A REVIEW OF THE UK FAST REACTOR PROGRAMME, MARCH 1979

R.D. SMITH

1. BACKGROUND TO THE 1979 REVIEW

Previous levels of effort and expenditure, in real terms, on LMFBR Research and Development in the UK were maintained during 1979. There have been delays in producing an agreed national fast reactor plan for the introduction of CFRs* in general and CDFR* in particular, so that it is now unlikely that the construction of CDFR can start before about 1985. The public inquiry, or inquiries, into fast reactors has not yet been arranged. It is also likely that there will be an inquiry into the proposed use of PWRs in the UK power programme and it is unlikely that this and the Fast Reactor inquiry can be carried out concurrently.

The Prototype Fast Reactor (PFR) at Dounreay has had its best year of operation so far generating about 40 GW-days of thermal power in the year ending in March 1979, with operation at power levels up to the design level of 600 MWt. There have been three shutdowns, the first in the Spring of 1978 for refuelling and plant maintenance, improvement and repair, while the second and third were concerned primarily with refuelling. Overall progress with the project in the past year has been good with lead pin fuel advancing to a burn-up of 9% by heavy atoms. The steam generators have behaved well with no new leaks. Plant trips have resulted from a variety of causes dominantly originating from conventional plant. During the year a total of 1800 members of the general public have been escorted around the plant as part of a plan to improve the public awareness of LMFBR issues.

The design of the CDFR has been evolving slowly. There is now less enthusiasm for any of the heterogeneous core designs although the option to change from the reference homogeneous design is being kept. A comprehensive series of physics measurements on some heterogeneous designs is being completed now in ZEBRA in collaboration with our DEBENE partners (see Section 10).

Work has continued on the design study of the fuel transport flasks for both fresh and spent fuel sub-assemblies, particular attention being paid to the prevention of sabotage or theft of fissile material. At this stage it is only necessary to establish feasibility: more testing and development work will be required before manufacture of the actual flasks.

The main research and development work for fast reactors in the UK is undertaken by the AEA, but other organisations, particularly NPC, CEGB and the NII, have made important contributions, many of which are referred to in later sections of this report.

The CEGB has continued to pay attention to the way in which CFR design is developing, and has shown particular interest in topics such as plant reliability, life expectation, in-service inspectability, availability and constructability. This has generally underlined the earlier CEGB views that great care must be taken in the selection of the initial design parameters and plant features, and that the necessary depth of resource must be devoted to the timely solution of the various technical issues. The extension of basic materials testing to the testing of components or simulated components is an important step in the verification of high temperature design methods and the selection and confirmation of materials failure criteria. An issue of importance in relation to the conservatism of present design code proposals lies in the effect on fatigue endurance of varying hold times at elevated temperatures: conflicting evidence exists in this area at present. Importance is also attached to the proper understanding of the behaviour of very large austenitic steel weldments and the ways in which defect tolerance can be affected by welding and heat treatment practices.

NB – Abbreviations are defined at the end of this review.

The Nuclear Installations Inspectorate (NII) has continued its examination of the safety of the CDFR, maintaining close contact with the development programme for CFR. In particular, the Inspectorate has completed an interim review of the safety of the reactor and is working on those areas requiring more detailed study. A study group has completed its work on molten fuel/coolant interactions (MFCI) and will report shortly. Work being done under external contract covers aspects of the behaviour of structural materials in a reactor sodium environment, under sodium inspection of structure, the MFCI phenomena and sodium rewetting, thermal hydraulics of fuel sub-assemblies, and whole core accident explosions and their containment.

The UK continues to play an active part in international collaboration on fast reactors, particularly in the safety area, both on a multilateral and bilateral basis, and many organisations in the UK strongly support the view that collaboration should be further increased in the future.

2. PROTOTYPE FAST REACTOR

2.1 Summary of operating experience during 1978

The past year has included much operation at higher power levels up to the design figure of 600 MWt. The reactor has again operated reliably with an availability of 88% during the past year and 81% of the time since it was taken critical in 1974. The reactor has now generated in excess of 121,000 MWd of thermal power. Electrical output maximum has been 200 MWe with the efficiency limited for the reasons given in last year's report. Improvements are however planned in the near future with the addition of a second hydraulic air pump to improve turbine vacuum and by full use of the high pressure feed heating system. The gross electrical output has so far totalled 28,000 MWd.

Irradiation of the core has proceeded to the point where high fuel burn-up levels are being attained. Reference design lead pins operating at the enhanced rating of 670 W/cm have reached a peak burn-up of 9% of heavy atoms. The main driver charge has attained 5·5% burn-up. No fuel pin failures have occurred in the driver charge fuel. There have however been two failures of experimental fuel. These items are awaiting examination in the irradiated fuel cave. The peak displacement dose in the driver fuel has attained 50 dpa, a dose previously reached in DFR in only those pins that exceeded 12% burn-up. As irradiation proceeds the effects of subassembly dilation and bow and length increases due to neutron induced voidage swelling became more important. There have been no difficulties to date in the removal of subassemblies from the core. Some subassemblies in the core have been rotated through 180 degrees, using in-pile machinery designed for the purpose, to counteract bow effects.

The primary circuit of the reactor has performed well. There have however been three trips associated with primary sodium pumps, two caused by a malfunction in the seal oil system on one pump and the third from a spurious reading of the primary pump motor speed logic.

The health physics position following the monitoring of radiation and contamination levels on the reactor top, and indeed throughout the plant, continues to be excellent. Gaseous effluent arisings from plant operation continue to be trivial.

There have been no boiler leaks in the past year and the boilers and associated secondary circuits have operated reliably. Significant improvements have been made in the procedures for setting up the relief pressures and performing statutory relief tests on the main boiler safety valves.

The stability of the feed heat train has become an increasingly important issue as higher powers have been reached. Two of the four direct contact (DC) feedheaters remain blanked off on the bled steam side (Numbers 1 and 3). Heater No. 4 is fed solely with bled steam from the exhaust of the main boiler feed pump turbine. Following two Station trips initiated by heater instability in July it was decided to alter the heater configuration so that the main boiler feed turbine exhausted partly to the condenser and partly to DC heater No. 4. Since this change the stability has improved and in operation at power levels up to 600 MWt there have been no further trips from this source.

There were two shutdowns for refuelling and other work in 1978. The main shutdown for refuelling and the completion of plant maintenance and improvement work began at the end of February and extended until the end of June. Operation then proceeded at increasing power levels through into the second half of September when the failure of an experimental fuel pin prompted a refuelling shutdown during October/November. Operation resumed in December and, in early January, reactor thermal power output was increased to 600 MWt. Shortly thereafter a second fuel pin failure occurred in the same type of experimental fuel as the first. Operation of the Station continued at less than half power while the delayed neutron signals were studied and location was achieved. In February the plant was shut down to remove the experimental fuel for examination and towards the end of the month high power operation has been resumed.

2.2 Spring shutdown

The principal objectives of the main shutdown in the Spring of 1978 were to refuel the core to meet the requirements set by reactivity and thermal hydraulics considerations and to reblade the last row of blades on all four main turbine LP exhausts.

The main fuel charge machine carried out a total of 125 single movements without significant difficulty. It is notable that the machine had been kept stored outside the reactor in a flask containing an inert atmosphere for 10 months since the previous shutdown. The demountable subassembly (DMSA) cluster transfer machine also functioned without fault for 16 cluster transfers. Difficulties had been experienced with this machine at the previous shutdown and modifications had been made after the earlier latching problems.

The last row of blades of the LP exhausts were all replaced on site and only small adjustments were necessary after test run-ups to achieve a satisfactory balance. The turbine shaft catenary was also reset following earlier bearing problems. Earlier tests of the 50% steam dump system revealed a shortfall in feed/spray water. This was corrected by reducing the leak-off at the main boiler feed pump to increase the supply and by fitting new spray nozzles in the desuperheaters to increase the mixing efficiency and thereby to reduce demand. An examination of cooling water pump No. 1 revealed severe erosion and graphitisation of the cast iron structure, and a temporary repair was effected using steel and concrete. It was decided not to remove the other pump for inspection at this stage. A restriction in steam flow at No. 2 reheater bypass, which had been observed during the previous run and traced to the movement of the thermal sleeve at the bypass outlet, was removed by fitting a new and modified sleeve. The other two will be modified later.

All three secondary sodium circuits were dumped, not simultaneously, for work on bursting discs, gas valves, sodium isolation valves, hydrogen detection membranes and to clear a blockage in a sodium phase detection loop (Circuit 3). The basket of the secondary cold trap loop was changed.

2.3 Operation – June/October

The resumption of power operation was delayed by a seizure of the main boiler feed pump immediately following the first rotor balancing run of the main turbine. The spare cartridge was installed. Early in the run an ingress of seaweed into the Station auxiliary cooling system caused an interruption to power operation. In the latter part of July, operation was restricted to two circuits as a consequence of a small leakage of sodium through a seal weld of a bursting disc assembly associated with Evaporator 3, which had been renewed during the shutdown. While the leakage was small the faulty seal weld was repaired to minimise damage to pipework. Trips were caused by instabilities in the feed heating system and by a failure of the heater drains pump in the July-August period, the latter causing an outage of one week. The new, failed pin detection location loop was commissioned in September and used to identify the first fuel failure. The loop uses beta precipitation from fission product gases stripped from a sample of the coolant. The fact that a fuel pin failure had occurred was detected by both cover gas monitoring, using beta precipitators and gamma spectrometer, and the bulk sodium delayed neutron monitoring system.

2.4 Shutdown – October/November

The Station was shut down in mid-October to remove the experimental fuel subassembly containing the failed fuel pin. In addition other core changes were carried out including rotations to counteract the effect of neutron induced voidage bow. On the power plant, pilot valves associated with all evaporator and superheater safety valves were refurbished, relief pressure set, and a test lift of each main valve satisfactorily completed.

2.5 Operation – December/February

Throughout December, operation proceeded at increasing powers up to 600 MWt. There were a number of interruptions due to trips. Following the second fuel pin failure in early January operation proceeded through into February when the Station was shut down for the removal of the experimental fuel item concerned. This was completed and at the end of February the reactor was operating in the range 500-600 MWt while commissioning of the HP feed heaters proceeded.

2.6 Irradiated fuel caves

The irradiated fuel caves have carried out their essential functions in the receipt and storage of irradiated fuel and the radiography and profile measurement of subassemblies. Window leaks, both of zinc bromide from the window tank in to the shielding penetration and of air past the inner alpha glass seal and in to the cave nitrogen atmosphere have continued to be troublesome. But both commissioning work and PIE work are continuing on items already in the caves.

2.7 Power plant

Studies continue of systems to allow a set-back of the power to within the capacity of the 50% steam dump as a means of avoiding reactor trips following certain classes of trip initiated from the power plant. Plant tests have been made to support these studies. In addition, improvements which are aimed at increasing Station availability and efficiency are under consideration.

2.8 Reactor measurements

Monitoring of normal operation and execution of carefully controlled experiments have allowed significant progress to be made in understanding the modes of reactivity feedback in PFR.

A small (approximately 6%) increase in the magnitude of the isothermal temperature coefficient has been noted as the core has gone from a clean situation to a partially burnt up situation. This in line with the expectation of the effect of burn-up on the fuel-clad gaps.

Careful measurements of power coefficient over a wide range of power up to near full power have shown the power coefficient to be significantly higher than predicted at low power, and slightly lower than predicted as maximum continuous full-rating conditions are approached. The discrepancy at low power has been identified with that part of the feedback which does not depend on coolant flow (i.e. the excess of fuel temperature above coolant temperature).

Low power calibrations of the PFR absorber rods at the start and end of runs were completed. The checks covered both individual rod worths and worth profiles. Careful analysis of these profiles confirms that no significant loss or redistribution of control rod absorber material has occurred.

Recordings of temperature fluctuations have been made using special probes located in the above-core plenum; the results have been used to validate rig experiments at RNL and to provide direct operational evidence of the magnitude of temperature fluctuations observed and hence the maximum thermal power levels achievable without risk of fatigue damage to structural components.

An incident recording system has been set up to record the transients in the primary circuit on reactor trips which thermally shock primary circuit structures. Signals have been recorded from a variety of transients and analysis is proceeding.

Preliminary analyses of the coolant temperature responses to small power perturbations at low power (approximately 160 MWt) have provided estimates of the fuel-to-coolant heat transfer coefficients. The thermal resistance between fuel surface and the cladding would appear to dominate the rate at which heat may be removed from the fuel. The gas-gap conductances (approximately 3 kW $M^{-2} K^{-1}$) are lower than the expected values by roughly a factor of 2. This may explain the higher fuel temperatures and higher power coefficients of reactivity feedback at low power.

Five effective natural circulation tests have been carried out. All tests have been initiated from low powers (3-5 MWt) and have depended upon prompt power. The results show that natural circulation is capable of removing powers of up to 15 MWt from the core once the pumps have stopped. Tests have been proposed to cover high powers and to study the initial transient following pumping failures.

3. DESIGN STUDIES

3.1 Commercial Demonstration Fast Reactor (CDFR)

A heterogeneous core has been designed which uses the same absorber rod lattice as in the reference design core. The two designs are interchangeable, which has the important advantage of allowing engineering design to proceed without risk of major alteration while the relative merits of the two types of core are considered further.

The possibility of annual instead of bi-annual refuelling is being examined with a view to simplifying fuel handling operations. An initial appraisal indicates that provided an advanced design of oxide fuel is used there should be no unacceptable penalties from the additional plutonium requirements. Work has also been proceeding on the evaluation of different blanket sub-assembly designs. The interim conclusion is that within current engineering and materials limitations, the optimum design for in-core and first row blanket sub-assemblies may be one with 127 pins. This would entail a small cost penalty when compared with the 85 pin design irradiated in PFR.

An alternative to hot cell refuelling and also to the reference design of triple rotating shields is a design based on the PFR concept of a small diameter single rotating shield utilising a pantograph fuel transfer machine. This is being examined with the initial emphasis being placed on a design which is used only off-power. Consideration has recently been given to a similar design which has the capability of remaining in the reactor during on-power operation and which is based on a preliminary design produced for the PFR. In addition to the economic advantages of minimising refuelling times, useful information from PFR could be provided for the commercial stations. The only extrapolations from the PFR design are a longer reach (due to the larger core) and higher operating loads (due to a restrained core).

NPC Whetstone have examined the logistics of transport and storage of irradiated fuel between the reactor and the reprocessing plant. The principal conclusion is that it would not be necessary to transport fuel assemblies at a higher heat rating than 7 to 8 kW in order to achieve the suggested out-of-pile time of 9 months. Also this fuel could be loaded in adequately shielded and protected flasks and subsequently delivered to the reprocessing plant in a satisfactory and safe manner providing the fuel was transported in sodium filled canisters within the transport flasks.

Progress has been made on the design of core support structures which are more tolerant to damage by incorporating a high degree of redundancy. The problems of in-service inspection and possible repair are also being studied.

A design of diagrid has been prepared in conjunction with Vickers Engineering (Barrow) in which manufacturing practicability has been assessed. This design provides for redundancy which inhibits both structural collapse and unacceptable loss of cooling in the core in the event of failure. Means of monitoring the structure in service are still under consideration.

The engineering design of a vessel supported by straps has progressed and has been integrated with interacting features of the reactor design without significant problems. The major uncertainty relates to the mass transfer of sodium vapour into the containment vault which is lined with ceramic insulation. This has necessitated lining the insulation with a Technigaz membrane which will act as the sodium barrier during normal operation and also in the event of a primary vessel leak. It will not however be regarded as a major barrier in reactor accident safety considerations.

Development of a fluidic valve for the burst pin detection location system is proceeding satisfactorily at Sheffield University and the next stage will be to test a bank of inter-connected switches. Discussions on the use of laser cutting for the production of the fluidic valve have been held with a potential manufacturer. This process is promising and may provide a route to the successful development of the system.

3.2 Design studies for later, advanced LMFBR

Early in the year a reactor design study was initiated to assess the capital costs of a system which is not constrained by current technical uncertainties. Following this stage it should be possible to select advanced design features for incorporation into a revised design of the CDFR which has advantages of both improved design and lower capital costs.

Particular design changes that could result in significant cost reduction are:

(a) A reduction in the number of secondary circuits from 8 to 4; the adoption of once-through boilers and a reduction in the number of boiler units from 32 down to a possible 4.

(b) A reduction in the standard of primary containment to that of a minimum guard vessel which will withstand sodium leakage from the primary vessel and also a roof design which is based only on operational considerations.

(c) The adoption of a rectangular layout for the secondary containment building thereby simplifying station layout, particularly active handling equipment, with resultant cost reductions.

(d) The reduction of active handling equipment assuming that components will be sufficiently well developed to minimise the need for frequent replacements.

(e) Some changes of the overall station layout which will minimise the length of secondary sodium and steam pipe runs.

(f) The use of a single turbine.

No major difficulties have been identified with the design using 4 circuits. The technical advantages of a large number of small units are considered to be out-weighed by the economic advantage of a smaller number of large units in spite of the greater operational effect arising from failures in the latter.

Designs of once-through steam generators are currently being evaluated with interest focused on a straight tube unit with no tube welds under the sodium. This would require a manufacturing capacity for tubes 35 metres long. Thermal sleeves developed for the PFR replacement units would be used at both ends of the unit to avoid the use of heavy tube plates and tube-to-tube-plate attachments. 9% Cr ferritic steel would be used on both the sodium and the steam sides of the units.

A design of primary vessel employing a torispherical end and larger diameter IHXs has led to significant reduction in overall height of vessel compared with CDFR. This is being evaluated and any benefits are still to be determined. If both primary and containment vessels are manufactured in steel and are operated at reactor outer pool temperature, the containment vessel can possibly be used as a secondary core support structure.

Pantograph refuelling machines designed to remain in the reactor would minimise shutdown time. It is proposed to install 2 machines in order to provide fall back capacity in fuel handling operations. Designs of the above-core structure incorporating a slot to accommodate the pantograph machine are progressing. Inconel 718 is being considered as a structural material for those parts of the system which would be subjected to maximum thermal shock and thermal striping.

3.3 Validation of structural integrity

During the past year some problems of structural integrity which concern both safety related structures and protection of investment have acquired greater prominence. Existing structural codes do not include post-weld heat treatment of structures in austenitic steel, and do not recognise the possibility of crack propagation from small defects becoming unstable during service life. However, the use of methods available for calculating critical crack lengths in ductile materials indicate that such a possibility exists in welds in a state of residual stress. This affects the validation of the main safety related structures – the diagrid, the core support structure, and the primary tank. Three programmes of work have been initiated to determine and mitigate the influence of residual stress in weldments on such fast crack propagation. A basic programme seeks to elucidate the effect of artificially induced residual stress starting within plate material, which can be compared with controls. Larger scale tensile tests to destruction are being made on wide plates with defects introduced into typical fabricated welds, to investigate the onset of fast fracture. A further programme, involving the manufacturer responsible for fabrication development of the core support structure, with specialist support from the Welding Research Institute and from a stress analyst contractor, will start from calculation of residual stress arising from fabrication and of the unloading path from deformation in order to predict defect propagation. These are intended to lead to further development requirements and proposals to mitigate residual stress and its effects. Some work has begun on the use of reliability analysis in the structural context, in parallel with failure mode analysis for validation purposes, initially on core support structures.

Work has continued on the development of simplified methods of analysis to provide an alternative to the elastic route of ASME Code Case N-47. Reference stress concepts and recent work on cyclic loading are being used to develop simplified design procedures for structures subjected to arbitrary cyclic loading in which elastic/plastic/creep deformation can occur. The reference stress techniques are applicable provided the loading lies within a creep modified shakedown limit. The magnitude of this limit has been investigated for a Bree type situation in which the ratio of primary to secondary loading, temperature, creep law and cyclic time were varied. Accumulated strain and creep damage predicted by full inelastic analysis and by the elastic route of CCN-47 have been compared with results of the simplified methods. Proposals have emerged which are less conservative and of wider applicability than those of Tests 1-4 of CCN-47 whilst requiring calculations no more complicated than those currently applied below the creep range.

Analysis has also been undertaken using full inelastic methods to determine the effect of additional thermal transient loading to the quasi-steady loading discussed by Bree. The shakedown boundaries and material damage factors are modified and the work indicates the limitations of Test 3 of CCN-47.

A simplified method of inelastic analysis of piping systems has been developed to reduce the volume of computer effort needed to investigate cyclic loads with creep relaxation and UK findings for an IWGFR set of piping benchmark problems have been reported.

A study into the constitutive relations of Type 316 steel subjected to cyclic loading is continuing, results obtained so far indicating a significant interaction between short-term cyclic plastic deformation and subsequent creep deformation, with marked effect on primary creep. Hold time and strain range effects on cyclic stresses, and softening effects, are being investigated in relation to the prediction of ratchetting. New proposals for constitutive relations, based on available data, are being prepared for immediate use. These will take some account of the observed influence of cyclic plasticity on creep and provide for a more realistic balance of hardening and softening mechanisms. More refined equations are being developed to incorporate the results of longer term experiments in current materials programmes.

Work on multiaxial creep deformation and rupture is continuing in order to determine the deformation and isochronous rupture surfaces of 316 SS. Preliminary work promises to lead to satisfactory correlations. Creep fatigue tests are being performed on 316 SS over the temperature range 550-625°C with a variety of strain ranges and with tensile dwell times up to 16 hours. Future tests will include compression dwells and hold times of longer duration.

A postulate that a cylinder subjected to linear movement of a severe axial temperature ramp (conditions not covered by the methods of T1320 in CCN-47) would ratchet radially was tested experimentally, and found to be correct. At the same time a computer code, adopted for the purpose, was used to analyse this phenomenon and gave good agreement; the effects of different material models was also shown.

The absence of extended plant service experience for fast reactor materials and conditions, and the inability to conduct component tests, has required structural validation to rely heavily on small scale experimental data. Proposals are, therefore, being examined to complement the thermal fatigue work already in progress in the induction heated facility at AERE Harwell. Difficulties remain, however, of combining a high throughput of tests with achieving realistic conditions giving results which can be satisfactorily analysed. These more complex and larger scale tests, for closer approximation to operating conditions in larger geometries with longer hold times, would seem to be candidates for further international co-operation on experiments which tie up expensive equipment for long periods.

During the year the proposed ASME Section XI Div. 3 Rules for Inspection and Testing of Components of Liquid Metal Cooled Plant have been examined for comment. The incomplete state of these proposals and the limited design geometries covered seem less of a guide to requirements than an indication of the need for further work on methods of inspection, testing and monitoring, on the evaluation of results, and on design arrangements for application. This also would seem to be an area where further international co-operation would be beneficial both to clarify the state of the art and to form a consensus of view on requirements.

4. ENGINEERING DEVELOPMENTS

4.1 Progress with the High Temperature Sodium Loop

The High Temperature Sodium Loop (HTSL) was dried out and ancillary services connected into the system following testing of the mechanical pump in water. The rig dump tank was later charged with 17 tonne sodium and the programme of sodium cold trapping and dry commissioning completed. Sodium was transferred into the main loop in July 1978 and initial circulating tests carried out using the mechanical pump. The rig has operated for a total of 800 hours employing pump speeds between 50% and 100% of full speed and at various temperature levels up to a maximum of 600°C. No major difficulties have been encountered and the rig fully meets the specification.

The HTSL will be used to mount various dynamic experiments with a common requirement for flowing sodium at high temperatures including measurements of the effects of thermal shocks. It consists of a single pumped loop containing about 8·5 tonnes of sodium, with storage and clean-up facilities. A sub loop with an EM high temperature pump and electric induction heater will also allow the generation of thermal gradients within the limits 400-600°C at a maximum rate of 25 deg C s^{-1}.

4.2 Sodium Components Test Rig

The Sodium Components Test Rig (SCTR) dry commissioning tests have been completed and sufficient sodium pre-treated and charged into the rig dump tank to meet the immediate experimental programme. Following sodium clean-up by cold trapping it is intended to begin commissioning of the Halip pump test section using sodium (see below).

4.3 Halip pumps

Work is continuing on the development of Helical Annular Linear Induction EM Pump (HALIP). The type of pump, which can be used fully immersed in sodium circuits at temperatures up to 600°C, has the advantage that the windings can be withdrawn and replaced without disturbing the pump duct and connecting pipework. Following tests in sodium on 32 mm diameter core pumps, and a cavitation test on a 57 mm diameter pump, previously reported, all components for the first two sizes of large experimental pumps have been manufactured. These are Pump A (89 mm diameter x 1000 mm long core) and Pump B (133 mm diameter x 1000 mm long core). The core of Pump A is assembled ready for winding. A suitable winding machine has been constructed and a trial winding of an 89 mm core has been successfully concluded. Winding tests on the 133 mm size are about to start.

4.4 Coolant circuit hydraulics

4.4.1 Thermal shock

Out of pile apparatus has been developed to simulate, without sodium, the type of thermal shock to which in-pile components are subjected as a result of a reactor trip. Initial work has been directed to studying the effect of shock on a 6 inch diameter tubular specimen with an internal boss and an external flange welded to it.

The test specimen is placed in a furnace and heated slowly to 600°C when it is quickly removed from the furnace into an array of tubes which direct an air blast to the external surface. This induces a surface thermal shock from 600°C to 400°C at a rate of up to 10 deg C s^{-1} (which is sufficient to simulate in-pile conditions). The rig is automated so that fast cycling without hold time or with any desired hold time at the upper temperature can be achieved.

4.4.2 Thermal striping

Flow and power variations between individual sub-assemblies, particularly at the core/blanket boundary, cause significant temperature differences between the streams of sodium emerging from adjacent sub-assemblies. These streams of sodium at different temperatures retain their thermal identity for a substantial distance downstream and, since the flow is unsteady, any structure immersed in this flow is subjected to fluctuating sodium temperatures. This phenomenon, known as thermal striping, causes the surface temperature of the structure to fluctuate, leading in turn to cyclic thermal straining, and possible surface crack formation and propagation.

The extent of these thermal fluctuations in PFR is being investigated both by in-pile instrumentation and by out-of-pile air and water models. The main model is one in which the complete above-core region is represented. In this model individual electrical heaters mounted in selected simulated core and blanket sub-assemblies permit the individual temperature levels in these assemblies to be set to the values which simulate detailed core outlet temperature patterns. The resultant flow velocities and temperature fluctuations are measured in the above-core region.

4.5 Steam generators

4.5.1 Thermal hydraulics

The thermal hydraulic work at AEEW has included completion of the analysis of results from the 10 cm radius bend U-tube tests on the High Pressure Rig and the twin U-tube stability test section on the 6 MW rig, together with comparison with model predictions. The results for the 10 cm radius bend U-tube led to further development of the analytically based bend dry out model which now successfully predicts results for 45 cm and 10 cm radius bends. The large amount of post-dryout data obtained from various U-tube test sections has been critically analysed and compared with correlations and an optimum correlation has been derived. Extensive data have also been obtained on temperature fluctuations at dryout. These are being analysed and models are being developed which should allow prediction of temperature oscillation behaviour and the subsequent stress cycling of the tube wall.

The first two stages of the test programme on the Freon rig using electrical and hot water heating to investigate the effect of the nature of the heating on dryout, post-dryout heat transfer and temperature fluctuations at dryout have been completed, and the analysis of the results is well advanced. It is likely that the third stage tests (electrical heating with a flux profile corresponding to water heating) will be required.

4.6 Instrumentation

4.6.1 Sub-assembly instrumentation

Work has been concluded on the development of a 1 mm diameter double-walled coaxial thermocouple, in which the Chromel and Alumel conductors are butt-welded together, sheathed, swaged down and then bent into a hairpin form with the junction at the tip of the hairpin. This type of construction gives a relatively robust thermocouple with a response time of $\sim$ 50 ms.

A commercially produced 60 sample batch of these thermocouples has survived over 5×10^5 400 to 600 to 400°C fast temperature cycles without failure. This type of thermocouple is now established as a fast response, highly reliable, temperature sensor.

Prototypes of a complete sub-assembly instrumentation thermocouple cluster have also been produced. The hot junctions of 6 coaxial thermocouples are enclosed in a 30 mm long x 8 mm diameter shroud tube with the 12 cable legs laid up helically around a 3 mm core cable to produce a robust, flexible multistrand assembly 5 mm diameter and up to 23 m long.

A test section for performance measurements on prototype sub-assembly Instrument Packages has been constructed and is presently being commissioned on the Mechanical Pump Test Rig in RNL. This simulates in sodium the outlet conditions of a sub-assembly for temperature transients and flow transients, either separately or together. The performance of the Instrument Package can be measured under normal conditions and also when it is displaced vertically or horizontally from its normal position. Two instrument packages, each having two flux distortion flowmeters and two coaxial thermocouples have been installed for the commissioning phase of the experiment. Recording and measuring channels have been constructed and commissioned which handle the signals for this commissioning package and for any subsequent above-core instrument package.

The flow and/or temperature transients in the test rig are generated by injecting sodium into the test section using a dc electromagnetic pump energised by current from a series of heavy-duty batteries switched in turn by solenoid switches in a controlled sequence by a digital timer. Commissioning experiments have shown that the form of the flow and temperature transients are satisfactory. The generated outlet flow transient is a uniform acceleration of up to 4 ms^{-2}, prevailing for 0·3 seconds at maximum accelerations and longer at lower values. The linear temperature ramp at constant flow is 60 deg C s^{-1} prevailing for 0·4 seconds, followed by a plateau which can be held for several seconds. This temperature transient can, if desired, be generated together with a flow transient, or a flow transient can be combined with a much larger temperature transient of less well defined form. Some initial measurements of temperature noise and flow response have been made using the commissioning instrument package at 50% flow.

5. MATERIALS DEVELOPMENT

5.1 Introduction

Studies of explosive welding as a method of fabrication have continued. In the field of inspection, a start has been made on the important but difficult task of developing a technique for the volumetric inspection of austenitic weld metal. The mechanical properties programme has been widened to include work on 321 austenitic and 2¼Cr ferritic steels, with some associated changes in emphasis, including a greater attention to the properties of weld metals, and to work connected with thermal striping/thermal shock problems. In waterside corrosion studies, considerable progress has been made in understanding the behaviour of 2¼Cr steel under caustic cracking conditions and in testing palliative treatments. Tribological work has been largely concerned with the problems arising in boiler tube support systems and promising results have been obtained with aluminised surfaces. Development of absorber materials is continuing, and the first PFR clusters and control rods are now available for examination.

5.2 Fabrication and inspection

Studies have continued on the explosive welding of 2¼Cr 1Mo and 9Cr 1Mo tubes into 2¼Cr 1Mo tubeplates. Welds 12-14 mm long can now be consistently produced in tubes with a 2·5 mm wall thickness and can withstand repeated pressure cycling and thermal shocks, typical of the most onerous service conditions, without leaking. The explosive welding procedures are considered suitable for use in the fabrication of steam generators.

Specimens have been produced by "narrow gap" welding in 50 mm thick 316 stainless steel plate in the horizontal position. The technique consists of applying TIG welding using argon shielding specially designed to provide a parallel-sided weld only 6 mm wide, which it is hoped will be easier to inspect ultrasonically than a conventional weldment. The grain orientation is not particularly favourable for minimum attenuation of ultrasound but this is offset by the considerably reduced volume of weld metal to be penetrated. Its ultrasonic properties are being compared with those of conventional manual metal-arc welds and the initial judgment is favourable. The technique is novel and was developed to produce higher quality welds than those made by submerged arc methods in ferritic alloy steel pipework.

RNL are undertaking a detailed investigation of the use of angled longitudinal ultrasonic waves for inspecting typical CDFR weldments. Initial studies employed 50 mm thick austenitic butt welds containing side drilled holes, while work on deliberately defected welds is now in progress. It has been shown that the angle of penetration of an ultrasonic beam into weld metal is effectively fixed by the dendrite orientation and not by the angle of incidence of the beam. However, the effective range remains approximately correct and defects may be located by a triangulation method. AERE are investigating the use of the ultrasonic diffraction technique. A theoretical approach has been developed which appears capable of explaining most of the observed ultrasonic properties of the weld in terms of grain alignment. These complementary studies provide an encouraging start to solving the important but intractable problem of supplying a method for the volumetric inspection of austenitic weld metal.

5.3 Mechanical and physical properties of circuit materials

5.3.1 General

Until recently the fast reactor programme was centred on Type 316 steel, and to a lesser extent, on 9Cr steel. The number of materials has now been increased to include Type 321 steel for PFR applications and 2¼Cr steel for CDFR and PFR purposes. Since the weld metal properties are important in relation to the integrity of structures, increased emphasis has been given to the testing of Type 316 weld metal, and the 2¼Cr programme also includes tests on weld metal.

In general, progress relating to air tests has been satisfactory during the last year, but environmental studies involving sodium or irradiation have been slow, partially because of the complexity of the work. However, a slightly reduced emphasis is now being given to the sodium work since, at an IWGFR Specialist Meeting held in late 1977, it was concluded that the effect of high purity sodium was likely to be small, based on short-term test results.

Currently it is considered that the three most important areas of work in relation to the satisfactory design of CDFR are strain controlled high cycle fatigue (for thermal striping evaluations), combined creep/fatigue (for thermal shock assessments) and the role of residual stress on the fracture behaviour (for defect size analyses).

5.3.2 Type 316 steel

Creep and rupture tests at 550, 575 and 625°C are continuing at RNL; test data extend to 44,000 hours on the wrought material and to 27,000 hours on weld metal. An evaluation of these and other data suggests that the previously agreed rupture strengths at temperatures around 550°C to times of up to about 20,000 hours are too high; lower values will be included in the data sheets. An analysis of the rupture ductility of wrought material by CEGB and others indicates that values should exceed 10% and that low values are generally associated with low grain boundary boron content, high carbon content and large grain size. The variability of microstructure and consequential variations in the mechanical properties of small test specimens are more significant with weld metal than with wrought material. Work is continuing in an attempt to reduce scatter in creep rupture values by testing relatively thick specimens where the thickness spans several weld beads. Detailed structural examinations are being carried out to identify phase transformations in various commercial and experimental welds subjected to stress relief heat treatments and/or thermal ageing. The rupture ductility of the weld metal is $< 5\%$ at long times. There is some indication that the rupture ductility might be improved by heat treatment; a programme on both wrought and weld metal is to be mounted to evaluate the properties after heat treatment (for example 10 hours at 800°C).

Strain controlled creep/fatigue tests on Type 316 steel in air have been carried out at SNL and RNL. Most work has been performed at 625°C but recently tests have been mounted at 550° and 570°C. Current indications are that the endurance at the lower temperatures may be less than at 625°C. Strain controlled creep/fatigue tests performed at RNL on Type 316 weld metal at 625°C have given endurances comparable to those observed on the wrought material. The rates of crack growth have been measured in load controlled creep/fatigue tests performed at RNL at 625°C on Type 316 steel in the as-received, 20% cold worked, and 10,000 hours aged conditions. Differences observed in the growth rates and endurances in creep/fatigue tests have been explained by SNL and RNL workers by applying a crack growth model to known differences in the microstructure.

Work at CEGB has shown that the introduction of creep interactions in high strain fatigue tests at 625°C, by superimposing ½ hour tension holds, increased crack growth rates. However, pre-ageing for 4,000 hours at 700°C reduced this effect. Further tests will assess other pre-ageing treatments, including service exposures.

Tests of crack growth under high cycle fatigue conditions have shown that large grain sizes and low yield strengths lead to higher values of the threshold stress intensity for crack growth and reduced crack growth rates. Pre-ageing at 650°C reverses these trends.

Thermal shock tests have been carried out by repeated water quenching of the inner surfaces of tubular specimens of 316 steel to identify end load effects.

Recent defect size assessments of stainless steel components have highlighted the fact that existing codes, which specify the use of initiation toughness levels and necessitate that residual welding stresses be regarded as equivalent to mechanical loads, can give rise to small tolerable defects in welds which are not stress-relieved. In view of the possible conservatism associated with these two assumptions and hence the derived defect acceptance levels, work is currently in progress in an attempt both to determine the influence of residual stresses on fracture of stainless steel and to identify the conditions under which ductile crack growth becomes unstable.

Measurements of base line residual stress values for manual metal arc austenitic weldments have been completed and the work is being extended to include heavy section pipe butt welds and a full-scale pressurised vessel. Residual stresses have also been measured in heavy section 2Cr Mo to AISI 316 transition joints as a function of time and temperature. Creep and thermal cycling experiments on such joints are continuing.

A variety of strain controlled fatigue endurance tests have been performed at RNL, mainly at 550°C, in both air and static sodium to endurances of about 10^6 cycles to evaluate the conditions for the avoidance of thermal striping damage. Longer term and lower ΔT tests are now planned at RNL and SNL to define high cycle fatigue performance.

Creep/fatigue tests have been performed in dynamic clean sodium on Type 316 steel at 625°C. It is found that with increasing hold times the endurance in sodium approached that found under similar conditions in air. CEGB tests presently underway in static or low-flow rate sodium are concentrating on exposure to contaminated sodium and involve both fatigue and creep testing of Type 316 steel. It is planned to include pre-aged material in these tests.

5.3.3 *Ferritic steels*

Mechanical property tests on 9Cr 1Mo in extreme conditions of heat treatment, overageing and cold work have continued to demonstrate that the material is extremely tolerant of treatment that often markedly reduces ductility in other ferritic steels.

High strain fatigue tests on 9Cr 1Mo at 550°C have shown no effect of dwell time on crack propagation rates. This is in contrast to the behaviour of Type 316 steel and is associated with the resistance of 9Cr 1Mo to grain boundary cavitation.

Stress relaxation behaviour of normalised and tempered 9Cr 1Mo has been investigated and the work is being extended to simulate creep ratchetting.

Preliminary creep rupture results on 9Cr 2Mo and 9Cr 2Mo (VNB) (EM12) demonstrate that the steels offer considerable strength advantages over 9Cr 1Mo. As in the 9Cr 1Mo test programme, behaviour of normalised and tempered materials will be compared with behaviour after adverse heat treatments.

5.4 Waterside corrosion

5.4.1 *General*

CEGB experience, arising from the AGR programme, of the waterside and steamside behaviour of ferritic materials, particularly 9Cr 1Mo, has been fed into the fast reactor development programme. Its waterside performance has proved encouraging because the material exhibits generally low corrosion rates with good tolerance of salts entering the boiler, and good resistance to stress corrosion cracking. The latter is critically dependent upon heat treatment, particularly after welding, and further work to set limits precisely is in process. In steam oxidation the dominant influence of system-strain on oxide integrity has been demonstrated and an explanation has been suggested.

5.4.2 *Stress corrosion of boiler steels*

2¼Cr 1Mo U-bend specimens (with and without a hard weld) and hard tube/tube plate welds have been exposed to 10% caustic soda at 200°C after a range of heat-treatments designed to indicate the minimum for stress relief adequate to prevent cracking. Separate experiments were initiated to determine the relationship between stress and time to failure for this material when exposed to the same refluxing medium. Tests also continued in this environment to establish the effect of potential on the crack susceptibility of 2¼Cr Mo in various metallurgical conditions.

The stress corrosion behaviour of hard 2¼Cr 1Mo steel in good quality water at 300°C has also been examined. For cracking to occur the material must be subjected to a high fraction of the yield stress. The work also indicates that the microstructure in respect of sulphide inclusions has an important part to play in promoting cracking.

At AERE the programme has concentrated on a comparative study of the stress corrosion cracking of 9Cr 1Mo, 2¼Cr 1Mo (Nb stabilised and unstabilised) and carbon steels, in the presence of concentrated solutions of sodium hydroxide. Limiting caustic concentrations exist above which the alloy steels do not crack. These critical values depend on the stressing conditions, but are likely to lie in the region of 35% NaOH at 120°C for 9Cr 1Mo steel and 60% NaOH at 160°C for 2¼Cr 1Mo 1Nb steel. The general corrosion resistance of the 9Cr 1Mo steel was usually good in boiling solutions up to 70% NaOH, although extensive attack could result from cathodic polarisation from the presence of oxygen or chloride ions.

5.4.3 *Corrosion in the absence of heat flux*

Isothermal exposure of 9Cr Mo and other steels to 1,000 p.s.i. (6·9 MPa) steam at 475 and 550°C has now exceeded 20,000 hours. Some difficulty is being experienced in descaling the specimens now that the oxide is thicker. The net weight gains for both 2¼ and 9Cr Mo at 550°C are in line with earlier results, i.e. following the departure from parabolic behaviour at about 8,000 hours, they continue to display linear kinetics. At 475°C the behaviour remains parabolic, with gains still less than 5 mg/cm².

5.4.4 *Corrosion in the presence of heat flux*

Experimental work in the RNL Model Boiler Test Facility indicates that under adverse hydrodynamic conditions steam blanketing occurs in the bend region of a pendant boiler tube, and dependent on the water chemistry can result in the generation of a corrosive environment from salt hideout. Operation in neutral water containing 3 ppb dissolved oxygen at boiler inlet has previously been shown to result in severe chloride pitting attack on 2¼Cr 1Mo (Nb) tubing in the bend region. The possibility of combating steam blanket corrosion resulting from salt hideout by operation in the ammonia-hydrazine water chemistry regime is under investigation.

The behaviour of 9Cr 1Mo tubing under Fast Reactor steam generator fault conditions is being studied in the Harwell Mild Steel Loop at high heat flux (660 kW/m²) using test sections which contain spark-eroded defects. Under alkali fault conditions at high heat flux, the protective magnetite layer on 9Cr 1Mo tubing behaves as expected from the magnetite solubility-temperature relationship. The present test simulates an acid sulphate fault with 2 ppm sodium bisulphate in water at 350°C, and has completed 1,000 hours of the scheduled period of 4,000 hours.

Studies of salt behaviour in once-through boilers have continued using the Miniature Boiler Corrosion Loop which has been operated with a 9Cr 1Mo test section. The deposition of soluble and insoluble iron in the feed has been studied using the radiotracer ^{59}Fe, different profiles being observed depending on whether the iron is introduced as $Fe(OH)_2$ or $FeCl_2/FeCl_3$. The behaviour of some sodium salts has been followed using the radiotracers ^{24}Na, ^{38}Cl and ^{82}Br. The equilibrium amount of sodium deposited, m, was found to depend on the concentration in the feed, c, according to the relationship $m = kc^n$, with $n = 0{\cdot}5$ for NaOH, Na_3PO_4, and Na_2HPO_4, $n = 1$ for NaCl and NaH_2PO_4, and $n = 2$ for $NaHSO_4$ and Na_2SO_4.

5.5 Tribology

Fretting studies have continued, mainly in relation to steam generator materials. Work on the rubbing and fretting behaviour of 9Cr 1Mo/aluminised counterfaces in sodium continues to demonstrate the benefits of this surface treatment for achieving low wear rates and low coefficients of friction.

Work is also being performed to assess more closely the tribological behaviour of the materials combinations of the existing PFR units. Rubbing tests in flat-on-flat geometry generally confirm earlier tests in crossed cylinders geometry and suggest that the specific wear rates for 316 stainless steel are lower then for stabilised or unstabilised 2¼Cr 1Mo steels under similar conditions. These in turn are slightly less than for 9Cr 1Mo. In simulations of the thermal sliding of evaporator tubing in 2¼Cr 1Mo bushes, local displacements and irregular transfer can results in interference to sliding. Such effects have not been found with 316/316 combinations with similar tube/bush clearances, but short dwell periods under load at 560°C have been found to give adhesion coefficients of up to 2·7.

Work is being performed on the dependence of the tribological behaviour of chromium-containing alloys, such as stainless steel and Stellite, on the oxygen level in sodium. A sharp transition in the coefficient of friction which is assumed to be due to the temperature-dependent formation/disappearance of sodium chromite has been demonstrated for Stellite/Stellite pairs in the temperature range 650°-500°C.

5.6 Absorber materials

The first six PFR clusters containing boron carbide and tantalum specimens were unloaded from the reactor at reload 1, following about 136 equivalent full power days (efpd) exposure. Three new clusters containing samples of boron carbide and monoclinic europia, in the form of model pins and capsules are being loaded at reload 3. Three boron carbide control rods have been withdrawn from the reactor for PIE after 65, 136 and 196 efpd respectively.

Examination of monoclinic and stabilised cubic europia pellets irradiated in PFR to a dose of 3×10^{22} n/cm² (total) at 650°C has demonstrated that the former material grows approximately linearly with dose at a rate of $\sim 0 \cdot 5\%/10^{22}$ n/cm² (total), whereas the latter is dimensionally stable or shrinks slightly. Irradiation did not affect the conductivity of samples exposed to doses of $\sim 1 \times 10^{22}$ n/cm² (total).

A computer model, known as BORCON, has been developed to calculate the temperatures, stresses and strains in operating boron carbide control rods or static irradiation capsules. Comparison of the model predictions with 479 series of DFR irradiations shows good agreement with burn-up at which cladding strain begins and the cladding strain rate after contact of pellet and cladding.

6. CHEMICAL ENGINEERING/SODIUM TECHNOLOGY

6.1 Corrosion, mass and activity transfer

6.1.1 Corrosion studies

Compatibility of aluminised Inconel 718 with sodium has been investigated as a function of temperature, sodium velocity and oxygen level. Tests at RNL have shown that its corrosion behaviour is sensitive to changes in all three parameters but the velocity effect disappears at low oxygen levels. Detailed metallography of specimens is in progress.

At AERE, brazed thermal sleeves of 9Cr 1Mo steel have been tested in 350°C and 550°C sodium hydroxide saturated with sodium oxide and sodium under equilibrium hydrogen pressures. No enhanced or localised attack of the braze or adjacent steel was detected even in runs where the reaction mixture was exposed until dry. The extent of corrosion was limited by rising concentrations of corrosion products. In supporting experiments using rotating discs the rate of corrosion could be substantially reduced by additions of iron corrosion products; additions of sodium chromite had no influence. Measurements showed that the solubilities of sodium ferrate and chromite in hydroxide melts increase greatly when oxide is present. The larger effect on chromite solubility is in line with results of CEGB studies which showed that iron corrodes more in pure hydroxide melts but that the addition of oxide increases the corrosion rate of chromium relative to that of iron.

6.1.2 Mass transfer studies

The Large Mass Transfer Loop at RNL has completed an 18-month programme of operation under representative PFR conditions. Specimens removed at different times from various parts of the loop have now been examined. The "core" test section has consistently shown a cross-over between bulk deposition at the upstream low temperature end and corrosion release at the downstream high temperature end. At the peak temperature corrosion release has been consistent with predictions based on other RNL Loop work. Surfaces here exhibit structures consisting of sub-grains of α-iron and M_6C carbides typical of those found in non-heat flux corroding conditions: alloying elements (principally nickel) have been preferentially removed from substrates. Elsewhere corrosion rates have been lower than predicted and for most of the period activation energies for the corrosion process have been about 130 kJ rather than 75 kJ as recommended in current RNL correlations. These surfaces are characteristic of oxide which – depending on temperature – is in various stages of removal. There is evidence that the position of the cross-over point may be moving slowly upstream with time, tending to restore the corrosion rate and activation energy to their expected values. In deposition zones, iron-rich particulate material continues to predominate over Mn-Ni deposits found initially. Towards the end of the period, deposited material started to break away from the upstream, hot, ends of these test sections and to redistribute towards cooler parts. Pure nickel specimens in deposition zones showed weight gains many times greater than those found on adjacent stainless steel specimens. Manganese predominated at the surface of these specimens whereas very little was found on steel specimens.

To investigate whether silicon is removed from steels in sodium a number of ferritic and austenitic steel samples were examined after exposure in the Large Mass Transfer Loop. Some depletion of silicon in surface layers (20-30 μm) was found, reducing the surface concentration to about half its as-received value. Silicon loss occurs in the initial stages of exposure and no time dependence is found over periods of up to 18 months. Analyses of corrosion products removed from this and other loops do not show the pattern of silicon deposition found by US workers.

The influence of microstructure upon the decarburisation behaviour of 2¼Cr 1Mo ferritic steel in sodium has been assessed at BNL. Changes in heat treatment and of compositional specification with respect to minor elements (silicon and nitrogen) can improve the resistance observed by various workers in the decarburisation of 2¼Cr 1Mo steel specimens. It has also been shown that decarburisation kinetics may obey a linear rate law, after an initial period, rather than a parabolic law as commonly assumed. The effect on ageing reactions of variations in carbon and boron levels in Type 316 steel continue to be investigated, with particular attention being given to the influence of carbon levels on crack initiation. The effect of carbon level in sodium is also being investigated.

BNL have developed a computer simulation of carbon transfer in austenitic steel circuits. The effects of variations in carbon diffusivity, solubility and carbon activity/concentration relationships have been examined, enabling their relative importance to be assessed. Values most consistent with available experimental information have been identified.

6.1.3 Corrosion and fission product activity transfer

Follow-up and decontamination studies continue using specimens cut out of the Harwell Active Mass Transfer Loop. Enhanced ^{54}Mn release and slightly sub-stoichiometric ^{60}Co release from the activated steel source specimen has been confirmed. The bulk deposition rate and particle size in deposits on IHX tube specimens were substantially greater at the cold end than at the hot end, with cold end deposits enriched in nickel and hot end deposits depleted in chromium relative to stainless steel. ^{54}Mn activity, although preferentially deposited at the cold end penetrated less far into the substrate here than at the hot end. The profile of ^{60}Co deposition has been shown to be consistent with boundary layer mass transfer control.

At BNL a study of caesium partition between sodium and steel is in progress using ^{134}Cs and ^{137}Cs tracers in a small pumped non-isothermal sodium loop. Reversible adsorption of Cs on Type 316 stainless steel has been confirmed. Results from a low temperature run yielded in-situ partition data for fresh steel surfaces in the middle of the wide range of published values. Deactivation of steel was found after pre-exposure to high-temperature sodium. Gamma scans indicate that most caesium is associated with the sodium surface film; very little remained on steel after draining sodium. Contrary to experience in BOR-60 no adsorption of caesium on sodium oxide in the cold trap was detected: cold trapping and partition measurement are continuing.

Modelling and assessment of activity distribution around the primary circuit continues at BNL and at Risley. Computer codes for activity source term calculations are under development: these make use of data generated by developed thermal hydraulics and physics codes. A comprehensive model developed at BNL accounts for the observed enhanced release of ^{54}Mn and substoichiometric release of ^{60}Co from corroding surfaces, and for their different deposition patterns in experimental facilities. Quantitative application to reactor circuits requires knowledge of partition coefficients and solubilities which is at present incomplete. By utilising simplified models it has been estimated that deposition of induced activity in a PFR pump could generate a dose rate of $\sim$ 5 R/hour at 1 meter from the drained pump axis. A 60°C reduction in CDFR inlet sodium temperature has been assessed to reduce individual nuclide release by a factor 3, except for manganese which is reduced by a factor between 5 and 10.

6.2 Sodium removal and decontamination

Removal of sodium from the tube bundle of PFR No. 3 reheater has been resumed. Following a heat soak at 100°C, austenitic and ferritic steel parts were held at 125°C and 150°C respectively. 990 kg of sodium was drained but internal inspection revealed that many tubes remained blocked. Preparations are now in hand to flush the unit with hot sodium to improve temperature distribution prior to drainage at a higher temperature. The unit will be cleaned subsequently in the alcohol cleaning plant now available. In support of proposals to install a water vapour/inert gas sodium removal process at Dounreay, experimental work has started at RNL in which test sections simulating critical PFR IHX design features are exposed to process conditions, particularly contaminated with sodium and exposed to air. Similar test sections are being used to explore the efficacy of sodium removal by vacuum distillation. Preliminary tests have shown that sodium can be distilled through thick oxide crusts.

6.3 Sodium chemistry

6.3.1 Metals and oxides in sodium

Support studies at AERE have included re-measurement of iron solubility in sodium. A critical review of world data concludes that these measurements are consistent with predictions from a theoretical model which is known to be valid for other metal solubilities in sodium. These data also agree in magnitude and temperature dependence with some US work. Measurements of manganese solubility in sodium are now utilising inert crucible liners rather than apparatus plated with pure manganese.

Electrochemical meters have been used at BNL to measure equilibrium oxygen potentials in sodium – metal – ternary oxide systems. Data have been obtained for the sodium/oxygen/chromium – sodium chromite system. The implications in terms of tribological requirements are being evaluated. Electrochemical meters have also been used to monitor the activities of soluble reaction products in a study of the kinetics of sodium hydroxide decomposition in liquid sodium. Results show that transient formation of sodium hydroxide in the event of water leaks into sodium is unlikely to restrict the use of oxygen meters for leak detection purposes.

6.3.2 Carbon in sodium

Carbon activities in sodium have been measured as a function of dissolved carbon concentration and temperature by using the BNL carbon meter. The nature and solubility of carbon species and the temperature coefficient of carbon activity are being determined.

The BNL carbon meter responds to injections of oil into sodium in the predicted fashion. A new sodium – carbon – hydrogen compound produced when sodium and hydro-carbon oil react at 400°C has been identified. These studies have been extended to cover silicone oil behaviour. At 510°C, reaction with sodium is extensive with release of hydrogen and methane and formation of elemental silicon, amorphous carbon and sodium oxide. No sodium/silicon or sodium/silicon/oxygen products have been detected. Iron silicide has been found on Type 316 steel foils exposed to the sodium-oil mixture.

Effects resulting from contact between sodium and kerosene (used for sodium plant remedial operations) have been examined at RNL. Tests were performed to establish thermal stability in the presence and absence of sodium, and the temperatures at which kerosene or its residue would react with stainless steel. Most kerosene distils in the temperature range to 300°C leaving 1% involatile residue (odourless kerosene; 6% with the commercial grade). Assessments suggest that carburisation following ingress of small quantities of kerosene would not present a serious hazard to the more vulnerable higher temperature plant items.

6.4 Sodium impurity instrumentation

6.4.1 Plugging meters

A statistical analysis of plugging meter flow rate data obtained using the experimental meter in the BNL loop has been completed at CEGB Marchwood. These rate correlations are being used in the development of an adaptive control system for plugging meter use. They support theoretical predictions of methods for increasing plugging meter sensitivity but indicate difficulties in the use of plugging meters for obtaining crystallisation kinetic data for applications to cold trap design and cast doubt on a previously recommended method for separately obtaining the saturation temperatures of several solutes.

6.4.2 Oxygen meter

Development of the electrochemical oxygen meter based on a thoria/yttria ceramic electrolyte and a variety of reference electrodes (air/platinum and metal/metal oxide) continues at AERE. Ceramic thimbles of uniform geometry and high purity (including a shortened version suitable for incorporation in brazed assemblies) are now in routine production. Development of a robust brazed cell is now in progress supported by laboratory tests to identify the most suitable reference electrode. Performance and endurance tests of AERE, BNL and Westinghouse meters are in progress. Failures occurring in the course of these tests are generally associated with transgranular cracking of the ceramic. A joint AERE/BNL assessment to identify the principal failure modes is in hand. General Electric (US) meters have been on test in one of the RNL loops. The average exposure time to failure of all meters tested to data at AERE, BNL and RNL is about 175 days. Construction of the PFR secondary circuit instrument loop is complete and 4 electrochemical O-meters will be installed.

6.4.3 Hydrogen meter

The galvanic hydrogen meter has been under development at BNL for some years. An engineered version has been produced and 4 examples are to be installed alongside oxygen meters in the PFR secondary circuit instrument loop. The laboratory model continues in use as a research tool (6.3.1 and 6.4.1). A re-examination of nickel diffusion membrane performance has been completed at DNE to establish and confirm the conditions necessary for quantitative detection of hydrogen at the katherometer, particularly in conditions during Super-NOAH runs when high hydrogen concentration develop.

6.4.4 Carbon meter

The AERE (diffusion) carbon meter installed in the PFR primary circuit has operated at primary sodium temperatures in the range 485-550°C giving indicated carbon activities of about 10^{-2}. However, whilst calibration measurements are complete at 600°C, they are not at these lower temperatures: extension and completion of this work is in hand. The BNL (electrochemical) carbon meter in PFR has shown behaviour consistent with that observed in the laboratory. There have been delays during periods when the primary sodium temperature was below the melting point of sodium/lithium carbonate eutectic electrolyte mixture, and a low melting point ternary electrolyte including potassium carbonate (MP 400°C) is now being developed at BNL. Provision is being made in the PFR secondary circuit loop to accommodate an AERE carbon meter. Two more instruments have been in use throughout the year at RNL as research tools in loops dedicated to interstitial transport studies.

6.4.5 PFR measurements

In the primary circuit a duplicate carbon meter iron membrane has been used in a chemically reduced condition to pass hydrogen for detection by katharometer. The indicated hydrogen level is consistent with plugging meter records and argon gas blanket analyses. The same membrane has been used to segregate tritium but further development of the sweep gas/counting arrangement is necessary before preliminary values can be confirmed. Arrangements are being made to use one of the installed secondary circuit hydrogen diffusion membrane to segregate and monitor tritium.

6.4.6 Module development

The need to minimise oxygen and hydrogen meter signal instabilities in steam/water leak detection applications has concentrated attention on means to control sodium temperatures at instrument heads. The concept of an instrument module which will do this, and provide protection against thermal and hydraulic shocks, has been developed by Risley and BNL jointly to the stage of an engineered design for electrochemical oxygen and hydrogen meters in pairs on the PFR secondary circuit loop Similar modules are being built into the RNL and AERE Loops.

6.5 Sodium technology

6.5.1 Control of impurities

Experience in cold trapping PFR sodium circuits has been encouraging. Four secondary circuit cold trap (SCTL) basket changes have been performed. During its operational period the fourth basket collected sodium hydride/oxide mixture at an average trapping efficiency of 70%. This basket was fitted with graded mesh, presenting a layer of coarse weave to pre-cooled sodium before it passed to the fine mesh close packed region. The impurity loading of 100 kg was substantially greater than that formed on previous baskets with uniform mesh. Deposit distribution was more uniform and additional capacity remained at the time of discharge.

CEGB (Marchwood and BNL) are collaborating with GEC in studying the control and monitoring of impurities in sodium. A rig has been constructed and commissioned: this incorporates a cold trap ($0{\cdot}1\ m^3$ volume, $1{\cdot}7 \times 10^{-4}\ m^3\ sec^{-1}$ sodium flow) and an experimental plugging meter ($1{\cdot}7 \times 10^{-5}\ m^3\ sec^{-1}$ flow) and the experimental programme is starting.

6.5.2 Cold trap regeneration

Cold trap regeneration proposals using a process loop built onto the Small Water Leak Rig (SWLR) at DNE are being examined: the technique applied would involve flushing of the trap with hot sodium. BNL are looking at the feasibility of regenerating tritium laden cold traps by vacuum decomposition and argon purging in the temperature range 280-400°C. Preliminary results indicate that the kinetics of hydrogen desorption from sodium/sodium hydride mixtures under dynamic vacuum are sufficiently rapid for this technique to be practicable.

6.6 Sodium leaks and fires

At AEEW, acquisition of equipment for the generation of sodium fire aerosols and fumes is proceeding. A stainless steel facility for aerosol generation has been manufactured and design of sampling equipment is in hand. In order to calibrate this equipment 3 techniques for the generation of standard aerosols are under study. A start has also been made on the design of a system for studying the kinetics of chemical reactions of sodium oxide with water and carbon dioxide and in examining the chemical composition of aerosols produced in sodium fires under varying humidities. In conjunction with DNE, work is proceeding on the evaluation of cascade impactors for, and the monitoring of aerosols produced in, Super-NOAH. Techniques for monitoring the aerosol after release from the stack are to be investigated at DNE.

At BNL a refined relationship describing the burning of free ambient sodium pools has been developed: it predicts the burning rates of pools greater than 0·1 metre diameter. Studies have continued on the chemical composition of reaction products formed on the pool surface during combustion. Chemical interactions between Graphex CK23 extinguishant and burning sodium pools have been analysed. Corrosion of steels in the vicinity of burning sodium has been simulated by treating steels with mixtures of sodium hydroxide, monoxide and peroxide.

6.7 Sodium vapour/condensation aerosol studies

Results of basic water analogue studies of heat and mass transfer to the roof using the 2-pool rig at AERE agree with predictions from single pool work after scaling by the ratio of pool areas. They are consistent with the predictions of a theoretical model which assumes convection stagnation in the outer annulus when the outer pool temperature is less than that of the roof. Extension of this work to estimate corrections for wall and roof re-radiation effects is under consideration. Water analogue studies of closed cylindrical geometries above hot cover gas give measured axial heat and mass flux profiles corresponding in form to code predictions. Although this represents an improvement over the use of average value predictions based on empirical correlations a discrepancy remains since measured values exceed predictions based on the vapour temperature at the tube mouth. The discrepancy is largely removed if the hot pool surface temperature is used as input to the code. A review is in hand to rationalise the situation and to identify the correct base temperature to use for code calculations for reactor application. Water analogue studies in the large annulus rig have commenced. Large temperature fluctuations and circumferential variations around the mouth of the annulus are found to damp out rapidly with distance. The large flat plate rig is being modified during commissioning.

The validation of heat and mass transfer models developed from water analogue work will be undertaken in sodium rigs. Detail design of a 600 mm rig with movable cold roof is proceeding, and components are being manufactured and tested. These include a prototype black body cavity needed as a reference source for pyrometry measurements in recognition of the importance of the radiation component in heat transfer. Supporting emissivity measurements will be undertaken in a small rig containing static, mobile and contaminated sodium pools with both clean and oxidised metal surfaces. Detail design of this rig is proceeding.

At CEGB Marchwood Laboratories a small rig has been in use to measure sodium deposition rates at a cool roof from helium and argon in contact with a hot sodium pool. Sodium loadings in argon have been measured and shown to be less than the values in helium under comparable conditions despite the much greater visibility and lower deposition rate found in helium systems: this result is tentatively attributed to the larger aerosol particle size in helium.

6.8 Cover gas clean-up studies

Following measurements of charcoal adsorption bed capacity for krypton a complementary series of measurements has been made with xenon. Because its uptake is much greater compared to krypton smaller experimental quantities of charcoal have been used: results are sensitive to the quality of charcoal used in different runs. The temperature range was the same as that covered in krypton work, but the partial pressure range was extended. The amount adsorbed correlates with the partial pressure according to equations of the Freundich type. This sequence of equilibrium studies has been completed with measurements of argon uptake undertaken to assess the effect on blanket gas pressure of changes in charcoal bed temperature during reactor operation. Attention is being given to the components required for larger scale dynamic gas flow experiments. If injection and mixing difficulties prevent the use of radiotracers the next best detection method appears to be mass spectrometry linked to in-line sampling.

6.9 Water circuits

6.9.1 PFR water treatment plant

Assessment of the DNE raw water treatment plant data has established a relationship between aluminium, pH and organic levels. Detailed improvements to operating procedures have followed: in particular it is intended to operate the plant in a continuous mode ultimately. Engineering and process changes to achieve this are well advanced. Aberdeen University has been investigating the nature and properties of organic impurities in the raw and treated waters. These are found to contain groups capable of complexing iron, aluminium and manganese metal ions. These complexes are more stable than the corresponding resin-organic complex and may account for their passage through some ion exchange resins in the presence of soluble iron. The release of non-ionised chloride-containing impurities from resins has previously been noted: during the year all batches of PFR condensate polishing plant resins were changed and a pre-treatment procedure was developed and applied before new resins entered service with the result that chloride levels in drum feed remained below 0·1 ppb.

6.9.2 Chemical cleaning of steam generators

In support of high temperature chemical cleaning processes work at CEGB Leatherhead Laboratories is showing that EDTA and other strong complexing agents in moderate concentrations are aggressive to the underlying metal at temperatures above 150°C. At low concentrations and temperatures they are suitable reagents provided pH stability is maintained. Theoretical studies show that some chelating agents should retain their complexing ability at high temperatures; confirmatory rig tests are planned at CERL. A limited amount of work at AERE is also directed towards this problem.

7. FUEL DEVELOPMENT

7.1 Introduction

With PFR fuel now at significant burn-up levels, a substantial amount of the development effort has been deployed in making predictions about the PFR fuel behaviour, in checking those predictions against post-irradiation examination data and in considering the relevance of the results to the CDFR fuel design choices. In addition a substantial volume of data has come from post-irradiation examination of experimental fuel irradiated in DFR.

7.2 DFR irradiations

A number of sub-assemblies have received some attention during the year. It has been reported previously that a DFR sub-assembly of 60 wire-wrapped pins showed a high failure incidence when examined at 8·6% burn-up. All these pins were clad in stainless steel with a peak rating of around 600 w/cm. Although substantially more detail has been obtained on the nature of the failures, especially in terms of fuel redistribution and attack on the inner surface of the cladding, it cannot be said that the reason for the failures is clear. It has been established that in addition to the type of broad front oxidative seen in lower rated pins, there is present a notably different kind of attack on the cladding. A related sub-assembly has undergone preliminary examination: this also has a peak linear rating approaching 600 w/cm and was discharged from the reactor at 5·8% burn-up with one pin failed. The interest of this sub-assembly lies in that it contains both pellet and vibro-fuel, different clad materials and a range of oxygen to metal ratios.

7.3 PFR irradiations

The peak burn-up reached in PFR in experimental fuel is about 9%, with the maximum exposed driver-type fuel at over 5%. Fuel has been removed from the reactor for detailed post-irradiation examination at the appropriate time. To date, attention has been turned most strongly to the calculation and monitoring of distortions in the fuelled sub-assemblies, and in the absorber rods and their guide tubes. It should be appreciated that although the nominal concept for the PFR fuel design is that of 'free-standing' units, in practice the core has been allowed to interact and it now has the essential attributes of a passively restrained core. Calculation to the date of the last reload at the end of 1978 showed that shoulder bow values (equal to half the bow at the sub-assembly top) of 10 mm could be expected, taking account of the variability of void swelling and irradiation creep data deduced from DFR studies. In fact, a sub-assembly showing just that magnitude of bow and removed to check bowing has been examined in the PFR caves. Data on sub-assembly length changes have been obtained by the operators by monitoring the fuel in its normal core position. This is commended as a useful and rapid technique for getting distortion data on the same items at successive reactor shutdowns, and in the PFR case, data at the next reload are expected with interest: to date the increases measured have been up to around 5 mm.

A further topic which has received substantial attention is the distortion of the control rod and shut-off rod guide tubes in an interacting core, and the progression towards possible interference loading between the absorber rod and its guide tube. The general strategy envisaged here at the appropriate time is to rotate the guide tubes to limit the bow. A substantial number of rotations have been carried out on PFR fuel both to limit bow and interaction with neighbours, with no handling problems occurring.

So far one sub-assembly and one demountable sub-assembly (DMSA) have given failure signals since the start of operation. Each contained experimental pins of a different design from the pins of the driver charge. The failed pins were operated at a high linear rating and the first failure occurred at a low burn-up (about 1·3%). The failed sub-assembly and DMSA were removed from the reactor and are scheduled for PIE in due course.

7.4 CDFR design

Development studies have continued on the 2 design styles noted in the last report, namely, a 265 pin design peak rated at 550 w/cm and a 325 pin reference fuel rated at around 450 w/cm. The concept of fuel of higher smear density than the reference 80% and a peak linear rating of around 500 w/cm is being put forward more firmly following fuel cycle studies.

The supporting programmes have included the preparation of irradiation experiments in both gridded and wire-wrapped pins of the larger diameter, and sub-assemblies embodying much of the detail of the CDFR pin in the smaller size: for example the incorporation of the upper axial breeder in the core pin, as opposed to PFR practice where a separate pin is used.

Consideration has continued of more flexible grid styles, both to obviate pin loading damage and to facilitate pin removal during reprocessing. Irradiation sub-assemblies are being prepared with flexible grids which have now been made. In the consideration of methods to reduce sub-assembly dilation, bihexagon wrappers have been included in layout and stressing studies.

7.5 General topics

It is a little premature to make any firm statements from PFR evidence on the behaviour of cladding and wrapper materials. With regard to the latter it may be said that stainless steel appears to be performing reasonably to expectation, with rather higher distortions in PE16 than were predicted. However, it must be recorded in both cases that the doses achieved in the material so far monitored are so close to the 'threshold' dose that firm conclusions are impossible.

The UK programmes are continuing to include FV 548 and the silicon and titanium adjusted 316-Type steel designated P316 in pin programmes and planned wrapper programmes. The possible merits of ferritic alloys for wrappers have been pursued to the point of preparing ferritic wrappers for a PFR experiment. Some evidence of recrystallisation in cold-worked FV 548 has been seen which is under review in the context of higher temperature strength.

The fuel development field is currently seen as somewhat crowded with considerations of fuel density (for performance reasons), fuel form, pin linear rating, process route and operator dosage and security, all appearing as relevant to the final choice of routes and parameters. Irradiation experiments have been prepared for PFR with fuel made by gel precipitation and loaded into pins in vibroform. In addition, studies of pellets up to 90% smear density are proceeding starting both from conventional granules and from gel feed.

7.6 Supporting topics

In support of the CEGB role as an informed future customer for Fast Reactors, work is undertaken by their laboratories in important areas of fuel behaviour: examples are given below.

7.6.1 Characterisation of ternary uranium oxides

A wide range of ternary uranium oxides involving alkali, alkaline earth or transition metals of the type observed as fission-product inclusions in irradiated UO_2 fuel have been prepared and characterised by X-ray diffraction. Detailed spectroscopic studies have been reported for the M-U-O system where M is Mg, Sr or Ba.

7.6.2 Thermomigration effects in mixed oxide fuel

The temperature gradient in a mixed oxide fuel pin during service produces a redistribution of oxygen by the phenomenon known as thermomigration. Thus the fuel O/M ratio changes from being initially uniform until it reaches a state in which composition varies with temperature along the gradient. A cluster model has been developed to describe this oxygen redistribution which accounts well for available experimental data and it has been applied to fast reactor fuel pins. Typically, radial redistribution occurs in minutes but axial redistribution, which depends mainly on the rating profile, only in times $\sim$ 150 days. There are consequences for fuel centre temperatures, for fuel-clad interaction, for fuel-coolant interaction in the event of failure, for the formation of plutonium-containing precipitates, for the expected location of fission product phases, for plutonium redistribution and for buffer performance.

7.6.3 Irradiation creep in metals

Results of theoretical studies and experiments on the creep-swelling interaction in pure nickel have been published. Current interest is concentrated on the role of alloying on irradiation creep again using nickel as a model material. A broad study of simple solid solution, dispersion hardened and conventional age hardened alloys has been undertaken at irradiation temperatures where swelling is small.

The test results show that transient irradiation creep was reduced by an order of magnitude and the steady state irradiation creep rate was effectively suppressed when compared with the available data on high purity nickel. Similarly, under thermal creep conditions these alloying elements decreased the steady creep rate by a factor of about one hundred.

In addition, the influence of small amounts of solid solution alloying additions on the creep-swelling interaction has been examined over a range of accurately controlled irradiation temperatures.

A study of the swelling and irradiation creep behaviour of nickel containing small solute additions of Fe, Mn, Co and Si is continuing in the Siloe reactor at Grenoble as part of a collaborative programme with French workers.

At 250°C irradiation creep occurs in the absence of detectable void swelling; at 350°C void swelling is strongly suppressed in comparison with data on high purity nickel.

The results again indicate that solutes providing strong point defect binding are effective in reducing irradiation creep rates over a significant temperature range.

8. REPROCESSING

8.1 Reprocessing of PFR fuel

The work to modify the DFR reprocessing plant at DNE to reprocess PFR fuel has continued and commissioning of the equipment installed in cave is well advanced. The plant will be actively commissioned by late Summer 1979 and will then be ready to receive PFR fuel.

The solvent extraction plant has been extensively modified to cater for the higher throughput required for PFR spent fuel together with flowsheet modification for high Pu/U ratio feeds and the fuel receipt and breakdown cave has been completely rebuilt. The sub-assembly breakdown route depends on partial removal of the wrapper using a laser followed by pulling of individual pins using a type of large lathe bed and travelling pin pulling tool stock. These breakdown machines underwent mechanical functional trials out of the cave and are now installed in the reprocessing plant where inactive commissioning has commenced.

The entire plant containment ventilation system has been redesigned to reduce the treated ventilation air volume to a minimum and a fluidic vortex amplifier control system has been introduced to cater for any emergency situation. The reduction in air flows through the active plant will reduce considerably the number of contaminated filters to be handled in a new remote filter change facility.

The development programme to confirm the PFR fuel reprocessing flowsheet is now almost complete and will in future be closely associated with the commissioning and monitoring of the early reprocessing runs using irradiated DFR and PFR sub-assemblies. To this end instrumentation has been installed in selected positions in the plant and the results will be analysed and compared with the predicted PFR flowsheet performance. As part of the development programme a number of novel features have been examined; these include a 20,000 rpm centrifuge designed to separate fission product insolubles from the dissolver liquor prior to solvent extraction. Units have been installed and are being commissioned in the cave. Work is also in hand to determine the temperature distribution in the centrifuge bowl and in subsequent storage facilities due to the self heating of the separated insolubles.

The use of a laser to remove the end of the sub-assembly wrapper prior to pulling the fuel pins from the assembly and also to cut the wrapper and grids into suitably sized pieces for waste management has been demonstrated and in cave commissioning of a 400 watt laser is in hand. A "magnetic filter" will be used to remove the activated steel particles from the laser smoke.

It is convenient in the PFR flowsheet to use flocculation treatment of the medium and low active liquid effluents to remove fission products and residual traces of plutonium. Studies have confirmed adequate decontamination factors and defined plant operating conditions. In order to obtain a high density floc and hence low volume storage, precipitation at $\sim$ 80°C in the presence of sulphuric acid is required. pH is an important parameter and work is in hand to develop a metal hydride electrode to replace the fragile conventional glass electrode.

The extensive use of plastic materials in radioactive facilities gives rise to problems of treatment, storage and disposal; methods of reducing such arising are being investigated. A major contribution is expected to come from the use of "bagless posting" and a prototype unit extending this system to the changing of HEPA filters has been incorporated in DNE facilities.

Reductions in the level of contamination on plastic waste arisings can realise saving in storage costs particularly if it is possible to reduce the activity to levels which are appropriate to storage in unshielded stores. A washing/shredding machine has been developed for this purpose and a unit is being installed in a highly active cell for operational use.

Because of the possible adverse effects of the chloride content of PVC on storage and incinerator construction materials, alternatives to the use of PVC in active areas have been examined. The results have enabled other preferred plastics to be substituted for PVC in many applications.

8.2 Generic programme

The objectives of the generic fast reactor fuel reprocessing development programme include:

(i) the provision of design information for a Reprocessing Unit, of sufficient capacity to cater for the fuel discharged from CDFR, and

(ii) the demonstration of specific unit operations at a scale equivalent to that required for a future larger plant or having easy and reliable "scale up" capability.

A reprocessing plant, with a capacity of 250 Kg Pu + U/day, could cater for spent fuel from about 4 GW of fast reactors and a number of parallel units of this size could constitute the plant units of the future, to ensure the level of plant availability essential to support a system of operating fast reactors.

A fuel transport study has been completed against an agreed set of ground rules and reactor installations scenarios. The major conclusions are that it is considered, on present evidence, feasible to handle and transport irradiated fuel in sodium filled canisters at decay heat ratings in the range 6-8 kW. The logistics and costs, used in the study, suggest that there are economic and technical benefits in storing the fuel in the actual transport flasks prior to sodium deactivation. The report emphasises that high process reliability is required if excessive storage capacity and major economic penalties are to be avoided. A programme of work has started to examine the decay heat transfer behaviour of sub-assemblies up to 10 kW rating in different cooling media and the effects of fuel pin removal from a sub-assembly on pin cladding temperature due to preferential coolant streaming.

The massive shearing of complete sub-assemblies prior to dissolution has been operated actively in large scale thermal reactor fuel reprocessing plants in the US. UK fast reactor fuel, with its heavy wrapper and small pins in grids, poses particular problems for this dismantling method and development has been concentrated on dismantling the sub-assembly before fuel pin cropping and dissolution. However some work is planned to obtain fundamental information on the mechanisms of shearing and also some inactive experience of shearing fast reactor sub-assemblies. The introduction of new concepts in fuel design such as compliant grids, end tube plate pin fixtures and grid support features, encourages the hope that withdrawal of the pins and grids from the wrapper in one operation will be a future possibility. Embrittlement of grids and fuel cladding is being investigated as a possible aid to sub-assembly breakdown and fuel exposure. Encouraging results have been obtained from a process in which cladding is exposed to ammonia at 500-800°C for some 72 hours and also from exposing irradiated cladding to zinc.

A study of dissolution behaviour of mixed Pu/U oxide fuel, both unirradiated and irradiated, is in hand and includes examination of the influence of the fuel fabrication route and characterisation of feed material to the fabrication process on the amount of insoluble plutonium remaining after dissolution. These results have been used to define a dissolution/acceptance test for inclusion in the Fuel Manufacturing Specification.

A number of solvent extraction flowsheets are being examined based on the PUREX system including the use of dilute TBP/OK to improve fission product decontamination and reduce solvent degradation effects as demonstrated in earlier DFR reprocessing operations. While no decision has been taken on the type of contractor to be used in the fast reactor fuel separations plant the possible use of pulsed columns and an electrostatic column is being considered. Possible flowsheets are being identified for the recycle of nitric acid and water in the reprocessing plant to minimise environmental discharges.

In order to improve on the control of gaseous releases to the atmosphere from a once-through ventilation system, some preliminary work is under way on a small closed cycle thermal flow ventilation loop. Information of value to the CFR reprocessing development will arise from the commissioning and operation of the PFR reprocessing plant. To this end, sampling points have heen incorporated in the PFR reprocessing plant to obtain information on the amount of fission product gases and volatiles evolved during fuel breakdown and dissolution to provide data for the design of an off gas clean up system. The provision and testing of a solvent clean-up facility using solid absorbents is being considered. The high specific activity of FR high active raffinates poses a serious problem in the design of storage tanks. A number of novel high active raffinate storage systems are being considered which would allow simpler construction and high quality assurance standards and give increased heat transfer area/unit volume without the provision of individual cooling coils. One example, utilising a helical coil principle has been assembled for inactive circulation investigations using a fluidic pump, and this work will be extended to include studies of solids in suspension and heat transfer.

9. SAFETY

9.1 General survey

Discussions on the CDFR safety case have continued throughout the year and presentations have been made to the Advisory Committee on the Safety of Nuclear Installations which includes representatives from public bodies, industry, the universities and trade unions. The safety case that is emerging on containment of whole core melt-down accidents is that the chance of any such accident occurring can be made acceptably low so that there is no special requirement to modify or strengthen the containment substantially. Nevertheless the containment will be made strong and able to withstand a particular size of excursion known as the Design Basis Excursion. The course of whole core accidents will continue to be studied and their calculated energy yields will be compared with that of the Design Basis Excursion.

Although the safety case is made in the context of a probability analysis, the limitations of this approach are not forgotten. Thus gross operator error, catastrophic structural failure and common-mode failure by as yet unidentified means are mechanisms which can in principle substantially reduce the safety margin of the plant. This has implications for the way in which the safety argument is assembled. Some common-mode failures (meteorites, earthquakes) may affect all parts of the safety protection simultaneously, but an important group (e.g. mistakes in servicing) affect only certain parts. There is a case for categorising such failure and ensuring that no one category on its own is responsible for too high a proportion of overall safety. This approach, described in terms of "levels of assurance" or "protective barriers" is well known. It leads to a justification for the requirement for containment and PAHR provision, although a probability approach also leads to a similar justification if due allowance is made for uncertainties in the estimates of probability.

More detailed studies of the processes involved in a whole core accident continue to indicate that the nuclear disassembly part of the accident is less energetic than suggested by earlier results using simplified idealised representations. A major factor has been the use of results derived from experiments on rapid transients and failure of pins in the large in-pile experimental rigs SCARABEE and TREAT. Of particular importance is the axial position of failure of pins during such transients. In particular, the slow rod withdrawal fault (without trip) which had previously yielded a mechanical disassembly energy of 13 GJ has been reduced to 3 GJ as a pessimistic estimate and 0·5 GJ as a more likely estimate. These estimates may be compared with an estimated containment strength of about 1 GJ. The emphasis placed on obtaining a design of core with a low sodium void coefficient is thus somewhat reduced. Further attention is now being given to the possibility that for some types of accident the initial excursion energy will be so low that a major core disassembly would not occur and a boiling pool of material would result. The modes of possible development of such a pool will require study to determine whether an energetic excursion can occur at a later stage.

The use of event trees for analysing accident development sequences has been applied to the single sub-assembly fault. Although only one particular sequence has been analysed in detail, the indications are that the probability of a whole core accident arising from local pin failure developments in a single sub-assembly are extremely low. This in turn has led to continuing discussions as to how far protection in each sub-assembly is required. The event tree approach has proved to be a very useful tool, and will probably be widely used. It involves a large amount of detailed work and focuses attention on the necessity to attach degrees of confidence to the probability estimates made by various experts. One factor encouraging a more optimistic interpretation to be placed on the development of sub-assembly faults has been the stable and relatively innocuous boiling modes obtained in the DFR special experiments.

The demonstration of the required very high integrity of the core support structure continues to be a matter of some difficulty. Design and other solutions are being explored.

9.2 Fuel failure

9.2.1 Analysis of fuel failure experiments

The work on this subject has covered four international collaborative in-pile experiments:

(i) SCARABEE: now almost completed.

(ii) CABRI: the KfK-CEA programme in which the AEA is a junior partner.

(iii) PINEX: the US benchmark experiments in TREAT for which predictions were submitted by the UK.

and (iv) PFR/TREAT: still at the planning stage but for which pre-test calculations are under way.

9.2.2 CABRI instrumentation – failure location by acoustic methods

An important requirement in the CABRI experiment is to know the exact time of fuel failure, so that the integrated power input and the place where rupture starts can be determined. This information may be masked at post-irradiation examination if the failure propagates along the pin. As a support to nucleonic imaging using the hodoscope, acoustic monitoring is being investigated by RNL. Two special microphones, able to withstand temperatures of up to 560°C, are being used to time the arrival of acoustic pulses so that the position of the sources of noise is obtained from a knowledge of the acoustic propagation velocity. Interference arises from the effect of the radiation transient on the measuring system and also from the associated noises, for example from the depressurisation of the helium circuit to initiate the transient. The results from both the preliminary tests in the TRIGA reactor at Neuerberg, in the "Instrument Check" test section and also the first 'A' series test have been encouraging. It is proposed to continue the investigation of the acoustic technique at least until the first pin failure provides an independent check.

9.3 Sodium boiling

9.3.1 Sodium boiling theory

This work has continued to concentrate on the development of a number of different versions of the SABRE code of increasing complexity and mathematical difficulty. Excellent progress has been made in a number of areas. These include: a possible solution to the problem of false diffusion; the explicit representation of spacer grids or wire wraps; methods for dealing with the rapid changes which occur near a boiling or condensing boundary; and completion of a transient version (SABRE 2) which covers transients in single phase conditions.

Development of a transient boiling capability has progressed slowly. Pressure dependence of thermodynamic properties is seen to be an important factor which stabilises rapid boiling excursion, and the Los Alamos ICE technique has been incorporated to deal with the effects of compressibility.

Comparison with water experiments which use a single row of pins has given good visual support for the predicted flow patterns even in complex situations with a leaking blockage and multiple wake vortices. The experiments at KfK both in water and sodium are the most important for checking temperature rise predictions. At present there are sufficient differences between experiment and theory to make it clear that either important aspects of the theory, e.g. turbulence modelling, need improving or that the experiments contain detailed features whose significance is not yet recognised. Very generally, it can be said that experiments with central blockages in full hexagonal sub-assemblies confirm the predictions, but when the blockage is moved to the corner the measured temperature rises are much higher and this effect is only partly predicted by the theory. The data now available however provide an excellent basis for validation of theoretical models up to the stage of modest degrees of boiling.

9.3.2 Water modelling and associated work

Work on the Visual Flow Rig at AEEW has been concerned with the determination of the axial flow friction factor of the sub-channels with two different values of pin-to-wall gap. The experimental values fall significantly below the values obtained from the standard SABRE I correlations. The pressure drop caused by the pin support grids and by a solid blockage has also been determined for use on the SABRE code modelling of the assembly. Work has started on the determination of velocities and turbulence parameters in the wake behind a blockage in the 11-pin array. Experiments on the heated 11-pin test section in the Low Pressure Water Modelling Rig have included single phase tests to determine the effect of unheated pins on the temperature distribution in the wake behind an impermeable blockage. The object of this work was to determine the extent to which unheated pins can be accepted in a multi-pin experiment, either deliberately to reduce the power requirement, or accidentally as a result of pin failure.

9.3.3 Blockage formation

A 7-pin test section using CFR style grid spaces has been built for preliminary investigations into the mechanism of blockage formation. A number of tests have been made in water using sand of specific size ranges in order to determine the minimum size of material which will form an initial blockage. Work is now being done on two- and multi-component mixtures. Techniques for observing the growth and shape of the blockages have been developed, and an attempt will soon be made to apply these to blockage formation in bundles containing more pins.

9.3.4 DFR special boiling experiments

An important aim of this programme was to demonstrate the unambiguous detection of boiling in a realistic operational environment by both thermal and acoustic noise techniques. It was found that the incipience of boiling coincided with a marked increase in the level of thermal noise measured by rig outlet thermocouples. Measurements of thermal noise levels were taken as the guide for the duration of each boiling run. Visual confirmation of the correlation between the incipience of boiling and an increase in thermal noise was obtained from an out-of-pile water rig at BNL.

The PIE of all the rigs will take a long time to complete but it is already clear that one of the major programme objectives, to demonstrate the durability of the fuel pins, has been achieved. Pins that have been subjected to temperatures in excess of 900°C for several hours show on examination no trace of failure or even marked dilation. Furthermore it has been shown that when failure does occur it appears to be in a manner which precludes rapid propagation of the incident; even when fuel debris has been released from failed pins there has been no sign of further blockage build-up at downstream grids. It is encouraging that coolant boiling, even in experiments with very low pressure heads, appeared to be quasi-stable for a period of an hour or more.

Conditions in the DFR experiments are considered to be pessimistic compared with PFR/CDFR, i.e. they involved low downward flows and there was a susceptibility to gas entrainment. The results obtained from the programme so far support the contention that an adequate safety argument for the sub-assembly incident can be based upon the observed behaviour of fuel under incident conditions.

9.4 Signal detection and processing

9.4.1 *Acoustic detection of boiling*

The investigation into methods of achieving low pump background noise has proceeded and work at Northern Engineering Laboratories (NEL) is clarifying the laws which govern the scaling of pump noise from model tests to full scale pumps.

In the DFR special experiments, in addition to the direct noise measurements, coolant boiling was also detected by an acoustic location technique which relies upon the time of flight of a pulse to different detectors, the relative difference in arrival times giving a guide to the location of the signal source. In the experiments, the boiling had to be detected against a background of cavitation noise from a nearby valve. The RNL pulse location system successfully discriminated between the valve noise and the boiling pulses in all tests. This technique has proved mosr valuable in rig experiments and should prove a valuable addition to a boiling noise detection system enabling, for example, waveguide rattling noise to be rejected.

9.4.2 *Temperature noise detection of boiling*

The fluctuations in outlet coolant temperature may be used as an early warning of a blockage in a fuel cluster. As one attempt to put the method on a theoretical basis, a computer model of turbulent flow in a pipe using random walk techniques has been prepared. Preliminary attempts to apply the technique in practical situations have been reasonably encouraging. At BNL an experiment to measure the noise decay rate in sodium has been designed for the No. 4 Loop. The factors which govern the stability of local boiling hehind the blockage are being investigated through an analysis of the results of the KNS electrically heated pin bundle experiments at KfK Karlsruhe.

9.4.3 *Processing of sub-assembly signals*

Continued progress has been made in the development, demonstration and proving of disparate logic systems for CDFR which possess both the fail-safe characteristics and self-diagnostic properties essential for sub-assembly protection. It has now become clear that there is an overwhelming case for the transmission of thermocouple signals from the rotating shield by means of multiplexing and proposals have been made for the construction of a rig to enable earlier studies and theoretical evaluations to be checked by obtaining experimental data on performance and reliability.

9.4.4 *Pulse coded logic for safety circuits*

A demonstration system with 14 parameters has now been manufactured, bench-tested and delivered to the CEGB power station at Oldbury for future operation in a passive role. The system has been engineered to an appropriate high standard for reactor use and includes printed circuit boards. Adequate immunity from interference has been demonstrated, but it is recognised that further engineering development of chassis construction is desirable to provide further protection against interference.

9.4.5 *Improved Laddic safety circuits*

A demonstration system (with around 10 parameters) using 'CANDI' Laddic has been completed for installation in DIDO and successfully commissioned. The system includes self-diagnostic facilities which is an important new development incorporated into 'CANDI' Laddic to make them suitable for CFR sub-assembly protection duties.

9.4.6 *Computer-based systems*

Development of earlier proposals for fail-safe, self-diagnostic microcomputer-based systems has continued. A highly selective pattern recognition circuit has been evolved which generates a dynamic (healthy) output in response to the uniquely coded serial input pattern for which it is designed, and a static (tripped) output, (i.e. continuous logic 1 or 0) or if any of the input bits, which comprise the pattern, is abnormal. A failure-mode-and-effect-analysis (FMEA) has shown that it is fail-safe for all simple logic failures.

9.4.7 *Reactivity anomaly detection*

On the basis of neutron flux noise data in PFR, a rough estimate has been made of the likely reactivity variation which could be detected by a simple neutron flux increment trip instrument having a trip margin of 7 or 8 times the rms noise level. On available evidence it would seem reasonable to detect reactivity variations of the order $\pm$ 1 cent in PFR. However, this figure may have to be increased when other effects such as control manoeuvres are taken into consideration.

9.5 Molten fuel coolant interactions

9.5.1 *Metal/water experiments in THERMIR*

Experimental work in THERMIR has concentrated on characteristics of the propagation stage of metal/water interactions. Recent tests in which a detonator was used to trigger the interaction at one end of a trough shaped interaction vessel have confirmed that propagation through a dense mixture of tin and water is associated with the passage of a shock wave. High speed films have shown that interaction propagates at a speed of order 100 m/s, and that the observed front corresponds with the passage of a shock wave which is observed by an array of transducers along the length of the interaction vessel. These data are consistent with the thermal detonation model of energetic thermal interactions. This work in THERMIR is continuing with aluminium/water mixtures which are expected to result in more energetic interactions and higher pressures. In addition small scale experiments are planned to investigate the role of vapour collapse in producing rapid fragmentation.

9.5.2 *Studies using Thermite charges to produce molten UO_2*

Experiments in Thermite Rigs A and B have investigated the release of molten UO_2 under both water and sodium. In the water rig a systematic investigation of the behaviour of 0·5 kg charges has almost been completed. The degree of dispersion of the molten charge has been varied by changing the pressure and volume of the cover gas region. For low pressure and large volume the initial expansion of the charge results in wide dispersion, and there are no indications of any interaction. Increasing the pressure and reducing the volume of the cover gas both tend to reduce charge dispersion. Under some conditions the impact of the end cap of the charge container on the base of the vessel provided a sufficient disturbance to trigger a small coherent interaction. Experiments with similar conditions in the sodium rig are still proceeding, and so far there has been no evidence of energetic interaction in the sodium rig.

Construction of the new facility for assembling and firing large (up to 20 kg) Thermite charges is in progress. This rig will be used both for charge development and for studying problems associated with entrapment of sodium in molten core debris.

9.5.3 *Fuel coolant interaction theory*

The detonation model of MFCIs based on a self-sustaining shock wave concept has been developed further by BNL. It has been shown that the effects of thermal disequilibrium in the coolant at the CJ plane allow low pressure, low efficiency detonations at least in poor conductivity coolants. The possibility that similar effects could occur in sodium is being investigated. The efficiency of hydrodynamic fragmentation at low shock strengths has been demonstrated for single drops of mercury in water. This work is now being extended to include the effects of a dense dispersion. The role of other fragmentation mechanisms, e.g. vapour blanket collapse and violent boiling, is under consideration. The possibility of large scale tests of the detonation model, (e.g. at JRC Ispra) has led to the initiation of feasibility studies of relevant contact modes using simulant fluids.

The detonation model proposed by BNL has been considered in the context of 3 possible modes of particle fragmentation. These were:

(i) relative velocity causing break-up of droplets.

(ii) vapour film collapse providing impact energy to cause break-up.

(iii) vapour collapse terminating in an asymmetric manner and giving rise to jet formation capable of penetrating droplets.

In all cases the distributed particles must retain their heat during the time taken to set up the distribution. This condition could be satisfied if UO_2/sodium could support stable film boiling but this is considered unlikely. There is therefore an indication that while these processes could give an energetic interaction with metal/water and also in rather contrived conditions of particle distribution and heating, the conditions in the fast reactor would not be appropriate to such processes.

9.6 Core structure tests

9.6.1 Sub-assembly accidents

The UK experimental programme of work on the crushing of sub-assemblies following a localised MFCI has proceeded slowly, principally because of continued difficulties in procuring low ductility materials for the simulation of irradiation embrittlement. Useful work on fuel pin simulation and grid crushing response has been completed. A detailed study of the manufacturing process being employed for brittle materials has been undertaken in order to identify variations, process limits and the potential for obtaining really low ($\sim 2\%$) ductility levels. This study will continue until a satisfactory route has been established.

Supporting theoretical studies have continued but further development of the core damage propagation and springback codes (SPOKE and SPRING) is necessary. Some work has commenced on a theoretical appraisal of the effect of defects (surface cracks) on wrapper response to internal/external loadings.

Experimental work has continued on investigating the compaction that may result from an energetic fuel-coolant interaction in a single sub-assembly. Earlier tests using a 1/5 scale model of a free-standing core with 126 sub-assemblies, but without singularities within the core, showed that some small inswing could occur following the initial outswing caused by the sub-assembly energy release. Tye introduction of 7 semi-rigid leaning post singularities within the core virtually eliminated the springback in the model tests in cold water. Consideration is being given to ways of simulating the reduction in viscosity applicable to full scale hot sodium conditions, to the effect of a condensible vapour energy source instead of an energy source using non-condensible gas and also to the tests required to simulate a restrained core.

9.6.2 Wrapper fracture

The irradiation induced loss of ductility of wrapper materials will render them susceptible to fast fracture as a consequence of impact loading during a sodium vapour explosion, or accidental stressing during unloading. The fracture mechanics concept of critical crack opening displacement (COD) measurement is to be employed to evaluate the critical defect size required to initiate rapid propagation in irradiated wrapper materials. Specimens of Nimonic PE16 have been prepared and are awaiting irradiation in PFR. In the meantime specimens have been machined from a DFR control rod carrier of 321 stainless steel and fracture testing has begun.

9.7 Accident development

9.7.1 Nuclear excursion-yield calculations

This work is aimed at developing modules for the whole core accident code FRAX. Work has concentrated on aspects of fuel coolant interaction after failure. Revised failure criteria have been formulated for FRAX 2 and these make a substantial difference to predictions of accident sequences. The incoherence of failure occurrence and location of failure near the top of the core lead to much reduced yields. A fuel motion model allowing for fission gas release and fuel expansion on melting is now being tested. This incorporates transient modelling of the fuel motion of the type observed in the PINEX experiments in the US. The next development being undertaken is to expand code capability to accept 3-dimensional reactor model data coupled with a channel by channel disassembly model. These developments will enable the code to be utilised for studies of heterogeneous cores.

The European comparative calculation study instigated by the whole core accident sub-group of the EEC Safety Group has continued to show that all the codes considered give reassuringly consistent results. It is proposed to extend the study to more realistic reactor models.

Studies made with the FRAX 2 for the rod withdrawal fault in the reference CDFR design have produced estimates of nuclear energy yield substantially lower than those computed using earlier versions of the code. This reduction in yield is mainly a consequence of current fuel failure modelling which predicts pin failures at the top of the core. Present yield values are within the potentially containable range. Work envisaged in this area in the coming year includes the assessment of advanced CFR designs including a heterogeneous core proposal, although current nuclear yield estimates for CDFR are such that the case for heterogeneous core designs, on low void safety related grounds, would appear to be reduced.

The code SARAZE-2 has been used to investigate the effect of sodium remaining in the outer core region during a hypothetical transient. It was shown that, for a variety of input conditions including a range of realistic reactivity ramp rates, the presence of sodium can either reduce or increase the energy release depending on whether the accident is mild or more severe respectively but that any increase compared with the voided core case is not great.

The code SARPIN analysing the movement of fuel within intact fuel pins has been modified to investigate the effect of a threshold pressure restraining fuel movement. Calculation using a range of threshold pressures showed that reasonable values for such a restraint, arising from friction or stiction perhaps, could not affect the energy release.

9.7.2 Pool boiling

Work has continued on the possible formation and properties of a boiling pool consisting of a mixture of liquid heat-generating fuel and 2-phase steel. If the core, during a hypothetical accident phase, could reach such a configuration, it might markedly affect the characteristics of any final energetic disassembly. For example, the compression of a pool by an external MFCI might add reactivity at a high rate and lead to energetic disassembly. Various qualitative scenarios of possible pool formation which have been postulated elsewhere have been considered.

A description has been written of the code 1DPHASE which is being developed to model the heat transfer and hydrodynamics of a mixture of heat generating single phase incompressible fluid (fuel), yielding energy to and moving through a 2-phase mixture (steel). The code is still at the development stage and has not yet been used for production calculations.

9.7.3 Equation of state

An IWGFR Specialist Meeting on this subject was held at Harwell in June 1978. As a result, a vapour pressure-temperature relationship, with an error band accounting for available data, was defined for urania, and good definitions of the thermodynamic regions of interest also being defined for sodium and stainless steel. The effect of incorporating plutonium needs further quantification. A paper presented to the meeting by the Safety and Reliability Directorate suggested that the observed rise of the specific heat of solid UO_2 with increasing temperature is better explained as being due to an electronic contribution than to defects in the lattice as presently assumed. This could significantly affect the thermal capacity and thermal conductivity of molten UO_2.

At BNL a sensitivity study of the effects of uncertainties in the equation of state has been completed. A simple model for determining the pressure-volume relationships in the expanding core bubble given an initial core energy distribution has been developed. Calculations of the production of aerosol from a failed primary containment have been improved. The requirements for further experimental data on hydrodynamic fragmentation are being identified. A model has been developed to calculate the reactivity effects of a small MFCI which removes the sodium from the spaces between sub-assemblies.

9.7.4 *Effects of fission gases on fuel failure modes*

The VIPER programme has been directed towards the resolution of experimental difficulties associated with the reliable measurement of fission product pressures during fuel melting. A satisfactory experimental configuration has been found in which irradiated fuel from DFR can be melted in the VIPER experimental capsule without damage to the pressure measuring transducer. The pressures developed can be measured reliably and can be resolved on a millisecond timescale starting from the initiation of the reactor pulse. Substantial pressures have been observed from experiments in which irradiated fuel has been melted. Pressures are developed within a few milliseconds of the reactor pulse and analysis of the released gas has shown that complete fission product gas release has occurred. However the released gas is mainly non-fission product gas, e.g. CO and H_2, with the result that the timescale for the release of fission product gas cannot be resolved.

Studies with unirradiated $(U,Pu)O_2$ have identified some of the sources of the impurity gases and fuel melting experiments have been completed in which the gas release has been commensurate with the known impurity content of the fuel. Preparations are in hand to restart experiments with irradiated fuel using improved fuel handling techniques. In the longer term, fuel is to be irradiated in the AERE reactors, and this fuel is to be transferred directly to the VIPER experimental capsule without exposure to handling cell atmospheres.

9.8 Containment

9.8.1 *Excursion containment experiments*

The 20 tests of the original Winfrith Code Validation COVA programme have now all been fired as have the majority of the corresponding Foulness COVA tests. Experiments designed to investigate means of reducing coolant slug impact on the roof have started and firings in three geometries have been carried out. Problems in interpreting load cell data have necessitated the design of a structure in which the plug is mounted independently of the vessel; and it is hoped to have this new support ready during 1979.

A successful test has been carried out at Foulness on a third 1/22 scale CFR model with steel outer containment. This model approximates to the more open pool of the present reference design whereas previous tests have been with the core supported from the roof by a thin cylindrical tank. The test which approximated to a $2\frac{1}{2}$ to 3 GJ excursion severely damaged the above core structure and caused some splitting of the thin sodium containment jacket, but left the outer guard jacket intact.

9.8.2 *Analysis of experiments*

Most of the analysis effort has been aimed at bringing SEURBNUK to a state where it can be considered to be a working tool. SEURBNUK calculations have now been completed for about 10 COVA tests representing various configurations, and for one SRI experiment. Comparisons of measured and predicted roof pressure loadings indicate that SEURBNUK is able to predict fairly realistic roof pressures and impulses for short tank COVA tests with small cover gas gaps. However, several calculations have been completed for long tank tests with larger gas gaps and in these cases predicted impulses are too high by 30-50% and pressures are not representative. In the first stage of a new collaboration agreement with JRC Ispra for the further development of SEURBNUK there has been an emphasis on COVA work. SEURBNUK calculations for a number of COVA tests were satisfactorily completed during an initial consolidation phase. Continuing work to incorporate the CHAM model for fluid flow through permeable strucrures is one of the topics identified for the next stage of the agreement and this work is well advanced.

There has been a continued effort to resolve errors in strain predictions for thin outer containment vessels in COVA tests. Collaboration with Germany under the KNS/COVA agreement has focused on this aspect and has resulted in an agreed collaborative material test programme between UKAEA, KfK, Interatom and JRC Ispra which aims to improve the data for the thin vessels. In addition to modelling problems associated with the thin vessels, it has also been found necessary to represent the elastic response of the thick COVA vessels in order to produce adequate predictions of floor pressures and impulses (i.e. the vessels are overstrong but not rigid).

9.8.3 *Accident containment theory*

Development of a rezoning method in ASTARTE has recently reached the stage which enables computation to proceed beyond roof impact by using one stage of rezoning. The resulting small error in conservation of energy seems quite acceptable. Some improvement of the wave shape of impact pressure can be obtained by introducing viscosity of a similar character to the false viscosity in Eulerian codes such as SEURBNUK.

The SEURBNUK code has now been brought into use for COVA analyses and reactor calculations. The complexity of possible bubble configurations caused a number of initial failures but in collaboration with JRC Ispra a substantial number of demonstration cases have now been successfully completed. An analysis of the finite difference scheme used in the code has shown that false viscosity is introduced. This is very common in Eulerian methods and accounts for the rather smoother behaviour predicted for roof impact. A better understanding now exists of the role of this effect and how to modify it by choice of time step.

In order to couple the code to more substantial structural calculations than the thin shell theory used initially, the finite element code EURDYN from Ispra has been linked to the hydrodynamical calculation. Calculations for the linked system agree with the thin shell theory calculation on an appropriate test problem. A collaborative agreement with JRC Ispra to cover further development has been established.

9.8.4 *Containment analysis*

The CADROS dynamic finite element structural response code development has continued. Proposals for application of the code to the analysis of the CDFR rotating shield and hold down system response to accident loadings have been prepared and this work should be under way in the coming year. The complementary 1D rotating shield response code ROSHOD has been developed further. Roof loading data can now be input in a form compatible with hydrodynamic loading code output. The code has been used to define roof impulse history accuracy requirements for the SEURBNUK code using COVA data.

9.8.5 *Missile studies*

A launcher, driven by compressed air, has been constructed at AEE Winfrith which enables missiles up to 150 mm diameter to be fired at concrete target panels up to 2·4 m diameter. Velocities up to 300 m/s can be achieved depending on the mass of the missile. A laboratory for the manufacture of concrete targets will shortly be completed. The scope of the missile studies has been extended to include impacts on metal targets and a series of drop tests with impact velocities up to 19 m/s is nearing completion. The velocity range will be extended following commissioning of the missile launcher. Transient deflection measurements from these tests are being used to validate the finite element codes EURDYN/2 (axisymmetric) and CADROS/DPS (3-dimensional) for use in reactor accident calculations.

9.9 Post-accident heat removal

The particular objectives are to investigate dryout behaviour of heated particulate beds; fluidisation characteristics such as the formation of vapour vents as in 'wick' boiling; dynamic behaviour such as 'bumping' resulting from sudden release of vapour; and improvements to bed heat transfer resulting from the use of catchment trays with perforated bases to allow the liquid to circulate more freely. Two approaches for simulating, experimentally, heat generation in a particulate bed are under active development: these are electrical resistance heating and controlled depressurisation of beds.

Tests have been performed which confirm that consistent results can be achieved by passing electric current through beds of tinned metal shot. Apparatus has been set up to obtain data on the performance of such beds immersed in water, which is a good simulant of sodium. Another idea examined is the use of a bed of copper particles immersed in carbon tetrachloride. When this system is slowly depressurised, starting with the liquid at boiling temperature, the majority of the heat evolved comes from the copper and thus simulates very closely a heat-emitting bed of particles. Apparatus to obtain data by this method has been constructed.

Small scale interaction experiments at Winfrith indicate that although concrete, basalt and alumina each have the potential to dissolve UO_2, concrete and basalt, close to the liquidus temperature, appear to be slow in distributing the UO_2 through the melt. The potential of these materials as sacrificial material is also complicated by their thermal decomposition which gives rise to large quantities of water. Initial experiments with alumina, particularly in the form of a silica containing refractory, indicate that the UO_2 dissolves quickly and distributes throughout the melt.

Work at Harwell is directed towards the determination of the constitution of a molten oxide-fuelled fast reactor core, and the associated redistribution of phases on cooling to the solid state. The results of arc-melting simulated irradiated coria have shown that the alloy Mo-Tc-Ru-Rh-Pd, which occurs as a separate phase in the solid state, is also immiscible in the liquid state with molten urania. The transition metal alloy is denser than urania. Thus a molten oxide core would not be homogeneous.

9.10 Active gases and aerosols

Basic work is in progress on the techniques of producing and measuring aerosols in order to be able to trace their behaviour if they are released into the secondary containment. A number of impactors of different design have been obtained together with a range of equipment for generating standard aerosols. Work is now proceeding on intercomparison of instruments. Infra red studies are continuing on container boundary effects ' The uncertainties or inadequacies associated with existing computer codes have led to the writing of a new code AEROSIM in which there are no restrictions on the form of particle size distribution and which uses efficient numerical techniques. At present, AEROSIM is restricted to dry aerosols, and thus to secondary containment applications. It will require further development to include evaporation/condensation for application to the primary containment. It is also intended to develop links between AEROSIM and the sodium burning code FRASC, and between AEROSIM and the atmospheric dispersion code TIRION.

There has been participation in the activities of an expert group on nuclear aerosols in reactor safety which was established during the year by CSNI. Although the terms of reference include thermal as well as fast reactors, the major emphasis has been on the LMFBR. The major activity of the group has been to produce a state of the art report and identify areas where more information is needed.

9.11 Effect of heated plume rise

An investigation has been performed by the Safety and Reliability Directorate of the UKAEA on behalf of the EEC, using the code TIRION, to examine the effect of plume rise following release of radioactive material from a fast reactor accident. The plume rise is assumed to be caused by the heat of combustion of about 10 tons of sodium, the radioactive decay heat being found to have little additional effect. The number of casualties depends on the prevailing weather conditions and the characteristics of the release. It is proposed to continue the study, including the effects of rain, radioactive particle deposition velocity, the composition of the radioactive material and various turbulent diffusion models.

10. REACTOR PERFORMANCE STUDIES

10.1 Neutronics design data

10.1.1 Nuclear physics data

The programme to provide differential nuclear data relevant to fast reactor design and fuel behaviour has continued. The new 136 MeV electron linac is expected to be installed shortly at AERE and, following a commissioning period of a few months, experimental use should begin during 1979.

For six months during 1978 neutron time-of-flight experiments on the Harwell synchrocyclotron were devoted to the measurement of total cross-sections of structural materials. Analysis of the data will yield resonance neutron widths which are used for multiple scattering corrections in capture cross-section measurements.

The previous analysis of ^{58}Ni and natural nickel transmission data has been extended in the case of ^{58}Ni from 300 keV to 1000 keV, with emphasis on the determination of s-wave resonance parameters but with the necessary inclusion of resonances from higher angular momenta to ensure adequate fits in the shape analysis by a single channel, multi-level R matrix code. Parameters for 62 s-wave resonances and 228 $\ell > 0$ resonances have been found below 1000 keV. Values above 700 keV are not final and this analysis was carried out to reduce the distant level effects for the 41 $\ell = 0$ resonances in the 10-640 keV range where the spacing curve is linear. Parameters are given for this range where the observed ($\ell = 0$) spacing and strength function are 16·3 keV and $2{\cdot}9 \times 10^{-4}$ respectively.

An evaluation of differential neutron data has been carried out for ^{241}Am, which will be incorporated into the UK Nuclear Data Library, for the range 10^{-5} eV to 15 MeV and includes the total, capture, fission, elastic and inelastic scattering, (n,2n) and (n,3n) cross sections, $\bar{\nu}$ and the fission neutron spectrum. Also included are the half-life and decay of ^{241}Am, the branching of the radiative capture cross-section to form the ground and isomeric states of ^{242}Am and resonance integrals. Where possible, the evaluation is based on measured data but much use has been made of nuclear reaction theory and the systematic behaviour of the nuclear parameters of the actinides to calculate the required data where no measurements exist.

Measurements of the integral (n,γ) cross-sections of ^{241}Am and ^{243}Am were completed and are being evaluated.

10.1.2 Chemical nuclear data

A method is being developed for the measurement of complete mass-yield curves by γ-ray analysis, and its application to an examination of the effect of neutron spectrum changes on fission yields. The method has been applied to the measurement of complete yield curves for fission of ^{235}U in differing fast neutron spectra in ZEBRA. Results are now complete for two of the spectra and have shown that the method is capable of giving excellent results with much less effort than the techniques used hitherto. The measurements are being extended to other spectra and to fission of ^{238}U and ^{239}Pu.

The code GAMANAL, for computer analysis of γ-spectra, has now been given a variety of input and output options and a number of different diagnostic procedures to aid in the analysis of difficult peaks. Additionally, a code has been written to extract half-life data from γ-spectra analysed by it.

A highly sophisticated code has been written which allows a PDP-11 computer with 28 k storage to act as four completely independent 4096-channel pulse-height analysers for automatic X- and γ-ray analysis. Considerable off-line spectrum analysis is possible while counting is simultaneously in progress.

Development of the fission product yield library continues. A comparison has been made by B. F. Rider of the USA with the corresponding US evaluation and all discrepancies were subsequently reconciled. Work continues on the fitting code for satisfying the constraints that must be obeyed by a consistent set of yields and two methods are being examined for solving the required set of approximately 950 simultaneous equations, the matrix of which is very sparse. The independent yields in the adjusted sets have not previously been divided into separate isomer yields because of the paucity of experimental results. A US method for estimating the required branching ratios, as used by Rider and Meek, is being evaluated for possible use in the UK file.

The method developed for "thermal" work on the fission yield of tritium could not be applied to "fast" irradiations because this involved the irradiation of solutions of the fissile material in polythene containers. Samples of pure ^{239}Pu metal and pure ^{235}U metal have been obtained and sealed into cans made from ultra-pure aluminium. These are now ready for irradiation in ZEBRA. A number of blanks will be irradiated with the samples including cans made from readily available "reactor grade" aluminium. Samples for parallel irradiations in GLEEP are also ready. Samples of enriched ^{240}Pu and ^{241}Pu have been ordered.

10.1.3 Data file management and evaluations

The main task under this heading is the maintenance and continual improvement of the Nuclear Data Library and its associated computer codes. The most important advances in 1977/78 were the following:

(a) Evaluations have been completed for ^{233}Pa, Au (an interim version), ^{58}Ni(n,2n); and are well under way for the natural isotopes of Fe.

(b) New data files for ^{241}Am, ^{238}U, ^{59}Co, ^{149}Sm and ^{197}Au have been added to a revised version, NDL-1*, of the main library tape NDL-1. Eighteen files from the US ENDF/B-IV library have been added to the dosimetry library tape NDL-3.

(c) The computer code SIGAR, which calculates Doppler-broadened cross-sections from resonance parameters, has been extensively revised. It is currently in two versions: SIGAR-5 has been sent to the NEA Computer Programme Library, and a User's Guide is in draft; SIGAR-6 includes further improvements, such as a more accurate Doppler-broadening routine, and distant-level corrections.

(d) Studies of the theory of resonance spacings have been of much value in evaluating resonance data.

There is a substantial effort on the production, development and testing of data and methods for calculations of decay heat, higher actinide build-up and of activities and neutron sources for fuel transport and reprocessing. The main achievements in 1977/78 were the following:

(a) A new library of fission product data has been established for use with the FISPIN inventory code. The library uses recently evaluated fission yield data and the first version of the UK Fission Product Decay Data (UKFPDD-1). In fact two libraries are available, one with yield distributions relating to a fast spectrum (near 0·5 MeV) and the other to a thermal spectrum, the latter being recently recommended at a meeting of interested specialists.

(b) A new library of actinide data has also been produced for use with the FISPIN inventory code. The main differences from the previous library are that gamma-ray decay data are included, and about 30 extra nuclides are calculated. In addition, the data for all the nuclides has been reviewed, and up-dated where necessary.

10.1.4 Reactor physics data sets

For reactor physics use, the differential nuclear data has to be converted into group form by integrating over the appropriate energy spectrum. The latest fine-group cross-section set FGL5 (and the associated broad-group set FD5) was produced in 1972. The data for important nuclides in this set were adjusted to take acount of the results of a wide range of measurements made in zero-power reactors. Since then, these fine and broad-group data sets have been extended and improved, together with the codes used for treating the heterogeneous cells of ZEBRA. Recent improvements include:

(a) The MURAL code has been modified to enable calculations to be made with cross-sections for a material at different temperatures in different regions of the cell. A routine has been written to interpolate between MURAL runs at the different temperatures available in the library.

(b) For alternative fuel cycle studies, the FD5 library has been extended to include data for ^{232}Th, ^{233}U and ^{233}Pa.

(c) Early in 1978 the NEACRP organised an international comparison of calculations made for a large fast reactor (about 1200 MWe). The main properties calculated were k_{eff}, sodium-void worths, Doppler effects, power distribution, breeding gain and the reactivity worths of a central control rod. Calculations were made using both FGL5 and FD5 and the results compared with calculations made in other countries using 14 other data sets. There were some surprising differences – particularly in radial power distributions and central control-rod worths.

(d) The nuclear data requirements regarded as first priority in France, Germany, Japan, the UK and the USA have been compiled and compared with the cross-section measurement programmes of these countries and Euratom. The aim is to draw the attention of laboratories to requirements which are not receiving attention.

It is intended to make continual improvements and extensions to the data sets FGL5 and FD5 and to the processing codes used with them. A work programe is being defined for a fully revised FD6.

10.2 Experimental reactor physics

10.2.1 The BIZET programme in ZEBRA

At the time of the previous report, the experiments in the BZB assembly modelling a large power reactor core of conventional design had just been completed. A 3-month reload then followed to install BZC, the first of the two heterogeneous cores being studied in BIZET. The BZC work finished in mid-September 1978, when the lattice was re-arranged to a single-annulus configuration (BZD) which will occupy the reactor until the end of the currently-planned collaborative programme.

Two types of heterogeneity characteristic of alternative designs of large fast reactors are being studied. In BZC, small groups of fertile elements representing blanket islands of a few sub-assemblies are distributed throughout the fissile region. In BZD these fertile elements will be gathered together to form a large central blanket, equivalent to about 36 sub-assemblies, surrounded by a fissile annulus. Both assemblies are fuelled entirely with plutonium at a single enrichment of 24% (Pu/Pu + U).

The localised heterogeneity of BZC results in only a modest reduction in sub-assembly void worth relative to a homogenised version. A key factor in this reduction is that, in accident studies, voiding of the fissile elements only needs to be considered since the internal blanket groups which contribute a significant positive reactivity would not be expected to void. In contrast to the behaviour of BZC, the strong radial flux gradients caused by the large central blanket in BZD result in a marked reduction in void worth, with a zone of negative sign adjacent to the central blanket as well as that occurring (as in the conventional core) in the outer core region. A favoured design of heterogeneous fast reactor is based on a combination of these two heterogeneities, a large central blanket being the main source of the reduced sodium-void reactivity with small fertile groups distributed throughout the fissile region serving to flatten the radial power distribution (which is undesirably peaked in the simple single-annular arrangement).

The majority of the BZC measurements were made in version BZC/1 which was critical with all 16 simulated power-reactor absorber positions occupied by sodium. In addition to the obviously interesting experiments, the following were included in the programme: simulation of plutonium build-up (2% Pu/U) in a blanket group with re-measurement of some properties; gamma-ray energy deposition scans through fissile, fertile and steel regions; and Doppler spectral indices (Mn, Ta, Au) in fissile and fertile zones.

Following the BZC/1 work, the fissile loading was increased to provide an excess reactivity of approximately 2% dk to permit a series of six absorber arrays to be studied at critical for power distribution and relative worth studies.

The experiments currently being planned for the single annular core (BZD) cover the same general topics as for BZC, with particular emphasis on demonstrating the reduction in sodium-void reactivity predicted for this arrangement.

Conclusions must largely await the completion of the detailed theoretical analysis. A few pre-analysis observations may, however, be made relative to the BZB conventional core:

(a) The introduction of distributed absorbers (boron, molybdenum), simulating fission product build-up, leads to a significant increase in sodium-voiding reactivity.

(b) The introduction of hydrogeneous material (polypropylene) simulating oil ingress produces a positive reactivity change in all fuelled positions, of a magnitude influenced by the proximity of control absorbers. The results show the complex non-linear effects of hydrogen arising from competing moderation and leakage components.

(c) The spatially-corrected count-rates of detectors located in the outer radial blanket yield shutdown reactivities resulting from a series of 15 different absorber arrays agreeing well with those from in-core detectors over a range 0-26 $.

10.3 Methods development

10.3.1 The calculation of reactivity effects due to sodium voiding

Extensive studies of the sodium void effect in a CDFR of conventional design have been completed covering many aspects of physics and calculational methods, including an assessment of current ZEBRA experience, the effects of sub-assembly heterogeneity, fuel temperature, fission products, higher plutonium isotopes, control rods, approximations in energy group structure and spatial mesh, and the use of diffusion theory.

10.3.2 Sub-assembly wrapper damage and bow

Improvements have been made in the implementation of COSMOS tasks for bowing calculations. The route which is now in production use allows the PFR irradiation history to be followed in detail for each sub-assembly. Core reloads, including sub-assembly shuffling, sub-assemhly rotations and power and coolant flow changes of any type can be followed. Sub-assembly interactions are also calculated. Versions of this route are available both for following actual loadings and for planning new loadings.

10.3.3 Control rod reactivity effects

Improved methods for averaging control rod cross-sections over a lattice for whole-reactor calculations are being developed. These involve weighting with an adjoint flux. Assessments have been made of the corrections required to diffusion theory estimates of control rod array worths in CDFR to allow for the finite mesh present in the calculational models used, and of the corrections required to allow for the use of diffusion theory rather than transport theory.

Some comparisons of transport theory (S_N and P_N methods) with diffusion theory for calculating eigenvalues and reaction rates have been extended to include a study of the reaction rates near control rods.

10.3.4 Finite element methods

The collaboration with Queen Mary College on finite element methods for neutron transport has been extended to include Imperial College. Queen Mary College are working on the eigenvalue problem and Imperial College on a two-dimensional multi-group shielding code. Comparison with benchmark calculations shows that the finite element method is quick, accurate and free from anomalies such as ray effects which can be troublesome with S_N methods. Both discrete and continuous angular representations are available for the angular dependence of the flux.

10.3.5 Developments and comparisons of diffusion theory codes

The codes TIGAR and SNAP have been compared with other diffusion theory codes from overseas (mesh-centre and mesh-edge numerical techniques) through the medium of an international series of benchmarks.

The three-dimensional diffusion code SNAP has been converted for use on the ICL 2900 series computer, to a full specification, with all system facilities such as dynamic core allocation and system-independent sequencing. Additional SNAP facilities have been made available in COSMOS. These include one-dimensional slab- and cylindrical-geometry, rotational symmetry for 60° sectors and all reflexive symmetry conditions in rectangular and triangular geometries.

In addition, SNAP has been provided with neutron balance and Rayleigh quotient edits for estimating k-effective with high precision. These edits allow finite difference results to be extrapolated to infinitely fine mesh, which in turn allows the transport correction for control rods to be estimated. Proposals have been put forward for a general perturbation module to work with SNAP (and thereby with COSMOS).

10.4 Energy deposition and shielding

10.4.1 Introduction

Work has continued during the year on the assessment of multi-group data sets and this has involved some exploratory data adjustments utilising the ASPIS iron benchmark experiment. At the same time, code development has continued at Risley and Winfrith, the former concentrating on diffusion codes (SCOREM and SNAP) and the latter on the Monte Carlo code McBEND.

10.4.2 The analysis of data requirements

Following the preliminary analysis of the CDFR radial shield, in which upper and lower bounds of the uncertainty were established on the key design parameters, consideration has been given to the ways in which the multigroup neutron data can be improved to meet the current target accuracies. The most difficult problem is to determine the effect of correlations which exist between the errors in the cross sections at different energies. Correlated error files now exist in the ENDF/BIV library and a processing code (MCOVE) has been written to produce multigroup covariance matrices. Of the materials relevant to CDFR, only the file for oxygen is currently available in the UK; the effects of including these correlation data were accordingly investigated in a notional concrete biological shield – the uncertainty in dose was close to the lower bound calculated on the assumption of no correlations between group cross-sections. A compilation of 15-group covariance matrices has now been received from Oak Ridge for iron and sodium, and calculations are in hand to investigate their effect in the CDFR radial shield problem. Before reliance can be placed on these data it will be necessary to establish their accuracy in benchmark experiments. In the meantime, sensitivity studies will continue on each aspect of the CDFR shield design to determine data requirements for the limiting cases of zero and full energy-dependent correlations.

10.4.3 ASPIS experiments on NESTOR

The first set of adjusted multigroup data has been produced from the iron benchmark experiment. The scalar absorption, inelastic and elastic scattering cross-sections in each of the 100 groups was subject to adjustment in a P_3 multigroup flux calculation. The most important trend in this preliminary analysis was a general lowering of the inelastic cross-section (in data file DFN 908) by as much as 50% between theshold and 1·2 MeV, and by about 20% at higher energies. A further series of measurements has accordingly been carried out to improve the accuracy of the spectrum between about 4 and 10 MeV. The sharp monotonic decrease with energy observed in this high energy range has necessitated some refinement in measuring techniques. The analysis is now well in hand and the fine-group variance-covariance matrix derived from the experiment will be condensed to 15 groups for comparison with the available ENDF/BIV data. This will be the first independent check of the correlated error files now being developed in the US.

10.4.4 PFR shielding

The experimental programme has now been largely completed but further experiments involving demountable sub-assemblies are planned to study the leakage spectrum in the vicinity of the blanket boundary.

A new spectrum-unfolding code, SENSAK, has been developed which seeks maximum-likelihood consistency between measured reaction-rates and their predictions by the survey methods for which the accuracy may only be known approximately. The variables adjusted are: source strength, group fluxes, group detector cross-sections and counting efficiency. Provision is made for the correlation of the fluxes to be input over the whole energy range. Tests have been carried out in a simulated spectrum which can be measured directly in ASPIS using fourteen activation detectors. This code should enable the accuracy of spectrum calculations in difficult positions remote from the PFR core to be considerably improved with the aid of activation monitors.

10.5 Fuel management and general neutronics calculations for CDFR

The average discharge burn-up of the first charge of CDFR operating on a nominally continuous refuelling scheme is about half that of the fuel discharged at equilibrium. The possible ways of improving this (with significant cost benefits) have been explored. One method is to reload partly burnt-up sub-assemblies. It was found that when sub-assemblies were re-charged into their original zones (outer zone sub-assemblies into the outer zone; inner zone sub-assemblies into the inner zone) short operating periods between refuellings were obtained and it was only worth re-using about one-third of the first charge. In the second phase of the assessment, outer-zone sub-assemblies were recharged into the inner zone provided that their power did not thereby exceed the limit. A scheme has now been produced that allows all of the first charge to be re-used.

Consideration has been given to ways of reducing the rather large sodium outlet temperature variations between fresh and irradiated blanket sub-assemblies without seriously reducing the effectiveness of the blanket region. Estimates of the effects of a number of schemes have been made using existing CRACKLE calculations and some new SNAP calculations. Detailed CRACKLE calculations using new libraries of diffusion data have now been set up for a promising shuffling scheme in which fresh radial blanket sub-assemblies are loaded intially to positions in the inner ring and after several cycles progressively moved to the middle and then to the outer ring from which they are finally discharged.

The current reference design has a restrained core and an oxide radial blanket, whereas the previous calculational model represented sub-assemblies supported by spikes at their lower extremities and a carbide radial blanket. The breeding gain and other parameters have been recalculated for a 6-batch refuelling scheme, 10% peak burn-up in the core, 2% peak burn-up in the radial blanket and a 5-year limit on dwell time. The axial blankets were reduced in length from 450 to 300 mm. The breeding gain of the new design is lower by about 0·07 than for the carbide blanket design. This results from reductions of about 0·02 for the radial blanket regions and about 0·05 for the core plus axial blanket regions.

A preliminary study of the use of thorium in fast reactors has been completed. The results for the base case and four variants are summarised in Table 1. The sodium-void effect is clearly much more favourable for the ^{233}U – thorium reactor (Case 5) though this reactor is a poor breeder. The plutonium-thorium reactor (Case 4) shows a marked improvement on the overall void effect, but the maximum positive effect is only slightly better than that of the standard design of Case 1. Case 4 is one of the best of the designs as regards gain and doubling time.

TABLE 1

Comparison of thorium and depleted uranium as fertile material

Case	1	2	3	4	5
Core fissile feed	Pu	Pu	Pu	Pu	^{233}U
Core fertile feed	U_d*	U_d	U_d	Th	Th
Axial blanket feed	U_d	U_d	Th	Th	Th
Radial blanket feed	U_d	Th	Th	Th	Th
Sodium-void effect (%)	1·5	1·1	1·0	0·4	− 1·7
Maximum positive void effect (%)	1·8	1·8	1·7	1·3	0·1
Doppler coefficient (x 10^{-3})	− 10·0	− 9·8	− 9·8	− 8·6	− 11·7
Breeding gain	0·16	0·16	0·21	0·20	0·02
Doubling time (years)	39·0	37·0	29·0	33·0	−

*U_d = depleted uranium.

10.6 Economics and design assessments

Work has continued throughout the year to establish the fast reactor as an essential contributor to the future UK nuclear programme and to investigate the potential advantages of variations in design parameters. Studies are carried out on a national system basis and standardised conceptual designs have been defined. These include standard and low-void-reactivity layouts of fuel with pin diameter 5·08 mm, more advanced designs with 6 mm pins and a carbide-fuelled design with 10 mm pins.

Studies using the FROVE code made over the last three years to assess the effects of various parametric changes – in pin diameter, size of fission gas plenum, axial breeder height, fuel density, linear rating and burn-up – have been completed. In the past year, it has been shown that the optimum pin diameter is near 6 mm in nearly all cases, that uranium ore consumption by the system increases monotonically with core height from 0·8 m upwards, and diminishes with axial blanket height from 0·3 m upwards, though the change is small; after 0·4 m. Doubling the plenum size increases ore consumption by 5%.

A comparative exercise has been completed to assess the risks of plutonium diversion when plutonium is either not used or burned in thermal or fast reactors. The plutonium contents of reactors, ponds and stores were calculated for the following assumed reactor installation programme:

rising from	47 GWe in year 2000
reaching saturation at	310 GWe in year 2065,
falling from	310 GWe in year 2095,
to	zero in year 2140.

The study showed that the risks were least for plutonium burning, especially in thermal reactors, through the consumption of uranium in the latter case was unrealistically high.

The feasibility of reducing core temperature rise from 170 to 140°C, to ease thermal striping problems, has been assessed for a range of pin diameters and core heights. The optimum pin size remained near 6-7 mm, though uranium consumption rose by 50-80% because of the poor specific inventory and breeding gain of the optimum design.

The effect of out-of-reactor time in the range, 9-36 months, on uranium consumption has been studied for a limited installation programme (40 GWe in year 2000, 125 GWe in 2025) to determine the possible trade-offs in plutonium tails (1-4%) as functions of pin diameter (5-7 mm) and fuel density (70-90% of theoretical). The dependence on out-of-reactor time is very strong relative to other parameters and a one-month change in out-of-reactor time has a similar effect to a ½% change in plutonium tails or a 2% change in fuel density. To keep the UK ore requirements for this small nuclear programme below 250 kt, the plutonium tails would have to remain below 2% of total throughput, whilst the out-of-reactor time would have to remain below 12 months (at 70% fuel density) or 18 months (at 90%).

10.7 Engineering analyses

10.7.1 Intermediate heat exchangers

A more fundamental study of transient flow, temperature and stress distributions in IHXs has been undertaken in order to clarify problems and assist in design validation. A computer code IETA has been developed to represent tube, tube plate and shell geometry but requires flow pattern input. A more elaborate code ANTHEA is being developed on extramural contract by CHAM Limited. Validation of these codes will be based on PFR data and on model experiments.

10.7.2 Natural convection heat removal

A lumped-parameter computer code has been used to examine the effects of trips from part load on the ability of natural circulation to remove the decay heat from the core in the unlikely event of a loss of all pumping power. The results of this study indicate that a careful choice of the primary and secondary pump halving times is necessary to ensure a smooth transition to natural circulation.

10.8 Control and dynamics

10.8.1 Plant kinetics

Until about ten years ago, the limitations in analogue and digital computing equipment required that greatly simplified mathematical descriptions were used in the large scale simulation of nuclear power plant. Digital computer developments by way of increased storage and switching speeds have reached the point where such restrictions no longer obtain. Plant definition now seems limited by the data available and the ingenuity of the scientist or engineer. These advances in digital computer technology were foreseen and a highly detailed modular simulation for fast reactor power plant was initiated. This has since acquired the title JCBARK – a programme of Jointed Construction for Boiler and Reactor Kinetics.

So far the model consists of: reactor, intermediate heat exchanger (IHX) and boiler which are coupled together by means of simulated pipework and pumps.

The reactor is characterised by multigroup one-dimensional diffusion neutron kinetics with distributed thermal hydraulics. Due to the adoption of a novel integration algorithm, this module tested against international bench marks returned the fastest computation time with a consistent accuracy. Five different zones of water-side heat transfer are contained in the boiler simulation along with a radial characterisation of heat conduction. This level of detail enables the boiler module to be utilised for heat transfer surface design. JCBARK is the accepted reference dynamics code and collaboration with NPC and the CEGB continues with regard to its validation against other programmes and plant data.

Because the boiler calculations completely dominate the run-time of a power plant simulation, attention has recently been directed at the use of special interpolating functions to reduce the number of spatial mesh points employed. A method has been developed, and the trivial logical errors which are present in the coding are currently being systematically removed. Initial results are very encouraging, and a ten-fold reduction in the number of mesh points has been achieved with a negligible real loss of accuracy and a reduction by a factor of three in computing time. This outcome considerably improves the prospect for future developments in JCBARK aimed at evaluating asymmetric plant transients with several boiler and IHX units.

Although a steam drum model already exists as a JCBARK module, it is rudimentary and many important phenomena are incorrectly represented or totally absent. The CEGB and Babcock-Wilcox have each made available further experimental drum data on steam carry-under, draw-down and liquid pool stratification which has enabled the formulation of a very realistic asymmetric JCBARK module. It is planned to implement its FORTRAN coding in the near future.

A dynamic steam turbine model for JCBARK, which is based on the expansion of steam through choked and non-choked blade stages is nearing completion. It is planned to compare predictions of this distributed model against those of the partially lumped turbine simulation currently used in PFR studies.

At the moment, improved recording arrangements are being evolved for PFR. It is expected that their completion will approximately coincide with the above code developments, so that whole plant model comparisons can be effected against normal and fault manoeuvres in the real system.

10.8.2 Modelling laws for thermal transients

The overall aim of this work at BNL is to obtain a clear understanding of the modelling laws for the extrapolation of the results of simulant fluid tests to reactor conditions.

Water model testing in axisymmetric geometries relevant to the hot pool of the reactor has continued. The model uses the change from a pure water to a brine flow to simulate the sodium density increase produced by the fall in temperature of the flow emerging from the core following a reactor trip.

It is found that for Richardson numbers (ratio of buoyant-to-inertia effects) above a critical value the flow regime changes markedly during the transient.

Complete thermal similarity between a water model and the sodium-cooled reactor requires that the dimensionless diffusivity of heat be equal in the two flows. The experiments, one in water and the other in sodium, are designed to produce detailed data on the turbulent diffusion of heat and momentum in a recirculating flow which is characteristic of that in a reactor pool. Measurements of the mean velocities and temperatures have been completed for the water experiment. A two-channel, laser-doppler anemometer has been commissioned to allow the precise measurement of the turbulent shear-stress in the water model. A finite-element code has been developed to analyse the results of the experiments. It includes an algebraic model of turbulence and has shown a considerable success in predicting the observed flows and temperatures. The code is now being modified to incorporate a (k,ε) turbulence model and has been used to predict a simple jet flow.

10.9 Neutron flux instrumentation

10.9.1 Developments in 1978

Design changes of significance to the neutron flux instrumentation system for CDFR considered during the past year include the introduction of a core restraint system, the reduction in size of the annular fuel store, the abandonment of proposals for heterogeneous core configurations with loosely coupled regions and a reduction in the core mixed-mean outlet temperature. The overall effect of the first three has been to restrict the available choices of instrumentation system and to narrow the range of neutron flux levels at possible detector sites without changing markedly the scope of the development programme. The lower sodium temperature eases the problems of developing higher temperature neutron detectors and cables by reducing the survival temperature requirement from 625°C to 575°C, which is nevertheless still above the capability of existing proved designs.

Optimisation of the design of pulse fission chambers to achieve the required discrimination against gamma radiation has been pursued in three ways. The use of "fast" gases will probably yield only about half the desired improvement over existing argon-filled types of chamber. A halving of the inter-electrode gap to 0·5 mm could achieve a further factor two improvement. (The forthcoming high temperature irradiation trials, starting in WAGR in the Spring of 1979 will test, among other things, the practicability of operating with the smaller gap). Finally, to determine what further reduction in gamma-pulse pile-up can be achieved by more careful optimisation of filling-gas pressure and pulse-shaping circuit design, a theoretical study has begun of the pulse pile-up mechanism in ionisation chambers assuming practical pulse shapes. It is clear that the low-power-channel detectors need to be as sensitive as possible and a view has to be taken on the maximum neutron sensitivity which can reasonably be achieved.

The manufacture of high-performance mineral-insulated cables in general is now under much better quality control. However, quadraxial cable has turned out to be much too stiff to enable the experimental pulse fission chambers to be loaded into the WAGR irradiation rig. It has, therefore, been necessary to construct a thinner cable which has three outer screens in direct contact. Though this 'trilaminax' cable is easier to make and is adequate for the WAGR trials, it does not provide sufficiently good screening against electrical interference for the installed pulse-Campbell channel for which a super-screened version will be developed. Whilst there seems to be no obvious gross noise effects in mineral-insulated cables which could affect Campbell-mode operation further, more careful, cable-noise and reactor-noise spectra measurements still need to be carried out. The pulse-Campbell channels installed in WAGR have performed well and no significant difficulties have been reported.

Shutdown amplifiers using some of the circuit configurations proposed for CDFR have continued to operate satisfactorily in WAGR. In the meantime, a preliminary examination is required of the circuit techniques needed to achieve reliably the kind of trip profile pertinent to CDFR for which the neutron-flux trip level will be a function of other primary circuit parameters, such as sodium flow.

Measurements on single small proportional counters have already established adequate neutron performance but gamma response measurements have still to be carried out and clusters built and tested. Development of the associated flexible radiation-resistant superscreened cable is being pursued using a dielectric of polyethersulphone (PES). The feasibility of extruding PES to the sizes and thicknesses required is currently being explored and high temperature radiation testing of dielectric samples also needs to be carried out.

10.9.2 Shutdown monitoring

A proposed new method of monitoring shutdown reactivity margins during refuelling involves the measurement of neutron flux at several locations around the core before and after the insertion of a fresh sub-assembly. It is argued that the ratio of counting rates before and after insertion can be used as an indication of the subcritical reactivity margin and possibly to detect inadvertent removal of a control rod. (The emphasis is put on fresh sub-assemblies so that the neutron source term need not be recalculated at each stage). The nucleonic and electronic implications of implementing an adequately reliable system of measurement are to be pursued jointly with NPC.

11. ABBREVIATIONS USED IN THIS REVIEW

AEA	United Kingdom Atomic Energy Authority
AEEW	Atomic Energy Establishment, Winfrith
AGR	Advanced Gas Cooled Reactor
AERE	Atomic Energy Research Establishment, Harwell
BNL	Berkeley Nuclear Laboratories (CEGB) Gloucester
CDFR	Commercial Demonstration Fast Reactor
CEGB	Central Electricity Generating Board
CFR	Commercial Fast Reactor (The future series of LMFBR following CDFR)
COVA	Code Validation Experiments (Containment studies)
CSNI	Committee on the Safety of Nuclear Installations
DFR	Dounreay Fast Reactor
DNE	Dounreay Nuclear Power Development Establishment
GEC	General Electric Company
IHX	Intermediate Heat Exchanger(s)
MFCI	Molten Fuel/Coolant Interaction
NII	Nuclear Installations Inspectorate
NPC	Nuclear Power Company
PFR	Prototype Fast Reactor, Dounreay
PIE	Post Irradiation Examination
RNL	Risley Nuclear Power Development Laboratories
SNL	Springfields Nuclear Power Development Laboratories
SRD	Safety and Reliability Directorate, UKAEA
TIG	Tungsten-Arc Inert Gas (Welding)
WAGR	Windscale Advanced Gas-Cooled Reactor
ZEBRA	Zero Energy Breeder Reactor Assembly