamics

11.7.1 PFR dynamic modelling

Mathematical models of the PFR plant have been widely used for many purposes. The Winfrith Dynamics and Control Group has devised procedures for quantitative validation of these models, known as the 'Distortion Method'. Following general interest in the method, a paper on an implementation based on a PFR trip is due to be given at the forthcoming BNES Conference on 'The Science and Technology of Fast Reactor Safety'.

11.7.2 Instrumentation cables

An investigation has continued into the causes and cure of microphony in the newly developed superscreened cables using the radiation-tolerant polymer, polyetheretherketone (PEEK). Attempts have been made to construct a PEEK core with a conductive coating to produce anti-microphonic PEEK cables. Clearly the mechanisms involved in the generation of charges are not yet fully understood. It has been established that the microphony of an initial batch of cable varies by an order of magnitude over its length.

STATUS OF FAST BREEDER REACTOR DEVELOPMENT IN THE UNITED STATES OF AMERICA
March 1986

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1. INTRODUCTION

The Federal government remains firmly committed to a viable nuclear industry as well as maintaining a significant presence in advanced reactor technology. At the end of 1985, the United States had 91 operating nuclear power plants. By the end of 1987, 24 more nuclear power plants are expected to become operational. This will result in over 20% of the nation's electricity being generated by nuclear means. The use of nuclear power has substantially increased energy security by greatly reducing the dependence on imported oil in the United States.

In the early 1970's, when the U.S. nuclear industry was thriving on high load growths, assumed economies of scale drove plant ratings from the intermediate range (400 - 800 megawatts electric (MWe)) to about 1000 MWe. The lower load growths projected today, combined with the utility financial situations, may indicate that plants with smaller ratings and lower capital cost are needed if the long term U.S. nuclear option is to be preserved. This thinking has led the Federal government to support studies for advanced reactors having smaller basic power levels which could be serial-constructed for larger power stations.