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THE BRITISH NUCLEAR ENERGY SOCIETY

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Submission of Papers for the Journal

THE Society accepts papers concerned with developments in nuclear power technology and in a wide range of related topics in nuclear science and engineering. Contributions should be of original work or, where appropriate, of critical surveys indicating developments in a particular field of study.

Papers, together with illustrations, should be submitted in duplicate to the Editor for consideration. A leaflet is available on application to give guidance to authors on the preparation of papers.

Letters to the Editor are welcomed and can be given very prompt publication. These may report important new facts or discoveries in advance of a detailed paper, briefly discuss some nuclear topic of interest to Members, or comment on papers which have appeared in the Journal.

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The Journal of THE BRITISH NUCLEAR ENERGY SOCIETY

Correspondence is welcomed from members on any of these subjects

Principal Contents

Meetings

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comment

The Rt. Hon. Anthony Wedgwood Benn, MP

(*Minister of Technology*)

I WAS interested to read in Comment, in the July issue of the Journal, that the division between Government and Industry in the nuclear field is not so sharp in some European countries as in the UK. The relationship between the public sector and the private sector in this country is based on an extremely complicated network of interests, and the reorganization of the nuclear industry, on which we are now embarking, will involve a good deal of adjustment in this relationship.

At present there are three consortia and their parent firms the plant manufacturers, the Atomic Energy Authority, the Central Electricity Generating Board and the South of Scotland Electricity Board. The new structure envisages the establishment of two new design and construction organizations in place of the three commercial consortia and the Atomic Energy Authority's design teams, and the aim is to secure close integration between these two organizations and the manufacturers of the main constituent elements of nuclear 'islands'.

The new companies would also need to work in close conjunction with the highly successful fuel organization set up by the AEA, which is already operating on a commercial basis. The Government has decided that to make this co-operation more effective in the exploitation of reactor systems, it would be advisable to establish the Authority's fuel business as a publicly owned company under the Companies Acts with the initial share capital wholly subscribed by the Government. In order to emphasize the interdependence of fuel and reactor design and supply the Government intend that the fuel company should take up and hold a minority shareholding in each of the two design and construction organizations. Since the establishment of the fuel company will require legislation, it is proposed that initially the Government shareholding should be taken by the AEA.

In addition, when the proposed new industrial structure takes shape, the Government have it in mind to set up an Atomic Energy Board on which the AEA, the design and construction companies, the fuel company and the generating Boards would all be represented. The Board would concern itself with the composition and objectives of research and development programmes (to which the two new design and construction organizations and the fuel company would be expected to contribute), the co-ordination of activities in the export field and major matters of policy.

In view of these arrangements and the legislation proposed, some modification in the organization of the AEA will be needed. This must be designed to ensure that the most effective possible use should be made of this great national asset, not only in support of the new organization of the nuclear industry, but also in conjunction with the Government's own research establishments in the development of industrial technology.

The Government's objectives, which these proposals are designed to further, may be summarized as follows:

(a) to make the best possible use of all the existing resources in this field, cutting out overlapping and duplication, of which there manifestly is some;

(b) to allow those who have worked in the Atomic Energy Authority full scope for carrying their work forward into the exploitation and sale of the systems which they have developed;

(c) to get the maximum possible advantage from technical standardization, coupled with the most effective design competition in engineering detail and construction methods;

(d) to try to link and co-ordinate the efforts in such a way as to relate reactor systems to the fuel elements and reprocessing business at which the AEA has excelled, both technically and commercially, through its fuel production group;

(e) to create an organization which permits the sort of international industrial links which will be of critical importance in all sectors of advanced industry and not just in atomic energy;

(f) to do this with a special eye upon the future of the European nuclear industry in co-operation with our partners in Europe;

(g) to increase the emphasis of our national effort on exploitation and see that future nuclear research is guided and shaped more directly by the needs of the market at home and abroad;

(h) to establish a creative partnership between the public and private sectors as far as possible by reaching a consensus of agreement so as to allow all the other objectives I have described to be achieved.

The creation and growth of this industry is one of the major technological developments of the second half of the twentieth century: the AEA is one of the great success stories of engineering and science. I think one can reasonably regard the award in 1965 of the Dungeness B contract to an AGR design as marking the end of the exploratory phase in the development of nuclear power: it demonstrated that nuclear power had become the most economic source of electricity generation and that British reactor designs were fully competitive. Today almost half the world's total operating capacity for generating electricity from nuclear energy is in Britain, while the Authority's fuel service has built up a substantial international business.

Why then, it may be asked, are the proposed changes thought necessary? The answer, quite simply, is that we are determined to exploit to the full the success which the AEA engineers and scientists have achieved and to do all we can to build up a flourishing export business on the basis of the research and development programmes in which the nation has invested hundreds of millions of pounds. The picture up to now is that of a typical British problem of brilliant research, generously financed, and an inadequate structure for its exploitation. We badly need a successful sales effort abroad, which could do more to support the activities of the engineers and the industry than any action that the Government itself can take. The emphasis now must shift to exploitation, particularly in the field of exports—where the record has been disappointing—if we are to reap the harvest which should result from the endeavours of those engaged in the Authority and the industry since the war.

The problems of the nuclear industry are particularly complex, but some of its features have close parallels with those in other industries whose interests the Ministry of Technology seeks to promote. As is so often the case, the first problem to be tackled, with the help of the Industrial Reorganization Corporation, is the basic structure of the industry: we must have industrial units strong and vigorous enough to carry through the costly research and marketing needed to support their sophisticated production and to sell their products against international competition. The two new design and construction organizations, which will be primarily responsible for the exploitation of British reactor systems at home and abroad, should be well able to stand up to international competition and should be capable of a powerful effort in overseas markets. So far as the Ministry of Technology is concerned, our aim is to gear our research efforts to the needs of industry and to help industry to improve its competitiveness, its productivity, and its profitability, so as to secure the export orders we need to help free us from the balance of payments restraints which have hampered our economic growth.

I hope that a reorganized and revitalized nuclear power industry will help lay the foundations for Britain's economic expansion in the 1970s. And the 1970s begin in little over a year's time.

BNES Newsletter

Gas-lubricated bearing technology

ALTHOUGH gas-lubricated bearings have existed for over 70 years, the major developments in both theory and practice have occurred during the last 15 years, prompted initially by their application to nuclear devices. Gas-bearing circulators are used in the DRAGON reactor at Winfrith Heath, and although they have not been used in the magnox stations, their use is being seriously considered for future gas-cooled reactors both with advanced thermal and fast reactor concepts. Direct cycle gas turbine systems are also being considered in conjunction with high efficiency HTR reactor systems, and gas bearings may well be used in such applications.

The need for both gas circulators and liquid metal pumps which would not contaminate the working fluid in the system resulted in the study of gas bearings as part of the nuclear research programme. Freedom from leaks and extended bearing life are additional features which are important in many nuclear applications. The development of units with gas bearings by the United Kingdom Atomic Energy Authority and other organizations in Britain and abroad has led to over 60 different types of circulators being made. Sizes of units vary up to 250 hp with circuit gases such as helium, nitrogen and carbon dioxide. Manufacturing and design techniques were developed so much during these years that the design of very large units, say 50 MW, does not now seem unreasonable.

About nine years ago the characteristics of self-acting bearings of long life, little vibration and no contamination were recognized as having another important application, namely to precision gyros. Not only have gas bearings increased the usable life of a gyro to many times the 1000 h of a ball bearing unit, but, by virtue of the constancy of the characteristics of stiffness and rotational accuracy, the overall accuracy of the gyro unit has been increased by an order of magnitude.

The work of the UKAEA in the gas-bearing field was not confined to self-acting bearings, however. Externally pressurized bearings, which are similar in principle and properties to hydrostatic liquid bearings, were developed and applied to a number of research devices. In recent years this type of bearing has been applied to many industrial machines as well as to research equipment. Air bearing dental drills, turbine flowmeters, grinding spindles and roundness machines are a few of the devices which have been developed by industry from the original work done by Shires, Robinson and Sterry in the UKAEA.

The impact of gas bearings on the machine tool industry is the most outstanding example of what gas bearings can do. Although the bearings were initially applied to grinding machine wheel spindles to take advantage of the low friction and thus low temperature rise characteristics, the greater accuracy of rotation, long life and high speed capability, compared with those of conventional bearings, have led to surprising improvements in overall performance of grinding machines. One of the largest grinding machine manufacturers in Britain is now offering air-lubricated bearings on all its machines.

Another similar large production air-bearing device is the high-speed turbine-driven circuit board drill which is manufactured by a British company specializing in the production of gas-bearing machines.

From the early stages of gas-bearing technology which, in Britain, was mainly in the hands of the UKAEA, a few of the developments based on their work have been mentioned. Other forms of bearing have been investigated, particularly in the USA, and are now being applied.

A form of bearing which is peculiar to gas lubrication is called the squeeze film bearing. It is a bearing which is unlike the self-acting and externally pressurized bearings in that it can support loads even at zero speed without an external supply of gas. By arranging for one of the bearing surfaces to vibrate at very high frequencies by means of an electromagnetic or piezo-electric system, a volume of gas is, in effect, trapped in the bearing clearance. A net positive load capacity is obtained which can be simply explained by applying Boyle's law. Although the load capacity is very small (of the order of a few lb/in^2), the bearing is finding application to things like gyros and space machinery where the advantages of low friction and no contact are required, and where an electrical supply is more convenient than a gas supply.

The foil bearing, which was originally demonstrated as an oil-lubricated bearing in 1953, is now being used commercially as a gas-lubricated bearing in computers and for very high speed rotors. The tape transport mechanisms in computer magnetic tape memories have been adapted, using the principles of gas bearings, so that the tape or foil is prevented from contacting the guides and magnetic heads by a very thin film of gas. External pressurization has also been applied to assist during start/stop cycles.

Probably the most important subject of development in gas lubrication at this time is the application of steam as a lubricant; this is of special interest in the turbo-machinery of power generators. Several research centres and companies are engaged in overcoming the basic problems of using a condensing vapour in forms of externally pressurized bearings. It is hoped that when these are solved we will have a bearing system which can be widely used for bearings required to use a process fluid which is a condensing vapour. Also the ease with which high pressures can be obtained by

evaporation of liquids can lead to other applications where high pressures are required to achieve high load capacity and stiffness.

Gas-bearing technology has both expanded and developed considerably from the early semi-empirical work done in the UKAEA. There are in existence hundreds of papers and several books[1] on the subject and regular meetings and symposia[2] are held throughout the world.

In Britain a group of gas-bearing technologists meets regularly for informal technical discussions, and the fourth of a series of biennial symposia will be held in the University of Southampton in April 1969.

To make better use of all the research and experience which has been gained in gas-bearing technology through the years, Southampton University has also set up a new Gas Bearing Advisory Service within its Department of Mechanical Engineering. This is intended to provide a design and consultancy service to industry and seeks to disseminate information on gas bearings through short courses and by lectures and demonstrations in industry.

Thus gas-lubricated bearing technology has grown quickly during the last 15 years from its rebirth with the UKAEA, and has blossomed into a very active and useful engineering technology.

References

1. GRASSAM N. S. and POWELL J. W. (Eds.) *Gas lubricated bearings*, Butterworths, 1964
2. Symposium on gas bearing design methods and applications, Southampton University, 25–28 April, 1967

New Zealand

NUCLEAR power is now being seriously considered in New Zealand. Its Power Planning Committee has recently recommended that nuclear power be included in the next 10-year development programme. The Committee suggests that the first station should consist of four reactors of 250 MW(e), and that one reactor should be commissioned each year from 1977. New Zealand has a number of possible sites, and areas around Wellington and Auckland are being studied. These proposals depend, however, on the availability of natural gas, and if new finds of natural gas are made the nuclear proposals would probably have to be substantially modified.

DRAGON

THE Foreign Ministers of the countries of the European Community have approved the Commission's proposal for the partition of costs of DRAGON, the high temperature gas-cooled reactor, at Winfrith. The UK proposal that it should carry 46·8% of the total project costs from 1 January, 1968 to 31 March, 1970 was accepted by the Commission. This compares with the previous UK contribution of 40·8%. The total project costs of this latest agreement are estimated to be £4·45 million.

Rumania

THE Rumanian Government has now signed an agreement with the IAEA in which it undertakes not to use a nuclear facility and its associated fuel for military purposes. This safeguard agreement will now enable Fairey Engineering Ltd to complete its £100 000 contract for a HELEN FI assembly and the UKAEA to supply the fuel.

The Institute of Atomic Physics at Magurele, 15 km from Bucharest, has previously based its reactor research programme on the VVRS–C reactor supplied by the USSR. This 3 MW H_2O tank type reactor with a peak thermal flux of 3×10^{13} n/cm^2/s first achieved criticality in July 1957. The new facility is a subcritical assembly for training purposes and basic reactor physics experiments. In its simplest form only natural uranium and light water can be used, but extension to enriched fuel and other moderators is not excluded. The assembly will permit measurements of material buckling, thermal fine structure, and the effect of control absorber arrays. Instrumentation is limited to that required by the neutron source generator, since there is no criticality hazard with this subcritical system.

Rumania has substantial natural reserves of oil and brown coal, so that the introduction of nuclear power will be based on its economic merits compared with fossil fuels, and consideration of foreign exchange problems.

Argonne National Laboratory

RECENTLY the Annual Report for 1967 of the Argonne National Laboratory has been published. Like many other American laboratories the activities of the ANL cover a wide field, ranging from reactor technology and the physical sciences to high energy nuclear physics and environmental studies.

Fast reactors

The Argonne National Laboratory has always been interested in fast reactors, and work in this field still occupies a substantial part of its effort. It operates the Experimental Breeder Reactor II (EBR II) at the National Reactor Testing Station, Idaho, which continues to play a major role in the US programme for the development of fast breeder reactors. It is the only facility for the fast neutron irradiation of fuels and materials for the AEC's liquid metal fast breeder programme. At the end of 1967 the reactor contained some 22 sub-assemblies containing 305 capsules, sponsored by the Argonne Metallurgy Division and about half a dozen other laboratories and companies involved in fast breeder reactor development. Supporting the irradiation work are the basic reactor physics studies using the zero-power reactors ZPR 3, 6 and 9. ZPR 3 in particular has been employed in specific studies of the core design for the Fast Test Reactor (FTR) which will be constructed at Hanford, Washington, to study plutonium-fuelled systems, and Argonne is constructing at Idaho a new critical facility Zero Power Plutonium Reactor (ZPPR), which ultimately should allow measurements on reactors as large as 14 ft × 14 ft × 8 ft. By the end of 1967 the buildings for this assembly were nearly complete.

Considerable work is also done on fast reactor safety studies mainly using liquid sodium, to study the problem of superheat and sodium expulsion. Acoustic detectors are being developed which should be able to detect sodium boiling, and lithium niobate, which can function as acoustic sensing element at over 1000°C, has been tentatively chosen for study.

Of interest is the laser sampling technique of reactor fuels. As developed at the Argonne, the laser beam is used for obtaining extremely small samples of irradiated fuel by vaporization. Fuel samples as small as one ten billionth of a gram, much smaller than from conventional means, can be obtained. This enables detailed information on the microstructure of the fuel to be obtained. Interesting results have been obtained on fission product distribution and the technique will undoubtedly be most useful to metallurgists. The report points out that this laser sampling technique should also be useful in a variety of other fields, such as solid state sciences, biology and medicine, if suitable analytical methods such as activation analysis and mass spectrometry can be applied.

Computing

The use of high speed computers has become an essential tool for those concerned with reactor design. In 1967 the laboratory installed an IBM/360 computer to replace their CDC 3600. The new machine uses a model-75 processor as a 'number-cruncher' with a model-50 processor to deal with peripherals and housekeeping. A larger memory than any previous machine is used in a two-level hierarchy, and there are 16 discs, a drum, two data cells and six magnetic tapes.

Considerable attention has been paid to convenient communication with users. Computer information can be displayed on nine graphic display screens, one of which is equipped with a light pen. A conversational timesharing software system is being developed to enable 15 users at electric typewriters access to the computer simultaneously. This is done by allocating each user in turn a 'time-slice' of a few hundredths of a second, and the system is expected to have a response time of 2 s, sufficiently rapid to make desk calculator simulation possible.

There will also be a remote access data system to provide users with input/output facilities at remote locations. Direct sampling to experimental equipment will also be allowed. A scheme of this type was used with the CDC 3600 computer, which was linked to the small digital computer used for data acquisition on the Tandem Van de Graaff accelerator. The use of small computers in this manner is finding increasing application in the whole field of experimental science, and the ability to communicate with the /360 is a most necessary feature, permitting major data analysis to be initiated from the experiment, in some cases using the results to control the future course of the experiment.

Some work is also in progress in devising new computer memory systems. The possibility is being explored of using photographic plates to store binary information using holographic techniques. By using the coherent light from lasers, three-dimensional pictures can be recorded on photographic plates in the form of diffraction patterns. When reconstructed with a laser beam, the image of the original subject appears to have depth, so that it can be viewed from different angles just like the original. By applying this method to binary information in the form of bright and dark spots, the Argonne computer scientists expect to pack 16 000 000 binary bits on a 5 in. square photographic plate. Applications for this development are limited to read-only store, since it takes 4 h to produce the photographic plate, but standard programmes and system software stored in this manner are accessible in a few microseconds with a laser beam.

High energy physics

The Argonne Laboratory is a major centre for the study of high energy nuclear physics, in association with many universities and colleges. The experimental programme is centred on the 12·5 GeV Zero Gradient Synchrotron (ZGS), some 22 experimental runs being carried out during the year. In one of these a new particle was discovered, the Y* with energy of 1680 MeV, from the analysis of 400 000 photographs of events in the 30 in. bubble chamber.

The development of experimental equipment for this work continued during the year. The proton beam intensity from the ZGS was increased by a factor of three, and very good short-term beam stability was achieved. Work also started on the construction of the world's largest bubble chamber, with a diameter of 12 ft. Hydrogen, deuterium, or hydrogen/neon fillings can be used. A superconducting magnet will provide an 18 kGs field for the chamber, the design being based on techniques pioneered for the 10 in. helium bubble chamber in 1966. Taking accurate stereoscopic photographs of tracks in this very large system presents further problems; special camera lenses with 140° angle of vision have been designed, and four of these will be used to cover the maximum volume of the chamber.

Environmental studies

The effect of radiation on living organisms has been studied for many years at Argonne. The formation of antibodies which provide a high degree of immunity to the introduction of foreign matter is being studied, together with the effect of radiation doses. The present programme is aimed at untangling the basic mechanism underlying the immunity provided by antibodies, which, if successful, will have wide application in such fields as control of infectious diseases and organ transplantations. Work is also being carried out on genetic defects arising from chemical deficiencies, the mechanism of fat transport and the mechanism of tumour formation. The techniques which have been developed in nuclear research find wide application in these studies. Not only are radioactive tracer techniques in constant use, but computer and instrumentation methods developed in the exploitation of nuclear energy are also of great value in these studies.

This work has now been extended to include public health problems in collaboration with local and government authorities. In Chicago the pollution of the atmosphere by coal and oil-fired plant has become a serious problem. The initial aim is to develop a computer program to predict the level of sulphur dioxide from the known sources of pollution and the meteorological conditions. This will allow a pollution warning system to be operated, and eventually it will indicate the minimum cost strategy for reducing pollution to a tolerable level.

The distribution of atmospheric lead from motor car

exhausts and its absorption by plants is also being studied. It has been found that the relative absorption from the air and the soil varies considerably from one plant to another, and the evaluation of toxicity in food arising from the lead content of car exhausts will require the study of many types of plant.

Conclusion

It is clear from the wide and expanding range of topics being pursued that the Argonne National Laboratory is continuing its very active role. The expanding collaboration with university scientists, industrial organizations and Federal and State agencies is enabling full advantage to be taken of the fall-out from nuclear research. With the total staff stabilized at about 5400 and an annual budget of $100·3 million, the laboratory is maintaining its position in the forefront of nuclear research.

Development of nuclear power in Germany

German scientists played a leading part in the international speculation following Otto Hahn's discovery of fission late in 1938. Within a month, Hahn and Strassman submitted a second paper to *Naturwissenschaften* postulating that the fission process would release further neutrons, making possible a fission chain reaction. In April 1939 this was confirmed experimentally by Joliot, Halban and Kowarski. Official interest in these developments was rapidly forthcoming. Professor Joos at Göttingen wrote to the Reich Ministry of Education drawing their attention to the possibility of an energy-producing reactor, and a first meeting with university physicists was held on 29 April, 1939.

A parallel approach was made by Professor Harteck of Hamburg on 24 August, 1939, in this case to the War Office, pointing out the possibility of making explosives based on fission which would be many orders of magnitude more powerful than conventional ones. Official thought was further stimulated in June by a semi-popular article in *Naturwissenschaften* by Flügge, an assistant to Otto Hahn, which outlined in layman's language the immense energy released by fission. During the summer, the Army Ordnance Department considered these developments, and by the time war broke out, only Germany had a military office devoted exclusively to the military applications of fission.

On 16 September 1939 the Army Ordnance Department convened the first of a series of meetings to discuss fission devices. A programme of basic measurements was initiated to measure cross sections and the properties of moderators; Professor Heisenberg was commissioned to investigate the theory of neutron chain reacting systems, and the first work on the separation of ^{235}U was started. During the succeeding months this secret programme derived considerable benefit from the work in the USA which continued to be freely published until June 1940. With no large accelerator of their own, the experiments on the Berkeley cyclotron which confirmed the existence of plutonium were of particular value, and provided an alternative to the separation of ^{235}U.

During 1940 a vast quantity of uranium compounds became available when the Union Minière stockpiles in Belgium were overrun. Although only a small quantity of heavy water was available, the Norwegian plant at Vemork was now in German hands, and plans were made to expand the output. Various experiments were designed to prove that a chain reaction was possible; by Harteck at Hamburg in June 1940 using UO_2 and solid CO_2; by Wirtz at the 'Virus House' in Berlin during December 1940, with UO_2 and paraffin wax; and by Bothe at Heidelberg using light water and UO_2. None of these assemblies showed positive neutron multiplication, but they provided a good deal of incidental information on the behaviour of neutrons, and the conviction grew that with heavy water a chain reaction could be established using natural uranium. In the autumn of 1941, Heisenberg and Döpel, using the first heavy water from Norway, made careful measurements at Leipzig on a 75 cm diameter sphere of heavy water and UO_2, and concluded that there was positive neutron production.

At this time the German programme was distinctly ahead of the work in the US, but the seeds of failure had already been sown. An erroneous measurement of the thermal diffusion length in graphite by Professor Bothe at Heidelberg led to the conclusion that natural uranium/graphite systems could not sustain a chain reaction. The German effort was therefore devoted to heavy water as a moderator, and the supply of this material proved to be a continual bottleneck. Perhaps the most serious deficiencies, however, were in the overall management of the German programme. The scientific effort was fragmented between several independent groups, competing for materials and duplicating experiments. The military authorities were never convinced of the value of the work, and the debate on ^{235}U separation versus plutonium production in a reactor led to neither line being given priority. There was no practical single-minded guiding hand to direct the programme. Thus, despite the flying start in 1939 and the excellent theoretical work of Heisenberg, poor management together with the wartime shortages during the later years resulted in a very slow rate of progress.

The last wartime attempt to establish a chain reaction took place at Haigerlock in March 1945, using materials hastily transferred from Berlin. A graphite-lined tank was placed in a water-filled pit, with 664 5 cm cubes of uranium metal suspended in a central cylindrical cavity. When all the 1500 kg of heavy water had been pumped in, the system was still subcritical, and a further 750 kg was estimated to be needed for criticality. It was not available. The first phase of development was over.

During the immediate post-war years Germany had more urgent problems than the utilization of nuclear energy. Moreover, the Allied government of occupation passed laws forbidding the pursuit of these matters, and this ban was continued even after the Federal Republic was formed in 1949. However, during the early 1950s the whole world began to realize that the exploitation of nuclear power might soon become an important commercial activity. In the United States the declassification of many reports on research reactor technology, the demonstration of electricity generation by a reactor, and the Atom for Peace programme in 1953 all quickened German interest in restarting nuclear research. Heisenberg's Max Plank Institute at Göttingen became the centre of development. A design study for the first German reactor was started, and there was even some

Table 1: *Research and training reactors*

Reactor	Location	Type	Power	Critical	Financed	Comment
FRM	Munich Tech. Univ., Garching	H_2O pool	1 MW	Oct. 1957	Land Bayern	AMF Atomics design
FRF	Univ. Frankfurt	Aq. homogen.	50 kW	Jan. 1958	Land/Farbwerke Hoechst	Atomics International L54 type
BER	Hahn–Meitner Institute, Berlin	Aq. homogen.	50 kW	July 1958	Land/Bund	A.I. L54 type
FR2	Karlsruhe	D_2O/Nat. U	12 MW	Dec. 1959	Industry/Land/ Bund	—
FRG	Geesthacht	H_2O pool	5 MW	Oct. 1958	Bund (60%)/ Land	Babcock and Wilcox design
MERLIN	Julich	H_2O pool	5 MW	Feb. 1962	Land/Bund	TNPG design
DIDO	Julich	D_2O	10 MW	Nov. 1962	Land/Bund	Identical to Harwell DIDO, now 15 MW
SAR1	Garching	ARGONAUT	1 kW	June 1959	Siemens	—
SAR2	Karlsruhe	ARGONAUT	10 W	Jan. 1963	Bund (75%)/ Land	—
AEG–PR–10	Gross-Welzheim, Bavaria	ARGONAUT	10 W	Jan. 1961	AEG	—
FRMZ	Maing Univ.	Triga MKII	100 kW (250 MW pulsed)	—	Bund	G.A. design
SUR–Munich	Garching	Solid homogeneous	0·1 W	Feb. 1962	Siemens	20% U/polythene
SUR–Berlin	Berlin Univ.	Solid homogeneous	0·1 W	July 1963	Bund	Siemens design
SUR–Darmstadt	Darmstadt Tech. Univ.	Solid homogeneous	0·1 W	Sept. 1963	Land Hessen	Siemens design
SUR–Stuttgart	Stuttgart Tech. Univ.	Solid homogeneous	0·1 W	1964	Bund/Land	Siemens design
SUR–Aachen	Aachen Tech. Univ.	Solid homogeneous	0·1 W	—	—	Siemens design
FRMB	Brawnschweig	H_2O tank	1 MW	1966	Bund	Deutsch Babcock and Wilcox
CFG	Geesthacht	Solid homogeneous	100 W	May 1964	GKSS	For KNK design
STARK	Karlsruhe	Fast/thermal	—	Jan. 1964	—	Modified Argonaut with central fast core
SNEAK	Karlsruhe	Fast critical assembly	—	Dec. 1966	—	—

oblique experimental work with neutron pulses from an accelerator.

After high-level negotiations, the declaration of sovereignty was issued by the former occupying powers on 5 May, 1955. The Federal Republic was free to engage in reactor physics and build reactors. In August, a small group of delegates attended the first Geneva conference, where most of the secrecy concerning peaceful uses of nuclear power disappeared. Optimism concerning cheap atomic power abounded. The second phase of development in Germany was beginning.

Although a Federal Ministry of Atomic Affairs was established in October 1955, it was the Länder, with their considerable degree of autonomy within the Federal system, who provided the major initiative in providing research reactors for the universities and technical colleges. Bavaria provided the FRM swimming pool reactor for Munich Technical University at Garching. Land Hessen, with the support of the local chemical industry, built the homogeneous aqueous FRF reactor at Frankfurt University, and a similar reactor was also built at the Hahn–Meitner Institute, Berlin. On the north coast, Hamburg, Bremen, Schleswig-Holstein and Lower Saxony founded GKSS at Geesthacht near Hamburg with the aim of developing ship propulsion, and the 5 MW FRG H_2O pool type reactor was procured.

All these reactors were of US origin and went critical in 1957–58; however British reactors were selected by Land Northrhine-Westphalia for their Nuclear Research Establishment at Julich where the MERLIN light water and DIDO heavy water reactors achieved criticality in 1962.

The FR1 graphite research reactor design started at Göttingen was supported by German industry, and with financial support from Land Baden-Wurttemberg and the

Bund, Karlsruhe was founded as a national laboratory. The design evolved into the heavy water FR2 reactor, similar to NRX, which achieved criticality in December 1959. An important aspect of the Karlsruhe laboratory was that it should act as a training centre for young engineers seconded from industry. Some low-powered research reactors were also obtained for several universities for educational purposes, and both Siemens and AEG financed ARGONAUT reactors. Table 1 summarizes the German research reactors.

The promotion of scientific and basic technical research was regarded as largely the province of the universities and the reactor research establishments financed by public funds, but the building of prototypes and commercial stations devolved mainly on industry and the utilities. Although the Federal Government and the Länder owned a substantial shareholding in most utilities, these had developed a tradition of independence from public direction. The German power programme was not therefore directed by central government, but developed on a commercial basis. Thus most of the prototype power reactors are financed by industry and/or the utilities, unlike the US and UK development. Similarly, the rate of installation of nuclear plant depends on commercial considerations, rather than national policy. The total capacity likely to be installed by 1975 is currently estimated to be 5000 MW(e), though the situation could change significantly under the influence of economic pressures.

As with the installation of research reactors, the first power reactors depended largely on systems developed overseas. The two major construction companies are AEG and Siemens, and they arranged to become licensees of GE and Westinghouse respectively. Also BBC, the German associate company of the Swiss Brown Boveri, have a co-operation agreement with APC in the UK for the construction of advanced gas-cooled reactors, and Demag with Atomics International have formed the joint Interatom company. Other firms of prominence in reactor construction are Krupp and the Deutsche Babcock and Wilcox.

In parallel with the research reactor programme, discussions took place in 1956–57 between the leading personalities of German industry, and later with the Federal Government, on a co-ordinated First Nuclear Programme. The outcome was a decision in principle to construct five stations each of approximately 100 MW(e) covering the full range of reactor types thought to be of commercial interest. The design work for this 500 MW(e) programme was carried out with some financial support from Federal sources, but decisions on construction were delayed, both by reticence on the part of the utilities and by the advances in reactor design during this period. The first power reactor to achieve criticality was not included in this initial programme; two utilities, RWE and Bayernwerk, jointly financed the 15 MW(e) Kahl/Main BWR. Eventually the 500 MW(e) programme led to the 100 MW(e) KKN CO_2-cooled heavy water pressure-tube reactor at Niederaichbach and KWL reactor at Lingen, a BWR with oil-fired superheat. A further experimental reactor was also initiated predominantly by the utilities; this was the 15 MW(e) AVR, a pebble bed HTR cooled with helium.

By 1960 it was clear that a revised reactor programme was needed, and discussions between the construction companies and the Federal Government yielded a modified programme. This has led to the construction of three reactors so far. The experimental HDR reactor at Grosswelzheim, a 25 MW(e) BWR with nuclear superheat, is due to achieve criticality this year. At Karlsruhe the construction of the 19 MW(e) KNK reactor is well advanced; this is a sodium-cooled system with zirconium hydride moderator. A further HTR of 25 MW(e) with a closed cycle gas turbine has been ordered for Geesthacht.

Apart from these essentially experimental systems, two other projects were initiated during this period. A full scale BWR station was constructed at Gundremmingen on the upper Danube. This was a logical development from the Kahl/Main project, and it derived great benefit from the experience with the earlier experimental reactor. The second system was the MZFR at Karlsruhe. Again this was a logical extension of an earlier development, following from FR2. Although developing 50 MW(e), this is really a multi-purpose research reactor, which can be used for testing new fuel designs under realistic conditions. This heavy water PWR went critical in 1965; some difficulties have been experienced with excessive D_2O losses, and a ton of D_2O was lost as steam in a refuelling incident.

The post-war period up to 1963 had seen a tremendous build-up of nuclear power technology from the first post-war experiments of Heisenberg's group at Göttingen. By judicious buying in of know-how from abroad a substantial number of projects had been started, with most soon due to come to fruition. However, to some extent the rate of development of fully viable nuclear power systems had suffered from fragmentation of effort: weak central direction combined with the complicated interaction of industrial interests with the utilities and the somewhat unco-ordinated support from the Länder had spread the available resources over a very wide range of projects. From some aspects the method of development shows remarkable similarities with the 1938–45 phase. The formulation by the Federal government of its Second Nuclear Programme for 1963–67 marked the third phase of German development: although still strictly only advisory in nature, commercial interests recognized the need for better co-ordination of a national programme. Also the lines of development were influenced appreciably by the £300 million from Federal funds devoted to nuclear energy during this five year period.

The Second Nuclear Programme was largely concerned with the consolidation of projects already started. Extra effort was devoted to fuels, including the work at Karlsruhe on a first German reprocessing plant (WAK) now nearing completion. In 1967, two major nuclear power stations were ordered by the electricity undertaking with a total power of 1200 MW(e), bringing the total generation from nuclear sources by the early 1970s to over 2000 MW(e).

At the beginning of this year the Federal Government announced its Third Nuclear Programme covering 1968–72. This is a logical continuation of the Second Programme, but with a greater concentration on a small number of key projects. The programme continues to rely on the co-operation of commercial interests, the Länder and the universities. It assumes that by 1980 the installed capacity will be 100 000 MW, with nuclear stations accounting for some 25% of this total, implying a rapid acceleration in the rate of nuclear construction.

It is anticipated that the investment in light water reactor technology will continue to be exploited, particularly in view of the success of these systems in the US. Further improve-

Table 2: *Power reactors*

Reactor and Location	Type	Output	Critical	Designer	Comment
Kahl/Main	BWR	15 MW(e)	Nov. 1960	AEG with GE	Experimental
AVR, Julich	Pebble bed HTR	15 MW(e)	1964	BBC/Krupp	Experimental
MZFR, Karlsruhe	D_2O-PWR	50 MW(e)	Sept. 1965	Siemens	Multi-purpose research reactor
KRB, Gundremmingen	BWR	237 MW(e)	1966	AEG/IGEOSA	Owned by RWE and Bayernwerk
KWO, Obrigheim	PWR	300 MW(e)	1968	Siemens	Ordered by group of utilities
HDR, Grosswelzheim	BWR	25 MW(e)	Autumn 1968	AEG	Nuclear superheat, experimental
KWL, Lingen	BWR	240 MW(e)	1968	AEG	Oil-fired superheat, owned by VEW/AEG
KNK, Karlsruhe	Na cooled ZrH_2	19 MW	Autumn 1969	Interatom	—
KKN, Niederaichbach	D_2O gas-cooled	100 MW(e)	mid-1970	Siemens	Prototype, pressure tube design, 1·15% U as oxide
Stade	PWR	630 MW(e)	1972	Siemens	For HEW/NWK utilities
Würgassen	BWR	612 MW(e)	1972	AEG/Telefunken	For PREAG
KSH, Geesthacht	HTR	25 MW(e)	?	GHH	Closed circuit gas turbine, second ordered
ITR (Julich?)	In-core thermionic	20 kW(e)	?	Siemens/ Interatom/BBC	Prototype, sodium-cooled 10–15% efficiency, for space applications

ments in design and reliability are felt to be required to improve competitiveness with other systems, and computers are expected to be used to optimize control. Heavy water reactors using natural uranium reactors are considered to have reached the commercial stage, and are also attractive because of the export potential of natural uranium systems. With the test facilities provided by the MZFR multi-purpose research reactor and the experience from the Neideraichbach prototype reactor, it is likely that Federal funds will be made available to pursue this line of development.

The long-term aim of the Third Nuclear Programme is to develop high temperature reactors (HTRs) and fast reactors, both of which will be needed in a balanced programme. The AVR reactor experiment represented a first step in attempting to move directly to advanced thermal systems, omitting some of the intermediate stages of the US and UK programmes. The industrial firms in the Thorium HTR Association, with Euratom, have strongly supported the design of a 300 MW(e) pebble-bed gas-cooled HTR. The design of this system has been frozen, and negotiations with the utilities are reported to be well advanced. The possibility of breeding with this reactor type is being actively pursued at Julich, using the thorium-^{233}U cycle. Increased efficiency can be obtained using closed-cycle gas turbines with the high gas temperatures obtained in this system. The 25 MW(e) KSH experiment at Geesthacht is expected to test the theoretical predictions and provide operational experience.

Serious study of fast breeder reactors started in 1960 at Karlsruhe, which has become the main centre for developing fast reactors. Participation in the American SEFOR project was negotiated in 1964. The Karlsruhe ARGONAUT reactor was modified to the coupled fast/thermal system STARK; the subcritical SUAK fast assembly was built to study the physics of fast reactor systems; and in 1966 the SNEAK fast critical zero-energy assembly achieved criticality. On the basis of this earlier work the Third Nuclear Programme anticipates the construction of two prototype fast breeders, one sodium-cooled and one steam-cooled. Work on the former is already subject to a collaborative agreement with Dutch and Belgian firms, and the Germans are apparently anxious for some degree of international collaboration on the steam-cooled type also.

The setting up of the Geesthacht establishment to study marine applications has already been noted. This led to the construction of the *Otto Hahn*, Europe's first nuclear powered merchant ship, scheduled to make its first Baltic cruise in November. The vessel is not expected to be fully economic, but will provide valuable operating experience. Development work on marine applications is planned to continue, but no guess is advanced as to the date when nuclear ships will be commercially viable to justify further vessel construction.

A notable new reactor type developed by AEG, Siemens, and BBC collaboration is a sodium-cooled thermionic reactor. This prototype will probably be built at Julich, with aerospace applications in mind. The Federal authorities are providing £5 million over the next six years to support this project. An efficiency of 10–15% is claimed with 19 diode chains operating at 1500°C.

It is clear that a most ambitious third programme is being attempted to achieve the expected 25 000 MW of nuclear capacity by 1980. In view of the delayed start to the German nuclear programme, this will be a remarkable achievement. There are already indications that the rate of ordering nuclear stations is increasing. In addition to the systems listed in Table 2, there are reports of four large ($\simeq$600 MW(e)) stations to be built at Recklinghausen, at a site on the Neckov river, in south east Bavaria where fossil fuelled electricity costs are high, and at Weisweiler in the brown-coal region of the Rhine. It will be interesting to see if purely commercial considerations result in the expansion of nuclear capacity considered desirable by the Federal Government.

BNES Conference Technical Report

Steam Generating and other Heavy Water Reactors

INTRODUCTION

THE Society organized a Conference on heavy water reactors which was held in London on 14, 15 and 16 May 1968. The timing of the Conference was specifically determined by the commissioning date of the prototype 100 MW steam generating heavy water reactor built by the UKAEA at its Establishment at Winfrith Heath. This plant was officially opened by HRH Prince Philip, Duke of Edinburgh, on 23 February 1968. Thus it seemed appropriate to provide a platform for discussion of the technical features of this reactor system at a time when a reasonable amount of operating experience had been obtained. It was further considered by the Society to be opportune to promote discussion on broader aspects of heavy water reactor technology, and contributions were invited from other countries active in this field.

This proved to be the formula for a very successful Conference. About 350 delegates from 24 countries attended, to hear and discuss 17 Papers (listed in full in the Appendix). These were presented in groups as follows: (a) SGHWR general design, (b) SGHWR performance, (c) Materials and operation, (d) The broader scene. These presentations provided the framework for discussion both on specific points of technology such as core physics, materials, water chemistry and fuel management and also on the strategy of heavy water reactors in relation to their design objectives and in relation to the provision of the special nuclear materials required to sustain large power reactor programmes. The following is an account of the principal points in the Papers and in the discussion which they evoked.

The Conference was opened by Mr E. S. Booth, Member for Engineering, CEGB, introduced as one of the world's major purchasers of power reactors. In his Address, Mr Booth said that despite the fact that a quarter of a century had passed since the first heavy water reactor went critical, and despite the interest shown by designers in heavy water systems, these had not so far (except in Canada) obtained a major position in the world's nuclear power programmes. The competing graphite and light water moderated systems had been the first to be exploited through the boost these had received from military programmes. However, despite the economic penalties attaching to the diversification of reactor types within a national nuclear power programme, it was clearer today than in the past that there was a place for advanced converter reactors. It now seemed improbable that there would be a direct switch from today's commercially established thermal reactors to fast reactors; thermal reactor construction would proceed alongside fast reactor construction for a considerable time and high fuel utilization would progressively become a more valued property of a system. The CEGB's provisional estimates indicated that another 30 000–40 000 MW of thermal reactor power would be added to the network, even allowing for a substantial fast reactor programme.

With a growing share of the installed generating capacity filled by reactors, safety for urban siting and load-following capability assumed greater importance, certainly in the UK and probably elsewhere. Urban siting would reduce transmission costs and improve rural amenity; in this respect, the minimum standard for safety assessment was the gas-cooled reactor in a concrete pressure vessel, for which the previous remote siting policy had now been relaxed in the UK. He hoped that the Conference would pay some attention to both these points.

GENERAL DESIGN OF THE SGHWR

Mr R. V. Moore[1] began his presentation of the SGHWR system by tracing the early history of the studies undertaken by the UKAEA which led to the ultimate selection of this system and the decision to construct a large power prototype. In the middle of 1957, on reviewing the design progress of the AGR, it was decided that the study of an alternative advanced thermal reactor concept should be undertaken by the UKAEA. This was initially regarded as an insurance policy against two points of potential weakness in the AGR as they were seen at that time, namely, the long-term irradiation behaviour of graphite and the use of large steel pressure vessels. There was, therefore, in these studies an early predilection for heavy water moderation in an unpressurized calandria, accompanied by a pressure tube construction for the primary circuit. A first study using steel cladding for the fuel and carbon dioxide cooling was considered to lead to a plant with limited economic attractiveness or development potential. It was followed by a study of a steam-cooled reactor in which steam flowing through the core in three or four passes was alternately superheated in the core and de-superheated in external heat exchangers before finally passing as superheated steam to the turbine. This system, though having the attraction that there was no phase change in the coolant in its passage through the core, had the economic disadvantage of retaining heat exchange equipment and the technical disadvantage of requiring a major development to define a fuel cladding which would withstand the required superheated steam conditions. It was then considered that the attractive points of the system could be retained and the disadvantages minimized by a direct cycle system. In this system, water was fed to the pressure tubes and boiling permitted in the core, the efflux from the pressure tubes being led to drums in which steam/water separation could take place, the water, together with feed water, being returned by pumps to the bottom of the pressure tubes and the steam led to the turbine, as in an assisted circulation fossil-fired boiler. For this system, studies indicated that lowest generating costs would be obtained with slight enrichment of the fuel. The feasibility aspects of the new problems of the effects of two-phase flow on physics, nuclear control characteristics, hydraulic and hydraulic/nuclear stability, corrosion, and heat transfer and dry-out were studied during 1958 and 1959.

Design studies of commercial SGHW plants in the range 100–600 MW were also carried out in order to determine the size variations of key components, so that the prototype

power plant, when built, would necessitate the minimum possible extrapolation of experience. In particular, it was decided that a pressure tube of 5·14 in. i.d. containing a fuel element in a single 12 ft length composed of 36 rods of 0·57 in. diameter pellets clad in Zircaloy would be suitable throughout this power range.

The size of the prototype, construction of which began in May 1963, was selected to be 100 MW, with some capacity in the power plant and other components for uprating. For further development, the prototype contains a number of experimental features, notably the provision of eight channels at the edge of the core which can eventually be used for superheat. Other possible developments are a natural uranium version (though this would require significant changes in core design and would have higher generating costs) and a system optimized for plutonium production and, possibly, plutonium recycling with natural uranium feed. It was believed that the reactor would have good load-following characteristics; in order to verify this belief, it was being planned to place the reactor in a load-following role early in its life.

Mr N. Bradley[2] next described the Winfrith Heath 100 MW prototype SGHWR and its principal engineering developments. The 104 boiling channels and eight superheat channels of zirconium alloy are arranged to pass vertically through clearance tubes in the aluminium calandria containing the heavy water. The pressure tubes are rolled at top and bottom into collars, which connect to the stainless steel primary coolant loop pipework, the tube diameter at the bottom being less than that at the top, so that, should tube replacement ever prove necessary, the pipework at top and bottom can be cut and the pressure tube drawn upward through the calandria. The steam/water off-take is a lateral branch at a position below a combined shield and locking plug in the head of the standpipe; the fuel element is solidly connected to this plug so that refuelling can be carried out by unlocking and lifting the plug. The fuelling equipment and the design of the standpipe are such that fuelling can be carried out with the reactor at power. The fuelling machine is mounted on a concrete shield rotating eccentrically within a second rotating shield to reach any of the standpipes; these shields form part of the primary containment and are each supported on hydrostatic bearings which also serve as seals.

Reactor control is provided by two systems, namely moderator level control for rapid changes and controllable boric acid concentration in the moderator for xenon override and long term reactivity control. The power control systems are designed for a 10% per minute rate of change of power. Emergency shut-down is also provided by two systems. A fast acting system injects lithium borate solution into special tubes built into the calandria when the safety circuits trip the valves holding this fluid in pressurized reservoirs; more slowly, the moderator is emptied into a dump tank. An emergency spray system provides fuel element cooling should the primary circuit suffer a rupture, with consequent loss of coolant.

Steel neutron shields of very rugged construction so that no repair provision is necessary, water filled and cooled, surround the calandria on all sides. The concrete biological shielding surrounding the neutron shields is arranged as a vented containment discharging through lutes into a pond which provides the first stage of decontamination; the plant building itself, suitably vented, provides secondary containment.

In the design of larger SGHWR plants, it would be intended to use the basic modules established for the core of the prototype. The core height would remain invariant, thus providing (at 15 ft with packing) a readily transported dimension. For outputs in the range 150–600 MW, the calandria would be split on a diameter, the two halves being seal-welded at site; for greater outputs, the calandria would be split on two chords. Pressure tubes and standpipes would be of the same general design as the prototype but arranged in shop-fabricated platens (of up to 14 standpipes, weighing approximately 8 tons) together with instrumentation pipework, including separate pipes for a BCD system, to speed site erection. The number of primary circuits would be selected according to output. Primary circuit pumps would be glanded in the larger sizes, incidentally easing the provision of additional inertia on the pump shafts to increase run-down times after an electrical failure.

Further studies have shown that attractive fuel cycle costs can be obtained with off-load refuelling. A typical proposal would be based on a nine-batch fuel cycle with fuel replacement at 21 000 MWd/t, requiring fuelling action twice per year. Off-load refuelling would in any case be required where, for larger plants, pressure containment instead of vented containment was selected. By sealing the tops of the standpipes into a membrane, a pond can be constructed directly above the core into which, when filled with reactor-quality water, the fuel can be drawn with the reactor off-load by a much simpler machine than that provided at Winfrith. At the same time, fuel that has been cooling for 6 months in another part of the pond would be removed from the containment building.

Mr D. O. Pickman[3] described the fuel element used in the Winfrith SGHWR, the considerations leading to the design, the finally chosen operating parameters and various developments in hand to secure improvements to the design. Amongst these was a design for an SGHWR using natural uranium. The Winfrith element and the proposed natural fuel design are illustrated in Figs 2–4 and pertinent design parameters given in Table 1.

Oxide was chosen for the form of the fuel mainly because it is compatible with steam and water. The Winfrith reactor had in fact been operated for three weeks with a failed pin, and the activity released was quite acceptable. Four power cycles had failed to increase the release. The only other limit arising from the fuel alone is its melting point of about 2500°C. High density fuel was chosen to keep the fission gas release down, a density of 10·65 g/ml being finally adopted. Limitations on the fuel were summarized as follows:

(a) In order to limit the net swelling of the UO_2 to 0·5% after the initial micro-porosity has been taken up, the burn-up should be limited to 25 000 MWd/t. (It has yet to be established that the proposed uprated design, with a centre temperature of 2400°C, will not exceed the swelling limit at a reasonable burn-up, although it is recognized that swelling may be governed by the performance of the cold rim of the fuel.)

(b) Gas release is diffusion controlled at all temperatures, with enhanced release rates accompanying equiaxed grain growth in the 1600–1800°C region, and 100% release at

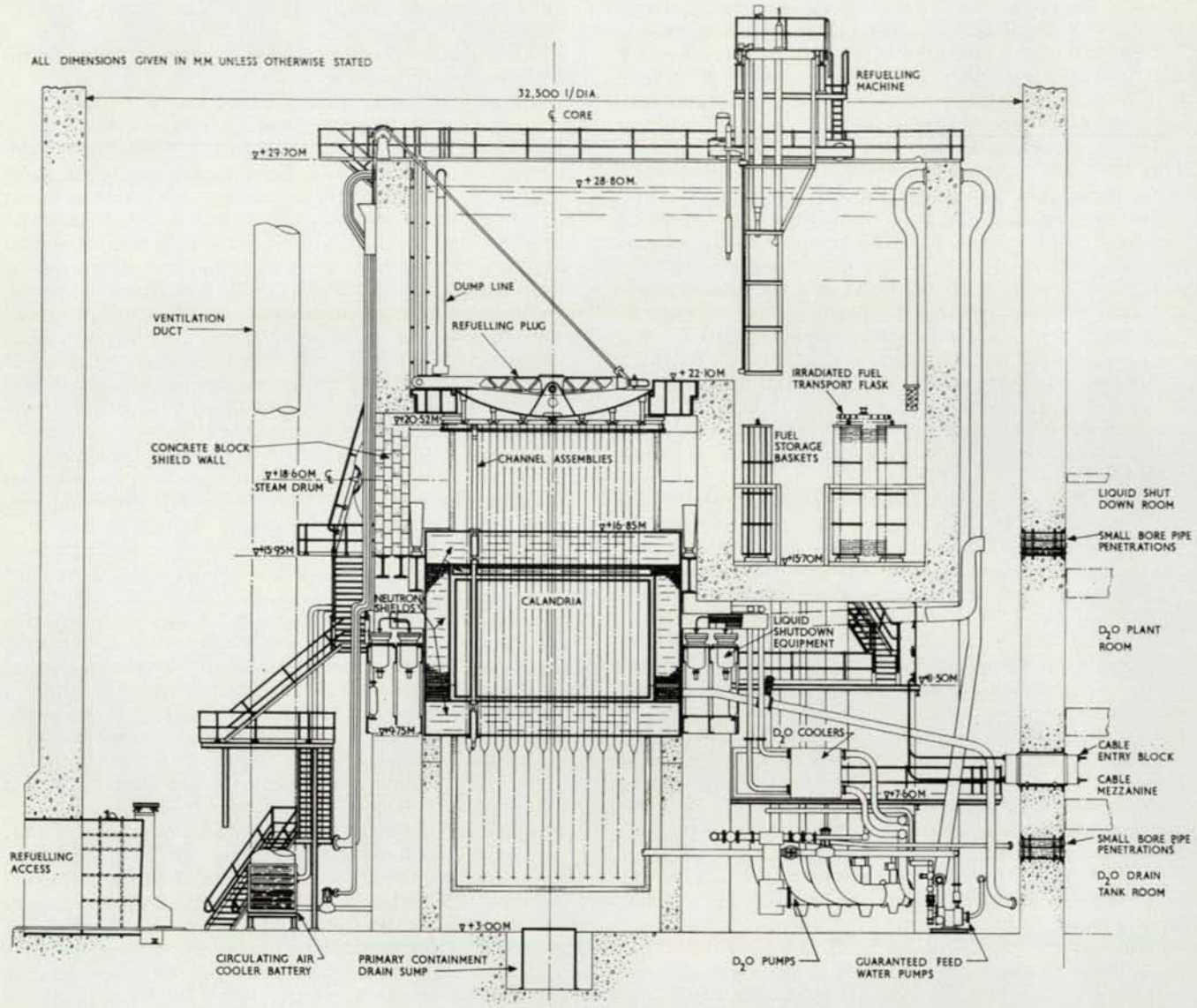

Fig 1 **Cross section through reactor (1500 MW(h))**

temperatures above 1800°C where columnar grain growth occurs.

Vibro-compacted UO_2 is under consideration for future enriched designs to eliminate inter-pellet gaps and, therefore, can wrinkling, to produce low density fuel and therefore reduce net swelling, and to reduce fuel costs.

U_3Si is being considered for natural systems because it is compatible with steam and water, has high density, low fission gas release, high thermal conductivity and a coefficient of expansion which yields very low can strains. However, it has shown high swelling which may necessitate the use of fuel with a central void.

Zircaloy-2 and stainless steel were considered for the cladding of the Winfrith SGHWR, but reports from the USA of stress-corrosion cracking in stainless steel in boiling water environments, coupled with the lower neutron absorption of the Zircaloy-2, led to the latter being adopted. From a corrosion point of view the commonly accepted temperature limit for such an alloy was quoted as 360°C, which gives a very large margin above the anticipated operating temperature for the Winfrith SGHWR can of about 304°C. However, this margin has to cater for any hot spots due to crud deposition in the fuel, to be monitored in the Winfrith reactor. As is mentioned later, PWR experience was said to suggest that significant crud deposition could occur and yield hot-spots.

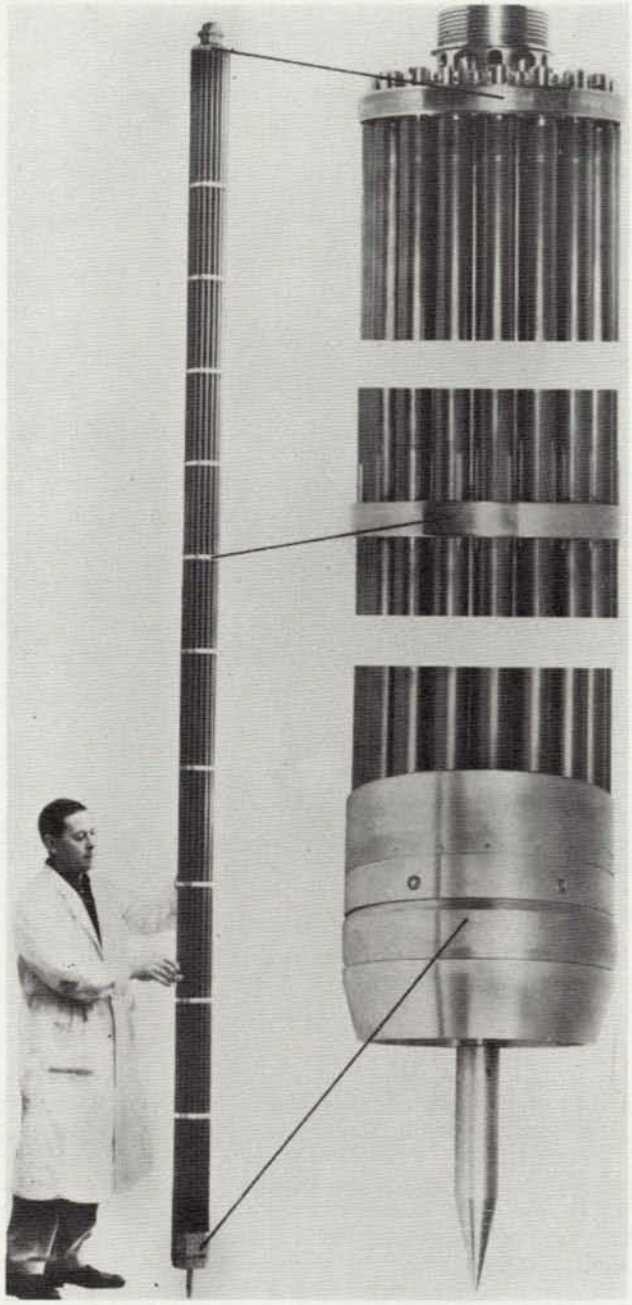

Fig 2 **General view of prototype fuel element**

Table 1: *Main parameters of enriched and natural fuel elements*

	Enriched	Natural
Fuel	UO_2	UO_2
Pellet diameter	0·570 in. (14·50 mm)	0·674 in. (17·12 mm)
Cladding	Zircaloy-2	Zircaloy-2
Cladding thickness	0·025 in. (0·635 mm)	0·015 in. (0·38 mm)
Length of element (overall)	167 in. (4·24 m)	45 in. (1·14 m)
Length of element (fuelled)	144 in. (3·66 m)	44·3 in. (1·125 m)
Number of elements per channel	1	4
Number of fuel pins per cluster	36	36
Pin diameter	0·63 in. (16·0 mm)	0·704 in. (17·88 mm)
Number of intermediate grids per element	10	4
Type of grid	Brazed stainless steel	Machined Zircaloy-2
Grid spacing	12 in. (305 mm) and 15 in. (381 mm)	9 in. (230 mm)
Material of central rod	Zircaloy-2	Zircaloy-2
Min. pin–pin spacing	0·080 in. (2·0 mm)	0·030 in. (0·76 mm)
Volume of gas plenum	1·28 $in.^3$ (21·0 ml)	0

Clad thickness was chosen to limit the tensile strain imparted by the fission gas pressure lifting the can off the fuel, and to limit the compressive creep in the can when the fuel is not supporting it against the coolant pressure. The anticipated fission gas pressures in the Winfrith design, the uprated Winfrith design and the proposed design for natural fuel were much lower than the 1900 $lbf/in.^2$ required to produce the limiting stress of 12 000 $lbf/in.^2$ in the can. In the case of the Winfrith design, the voidage provided inside each can (including plenum) will keep the fission gas pressure below the coolant pressure at all times. As regards power cycling, a reduction from 100% to 75% power would leave the can totally unsupported for a day, so that bigger power changes had no further effect on daily cycling. The strain thereby injected was such that a daily cycle of 100–75–100% was estimated to consume only 1% of the available fatigue resistance of the can, even when the effects of irradiation and hydrogen pick-up on creep rate and ductility are taken into account.

The can is also strained outwards by the swelling of the fuel and by any mismatch between the diameter of can and pellet at operating conditions. The latter is limited to 0·5% which, added to the anticipated swelling of 0·5%, gives a total strain of 1·0%. It was reported that hydride platelets, oriented radially, can form whilst the can is under a hoop stress, either tensile from the above expansion, or compressive from the coolant pressure, and thus embrittle the can. Manufacturing processes were therefore developed which produced the required preferred orientation in the can to minimize the subsequent formation of radial platelets. The can is designed to accommodate 300 ppm hydrogen absorbed from the corrosion process, to be compared with the data of Pashos *et al.* (Fuel Performance in Boiling Water Reactors, *Nuclear Applications*, 1966, **2** (Dec) 510) indicating that the hydrogen level will not exceed 90 ppm.

Work was reported on the asymmetrical collapse of the can under the external coolant pressure leading to the formation of longitudinal wrinkles which could rupture during subsequent straining. It was shown that by keeping the initial pellet–can gap down to a maximum of 0·005 in., which is acceptable from a manufacturing point of view, this problem

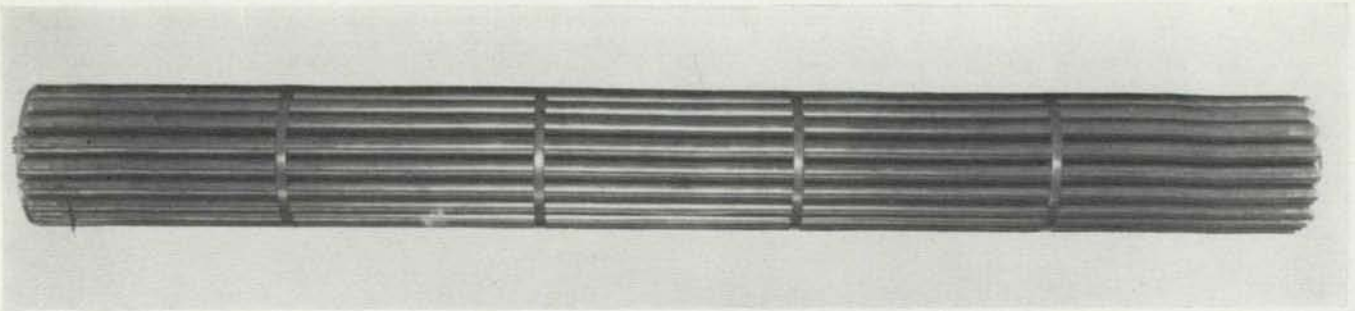

Fig 3 **General view of one 3 ft 9 in. (1·14 m) element of the natural design**

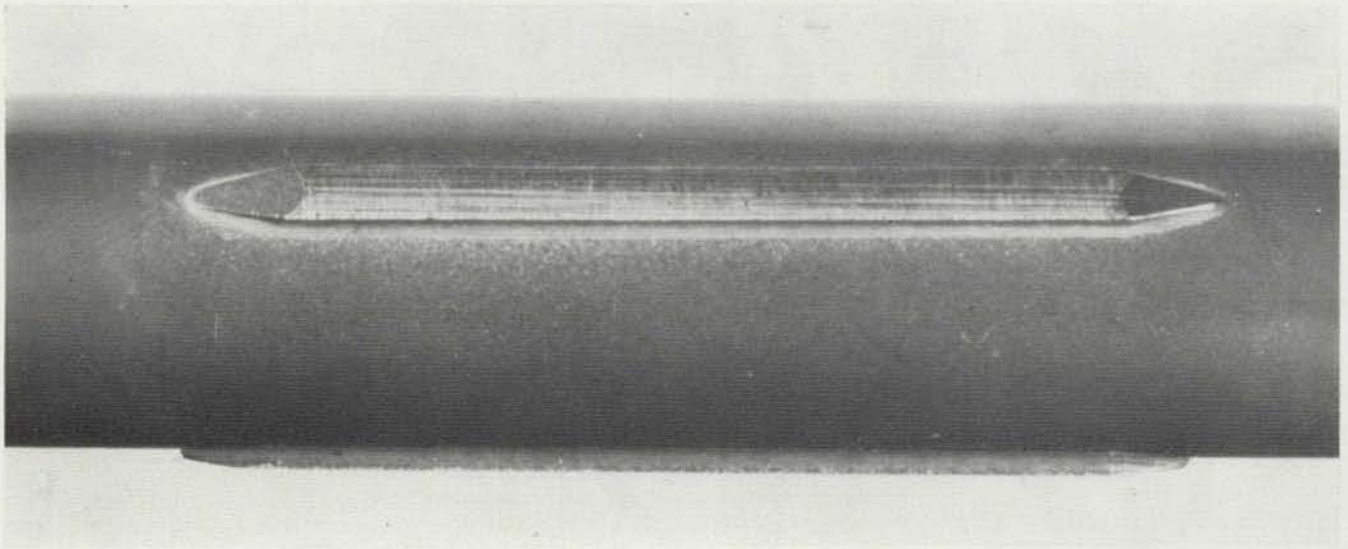

Fig 4 **Brazed wear pads on fuel pin, pickled and autoclaved**

could be avoided for the wall thicknesses and coolant pressures used in the Winfrith design and proposed for the uprated and natural designs. Gaps are formed, however, during reactor operation, and whilst these will not lead to wrinkling in the heavy cans of present designs it might put a lower limit to can thicknesses of the future.

It was reported that thermal cycling under high coolant pressures might lead to progressive ratchetting of the can along the fuel, leading eventually to failure, but out-of-pile work has not shown this.

Zircaloy-2 wear pads are brazed to the can with a Zr-5 w/o Be alloy, to prevent the can rubbing against the steel intermediate support grids, and tests show that these are very effective.

The Winfrith element contains 36 fuelled pins in three rings of 6, 12, and 18 respectively, plus an empty support tube through the centre. This tube carries the top and bottom support for the element and the ten intermediate support grids which space the pins and support them laterally. It also acts as a means for supplying emergency cooling water to the fuel element via a series of drilled holes. The pins are the full length of the core with a plenum at one end to accommodate fission gases. The merits of multiple short elements versus a single element are shown in Table 2.

In the natural uranium design much shorter pins are proposed, mainly to permit axial shuffling. The need to reduce the H_2O absorber in the cooling channels has also led to a much closer pin spacing (0·030 in.) so the wear pads have had to be rejected in favour of direct contact between can and Zircaloy-2 grid. Tests to date show acceptable fretting. The fission gas pressure is much higher due to the absence of the plena.

The natural fuel has a much reduced burn-up, the peak being estimated to occur in the range 7000–9000 MWd/t, so that very little UO_2 swelling is expected. The same can strain limit (1% total) is proposed as for enriched fuel, but, depending on can thickness, the initial can–pellet gap may have to be reduced to preserve stability in the initial hot pressurized situation. The cumulative strain, due to creep strain reversals arising from power changes, may however be the critical factor in determining the endurance of the can. Another factor of importance is that the cladding on this fuel is elastically unstable, even at the upper end of the thickness range. Irradiation experiments have indicated generally

Table 2

Advantages of long elements	Advantages of short elements
	Permits axial shuffling to even out burn-up
	Ability to re-use some pins when only one failed in a cluster
	Flux peaking at ends of each element may lead to hotspots
	Increased pressure drop over gaps between pins
	Increased absorber and loss of fuel between elements
	Cost increased due to the cost of the extra end fittings plus fabrication
Cheaper to produce the reduced number of pins	Cost of pin reject is less
Needs higher fuelling machine and more difficult handling route	Light to handle, eases layout of production line (0·5 ton from Winfrith bundle compared with 36 lb for Douglas Point bundle)
Can use gas plenum if needed to accommodate fission gas especially in a transient	Smaller quantity of fission products released for single pin failure

satisfactory behaviour for such cladding for the proposed natural SGHWR service, but more frequent pressure reductions during fuel lifetime may prove to be deleterious as a result of a ratchetting type mechanism.

Corrosion effects in the thinner cladding, operating at the reduced system pressure and for the shorter dwell time, are expected to be less important than in the enriched reactor. The lower corrosion rate and shorter time more than offset the reduction in thickness of cladding in terms of H_2 pick-up.

The possibility of the coolant flow causing the individual pins to move or vibrate, leading to fretting damage or fatigue, was examined by subjecting full size bundles to typical coolant flows in a high-pressure endurance loop. This work showed unacceptable fretting on early wire-wrapped cans, but the present design using wear pads is alright from this point of view. Oscillations were set up by an early design of spiral neutron scatter plug which had, therefore, to be re-designed. A residual vibration around 7 cycles/s still exists, the consequences of which will be established by examining fuel elements from the reactor.

The author concludes by listing two major areas in which improvements are being sought in pin and cluster designs, namely (a) with highly rated elements with associated minor fuel element design changes to prevent can strain and gas release limits being exceeded, (b) with thin cans down to 0·010 in.

SGHWR PERFORMANCE

This session was devoted to performance aspects of the SGHWR, including nuclear, thermal and hydraulic design, control, and fuel management.

The Paper on nuclear design resembled an iceberg in that it was not immediately obvious from the visible portion that it was the result of an extensive and carefully designed programme. Dr D. Hicks[4] explained how the simultaneous development of theoretical methods and a special programme of experiments had achieved the targets in predicting reactivity, void coefficient and power distribution with margins to spare. A somewhat surprising feature was the ability to predict, in strongly heterogeneous cores, power distributions to the required accuracy without recourse to sophisticated heterogeneous methods which can be expensive in computer time.

The characteristics of the SGHWR lead to strong interactions between the state of the coolant and the neutronics of the core. This limits the extent to which the principle of separability can be used in the design calculations, and consequently a system of modular compatible codes, PATRIARCH, has been developed to facilitate the iterative calculations. In spite of the care taken in designing this system of calculation, the expenditure of computer time was considerable, sufficiently so for the authors to consider that development of more efficient codes is desirable.

Satisfactory agreement between theoretical predictions and measured values was reported at all levels of experiment, culminating in the mock-up of the Winfrith SGHWR core in DIMPLE. This gives confidence in the ability of the methods to predict the behaviour of commercial enriched reactors based on the same size pressure tubes and fuel assemblies as in the prototype. However, the same confidence is not considered justifiable when dealing with the reactors using natural uranium as fuel. Consequently a special programme of reactor physics experiments is in progress with the objective of improving the accuracy of nuclear calculations for natural fuel. It is apparent that the millennium, when theoretical methods and computers will be so powerful that experimental verification of relatively small changes in reactor types will not be required, is still beyond the horizon.

Mr J. A. G. Holmes[5] said that, in general, the problems arising in the thermal design of the SGHWR were similar to those which had been met and solved with other reactor types and in other related fields. However, some extrapolation from previous experience was required.

The selection of an economically attractive design and evaluation of its performance calls for the ability to predict a variety of effects, notably fission gas pressure, condition for dryout, coolant flow distribution, and hydrodynamic stability margins. The comprehensive PATRIARCH system of computer codes was also used for steady state performance analysis. A code SPLOSH has also been devised for studying hydrodynamic instability and the behaviour of the coolant flow following a pump failure.

As many of the computer codes depend on empirically determined coefficients, an extensive experimental programme was launched to cover these aspects which were not already backed up by sound experimental work. Among the rigs used were the 1 MW and 9 MW electrically heated test rigs at Winfrith. In particular, the 9 MW rig could accommodate full size fuel elements. The tests carried out included steady-state dryout measurements, pressure drop measurements and transient dryout measurements. It was not possible to test the prototype reference design for parallel channel hydraulic instabilities on the full scale test rig as the margins against such instability in the design were too large.

The SPLOSH computer code predicts that hydrodynamic instabilities are well damped at channel powers of 5 MW as used in the current design, and it was concluded that

hydrodynamic instability is not an effective restraint to performance of the SGHWR.

Exploratory studies have indicated that the consequences of dry-out are unlikely to be catastrophic as regards a massive release of fission products, but the outage time due to fuel replacement and plant clean-up is of economic importance. The SGHWR was shown to be highly protected against dry-out at nominal full power.

The Winfrith SGHWR forms a strong basis for the design of commercial SGHWRs and reference parameters were quoted for both enriched and natural uranium commercial SGHWRs.

Mr D Wray[6] described the dynamics of SGHWR systems which led to specification of control system requirements for efficient normal operation and protection against fault conditions.

Alternative ways of satisfying the control system requirements were discussed, including coupled systems in which power demand is set by steam valve position and pressure is controlled by reactivity adjustment, and decoupled systems where power demand is set through the reactor power controller by reactivity adjustment and system pressure is controlled by system valve position. During the discussion it was agreed that a co-ordinated control scheme, combining coupled and decoupled systems, would afford a more rapid response.

In the Winfrith SGHWR three variables are controlled to demanded values. Reactor power, measured at low power by neutron flux and at high power by steam flow, is controlled by variation of moderator level. Drum pressure is controlled by varying the setpoint of the turbine throttle valve and drum water level by feedwater flow rate. Moderator height is maintained to cover the fuel by adjusting the boron content of the moderator to compensate for long term reactivity changes.

On detection of serious faults the reactor is automatically shut down by the very rapid injection of a lithium borate solution into tubes in the core accompanied by a fast moderator drain. Less serious faults, such as a turbine trip, automatically initiate a rapid setback to 20% power.

Much interest was shown in the control characteristics of larger commercial designs of SGHWR which are subject to spatial instabilities. Several methods of control were postulated, such as dividing the top portion of the moderator into separate level control regions by radial plates dipping into the heavy water; to deal with instabilities as well as providing capability of shaping the power distributions, the use of solid or gas-filled rods found considerable support. Mention was made of a modification under development which would enable the Winfrith SGHWR control system to increase power from 75% to 90% in 2·5 s.

A natural uranium fuelled SGHWR will have special problems. Because of large positive void coefficients means will be necessary for rapid reactivity control, and one method being examined is that of primary coolant flow variation. The primary circuit will have to be designed to avoid hydrodynamic instabilities and means provided to reduce the maximum depressurization rates for adequate control. Reactor noise is also a potential problem but no clear information was given on its effects.

Mr F. P. O'Dell[7] described the principles governing the selection of fuel management schemes for the 100 MW(e) Winfrith reactor, for a low enrichment commercial reactor and for a natural uranium design.

For the 100 MW(e) Winfrith reactor, the choice was affected by the dual function of power producer and irradiation test vehicle. To restrict fuelling to annual shut-downs, a three-cycle 'roundelay' scheme was adopted, giving a total radial form factor of 1·33 or better.

In low enrichment reactors, the simplified fuelling machinery, absence of element joint perturbations and flattening of the axial power shape during burn-up obtained with a full-length fuel element were considered to outweigh any advantages to be derived from axial fuel management using segmented elements. In the selection of radial fuel management schemes it is necessary, in addition to satisfying economic criteria, to ensure that the thermal performance and fission gas pressure limitations are not exceeded, by examination of rating histories of individual channels. The results presented were from calculations using computer codes from the PATRIARCH scheme, which allow individual channel representation.

The unit power cost differences between an on-load continuous charge–discharge scheme and 4-, 9- and 16-cycle off-load schemes were presented for a 500 MW(e) design. It was concluded that on-load fuelling is unlikely to be attractive in reactors of up to 500 MW(e) output, and that the off-load schemes show optimum behaviour at about 9 cycles/s.

The off-load schemes considered achieved power flattening by using two fuel enrichments. A single enrichment nine-cycle scheme, achieving power flattening by a radial fuel move, as in the Winfrith reactor, had also been considered, but the extra fuel handling increased the outage penalty.

The variation of power form factor with time for the nine-cycle scheme shown was due to boron poisoning in the reflector. The effects on form factor of an unpoisoned reflector and of power shaping by discrete absorbers were also being examined.

In natural uranium reactors, the achievement of a satisfactory fuel life is of paramount importance. Some form of axial fuel management must therefore be introduced, and the radial fuel management scheme must either be a continuous charge–discharge scheme or an off-load scheme of high order, although the latter may increase the outage penalty. Schemes utilizing natural uranium and natural uranium with recycled plutonium were under consideration.

MATERIALS AND OPERATION

Mr Nichols[8] described the work done in the UKAEA to show that the pressure tubes would not fail mechanically in their service environments under the combined effects of neutron irradiation, water corrosion, carbon dioxide corrosion and internal pressure. Irradiation strengthens and embrittles the Zircaloy-2. The hydrogen absorbed during the corrosion process does likewise, although the extent of the changes depends upon the orientation of the zirconium hydride precipitates in the tube wall, radial, as opposed to circumferential, hydrides giving the lowest ductility.

Four potential modes of failure were examined, namely, tensile yield or bursting, fast fracture from local defects, excessive creep deformation and fracture by creep.

On the first of these the authors showed that new tubes

required an internal pressure of 4200 lbf/in.^{2} to burst them, which has to be compared with an operating pressure of 950 lbf/in^2. Test work on the effects of hydrogen pick-up and irradiation upon the yield and ultimate tensile strength of the tube material showed that both increased slightly thus increasing the resistance to tube bursting.

The second mode of failure was studied by tube bursting tests on specimens with varying defects built into them. Some of the specimens were irradiated to a dose of 2×10^{20} n/cm^2 (1 MeV), which is above that at which embrittlement is claimed to saturate (1×10^{20} n/cm^2 at 1 MeV), and some were loaded with up to 900 ppm hydride which exceeds that anticipated in service (400 ppm). Unfavourable radial hydride orientations were also included in the tests. This work showed that a defect at least 4 in. long was required to propagate rapidly and produce tube failure at the design conditions (16 000 lbf/in.^{2}, 300°C). Such a defect size was said to be several orders of magnitude greater than that which can be detected by the inspection techniques applied prior to reactor commissioning or planned for intermittent use throughout its service. These included optical viewing, ultrasonic testing, measurement of scores and scratches and dimensional measurements. These would automatically enable the operator to discriminate between faulty tubes and leaks from rolled joints. Tests showed that defects much smaller than 4 in. would cause leakage into the insulating gas gap and thus enable them to be detected. It was also reported that work had been done to find the time taken for a defect whose size was on the limit of detection to grow to the critical 4 in., and this showed the period to be greater than the reactor lifetime. Thus the important step is to ensure the tubes are thoroughly inspected before they enter service. Earlier in the meeting it had been explained that it was feasible to replace a pressure tube should a fault escape detection and a leak develop.

For the third mode of 'failure' a limit of 2–3% circumferential strain in 30 years was set. This expansion of the pressure tube leads to small and acceptable changes to the coolant flow and insulation gas gap, and does not jeopardize the support of the element. Since irradiation has a large effect on the creep rate of the Zircaloy-2 tube material (typically a factor of 10), in-pile creep data were needed from which the permissible tube operating stress could be derived. Available data indicated that the SGHWR operating stress of 12 400 lbf/in.^{2} at between 280°C and 290°C is acceptable. An extensive programme of in-pile creep testing is continuing on this topic. In addition some of the pressure tubes of the SGHWR will be monitored for dimensional changes when the reactor is shut down. The possibility of flux and temperature gradients leading to non-uniform creep deformation and possibly bowing of the tubes was raised and clearly needs attention for each reactor design since it will depend upon local conditions. The authors said the SGHWR had been cleared from this point of view, and the dimensional monitoring would enable a check to be kept on it.

Lastly the possibility of the tubes rupturing before reaching 2–3% strain was assessed. Creep rupture tests on Zircaloy-2 as manufactured showed that the time to reach 2% strain is only about half that to rupture, that tertiary creep sets in at about 2–3% strain and that the strain at failure would be about 10–20%. The effects of hydrogen pick-up on the rupture ductility limit were not discussed, but evidence was quoted on the effect of irradiation. It was shown that under irradiation the extent of secondary creep is extended way past the strain level at which tertiary creep sets in. In the particular example given a specimen had reached 7% without passing into tertiary creep, to be compared with the 2–3% in the absence of irradiation. This clearly will extend the time to rupture even though it may not increase the ultimate ductility available.

It was concluded, therefore, that by selecting the tube thickness to fix the operating stress at 12 400 lbf/in.^{2}, the deformation in the tube would be limited to less than 2–3% and that rupture would not occur within the 30 year life of the plant, taking into account the effects of hydrogen absorption and irradiation.

In order to reduce the neutron losses in the pressure tube, the characteristics of the Zr–2·5% Nb alloy are being examined, and to date it appears that a wall thickness of 0·1 in. would be adequate for the SGHWR duty, compared with the 0·185 in. of the Zircaloy-2 tubes. Some Zr–2·5% Nb tubes have been installed in the Winfrith SGHWR to establish their potential.

A short report was given of tests comparing the bursting strength of welded and unwelded tubes which showed the welding made only a very slight difference. This underwrites the ability to replace any tube developing a fault during service.

Dr C. Tyzack,[9] analysed the compatibility problems between the water and the main circuit components, namely the pressure tubes, the primary circuit, the calandria and its auxiliary circuit, and the condenser-feed system.

The extent of metal loss from the pressure tubes due to corrosion in the coolant is difficult to assess, mainly owing to the conflicting data on the effect of irradiation upon the corrosion rate. It appears that at 280–290°C, in the absence of irradiation, the metal loss would be limited to less than 0·001 in. in the 30 year life of the reactor, whilst under irradiation this may be increased by a factor lying between 1 and 30. Should the factor be as large as 30 then clearly the deformation estimates made by Nichols *et al.* will be underestimates, although it is claimed they will still be acceptable. In order to decide on this important point an extensive experimental corrosion programme is being undertaken, which includes monitoring specimens placed in the fuel channels and vaults of the Winfrith SGHWR. The pick-up of hydrogen from the corrosion process is not enhanced under irradiation in the same way as the corrosion rate is, and the total quantity absorbed is unchanged. This yields an estimate of 400 ppm for the pressure tubes at end of life, made up of 300 ppm from the coolant side and 100 ppm from the vault gas in the thermal insulating gap between pressure and calandria tube. Complementary to the work on mechanical properties, the corrosion characteristics of Zr–2·5% Nb are being examined. It is recognized that the out-of-pile corrosion rate is worse than that of Zircaloy-2, but the rate of hydrogen pick-up is no greater, and in fact may be smaller. Furthermore, irradiation does not appear to enhance the corrosion rate of Zr–Nb significantly. This holds out considerable promise for the Zr–Nb alloy which the in-pile tests on samples and tubes should confirm.

The basic requirement governing the choice of material for the out-of-core primary circuit was for a steel with adequate corrosion resistance under operating and shut-down conditions. Information available from the Dresden BWR indicated that oxygen levels during running of about 0·1–0·2

ppm, together with the stoichiometric equivalent of hydrogen, would be attained in the recirculating water. This level was grossly in excess of that considered acceptable for mild steel in conventional practice but appeared acceptable for stainless steel. Consequently the latter was chosen for the SGHWR. Subsequently, however, experimental work in the US and UK has indicated that no difficulties are to be expected from the use of ferritic steels at high temperature (285°C) in an oxygenated coolant when a protective adherent oxide is produced. In fact, the choice of coolant chemistry for acceptable corrosion and release rates from mild steel may well be governed by conditions prevailing at shutdown when non-protective oxides are produced. Experimental work was reported which showed that control of water conditions during shutdown is of paramount importance and that removal of ionic matter is a far more effective way of arresting corrosion of steel than is raising the pH by ammonia additions. The efficiency of wet conservation of mild steel circuitry in aerated high-purity water is currently receiving a large scale practical demonstration on power stations at Hamburg, where no additions are used during shut-down. Based on this evidence it was argued that mild steel could be used for the primary circuit, thus effecting considerable economy and easing fabrication. Furthermore, since the bulk of the corrosion products arise from the feedtrain rather than the primary circuit, changing the latter to mild steel would not significantly increase the quantity of corrosion products carried into the fuel channels. Questions were raised on the consequences of crud formation on the fuel elements and its effect on reactivity, fuel pin friction and the dry-out margin. PWR experience was quoted where all these factors were found to be sensitive to crud deposits. No SGHWR experience was reported, although on the second point the authors claimed the effect would be very small because the channel resistance was only a small fraction of the total circuit resistance. For example, a 30% increase in channel pressure drop would only reduce the flow by 5% and consequently the dry-out margin by 2½%. Plans have been laid to monitor the Winfrith SGHWR to establish the extent of this important problem. The different water treatments and materials as between the various water reactors will of course mean this problem will assume different proportions in each case.

The additional factor governing the selection of materials for this part of the SGHWR circuit, compared with conventional power plant practice, is the need to minimize the quantities of corrosion products formed which may deposit on fuel elements. For LP feedheaters and the deaerator vent condenser 90/10 cupro-nickel was specified as no serious corrosion was expected to the steam side at the temperature and oxygen levels envisaged, and most of the corrosion products formed on the waterside should be removed on the powdered resin unit beyond the second LP heater. However, to avoid the corrosion exfoliation problems associated with cupro-nickel the HP feed heaters were specified as stainless steel. One mild steel unit has also been incorporated on an experimental basis.

The condenser was originally specified as Admiralty brass tubes and Naval brass end plates, but it was subsequently decided to fabricate the inlet and gas extract regions in 70/30 and 90/10 cupro-nickel respectively. This was to minimize the possibility of stress corrosion cracking in the event of the reactor being operated with an ammonia-dosed coolant, when concentrations of up to 1000 ppm ammonia may be encountered in these regions, compared with about 20 ppm in the remainder.

The material of construction of the calandria is required to have low neutron capture cross section, satisfactory welding characteristics and adequate resistance to corrosion and stress corrosion. The eventual choice was Al–2·8% Mg together with Al–3·6% Mg weld filler rod. Experimental studies have confirmed the suitability of this material in the environments which it experiences under irradiation both on the moderator side and on the side exposed to vault gas. The possibility of waterline pitting attack under heat transfer conditions is overcome at the moderator blanket gas interface by returning the D_2O over the tubes so that a continuous water film is maintained and the interface eliminated.

The remainder of the moderator circuit is fabricated in stainless steel which has been shown to produce no pitting or galvanic actions against the aluminium calandria when ion-exchange clean-up is maintained.

Mr G. K. Dickson[10] outlined the design requirements, chosen design solutions and potential developments of chemical control in the moderator circuit and of the control of primary circuit and condensate water conditions.

Chemical control in the moderator circuit is required (a) to control nitric acid, formed by irradiation of inleaking air, so that a pD of 5·5 could be maintained to minimize aluminium corrosion, (b) to prevent build-up of aluminium and steel corrosion products and (c) to control reactivity changes in the early stages of the fuel cycle by adjustment of the boric acid concentration in the moderator. The system chosen to meet these requirements consisted of a cation exchange bed to remove corrosion products and any lithium produced by the n–α reaction, with duplicate boron anion exchange beds for nitrate and borate removal, a column for deuterating and de-deuterating resins together with equipment for injecting deuterated boric acid, and for regenerating spent resins. Further plant was needed for preparation of deuterated regenerants and for the recovery of heavy water from spent regenerants. Use of enriched B_{10} enabled the D_2O inventory to be minimized by reducing anion exchange bed volume and boric acid injections volume. In the final system nitric acid removal is achieved on boron-saturated strong base anion exchange resins rather than separated on weak base resin.

A more elegant system requiring less plant and a lower D_2O inventory is being developed, in which resin, regenerants and heavy water are re-cycled in closed circuits. This is based on electrodialysis and has been christened BRANDY (boron regeneration and electrodialysis).

Given the facilities just described for controlling the boron level in the moderator, their use could be extended, the authors suggest, to

(a) xenon compensation during load changes;

(b) on-load refuelling reactivity compensation;

(c) power changes (e.g. two-shift operation).

Control of primary circuit water quality in a direct cycle reactor is necessary to minimize (a) corrosion in the circuit, (b) deposition of corrosion products which might otherwise affect fuel element heat transfer performance and mechanical reliability, and (c) activation of corrosion products and

problems associated with their accumulation in 'clean' parts of the circuit. Existing information indicated that the major contribution to impurities in the primary water would come from the feedtrain. Consequently the plant was designed to clean up the condensate returns at a point as high up the feedtrain as possible, and a point just prior to the de-aerator was finally chosen for this. Powdered resin units were chosen for the clean-up plant as these allowed it to be operated without regeneration in the vicinity of the de-aerator and combined the functions of ion exchange and filtration. They also reduced the amount of equipment required and minimized the effluent treatment and waste storage facilities. The primary circuit clean-up stream is 'blown down' into the condensate upstream of the powdered resin unit which provides full flow condensate polishing.

Development of 'direct contact' in place of 'tubular' feed heaters will reduce the load of corrosion products entering the condensate stream. This will reduce the need to operate the polishing units at high temperatures, thereby increasing the life and efficiency of the powdered resin. Alternatively high temperature impurity removal systems may enable a simple unit at the top of the feedtrain, or in the primary circuit itself, to maintain impurity levels even lower than those achieved at present.

Operational experience from SGHWR available at the time of the conference was briefly outlined by Mr R. Berry.[9] Moderator purity has been maintained at a high level by operation of the ion-exchange plant and metallic impurities nave been maintained between 10 and 50 ppm. Boron capacity and reaction times have been similar to those predicted. Continuous running of both anion and cation columns produces 'conductivity' water of pD approaching 7, and control of pD 5·5 has been maintained by intermittent dosing with DNO_3. Control of the primary circuit chemistry by the powdered ion-exchange units has raised operational problems associated with the coating of the beds. As a result, small quantities of resin have entered the main circuit; radiolysis of this produced temporary high N-16 levels at the turbine and increased crud levels in the circuit. Troubles of this type, however, are reducing with increasing operator experience. The beds have kept iron and other metallic impurity levels low, but have shown little capacity for SiO_2.

Mr D. R. Poulter[11] presented the safety aspects of the SGHWR prototype. The hazards analysis is being done on a probability basis. This shows that the most serious faults are the so called 'stagnation accident', in which a failure in the feeder header system produces persistent low flow in a larger number of channels, and single-channel accidents in which fission products might be released almost instantaneously.

Data for the hazards analysis were obtained from an extensive experimental programme. Tests using external missiles showed that the calandria tubes protect the pressure tubes against missiles, thus preventing the propagation of pressure tube failures. Experiments with replicas of fuel pins showed which combinations of temperature and internal or exterior pressure led to cladding failure. The effectiveness of the emergency cooling system was also studied. Other experiments determined the decontamination factors obtained by bubbling steam–air mixtures through water or passing them through iodine adsorption plants. The results of the experimental programme enabled the performance of the reactor containment system and the safety of the general public to be assessed.

Mr J. McCrickard[12] described the commissioning programme, which was divided into three phases.

Stage 1 included fuel and moderator loading, zero power physics measurements and hydraulics tests. Thermal movement, vibration levels, safety valve lifting pressure, circulator run-down and cavitation were measured. The run was terminated with a depressurization.

Stage 2. Programme considerations led to the safety circuits being completed and commissioned after zero power testing. Delays occurred when a superheat channel tube assembly had to be removed following a leak, and when a booster pump motor seized on starting. Better displays of motor current loadings should avoid the latter.

Stage 3, power raising, commenced 20 December 1967, the turbo-alternator was synchronized with the grid on 24 December and full power momentarily achieved on 7 January. Radiation surveys and turbo-alternator tests were performed. All ancillary equipment was loaded to capacity and tested, the vibration measurements were repeated and the performance of reactor, turbine and feedtrain checked. The automatic control system was commissioned.

Early operating experience has been satisfactory and since full power was attained on 25 January, 136×10^6 units have been generated with 75% availability. Since 1 March the figures were 101×10^6 units with 96% availability.

Radiation levels have not prevented essential work, contamination being localized near any steam or water leaks. The main activity is retained in the lagging, thus indicating leaks after reactor shutdown. The polishing plant is effective against iron, copper and chloride, but cannot remove silica under local temperature conditions.

The control systems had been tested by using a PACE analogue to simulate the plant, ensuring that they were correctly assembled. Plant responses to demand changes were used to optimize them. Some difficulties were experienced from noisy transducer signals due to pipework geometry but were reduced by re-routeing these pipes. Modifications and re-optimization of the pressure controller are proposed. The reactor fuel charge performed well apart from a leaking fuel element, which allowed operating procedures and monitoring to be checked.

Performance measurements showed that the zero energy criticality measurements agreed well with previous DIMPLE measurements, both hot and cold. At full power, hot, the reactor was 0·25% more reactive than predicted. Burn-up measurements indicated that the fuel cycle length 300 ± 30 EFPD will be achieved, while the measured void coefficient agreed well with prediction. It is likely that the fuel temperature coefficient is more negative than expected. The moderator and coolant temperature coefficients, as expected, were about 5 mn/°C less negative than predicted but the effect is not important in SGHWR. Moderator height coefficients were 25% smaller than predicted but are within control system tolerance. The worth of the (12) liquid shutdown tubes at hot, zero energy, was 3·4% while, if three tubes fail, the remaining nine tubes hold down 2·1%, ample to ensure shutdown.

Predictions of circuit coolant flow lay within the scatter of measurements; individual channel flows were predicted within $\pm 5\%$. However, at zero power with no steam voidage, channels near the pumps received flow 10–20% below

average. This flow starvation disappeared at significant power levels and rates of raising power were not affected.

Steam qualities are measured by Venturi meters fitted to individual channels. These were checked with inlet flow meters and 77 out of 104 proved satisfactory and were used to obtain the core power distribution. The standard and booster channels were operating at just under 4 and 5 MW respectively, as required. However, powers were about 10% high in booster channels and other high flux regions and low elsewhere. Additional full power measurements indicated a form factor of 1·56, about 8% high. The flux scanning is at least as accurate as the quality meters, which were estimated to give accuracies of 5–6% for 3 MW channel powers and better for higher powers.

THE BROADER SCENE

Mr L. R. Haywood presented the paper contributed by Dr W. Bennett Lewis,[13] who was unfortunately prevented through indisposition from attending the Conference. The Canadian programme of heavy water moderated and pressurized heavy water cooled natural uranium power reactors began with the commissioning in June 1962 of the 22 MW horizontal pressure tube Nuclear Power Demonstration plant at Rolphton, Ontario. Since then, 3000 MW of commercial power plants embodying the NPD principles have been committed in Canada, India and Pakistan. The first station in this programme at Douglas Point has now achieved its full load of 203 MW. A further 3000 MW power programme was being considered.

Little difficulty had been experienced in NPD or would be expected in the larger plants in respect to operation of the moderator, though tritium levels would be somewhat higher than in the many heavy water moderated experimental reactors which had operated so successfully in many countries. However, experience with the pressurized heavy water circuits was limited; such as it was, it had shown the need for refinements in design and in operating technique for larger plants intended for operating with a minimum of skilled staff. Although leaks need not become heavy water losses from the system, they imposed demands on the operators, and rigorous disciplines to prevent exposure to tritium, even that adhering to masks and protective clothing that had to be worn during entry into the reactor spaces. High capacity ventilation and speedy adsorber driers were necessary to draw heavy water and its vapour out of all spaces into which small leakages could occur; spaces into which large spillage could occur, e.g. refuelling vaults, were required to be sealed off.

Since one of the factors that could limit plant life was the life of the pressure tubes themselves, it was a point of basic design philosophy that the tubes should be replaceable after some years of service; this operation had been carried out successfully both in NPD and in Douglas Point. At an earlier stage, it had been thought that tube life might be limited by embrittlement through hydrogen pick-up. While this now seemed to be a lesser worry, more recently it had been found that cold-worked Zircaloy 2 under fast neutron bombardment exhibited accelerated creep which might exceed the 2% strain, currently set as the acceptable limit, in considerably less than the stipulated 30 year plant life. A new Zr-2·5% Nb alloy was now expected to show a much better performance. The most recent designs adopted vertical pressure tubes with single-ended refuelling from the top (as against the double-ended 'contra flow' refuelling of the earlier horizontal pressure tube designs) with reduction of capital cost and easement of pressure tube replacement.

The most basic attack on the problem of high pressure heavy water leakage is to change the coolant. A 20–40 MW organic cooled plant had been working successfully at an exit temperature of 400°C at White Shell, Manitoba. It appeared that the problems of hydriding and deposition of fouling films could be controlled and that such a plant, if developed in conjunction with nuclear superheat (in which greater international co-operation was desirable), would show a station efficiency of 40%. Nevertheless, lack of funds had determined a low priority for this work. The other major development is that of a natural uranium boiling-light-water cooled, heavy water moderated reactor of which a 250 MW(e) prototype is under construction at Gentilly, Quebec. The basic ingredients to resolve the conflict between the use of natural uranium, achieving negative overall reactor coefficients and achieving long burn-up are (a) increased uranium inventory, (b) high feedwater temperature, (c) several separate coolant loops, and (d) thin high-strength pressure tubes. Orifices are provided at channel inlets to control hydraulic instabilities.

A principal aim of all the Canadian developments has been the achievement of low refuelling costs, through good neutron economy, standardization of fuel bundles and development of manufacturing techniques. At a burn-up of 9000 MWd/ton, the refuelling costs of the Pickering plant are expected to be 0·63*d.*/kWh on the basis of zero value assigned to spent fuel, falling to an expected equilibrium value of 0·38*d.*/kWh when the reject plutonium is sold or recycled. These low fuelling costs will maintain the competitive position of the heavy water reactors when world uranium prices rise, even against the competition of fast reactors. A further economic prospect would be the gradual introduction of thorium in place of ^{238}U in order to produce ^{233}U.

Mr R. Naudet,[14] of France, first described the status of the carbon dioxide cooled heavy water moderated 70 MW(e) plant EL4, construction of which began in 1962 and which delivered power to the grid in July 1967. The present driver charge of fuel is stainless steel clad; development of low neutron absorbing cans is in hand and currently six channels are loaded with elements clad in zirconium with copper surfacing, the number of these to be increased over the next two years.

Economic studies have shown that for a large reactor, fuelling costs should be acceptable. However, the successful development of the light water reactors has made available low priced, well developed Zircaloy clad uranium dioxide fuel, which could form the basis of heavy water moderated water-cooled reactors with better economy than the gas-cooled version. For this reason, a technical and economic appraisal of the lines of development of heavy water moderated reactors using both heavy and light water cooling had been carried out in France.

Heavy water cooling had the advantage of better neutron economy and the avoidance of positive void coefficients, but the disadvantage of the need to retain a very expensive and heavily tritiated coolant. Both pressure vessel and pressure tube designs were under study. The pressure vessel version seemed currently to be limited to about 300 MW because of

the size limitations on steel pressure vessels; beyond this, concrete vessels would have to be developed, together with appropriate thermal insulation systems to keep the concrete at sufficiently low temperatures and with acceptable levels of thermal stress. Fuelling machinery would with advantage be placed inside the vessel (out of absolute necessity in the case of a concrete vessel) to limit heavy-water leakage; such equipment also required development to a reliable state. On the other hand, the pressure tube version was now well established and seemed to pose no problems other than those noted above.

As far as light-water cooling was concerned, the CEA agreed with the analyses carried out in Britain and in Canada, namely that either one designed for negative void coefficients, obtaining an operationally very safe reactor, but with the penalty of relatively high enrichment (as in the UK), or, if one designed for natural uranium, accepted the resulting loss of performance and the more exacting requirements to achieve acceptable safety (as in Canada). However, it appeared from these studies that optimum economy with satisfactory operation could be obtained at an intermediate position in which only very slight enrichment was used. In a specific case, using only 1·05% enrichment in a boiling-light-water cooled heavy water moderated reactor would give the same refuelling cost as in a natural uranium Candu type (though both would be higher than the optimum refuelling cost for the heavy water cooled reactor, which would also be obtained by slight enrichment); the cost of enrichment would be off-set by the greater burn-up attainable. There would also be worthwhile savings in capital cost.

Mr R. A. Stueger,[15] of Germany, described the contributions to the development of heavy water moderated reactors by his Company. The 100 MW(e) prototype carbon dioxide cooled heavy water moderated reactor under construction at Niederaichbach (KKN), due for commissioning in 1970, was briefly described. This plant uses steel clad fuel to give an exit temperature of 500°C. As in the case of EL4, there is a development towards a natural version using zirconium–copper cladding, beryllium having been finally abandoned about two years ago. An interesting line of development suggested for the system was the use of a carbon dioxide gas turbine as main power plant.

In the field of pressure vessel reactors, Siemens had constructed the 51 MW(e) (nett) MZFR, a natural uranium pressurized heavy water reactor which has been on power since March 1966. The components of the reactor, apart from the core and the on-load refuelling machine, do not differ greatly from those for a light PWR.

The MZFR and the experience gained in its operation were the foundations of a design for a 320 MW plant which was successfully offered in the summer of 1967 in competition with other reactor types for construction at Atucha on the Parana River in the Argentine. The pressure vessel for this plant, to contain a pressure of approx. 1750 lbf/in.^{2}, is of diameter approximately 17 ft 6 in. and weighs 500 tons. At this weight, transport by water is necessary direct from the manufacturer's works to the site. It is this transport problem, rather than the problems of fabrication, which currently sets the limits on the power output from this style of plant; larger outputs would require either some degree of site fabrication or concrete pressure vessels.

The economics of the system favour a relatively high power density, which is achieved by the use of small diameter rods (11·9 mm, compared with 15·9 mm for SGHWR) arranged in 37 rod bundles of undivided length, since axial shuffling would not show adequate economic gain. These bundles are attached to locking plugs and the whole assembly is withdrawn for refuelling and radial shuffling through the individual standpipes provided in the head of the vessel. Within the Atucha vessel, moderator and coolant are separated; the moderator is maintained at about 180°C, leading to a significant increase in burn-up to 8000 MWd/ton. Heat generated in the moderator is used for feed water heating. The independent moderator circulator pumps can be used for removal of shutdown heat. Control is by 29 rods arranged at an angle round the outside of the fuel standpipe region of the vessel head. At 300 MW, no spatial instabilities occur so that control at power can be exercised by only two rods. The plant will be housed in a spherical pressure containment designed to the criteria usual for PWRs; its nett efficiency will be 29%.

For future plants of larger sizes, Siemens have had under development concrete pressure vessels. A necessary change in reactor configuration with a concrete vessel would be the placing of the fuelling machine within the vessel, as in the Marviken BHWR. The concept could be used as a replacement for a steel vessel, retaining external heat exchanger loops or, by a further development, all primary plant could be integrated in the vessel. In both cases, control rods would enter the core from below. For the present, exit quality from the core of a large plant would be kept below 1% in order to ensure that no core instabilities occurred.

Dr S. Villani,[16] of Italy, provided some interesting details of his project. The CIRENE concept is similar in its general outline to the SGHWR, but fundamentally different in its reactor physics and control aspects since it aims at the use of natural uranium fuel. Thus its optimization is basically different, and solutions to the special problems of rapid control action, safety, and low-load and start-up are required. A 500 MW(e) reference design has been worked out in detail, out-of-pile experiments performed on the novel control scheme, and a 35 MW prototype is under consideration for construction on the site of the Latina magnox power station.

Relative to SGHWR (for which the relevant parameters are given in brackets), CIRENE has the relatively low rating of 12·8 (17·8) MW(t)/t, a burn-up of 8500 (18 000) MWd/t, a pressure of 50 (63·3) atm and the exceptionally high exit quality of 25 (10·75)%; this last is chosen to minimize the water content of the channels and is related to a dry-out margin of 1·5, including hot channel factors. A 19-rod segmented fuel cluster is used with collapsible Zircaloy-2 cladding.

Slow reactivity changes can be dealt with by moderator poisoning and level control, as in SGHWR. However, the positive void coefficient of the system demands a rapidly responding control system, which also serves to control spatial instabilities. This is provided by circulating through special U-tubes through the calandria a mixture of helium and borated light water, which, as tests have shown, can provide in a 1·5 cm bore tube stable flows throughout the range from 'all gas' to 'all liquid'. The designed response time for the system is 2 s. Provisions have to be made in the flow circuit for cooling and for dealing with hydrogen and oxygen formed by dissociation during the passage of the fluid through the core. Emergency shut-down is provided

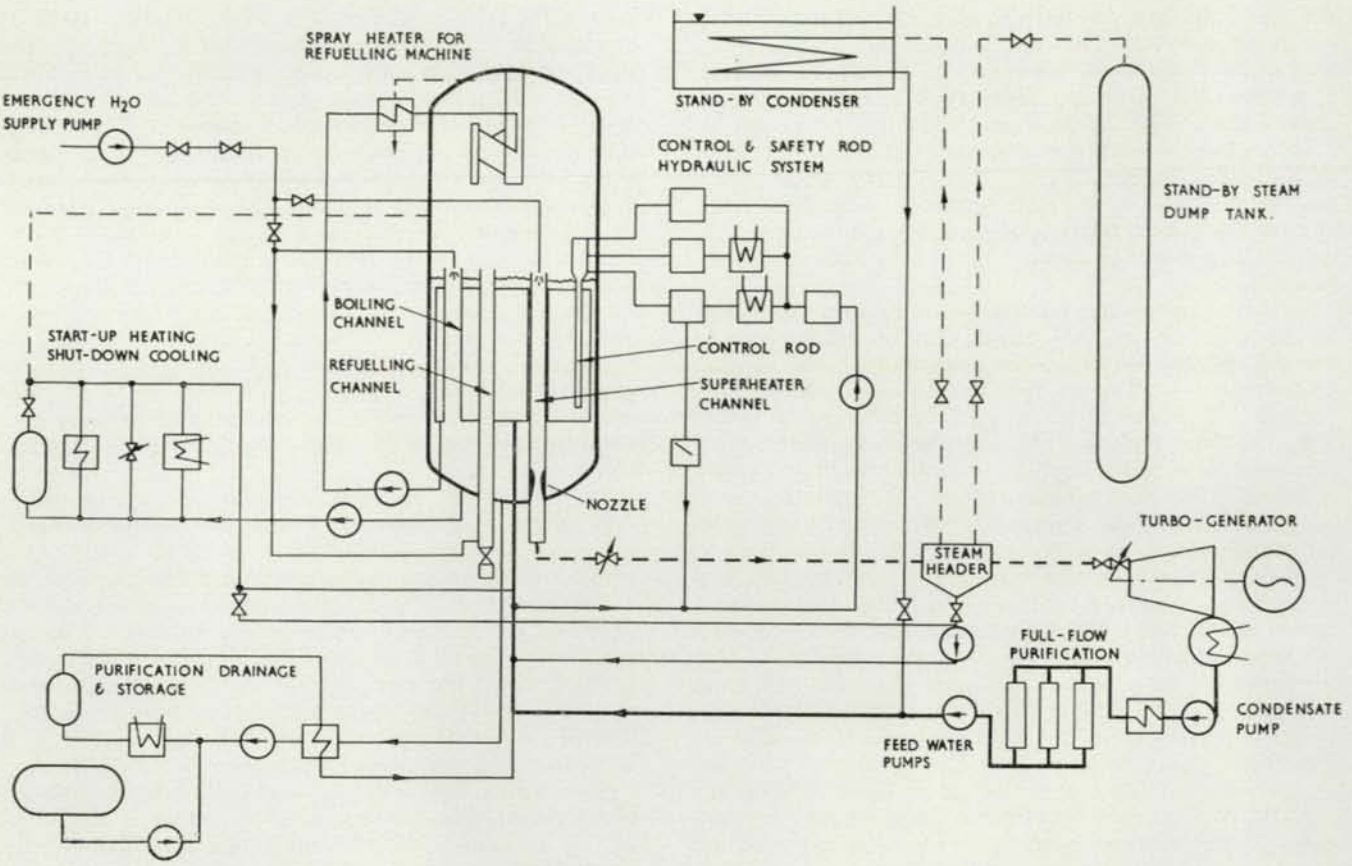

Fig 5 **Simplified flow diagram for Marviken BHWR**

by poison injection under gravity into other tubes also fitted in the calandria; the slower times compared with SGHWR are acceptable as large flywheels would be fitted to the primary circuit pumps.

In order to control the light-water content of the channels, it is proposed to operate at lower loads with falling pressure so that coolant density is maintained at the full-load value. Below 25% load, hydraulic instabilities could be expected; running at lower loads would be achieved by lowering the coolant density by means of a steam pump recirculating some steam to channel inlet. This pump would also be used at start-up for circulating steam through the channels, since with the pressure tubes full of water the reactor would be sub-critical.

Also included in the Paper were some aspects of on-load and off-load refuelling of the CIRENE reactor, future developments (including control by variation of moderator density by bubbling gas through it), the economic balance sheet for the development and the relationship between the CIRENE reactor, with its good plutonium production relative to light water reactors and fast reactor construction programmes. On this last point, other speakers drew attention to the fact that proponents of the natural uranium systems were making a virtue of necessity: high plutonium production was a concomitant of low burn-up and enriched reactors could, with better economy, also provide larger plutonium quantities, if the economic ground rules were adjusted to make plutonium more valuable, simply by restricting the target burn-up.

Mr P. H. Margen,[17] of Sweden, traced the course of heavy water reactor development in Sweden from the commissioning of a 600 kW research HWR in Stockholm in 1954. In 1958, decision was reached to commit a 65 MW(t) prototype HWR at Ågesta for power production (10 MW(e)) and district heating; this plant reached full power in March 1964. Natural uranium and pressure vessel construction (partly to reduce leakages from joints and partly to suit Swedish manufacturing preferences) were selected. The system is thus a PHWR, supplying a back pressure turbine. Fuelling is off-load during the long summer shut-down (enforced by lack of demand). Heavy water leakage has been very satisfactory, amounting from all causes to only 5–7 g per hour of operation at pressure.

In parallel with the committing of Ågesta, consideration was being given to a larger prototype intended to be more closely representative of commercial plants. This 200 MW(e) prototype is under construction at Marviken and is due for full power operation in the spring 1969. The features which, after prolonged study, it was decided to incorporate in this plant were:

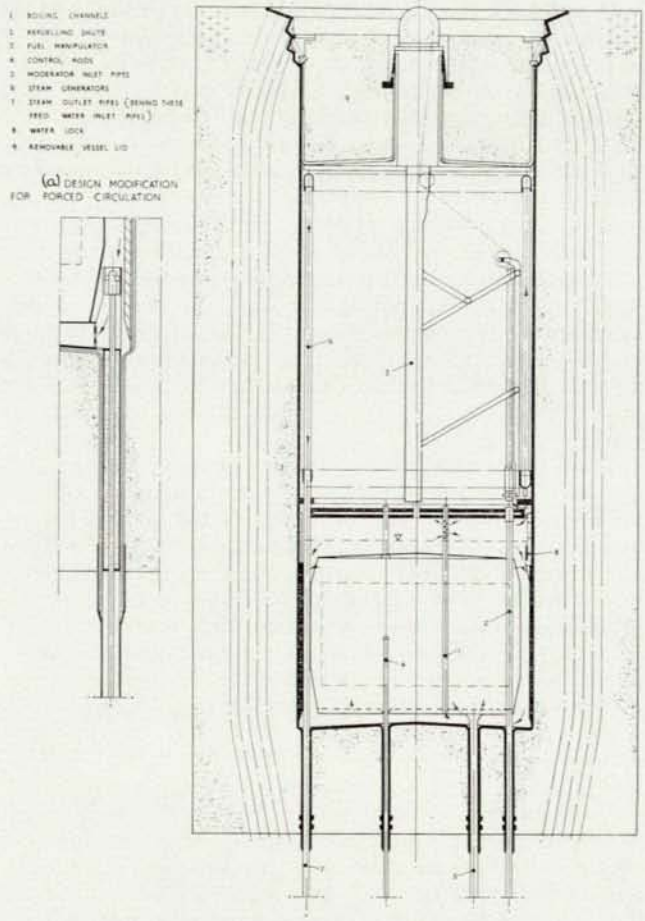

Fig 6 **600 MW(e) BHWR with concrete pressure vessel, integral design and natural circulation**

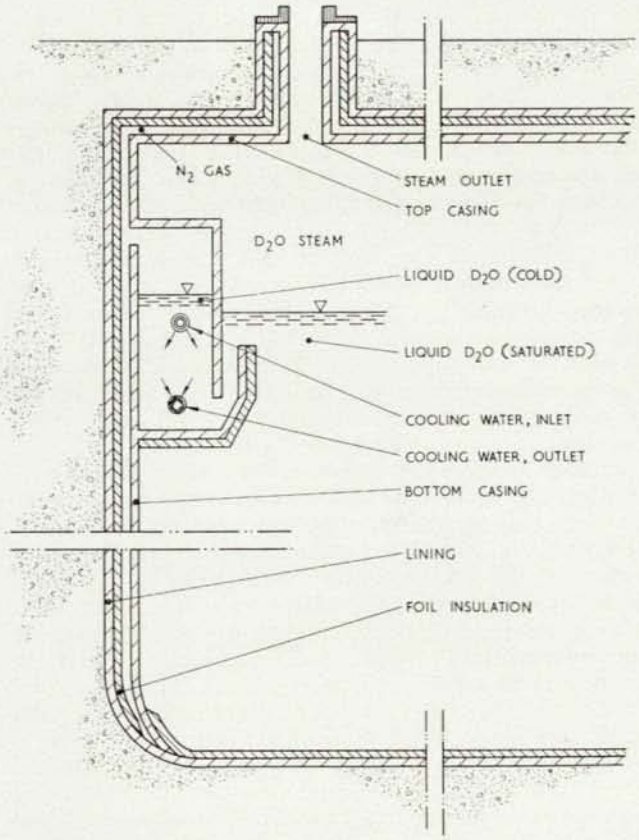

Fig 7 **Thermal insulation for concrete pressure vessel**

(a) boiling D_2O as opposed to pressurized D_2O
(b) natural circulation
(c) direct cycle
(d) on-load refuelling
(e) possibility for nuclear superheat
(f) pressure suppression containment.

Pressure vessel construction and a fuel technology closely based on plant developed in the USA for LWRs were retained. The direct use of D_2O steam in the turbine was confirmed after full-scale tests on turbine seals and a number of condensers, and statistical analysis of condenser tube faults. These suggested that overall leakage rates in D_2O turbo-generators should be very low. However, should potential clients wish to observe the operation of the turbine in the longer term, the BHWR could also be offered in indirect cycle form; the reactor would look the same and the generating cost would increase by about 2·5%.

As regards superheat, some prototype fuel assemblies have been tested in the steam loop of the Studsvik reactor without failure, but opinion on the attractiveness of this feature must await the operation of the Marviken reactor.

For BHWRs of larger output, on-site fabricated steel vessels or concrete vessels would have to be chosen. In the case of the indirect cycle BHWR in a concrete pressure vessel, the heat exchangers can be located within the vessel. Designs have been prepared for a vessel with removable head which will shortly be constructed at Studsvik. Tests have been carried out on a system of metal foil vessel heat insulation, kept free from D_2O by nitrogen pressure balanced with the vessel contents.

Mr H. Cartwright,[18] of the UKAEA, in his paper on the commercial SGHWR drew attention to the need to consider all the relevant cost factors. These were not by any means all controlled by purely technical factors and selected operating conditions. Construction times had to be reasonably short and achievable with certainty. A delay of six months in completion would eliminate a cost advantage of £3–4/kW; providing a six months' time contingency at the end of the construction period could give rise to a cost of about £2/kW as a result of the increased interest during construction. Once in operation, it was essential for the plant to have high availability. Fault-free running depended on the main technical features of the plant but equally on a multi-

tude of ancillaries which all had to function reliably. Simplification of equipment, elimination of all unnecessary features and standardization of components from plant to plant were factors that led to enhanced reliability. In the case of the SGHWR, larger units than the Winfrith prototype would use the same pressure tube size, same pitch, same fuel cluster and same type of control and could use the same ancillary components so that commercial plants should achieve high availability on the basis of the experience obtained at Winfrith. The refuelling system adopted also had considerable bearing on availability; on-load refuelling gave the lowest planned outage and also the lowest fuel cycle cost but, despite this, off-load refuelling was currently recommended for commercial SGHWRs when used in operating regimes providing reasonable periods of slack load, because of the simplifications possible with an off-load fuelling machine.

Another important attribute, bearing both on the achievement of high availability and securing the economic plant life, was high accessibility for maintenance and repair. Commercial SGHWRs were being designed with this point particularly in mind. The pressure tubes, which might last for less than the desired plant life, were readily replaceable. It was, however, to be noted that such replacement would not be necessitated on safety grounds. Cracks which could propagate into rapid failures were of such size that replacement of the tube on the grounds of leakage would have been undertaken well before a dangerous situation could arise.

Especially as nuclear power became a significant fraction of the total connected generating capacity, load-following capability became important in a commercial reactor. The SGHWR had rapid response rates to change of load demand; power changes occurred with little change in operating temperatures or pressure, so that no harmful effects of power cycling were to be expected.

The commercial SGHWR could be provided with a variety of containment forms according to size, location and the contractual arrangements under which the plant was being constructed. Vented containment was probably unsuited to plants greater than 600 MW.

CONCLUSIONS

A number of points of presentation and discussion are more conveniently presented here, rather than with the papers which give rise to the point.

A number of speakers made the point that heavy water moderated reactors, of any of the types described at the Conference, would be more economic in slightly enriched than in natural uranium form. On the other hand, while this is true on the basis of today's prices, future trends in uranium prices could well favour the natural uranium fuelled versions. In any case, a number of customers today would seem to be prepared to pay a premium to avoid the political or supply implications of using enrichment.

For the pressure tube reactors, optimizations were fairly flat in the region of the selected designs so that there was no case for 'tailoring' the design too closely to the precise economic ground rules selected. Economic benefit would result from the adoption of higher exit steam qualities, providing that the control and heat transfer aspects were satisfactory. The unanimous modern trend was to vertical tube arrangements.

If new large programmes of construction of heavy-water moderated reactors were to emerge, then the supply of heavy water would have to be assured, possibly through international co-operation. A special problem was that, while a large production capacity was needed to deal with D_2O inventories, subsequent consumption was very small so that the plants would be under-employed. Utilities would require guarantees of additional D_2O supplies in case a loss of moderator accident occurred. The scale of the problem could be seen in relation to the Canadian programme: this would require in total about 30 000 tons D_2O in 1988, 4000 tons being required in that year. This was to be seen against a production capability of the Savannah River plant of 200 tons/year, of 400 tons/year being commissioned in Canada in 1968, a further 400 tons/year commissioned late in 1969 and probably a further 400–800 tons committed in 1968.

There was a reasonably clear concensus of opinion that the best prospects for heavy water reactors would be achieved by making maximum use of the materials technologies developed for the commercially successful light water reactors. Nuclear superheat was clearly in the position that it might be developed in the future if an economic justification were shown. Superheat fuel would probably be based on high-alloy stainless steel (if stress corrosion proved to be no major obstacle) but the cans would have to be thin to give economic advantage for the concept.

In his closing Address, Mr J. C. C. Stewart, Board Member for Reactors, UKAEA thanked the British Nuclear Energy Society, the Institution of Civil Engineers and particularly the authors and speakers from abroad for a Conference which had produced so much lively discussion and which had demonstrated the very considerable interest around the world in heavy water reactors. One had to note the variety of heavy water reactor types now being offered by so many different reactor builders, and one could not but wonder whether the fragmentation of effort and the confusion thereby caused to potential clients would not increase the difficulties of any new reactor system in establishing itself against existing competitive types.

It would seem from the Conference that the future could see the emergence of two main types, namely, pressure vessel types with boiling heavy water and pressure tube types with boiling light water. It would also seem that the economic and reliable achievement of a natural uranium cycle brought with it a greater technical challenge. He himself doubted if the single advantage of the use of natural uranium would outweigh the simplicity achievable with enrichment; in the UK, it was believed that the slightly enriched pressure tube design offered the best prospect for early commercial exploitation.

A feature of the heavy water reactors was that they were potentially good plutonium producers. Thus the future of heavy water reactors might well be alongside fast reactors and providing the feed material for expanding fast reactor construction programmes.

APPENDIX: Papers Presented

1. Moore R. V. and Holmes J. E. R. The SGHWR system. Proc. Conf. on Steam Generating and other Heavy Water reactors. British Nuclear Energy Society, London, 1968

2. Bradley N., Dawson D. J. and Johnson F. G. Engineering design of SGHWRs. *ibid.*
3. Pickman D. O. SGHWR fuel design and materials. *ibid.*
4. Hicks D., Johnstone I., and O'Dell F. P. Nuclear design of SGHWRs. *ibid.*
5. Holmes J. A. G., Obertelli J. D. and Roberts H. A. Thermal and hydraulic design of SGHWRs. *ibid.*
6. Wray D., Butterfield M. H. and McMillan R. N. H. Control of SGHWRs. *ibid.*
7. O'Dell F. P., Hopkins D. R. and Allen F. R. Fuel management in SGHWRs. *ibid.*
8. Nichols R. W. and Watkins B. Pressure tube materials for SGHWRs. *ibid.*
9. Tyzack C., Berry R. and Campbell C. S. Some material compatibility aspects in SGHWRs. *ibid.*
10. Dickson G. K., Burton W. R. and Riley J. A. Chemical control of SGHWR circuits. *ibid.*
11. Poulter D. R., Johnson F. G., Cowking C. B. and Windle G. V. Safety aspects of the Winfrith SGHWR. *ibid.*
12. McCrickard J., Smith D. and English D. Commissioning and operating experience with the Winfrith SGHWR. *ibid.*
13. Bennett Lewis W. Economic heavy water power reactors. *ibid.*
14. Fonteray P. J., Goldstein S. and Naudet R. Prospects of water cooling in heavy water moderated reactors. *ibid.*
15. Keller W. K. F. and Stueger R. German contribution to the development of heavy water moderated reactors. *ibid.*
16. Villani S. Development of cirene heavy water reactors. *ibid.*
17. Margen P. H. The Swedish programme on vessel HWRs. *ibid.*
18. Cartwright H. The commercial SGHWR. *ibid.*

Corrigendum

It is regretted that in the Report on the BNES Conference on steam generating heavy water reactors (July, p. 199) the late head of Siemen's Reactor Department was named incorrectly. He was in fact Professor Wolfgang Finkelnburg.

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1st COCKCROFT MEMORIAL LECTURE

Sir John Cockcroft and Atomic Energy

Lord Penney, KBE, FRS

Rector, Imperial College of Science and Technology, London

INTRODUCTION

JOHN DOUGLAS COCKCROFT was an extraordinary man. Primarily he was a scientist, but his wisdom and breadth of vision, his humanity, his strength and his energy gave him an authority and an influence in national and international affairs of science which were quite outstanding. During his lifetime, science and technology were expanding at a phenomenal rate. Ever increasing sums of public and private money were being directed towards science and technology, and technological innovation became widely accepted as the corner-stone of industrial growth. The making of policy, the management and the leadership of projects involving large groups of scientists, engineers and administrators demanded a new type of man—a man who would take part in decision-making on the grand scale and could at the same time take direct charge of the planning and the management. Cockcroft was such a man, and for more than thirty years he played a leading role in influencing British science and technology. Yet he never lost his interest in the details of contemporary science and engineering—indeed his masterful grasp of what was scientifically significant gave his opinions the great weight that they had.

Cockcroft developed an interest in physics at school and formed an ambition to do research work in experimental physics. He was advised to study mathematics first, and in 1914 he enrolled at Manchester University. From 1915 to 1918 he was in the army, and survived some of the most bitter engagements of the First World War. He then returned to Manchester and obtained a technology degree in 1922, having spent part of his time as a College apprentice with the Metropolitan Vickers Company. He went to Cambridge, and in 1924 distinguished himself in the Mathematical Tripos, Part II, as a B* Wrangler and obtained the BA degree. The solid foundations in electrical engineering and in mathematics which he had laid stood him in good stead for the rest of his life. After graduating at Cambridge, Cockcroft began research in physics. He was one of the remarkable group of men inspired by Rutherford to work on the atomic nucleus. Cockcroft's research gained world recognition and, as everybody knows, it won for him and his colleague E. T. S. Walton a Nobel prize for Physics in 1951.

This lecture was given to the British Nuclear Energy Society on 13 June 1968.

Whether Cockcroft will be best remembered for his personal scientific work, for his influence in national and international science or for his work as a leader in atomic energy is perhaps not an important question because his contributions to all three were enormous. Certainly it is impossible to attempt to describe his work in atomic energy without telling much of the story of atomic energy and mentioning many names. I have chosen to speak about Cockcroft and Atomic Energy because I was closely associated with him for over twenty years in the British effort and this is the part of his life's work about which I can speak best.

ATOMIC ENERGY DURING THE SECOND WORLD WAR

In 1938, Cockcroft was drawn into radar by Sir Henry Tizard, and for the next few years he was almost wholly engaged on radar work, but he played a part in an even more secret matter, atomic energy. Scientists in several countries had seen the possibility that nuclear fission might enable a devastating weapon based on nuclear energy to be made. In Britain, during the first year of the Second World War, a military fantasy changed to a grim possibility through the work of a small number of theoretical and experimental scientists, including some who had come to Britain to avoid Nazi persecution and had become British. A committee, called the MAUD Committee of the Ministry of Aircraft Production, was formed to assess the feasibility and cost of nuclear weapons. Professor G. P. Thomson was the chairman, and Cockcroft was a member.

The MAUD Committee produced two reports in July 1941, one on 'Use of Uranium for a Bomb', and the other on 'Use of Uranium as a Source of Power'. The response of the British Prime Minister, the organization of responsibilities under Sir John Anderson, and the creation of Tube Alloys in the Department of Scientific and Industrial Research, the effect of the MAUD Committee reports and British Government actions on the United States and Canada, the remarkable growth of the United States Manhattan District project, and the political, commercial and procurement difficulties involving Britain, Canada and the United States are all well documented in several official histories and in various published books and journals. In particular, Margaret Gowing's book, 'Britain and Atomic Energy, 1939–45' should be quoted.

MONTREAL AND CHALK RIVER

The official British proposal that a team of British and French scientists and engineers should move to Canada to work with Canada on atomic energy was made on 24 September 1942, after preliminary soundings, including one to Dr V. Bush in Washington. The proposal was agreed and the President of the Canadian Research Council, Dr C. J. Mackenzie, quickly arranged that the collaborative endeavour would use laboratory accommodation in the unfinished Medical wing of the University of Montreal. The outlook was not promising because there was no agreement by the United States on the supply of essential materials or information. At about this time, relationships in atomic energy between the United States on the one hand and Britain and Canada on the other, were at a low ebb.

The Quebec Agreement of August 1943, and the move of Professor J. Chadwick to Washington to lead British atomic energy affairs in the British Mission greatly improved relationships. Chadwick had it in mind to get Cockcroft as director at Montreal and he explained to Cockcroft the plans he hoped would be agreed for Montreal and suggested that if they were approved, Cockcroft should lead the laboratory. Chadwick introduced Cockcroft to Major General L. R. Groves, the man in charge of the Manhattan District project, in December 1943, and Cockcroft then visited Montreal. Mackenzie was pleased by the possibility that Cockcroft would lead the Montreal laboratory: Groves said that the choice of the leader was a matter for Chadwick and Mackenzie.

The combined Policy Committee in Washington on 13 April 1944, with Mr H. L. Stimson in the chair, accepted the proposal that a heavy water pile of moderate but adequate size was to be built in Canada, with American help.

Cockcroft reached the Montreal Laboratory on 25 April 1944. He quickly visited the Chicago Metallurgical Laboratory, the Argonne Laboratory near Chicago and the Clinton Laboratory near Knoxville. He then visited various sites which were being considered for the Canadian pile, and approved the Chalk River site. Under his calm but energetic leadership the laboratory gained a firm sense of purpose. He personally guided the planning of the lay-out of the Chalk River site. At Montreal, the design and development of the 10 MW heavy water reactor NRX, and its back-up zero energy assembly ZEEP, were vigorously pursued, and construction of NRX at Chalk River began in the autumn of 1944. Cockcroft and many of the staff moved from Montreal to the new town site at Deep River towards the end of 1945. Chemical processes began to be devised for recovery of uranium from the irradiated fuel as well as for separation of uranium-233 from irradiated thorium rods in the reflector. American information on the chemistry of plutonium was not available and the primary separation processes for plutonium were worked out, but only on a semi-micro scale, using American slugs from Clinton. By the time Cockcroft left Canada in the autumn of 1946, the reactor was almost complete, the laboratories were fully occupied and the town had become a settled community. The NRX heavy water reactor later proved to be one of the most valuable research reactors in the world because of its high neutron flux: it also became the forerunner of successful developments in Canada on the use of atomic energy for power purposes. The lay-out of the Chalk River site has stood the test of time and is a physical testimonial to the vision and personality of Cockcroft.

ATOMIC ENERGY IN POST-WAR BRITAIN, 1946–54

Anderson in 1944 was already considering what should be done about atomic energy in Britain after the war. One part of the problem was easy. Britain must surely have a large research establishment and Cockcroft was the obvious man to take charge.

The end of the war came in August 1945, and the General Election in October returned a Labour Government, with Mr Attlee as Prime Minister. Attlee endorsed the offer to Cockcroft to become director of the research establishment and persuaded Lord Portal of Hungerford to become Controller of Atomic Energy (Production) and Mr C. Hinton to become Deputy Controller (Production). Sir Oliver Franks, Permanent Secretary of the Ministry of Supply, had agreed with Hinton that if it was possible to evolve civil applications for nuclear power, the engineering development of these should be the responsibility of the Production organization. Hinton invited his colleague W. L. Owen to join him.

Britain was to establish a research centre, to obtain practical experience of atomic energy on the industrial scale, and would make plutonium. No decision was made about starting an ordnance programme for atomic weapons. By implication, there was to be no immediate start on a diffusion plant.

The top level arrangements of responsibilities were exceptional and from the point of view of the Civil Service were untidy. Cockcroft was not under Portal, but reported to Franks; Portal, Cockcroft and Hinton were all in the Ministry of Supply, but Portal had direct access to the Prime Minister. Nevertheless, the full resources of the Ministry of Supply and, through this Ministry, of other Government Departments, were essential, were available and were used. Portal formed two committees and normally took the chair at both: in his absence, Cockcroft took the chair. These were the Atomic Energy Council whose membership included Cockcroft, Hinton, M. W. Perrin and F. C. How, and the Atomic Energy Technical Committee, a large body whose membership included the members of the Council, a few of the senior staff of Harwell and Risley and several of the senior people who had been involved in the British war-time effort on atomic energy and who were mostly back in the universities.

Harwell—early days

Early in the summer of 1945, M. L. E. Oliphant and Cockcroft had suggested that one of the permanent airfields in the Oxford area should be taken over as the site for the atomic research establishment and Cockcroft proposed Harwell. Anderson's advisory committee agreed and the Air Ministry gave immediate consent.

Cockcroft received the formal offer of appointment from Franks in November 1945. The salary was £2000 per annum and the appointment was for five years in the first instance. The letter included an assurance that subject to safeguarding information of military importance the establishment would have considerable freedom in regard to publications.

Cockcroft accepted the offer, and for nine months, until W. B. Lewis succeeded him in Canada, he was both Director of Harwell and Director of the Chalk River and the Montreal laboratories, and was constantly commuting across the Atlantic.

Possession of the Harwell site was taken on 1 January 1946. H. W. B. Skinner was the first Division head to arrive and he acted as Cockcroft's local deputy. Cockcroft was fully informed on the more important appointments and took an active part in finding good staff. Industrial relations were in the capable hands of the Chief Engineer, H. Tongue.

The initial programme of the establishment included

(a) a low powered graphite moderated pile (GLEEP) to be used mainly for testing the purity of uranium metal and graphite for future reactors (commissioned 1947);

(b) a 6 MW (thermal) graphite moderated research reactor (BEPO), to be designed by the Industrial group at Risley (commissioned 1948);

(c) an electromagnetic separator for uranium isotopes to produce small quantities of ^{235}U;

(d) a Van de Graaff accelerator to produce 6 MeV protons for nuclear data work;

(e) a synchrocyclotron to produce protons of energy 180 MeV;

(f) a 'hot' laboratory for radio-chemistry;

(g) office accommodation and some general purposes laboratories, to be obtained by converting existing buildings.

Some of the early planning of the development of the Harwell site was done at Montreal and Chalk River, and the detailed planning of the hot laboratory was mostly done at Montreal. The construction of buildings and facilities at Harwell was the responsibility of the Ministry of Works, and a special team was put at Harwell. The Director General, Sir Charles Mole, spent a lot of time at Harwell, and did much to ensure rapid and effective progress. Cockcroft was well informed on the building plans. He was interested in architecture, and was very knowledgeable, and he was an excellent critic of plans of buildings for scientific work. He already had considerable experience, with the Mond laboratory at Cambridge, and at St John's College, Cambridge, where he had been junior bursar while a restoration and new building programme had been in progress, and also of course in the lay-out and detailed designing at Chalk River.

Among the many urgent technological problems of atomic energy, where Cockcroft and his staff realized that solutions might be difficult to obtain, two where the uncertainties were most intangible may be mentioned—graphite and irradiated fuel chemistry.

The graphite for BEPO and the plutonium production piles needed to be of exceptional purity, especially as regards certain elements, determined by nuclear considerations. Cockcroft had formed early in 1945 a graphite committee in Canada; J. V. Dunworth led this work. Early in 1946, Cockcroft arranged for measurements to be made in ZEEP of the cross section of Canadian graphites coming from a plant at Welland, Ontario. These tests indicated a preference for a particular petroleum coke as raw material and the need to eliminate the metallurgical coke used in commercial graphitizing furnaces. The work led to the use of Canadian graphite in BEPO and, initially, in Windscale.

At Harwell, Cockcroft put J. M. Hutcheon in charge of a programme to develop a suitable production route, with co-operation from a number of industrial firms and Government and University laboratories. This work, with the particular help of the Shell Company, led to the replacement of residue cokes by distillate cokes as starting materials. Trial quantities of these cokes were made at Shell refineries and converted to graphite, first in a small pilot plant built and operated by British Aluminium Co. at Kinlochleven, Scotland, and then in the plants at Welland and the associated company, British Acheson Electrodes, at Sheffield.

This graphite proved to have a capture cross section about 20% lower than the previous nuclear grade and was satisfactory in other respects. It became a standard material for the Windscale, Calder and the magnox reactors, and was known as PGA (pure graphite grade A).

The second technological set of problems was related to the design and construction of the chemical separation plant at Windscale to separate plutonium and uranium from irradiated fuel, and deal with the fission products. The initial designs of this plant had to be based on the semi-micro scale work at Chalk River done on the Clinton slugs, but fuller information was needed. Cockcroft arranged that a team of British chemical engineers would build a pilot extraction tower at Chalk River, and the development of the butex process was done at Chalk River by a Harwell team led by C. R. Nicholls, using the first irradiated fuel coming from NRX for their experimental work.

Atomic energy policy and central direction

The original remit to Hinton about the plutonium production pile was to design and build one resembling the Hanford reactors and using similar siting criteria. A possible site was found but it was extremely remote and completely lacking in facilities. Attention was therefore turned to a gas-cooled reactor, and work by H. H. Gott and E. H. Lee at Risley and J. Diamond at Harwell showed that a scaled-up version of BEPO, using finned fuel cans and air cooling at atmospheric pressure was feasible, with much less demanding site conditions. The decision to build the two Windscale piles was made possible by this work, but the interest in gas cooling had also considered carbon dioxide cooling. A small effort at Risley was put on the carbon dioxide cooled reactor with help from Parolle (the subsidiary of C. A. Parsons and Co.). Cockcroft organized the basic supporting work at Harwell.

Meanwhile, tense relationships between the Soviet Union and the West had led the Chiefs of Staff to consult Portal in the late spring of 1947 about a nuclear weapons programme. Portal made some proposals and saw the Prime Minister. A start was made on planning the ordnance side and staff to undertake the work began to be assembled by W. G. Penney. The priority task of the British atomic energy programme was soon to become the production of a small number of nuclear weapons. Late in 1947, Portal and his Atomic Energy Council decided to strengthen the limited research and development effort which had been continued on diffusion plants, with the construction of a

major plant in view. Hinton and Owen at Risley began to re-assess the situation, using the flowsheets and the membrane work which ICI Billingham had done for Tube Alloys. By 1950, the Chiefs of Staff were asking for an increase in the established military programme. The construction of the diffusion plant at Capenhurst was authorized. The Production Group was given the formidable responsibility for its development, design, construction and operation. Owen chaired the design committee, of which Cockcroft was a member.

The British atomic energy project continued to expand rapidly. In 1951 Portal decided that he could relinquish his post as Controller and he handed over to his successor, Lieut. General Sir Frederick Morgan.

The General Election of 1951 brought the Conservatives back into power under Winston Churchill. Lord Cherwell became Paymaster General and Mr Duncan Sandys became Minister of Supply. Cherwell and Sandys agreed that Cherwell should set up an Atomic Energy Council, consisting of Cherwell (Chairman), and Morgan, Cockcroft, Hinton, Penney, How and a secretary. The increased military programme required by the Chiefs of Staff would not be fully met by the two Windscale piles and Capenhurst. A new possibility was seen. The small amount of design–assessment work being done at Risley, with help from Parolle, and the applied research effort at Harwell, on a power reactor with carbon dioxide cooling had been picked up by a composite team centred at Harwell under H. L. Goodlet, with participation by the Ministry of Works, the Central Electricity Authority and Babcock and Wilcox. Excellent progress on a conceptual design had been made by R. V. Moore on a power reactor called PIPPA. The requirement put by the Chiefs of Staff was for plutonium, and Hinton, Owen and Cockcroft agreed to change the basis of the PIPPA design to make plutonium production the primary consideration and power a by-product. Cockcroft readily agreed that Moore should be transferred to Risley. By late 1952, Hinton was able to propose to Cherwell's Atomic Energy Council that a dual-purpose reactor of a modified PIPPA type should be built at Calder Hall, primarily to make plutonium for the military programme but also to generate electricity for the national Grid. Government approval was given early in 1953, and for the first time in Britain the vision of nuclear power began to have some practical endorsement. Soon after the detailed design of the Calder reactor began, the advantages of building a pair of reactors became decisive, and the plans were modified. Construction work started and the first Calder reactor produced steam in the summer of 1956, barely 3½ years after authorization.

The Atomic Energy Authority

Cherwell had never wholly supported the arrangements which placed atomic energy in a government department. Indeed, when Attlee was Prime Minister, Cherwell tried to persuade him that an independent body under broad Government control would be more efficient; but after considering an analysis by How of the pros and cons, Attlee ruled in February 1951 that the Ministry of Supply arrangements should continue.

Later that year, with the Conservatives in power, Cherwell renewed his efforts, at first not successfully. However, a small Cabinet committee was set up to examine the matter, and this committee was strongly influenced by the performance expected from the Calder reactors. Radioactive isotopes, although much less important, also promised to have wide-ranging civil applications. The Cabinet Committee recommended in January 1953 that a small committee should prepare a detailed scheme for the transfer of responsibility to a non-departmental organization and work out the most suitable form for the new organization. The Prime Minister announced in the House of Commons that the Government had appointed a committee consisting of Lord Waverley, Sir Wallace Akers and Sir John Woods to devise a plan for the organization of atomic energy.

Waverley, Akers and Woods visited the atomic energy sites, talked with Cockcroft, Hinton and Penney, discussed the problems with government ministers and officials and consulted the Opposition and some industrialists. They recommended the formation of the Atomic Energy Authority. The drafting of the necessary legislation by How and the parliamentary lawyers closely followed their proposals and became law in the Atomic Energy Act of May 1954.

The Minister made responsible for the Atomic Energy Authority was the Lord President of the Council, who at that time was the Marquess of Salisbury. He appointed Sir Edwin Plowden as Chairman: Cockcroft, Hinton and Penney were appointed as full-time technical members, and Sir Donald Perrott became the Member for Finance and Administration. How took charge of the office of the Lord President of the Council (Atomic Energy Department) and was the Accounting Officer for the Atomic Energy Authority, although for their own purposes the Authority also had a commercial accounting system.

The work-load on the newly formed Atomic Energy Authority was heavy and was increasing rapidly. While the titles of the full-time technical members were clear, the Member for Research (Cockcroft) and the Member for Engineering and Production (Hinton) did not work solely or even mainly as functional members, with Authority-wide responsibilities matching these titles. Cockcroft remained the Director of Harwell, until B. F. J. Schonland succeeded him in this post in 1958; and Hinton, in addition to his Membership, was also Managing Director of the Industrial group with Owen as the Director of Engineering and Deputy to the Managing Director.

Harwell and Risley 1950–55: nuclear power

The 'hot' laboratory at Harwell was urgently needed, but it was a complicated building. By a tremendous effort from all concerned, a part of the building was handed over to the Chemistry Division under R. Spence in July 1949, but the plutonium wing for the Metallurgy Division, under H. M. Finniston, was not ready until the following year. The commitments of the production programme at Windscale could only be met by the narrowest of margins and by the Industrial group taking many short cuts. The first full-scale plutonium billet was made in the laboratory at Harwell on a Saturday evening in Cockcroft's presence on 12 December 1951, by which time the primary separation plant at Windscale was already operating inactively, and the construction of the plutonium purification and metal finishing plant was almost complete.

The technology of graphite periodically gave cause for anxiety. Harwell was studying the research problems and irradiation experience was being gained. Not enough had been known in Britain about the differential growth and shrinkage rates of graphite, about the storage of radiation energy damage, or about corrosion rates, to allow reliable estimates of the life of the Windscale piles to be made; and regular Wigner energy releases were to be made in these piles. More information on graphite was available by the time the Calder reactors were being designed and the design of the graphite core of these reactors kept all parts of the graphite structure as hot as possible: the predicted energy storage would never reach a level where energy release would be required. Work on graphite received a tremendous stimulus as a result of the accident to a Windscale pile in 1957. An Authority-wide graphite project under A. H. Cottrell of Harwell was organized on a crash programme. Cockcroft closely followed this work. Graphite work continued to have high priority for many more years, because of the steadily increasing severity of the reactor parameters in the nuclear power programme.

Uranium procurement was always in the minds of those in charge of the British atomic energy work. Nothing was ever stopped because the uranium supplies were not obtainable but the need for uranium conservation influenced the directions in which the programme was guided. The diffusion plant could, if necessary, use depleted uranium recovered from irradiated fuel. One of the arguments favouring the Calder type of reactor, rather than a third Windscale reactor, was the higher fuel rating. By 1951–52, the fears about procurement of uranium in the long term were at one of their peaks and at about the same time the Industrial group knew that some design effort would become available within a year or two. Hinton and Cockcroft agreed that attention should be paid to the fast reactor concept. A design committee was set up at Risley, with Hinton in the chair, to look at the design possibilities. Cockcroft nominated C. A. Rennie and F. W. Fenning to represent Harwell. Cockcroft arranged that the nuclear physics division would plan a programme of measuring certain cross sections, and, at Cockcroft's request, enough plutonium was to be borrowed from the weapons programme for Dunworth to start work in 1952 on a zero energy plutonium fast reactor assembly (ZEPHYR) for integral experiments. However, there was no 'industrial' programme until 1954 and the first step towards such a programme was a report written by J. M. Kay, G. R. H. Geoghegan and D. R. Poulter. Early in 1955, Government approval was given for building a large experimental fast reactor at Dounreay, and construction began in March 1955.

The year 1955 also saw two other important events in atomic energy. Three more pairs of reactors copying the pair at Calder Hall were to be built, to increase the rate of production of plutonium for the military programme. One pair was to be sited at Calder Hall, and the other two pairs at Chapel Cross. The second important event was the publication of the Government White Paper 'A programme for nuclear power' (Command 9389). The White Paper suggested the construction of 2000 MW of electrical power over ten years. At about this time and for several more years, world uranium resources appeared plentiful at least for a decade or more, and the Authority were able to place several advantageous long-term contracts.

The Research Group and the Authority, 1956–59: nuclear power

The major decisions of 1955 substantially increased the work load on the Authority and the total number of Authority staff continued to increase. British industry was organizing itself for a large programme of nuclear power stations at home and abroad. The Industrial group had a clearly defined programme which involved a great deal of new construction. Harwell was bursting with ideas about possible reactor systems. At Harwell, DIDO was commissioned in 1956, although by then a scheme to combine a heavy water distillation plant with an electricity generating station in New Zealand, based on geothermal steam, had been shown to be uneconomic. Thermal reactor systems being studied at Harwell for feasibility included the enriched oxide pressurized light-water system LEO, the aqueous homogeneous reactor, the liquid-metal fuelled reactor, and the high temperature gas-cooled reactor. Harwell set up a Reactor School in 1954 and later, in 1957, the Calder Operations School opened.

The Atomic Energy Authority decided that too many reactor types were being investigated. A committee called the Committee on the Reactor Programme was formed under W. Strath in 1956 to make recommendations about limiting the programme. By then, the organic moderated system and the gas-cooled heavy-water moderated system also had their protagonists, and the situation was getting out of control. There was considerable external pressure on the Authority to start a project on nuclear marine propulsion. Some assessments were made, but the Authority were unable to see any promising prospects of an economic system. Submarines, where the economics were secondary to operational performance, were, of course, another matter.

Cockcroft considered that it was the duty of the Research Group to study advanced reactor concepts and to test the most promising ideas in small experimental reactors. There was no prospect of building more reactors at Harwell. Dounreay was a possible site, but it was remote and staff were not easily attracted. Cockcroft recommended to the Atomic Energy Authority in 1957 that a new experimental reactor establishment, to be part of the Research Group, should be built in Dorset at Winfrith. The proposal was controversial but Cockcroft's view prevailed. Construction work at Winfrith started in 1958, and within a year reactor physicists and engineers began to move from Harwell. A number of zero energy facilities were built, and others were to be moved from Harwell and rebuilt, with improvements.

Among all the longer-term possibilities for land based thermal nuclear power stations, Cockcroft had begun to favour a high temperature helium-cooled converter type of reactor using dispersed fuel. The Industrial group in the shorter term favoured the advanced gas-cooled reactor (AGR) and the planning of a test reactor to be built at Windscale began in September 1957. The Authority did not have sufficient resources to design and build the AGR and also design and build an experimental high temperature gas-cooled reactor. Cockcroft therefore spoke informally to a number of the atomic energy leaders in Europe and enlisted their support for a collaborative scheme at Winfrith. Cockcroft also promised support for the Halden project in Norway. In 1958 the Authority agreed officially with the

European Nuclear Energy Agency to participate in the Halden project, and in March 1959, the Dragon agreement was signed at the OECD headquarters in Paris. Later on, after Cockcroft had become Master of Churchill College, his plan of Winfrith as a site for experimental reactors was further fulfilled by the siting there of the large zero energy critical facility ZEBRA for work on the physics of fast reactor cores, and by the siting there of the 100 MW(e) demonstration power station SGHWR.

Cockcroft resigned as a full time member of the Atomic Energy Authority at the end of May 1959, to become Master of Churchill College, Cambridge, but he wished to continue spending as much time as possible keeping in touch with developments in the Authority; and as part of the arrangements which were made, he became a part-time member of the Authority, and remained so until his 70th birthday in May 1967.

ACTIVITIES CONNECTED WITH ATOMIC ENERGY

While Cockcroft was engaged in atomic energy work he spent a considerable proportion of his time on matters which were closely connected with atomic energy. Immediately he went to Montreal, he arranged that some work would be done there on radiation biology. Later, he was the prime mover in having a unit of the Medical Research Council, under J. F. Loutit, sited next to Harwell. He served on various MRC committees on radiation matters, and several Harwell staff, notably Catherine Williams and W. G. Marley, also made substantial contributions. He suggested and helped to arrange the American–British–Canadian conferences on health and safety matters. Three conferences were held between 1949 and 1953, and notable progress was made on defining tolerance levels. Consistent with his philosophy of opening atomic energy to the outside world as far as possible, he was influential in declassification policy.

Cockcroft also took a keen interest in radioactive isotopes and in 1950, the Amersham site of the present Radiochemical Centre and its director, W. P. Grove, were taken over by the Ministry of Supply and came under Cockcroft. Isotopes were made in the Harwell reactors, and at Harwell research on the use of isotopes expanded under H. Seligman. Cockcroft formed a Technical Irradiation Unit at Grove Airfield, near Wantage, in 1956. Research workers and students came from many countries to work at the Harwell isotopes school.

High energy physics was another field where Cockcroft's influence was important. He was able to help several universities in the immediate post-war years, through his nuclear physics and electronics staff at Harwell and Malvern. He was one of the European scientists who worked to get approval of Governments for a high energy physics research centre, the outcome of which was CERN. He was also a central figure in the creation of the Rutherford Laboratory of the National Institute for Nuclear Science, and the ingenious and effective scheme which used the resources of the Authority to build up this laboratory owes much to him.

Plasma physics, or controlled fusion, was a field of science which Cockcroft was able to support directly through his position in atomic energy. The largest part of the effort in Britain in the 1950s was at Harwell, and before Cockcroft left the Authority for Churchill College, the plan of a separate, open site for controlled fusion work at Culham had been effectively agreed. Cockcroft persuaded J. B. Adams to give up his post of Director General of CERN to lead Culham.

Cockcroft was always active in international atomic energy affairs. He was one of the scientists who supported President Eisenhower's proposals for a bank of fissile material to be held by an agency of the United Nations and used only for peaceful purposes, and for a United Nations Conference on the peaceful uses of atomic energy. He served on the scientific advisory committees of both the United Nations and the International Atomic Energy Agency. He led the British delegation to the 1st and 2nd Geneva Conferences, and at the end of the 2nd Conference he gave a brilliant summary of the proceedings.

Cockcroft was also active in several other international committees or groups connected with atomic energy. One must be especially mentioned—not because it took up a lot of his time, or because his contributions were any greater here than in the others, but because the purpose and motives reflected his philosophy, especially his philosophy in the last few years of his life. He had become more and more deeply worried about the risks and consequences of nuclear war, and yet he was so completely convinced that nuclear power, sooner or later, was vital to mankind. The Pugwash Conferences gave him a forum for showing his thoughts, while staying strictly within a technical brief. His advice was sought about the British membership of the Continuing Committee, the body responsible for organizing the conferences and guiding the movement. In 1959 he became a member of the Organizing Committee for the 9th and 10th Conferences, which were held at Cambridge and London. He gave special attention in 1961 and thereafter to the problems of control of fissile materials and the development, under control, of the peaceful uses of atomic energy. He clarified many of the technical problems of a comprehensive nuclear test ban treaty.

Cockcroft was elected to the new post of President of the Pugwash Conferences on Science and World Affairs at the large Conference held at Ronneby in September 1967, at which extensive plans for new Pugwash activities were adopted. In undertaking the Presidency, Cockcroft expressed deep concern about the implications of the progress of science and technology. He gave one of the keynote papers on 'The Control of the Peaceful Uses of Atomic Energy'. The paper was precise, informative and stimulating, and was used as the basis for the first Pugwash Symposium on this topic, held in London in April 1968. At Ronneby he worked, with obvious enthusiasm, on plans for raising funds and implementing the new programme of Pugwash activities, but his death ten days later abruptly ended these endeavours.

Cockcroft's contributions to Pugwash completed his work on atomic energy. Inevitably, he had been drawn into atomic energy during and after the war, when the purpose was military, but as soon as he could his interests were focused on the peaceful uses. His contributions were immense. His lectures and speeches about nuclear power were always models of technical accuracy. When the international situation allowed purposeful discussions to be held on the possibility of disarmament and control of atomic energy against misuse, he worked hard to provide

for political leaders a sound evaluation of the technical problems that have to be solved before political actions can happen. He will always be remembered as one of the great scientists whose scientific originality led to atomic energy; who played a leading role in atomic energy; and who did so much to guide the world in using the energy of the atomic nucleus only for peaceful purposes and the betterment of mankind.

Many colleagues have given me substantial help in recalling to memory parts of the story I have tried to tell. My most grateful thanks are expressed to Miss J. M. Pye and Mrs L. H. Arnold, Sir James Chadwick, Lord Hinton, Sir Leonard Owen, Dr W. B. Lewis, Professor R. Spence, Professor J. Rotblat, Dr J. B. Adams and many colleagues in the Atomic Energy Authority who have helped me on specific points.

A loop designed to investigate the aqueous corrosion (particularly crevice corrosion) of mild steel under reactor radiation at 40°C is described. The loop was installed in an experimental hole in the BEPO reactor at Harwell, and the design conditions for the cooling water pipes of prestressed concrete reactor pressure vessels were simulated. Details of specimen preparation and the design, construction and commissioning of the loop are presented, with particular attention to water quality control.

Studies on the low temperature aqueous corrosion of mild steel under reactor radiation

Part I: Design, construction and commissioning of an experimental loop in the BEPO reactor

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INTRODUCTION

CURRENT designs of prestressed concrete pressure vessels (CPVs) for nuclear reactors incorporate cooling water pipes attached to the mild steel liner. Owing to the difficulty of cleaning the bore of flash butt welded pipes, sleeve joint designs, which inherently contain crevices, have been adopted. Whilst it was not expected that the bulk corrosion of the pipe surface would be a problem, some concern was expressed regarding the performance of the pipe joints, e.g. the differential water conditions existing between the main flow and the stagnant zones in the crevices might initiate corrosion cells, leading to enhanced localized attack, possibly aggravated by the radiation flux.

Because of the uncertainties in the performance of the proposed systems, it was decided to expose suitable corrosion specimens in reactor loops under conditions resembling those likely to be experienced in practice.

The chemical characteristics of the current systems are basically similar. All are designed to be operated with high purity water maintained at a pH value of $10{\cdot}5 \pm 0{\cdot}5$ with lithium hydroxide, and with a very low dissolved oxygen level. Two methods have so far been adopted to control the dissolved oxygen level. The first employs a vacuum de-aerator and the second an overpressure of hydrogen to promote recombination of radiolytically produced oxygen.

The BEPO reactor at AERE Harwell was chosen as the irradiation facility because of the need to use a mixed γ and neutron flux. A hole in this reactor, when fitted with a fast neutron booster, gave a radiation flux similar to the maximum value estimated to occur in a CPV cooling system.

Irradiation of the loop began in September 1964 and ended in June 1967. This Paper describes the design, construction and commissioning of the experimental loop. The results from specimens exposed for periods of up to 18 months will be published[1] in Part 2.

DESCRIPTION OF EXPERIMENTAL EQUIPMENT

The general arrangement of the loop is shown in Fig. 1. The pumping section cabinet contained two separate circuits, each consisting of a header tank, pump and flow controls. These provided a flow of water through four specimen assemblies, which were contained in a plug fitted into the experimental hole A2/1 in the BEPO reactor. The electrical control and protection circuits for the pumps were contained in the instrumentation control panel, which also housed two electrical safety systems to give warning of any water leakage from the in-pile components. A third cabinet, not shown in Fig. 1, contained the two meters for measuring the levels of dissolved hydrogen and oxygen in the loop water.

PUMPING SECTION

Fig. 2 shows a flow diagram for one of the two circuits contained in the pumping cabinet. This was called Rig 1 and provided flow through loops A and B. The circuit for Rig 2, which is not shown in Fig. 2, provided flow through loops C and D. The main flow route was as follows: from the header tank to the pump, through an out-of-pile control specimen, through the in-pile specimen and back into the

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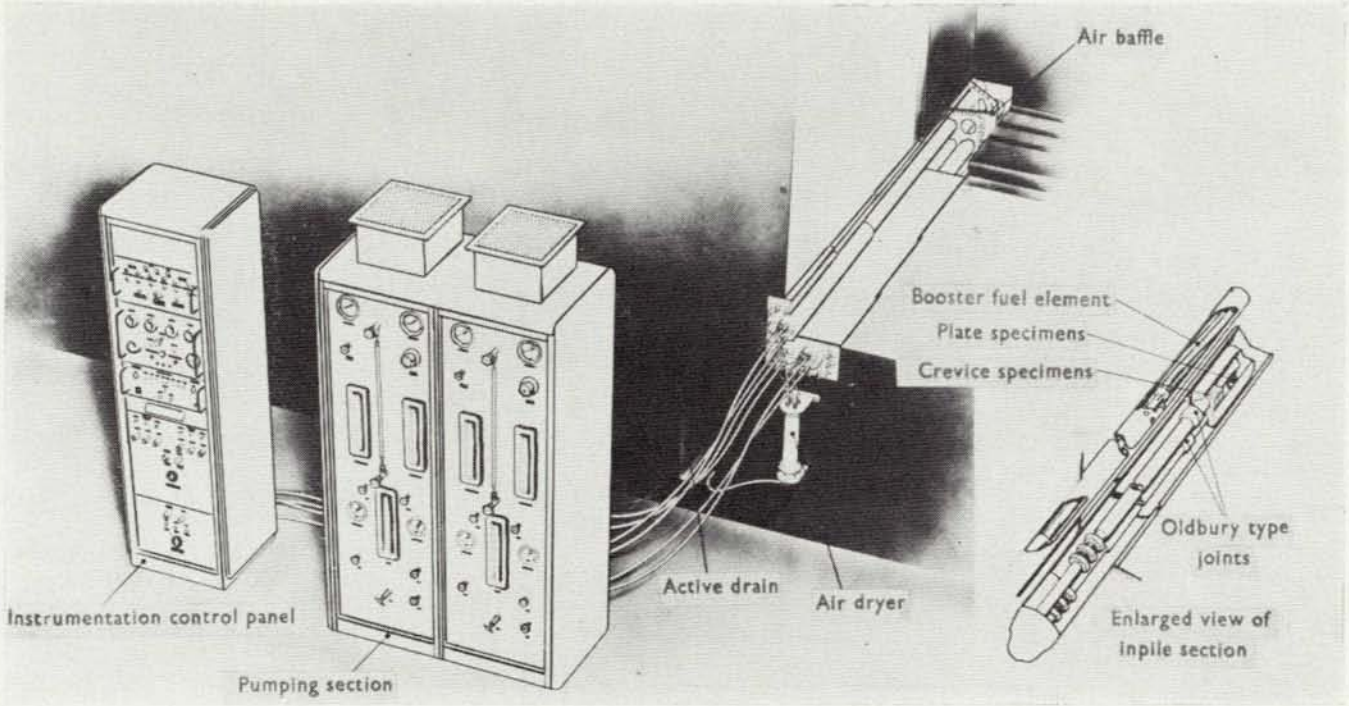

Fig 1 **General view of the rig**

top of the header tank. The flow to the loops was controlled by valves on the pump outlets and was measured by a Rotameter type flowmeter. There was also a delivery and return pressure gauge for each pump. Each header tank had a sight glass for water-level indication and a thermometer to measure water temperature.

There were two subsidiary circuits. The one shown to the left of the header tank in Fig. 2 was a water quality control and sampling circuit. It consisted of two ion-exchange columns which could be connected by valves into the bypass circuit when required, and two filters for collection of suspended solids. Flow through this bypass circuit was 1% of the main circuit flow. The other subsidiary circuit consisted of two ion-exchange columns in series supplying both rigs, and was used to control the quality of make-up water.

The free surface of water in the header tanks was covered with a hydrogen gas blanket in Rig 1 and a helium gas blanket in Rig. 2. The material used in the pumping section was mild steel except for carbon pump seals and rubber gaskets.

IN-PILE SECTION

The four specimen assemblies were housed in a plug unit (filled with iron-shot concrete and alumina-coated to prevent corrosion) which fitted into the stepped experimental hole. The four assemblies were also contained in aluminium thimbles within the plug unit, these projecting beyond the shielding into the graphite moderator and reactor coolant flow. (The thimbles also acted as secondary containment for the loop water.) The specimen assemblies were fitted to their plug units by compression ring fittings.

Because the fast neutron flux in the reactor was too low for this experiment, it was increased by placing 24 BEPO fuel elements on a 12 in. hollow square around the four specimen holders. Each element was mounted on a plug and provided with a thermocouple.

Supply and return between the pumping section and the plug unit was by trailing flexible hoses to facilitate handling procedures. The hoses were originally made of nylon but these were removed after in-pile commissioning troubles and replaced by stainless steel of convoluted construction, having an outer protective cover of stainless steel braiding.

INSTRUMENTATION CONTROL PANEL

This panel contained the electrical supply and control circuits for the circulating pumps, a fault indication unit and two leak detection units.

The first leak detection unit was an air humidity monitor which measured the moisture content of pre-dried air drawn through the specimen assembly plug thimbles. The monitor gave indication and warning of high humidity on any of the four channels.

The second unit gave a warning of loss of electrical insulation resistance between two woven glass insulated wires mounted on the specimen assemblies (two per specimen). Both these units were designed to provide an audible alarm and visual indication in the event of a water leak from the assemblies into the secondary containment.

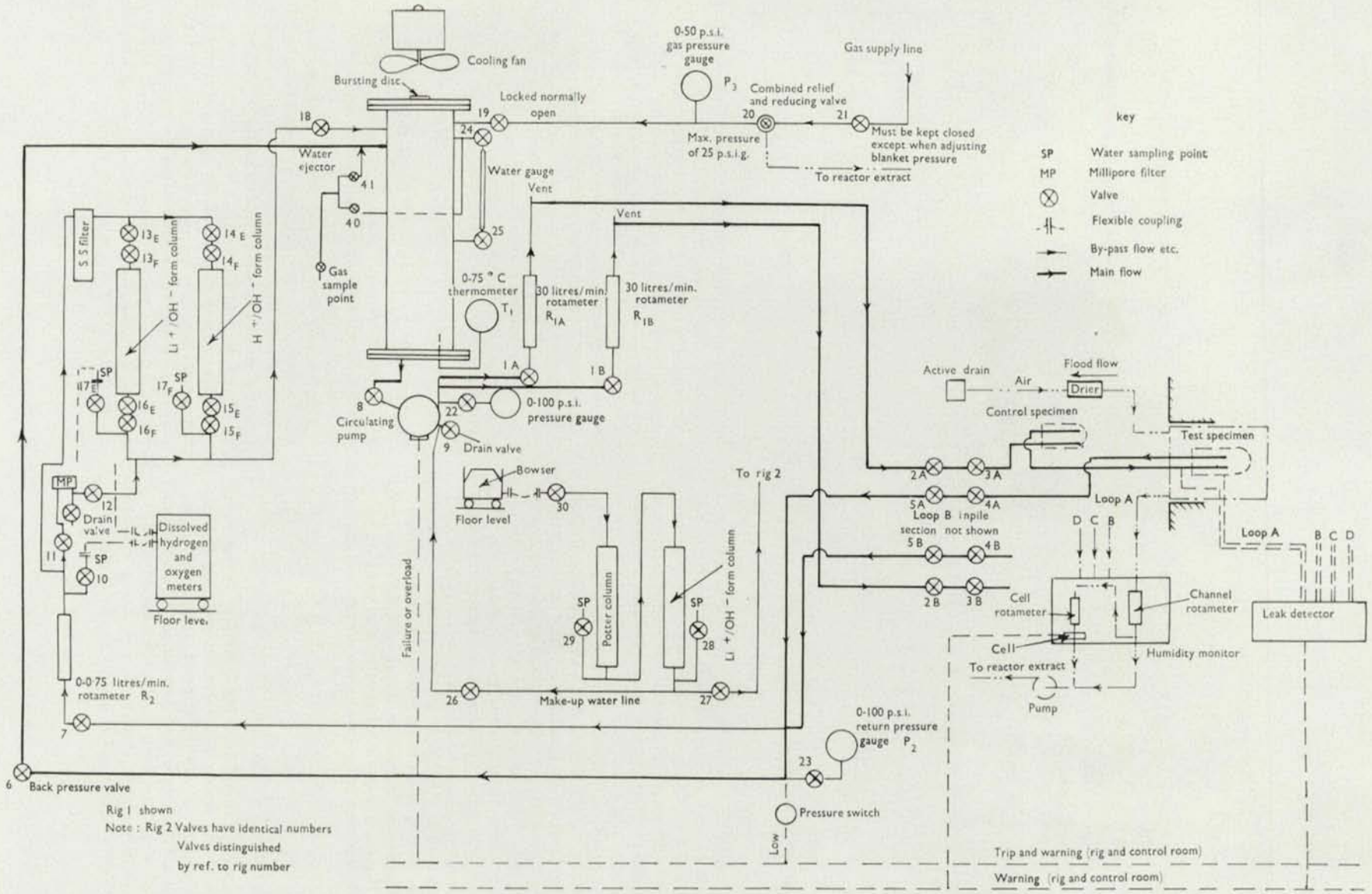

Fig 2 **Rig flow diagram**

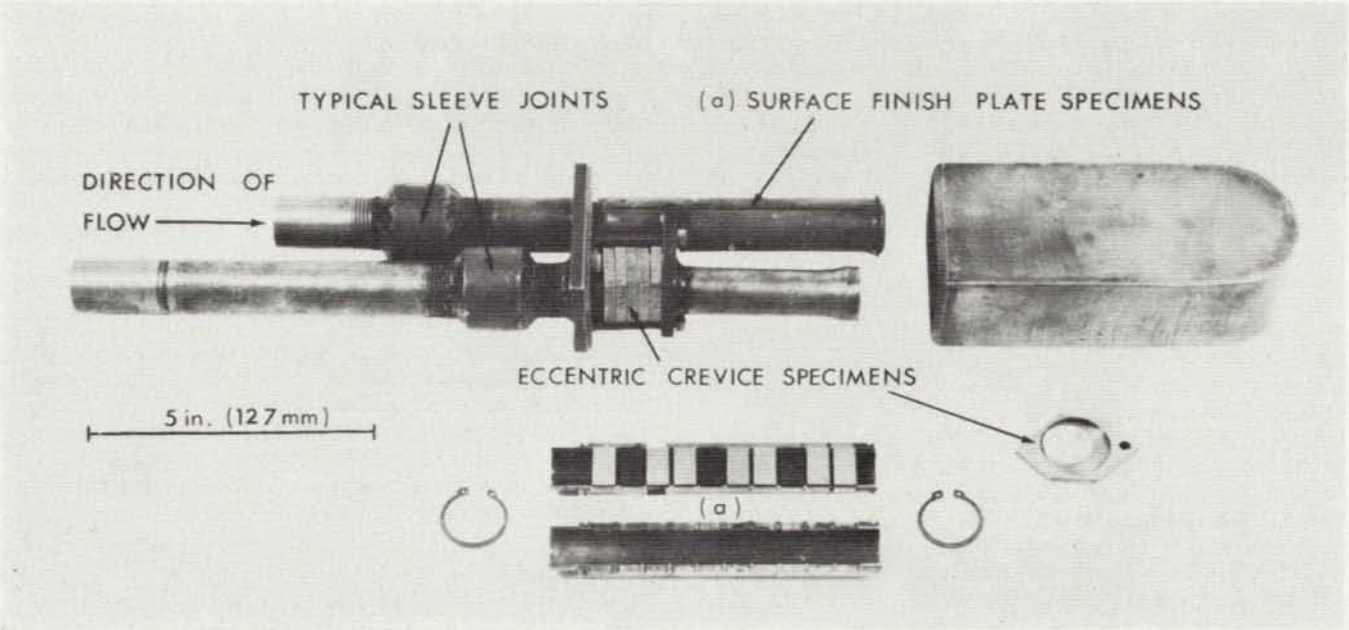

Fig 3 **Layout of a specimen assembly**

CONSTRUCTION OF SPECIMEN ASSEMBLIES

Typical joint pipe-runs were welded into an end plate, which in turn was welded to the assembly dome (Fig. 3). Crevice specimens were machined out of individual plates to form recesses eccentric to the bore of the return pipe, the plates being held together by a nut and screw thread. The plate quantitative corrosion specimens were held in a split barrel, which was located in the end plate. For each set of in-pile specimens, there was a corresponding identical out-of-pile arrangement. The specimens could be removed for inspection when the dome had been cut away.

The samples of typical sleeve joints when received were closed at each end with a polythene cap. They were cleaned and degreased with acetone and then dried in an air stream. In general, the surface of the pipe sections was covered with a dark, shiny, adherent oxide film broken in a few places by rust spots (often along the seam welds). In several instances, the welding process had resulted in melting of the inner surface of the pipe.

The simulated crevices were prepared by mounting together a series of plates with eccentric recesses of varying widths machined in them (Fig. 3). All surfaces had a fine turned finish and were cleaned and degreased in acetone prior to assembly. All the recesses varied in depth from 0 to 0·375 in. (0 to 9·525 mm) around the eccentric groove. Widths were 0·002, 0·005, 0·015, 0·025, 0·050 and 0·100 in. (0·05, 0·125, 0·375, 0·635, 1·27 and 2·54 mm respectively).

Quantitative corrosion specimens were of steel cut to $20 \times 10 \times 2$ mm thick. They were marked for identification purposes (1 to 10) heat-treated at 1000°C for 2 h under vacuum (10^{-5} torr) and then cooled to below the eutectoid transformation in about 20 min. The microstructure consisted of small equiaxed grains about 35 μm in diameter. The ten specimens contained in each holder (Fig. 3) were given three different surface treatments. These were:

(a) ground on all faces to a 600 grit carborundum finish (specimens 1, 4, 7 and 10);

(b) ground on all faces to a 600 grit carborundum finish, then metallographically polished to a 0·25 μm diamond finish on the two major faces (specimens 3, 6 and 9);

(c) ground on all faces to a 600 grit carborundum finish, then immersed in a stirred bath of a 5% solution of inhibited HCl for 0·5 h at room temperature (specimens 2, 5 and 8). The surface of the specimens then exhibited, in general, a yellowish brown coloration. Finally, the specimens were annealed at 240°C for 40 min under one atmosphere pressure of nitrogen.

This chemical and heat treatment was an attempt to simulate the site treatment of the pipe surfaces near the welds.

All specimens were degreased in acetone, dried, weighed and then loaded into numbered holders with specimen number 1 at the water inlet end (Fig. 3).

WATER CONDITIONS

Prior to in-pile operation, the loops were filled with demineralized water at a pH of 10·5 (using lithium hydroxide) which was circulated for several weeks to clear up the circuit.

At the start of in-pile operation, the loops were refilled with fresh demineralized water through the additional purification system shown in Fig. 2. This consisted of two

ion-exchange columns in series, the first a Li^+ form Potter column which deoxygenated the water, and the second a mixed bed resin in the Li^+/OH^- form to reduce further the impurity level. The former resin was prepared by converting H^+ form resin to the Fe^{++} form with ferrous sulphate, washing free from sulphate and treating with lithium hydroxide to convert the resin to the Li^+ form, leaving ferrous hydroxide as a deoxygenating agent on the resin.

During operation, the water purity and pH were maintained by a bypass set of mixed bed ion-exchange resins on each rig taking about 1% of the total flow (Fig. 2). These consisted of a bed in the H^+/OH^- forms mounted parallel to a bed in the Li^+/OH^- forms. A check was kept on the constancy of the pH and conductivity by periodically sampling the water from the loops and the amount of lithium in solution was determined by flame photometry.

Regular analyses of samples of the loop water for chloride ion, silica, soluble and total iron (as Fe^{+++}), ammonia, copper (as Cu^{++}) and peroxide were also carried out. These impurities were measured by standard colorimetric methods, using a Unicam SP 500 spectrophotometer, and the detection limits were 0·03, 0·06, 0·05, 0·05, 0·02, 0·001 and 0·05 ppm, respectively.

Filtering of the water for insoluble corrosion products was through a 20 μm stainless steel filter, while a 0·45 μm millipore filter was used for obtaining samples for analytical purposes. Both these filters were located in the 1% bypass circuit.

The hydrogen and helium gas blankets above the water were frequently monitored by taking samples and analysing them on a gas chromatograph.

Hydrogen dissolved in the loop water was measured by stripping it from a known closed volume of loop water with an air stream and, in the case of the rig with the hydrogen blanket, measuring the thermal conductivity of the resulting water-saturated air/hydrogen mixture on a calibrated katharometer. For the rig with the helium blanket the sample of air/hydrogen was analysed by gas chromatography.

Dissolved oxygen in the water was measured by an electrochemical technique using an instrument developed at the Admiralty Materials Laboratory which is now produced commercially by the North Hants Engineering Co. Ltd.

The sample water was passed through a measuring cell to obtain a typical sample of loop water, excess water being run to waste. This water in the cell was then electrolyzed between a platinum electrode and a carbon electrode, and, by selecting a suitable voltage across the cell, conditions could be obtained where the current at low concentrations of dissolved oxygen was limited by the rate of diffusion up to the cathode. The limiting current for oxygen reduction was then proportional to the dissolved oxygen concentration. Oxygen levels down to 0·002 cm^3/kg could be detected.

LOOP COMMISSIONING

In-pile operation of the loop began in September 1964, but within a few days the loop water turned yellow. This was shown to be due to organic material removed from the nylon hoses connecting the in-pile and out-of-pile sections, and the hoses were replaced by stainless steel. (The detailed examination of the cause of organic contamination has been published elsewhere.[2])

The loop was cleaned by running with frequent changes of water for several weeks, so that the organic film suspected to be on the metal surfaces of the loop would be removed, either by simple leaching or by attack by radiolysis products in the water. This procedure was continued until the loop water was satisfactory for corrosion tests. Specimen assemblies were then replaced and the experiment was successfully restarted in June 1965.

ACKNOWLEDGEMENTS

The assistance of Mr J. J. Stobbs (now at Electro-Watt Engineering Services Ltd. of Zurich, Switzerland) in the design stage of the experiment, and of various people associated with the operation of the rigs in the BEPO reactor is gratefully acknowledged. The Paper is published by permission of the Central Electricity Generating Board and the United Kingdom Atomic Energy Authority.

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APPENDIX

Table 1: *Data sheet*

Per rig	
Total quantity of water in circuit	21 gal (95·5 litre)
Quantity of water in dynamic circuit	3 gal (13·6 litre)
Quantity of water in dead circuit	18 gal (81·8 litre)
Volume of each ion exchanger	3·1 litre
Total area of mild steel surface in circuit (14 ft^2 dynamic, 14 ft^2 dead)	28 ft^2 (2·6 m^2)
Total area of stainless steel	3 ft^2 (0·28 m^2)
Total area of nylon	13 ft^2 (1·21 m^2)
Water pressure at specimens	$\simeq$50 p.s.i.g. ($\simeq$446 kN/m^2)
Hydrogen blanket gas pressure	10 p.s.i.g. (170 kN/m^2)
Maximum operating pressure	80 p.s.i.g. (653 kN/m^2)
Main flow from pump	9 gal/min (41 litre/min)
Flow through water treatment circuit	0·45 gal/min (2·05 litre/min)
Velocity in primary water circuit	6 ft/s (1·83 m/s)
Velocity in primary water circuit (in-pile section)	4 ft/s (1·22 m/s)
Reynolds No. in in-pile section	20 000
Transit time through reactor section	15 s
pH of water	10·5
Water temperature	40–45°C
Radiation flux at in-pile section	
Fast neutron	2×10^{10} n/cm^2 s
Thermal neutron	10×10^{10} n/cm^2 s
Gamma dose rate	5×10^5 r/h

A one-energy group Monte Carlo collision probability code (MONTE) has been written to calculate the ratio of $^{238}U/^{235}U$ fission rates in rod-cluster geometry. A reasonably exact representation of the lattice cell is adopted with a simplified representation of neutronic events. The predictions of the code have been compared with experimental data for UO_2 rod-cluster fuelled heavy water reactor lattices with a number of different coolants. Using a single fixed set of cross-section data, the variation of δ_{28} with coolant type and lattice pitch is satisfactorily predicted, but the situation with regard to cluster geometry variations is less clear. For the only 'small' cluster (seven rods) for which experimental data were available, there is very close agreement between experiment and theory. For the larger clusters (19 or more rods) more typical of practical power reactors, the MONTE predictions were on average some 8% lower than experiment. It is not resolved whether this cluster geometry effect arises from shortcomings of the MONTE model or from a systematic bias in the experimental data.

Monte Carlo calculations (one-group) of $^{238}U/^{235}U$ fission ratios in rod-cluster fuel elements

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G. Doherty, BSc*
E. W. Hesse, BSc(Eng)*

INTRODUCTION

THE METHUSELAH scheme[1, 2] of calculation for heavy water moderated pressure tube reactor lattice cells employs three fast and two overlapping thermal neutron energy groups. The first group, having energies greater than 821 keV, includes all ^{238}U fission events in the cell, and so makes an important contribution to the reactivity of the system, particularly in the case of natural uranium fuel.

These group 1 events are calculated by means of diffusion theory and a 'smeared' model of the fuel cluster geometry, using a fixed set of cross-section data similar to, but updated from, that due to Ombrellaro.[3, 4] It is found using this scheme that agreement with measured $^{238}U/^{235}U$ fission ratios δ_{28} for heavy water moderated cluster fuelled lattices can only be obtained by major adjustments (up to about 50%) of the ^{238}U group 1 cross-section data, the adjustment required being a function of both cluster geometry and type of coolant. This deficiency of the METHUSELAH scheme is probably due mainly to neglect of flux hyperfine structure in the smeared model.

When new systems for which experimental information on δ_{28} is not available are surveyed, the choice of suitable group 1 data for ^{238}U is ill-defined and can lead to uncertainties of 1–2% in k_∞ for the unknown lattices. The deficiency in experimental data can largely be repaired by fully detailed Monte Carlo calculations in exact geometry and many energy groups. These, however, make a very heavy demand on computer time, which becomes entirely prohibitive if many systems require investigation.

It was therefore decided to adopt a simplified Monte Carlo approach to the problem, using only one energy group, but allowing a reasonably accurate geometrical representation of the lattice cell, and to test its validity by checks against experimental δ_{28} values for a wide range of cluster geometries, types of coolant and lattice pitch, for heavy water moderated systems. If successful, this would then permit adjusted cross sections to be derived for METHUSELAH calculations on new systems, with little expenditure of computer time. The comparison has so far been made only for experiments with UO_2 fuel elements, this being the fuel material of greatest immediate practical interest.

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MONTE CARLO CODE (MONTE)

A typical rod-cluster fuelled heavy water moderated pressure tube reactor lattice cell is shown schematically in Fig. 1. The MONTE code essentially calculates the rod–rod collision probabilities for the cluster in one energy group only, using the 'virtual collision' method of Woodcock *et al.*[5] These collision probabilities are then used to derive the cluster averaged fission ratio in terms of the ^{238}U fission cross-section and relative fission rates (primarily ^{235}U thermal) in each rod of the cluster, supplied as input data. The simplifying assumptions of the model are:

(a) that the transport-corrected total cross section gives the effective collision rate in the fuel rods, and that a fraction Σ_f/Σ_{tr} of these collisions results in fission events;

(b) that collisions in the coolant, cans or moderator which do not absorb the neutron or reduce its energy below the ^{238}U fission threshold may be ignored;

(c) that the effect of surrounding fuel elements (i.e. dependence of δ_{28} on lattice pitch) may be accounted for by reflection at the boundary of a single cell, which may be either square or hexagonal.

Further details of the model adopted in the code are given in Appendix I.

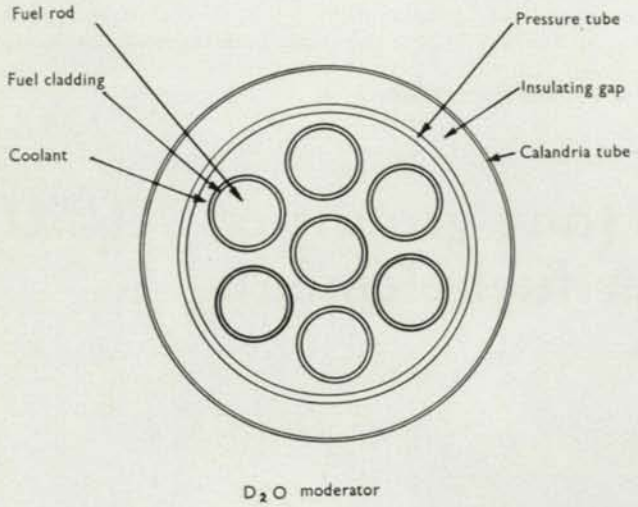

Fig 1 **Typical SGHWR lattice cell**

Table 1: *Microscopic cross-section data used in* MONTE *calculations (Methuselah II Library, Dec. 1964)*

Isotope \ Cross section* (barns)	σ_{trans}	σ_{rem}	σ_{abs}	σ_f
^{238}U	5·1635	1·9116	0·4282	0·3926
^{235}U	6·1340	1·6905	1·2484	1·1349
^{16}O	1·7070	0·3198	0·0220	—
^{1}H	1·5584	1·4492	—	—
^{2}D	1·6197	1·1143	0·0118	—
Al	2·4075	0·3871	0·0092	—
Zr	3·8040	0·7070	0·0633	—
Nb	1·8906	0·0823	—	—
Ni	2·7165	1·1154	—	—
Cr	2·8857	0·9494	—	—

* $\sigma_{g\to g} = \sigma_{trans} - \sigma_{rem} - \sigma_{abs}$

Table 2: *Experimental and calculated values of $10^4\ \delta_{28}$ for D_2O moderated UO_2 rod-cluster lattices*

Cluster	Coolant	Lattice pitch (cm)	Ref.	$10^4\ \delta_{28}$ experimental	$10^4\ \delta_{28}$ MONTE
7-rod (natural)	D_2O	18·1	AECL 2285	559 ± 20	584 ± 8
		19		579 ± 20	571 ± 8
		22		550 ± 19	547 ± 8
		28		519 ± 18	502 ± 8
		36		510 ± 18	516 ± 8
	Air	19·1		578 ± 20	594 ± 8
		22		560 ± 20	561 ± 8
	$C_{18}H_{22}$	19·1		580 ± 20	559 ± 8
		22		531 ± 19	532 ± 8
19-rod (natural)	D_2O	18·1	AECL 2285	564 ± 9	538 ± 6
		21		543 ± 9	504 ± 6
		24		528 ± 9	487 ± 6
		28		481 ± 8	448 ± 6
		36		475 ± 8	446 ± 6
	Air	18·1		659 ± 11	635 ± 7
		21		623 ± 10	550 ± 7
		24		565 ± 9	527 ± 6
		28		549 ± 9	499 ± 6
		36		532 ± 9	514 ± 6
	$C_{18}H_{22}$	18·1		581 ± 9	536 ± 6
		21		536 ± 9	489 ± 6
		24		521 ± 9	473 ± 6
		28		491 ± 8	452 ± 6
28-rod (natural)	D_2O	24·1	AECL 2636	580 ± 9	537 ± 5
		28		582 ± 8	506 ± 5
		32		554 ± 13	506 ± 5
		40		547 ± 8	485 ± 5
	Air	24·1		691 ± 11	626 ± 6
		28		632 ± 10	584 ± 5
		32		619 ± 9	569 ± 5
		40		624 ± 9	530 ± 5
	$C_{18}H_{22}$	24·1		567 ± 9	542 ± 5
		28		546 ± 8	509 ± 5
		32		543 ± 8	501 ± 5
37-rod (0·91% ^{235}U)	H_2O (20°C)	24·13·1	AEEW-R336	521 ± 11	482 ± 5
	H_2O (90°C)	24·13·1		500 ± 21	482 ± 5
'SG3' Series	H_2O/D_2O 'Beads'	24·13·1		547 ± 11	502 ± 5
		24·13·1		600 ± 13	531 ± 5
	Air	24·13·1		676 ± 14	631 ± 6
90-rod (1·14% ^{235}U)	Air	21·27□	STI/DOC/10/20	647 ± *	547 ± 5
	D_2O	21·27□		461 ±	391 ± 4
	80% D_2O/20% H_2O	21·27□		544 ±	387 ± 4
	60% D_2O/40% H_2O	21·27□		504 ±	381 ± 4
	H_2O	21·27□		458 ±	365 ± 4

* Experimental errors not quoted

DATA FOR MONTE CALCULATIONS

The microscopic cross-section data used in all calculations are those of the December 1964 METHUSELAH II data library and for convenience are listed in Table 1.

Relative fast neutron source strengths for each rod of the cluster are obtained from the 'power distribution' given by a METHUSELAH calculation for the appropriate lattice.

EXPERIMENTAL DATA

Few measurements of fast fission ratio, δ_{28}, in UO_2 rod-cluster lattices typical of pressure tube reactors, covering a range of different coolants, have been published, the principal series being those in support of the Canadian CANDU development programme. Bigham[6] reported measurements in the Canadian ZED-2 reactor on clusters of $7 \times 2{\cdot}40$ cm diameter and $19 \times 1{\cdot}421$ cm diameter natural UO_2 rods

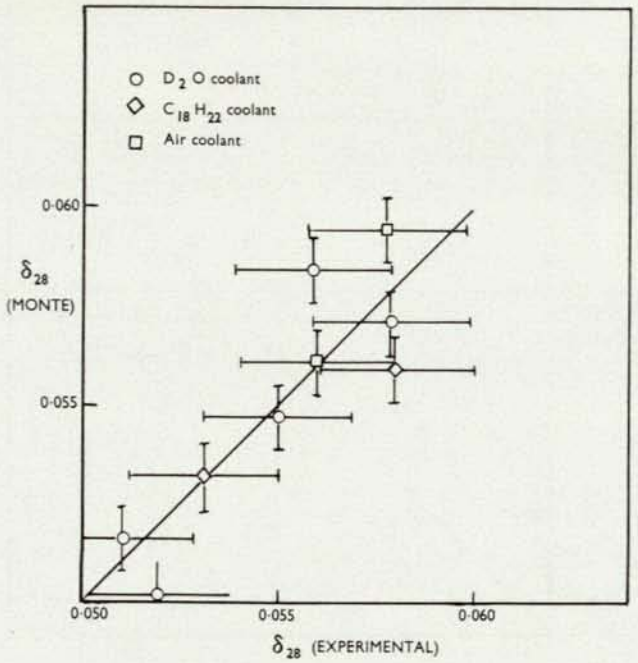

Fig 2 **δ_{28} for 7-rod clusters**

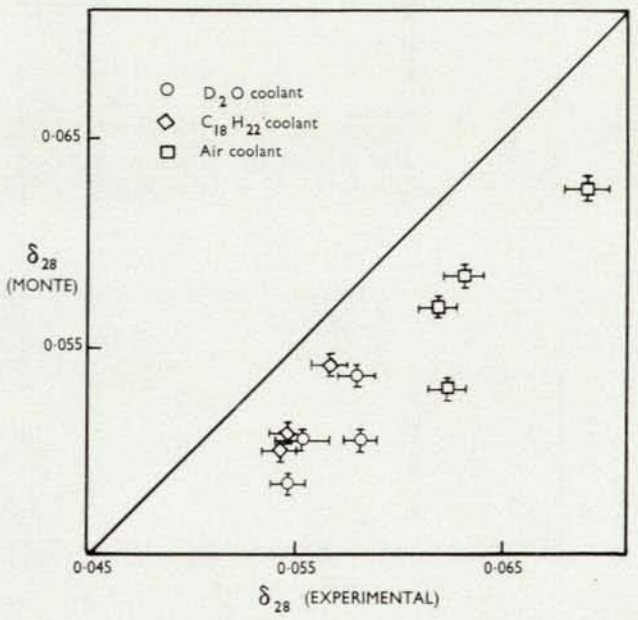

Fig 4 **δ_{28} for 28-rod clusters**

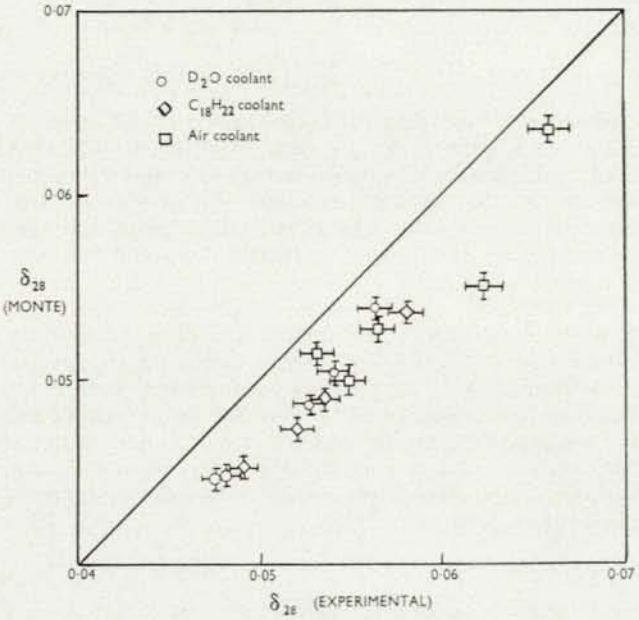

Fig 3 **δ_{28} for 19-rod clusters**

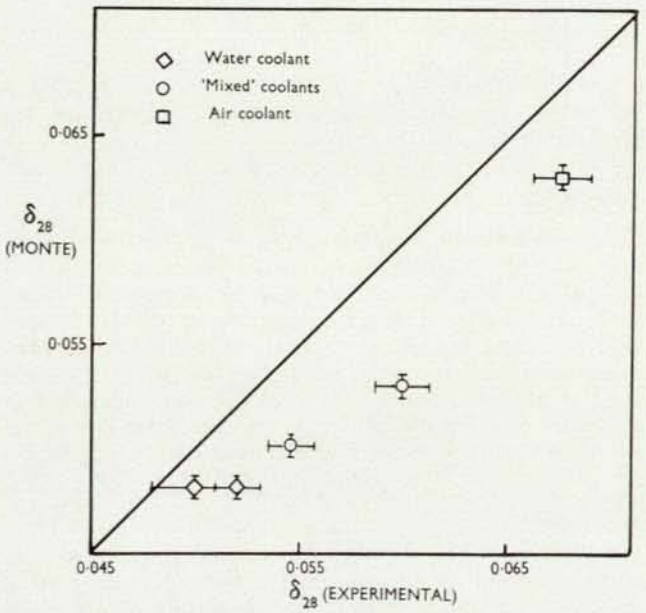

Fig 5 **δ_{28} for 37-rod clusters**

with air, heavy water and organic liquid 'coolants' in a variety of lattice pitches. It should be noted that the results in this reference revise and supersede those published earlier[7–10] for these clusters. Bigham[6] also reports a similar range of measurements on clusters of $28 \times 1{\cdot}421$ cm diameter natural UO_2 rods, but his quoted results have recently been re-analysed, with small revisions, in a publication by De Lange *et al.*[11]

UK measurements with air, heavy and light water and simulated reduced density water/steam 'coolants' have been reported for clusters of $90 \times 0{\cdot}762$ cm diameter rods[12] and $37 \times 1{\cdot}016$ cm diameter rods.[13] The experimental values quoted for the latter clusters differ slightly from those given by Campbell *et al.* as a result of revision of the time-dependent foil calibration factors.[14]

Relevant parameters of all the lattices for which measured

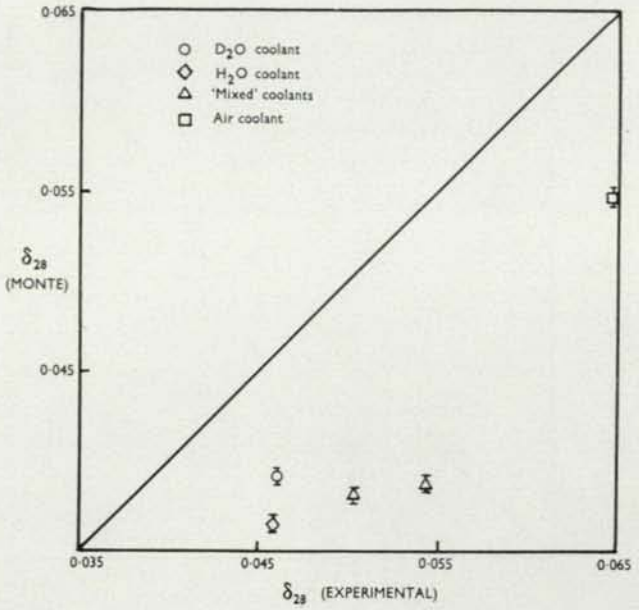

Fig 6 **δ_{28} for 90-rod clusters**

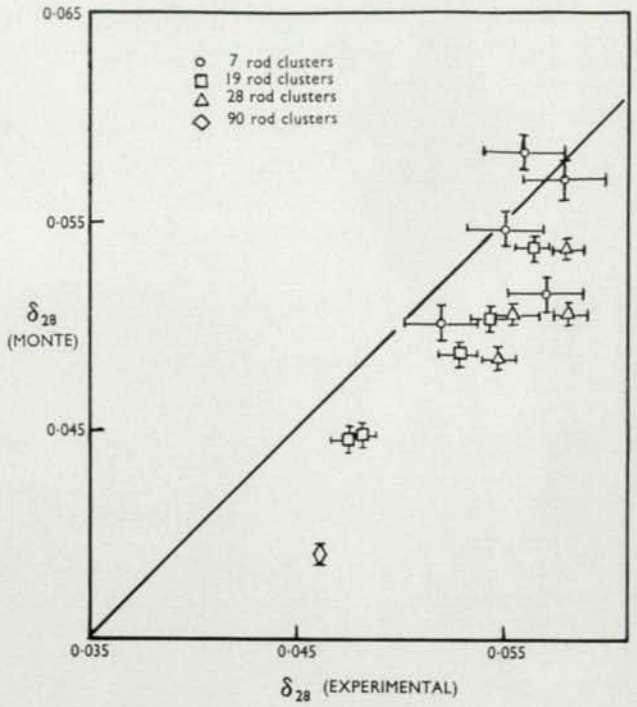

Fig 7 **δ_{28} values for D_2O-cooled clusters**

δ_{28} values were compared with MONTE calculations are given in Table 2.

RESULTS

The experimentally measured value of δ_{28} for each lattice and the corresponding value calculated using the MONTE code are given in Table 2. In Figs 2–6 the calculated values of δ_{28} are plotted against the experimental values for each type of cluster in such a manner that agreement between experiment and theory is denoted by all points lying on the line at 45° to either axis. Figs 7–9 are similar, but in this case the points plotted have been regrouped and correspond to air, heavy or light water/organic liquid 'coolant'.

DISCUSSION

It is clear from Figs 2–9 that two sets of experimental data stand out from the remainder. The first set is the DIMPLE 90-rod cluster measurements, for which there appears to be no systematic relation between experiment and calculation. The second set is the ZED-2 7-rod cluster measurements, which are in excellent agreement with the MONTE calculations.

The remaining data (19-, 28- and 36-rod clusters), whilst showing no significant systematic trends with either coolant type or lattice pitch, have calculated δ_{28} values tending overall to be some 8% lower than experimental values. These data are displayed in Fig. 10, which also shows the measure of agreement attained between experimental and calculated values of δ_{28} when the ^{238}U fission cross section used in MONTE is increased by 8%.

Newmarch[14] has stated that the DIMPLE 90-rod measurements were made at a very early stage of the UKAEA heavy water programme, when the experimental techniques were still in the development stage. He therefore regards the results as not fully reliable and recommends that they be disregarded in comparative studies. They will therefore be considered no further.

The Canadian 7-rod data were found by De Lange *et al.*[11] to be systematically higher than would be expected from comparison of the 19- and 28-rod data with the 'recipe calculation' code LATREP. These measurements were made with foils less depleted in ^{235}U (14-fold) than is usual, and this might possibly be the source of a systematic bias in the experimental results. It is, however, not possible to say categorically whether these or the remaining cluster data may contain an error.

CONCLUSIONS

(a) A simple one-group Monte Carlo code, MONTE, has been developed, and has been shown to give good predictions, using a single fixed cross-section set, of the variation with coolant type and lattice pitch of the $^{238}U/^{235}U$ fission ratio δ_{28} in UO_2 rod-cluster fuelled heavy water moderated reactor cells. A typical calculation requires 2 min on the IBM 360/50 computer for ±1% (standard deviation) statistical accuracy in δ_{28}.

(b) The variation of δ_{28} with cluster geometry may not be quite as well predicted, agreement with experiment being nearly exact for Canadian 7-rod clusters, but calculated values being some 8% lower than experimental values for other geometries. The source of this discrepancy is not

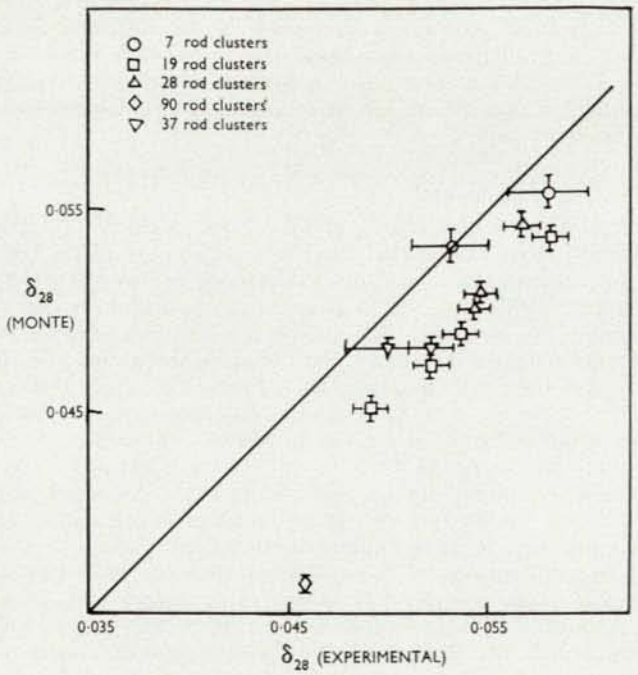

Fig 8 **δ_{28} for water- and organic-liquid-cooled clusters**

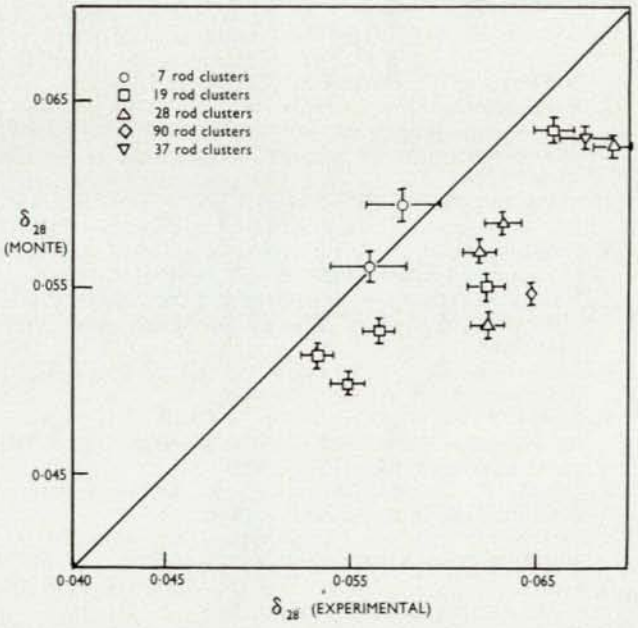

Fig 9 **δ_{28} for air-cooled clusters**

established, but may be due to systematic experimental bias in the 7-rod cluster results.

(c) If it is assumed that the 7-rod experimental data are in error, then very good agreement between MONTE and all the remaining UO_2 cluster data could be achieved by an increase of approximately 8% in the ^{238}U fission cross-section data used. This is not an unreasonable adjustment of the data, considering its method of derivation, which was by averaging over 54-group B1 slowing down spectra (MUFT-IV) for homogeneous uranium/light water mixtures. It is recommended that such an adjustment be made for MONTE calculations of the rather large clusters of rods proposed for current heavy water power reactor studies. The remaining scatter in δ_{28} values is then equivalent to only about 0·25% in lattice multiplication constant.

(d) In order to test with certainty the ability of MONTE to calculate variations of δ_{28} with cluster geometry, it is desirable that experiments be conducted on a range of clusters using a single technique. Such a series of measurements, covering clusters of from 7 to 61 rods has been planned and will commence shortly. Some improvement in accuracy over the usual level is anticipated from the use of a high resolution lithium-drifted germanium detector.

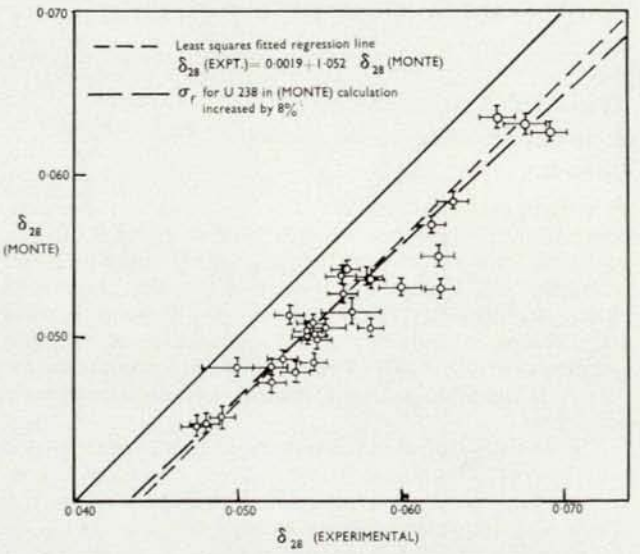

Fig 10 **δ_{28} for all clusters except Canadian 7-rod and DIMPLE 90-rod clusters**

ACKNOWLEDGEMENTS

The Authors acknowledge the contribution of J. E. Sinclair, (University of New South Wales), whose work with an early version of the MONTE code considerably reduced the effort required for the present comparative study. They are also grateful to Mrs J. I. Faulkner for her assistance with computations and data preparation.

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APPENDIX I

Some features of the MONTE model

Geometry

Fuel rods and cladding material are represented exactly, except that any gap between fuel and cladding is smeared uniformly into a thickened reduced-density cladding closely fitting the fuel. Collision probabilities to other rods of the cluster are explicitly calculated only for those rods which are specified as forming the minimum sector of rotational symmetry for the cluster. The complete collision probability matrix is then constructed via the rotational symmetry condition.

The coolant region is represented exactly, except that its outer radius is increased by the amount necessary to compensate for the removal cross section of the pressure tube which is not represented in the model.

The moderator region outer boundary is correctly specified, but an annulus is added to its inner boundary. This additional moderator annulus is chosen to compensate for the neutron removal cross section of the calandria tube, which is not specifically represented in the model.

These coolant/moderator modifications thus fix slightly modified dimensions for the pressure tube/calandria tube insulating gap.

Cross-section data

For each fuel rod, Σ_{tr}, Σ_f and $\Sigma_{g\to g}$ are specified. Σ_f is the fission cross section for only the ^{238}U content of the rod. The relative source strength in each rod is also given. The source within any rod is assumed to be uniformly distributed and isotropic. For each rod of the specified minimum sector of symmetry, the collision probability for all rods of the cluster is calculated. A fraction $\Sigma_{g\to g}/\Sigma_{tr}$ of these fuel collisions acts as secondary sources, and the total collision probabilities for these rods are obtained by iteration. The complete collision probability matrix for the cluster is then constructed from the symmetry condition, and the ^{238}U fission rate per source neutron is calculated via Σ_f and the relative rod source strengths specified.

For all regions of the cell other than the fuel rods, a single cross section only is specified, namely the cross section for absorption and removal below the ^{238}U fission threshold. This is equivalent to assuming that a neutron leaving a fuel rod continues in the same direction until it is absorbed, scattered to an energy where it can no longer cause fission in ^{238}U, or enters another fuel rod. On reaching the cell boundary, it is returned to the opposite side of the cell whilst retaining the same direction cosines of motion. In hexagonal geometry this is achieved by the use of three non-orthogonal co-ordinate axes perpendicular to the faces of the hexagon.

In the isolated cell (infinite pitch) option, the total remaining weight of the neutron is scored at the first moderator collision, and tracking is terminated.

It is not anticipated that errors due to these simplifying assumptions will be serious in practice. Modification of the programme to treat 'chequerboard' lattices of, for example, different enrichments, would be quite straightforward.

Fast fission ratio (δ_{28})

The fast fission ratio is calculated from the fissions per source neutron, N_8, obtained from the Monte Carlo calculation. Thus

$$\delta_{28} = \frac{\beta\nu_5 N_8}{1-\beta\nu_8 N_8}$$

where β is the fraction of fissions neutrons born above the fission threshold of ^{238}U and is assumed identical for both ^{235}U and ^{238}U fission, and ν_5 and ν_8 are the neutron yields per fission in ^{235}U and ^{238}U respectively. The standard deviations of N_8 and δ_{28} are also estimated.

A fast pulsed reactor, named VIPER, was commissioned at AWRE, Aldermaston, in 1967 and is now in use as a source of intense pulses of neutron and gamma radiation. The pulses are 400 μs wide with a peak power of 20 000 MW and can be produced at a frequency of one or two per day. The reactor has several novel design features including the use of the Doppler temperature coefficient to terminate the prompt critical pulse, and the commissioning experiments therefore have considerable interest in the field of fast reactor physics. In this Paper the reactor is described and in the following paper (Part 2) an account is given of the measurements of its physics characteristics.

The fast pulsed reactor VIPER

Part 1: General description

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H. Goodfellow*
M. H. McTaggart*
E. G. Warnke*

NOTATION

$F(t)$	total fissions to time t
F_T	total fissions in prompt pulse
$k(t)$	prompt reproduction factor at time t
k_0	prompt reproduction factor when pulse starts
$n(t)$	neutron population per unit volume at time t
t_m	time of occurrence of peak power
W	pulse width at half peak height
α	$(k-1)/\tau$
α_0	$(k_0-1)/\tau$
β_i	fractional yield of ith group of delayed neutrons
λ_i	decay constant of ith group of delayed neutrons
μ	temperature coefficient of reactivity
$\bar{\nu}$	average number of neutrons produced per fission
$\rho(t)$	$1-1/k$ = prompt reactivity at time t
ρ_0	$1-1/k_0$ = prompt reactivity when pulse starts
τ	average neutron lifetime

INTRODUCTION

THE term 'pulsed reactor' is generally taken to mean a reactor which is designed so that the reactivity can be quickly increased from a delayed critical or sub-delayed critical condition to a highly reactive condition, usually beyond prompt critical, and then quickly reduced again. Most pulsed reactors are operated by a single-shot process in which the reactivity is increased by rapid addition of fuel or rapid removal of absorber and is subsequently decreased by an inherent temperature-dependent property such as the expansion of the fuel or the decrease of the effective cross section for fission. However, a few pulsed reactors have been built to operate in a repetitive mode in which the addition and removal of reactivity are achieved by mechanically cycling a piece of fuel or absorber through the reactor core. Examples of pulsed thermal reactors are TRIGA,[1] TREAT[2] and SPERT.[3] The best-known example of a pulsed fast reactor is GODIVA[4]; other pulsed reactors of this type are reviewed by Wimett.[5] The repetitively pulsed fast reactor, IBR, is described by Blokhin *et al.*,[6] and SEFOR, a power fast reactor with pulsing capability, is described by Hafele *et al.*[7]

Pulsed reactors are used either as intense transient radiation sources or as means for studying the kinetic behaviour and the reactivity temperature coefficients of reactors undergoing fast transients at very high power levels. In the former case, the neutron or gamma radiation may be used for testing radiation effects on fissile or non-fissile components, for example power reactor fuel elements, or for measuring neutron scattering cross sections where high source intensity is an important consideration. Studies of temperature coefficients and kinetic behaviour are an essential part of the commissioning of any pulsed reactor and often provide important information of general application. It is usually necessary to limit these studies to those essential to the proving of a particular pulsed reactor design. However, in the case of SPERT and SEFOR, for example, such studies are the predominant feature of the experimental programmes, which are designed to provide information for the safety evaluation of power reactor systems.

The VIPER reactor, which began operating at AWRE Aldermaston in 1967, is a fast pulsed reactor operated on a single-shot basis at a frequency of not more than two pulses per day. It is designed for testing the behaviour of materials under intense transient radiation pulses. For example, the behaviour of enriched uranium metal during rapid expansion is being studied at various temperatures by measuring the displacement of a surface during the reactor pulse heating. The reactor also provides a useful

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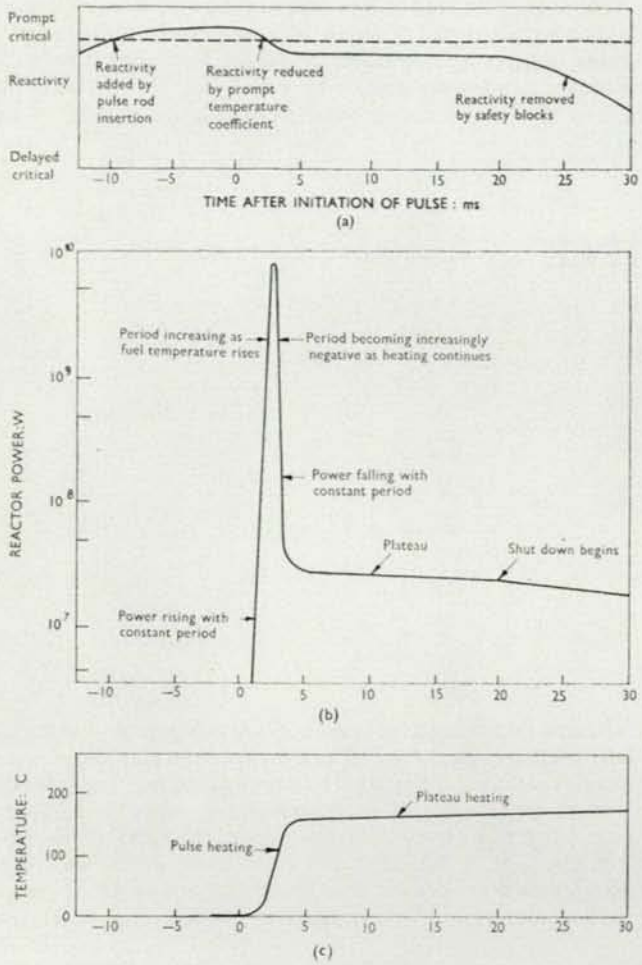

Fig 1 **Pulse characteristics**

simulant of the radiation burst from a criticality accident, and in this connexion it is being used to test the performance of accident monitoring instruments.

The VIPER design has a number of features which are novel to fast pulsed reactor design and in particular it uses the Doppler effect in ^{238}U as a principal component of the reactivity temperature coefficient. The commissioning tests therefore provide a direct demonstration of the efficacy of the Doppler effect in terminating a fast reactor transient and a means of checking the accuracy of kinetic calculations involving the Doppler effect.

This Paper is Part 1 of two papers describing the reactor and the experiments which have been done to measure its characteristics. The general principles of design of fast pulsed reactors are briefly reviewed and the application of these principles in the design of the VIPER reactor is described, with particular emphasis on the novel features. In Part 2 an account is given of the measurements which were done to study the physics characteristics of the reactor and the results are compared with calculated values.

PRINCIPLES OF PULSED REACTOR DESIGN

The characteristics of the pulse produced by a single-shot pulsed fast reactor are illustrated in Fig. 1. The sequence of events is as follows:

(a) The reactor is balanced at delayed critical.

(b) A reactivity insertion of rather more than one dollar is made by inserting an extra piece of fuel in the form of a pulse rod. The final part of the reactivity addition ramp is shown in Fig. 1 (*a*).

(c) A diverging fission chain reaction is initiated either by the existing neutron population or by the injection of a pulse of neutrons from an accelerator source. The power then rises exponentially as shown in Fig. 1 (*b*) and continues to rise with the same period until significant heating occurs.

(d) As the power increases the fuel temperature begins to rise (Fig. 1 (*c*)), and the prompt negative temperature coefficient immediately begins to reduce the reactivity of the system. The power continues to rise but with a steadily increasing period until it reaches a peak at the point where the prompt reactivity is reduced to zero.

(e) Heating continues at a high rate and as the prompt reactivity becomes negative the power begins to fall. The reactivity continues to decrease and the (negative) period continues to decrease (i.e. the pulse edge becomes steeper) until the power is reduced to a level at which the rate of reactivity change becomes negligibly small.

(f) The reactor reaches a state of transient equilibrium in which the negative period is constant and is approximately equal in magnitude to the period at the start of the pulse.

(g) The power falls with the transient equilibrium period until it reaches the level where the effects of delayed neutron emission become significant. The delayed neutron contribution can be regarded on this time scale as a source equal to the initial rate of decay of the precursors produced in the prompt pulse, with a prompt multiplication determined by the prompt reactivity condition of the reactor. Since the latter is now at a temporary equilibrium value the fission rate due to delayed neutrons is constant and the reactor power levels off at this value, which is the plateau shown in Fig. 1 (*b*).

(h) The reactivity is rapidly reduced by mechanical means and the plateau heating is terminated.

The kinetic behaviour of a single-shot pulsed reactor with a constant temperature coefficient is well represented by a space-and-energy-independent reactor model ignoring delayed neutron effects in the prompt critical part of the transient (e.g. Wimett[4]). The kinetic equation is thus:

$$\frac{dn}{dt} = \frac{n}{\tau}(k-1) \tag{1}$$

The prompt reproduction factor is related to the integrated power and to the temperature coefficient as follows:

$$k = k_0(1-\mu F) \tag{2}$$

Thus

$$\frac{dn}{dt} = \frac{n}{\tau}\left[k_0(1-\mu F)-1\right] \tag{3}$$

and

$$\frac{d^2F}{dt^2} = \frac{k}{\tau}\frac{dF}{dt}\left[1-\frac{1}{k_0(1-\mu F)}\right] \tag{4}$$

An approximate solution of equation (4) is

$$F = \frac{2\alpha_0\tau}{\mu k_0}\left[\frac{\exp\alpha(t-t_m)}{1+\exp\alpha(t-t_m)}\right] \quad (5)$$

The following relationships can then be derived:

$$\text{Total yield in pulse, } F_T = \frac{2\rho_0}{\mu} \quad (6)$$

$$\text{Peak power} = \frac{\alpha_0\rho_0}{2\mu} = \frac{\alpha_0 F_T}{4} \quad (7)$$

$$\text{Yield integrated up to peak} = \frac{\rho_0}{\mu} \quad (8)$$

$$\text{Reactivity at end of pulse} = -\rho_0 \quad (9)$$

Pulse width at half peak height,

$$W \simeq \frac{3{\cdot}52}{\alpha_0} \simeq \frac{7\tau}{k_0\mu F_T} \quad (10)$$

From equation (9), the value of α at the end of the pulse is equal in magnitude but opposite in sign to its value at the start. The prompt part of the pulse is thus completely symmetrical (within the limitations of the simple theoretical model).

The pulse duration is generally very much shorter than the mean lifetime (0·28 s) of the shortest-lived delayed neutron precursors. Therefore in times appreciably less than 0·28 s the rate of production of neutrons from precursors formed in the pulse can be written as a constant term

$$A = \Sigma_i\lambda_i\beta_i\bar{\nu}F_T \quad (11)$$

and the rate of formation of new precursors is small by comparison. The time variation of neutron population is therefore well represented by the equation

$$\frac{dn}{dt} = \frac{n}{\tau}(k-1)+\Sigma_i\lambda_i\beta_i\bar{\nu}F_T \quad (12)$$

and, since

$$\frac{k-1}{k} = -\frac{(k_0-1)}{k_0} = -\rho_0$$

the fission rate is constant and is given by

$$\frac{dF}{dt} = \frac{F_T}{\rho_0}\Sigma_i\lambda_i\beta_i = \frac{2}{\mu}\Sigma_i\lambda_i\beta_i \quad (13)$$

Thus the plateau power level is independent of the pulse yield.

At this stage it is useful to note certain features which are important in the design of pulsed fast reactors. These features follow from equations (6)–(10) by inserting typical values of the basic reactor properties. For example, if the mean lifetime, τ, is 10^{-7} s and the temperature coefficient, μ, is $10^{-5}/°C$, then for a fission yield of 3×10^{17} the required value of ρ_0 is about 20 cents. The initial prompt time constant, α_0, must therefore be of the order of 10^4/s or greater and, from equation (10), the pulse width at half height will be of the order of 400 μs or less. On this time scale the rate of heat transfer by conduction is negligibly small and the heat capacity of the fissile material itself determines the temperature rise. The maximum allowable temperature of the fissile material therefore determines the maximum fission yield. The quantities which determine the pulse width, W, can be seen from equation (10). The pulse can be shortened by increasing F_T or by reducing τ/μ, but the simple relationship of equation (10) is valid only if the shut-down coefficient acts instantaneously. For very rapid pulses there is a significant inertial delay before the fuel expands and there is a consequent change in pulse shape.[4] However this is not an important consideration in the design of the VIPER reactor.

Because of the very short duration of a prompt transient in a fast reactor the temperature coefficient must act promptly to terminate the transients, and this rules out any property dependent on slow conduction of heat. In all pulsed fast reactors before VIPER the prompt coefficient has been entirely due to fuel expansion. In VIPER the Doppler effect is used in addition to the fuel expansion, and in certain types of VIPER design the Doppler effect can be the dominant contribution.

Another important consequence of the short time scale is that significant heating will occur between the end of the pulse and the operation of the plateau-terminating mechanism if the latter is initiated by the pulse itself. However, this can be reduced by using a sequence timer to determine the interval between events. Provided that the start of the pulse itself is fixed precisely (by injecting a strong source of neutrons at the appropriate time), the sequence timer can be set to give the shut-down signal before initiating the pulse. The shut-down rods then begin to move immediately the prompt spike part of the transient is complete and the plateau heating is reduced to a very small proportion of the total.

A limitation on the efficacy of the sequence timer and on the control of plateau heating is imposed by the effect known as pre-initiation, that is, the initiation of a diverging chain reaction before the reactivity addition has been completed. If this happens, the pulse yield corresponds to the reactivity level existing when the fission chain initiates (since the pulse rod does not move appreciably during the duration of a pulse). The pulse yield is therefore less than intended, but the plateau length is greater than intended (because the timing of the shut-down mechanism has to be related to the time when the pulse is expected to start). It can be shown by statistical analysis that the probability of initiation depends principally on the rate of addition of reactivity, the source strength and the time for which the system has been above prompt critical. Following Hansen,[8] the probability of initiation occurring up to the time T, when the pulse rod reaches its fully-in position, is

$$P(T) = 1-\exp\{-pST^2\bar{\nu}[\overline{\nu/(\nu-1)}]^{1/2}\} \quad (14)$$

where S is the source strength and p is the rate of addition of reactivity. Assuming that p is constant during the pulse rod movement, pT is equal to ρ_0. Since ρ_0 is already determined by the pulse requirements and the minimum value of S is determined essentially by the amount and the enrichment of the fuel, equation (14) defines the maximum value of pulse rod insertion time, T, which will give an acceptably low pre-initiation probability.

DESIGN PRINCIPLES OF THE VIPER REACTOR

The VIPER reactor is designed to provide a radiation source of the order of 3×10^{17} fissions in a pulse a few hundred microseconds long and is required to accommodate experimental irradiation samples with sizes up to a maximum linear dimension of about 1 ft. Since it is desirable to be able to vary the size and shape of the reactor core and the

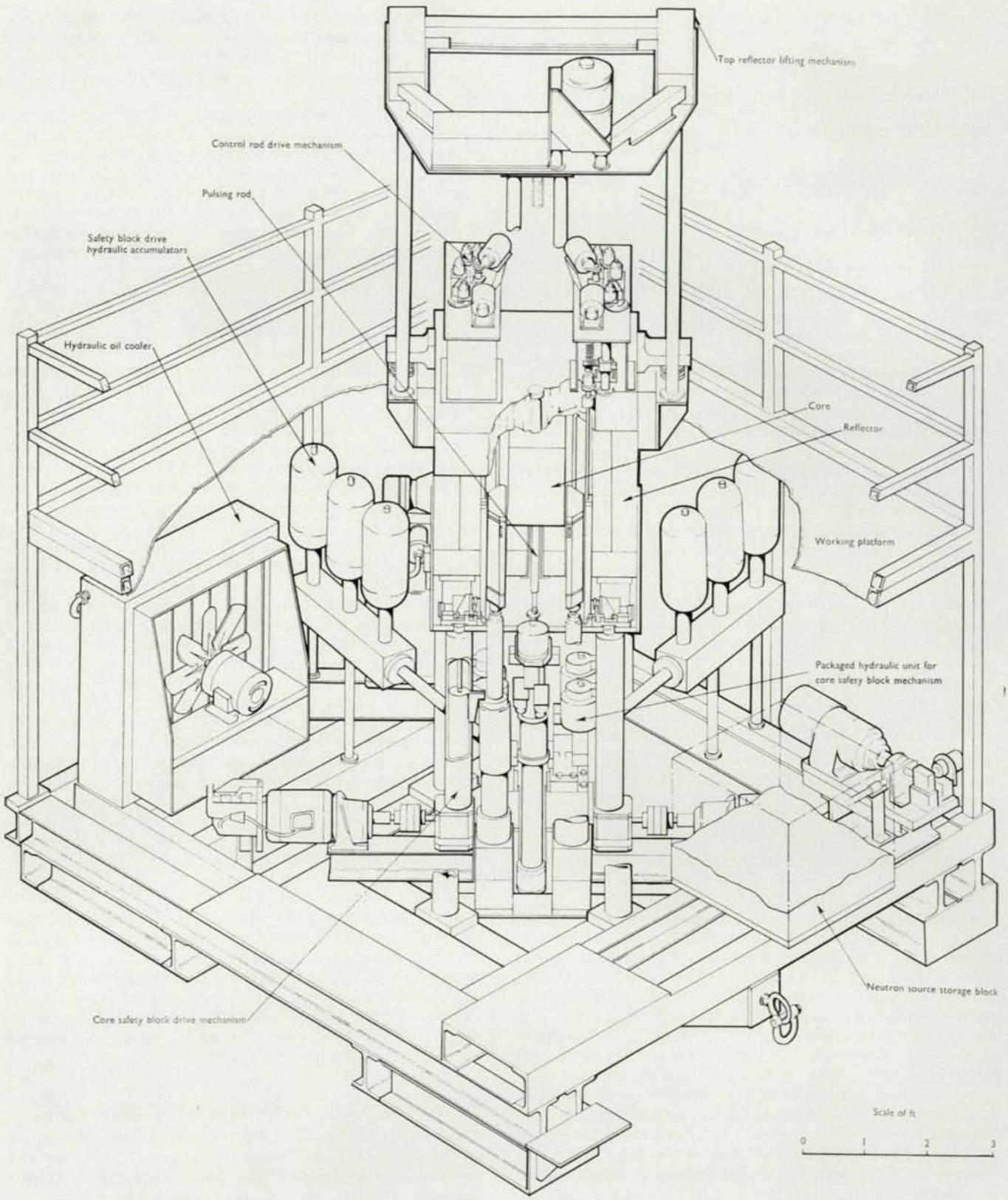

Fig 2 **General view of VIPER reactor**

THE BRITISH NUCLEAR ENERGY SOCIETY

1-7 GREAT GEORGE STREET - LONDON SW1 Telephone: 01-839 3611

INTERNATIONAL CONFERERENCE - 1969

THE PHYSICS OF FAST REACTOR OPERATION AND DESIGN

A three- or four-day Conference will be held in week commencing 23 June 1969 at The Institution of Civil Engineers, Great George Street, Westminster, London SW1.

The main theme of this physics conference will be the prediction and comparison with experience of the behaviour of fast reactors. Particular attention will be paid to the dynamic behaviour of existing reactors.

Sessions are planned on the following topics:

1. Basic physics measurements with zero power reactors
   Topics will include comparison of predictions with experimental results for zero power reactors, to test data and data sets. Adjustment of data and the use made of spectrum measurements will be discussed with the aim of reviewing the present discrepancies between integral and differential data.

2. The prediction of steady state and dynamic performance of power reactors
   Topics will include prediction of temperature and power coefficients, power distributions, control rod worths, gamma heating and similar characteristics. Both theoretical predictions and the use of zero power mock-ups will be included.

3. Measurements and experience in operation of power reactors
   Topics will include physics experiments and experimental techniques in power reactors, both for commissioning and power operation. In particular any discrepancies between prediction and observed performance should be emphasised and the implications of these on predictions for future reactors discussed.

Authors who wish to prepare papers for presentation at the Conference are invited to submit a 250-500 word synopsis not later than 1st October 1968 to:-

The Secretary, British Nuclear Energy Society,
1-7 Great George Street,
Westminster, London, SW1

Those who are interested in receiving further details of this Conference should complete and return the slip provided below.

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width of the pulse to suit a range of irradiation requirements, the design is based on a core made of rod fuel elements packed closely together in a matrix block. An alternative design, in which the core was divided into horizontal plates, was rejected because of the risk of buckling and the complications involved in cladding the plates and in locating them reproducibly. The rod design allows the core composition to be varied by varying the rod spacing and the composition of the matrix block. To ensure reproducible thermal expansion effects the rods are made as single pieces of metallic uranium alloy. This material will withstand repeated pulsing to 450°C without significant distortion and, taking this as a design maximum temperature, the minimum amount of fuel in the core to produce 3×10^{17} fissions is determined (by the heat capacity) as about 200 kg. Since in a fairly close-packed system the amount of ^{235}U required for criticality is much less than this, it is possible to use uranium of medium enrichment, which leads to the important property that, if a small amount of hydrogen is incorporated in the core, a large contribution to the prompt temperature coefficient can be obtained from the Doppler effect in ^{238}U. This is important because it overcomes the main disadvantage of using rod fuel—the fact that the expansion temperature coefficient is effective only in the direction of the rod axis. The use of medium-enrichment uranium also has the advantage that it makes possible the testing of fully enriched materials to high temperatures without risk of damage to the reactor fuel elements.

From equation (10) it is clear that a large temperature coefficient is important in obtaining a narrow pulse for any given yield and the inclusion of hydrogen may also make a significant change in the pulse width because of its effect on average neutron lifetime. There is therefore a fair degree of scope in varying the matrix block composition to obtain the required temperature coefficient and pulse width. Since the inclusion of very large irradiation samples in the reactor core would lead to undesirable operational complications and greatly increase the fuel requirement, provision was made only for the irradiation of small (2 in.) samples in the core. Larger samples may be irradiated in a cavity which can be made in the reflector adjacent to the core when required. The reflector is provided principally for this purpose but it also serves to de-couple the core from any changes in the surroundings which might otherwise introduce errors in the reactivity increment.

A series of trial calculations showed that a suitable enrichment would be 37%, giving a critical mass of about 80 kg ^{235}U. The core, designated VIPER 1, which was selected for the initial experimental programme has the following predicted characteristics:

Average neutron lifetime	$1{\cdot}5 \times 10^{-7}$ s
Expansion coefficient	-6×10^{-6}/°C
Average Doppler coefficient	-6×10^{-6}/°C
Critical mass	82 kg ^{235}U
Maximum fission yield (for a 400°C peak temperature rise)	3×10^{17}
Initial period of maximum pulse	100 μs
Width of maximum pulse at half height	400 μs
Maximum reactivity increment above prompt critical	22 cents
Peak power	20 000 MW
Plateau power	20 MW

Equation (6) shows that the pulse yield is directly proportional to the reactivity increment above prompt critical, and the reproducibility of the pulse yield is therefore determined by the reproducibility of the starting position and of the actual reactivity increment. To obtain 1% reproducibility of the maximum pulse yield in VIPER 1 the combined uncertainties of starting position and of reactivity increment must not exceed 1% of 20 cents, i.e. 0·2 cent, and this was used as the criterion for reproducibility in design and in operating procedures.

The reproducibility of all movable components is such that the reactivity change between two critical balance points separated by a reactor shut-down is less than 0·1 cent. However, from day to day there are deviations of up to 0·2 cent due to changes in the reactor temperature, and the reactivity balance point may change by up to 5 cents if the moisture content of the moderator is reduced by a series of full size pulses.

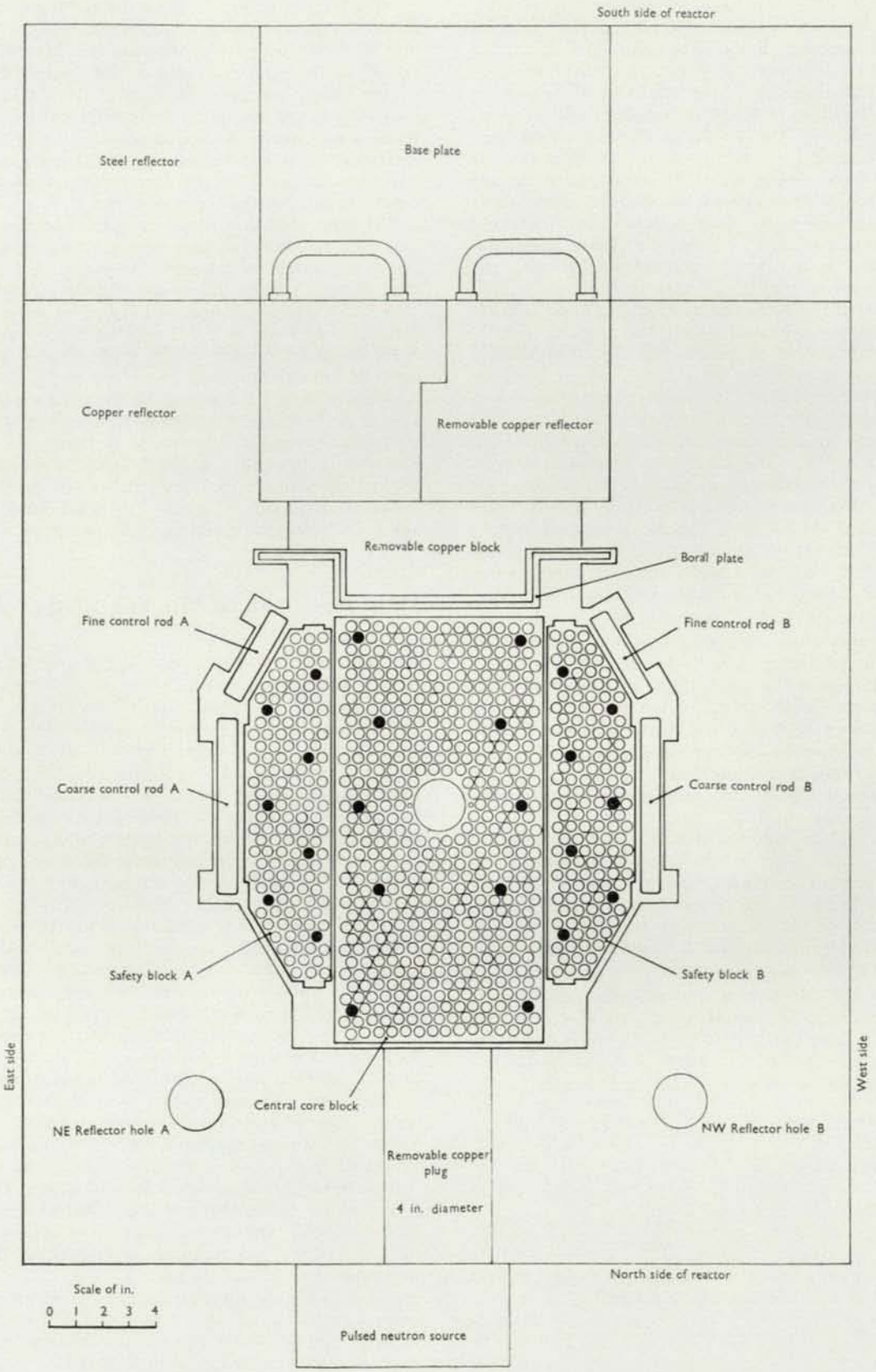
At the end of a full size pulse the reactor plateau power level is 20 MW and the plateau heating would be equal to the pulse heating if allowed to continue for 0·5 s. It is desirable to limit the plateau heating to less than 10% of the pulse heating, and the shut-down mechanisms are therefore required to remove a substantial amount of reactivity within 50 ms of the pulse initiation.

MAIN FEATURES OF THE REACTOR DESIGN

General features

An illustrative drawing of the reactor as a whole is shown in Fig. 2; plan and section views of the core and reflector are shown in Figs 3 and 4.

The core consists of a fixed central section, 8 in. × 15 in., with two smaller movable sections which are shaped to give an approximately cylindrical outline. The movable sections, called safety blocks, can be driven 12 in. vertically downwards to reduce the reactivity of the system during fuel loading or to shut down quickly after a pulse. Immediately outside the core boundary are four boron control rods with vertically acting drives, and the whole system is surrounded by a copper reflector which has a minimum thickness of 8 in. The reflector region below the core is extended downwards to act as a radiation shield around the safety blocks when they are lowered. The top section of reflector is a single slab and this can be raised to give access to the core. Part of the reflector on one side is in the form of removable blocks and the base plate is extended on that side, so that a large irradiation cavity about 13 in. square and 11 in. long can be provided when necessary by rearranging the reflector blocks. Since this cavity is directly adjacent to the reactor core, provision is made to insert a boron plate to de-couple the core from any slow neutrons produced in an irradiation experiment. The central core section has a 2 in. diameter hole at its centre. This provides a channel for the insertion of the pulse rod, which is driven up from below, and also provides at its upper end a cavity in which small samples can be irradiated. This central irradiation cavity is of diameter 1·75 in. and length 3·5 in., and it is 3 in. from the core centre at its nearest point. Two other experimental holes are provided in the corners of the reflector and provision is made for access to nine fuel element positions in the core central block.


South side of reactor
Steel reflector
Base plate
Copper reflector
Removable copper reflector
Removable copper block
Boral plate
Fine control rod A
Fine control rod B
Coarse control rod A
Coarse control rod B
Safety block A
Safety block B
East side
West side
NE Reflector hole A
Central core block
NW Reflector hole B
Removable copper plug
4 in. diameter
North side of reactor
Scale of in.
0 1 2 3 4
Pulsed neutron source

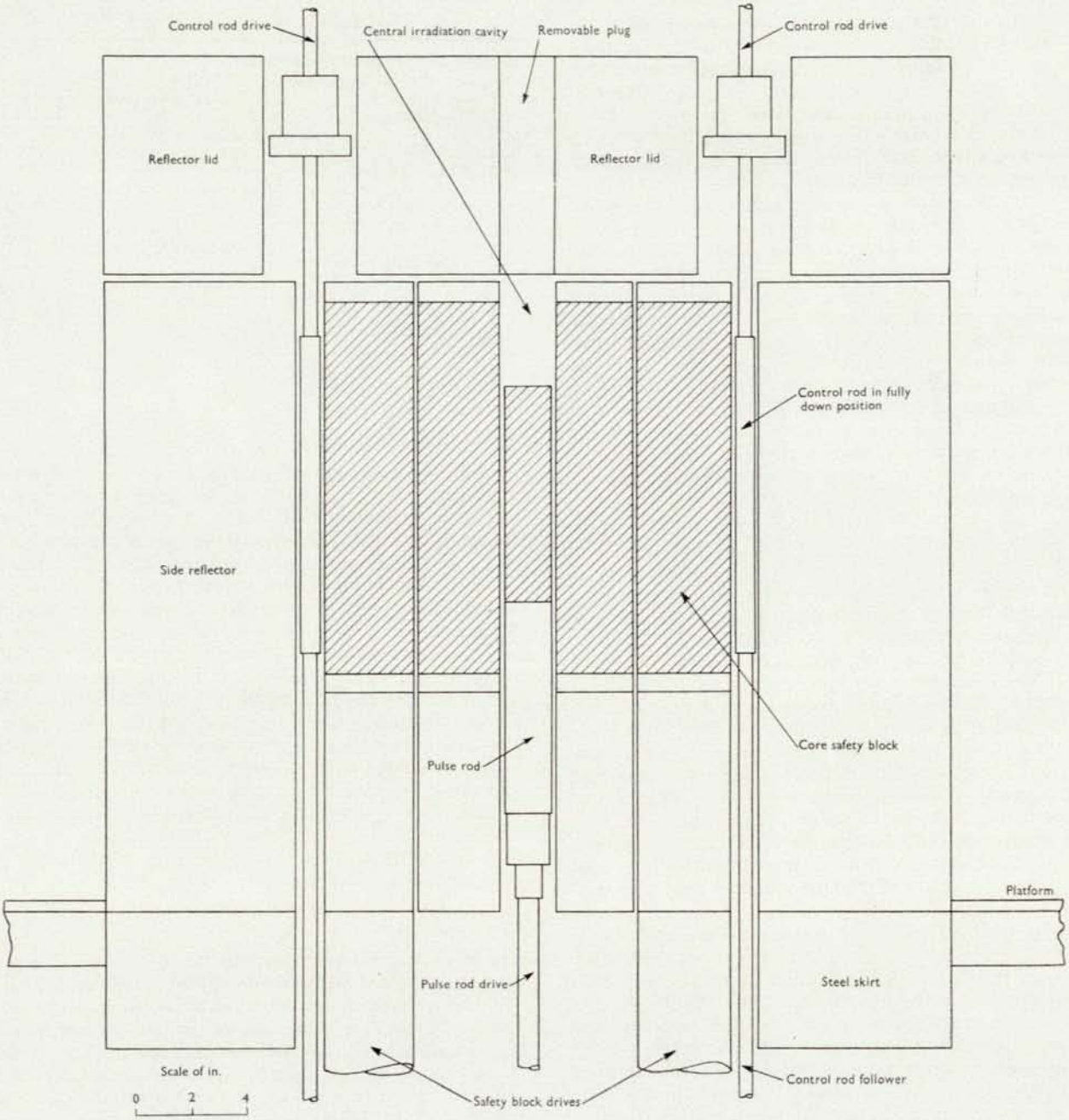


Fig 4 **Vertical section of the core and reflector**

Fig 3 (left) **Plan of the core and reflector**

The reactor is located in a large laboratory which has walls and roof made of thick concrete to provide the necessary radiation shielding. The same laboratory contains the low power experimental fast reactor VERA,[9] and the two reactors are operated as alternative facilities.

An adjoining laboratory contains an accelerator which provides a pulsed beam of deuterons and this can be directed to impinge on a titanium tritide target on either reactor system, thus producing a powerful pulsed neutron source. This is used on the VIPER reactor for neutron lifetime measurements or for time-of-flight spectrometry in commissioning experiments. A 4 in. diameter removable plug is provided in the reflector on the north side of the reactor to accommodate the pulsed source when required, and for the time-of-flight spectrometry a neutron beam can be extracted through the removable copper blocks on the south side.

For initiating prompt critical pulses the accelerator pulsed source is unnecessarily cumbersome and a smaller pulsed source of the type described by Gow and Pollock[10] is used instead. This is mounted outside the reflector on the north side as shown in Fig. 3.

Core design

The fuel elements (Fig. 5) are cylindrical rods of uranium, 11·4 in. long and of diameter 0·400 in., clad in stainless steel. The uranium is enriched to 37·5% in ^{235}U and is alloyed with 1·25% of molybdenum by weight. The function of the steel cladding is to prevent corrosion if the core is exposed to air when hot and to provide a means of holding each fuel rod at its mid-point so that the forces developed during the pulsed heating are balanced and do not load the reactor structure. The cladding is secured to the fuel rod by forming it into a circumferential notch which is cut into the fuel rod around its centre. If the rods were supported at their ends they would exert a total force of 50 tons on the reactor base plate during a pulse, and the maximum stress in the fuel would be four times greater than with support at the mid-point.

The construction of the core is illustrated in Figs 6 and 7. The elements rest on a steel base plate and are held down by springs which are strong enough to ensure that each element remains seated on the base plate throughout the pulse and the subsequent expansion and contraction of the fuel can. The elements are located in a triangular array by lattice plates of low-expansion steel at the top and bottom of the core and at the mid-plane. The holes in the mid-plane lattice plate are displaced radially outwards by 0·010 in. with respect to corresponding holes in the top and bottom plates, thus preventing simple inward bowing of the fuel elements due to the radial transverse temperature gradient. This arrangement does not prevent bowing of the rods between lattice plates but, as shown by calculations to be described in Part 2, this is a very small effect. The space between the fuel elements and the lattice plates is filled with a 'matrix material' which can be selected according to the required core composition. The matrix materials at present available are copper, which provides a non-moderating material to reduce loss of neutrons by streaming, and epoxy resin, which provides neutron moderation by virtue of its hydrogen content. The epoxy resin formulation selected was developed for its stability at temperatures up to 275°C. The matrix materials are in the form of plates of various shapes as shown in Fig. 6 and their thicknesses are either 0·375 in. or 0·187 in. The plate sizes are limited so that there is adequate clearance for their expansion when the core is heated. For VIPER 1 the matrix contains copper and epoxy resin plates in the volume ratio 1·5:1 and they are arranged in continuous planes across the reactor core as shown in Fig. 7. The critical fuel loading in VIPER 1 is 540 elements. The remaining fuel element positions are filled with copper rods and in these regions all the matrix plates are of copper. The element positions shown black in Fig. 6 are tie-rods holding the core box lids to their base plates and thus holding the fuel elements down. Each of the three core sections is contained in a steel box. The base is perforated and there is a gap around the lid so that argon gas can be flushed through the core before the reactor is pulsed. This is done to minimize chemical attack on the epoxy resin or on any exposed uranium metal.

Some numerical details of the core construction are summarized in Table 1.

Table 1: *Reactor core details*

Weight of ^{235}U per fuel element	155 g
Number of element positions:	
Central section	478
Safety blocks	133 each
Total	744
Lattice pitch	0·510 in.
Maximum volume of core	36 litres
Maximum volume fraction of uranium	0·45
Minimum void fraction	0·126
Average volume composition of VIPER 1:	
Fuel	0·45
Stainless steel	0·14
Copper	0·14
Epoxy resin	0·095
Void	0·175
Ratio of hydrogen to ^{235}U atoms in VIPER 1	0·46

Core materials

The fuel material requirements are as follows: adequate strength to resist deformation under dynamic loading at high temperature; minimum dimensional changes as a result of thermal cycling; stability in the expected irradiation environment. The three possible materials considered in the initial survey were 'adjusted' uranium containing 250–500 ppm iron and 500–1200 ppm aluminium, uranium alloyed with 1·25% molybdenum by weight, and uranium alloyed with 10% molybdenum by weight. The adjusted uranium was rejected because of its inferior resistance to thermal cycling effects and the 10% molybdenum alloy was considered undesirable because of the neutronic disadvantage of dilution with molybdenum and because of its tendency to brittleness under certain conditions. The 1·25% alloy therefore became first choice and its suitability was confirmed by practical tests of its strength and of its resistance to thermal cycling effects.

When the fuel rods are heated rapidly by a reactor pulse the material is unable to expand rapidly because of its inertia and a compressive stress is therefore established. This is then followed by a tensile stress of the same magnitude when the material recovers and overshoots its unstrained

state. The maximum local stress occurs at the centre of the rod where the calculated nominal stress for a full size pulse is 2 ton/in.^{2} with a peak of 8 ton/in.^{2} at the notch. The centre fuel temperature is about 100°C at the compression peak and about 320°C at the tension peak. The fuel strength was measured by plain bar tests of the fracture stress and the elongation at temperatures up to 600°C. Further tests of a similar type were made on bars notched in the same way as the actual fuel rods and these bars were also tested at the strain rate corresponding to a full-size reactor pulse, (2 in./in./s). These experiments showed that the 1·25% molybdenum alloy has adequate margins of strength and ductility for regular use in 400°C pulses and that damage by local yielding at the notch would not be expected in pulses up to 500°C.

The growth of the uranium alloy due to thermal cycling was measured by electrical resistance heating in vacuo and in an argon atmosphere. The samples were cycled between 60°C and 500°C every 2 min, the heating taking place in 2·5 s, and the growth was measured after 500 and 1000 cycles. The results showed that the thermal cycling growth resistance of the 1·25% molybdenum alloy is satisfactory; the material is unlikely to extend by more than 0·1% as a result of 1000 cycles to 500°C.

The stability of the fuel after prolonged radiation has not been tested. In 1000 full size pulses the accumulated neutron dose is about 10^{18} n/cm^2 and it is considered desirable to check the mechanical properties of the fuel before this stage is reached. Other fuel effects considered were burn-up, which amounts to 0·0001% after 1000 pulses and is therefore not important, and chemical reaction between uranium and stainless steel, which is also regarded as unimportant since it proceeds very slowly at temperatures below 520°C.

The requirements of the moderator are: it should contain an adequate proportion of hydrogen; it should be capable of fabrication into the necessary shapes for the lattice; and it should withstand heating to temperatures corresponding to a full size pulse without damage, significant loss of hydrogen or distortion. The material chosen for this purpose was an aluminium-filled epoxy resin which has the composition listed in Table 2. Small blocks of this material were tested by heating in an inert atmosphere for 10 h at various temperatures. Above 200°C there was a small weight loss which increased slowly with temperatures up to 300°C and fairly rapidly at higher temperatures. However, allowing for the total time spent at maximum temperature in each pulse, the total weight loss in 1000 pulses to 300°C would be limited to about 1% which is an acceptable figure. The maximum moderator temperature in a full size pulse is 275°C.

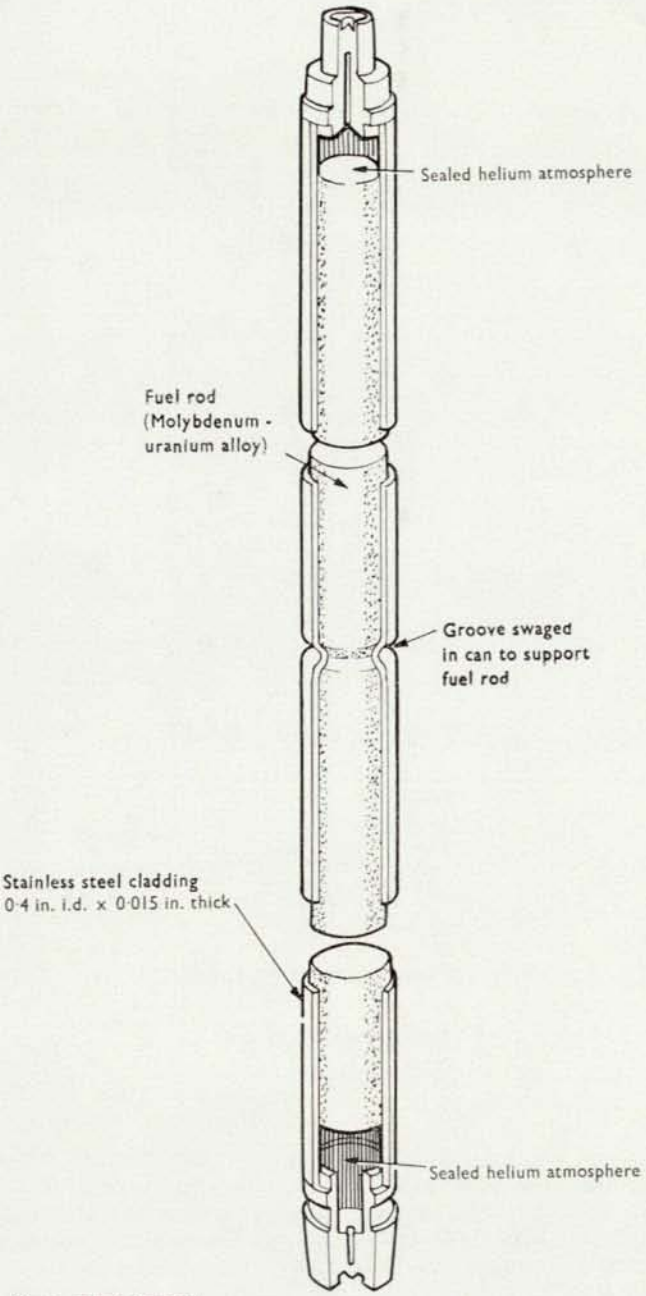


Fig 5 **Fuel element**

Control mechanisms

The reactor is controlled by movement of the safety blocks, the control rods and the pulse rod. The safety blocks, each worth 5% in reactivity, can be driven up or down slowly by a nut and lead-screw system or can be driven rapidly down to terminate the plateau heating after a pulse. The rapid drive is a hydraulic system which imparts an acceleration of 5*g*. Each block is driven by a double-acting piston with the pressures on the two sides adjusted so that the net upward force is sufficient to raise the block. When rapid

Table 2: *Composition of epoxy resin moderator*

Composition	% by weight
Aluminium	28·2
Carbon	50·8
Hydrogen	4·3
Oxygen	16·7
Hydrogen content, g/cm^3	0·062
Density, g/cm^3	1·44

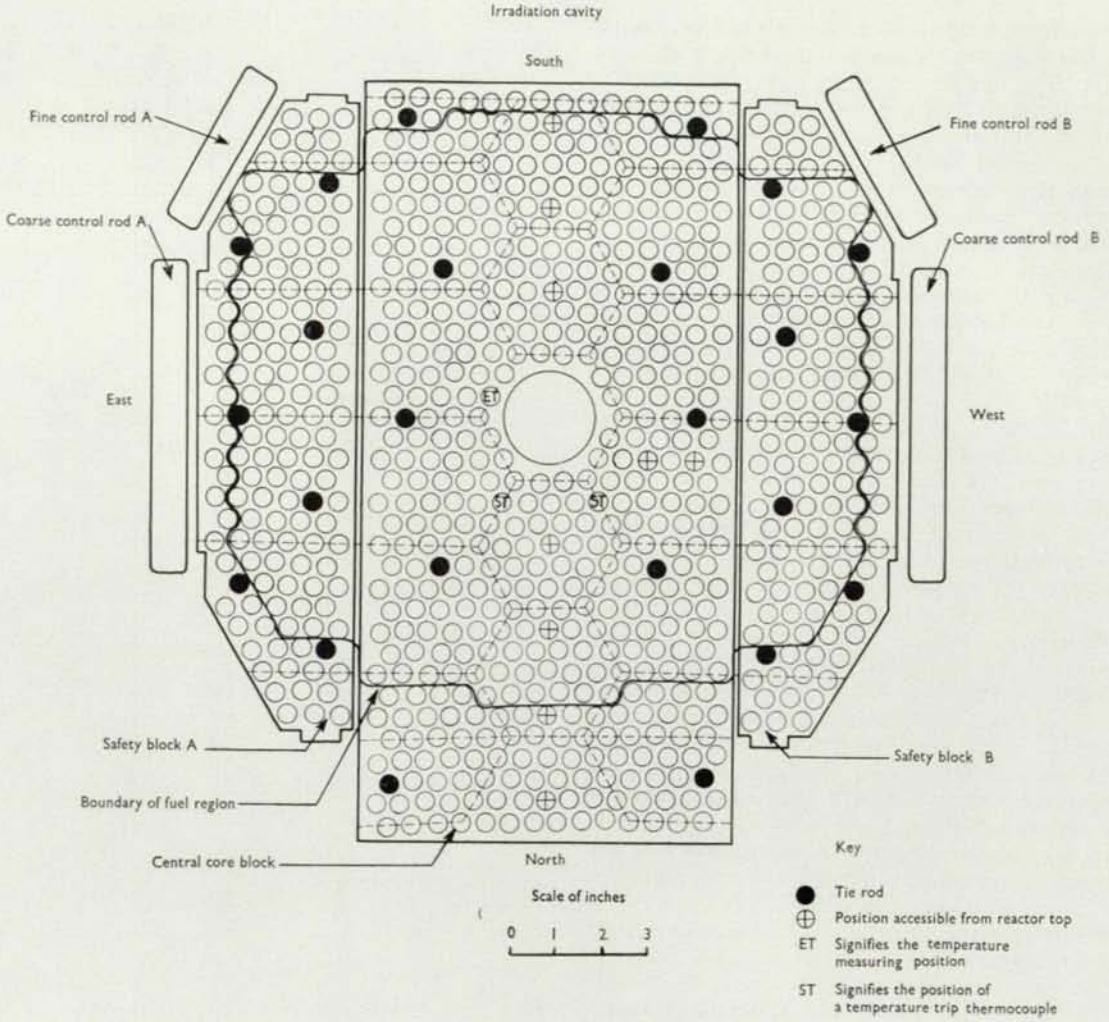


Fig 6 **Detailed plan of the core**

shut-down is required the trip signal opens a pair of vent valves in the lower cylinder, and the pressure in the upper cylinder then drives the block down using stored energy from a hydraulic accumulator. The blocks begin to move 20 ms after the trip signal has interrupted the current in the vent valves, and the time required to remove the first dollar of reactivity is 47 ms from the trip signal. To ensure high reliability in the shut-down process each block is provided with a separate set of three hydraulic accumulators and the vent valves are duplicated.

The control rods are boxes packed with boron powder enriched in ^{10}B. Each coarse rod contains 440 g of ^{10}B and is worth 1·2 dollars reactivity and each fine rod contains 70 g of ^{10}B and is worth 0·2 dollars. The rods are attached to their drives by electromagnets and they are released to drop under gravity into the core if the reactor safety lines are broken.

The pulse rod consists of a steel tube containing a cylinder of uranium with the same enrichment and alloy composition as the core fuel. The uranium cylinder is of diameter 1·55 in. and length 4·125 in. and weighs 2·35 kg. The reactivity worth (1·27 dollars) is chosen so that a full size pulse can be produced by driving the rod from its fully out to its fully in position, starting with the reactor at delayed critical. A small margin of extra reactivity is incorporated to allow for variations in pulse rod worth due to core rearrangements. For the rapid insertion required in pulsed operation the rod is driven pneumatically and it covers its full 14 in. stroke in 220 ms. This is sufficiently rapid to reduce the probability of pre-initiation to an acceptable level, provided that there is no neutron emission from experimental components in the irradiation cavities. For reactivity balance checks, the rod is provided with a slow speed drive using a nut and lead-screw.

Control and safety instrumentation

For start-up and steady-state operation the reactor is provided with conventional instrumentation which consists of

three low-power pulse channels and three medium-power current channels. The detectors are fission chambers for the pulse channels and boron-coated ion chambers for the current channels. All the detectors are in polythene cylinders at the outside of the reflector. Each system provides low, high and period trip circuits and all the outputs can be recorded. Together the steady-state power channels cover the power range from 0·2 mW to 2 kW.

For transient and pulsed operation it is necessary to provide several specialized instruments to ensure reactor safety. These are the instruments which detect the start of the pulse and immediately initiate shut-down action to prevent the reactor being overheated by plateau heating. They comprise a scintillation counter channel, a thermocouple channel and a time-sequence channel. To obtain the required high reliability in initiating the shut-down signal each channel is duplicated and provision is made for frequent functional checking. By using three different types of instrument we obtain protection against failure arising from a common fault. The design specification is based on limiting the plateau heating to 10% of the pulse heating and this requires the safety blocks to start moving within 70 ms of the pulse rise. Since 20 ms elapse from the breaking of the valve circuit to the start of block movement, the interval between the pulse rise and the actuation of the shut-down instrument relay must not exceed 50 ms. All three channels achieve this.

The two scintillation counters are mounted outside the reflector on the north side of the reactor. Each counter has a 2 in. long, 1 in. diameter crystal of sodium iodide mounted on a 2 in. photodiode.[11] The sensitivity is determined by the operational requirement to trip the reactor at a power which can be pre-set between 300 W and 3 MW. A photodiode rather than a photomultiplier is used because it is more reliable and because its current output does not vary appreciably over a wide range of polarizing voltage levels. The counters have shut-down amplifiers with variable trip levels. For prompt critical pulses the trip is set at 10 kW to provide a signal as soon as the pulse starts to rise. For sub-prompt-critical transients the trip has to be set at levels up to 3 MW to allow the fuel temperature to rise sufficiently that the temperature coefficient can be measured from the change in reactor period. The facility to set the trip level at 300 W is required for reliability checks in which the reactor is operated on relatively long-period transients to test the correct functioning of the trip channel and to calibrate its power signal against the steady-state instruments. This is done at intervals of two weeks. A facility is also provided for daily checks; this uses a pulsed light source which is built into the detector. For prompt critical pulses the operating time of the unit from the pulse initiation to the opening of the output relay contact is 20 ms.

The temperature safety channel[12] is based on an intrinsic thermocouple. This design is preferable to the normal sheathed thermocouple because of its faster response and because failure can be reliably detected by an electrical continuity check. The construction is illustrated in Fig. 8. A cylinder of uranium metal, of diameter 0·375 in. and length 0·25 in., is held in a stainless steel mounting which fits into the top end of a modified fuel element. The uranium is enriched to 37·5% in ^{235}U and is alloyed with 10% by weight of molybdenum to improve oxidation resistance at

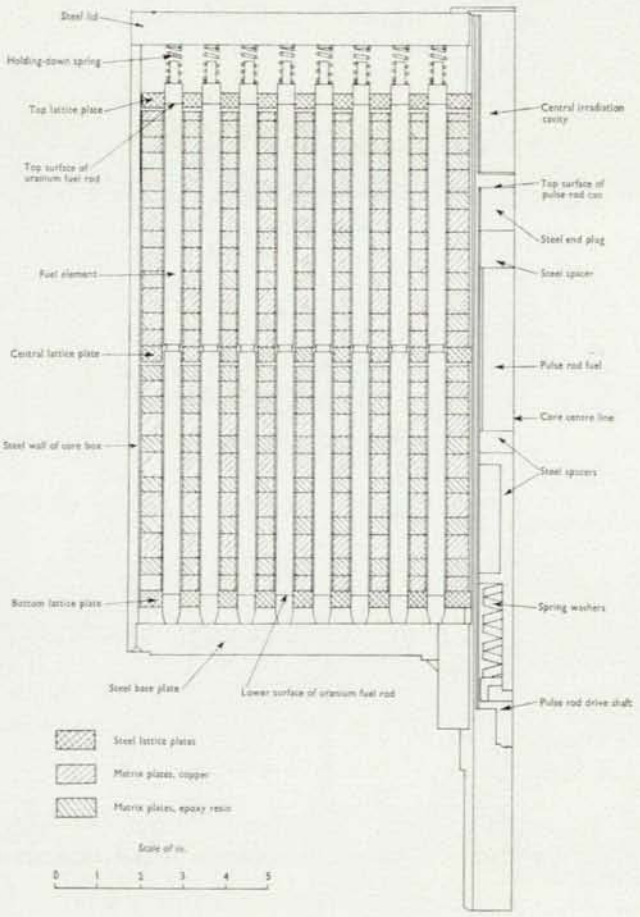


Fig 7 **Vertical section through the reactor core**

high temperature. A steel-sheathed 0·060 in. diameter chromel–alumel thermocouple cable passes through an axial hole in the uranium and the chromel and alumel wires (diameter 0·010 in.) are welded to the rim of the uranium. The choice of wire diameter is a compromise between the conflicting requirements of fast response and ruggedness. The 0·010 in. diameter wires have 2 ms response, which is small compared with other delays in the system, and at the same time they are sufficiently robust that the risk of accidental damage is very small. The temperature rise in the thermocouple uranium cylinder is approximately the same as that at the end of a standard fuel rod in the same position. The two thermocouple elements are placed in the positions marked ST in Fig. 6 and the temperature rise in these positions is 215°C for a full size pulse. The response time of the thermocouple trip amplifier from application of a step voltage at the input to opening of the output relay contacts depends on the ratio of the applied step voltage and the trip level. For typical conditions of a 215°C temperature rise and a 50°C trip level the response time is 35 ms, and for a prompt critical reactor pulse the time from pulse initiation to the opening of the output relay contact is 50 ms.

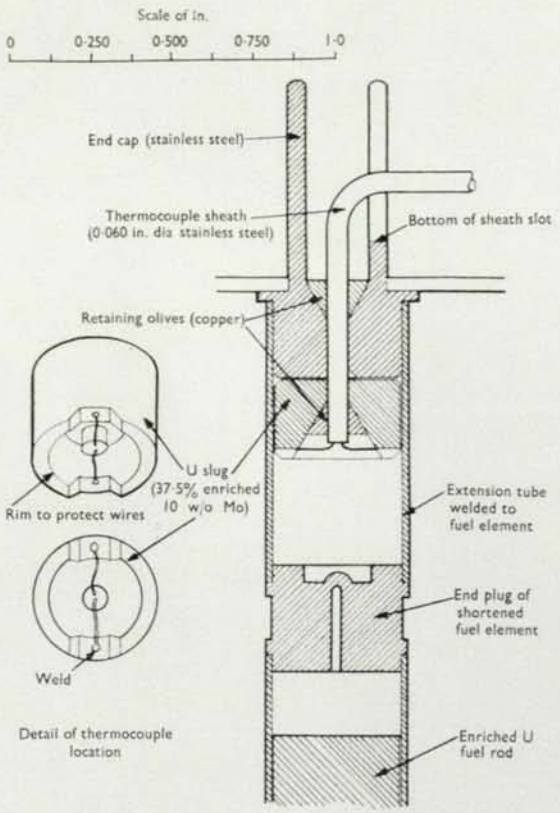


Fig 8 **Construction of intrinsic thermocouple for fuel temperature measurements**

The sequence timer provides more than a shut-down system; it is also used to determine the time of initiation of the reactor pulse in relation to the fire signal and the insertion of the pulse rod. The fire signal triggers the insertion of the pulse rod and starts the timer; at a pre-set time, T_1, afterwards the timer triggers the pulsed source which initiates the reactor pulse; at another pre-set time, T_2, the timer breaks the current supply to the safety block vent valves. T_1 is normally set at a value equal to the insertion time of the pulse rod, 220 ms, and T_2 is set at a value about 10 ms less than T_1. Thus the reactor shut-down signal is generated before the reactor pulse is initiated and the pulse occurs during the interval between the shut-down signal and the beginning of safety block movement. The advantage of this system is that it enables the plateau heating contribution to be reduced to acceptable limits. Because of its fast response time the timer is normally the effective shut-down signal and the scintillation counters and thermocouples have only a safety back-up function.

INSTRUMENTS FOR MEASURING REACTOR PULSE CHARACTERISTICS

General features

Accurate measurements of the reactor pulse characteristics are essential for the proper understanding of the reactor physics. They are also important in checking the safety principles of operation; for instance, the measurements which determine the reactivity temperature coefficient must be made at an early stage of the commissioning in order to prove the safety of the reactor. The pulse characteristic measurements require specialized and novel instrumentation designed to cope with the short time scale and the wide power range of the transients. It is necessary to measure the time variation of reactor power and temperature throughout the pulse and the total energy produced. The temperature is important because it gives an immediate and direct measurement of the pulse yield. The reactor period, measured in the early stages of the pulse before appreciable heating occurs, is important as a direct indication of whether the intended reactivity step was achieved. Useful information about the pulse yield and about the rate of heat distribution can be obtained by measuring the time dependence of the expansion of the fuel rods and their steel claddings during the pulse.

Power and period measurements during the pulse

The reactor power is measured by two scintillation counters using plastic scintillators,[11] which respond mainly to the neutron radiation from the reactor. To obtain the required accuracy of measurement both channels are used as linear, rather than logarithmic, systems and the wide power range is covered by taking parallel signals through different levels of amplification.

The low power counter uses a scintillator of diameter 2 in. and thickness 0·0625 in., mounted on a 2 in. diameter photomultiplier, and is designed to cover any required two-decade section of the power range from 300 W to 50 MW. The lowest power is a convenient one at which to calibrate the counter against a standard fission counter and the highest power allows a measurement of the plateau region of the pulse; these limits also include all the requirements for power measurements in sub-prompt-critical transients. The power range is selected by a remotely controlled switching system which varies the photomultiplier load resistor in the range 3·3 kΩ to 100 MΩ, and the output is taken from the detector via an emitter-follower stage to a fixed-gain amplifier in the control room. Here it is connected to an analogue/digital converter for processing by an on-line computer and to an ultra-violet recorder for display. The computer is used to provide an immediate display of any selected portion of the transient within the detector range and to calculate the reactor period at any specified point. The analogue/digital converter produces an 8-bit signal and has a conversion time of 0·8 μs. It is fitted with a multiplexer so that it can sample and read directly into the computer the instantaneous values of up to 10 different analogue signals. The minimum cycle time of this process (for 10 signals) is 15 μs, which is sufficiently short to deal with the fastest rising reactor pulse events. This system is designed to deal eventually with all the reactor pulse data automatically, but at present some of the data are also recorded on oscilloscope displays and analysed by hand to obtain greater accuracy. The power records obtained with the scintillator have been compared with measurements using ^{235}U and ^{238}U fission counters located in the north east reflector hole, and there is no significant difference between the results obtained by the two methods. For example, the reactor periods do not differ by more than the standard error, which is 1%.

The high power scintillation counter uses a 2 in. diameter scintillator, 6 in. long, mounted on a 2 in. diameter photodiode. It is designed to cover any required two-decade section of the power range from 10 MW to 30 000 MW. The lowest power range allows a measurement of the plateau power level and the highest range includes the peak of the full size reactor pulse. The photodiode current is connected directly to a load resistor in the reactor control room and the voltage across this resistor is measured by an oscilloscope. The sensitivity is controlled by the choice of load resistor, normally in the range 100–10 000 Ω, and by the choice of oscilloscope amplifier. This system is normally used to measure the initial period, the plateau level, the symmetry of the pulse and the pulse width at half peak height. The complete pulse record for powers above 10 MW is integrated to determine the relative contributions of the prompt-pulse and plateau heating.

Initially the counter was mounted on the top framework and the pulse shape records then showed a pronounced asymmetry due to a component in the trailing edge which had a significantly longer period than the rising edge of the pulse. This effect was largely eliminated by mounting the detector in the large irradiation cavity and was therefore attributed to room-scattered radiation. The scintillation counter results have been checked by making similar measurements with a parallel-plate fission counter located just outside the reflector. The counter is operated without any filling gas to obtain a fast-response in direct current operation. The chamber envelope contains two parallel-plate systems, one comprising a ^{238}U plate and a nickel plate, the other comprising two nickel plates. The signals from both systems are recorded and that from the non-fissile chamber is used as a correction for any spurious electrical effects in the ^{238}U chamber. The corrections are less than 10%.

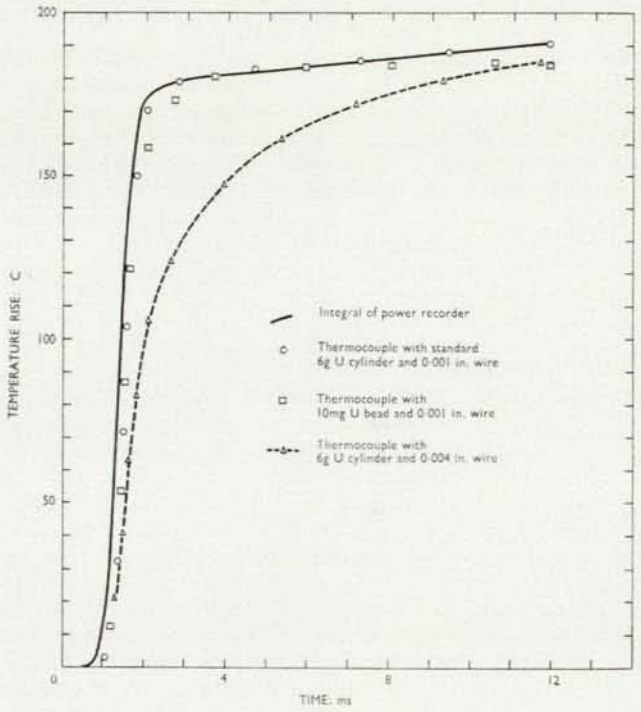


Fig 9 **Time response curves of various intrinsic thermocouples in a 22 cent VIPER pulse**

Temperature measurements

A thermocouple attached to the fuel is used to measure the time dependence of the reactor heating transient and the final temperature reached. For this purpose a response time not greater than 2 ms and preferably of about 0·2 ms is required, and loss of heat from the junction region during the first 30 ms after the pulse must be kept to a minimum. Regarding response time, Morrison[13] has shown that a response time of 0·2 ms can be obtained if the thickness of the thermocouple wires is reduced to 0·001 in. Regarding heat losses, Mullender[12] has shown by experiment with various sizes of uranium cylinder that the one used for the temperature-sensitive shut down channels (0·375 in. diameter and 0·25 in. long) has negligible drop in temperature over a period of 50 ms. The design therefore follows that shown in Fig. 8 with the modification that the thickness of the thermocouple wires is reduced to 0·001 in. for the actual junction and for a length of about 0·1 in. from there to the main cable. Temperature against time curves for several types of intrinsic thermocouples are shown in Fig. 9, with a curve indicating the integral of the measured reactor power data for comparison. The disadvantages of the small bead thermocouple from the point of view of rapid cooling and of the thicker wire thermocouple from the point of view of response speed are apparent. The thermocouple used for the temperature measurements is the one with large U cylinder and thin wire. It is mounted in the position marked ET in Fig. 6. At the outside of the reactor reflector the thermocouple wires are welded to copper extension wires which carry the signals to the amplifier and recording equipment in the control room. The chromel–copper and alumel–copper junctions are maintained at 0°C in a constant-temperature unit. The thermocouple signal is amplified by an integrated circuit system with a feedback loop having an overall gain of 10 over the frequency range 0 to 8 kHz. The output signal is recorded on an ultra-violet chart recorder which has a flat frequency response to ±5% up to 1 kHz and a minimum time resolution of about 0·1 ms. Provision is also made to read the signal into the on-line computer via the multiplexer-ADC system referred to in the previous section. The rms noise level of the complete measuring system depends on the gain being used, but is normally equivalent to a fuel temperature change of 1 to 2°C. The accuracy of the measurements is better than 0·5°C standard error.

For the sub-prompt-critical heating transients used in temperature coefficient measurements the rate of heating is relatively slow; for example, in an 85 cent transient with an initial period of 0·5 s the major part of the 35°C temperature rise occupied 0·9 ms. It is necessary therefore to make an allowance for the cooling of the thermocouple when analysing the sub-prompt transients.

To determine the reactivity temperature coefficient and the maximum fuel temperature the thermocouple measurement must be related to the average and peak values of the

temperature distribution. This is done by using ^{235}U foil activation as a measure of the relative fission rates at the thermocouple position and at other positions throughout the core. In the VIPER 1 core the thermocouple temperature is 0·69 times the core average and 0·50 times the core maximum. These ratios include allowance for the difference between the specific heats and ^{235}U contents of the fuel and of the uranium used in the thermocouple cylinder. The temperature rise in the fuel is mainly due to fission heating, but thermocouple measurements with a tungsten cylinder replacing the uranium show that about 7% of the heating is due to gamma rays.

Yield measurements by activation

The total yield of a reactor transient can be determined by the temperature rise or by integration of the pulse shape measurements. However, it is useful to have another method available to check discrepancies between these two or to provide for the contingency of instrument failure. For this purpose we use the activation of a disc of magnox AL80 placed in a reproducible position in the north west reflector hole. The amount of ^{24}Na produced in the Al(n, α) and Mg(n, p) reactions is measured by a gamma counter with a discriminator admitting only those pulses corresponding to energy deposition between 1·25 and 4·5 MeV. The activation measuring system is calibrated against a fission counter in a slow-transient calibration experiment.

Measurements of fuel and cladding expansion

Provision is made for measuring the displacement of a fuel rod during the pulse. This measurement is intended to determine if the rod behaves as predicted during its rapid expansion and to measure the expansion as a function of time. The displacement of one half of the rod is measured in a special fuel element in which the top end of the stainless steel sheath is removed and a small coil is mounted just above the end of the uranium rod to act as a variable-reluctance transducer. The movement of the uranium during its expansion causes an electrical signal in the coil and this is recorded by a high frequency bridge circuit which has a carrier frequency of 1 MHz. A similar arrangement is used to measure the expansion of the steel cladding. In this way it has been shown that in a full size pulse the steel does not start to expand until about 10 ms after the pulse.

PROCEDURES FOR TRANSIENT AND PULSED OPERATION

The operation of a reactor on short period transients and in the prompt critical region requires special procedures, and it is convenient to distinguish two types of transient. The first includes those in the reactivity increment range from 50 cents to 90 cents, that is, periods from 6 s to 0·2 s, and the second includes the faster transients and the prompt critical pulses.

The range of transients up to 90 cents is of most importance during the commissioning of a new reactor core because it includes the precise determination of the prompt critical point by extrapolation of a period against reactivity relationship, and the determination of the prompt inherent temperature coefficient of reactivity from the variation of reactor period with temperature. Because of the wide range of power levels, periods and time scales involved in these tests, the time-sequence system cannot be used effectively as a means of limiting excess heating. However, the absence of this protection is compensated by the fact that manual control is possible down to periods of 0·2 s, and these transients are therefore referred to as 'manual transients'. In manual transients the reactor is also protected by the scintillation counter channels which trip at a pre-set power level chosen to correspond to an integrated power well within the reactor temperature limit, and it is also protected by the fuel temperature trip channels.

Because of the small margin for error in manually controlled transients at the short-period end of the range, a procedure has been developed to double-check the starting condition of the reactor and the size of the reactivity increment. All the transients are initiated by driving the pulse rod rapidly from its fully out to its fully in position. The pulse rod would generally be smaller than the full size one, but the adjustment of the reactivity increment over delayed critical is achieved not by varying the pulse rod size but by introducing a reactivity bias to vary the starting condition. This reactivity bias is obtained by control rod adjustment and only negative values are allowed, i.e. the starting condition must not be above delayed critical. Thus the reactivity step above delayed critical is given by:

$$\Delta\rho = (\text{pulse rod worth}) - (\text{control rod adjustment})$$

The double-check procedure consists in balancing the reactor at delayed critical first with the pulse rod fully out and then with the control rod adjustment made and the pulse rod raised sufficiently to balance. The control rod adjustment, based on the calibration of that rod, is thus checked against the equivalent movement of the pulse rod based on its independent reactivity calibration. When this process is completed the control rods are locked in position and the pulse rod is driven fully down ready for the initiation of the transient by its rapid insertion. One advantage of this system from the safety point of view is that the starting position of the pulse rod is always the same; if the pulse size was varied by varying the starting position of the pulse rod there would be a greater risk of making an incorrect reactivity increment.

The normal mode of reactor operation is the production of prompt critical pulses. For these and for sub-prompt transients involving reactivity increments of more than 90 cents over delayed critical the sequence of events is too rapid for effective manual control over the reactor. These transients are therefore operated under the automatic time-sequence control system described earlier, and they are referred to as 'automatic transients'. To reduce the probability of pre-initiation during pulse rod insertion the super-prompt pulses must be started at a very low power level, and provision to reduce the power level is therefore included in the automatic transient sequence. As for manual transients, the pulse rod is always driven from its fully out to its fully in position and the increment over delayed critical is determined by a starting bias produced by control rod adjustment. The control rod adjustment is also double-checked in the same way. However in the automatic sequence, after the control rods have been locked in position and the pulse rod lowered, one of the safety blocks is driven down (thus removing 5% reactivity) and is kept down for

15 min to allow the neutron population to fall to a low level. The block is then raised and immediately afterwards the pulse rod is driven in fast. At the time when the pulse rod reaches the core centre the reactor transient is initiated by firing the pulsed neutron source, and in this way the time of occurrence of the pulse is precisely determined.

The correct sequence of operations in both types of transient is ensured by a system of interlocks and keys. The crucial stage, that is the choice of control rod adjustment and its verification by reactivity balance against the pulse rod, is checked independently by two senior members of the reactor staff. A temperature-controlled interlock is provided to prevent pulsed operation if the core temperature is more than 20°C above ambient. This prevents overheating by repetitive pulsing and reduces to acceptable limits any errors in the reactivity balance checks due to temperature drift.

ACKNOWLEDGEMENTS

The VIPER reactor was designed and built at AWRE Aldermaston and we should like to acknowledge the help provided by the Engineering Divisions and the Material Development Divisions. We should like to mention the special contributions of Mr J. Howden, Mr E. Cann and Mr B. H. Martin in engineering, of Mr R. Willows, Mr B. R. T. Frost and Mr N. G. Griffiths in fuel element development and of Mr L. J. Dalby in control and safety instrumentation.

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This is the second of two papers describing the pulsed fast reactor VIPER and its physics characteristics. In Part 1[1], a general description of the reactor and its instrumentation was given. The present Paper describes a series of experiments and calculations to determine the physics characteristics of the reactor and the fission yield and radiation doses in a full size pulse.

The fast pulsed reactor VIPER

Part 2: Reactor physics measurements and analysis

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H. Goodfellow*
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J. W. Weale*

NOTATION

ρ	reactivity
τ	average neutron lifetime
β	effective delayed neutron fraction
T	reactor period
a_i	β_i/β
β_i	fractional yield of ith group of delayed neutrons
λ_i	decay constant of ith group of delayed neutrons
$n(t)$	neutron population at time t
α	$(k-1)/\tau$
k	prompt reproduction factor
F_T	total fissions in prompt pulse
ρ_0	prompt reactivity when pulse starts
μ	temperature coefficient of reactivity
$T(Z)$	temperature of fuel at height Z above the central plane
b	linear expansion coefficient of fuel
$\bar{\nu}$	average number of neutrons per fission

INTRODUCTION

The reactor physics studies on VIPER have centred around the following topics:

(a) The kinetic behaviour of the reactor has been analysed in the reactivity range from delayed to prompt critical. Essentially this is an experimental check of the accuracy with which the reactor behaviour is predicted by the inhour equation using the standard delayed neutron data.

(b) The kinetic behaviour has been analysed in the reactivity range beyond prompt critical using a point-reactor model, with and without allowance for the temperature dependence of the temperature coefficient of reactivity.

(c) The reactivity temperature coefficient has been measured in reactor transients both below and above prompt critical and the results have been compared with the predicted Doppler and expansion coefficients.

(d) The neutron spectrum in the lower energy range and the prompt neutron decay constant at delayed critical have been measured and the results are compared with calculation. These comparisons are important in assessing differences between predicted and measured prompt kinetics and Doppler reactivity coefficients.

An account is given of the work done under these headings and this is followed by a section describing the measurements of fission yield and of radiation doses. Finally there is a section summarizing and discussing the chief discrepancies between experiment and calculation.

REACTOR KINETICS UP TO PROMPT CRITICAL

The kinetic behaviour of the reactor can be described by the standard inhour equation

$$\rho = \frac{\tau}{\beta(\tau+T)} + \frac{T}{\tau+T}\Sigma_i \frac{a_i}{1+\lambda_i T} \qquad (1)$$

This equation[2] is based on the assumption that the flux can be partitioned into spatial and time-dependent functions. This leads to an asymptotic solution of the form

$$n(t) = n(0)e^{t/T}$$

By making small changes to the standard delayed neutron data, equation (1) can be forced to agree closely with the experimental observations. It therefore appears that the underlying assumption is valid for the VIPER reactor in the range up to prompt critical.

The delayed neutron data of Keepin[2] were used for the initial analysis. The contributions from ^{235}U and ^{238}U fission were combined into six groups, whose effectiveness was calculated by the multi-group diffusion theory code

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Table 1: *Delayed neutron data, Keepin and adjusted set for VIPER 1*

Group no.	Group decay constant, s^{-1}	Effective fraction of delayed neutrons: Keepin[2], ^{235}U	Effective VIPER 1 based on Keepin[2]	Adjusted set for VIPER 1
1	0·0127 ± 0·0002	0·000205 ± 0·00002	0·000253	0·000246
2	0·0317 ± 0·0008	0·001363 ± 0·00007	0·001449	0·001411
3	0·1150 ± 0·003	0·001221 ± 0·00014	0·001340	0·001305
4	0·3110 ± 0·008	0·002458 ± 0·00011	0·002928	0·002903
5	1·4000 ± 0·081	0·000717 ± 0·00006	0·001057	0·001100
6	3·8700 ± 0·369	0·000260 ± 0·00003	0·000253	0·000315
Total delayed fraction		0·0065 ± 0·0003	0·00728	0·00728

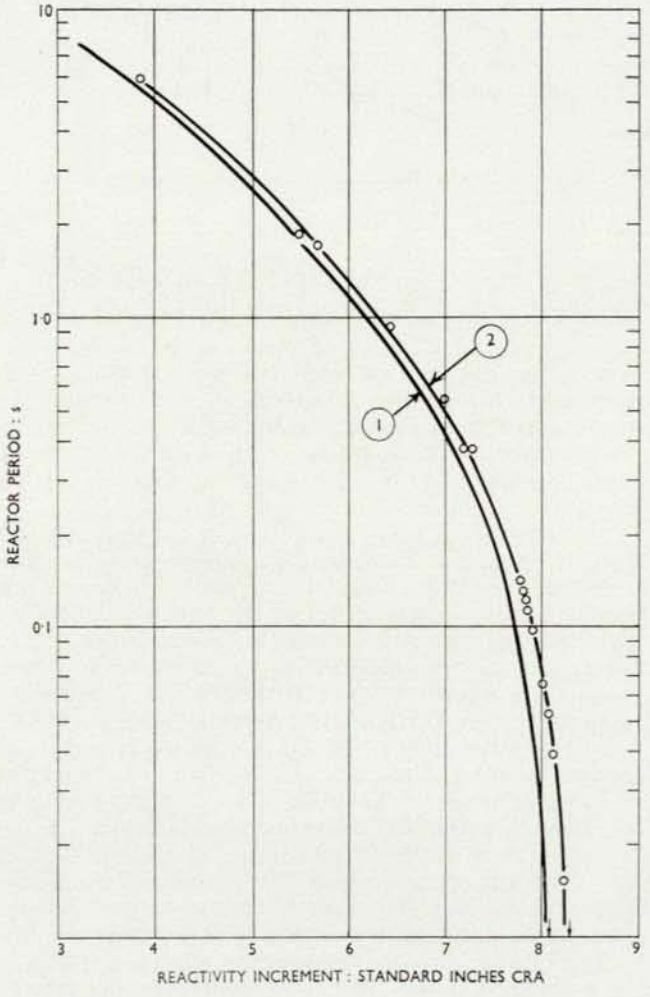

Fig 1 **Reactor period against reactivity increment (sub-prompt critical)**

Table 2: *Accuracy of period measurements, delayed critical region*

Reactivity increment, cents	Detector	Time interval analysed, s	Reactor period, s
94	Scintillation	0·88–1·08	0·147 ± 0·001
	Scintillation	1·08–1·31	0·146 ± 0·0003
	^{235}U fission	0·12–0·60	0·145 ± 0·002
63	^{238}U fission	27·0–32·0	2·22 ± 0·07
	^{235}U fission	8·0–20·0	2·26 ± 0·01

SWAN.[3] The basic Keepin data for ^{235}U and the effective data for the VIPER core are listed in Table 1.

The experimentally observed periods resulted from various step additions of reactivity which were made by driving the pulse rod from its fully out to its fully in position. Four pulse rods of varying fuel contents were used, and the precise reactivity addition was obtained by adjusting a control rod to make the starting position a known amount below delayed critical. It was convenient to use one control rod, CRA, as the reference for all reactivity calibrations. Its worth per unit length did not vary by more than 2% over its position range from 4 in. to 8 in. The standard reactivity unit was taken as the worth of CRA movement from 6 in. to 7 in. The measured reactor period corresponding to this increment is 67·15 s and, using the effective VIPER 1 delayed neutron data, this is equivalent to 12·35 ± 0·10 cents. This unit was shown to be independent of the pulse rod position and of the positions of the other control rods within the accuracy of calibration. The total worths of each of the four different pulse rods were determined by balancing them against CRA and the accuracy achieved was ± 0·02 standard in. of CRA (i.e. ± 0·24 cents). Some direct calibrations in terms of reactor period were done to check consistency. On this basis we estimate that the standard error of the reactivity increments was 0·20 cents, although the relative values of reactivity increment for a single pulse rod were more accurate, generally within ± 0·005 in. (i.e. ± 0·06 cents).

The reactor periods were measured by recording the time dependence of the neutron flux using pulse fission chambers

Table 3: *Results of period measurements in delayed critical region*

Pulse rod details		Reactivity increment above delayed critical, standard CRA in.	Measured reactor period, s	Reactivity insertion (cents)			
				Expected increment based on 1 in. CRA = 12·35 cents	Effective increment using Keepin VIPER data in equation (1)	Expected increment based on 1 in. CRA = 12·06 cents	Effective increment using adjusted data in equation (1)
Mass of ^{235}U g	Total worth, standard CRA in.						
421	5·67 ±0·03	3·87	5·84	47·79	47·05	46·65	46·22
		5·50	1·84	67·92	67·47	66·30	66·53
557	7·29 ±0·01	5·69	1·70	70·27	68·75	68·60	67·82
		6·44	0·92	79·43	77·80	77·54	76·94
		6·998	0·54	86·42	84·18	84·36	83·44
		7·29	0·372	90·03	87·81	87·88	87·17
789	9·824 ±0·02	7·201	0·372	88·94	87·81	86·81	87·17
		7·777	0·140	96·05	94·35	93·75	93·78
		7·824	0·130	96·63	94·69	94·32	94·34
		7·844	0·122	96·88	95·02	94·56	94·65
		7·858	0·112	97·05	95·35	94·73	95·03
		7·924	0·0967	97·86	95·90	95·52	95·62
		8·024	0·0647	99·10	97·16	96·73	96·95
		8·124	0·0386	100·34	98·27	97·94	98·15
		8·224	0·0152	101·57	99·37	99·15	99·38
867·5	10·654 ±0·02	8·097	0·052	100·00	97·68	97·61	97·52

or a scintillation counter. A ^{235}U fission chamber covered the range 1–50 W, and the range up to a few kilowatts was covered by a ^{238}U fission chamber and by the low power scintillation counter described in Part 1. The counter signals were recorded in the on-line-computer store and were fitted to an exponential function. The asymptotic behaviour was clearly established by the reproducibility of the period over various time intervals and by the comparison between different detectors. This is illustrated in Table 2, which lists the observed period as a function of time from commencement of the transient.

The reactor can be operated in two types of transient, manual and automatic, the latter being restricted by a time sequence interlock which limits the duration of the transient to a maximum of 1 s from the start of the reactivity insertion. The manual mode was used for reactivity increments up to 95 cents, equivalent to a period of 0·15 s, and the automatic mode was used for greater increments.

The variation of reactor period with reactivity increment expressed in standard inches of CRA can be obtained from equation (1) using the calibration figure of 12·35 cents per standard inch. This, shown as curve 1 in Fig. 1, represents the expected relationship for increments larger than 10–12 cents as used in calibrating CRA. The observed relationship is shown as curve 2. There is a noticeable discrepancy for periods shorter than 1 s, the observed periods being longer than expected. Evidently there is a need to adjust the calculation method or the data to bring the prediction into agreement with experiment and, since it is convenient to continue using the simple inhour equation, we chose to adjust the relative abundances of the delayed neutron groups to obtain agreement. The required changes in all cases are less than two standard deviations of the calculated effective values. This standard deviation includes a contribution due to the basic data quoted by Keepin[2] and a contribution of 2–3% due to the calculated effectiveness factors.

The data shown in Fig. 1 are listed in a slightly different form in Table 3 which shows the complete set of measurements. The reactivity increments are listed in terms of CRA standard inches as determined by the comparison between pulse rod and control rod worths at delayed critical. These increments are first converted to the absolute (cent) scale using 1 CRA inch ≡ 12·35 cents, and are listed under the heading 'expected reactivity increment' (column 5). By substituting the observed period in equation (1) using the delayed neutron data based on Keepin, we obtain the 'effective increment' (column 6). The discrepancy now takes the form of a reactivity difference between columns 5 and 6 varying from about 0·5 cents for a 50 cent insertion to 2·2 cents near prompt critical. The purpose of the adjustment exercise is to find a set of delayed neutron relative abundances which make these two columns identical.

The form of equation (1) shows how this adjustment may be accomplished. The first term, containing the prompt neutron lifetime is negligible below prompt critical. The contribution from a particular delayed neutron group i varies most in the range from $T \simeq 1/(5\lambda_i)$ to $T \simeq 5/\lambda_i$. For $T < 1/(5\lambda_i)$, the ith group contribution is virtually constant and for $T > 5/\lambda_i$ the ith group contribution rapidly becomes negligible. Thus it is easy to determine which groups require

Table 4: *Results of period measurements in prompt critical region*

Reactivity increment, cents	Reactor period, μs	Reciprocal period, $s \times 10^{-3}$	Pulse width ($W_{1/2}$) at half peak, μs	Ratio of $W_{1/2}$ to initial period
101·7	1170 ± 100	0·855	4500 ± 250	3·8 ± 0·4
101·7	1500 ± 100	0·667		
105·6	460 ± 15	2·18		
109·7	235 ± 5	4·24		
111·9	187 ± 3	5·35		
111·9	250 ± 4 P	4·00		
115·9	191 ± 4 P	5·23		
115·6	146 ± 2	6·86	620 ± 20	4·2 ± 0·2
118·8	117 ± 2	8·55		
120·5	109 ± 2	9·17		
121·8	103 ± 2	9·71	465 ± 20	4·5 ± 0·2
121·8	109 ± 2	9·17	505 ± 20	4·6 ± 0·2
121·7	132 ± 2 P	7·58	600 ± 20	4·5 ± 0·2
121·8	108 ± 2	9·26		
121·8	107 ± 2	9·35		
121·8	125 ± 2 P	8·00		
121·7	120 ± 2 P	8·33		
123·0	100 ± 2	10·0		

adjustment simply by inspecting the variation in the differences between the 'expected' and the 'effective' reactivity increment (i.e. between columns 5 and 6). By adding about 0·7% to group 4 and about 1% to each of groups 5 and 6, re-calculating the CRA calibration and repeating the above analysis of reactivity increments, we obtained close agreement between the expected and calculated values as shown in the last two columns of Table 3.

The adjusted set of delayed neutron data shown in Table 1 leads to a calibration of 1 CRA inch ≡ 12·06 cents and to a definition of prompt critical as the insertion of 8·292 standard CRA inches. The standard deviation of this number, based on the fit achieved to a set of reactivity insertions with individual standard deviations of approximately 0·2 cents, is 0·01 standard CRA inches or 0·12 cents.

REACTOR KINETICS IN THE PROMPT CRITICAL REGION

The kinetic behaviour of the reactor has been measured in the prompt critical region for reactivity increments varying from 1·00 to 1·23 dollars above delayed critical. For each pulse the quantities measured were the initial reactor period, i.e. before significant heating starts, and the reactor power as a function of time throughout the pulse.

The methods used for period measurement are described in the section on 'Reactor kinetics up to prompt critical' and in Part 1. The most reliable measurements are obtained from the low power scintillation counter which has a maximum range of 50 MW. At this power level in the slowest transients measured, which had a period of 1 ms, the fuel temperature rose by less than 1°C. The periods obtained from the high power scintillation counters are up to 5% longer than the low power periods. This discrepancy is partly due to difficulty in excluding background effects in the short range available to the high power instruments before significant heating begins.

The measured initial periods for a range of typical pulses as a function of reactivity increment are listed in Table 4 and the reciprocal period α as a function of reactivity increment is shown in Fig. 2. The reactivity scale is based on the adjusted delayed neutron data for which the increment between delayed and prompt critical is equivalent to 8·292 standard inches of CRA. Most of the experimental points in Fig. 2 lie on a straight line. For those marked P the time-sequence records show that pre-initiation occurred, i.e. the pulse developed spontaneously from the intrinsic neutron population before insertion of the pulse rod was complete. For these, α is smaller than expected because the pulse corresponds to a reactivity insertion which was smaller than intended. The standard error of each point, indicated in the usual way, includes the statistical error involved in fitting an exponential to the curve of reactor power against time and the error arising from the uncertainty in instrument bias which has to be subtracted before the curve fitting is done. The line shown in Fig. 2 was fitted to the points by a least squares method, excluding the points marked P and the prompt critical point. Equation (1) can be made to fit the observations by setting the ratio β/τ equal to the slope of this line. The observed slope is $\beta/\tau = 4{\cdot}36 \pm 0{\cdot}05 \times 10^4$/s.

The calculated slope, based on the calculated delayed fraction β given in Table 2 and the lifetime τ calculated by multigroup diffusion theory[3] described in the next section is $5{\cdot}49 \times 10^4$/s.

The reactor power as a function of time throughout the pulse (the 'pulse shape') was recorded by the high power scintillation counter and by the evacuated fission chamber. Initially the scintillation counter was placed outside the reflector, about 2 m from the core. In this position the pulse shape differed from that observed when the counter was placed inside the reflector, in the time-of-flight exit beam and shielded from room radiation. In larger pulses, the peak signal from the unshielded counter appeared to be about 100 μs later and the pulse had a rather high tail which broadened it and obscured the transition from pulse to plateau. The effect was attributed to capture gamma rays

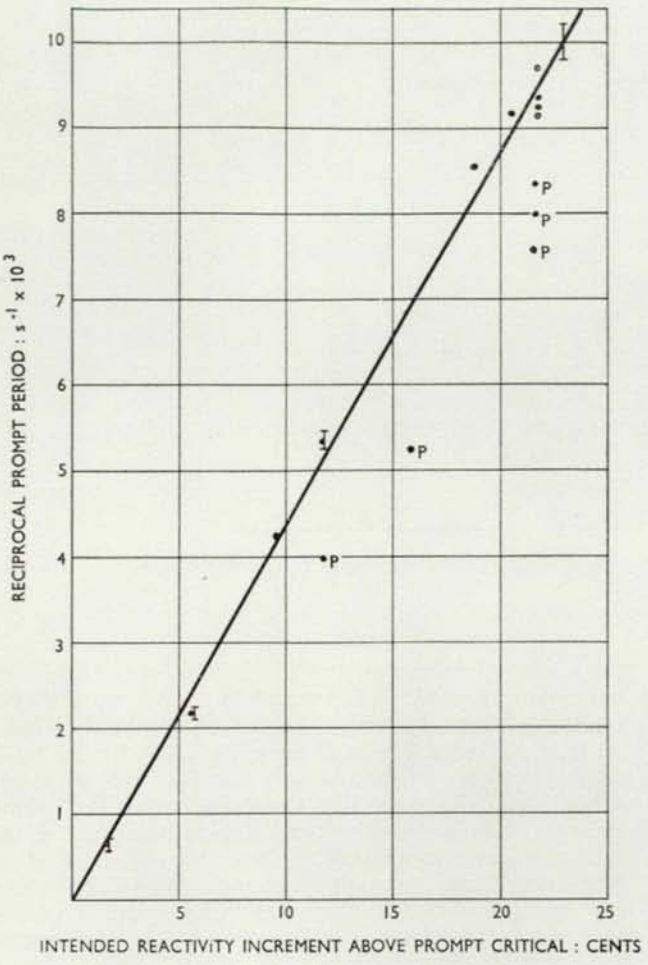

Fig 2 **Reciprocal prompt period against reactivity increment**

produced by room-scattered neutrons. Since the pulse shape measured by the shielded scintillation counter and by the evacuated fission chamber agree closely, we consider this to be a true representation of the pulse shape. Figs 3 (*a*), (*b*) and (*c*) show typical pulse shapes measured in this way for a range of pulse sizes. Table 4 shows the complete set of results including the pulse width measurements for those pulses in which the shielded scintillation counter and the evacuated fission chamber were used. According to the simple kinetics model with constant temperature coefficients (Part 1), the pulse width at half peak height is equal to 3·52 times the initial reactor period. However, the experimental results show that the ratio is larger than this and that it tends to increase with pulse size. This discrepancy is discussed in the summary and discussion.

Temperature coefficient of reactivity

The main contributions to the temperature coefficient of reactivity are: axial expansion of the fuel, changes in effective resonance absorption in ^{238}U (the Doppler coefficient),

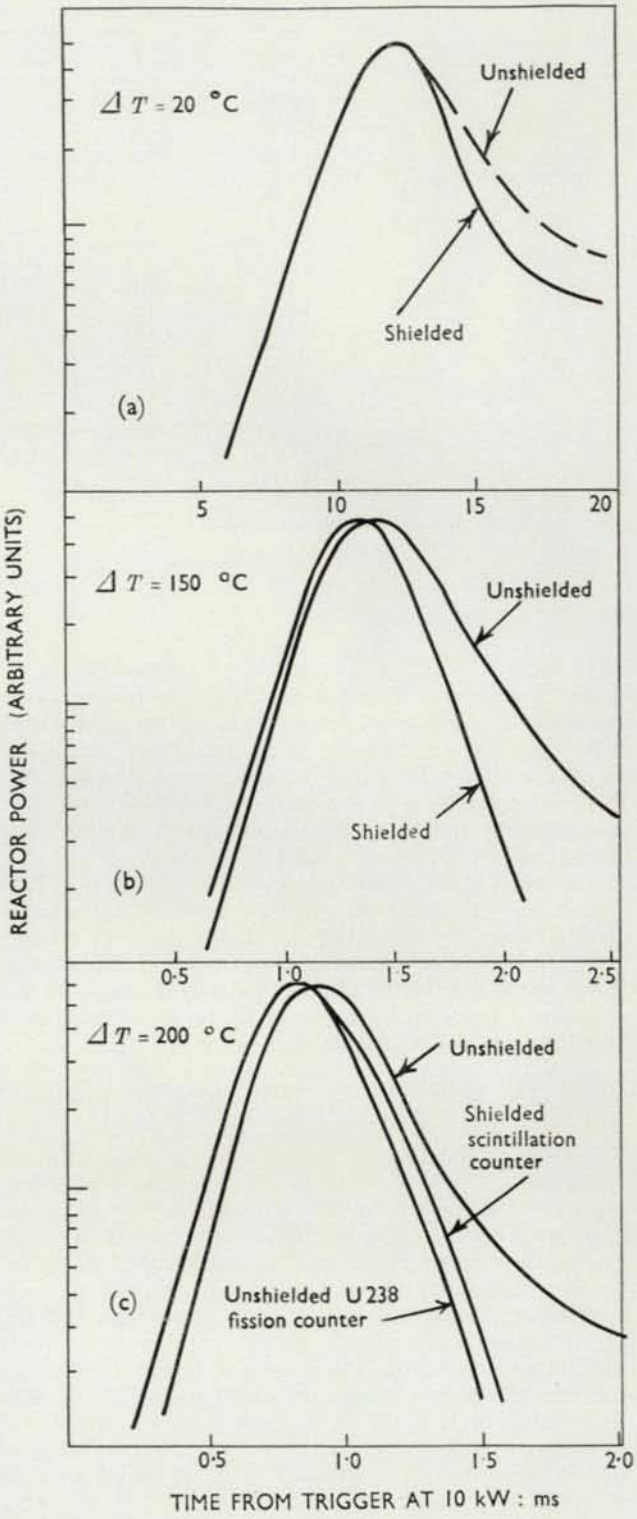

Fig 3 **Pulse shape**

and structural effects such as fuel can expansion or relative movements of reactor components. Of these, the first two can be regarded as prompt effects since they respond effectively instantaneously when the fuel is heated. The structural effects take place only as heat is transferred from the fuel to the rest of the reactor and the rate of heat transfer is slow compared with the time scale of fast transients. The chief characteristics of a pulsed reactor are therefore determined only by the prompt components of the temperature coefficient, and for this reason the experimental and theoretical work has been concentrated on measuring and understanding their effects.

In the following sections are described the determination of the prompt temperature coefficient in fast sub-prompt transients by measuring the variation of reactor period with fuel temperature, the determination of the prompt temperature coefficient by measuring the yield as a function of reactivity increment in super-prompt pulses, and the calculation of the fuel expansion and Doppler effect components. We also include the results of an experiment to determine the effective temperature coefficient when the reactor is heated relatively slowly. Although not required for an understanding of the pulse behaviour of the reactor, this coefficient is needed to determine the temperature change which can safely be allowed between the time when the reactor is balanced at delayed critical and the time when the pulse is fired.

Derivation of the prompt temperature coefficient from sub-prompt transients

When a transient is allowed to reach a power level such that significant heating takes place in the fuel, the change in reactivity produced by the temperature coefficient causes the reactor period to lengthen. This change in the power/time curve, which would otherwise be exponential, is a measure of the prompt temperature coefficient. The particular importance of this experiment is that it provides a means of checking the prompt component of the temperature coefficient before operating the reactor in the prompt critical region. Several measurements of this type were made during the commissioning phase and two of them were repeated later with extra instrumentation to obtain an accurate comparison with the coefficient measured in the prompt critical region.

The transient signals from the thermocouple and the low power scintillation counter were recorded in the on-line-computer store. The power record was checked by recording the output from a ^{238}U fission chamber using a 400-channel time sorter and this agreed with the scintillation result within the experimental errors. The results were compared with a set of calculations of the power and temperature variations corresponding to a range of temperature coefficients for the same reactivity increment. The calculations were done with the one-group space-independent kinetics code RKSF[4] using the following input parameters:

Initial power	0·1 W
Initial core temperature	20°C
Initial reactivity	0 (i.e. delayed critical)
Heat capacity of fuel	$2{\cdot}95 \times 10^4$ J/°C
Neutron lifetime	0·17 μs
Delayed neutron parameters	adjusted set from Table 1
Temperature coefficient (per °C average temperature)	(a) $-1{\cdot}0 \times 10^{-5}$
	(b) $-1{\cdot}1 \times 10^{-5}$
	(c) $-1{\cdot}2 \times 10^{-5}$

The reactivity was added as a step input. The code calculates the average fuel temperature assuming 166 MeV per fission and produces a set of values of the reactor power and temperature as a function of time. The results of one set corresponding to an insertion of 84 cents, i.e. a reactor period of 0·5 s, are shown in Fig. 4.

To facilitate comparison between calculated and experimental results and to avoid normalization difficulties the deviation of power from exponential was plotted as a function of average temperature rise. This is shown in Fig. 5 as a series of curves for different temperature coefficients at the same reactivity insertion. Here P_Z is the power corresponding to zero temperature coefficient, i.e. the simple exponential shown in Fig. 4, and P_{tc} is the power corresponding to a temperature coefficient 'tc'. Thus if the temperature coefficient is zero, $P_Z:P_{tc}$ remains unity, and if it is negative, $P_Z:P_{tc}$ increases with fuel temperature. The results of two transient measurements, both corresponding to a reactivity step of 84 cents but one terminated at 1 MW and the other at 2 MW, are shown in Fig. 5 in the form of $P_Z:P_{tc}$ curves. A cooling correction has been applied to the temperature measurements, increasing them by 15%, and the average fuel temperature is taken to be $1{\cdot}45 \pm 0{\cdot}08$ times the thermocouple temperature. The results are consistent with a temperature coefficient in the range $1{\cdot}0$–$1{\cdot}1 \times 10^{-5}$/°C. The main source of error is in the ratio $P_Z:P_{tc}$, which is derived for the experimental curves by measuring the period before heating begins and extrapolating to the higher power levels. Assuming errors of 2% on the initial period and 2% on the measured power, the standard deviation of the ratio is about 3%. At a temperature rise of 30°C the difference between two adjacent calculated curves of $P_Z:P_{tc}$ is 7%.

Measurement of the prompt temperature coefficient in super-prompt pulses

The simple theory outlined in Part 1 predicts the following relation between pulse yield, reactivity and temperature coefficient:

$$F_T = \frac{2\rho_0}{\mu}$$

Thus the temperature coefficient can be determined from a set of values of yield and reactivity. The yield is measured in terms of temperature rise at the core thermocouple. To determine the true core-averaged temperature coefficient it is necessary to subtract the plateau heating contribution from the measured temperatures and then to convert them to core-average values.

The plateau heating rate, which is constant according to the simple theory, has been obtained by four different methods and the results are:

From the gradient of the temperature record	0·4–1·2 °C/ms
By integration of the pulse shape records	0·5–1·0 °C/ms
By comparing total yields for different plateau lengths	$0{\cdot}5 \pm 0{\cdot}2$ °C/ms
By calculation, using equation (13 of Part 1	$0{\cdot}54 \pm 0{\cdot}10$ °C/ms

Table 5: *Prompt-critical pulse yields as measured by the fuel thermocouple temperature*

Reactivity insertion, cents	Temperature rise, °C		
	Total	Plateau correction	Pulse
101·7	17·8 ± 0·5	5 ± 2	13 ± 2
105·6	61 ± 1	8 ± 3	53 ± 4
109·7	103 ± 3	12 ± 4	91 ± 4
111·9	112 ± 2	6 ± 2	106 ± 3
118·8	188 ± 2	6 ± 2	182 ± 3
120·5	196 ± 2	6 ± 2	190 ± 3
121·8	202 ± 2	6 ± 2	196 ± 3
121·8	204·5 ± 2	6 ± 2	199 ± 3
121·7	214 ± 2	14 ± 4	200 ± 5
123·0	222 ± 2	5 ± 2	217 ± 5

The temperature gradient measurement is subject to uncertainties arising from heat loss along the thermocouple wires and temperature redistribution in the uranium cylinder. In large pulses the shape is uncertain in the plateau region because the intense background of γ-rays from pulse fission products obscures the early part of the plateau. However the comparison of total yields for identical pulses with different plateau lengths eliminates the uncertainties arising in the early part of the plateau and, since the plateau lengths are quite accurately measured, we consider that this is the most reliable direct measurement. The calculated plateau heating is derived from the re-arranged expression

$$\frac{dT}{dt} = \frac{2\Sigma_i \lambda_i a_i}{\mu}$$

using the adjusted delayed neutron parameters given in Table 1 and a value of $1{\cdot}5 \times 10^{-5}$/°C for μ. The calculation is subject to some errors arising from the over-simplification of the calculation and these are discussed later. The best value of the plateau heating rate is taken to be 0·5 ± 0·2 °C/ms for full size pulses and 0·4 ± 0·15°C/ms for smaller pulses.

The total yields, plateau contributions and pulse yields for a range of reactivity insertions are listed in Table 5. The standard deviations of the pulse yields include the temperature measurement uncertainty, which ranges from 2°C in large pulses to 0·5°C in small ones, and the uncertainty in plateau contribution, which is generally the dominant item. Fig. 6 is a graph of pulse yield as a function of reactivity insertion. The straight line, which has been fitted to all the points except the origin by a least squares method, has a gradient of 0·107 cents/°C. This corresponds to a temperature coefficient of $(15{\cdot}6 \pm 0{\cdot}4) \times 10^{-6}$/°C temperature rise at the core thermocouple. There is no evidence of curvature in the graph of yield against reactivity, indicating that the simple theory is valid to this extent.

The measured temperatures are converted to core-average values by dividing by the factor 1·45 ± 0·08. This includes a factor of 1·173 ± 0·065 relating the core-average fission rate to the fission rate per gram of fuel at the thermocouple position, and a factor of 1·236 ± 0·036 relating the specific heats and ^{235}U contents of the fuel to those of the thermocouple uranium cylinder.

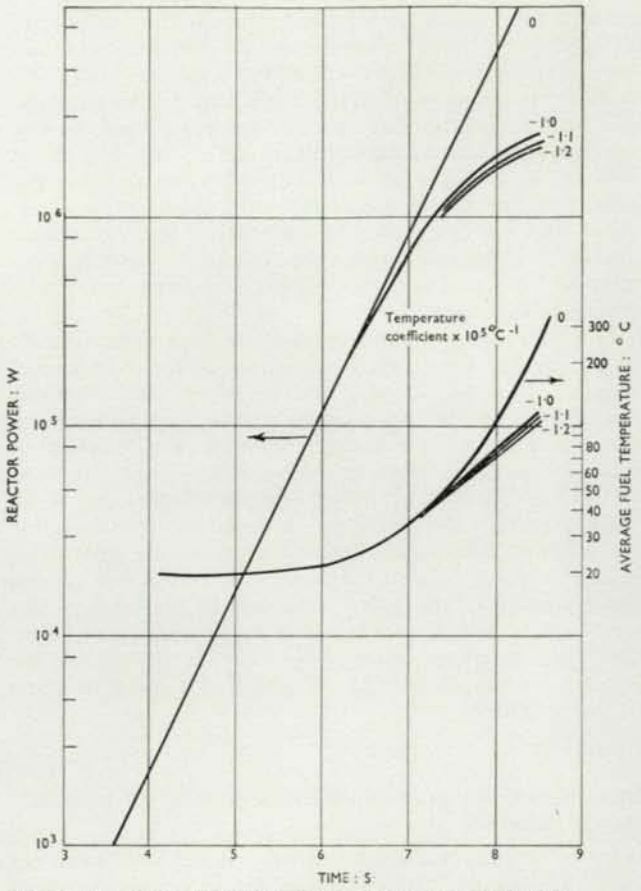

Fig 4 **Reactor power against time (sub-prompt transient)**

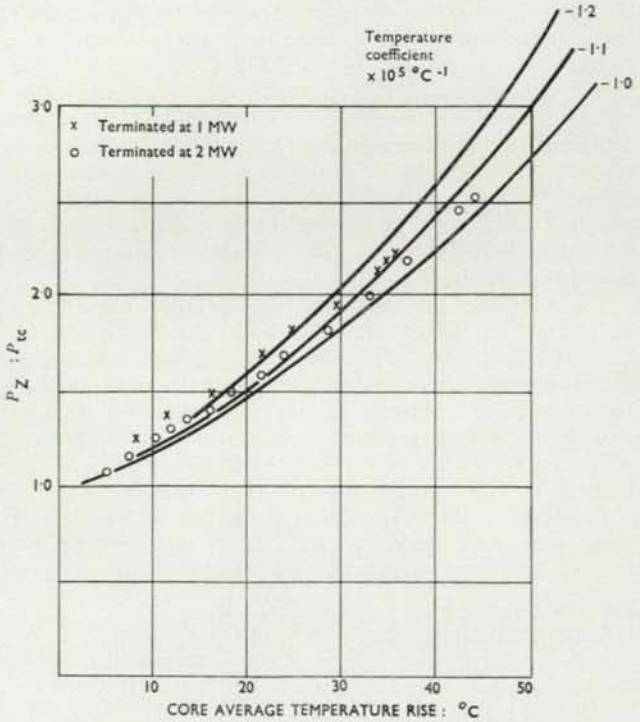

Fig 5 $P_Z : P_{tc}$ **against core average temperature rise**

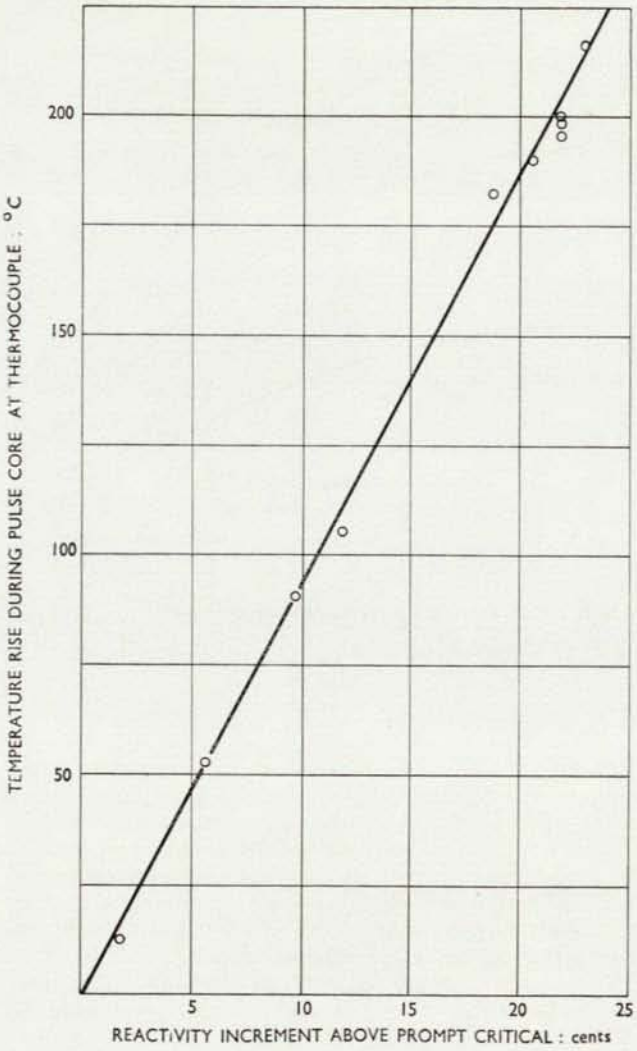

Fig 6 **Pulse yield (temperature rise at core thermocouple) against reactivity increment**

The core-averaged temperature coefficient derived from the super prompt pulses is therefore $(1{\cdot}08 \pm 0{\cdot}07) \times 10^{-5}/°C$ average fuel temperature rise. This figure is smaller than that quoted in an earlier interim report[5] because in the earlier analysis the thermocouple temperature losses were underestimated and the relation between the core-average fission rate and the fission rate at the thermocouple position was inaccurate.

Calculation of the prompt temperature coefficient due to fuel expansion and bowing

The temperature coefficient due to expansion and bowing of fuel rods has been calculated for a temperature distribution corresponding to the measured ^{235}U fission distribution. This assumes that the temperature distribution is the same as the fission heating source, which is true if heat transfer along the fuel pins or from the fuel pins to the cans is negligible, if all fission fragments are retained within the fuel and if the heating contribution from prompt or short-lived gamma emitters has the same distribution as the ^{235}U fission. Direct measurement of the fuel can expansion shows that heat transfer to the can is negligible until at least 20 ms after the pulse. Calculation shows that heat transfer along the length of the fuel pin is not appreciable until a few hundred milliseconds have elapsed. Since the fission fragment maximum range is about 12 mg/cm^2 and a layer of this thickness represents only 0·3% of the rod volume, effectively all fission fragments are retained. Since gamma heating accounts for about 10% of the total temperature rise in the fuel it does not appreciably perturb the ^{235}U fission heating distribution.

The information required for the temperature coefficient calculation is the fuel temperature distribution and the reactivity worth per unit mass of fuel. For the bowing coefficient it is also necessary to take into account the constraints applied to prevent lateral movement of fuel rods, such as the displaced holes in the lattice plates (Part 1).

The ^{235}U fission distribution was measured using 0·25 in. diameter fission chambers and thin foils of ^{235}U. The foil method was calibrated against the fission counters and the fission counters themselves were calibrated against standard foils.[6] The overall standard deviation of the fission distribution measurements in terms of the count rate of a standard reactor power monitor fission counter is approximately 4%, arising from the absolute calibration of the fission chambers ($\simeq 2\%$), the relative accuracy of the foil measurements (3%), the effects of heterogeneity of core structure (2%), and possible streaming effects due to the empty fuel channel (2%). There are some irregularities in the distribution due to the lattice plates, the fuel pin notches, the effect of the pulse rod hole and the effects of the control rods. The measured distributions are shown in Fig. 7.

The reactivity worth of fuel as a function of position was determined by perturbation experiments using small cylindrical samples which could be driven from outside the reflector into empty fuel element holes in the core. Measurements were made at the level of the central plane and at heights of 2 in. and 4 in. above it in three of the experimental access holes shown in Fig. 6 of Part 1. The experiments included worth measurements of ^{235}U and ^{238}U separately, studies of streaming effects, sample size effects and the change in sample worth due to the insertion of the (0·421 kg ^{235}U) pulse rod. The absolute accuracy of the worth measurements ultimately depends on the calculated delayed fraction used in calibrating the control rods, but the relative accuracy is 2%, standard error. The measured distributions are shown in Fig. 8.

In calculating the temperature coefficient the core was treated as a cylinder divided into annular zones 1 cm thick. For each zone the movement of fuel at a height Z above the central plane was calculated as

$$E(Z) = \int_0^Z bT(Z)\mathrm{d}Z$$

using the observed ^{235}U fission distribution to determine the temperature distribution. Similarly the displacement due to bowing caused by the radial temperature gradient was calculated, first for the situation where the fuel rod centres are fixed but their ends are unrestricted and, second, for the situation in which both the centres and the ends are fixed by the offset arrangement of the positioning holes in the lattice plates. The reactivity changes due to axial expansion

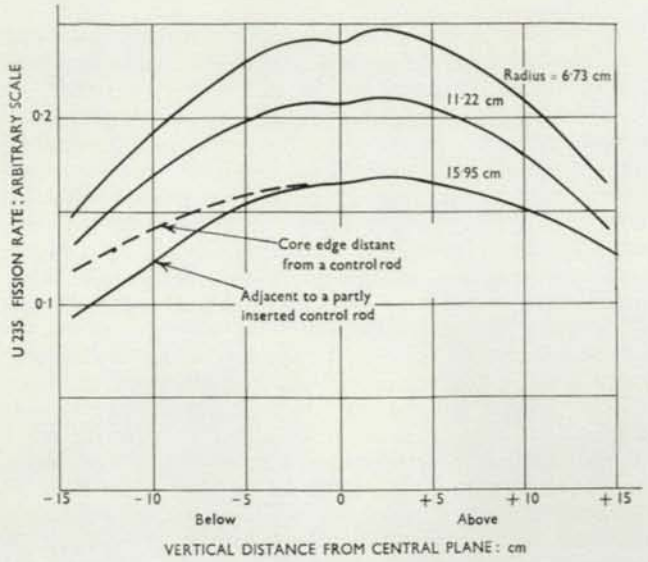

Fig 7 **^{235}U fission distribution**

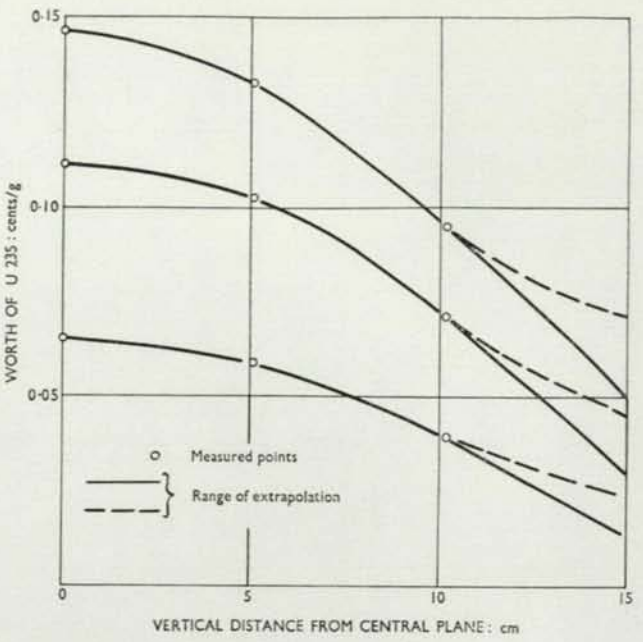

Fig 8 **^{235}U worth distribution**

and radial bowing displacements were calculated by numerical integration of the products of the displacements and the vertical and radial worth gradients.

The value used for the linear expansion coefficient of the fuel was $(15{\cdot}3 \pm 0{\cdot}15) \times 10^{-6}/°C$ which is the average value from 25°C to the core average temperature in a full size pulse (325°C). The calculated axial expansion coefficient is $(-7{\cdot}0 \pm 0{\cdot}9) \times 10^{-6}/°C$ (average fuel temperature rise) and the bowing coefficient, allowing for the lattice plate constraint, is $(+0{\cdot}03 \pm 0{\cdot}01) \times 10^{-6}/°C$. Clearly the bowing coefficient is negligible, but without lattice plate constraint it would be $0{\cdot}7 \times 10^{-6}/°C$, the sign depending on whether the ends of the rods moved away from the reactor centre or towards it. The quoted standard errors are due mainly to uncertainties in the extrapolation of the fission and worth distributions to the outer regions of the core and in defining the core boundary. Smaller errors arise from the assumption of symmetry in the distributions and from the use of an average value of the expansion coefficient.

Measurements of the displacement of the end of a central fuel pin with a transducer provide a check of the validity of the expansion coefficient. The temperature profile along the pin and the temperature rise of the in-core intrinsic thermocouple can be related directly through the measured fission distribution. For a temperature rise of 188°C at the thermocouple the observed expansion of the upper half of one of the fuel pins was $27{\cdot}3 \pm 0{\cdot}3 \times 10^{-3}$ in., compared with a calculated expansion of $27{\cdot}5 \pm 2{\cdot}7 \times 10^{-3}$ in.

Calculation of the Doppler temperature coefficient

Doppler coefficients were calculated using the Brissenden–Durston method[7] with variations to take account of the VIPER geometry.[8] Resonance cross sections were calculated for ^{235}U and ^{238}U at 300°K, 600°K and 900°K, and for each temperature the neutron fluxes and flux-weighted cross sections were calculated for the fuel and moderator regions of the VIPER core. These were then combined by flux and volume weighting to produce effective group cross sections for a homogeneous criticality calculation at each temperature. The fractional changes in k corresponding to changes of (uniform) core temperature from 300°K to 600°K and from 600°K to 900°K are listed in Table 6.

Table 6 also includes the result of a calculation in which the higher temperature situation was characterized by 600°K data for ^{238}U and 300°K data for ^{235}U. This was done to separate the effects of ^{235}U and ^{238}U, and it shows that the predominant contribution is due to the ^{238}U. The ^{235}U contribution is small and positive. The results of the 300–600°K and 600–900°K steps show that the overall Doppler coefficient is not a linear function of fuel temperature. As shown in Fig. 9, the results can be fitted to a curve of the form

$$\frac{\partial k}{\partial T} = AT^{-3/2}$$

Because the temperature distribution in VIPER is not uniform, the average Doppler coefficient was calculated by averaging over the whole core the quantity

$$\Delta k = \int_{300°K}^{T} AT^{-3/2} \mathrm{d}T$$

This, converted to the reactivity change per °C average fuel temperature rise, is also shown in Fig. 9. The average Doppler coefficient for a full size pulse, i.e. an average fuel temperature rise of 300°C, is $5{\cdot}77 \times 10^{-6}/°C$ (as a measure of the non-linearity the corresponding coefficient for a 50°C average temperature rise is $8{\cdot}8 \times 10^{-6}/°C$).

There is evidence[9] that the resonance data used in the calculation overestimate the (positive) ^{235}U contribution.

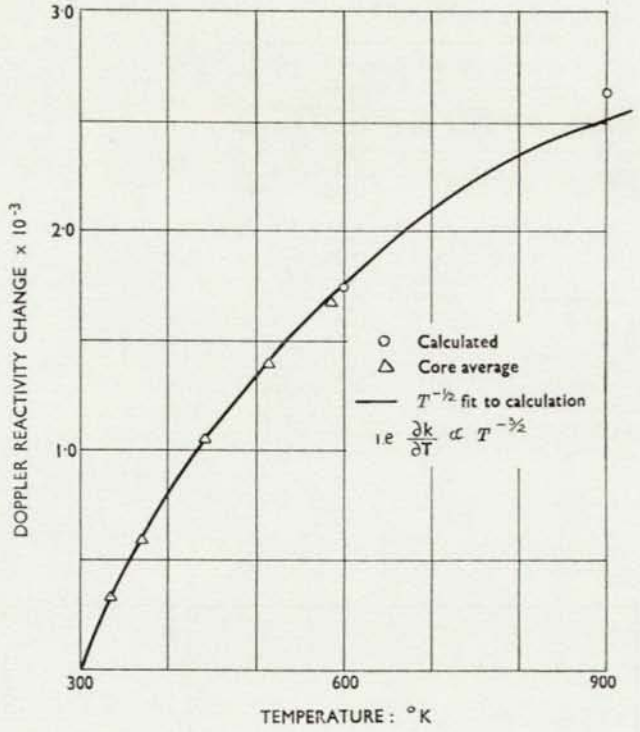

Fig 9 **Variation of calculated Doppler temperature coefficient with temperature**

The estimated accuracy of the Doppler coefficient calculation method is 10–15%, but no adequate assessment has been made of the accuracy of the basic data.

Measurement of the temperature coefficient corresponding to relatively slow temperature changes

The temperature coefficient corresponding to low rates of temperature change was measured by operating the reactor at a power of 1·5 kW for 13 min and recording the temperature and reactivity during the subsequent cooling. For this experiment extra thermocouples were placed in the core. By the end of the 13 min power run the core thermocouple indicated a rise of 25°C and, although the power was then reduced to 1 W, the core edge thermocouple continued to heat owing to conduction from the core centre. After about 30 min, when all the thermocouples in the core appeared to be cooling steadily, the reactor was maintained at delayed critical at a power of 1 W by adjusting a calibrated fine control rod. During the next hour, while the reactivity adjustment was 0·61 cents, thermocouples in the central region cooled by between 6·5 and 7·5°C. A similar thermocouple near the edge of the core, where the temperature rise was only 70–80% of that observed on the central thermocouples, cooled by 5°C. On this basis the temperature coefficient is $0{\cdot}1 \pm 0{\cdot}02$ cents/°C core average temperature change.

This result was checked in a later experiment in which the reactor was re-balanced at delayed critical after a pulse while the core was 35°C above the starting temperature.

Table 6: *Calculated Doppler reactivity coefficient*

Temperature change, °K		Reactivity change	
From	To	$\frac{\Delta k}{k}$	$10^6 \frac{\Delta k}{k} / °C$
300	600	−0·00174	−5·8
600	900	−0·00090	−3·1
300	$\{^{238}U\ 600\}$ $\{^{235}U\ 300\}$	−0·00196	−6·6

The reactivity change due to this temperature difference was 4 ± 1 cents, i.e. a temperature coefficient of $0{\cdot}1 \pm 0{\cdot}025$ cents/°C.

Summary of the temperature coefficient data

The results of the various experiments and calculations which have been done to determine the overall temperature coefficient and its separate components are summarized as follows:

Source of information and type of coefficient determined	Temperature coefficient per 10^5 °C core-average temperature rise
Prompt temperature coefficient derived from sub-prompt transients	−1·0 to −1·1
Prompt temperature coefficient derived from the yield-reactivity relation in prompt critical transients	$-1{\cdot}08 \pm 0{\cdot}07$
Calculated prompt temperature coefficient due to fuel expansion and bowing	$-0{\cdot}70 \pm 0{\cdot}09$
Calculated prompt temperature coefficient due to the Doppler effect	$-0{\cdot}58 \pm 0{\cdot}09$
Measured temperature coefficient corresponding to slow heating rates	$-0{\cdot}73 \pm 0{\cdot}14$

MEASUREMENT OF THE NEUTRON LIFETIME AT DELAYED CRITICAL

The prompt neutron decay constant, α, was measured in the range 0·2–2 dollars sub-critical by the pulsed source method and in the range from critical to 0·9 dollars sub-critical by the Rossi-α method. The Rossi-α measurements were done in the standard core, but for the pulsed source measurements the core was re-arranged to accommodate the 14 MeV neutron source as shown in Fig. 10(*a*). The Rossi-α measurements were made with a pair of BF_3-filled proportional counters inserted in the pulse rod hole. The

Table 7: *Rossi and pulsed α prompt decay measurements*

Degree of criticality, cents	Type of measurement	Details of counters used	Decay constant × 10^{-4}/s at time t, μs		
			t = 20	t = 50	t = 80
Delayed critical	Rossi	Two BF_3 counters in pulse rod hole	3·99 ±0·04	3·56 ±0·06	3·10 ±0·10
−19·5	Pulsed	^{235}U and ^{238}U fission counters in core	4·89 ±0·04	4·38 ±0·06	3·59 ±0·10
−19·5	Pulsed	^{238}U fission counter in core with pulse rod inserted	5·15 ±0·02	4·60 ±0·06	3·91 ±0·14
−19·5	Pulsed	^{235}U fission counter in copper reflector	4·71 ±0·02	4·43 ±0·06	3·76 ±0·12

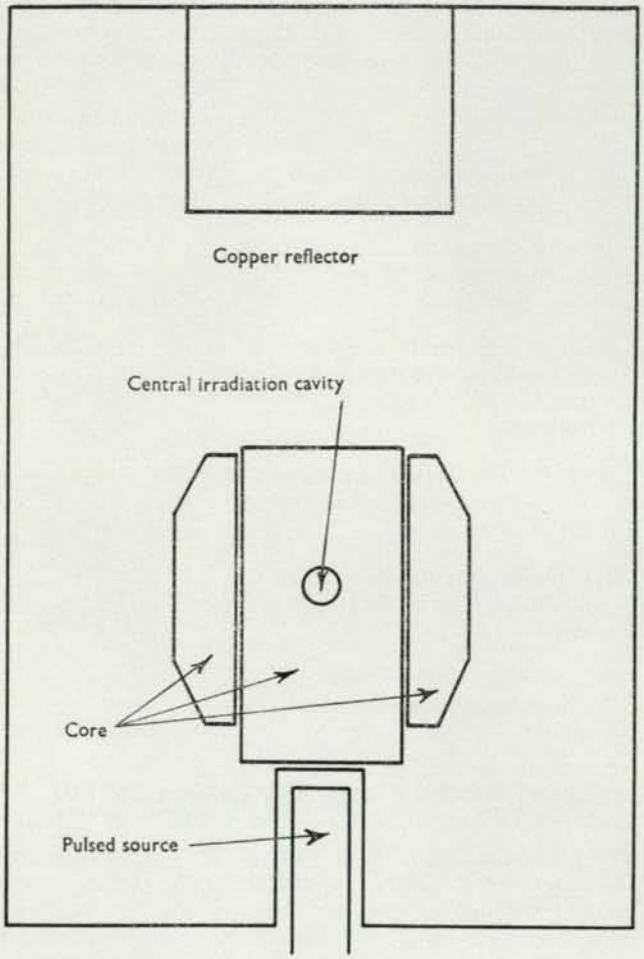

Fig 10(a) **Core arrangement for Rossi and pulsed alpha**

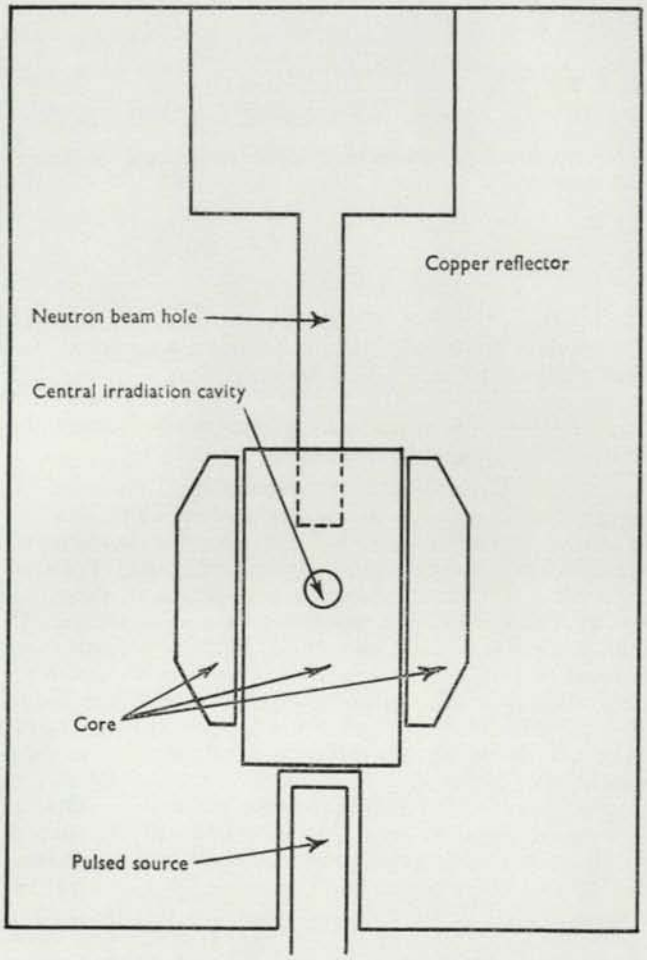

Fig 10(b) **Core arrangement for time-of-flight spectrometry**

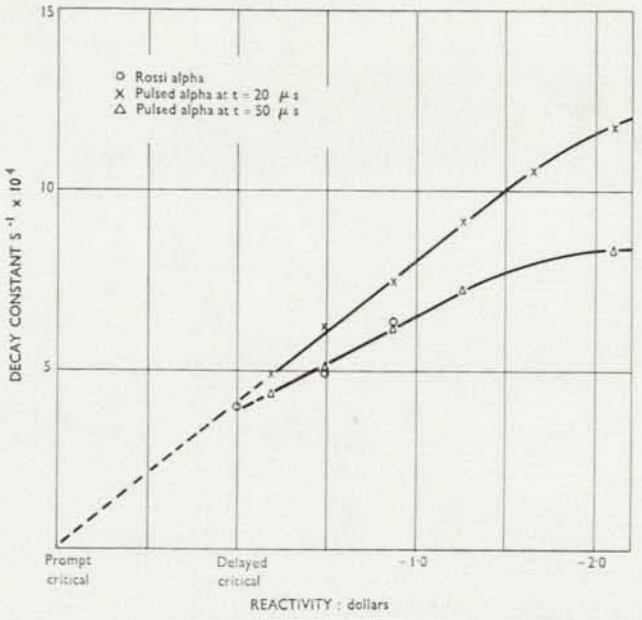

Fig 11 **Extrapolation of pulsed and Rossi alpha to delayed critical**

pulsed source measurements were made with 0·25 in. diameter fission counters, coated with ^{235}U and with ^{238}U, placed in holes in the core and in the reflector. In both methods the decay curves extended for a period of between 50 and 100 μs from the initiating event until obscured by background, but the curves obtained after subtraction of the background could not be fitted by single exponential functions because the apparent decay period increased steadily with time. For example, the values of the decay constant of the critical core obtained from the Rossi-α decay curve at 20, 50 and 80 μs after the initiating event are listed in Table 7. In each case the value has been averaged over a period of 50 μs.

According to the point-reactor kinetics model, α is directly proportional to the reactivity and has the values β/τ and 0 respectively at delayed and prompt critical. The pulsed source α values measured at 20 μs after the pulse are directly proportional to reactivity and, when extrapolated to delayed critical, yield a value of β/τ which is in close agreement with the value at delayed critical measured by the Rossi-α method. The pulsed source α values measured at 50 and 80 μs after the pulse are not directly proportional to reactivity, but the value of β/τ obtained by extrapolating the 50 μs values is in good agreement with the other results. The extrapolation is shown in Fig. 11.

On the basis of these results the reactor-averaged value of β/τ is $4{\cdot}05 \pm 0{\cdot}05 \times 10^4$/s and using $\beta = 0{\cdot}728\%$ the mean lifetime τ is $0{\cdot}180 \pm 0{\cdot}022$ μs.

There is some evidence from measurements in the reflector with detectors sensitive to low-energy neutrons that the low energy flux in the reflector decays more slowly than that in the core.

The Rossi-α results give a direct measure of the delayed neutron fraction by using the expression[10]

$$A = \frac{E\{\overline{\nu(\nu-1)}\}k^2\alpha}{2\bar{\nu}^2\beta^2}$$

where A is the amplitude of the Rossi-α signal at time zero and E is the counter efficiency (counts per reactor fission). E was determined by comparing the count rates in the BF_3 counters with the measured total fission rate in the core and, using $\alpha = 4{\cdot}05 \times 10^4$/s the value of β derived from the Rossi-α amplitude is $0{\cdot}63 \pm 0{\cdot}04\%$.

NEUTRON ENERGY SPECTRUM

The neutron energy spectrum in the reactor core was measured over the energy range 140 eV to 100 keV by the time-of-flight method,[11] using the arrangement shown in Fig. 10(*b*). The neutron beam emerged from a surface at the core mid-radius via a 2 in. square hole and the pulsed source of 14 MeV neutrons was located close to the core surface on the opposite side of the core. The measured spectrum is shown in Fig. 12. The flux is normalized to a standard reactor power (a fission rate of unity per gram of ^{235}U at the point of measurement) by a measurement of the flux at 337 eV using the manganese foil sandwich method.[11]

The neutron energy spectrum in the core was calculated by transport theory and by diffusion theory methods. The transport theory calculations used the S_n approximation in a two-dimensional code, TURTLE,[12] with the FD2 data set[13] but without allowance for heterogeneity effects in determining the resonance self-shielding. The diffusion theory calculations used the one-dimensional code SWAN[3] with group cross sections in which heterogeneous self-shielding of resonances had been incorporated as for the Doppler coefficient calculations. The calculated spectra are shown in Fig. 12, normalized to the standard reactor power scale. The diffusion theory spectrum agrees with the measured spectrum within the experimental error (estimated at 10%) but the transport theory spectrum contains too little flux in the low energy region. The difference between the calculations is probably due to the inadequate treatment of resonance self-shielding in the transport theory data.

YIELD AND RADIATION DOSES IN THE FULL SIZE PULSE

The maximum yield in a pulse in VIPER 1 is determined by the maximum temperature (275°C) to which the epoxy resin components can be heated without significant deterioration. Assuming a starting temperature of 25°C, the peak temperature at the core thermocouple then is 237°C, the fuel average temperature is 332°C and the peak fuel temperature is 456°C.

The total fission yield corresponding to a full size pulse has been measured in three ways, two based on fission measurements and one based on the total heat generated.

In the first method the pulse yield for a given temperature rise was measured in terms of the activation of a magnox disc, as described in Part 1, and the magnox disc activation was related to fission yield in a steady-power calibration experiment. The power integral in the steady-power experiment was measured by a monitor counter which was itself

Table 8: *Measurements of fission yield in a full-size pulse*

Method	Fission yield per °C temperature rise of core thermocouple, fissions $\times 10^{-15}$	Fission yield in full size pulse, fissions $\times 10^{-17}$
Steady power fission counter measurement scaled to full size pulse by magnox activation detector	$2{\cdot}06 \pm 0{\cdot}20$	$4{\cdot}36 \pm 0{\cdot}42$
Radiochemical fission yield in small sample irradiated during a pulse	$1{\cdot}71 \pm 0{\cdot}19$	$3{\cdot}63 \pm 0{\cdot}41$
Measurement of total heat energy generated	$1{\cdot}57 \pm 0{\cdot}13$	$3{\cdot}33 \pm 0{\cdot}28$
Mean fission yield	$1{\cdot}73 \pm 0{\cdot}16$	$3{\cdot}66 \pm 0{\cdot}35$

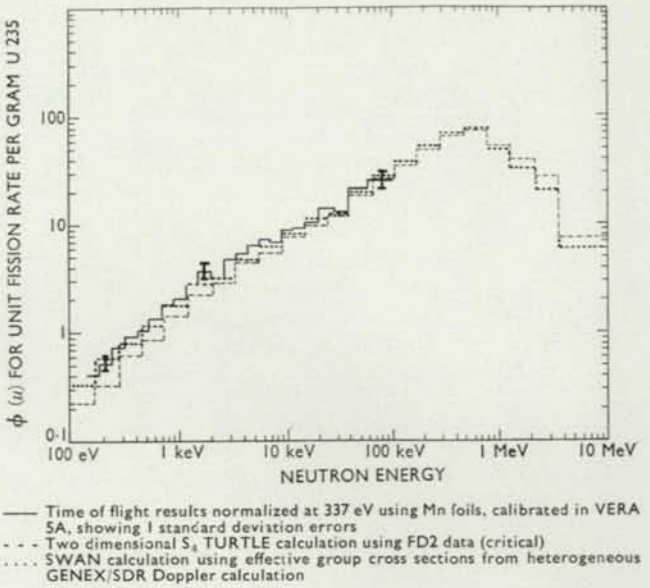

Fig 12 **VIPER spectrum by time of flight**

related to a standard fission counter and thus to the total fission rate in the core.

In the second method a ^{235}U foil was irradiated in the core during a pulse and the number of fissions was measured by radiochemical analysis of the fission products. From the fission distribution measurements the total number of core fissions was thus determined and this was related to the measured temperature rise in the pulse.

In the third method the temperature rise for a full size pulse was measured at the core fuel thermocouple and the corresponding temperature rise as a function of position throughout the core was obtained from the measured fission distribution. The total heat generated was then calculated from the specific heat of the fuel. The equivalent fission yield was calculated on the basis that each fission deposits 166 ± 6 MeV energy in the fuel. This allows for some loss of γ-ray energy and some heat transfer to the fuel-can during the plateau and neglects the contribution from radioactive decay of fission products. The results of the fission yield measurements are shown in Table 8. All three methods depend on the fission distribution to determine the total core yield from a measurement at a single point, and a standard error of 6% is attributed to each result to allow for the uncertainty in this measurement. In the first method there are errors due to the absolute calibration of the fission counters (3%) and due to the necessity for intercalibration of detectors. In the second method there is a standard error of 5% associated with the measurement of fission yield by radiochemistry. In the third method there are uncertainties in the specific heat of the fuel and in the ratio of the specific heats of the fuel and the thermocouple uranium cylinder. Since the standard error in the third method is smaller than in the other two, and since it is also a different kind of measurement we have given it double weight in calculating the mean yield. The best value for the full size pulse yield is therefore $(3{\cdot}66 \pm 0{\cdot}35) \times 10^{17}$ fissions.

The integrated doses of neutron and gamma radiation at various positions in and around the reactor have been determined partly by measurement and partly by calculation. The gamma radiation was measured by thermoluminescent dosimeters using lithium fluoride with lithium enriched to 99·9% in the^{7} Li isotope. The neutron radiation was measured in terms of the total counts recorded in ^{235}U and ^{238}U fission chambers. The integrated neutron flux was also calculated at various positions. The radiation doses for the full size pulse yield of $3{\cdot}66 \times 10^{17}$ fissions are listed in Table 9 and the calculated neutron spectrum within the core is shown in Fig. 12.

SUMMARY AND DISCUSSION OF THE PHYSICS EXPERIMENTS AND THEIR ANALYSIS

The following points emerge from the comparison of calculation and experiment.

(a) The kinetic behaviour of the reactor up to prompt critical can be accurately represented by a point reactor model (the inhour equation) provided that some fairly small adjustments are made to the effective relative delayed neutron yields particularly affecting the short-lived groups.

(b) The kinetic behaviour of the reactor in the region beyond prompt critical shows the linear relation between reciprocal initial period and reactivity predicted by the point reactor model. The derived value of β/τ is 7% larger than the value obtained in Rossi-α and pulsed source experiments but the calculated value is 1·3 times the average of the two experimental figures.

(c) The observed pulse shape in super prompt pulses differs from the shape predicted by the point reactor model with temperature-independent coefficients. The ratio of pulse width to initial period is 20% larger than the value of 3·52 predicted by the simple theory.

(d) The sum of the calculated expansion and Doppler coefficients is $(1{\cdot}27 \pm 0{\cdot}15) \times 10^{-5}/°C$ compared with the total coefficient of $(1{\cdot}08 \pm 0{\cdot}07) \times 10^{-5}/°C$ measured in the super-prompt and sub-prompt transients.

Table 9: *Radiation doses produced in a full size VIPER pulse*

Location	γ radiation dose, r	Integrated neutron flux, n/cm^2	Fissions per gram of ^{235}U	Fissions per gram of ^{238}U
Central irradiation cavity (bottom surface)	$1\,3\times10^5$	1×10^{15}	$4{\cdot}5\times10^{12}$	$2{\cdot}5\times10^{11}$
Core–reflector interface (on reactor mid-plane)	5×10^4	$0{\cdot}9\times10^{15}$	4×10^{12}	$1{\cdot}6\times10^{11}$
Large irradiation cavity ($13\times13\times11$ in. cavity, with boron plate on core surface)				
(a) inner face	$3{\cdot}5\times10^4$	5×10^{14}	$2{\cdot}2\times10^{12}$	7×10^{10}
(b) centre	$1{\cdot}1\times10^4$	—	$1{\cdot}9\times10^{12}$	—
(c) outer face	8×10^3	—	$1{\cdot}7\times10^{12}$	—
North west reflector hole (on reactor mid-plane)	6×10^3	—	1×10^{12}	4×10^9
Outer surface of reflector (top surface, near reactor centre line)	$1{\cdot}1\times10^3$	3×10^{13}	7×10^{11}	$3{\cdot}5\times10^9$

Points (b), (c) and (d) call for some discussion. Considering first the neutron lifetime, the most striking aspect is the discrepancy between the calculated value and the value derived from the two experiments. Similar discrepancies between Rossi-α measurements and point-reactor calculations in reflected fast reactors have been reported before[10, 14] and it has been noted that the point-reactor model is particularly inappropriate when the reflector is a neutron moderator rather than a neutron absorber because the decay of the low energy neutron flux is then slower in the core than in the reflector. In that case the observed value of β/τ would be time-dependent and would vary according to the relative contributions of the core and reflector fluxes at the detector position, whereas the steady-state lifetime calculation would give a time-independent space-averaged value. However this effect seems unlikely to explain the 30% discrepancy because the reflector effects are most marked when the reactor is sub-critical and there is close agreement between the delayed critical values and those obtained from super-prompt-critical transients. The spectrum measurements show that the discrepancy cannot be explained in terms of an error in the calculated core spectrum because, when heterogeneity effects are taken into account, this agrees closely with the measured spectrum (and the same calculation was used to obtain the lifetime). However, since about half the contribution to the calculated average lifetime is due to reflector neutrons, the discrepancy could be explained if the relative number of low energy neutrons in the reflector was appreciably larger than predicted by calculation. The reflector neutron energy spectrum has not been measured but the ratio of calculated to measured ^{235}U fission rate in the reflector at a radius of 7 cm and a height of 25 cm above the core mid-plane is 0·9, indicating that the reflector spectrum is probably softer than calculated. The 7% difference between the β/τ values from the two experiments is small compared with the discrepancy between experiment and theory but it does appear to be significant compared with the 1·2% standard errors of the measurements. The difference may be partially accounted for by the fact that the super-prompt measurement is made with the pulse rod in, thus giving greater weight to the (shorter) core lifetime, but it could also be attributed to the over-simplification of the one-point model used in deriving the results.

The discrepancy between observed and predicted pulse shapes can be considerably reduced by introducing a more accurate representation of the temperature coefficient. The simple theory assumes that the temperature coefficient is independent of temperature and, although this is a good approximation for the fuel expansion component, it is not an accurate representation of the Doppler component. The Doppler coefficient, which amounts to about half the total, is approximately inversely proportional to the absolute temperature to the power 3/2, and when this is allowed for in the calculation the predicted pulse shape is appreciably changed. This effect has been studied by a series of calculations of pulse shape and yield using the reactor kinetics code RKSF[4] and the prompt kinetics code DOPPELAS.[15] Attributing a temperature dependence of $T^{-3/2}$ to half the total temperature coefficient, the following results are obtained:

(a) The pulse is broader than predicted for a temperature-independent temperature coefficient; the ratio of pulse width to initial period is increased from 3·52 to 3·9 for all pulse sizes.

(b) The relationship between yield and reactivity insertion becomes slightly non-linear.

(c) The plateau power level becomes dependent on pulse yield: for small pulses corresponding to 1 or 2 cent reactivity increments the predicted level is 0·45°C/ms and for full size pulses, corresponding to 20 cent increments, it is 0·75°C/ms. The calculations with a temperature-independent temperature coefficient predict a plateau level of 0·7°C/ms independent of pulse yield.

The introduction of temperature-dependence into the calculations clearly improves the agreement between experimental and calculated pulse widths. However, the experimental results are not accurate enough to show the slight non-linearity of the yield-reactivity relation or the variation of plateau level with pulse yield.

Since the discrepancy between the calculated and measured temperature coefficients is less than the sum of the standard errors it is reasonable to conclude that the calculated expansion and Doppler coefficients are correct within the accuracies claimed. However it is worth noting that the measured temperature coefficient and the calculated expansion coefficient both depend on the accuracy of the delayed neutron fraction whereas the calculated Doppler coefficient does not. Therefore an error in the delayed neutron fraction would change the calculated and measured total coefficients in different proportions. Also the Doppler calculation refers to a uniformly heated core and does not include the effects of the temperature distribution.

ACKNOWLEDGEMENTS

It is a pleasure to acknowledge the contributions made by Dr R. L. Long (on leave of absence from the University of New Mexico) particularly to the commissioning phase, by Mr W. J. Paterson for the time-of-flight spectrometry and neutron lifetime measurements, and by the members of the VIPER reactor staff.

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Occasional equipment failures occur inside gas-cooled reactors necessitating an internal inspection and possible recovery operation. Physical entry into the reactor is extremely difficult and remote handling and viewing equipment has been developed. Television cameras are available for inspection and recovery operations in fuel channels. Flash photography has been used extensively in-pile and special cameras have been developed for this purpose. The requirements for such equipment are discussed, together with a number of recovery operations undertaken at different nuclear power stations.

In-core viewing and remote handling problems of gas-cooled reactors

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INTRODUCTION

THE power reactors in Britain, which are owned and operated by the Central Electricity Generating Board, are intended to run continuously, with only biennial shut-downs for inspection and maintenance purposes. On-load fuelling is undertaken on all reactors. The routine replacement of control rod mechanisms at most stations is carried out at power, and there is a strong financial incentive to keep reactor shut-down periods to a minimum.

Equipment failures occasionally occur, making it necessary to inspect inside the reactor pressure vessel and perhaps carry out a recovery or repair operation. After the start of power operation of a gas-cooled magnox reactor, physical entry inside the reactor pressure vessel is extremely difficult, and would require a prolonged shut-down. Because of this, remote viewing and handling equipment is used to deal with these breakdowns. The design and development of this equipment and its use in dealing with some reactor faults are described in the Paper.

VIEWING EQUIPMENT

For a gas-cooled magnox type reactor the on-load conditions, i.e. gas temperatures in the range 200–410°C and fast neutron radiation levels of 2×10^{13} n/cm^2 s, raise severe problems for the introduction of viewing or other operating equipment into the reactor pressure vessel with the reactor operating at power. There is an obvious financial incentive to undertake any simple inspections inside the pressure vessel with the reactor on-load, and developments have been undertaken to produce an on-load television camera to withstand these conditions. The results achieved were very encouraging although the ultimate limit appeared to be in the generation of Cerenkov light in the face plate of the vidicon tube. However, owing to the very high cost of development and the limited usefulness of such a camera, the development has been abandoned.

As a result, all inspection and other operations are carried out with the reactors shut down. The conditions to be withstood are then: gas temperatures as high as 200°C and gamma radiation levels 24 hours after shutdown up to 10^6 rad/h with an atmosphere of carbon dioxide or air. In spite of these less arduous conditions, problems still arise in cooling the equipment and making it able to withstand gamma radiation.

To view an object inside the reactor pressure vessel, the instrument must have an adequate and clear field of view and be capable of being directed and focused, and adequate illumination must be provided. These requirements are discussed briefly below.

Field of view

The required field of view is dependent upon the object being viewed. For example, viewing the inside of the top dome of the reactor pressure vessel would probably require a narrower angle of vision and a longer focal length than, say, viewing inside a fuel channel. Generally access is restricted to that obtainable through the pressure vessel standpipes from pile cap level. Owing to the rigidity of lifting and directional attachments and possible cooling hoses, the instrument will hang approximately in line with the standpipe. For entry into a fuel channel a special chute may be required to direct the instrument into position. Stations vary considerably in the permissible fields of view from feasible instrument positions because of differences in the detailed design of the reactor structure and pressure vessel.

Optical path

One of the problems likely to be encountered even in a shut-down reactor is the effect of 'shimmering', which is caused by changes in distortion as density gradients in the

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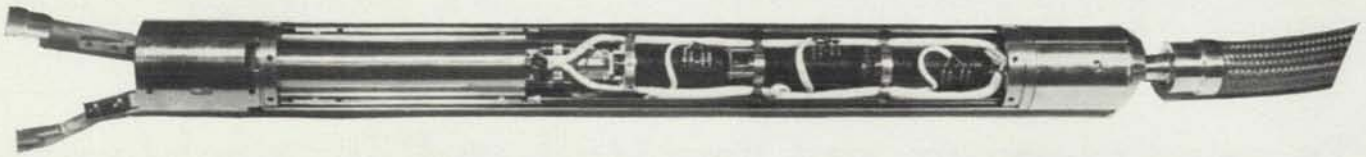

Fig 1 **In-pile television camera with manipulators**

gas vary with time. An 'instantaneous' picture, for example a photograph taken by a flash bulb, may give a better image, but still would not be free of distortion. Possible improvements may arise from selecting alternative viewing positions or perhaps controlling the gas conditions in the viewing area. At one time it was thought that the presence of entrained dust would give rise to viewing difficulties, but this has been found to be negligible.

Orientation

An operator using a viewing instrument can interpret the image presented to him without necessarily knowing the orientation of the instrument. He achieves this by scanning the area required, pointing the instrument and focusing onto the detail required. This applies to a continuous or intermittent picture. However, once the instrument is moved, in order to enable the field of view to be identified it is necessary to provide means of indicating the position of the objective and its direction with respect to a reference line outside the reactor. In some difficult viewing situations, more than one sighting may be required to give an intersection from other viewing positions. Within the reactor pressure vessel it may be possible to place markers on fixed parts of the reactor structure to identify particular positions. In addition, suspended lights can frequently assist in giving the bearing of the instrument. When the viewing camera has a stationary body with a rotating head, a scale attached to the body and in the field of view of the instrument can be used to give a bearing.

Illumination

Because of the black background inside the reactor pressure vessel, a strong source of illumination is required. For viewing close to the instrument a built-in light source is necessary and for continuous viewing this raises problems of cooling and space requirements. In the case of television cameras designed to go down a fuel channel, the lighting arrangement must be compact, and additional cooling is needed. For more general viewing, such as the inside of the top dome of the pressure vessel, separate light sources can be employed. These can be introduced via other standpipes and the best positions can be selected to reduce shadows and provide reference objects for orientation.

As in the case of exposure time, scan time of a television tube is related to the intensity of the illumination, so attempts to cut down shimmer by shortening the time of a single scan will require corresponding increases in the intensity of illumination.

Photographs may be taken using flash bulbs or electronic flash and the intensity of illumination is a matter of trial and error depending upon the object, general surroundings, film speed, lens and aperture.

Method of viewing

Several techniques are available for presenting an image remote from the objective: direct viewing via a form of telescope or fibre optical instrument, television or photography. Direct viewing with a telescope or fibre optical instrument is limited in the field of view available and the flexibility of operation. As a result, more emphasis has been placed on the development of suitable television cameras and photographic cameras for use within the reactor pressure vessel.

As has been mentioned earlier, introducing a television camera into the reactor structure raises problems in cooling and the ability of the unit to withstand gamma radiation. The camera must also be capable of being erected and focused onto the required object. Similar problems exist for a still camera using flash for illumination. There is also the danger of fogging the film due to gamma radiation. Experiments have shown that slow-speed films are more resistant to fogging from radiation.

STATION VIEWING EQUIPMENT

The equipment used at the operating nuclear stations for viewing inside the reactor pressure vessel can be described under the general headings of television equipment and photographic cameras.

Television equipment

Each CEGB nuclear power station is equipped with closed circuit television equipment which is used for viewing inside the reactor under shut-down conditions. Considerable developments have taken place in the design of television cameras, and the present-day camera is capable of withstanding gamma radiation up to 10^7 rad/h with alpha contamination present. It is built to withstand temperatures up to 300°C and gas pressures up to 350 lbf/in.^{2} The camera is sufficiently small to pass down a channel of internal diameter 3·5–4 in.

A typical television camera is shown in Fig. 1; this was developed by Messrs Pye of Cambridge. The outer shell

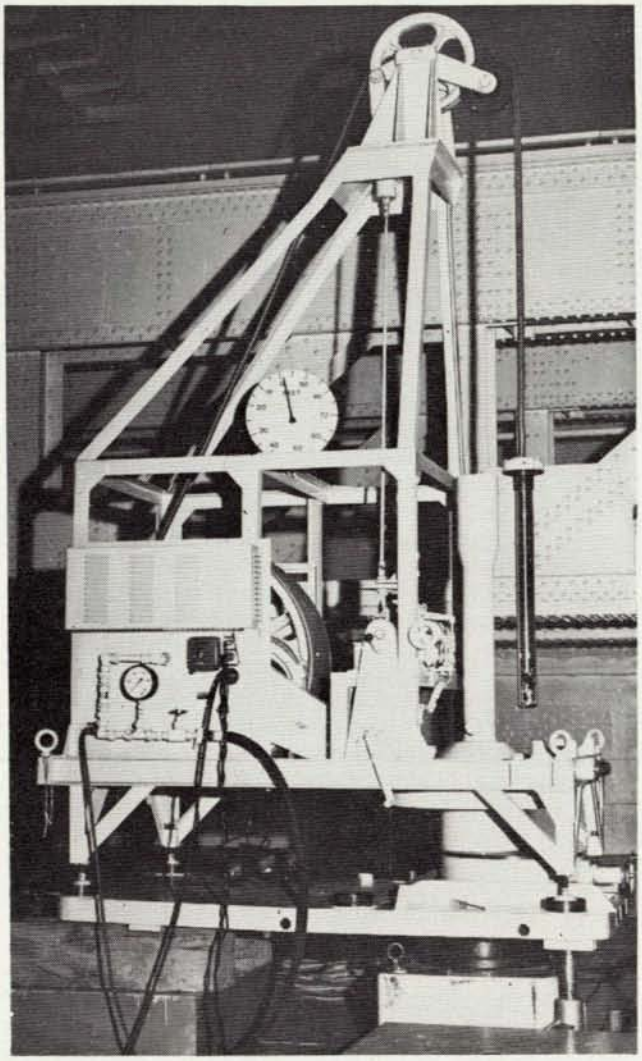

Fig 2 (left) **Bradwell in-pile television rig**

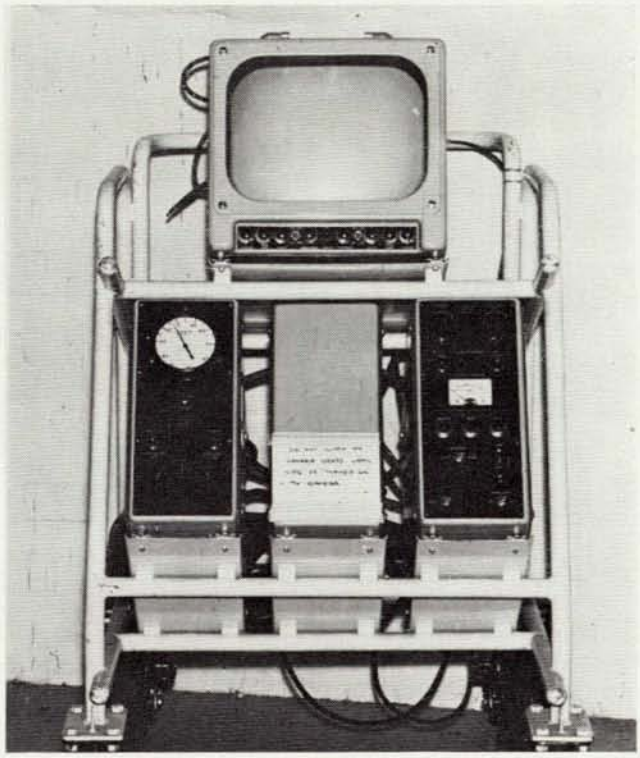

Fig 3 **Bradwell in-pile television monitor**

of the camera is of stainless steel with a front window of stabilized glass. Focusing is achieved by moving the camera capsule—vidicon tube, deflexion and focusing coils and video amplifier—relative to the fixed lens. The lighting for the camera is self-contained in the form of two tungsten iodine lamps, arranged one on either side of the lens. Various small grabs and manipulators can be fitted to the front end of the camera as illustrated and these are operated by a small electric motor. A further electric motor operates the right-angle viewing mirror. All-round viewing is provided, enabling fuel channel walls to be inspected and viewing and recovery operations may be undertaken in line with the camera. The camera is attached to a thermally insulated stainless steel flexible hose which contains the camera cable and the cooling gas supply to the camera.

The television camera does not become radioactive in itself under reactor shut-down conditions but can become contaminated with radioactive material. For subsequent handling and maintenance, decontamination is important to reduce the ingestion hazard.

Recent experience has shown that a video tape recorder can be a most useful adjunct to the in-pile television system. By this means an operation can be recorded on tape so that it can be played back later on a video monitor. This record can be invaluable in a recovery operation for checking the sequence of events taking place or preserved for later reference in a subsequent operation of the same type. The in-pile television rigs used at two power stations are described below.

Bradwell in-pile television rig

The television viewing equipment at Bradwell is a self-contained assembly consisting of a camera positioning rig, motor and control unit (Fig. 2). The camera rig, which can only be used when the reactor is off-load and depressurized, contains a winch unit and winding gear for lowering the camera down a standpipe into the reactor. An indicator on the winch unit gives the position of the camera during loading and unloading. The controls to the camera and winch unit and the television monitor are mounted on a separate trolley (Fig. 3) which can be positioned anywhere on the pile cap. The camera rig can be positioned and used

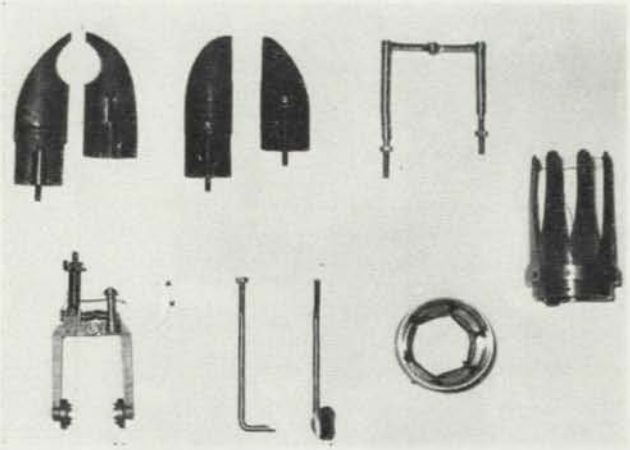

Fig 4 **Grabs for in-pile television camera**

independently of the fuelling machine, leaving the fuelling machine free for other in-core handling work, which can be carried out simultaneously.

The television camera supplied with this unit is of diameter 3·5 in., and can be used inside the fuel channels. It is designed for use at reactor pressures slightly greater than atmospheric, and in temperatures up to 200°C. Carbon dioxide coolant maintains the camera internal temperatures below about 65°C. The camera can view along, and at right angles to, its axis and includes its own source of illumination provided by four 25 W lamps. This is sufficient for the camera to view objects up to 3 ft away when operated within a black graphite fuel channel. Various attachments are provided with the camera, including a manipulator head which, when equipped with suitable grabs, may be used for extracting objects from within the reactor. A right-angle viewing head can be used which permits a view straight ahead of the camera or at right angles to it. Fig. 4 illustrates some of the types of grabs which may be used with the television camera manipulator head.

Trawsfynydd in-pile television equipment

Trawsfynydd has a variety of television equipment which can be used for in-core work. The station has an off-load in-pile television camera rig similar to that at Bradwell, but in addition the fuelling machine has a television camera and hoist which can be used under shut-down conditions with the reactor pressurized, and at temperatures up to 200°C. This camera carries its own lighting source and is small enough to pass down a fuel channel. In addition to this, television is used with special trailing lead thermocouple chutes. These chutes have been designed and manufactured for charging fuel elements into the reactor with trailing lead thermocouples attached. Television monitoring of these charging operations is considered necessary to ensure the integrity of the thermocouple assembly as it is loaded. The chute consists of a 6 in. nominal bore tube with a short extending arm so that a fuel element can be directed into the channel to which it is to be charged. A television camera and lighting unit is fitted to the bottom of the main tube which allows an examination of the assembly to be made as it is lowered into position. The assembly can be operated off-load at atmospheric pressure, and at temperatures less than 200°C.

Photographic cameras and techniques

Excellent photographs have been taken inside CEGB reactors under shut-down conditions using equipment varying from a simple Brownie camera to much more sophisticated cameras specially developed for the purpose. The detail revealed by in-pile photography is generally better than that obtained by using television. Flash attachments adjacent to the camera lens are used, so that parallel light is cast in front of the camera, thus avoiding the production of misleading shadows. However, for specialized applications there is no reason why multiple flash units should not be used at different positions in the reactor.

At Bradwell Power Station a Kodak Brownie Flash 20 camera has been used, mounted in a suitable framework suspended from a length of conduit, for taking in-pile photographs. The shutter was operated by a length of wire threaded through the conduit. Normal flash bulbs were employed, and to guard against a burst bulb the flash bulb and reflector were covered by a piece of stainless steel wire mesh. Black and white and also colour photographs have been taken using this type of equipment. Owing to the very limited range of colours inside the reactor, the colour photographs were found to be of little additional value.

At Trawsfynydd Power Station a Robot 35 mm camera has been used in a similar way to take in-pile photographs. This has the advantage of a small motor, enabling a sequence of up to 25 photographs to be taken without the camera having to be reset.

Experience at the various stations showed that small commercially available cameras were limited in their application owing to possible damage from excessive temperature and the time taken to develop and print a film. An investigation was undertaken at the CEGB Berkeley Nuclear Laboratories on the use of Polaroid Land self-developing film, and the development of a suitable camera. Experiments showed that the slower speed films were more resistant to radiation, and the maximum permissible film temperature was about 80°C. Both flash bulbs and electronic flash were considered, but flash bulbs were eventually chosen owing to the greater integrated illumination available and the simplicity of the associated equipment.

The camera and its handling rig were designed for flexibility in use inside the reactor, permitting the camera to be directed by the operator. The camera (Fig. 5) is of light alloy of welded construction, and fits inside a sheet steel pod. A standard Schneider Angulon wide-angle lens is fitted, together with a Polaroid Land film back. The camera focus is adjusted by a focus ring which is then locked. The shutter lever is operated by a small pneumatic jack, actuated by a push button on the operating assembly. The pod also contains the flash unit and battery. The flash bulb is mounted alongside a gauze-covered window in the side of the pod. The pod containing the camera is suspended from an operating arm together with a counterbalance weight (Fig. 6), and the whole assembly is lowered down a standpipe. From the pile cap the operator can orient and

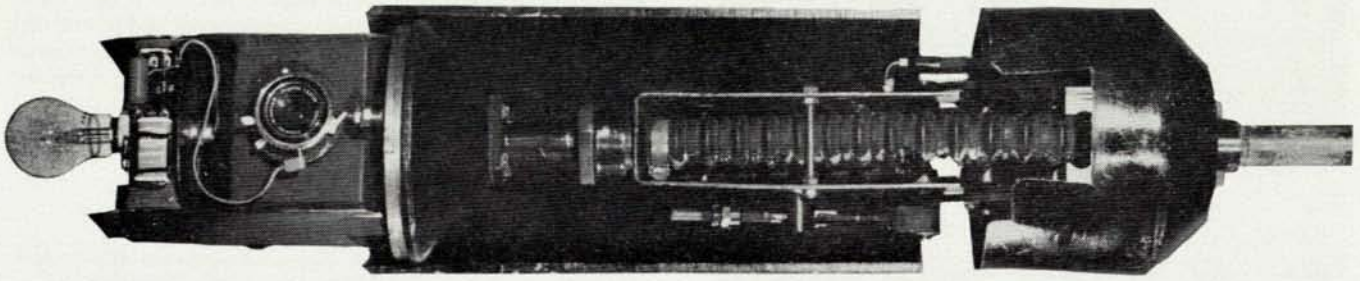

Fig 5 **BNL flash camera**

Normal mode of use of the BNL camera is vertical.

angle the camera as required within the reactor to take a photograph.

In the processing of the exposed film the developing times are generally kept as standard but the printing times and quality of the paper can be varied. Surprising variations in the quality of the prints can be produced by using different qualities of paper, and it is important that for a given negative the appropriate type of paper is used to make the print. The printing technique must ensure that the important part of the photograph is revealed with the maximum amount of detail, if necessary at the expense of the remaining areas of the photograph. Depth of focus down fuel channels and across the charge pans can be achieved by using very high intensity flash, but this tends to burn out objects closer to the camera.

Three-dimensional photography has been carried out at Trawsfynydd Power Station with the object of obtaining a contour map of the charge pan area showing the heights above and below a certain datum level. The technique was to take photographs simultaneously from two cameras mounted vertically about 5 ft above the charge pan and 1 ft apart, so that the photographs overlap in the required area. Analysis of the results is carried out by specialists in the technique using a special viewer and the results are presented in the form of a contour map of the area showing heights above or below the datum to an accuracy of about $\pm 0{\cdot}20$ in. Accuracies of about $\pm 0{\cdot}01$ in. can be obtained if the optical axes of the two cameras are known accurately.

RECOVERY TOOLS

Besides the equipment required for viewing inside a reactor, a variety of specialized tools and grabs is required for recovery and repair operations. Each station is provided with a number of recovery grabs for the more predictable operations such as recovering damaged fuel elements. A range of typical grabs for attachment to a television camera is illustrated in Fig. 4. Experience during recovery operations has shown that improvised grabs and tools made at the station can be just as valuable as their commercial counterparts.

In recovery exercises, care must be taken to ensure that the situation is not made worse by the failure of recovery equipment inside the reactor. Recovery equipment is usually designed so that it can be collapsed and pulled through a standpipe in the event of a failure. Where any crushing operations are to be carried out, screw jack arrangements are designed so that no resultant external force acts on the item which may cause buckling or bending. Wherever possible during these operations, terylene ropes are used in preference to wire cables because in the event of an entanglement they may be cut easily.

Grinding operations were carried out by remote control in the reactor at Trawsfynydd when the failure of a fuel chute occurred. Grinding was chosen in preference to any other machining operation to help in dismantling the fuel

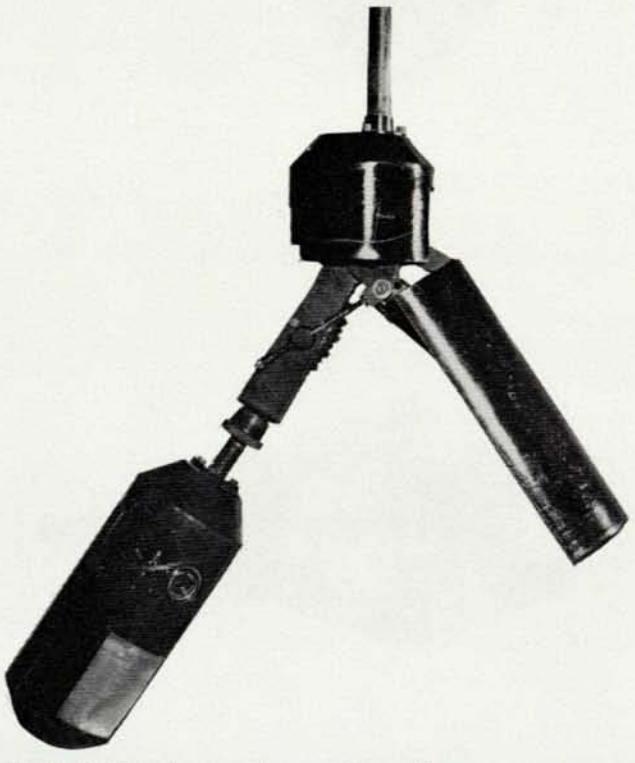

Fig 6 **BNL flash camera in operating position**

Fig 7 **Flux scanning tube seen in television monitor**

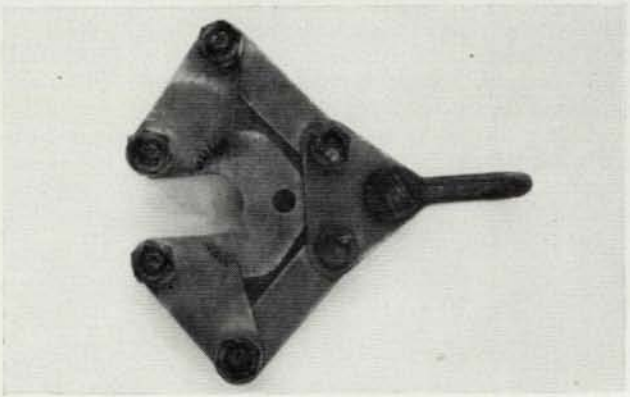

Fig 8 **Mechanical grab—jaws open**

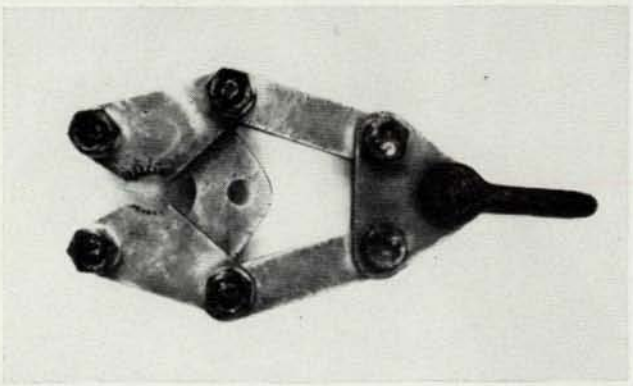

Fig 9 **Mechanical grab—jaws closed**

chute because the cutting forces involved were lowest. A general principle of applying loads inside the reactor is to ensure that no uncontrollable strain energy is present, the release of which could cause entanglement of cables or further damage. Care must be taken in the choice of materials for use inside the reactor to ensure compatibility with magnox fuel.

VIEWING AND RECOVERY OPERATIONS

To illustrate the use of viewing and recovery equipment, and to show what is possible by remote handling techniques, the case histories of a number of recovery exercises are described.

Recovery of flux scanning guide tube at Bradwell reactor 2

In the Bradwell reactors, each standpipe assembly locates with one channel in a group of 36 and provides access for trailing lead fuel element thermocouple cables. In addition each standpipe assembly has a flux scanning guide tube welded to its side. This tube is approximately 19 ft long, 1·25 in. in diameter, and is made of mild steel. The function of the guide tube is to provide access and a guide to a fuel channel for flux scanning wire and bob weight.

During routine on-load absorber handling operations on reactor 2 in May 1963 it was noted that a standpipe assembly was damaged and its flux scanning guide tube was missing. An inspection of the damaged assembly revealed that about 18 ft of the flux scanning guide tube was missing. The reactor was shut down, cooled and depressurized so that a search with the in-pile television camera could commence. Inspection of the charge pan areas revealed two lengths of flux scanning guide tube, one approximately 12 ft long and badly bent, and the other piece relatively straight, approximately 3 ft long. Fig. 7 is a photograph taken of the television monitor screen showing one of the lengths of tube lying across the charge pan in the reactor. The light source used was that normally carried in the camera head.

The longer piece of tube was picked up by a grab from a standpipe adjacent to that being used for the television camera. A maximum grab load of 1·5 tons was recorded as the tube section entered the bottom of the reactor standpipe. The grab mechanism was designed so that the greater the load on the grab the greater was the gripping force exerted by the grab jaws (Figs 8 and 9). This type of grab was essential for the recovery operation as the tube had to be bent into a U section before it could be withdrawn through the 8 in. diameter standpipe. The smaller piece of tube was recovered in a similar fashion through a different standpipe. An examination of the pieces of tube recovered showed that a further section of tube approximately 10 in. long was still missing and a further search was made. After about 80 channels had been inspected by television a piece of tube was located in a fuel channel. This piece of tube was retrieved using a special pair of jaws fitted to the television camera manipulator, and from this length it was apparent that no further tube sections were missing. The cause of the failure of the flux scanning guide tube was established as faulty welding of the tube to the standpipe assembly, and subsequently all the remaining standpipe assemblies were modified.

Dropping a fuel element axial stringer at Bradwell reactor 1

At Bradwell a fuel element axial stringer thermocouple assembly consists of seven elements, each of which has two thermocouples contained in a specially thickened fin. These assemblies are loaded into the reactor as a complete unit. During the loading operation for an axial string assembly in August 1963, it was discovered that the assembly had been dropped from the insertion tool into its fuel channel. The reactor was shut down and arrangements made to discharge this particular channel of fuel. The contact assembly and the top six fuel elements were recovered without any difficulty using the normal fuelling machine. When an attempt was made to remove the bottom fuel element it was found that the normal grab would not pick it up. It was concluded that the lifting head of this fuel element was probably damaged and special arrangements would be required to discharge it.

A special grab known as a 'ferret' (Fig. 10) was used to remove this element. This grab has serrated jaws which can move vertically inside fixed limbs and are designed to slide over the normal element lifting head. When this grab is in position the jaws grip the fins on the fuel element can; the head of the ferret grab is shaped so that it can be picked up by the normal fuel element grab. Fig. 11 shows the element being lifted by the ferret grab. Inspection of this fuel element showed that its lifting head was extensively damaged and that the debris from this was likely to be at the bottom of the fuel channel, and a search was made using the in-pile television. The first examination of the bottom of the fuel channel revealed pieces of the element's lifting head (Fig. 12, taken from television monitor screen). A large part of the lifting head was removed by careful manipulation of a grab attachment on the television camera. A further television inspection showed that the remaining pieces of lifting head were no longer visible, and it was concluded that these pieces must have fallen down the space between the support strut and the channel wall. The support strut at the bottom of the channel was removed and further inspection showed that the remaining pieces of lifting head were sitting on top of the channel gag strut. These pieces in turn fell into the bottom of the gag lantern assembly. The gag strut was then removed from the channel and these fragments were retrieved using a specially designed pair of long jaws on the television camera manipulator. The very small fragments of the lifting head were removed by using a probe filled with an epoxy resin adhesive. The probe was attached to the television camera manipulator (Figs 13 and 14). Recovery of these small fragments from the gag lantern assembly completed the recovery operation.

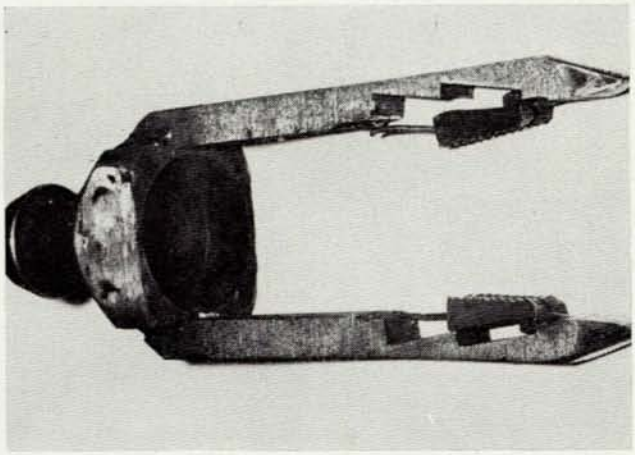

Fig 10 **Ferret grab**

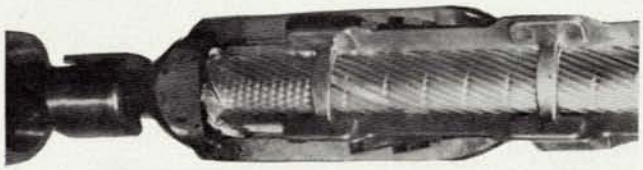

Fig 11 **Ferret grab located on fuel element**

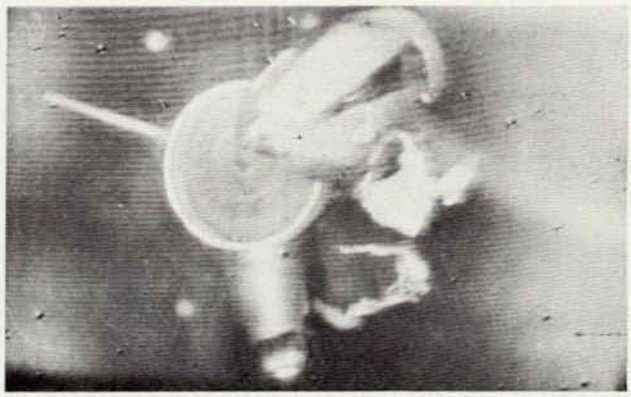

Fig 12 **Fuel element lifting head seen on television monitor**

Charge pan recovery operation at Trawsfynydd reactor 2

At Trawsfynydd, the charge pan consists of a heavy steel plate sitting on a number of cast steel charge pots, located over each fuel channel. The pots have rectangular bases and are supported on the graphite core with locating spigots for each fuel channel. The tops of the charge pots are circular, locating with holes in the charge pans. Fig. 15 has been taken in the reactor with two charge pans lifted off their locating pots, and shows the arrangement. Each charge pan is supported on three hemispherical seats. The pipework seen in the photograph is for BCD sampling on each fuel channel and the pipe runs across the reactor lie between the charge pans and the square bases of the charge pots.

During refuelling operations in July 1967, some difficulties were experienced in locating the charge chute in two fuel channels in a particular charge pan on reactor 2. A photographic survey was made of the offending charge pan with the reactor shut down using the Robot camera rig. Fig. 16

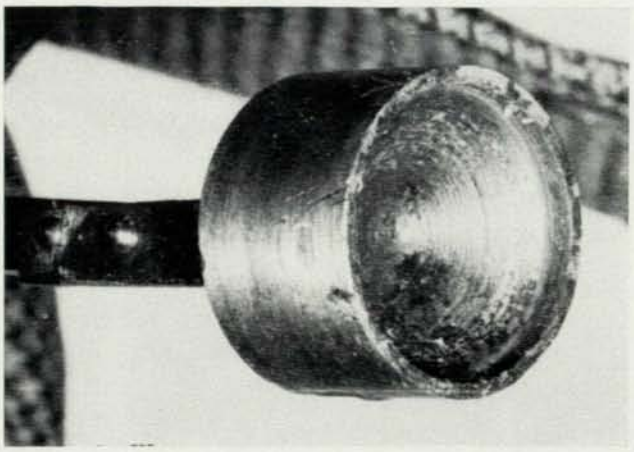

Fig 13 **Adhesive probe for recovering fragments**

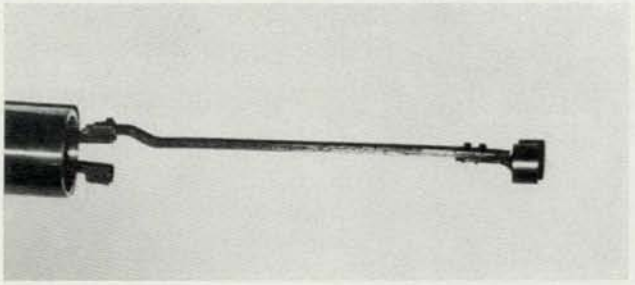

Fig 14 (left) **Probe attached to television camera**

Fig 15 **Trawsfynydd charge pan and charge pots**

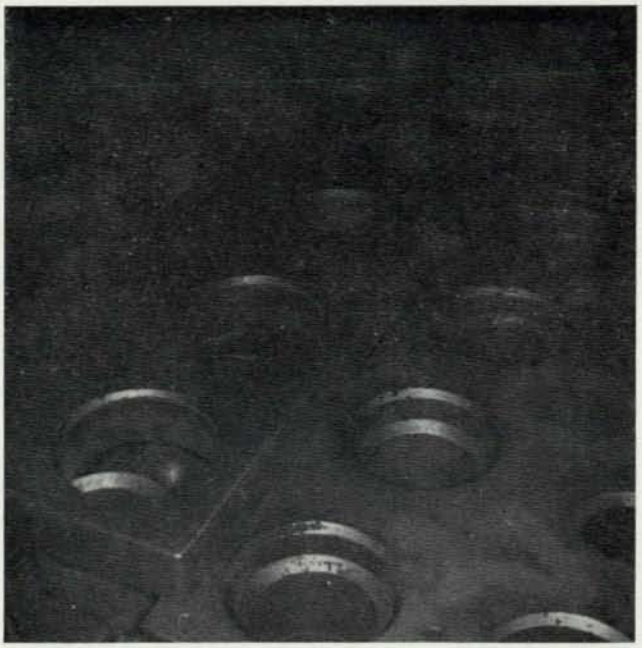

Fig 16 **Trawsfynydd charge pan lifted**

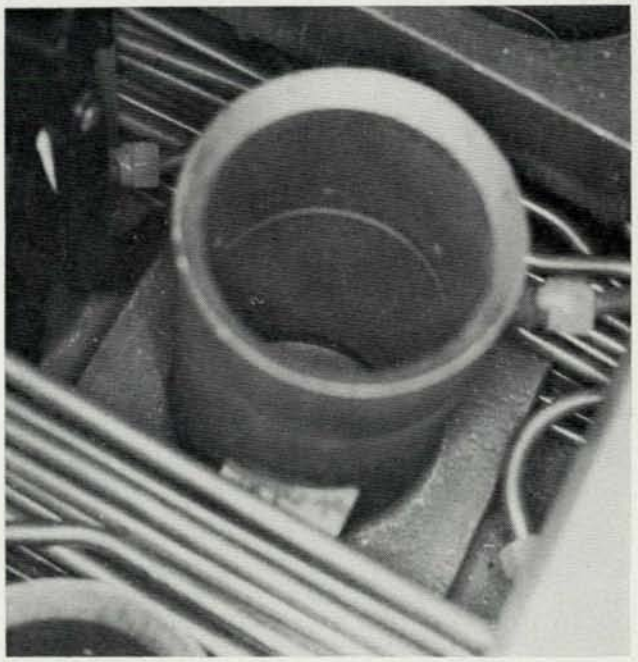

Fig 17 **BCD tubes trapped under charge pot**

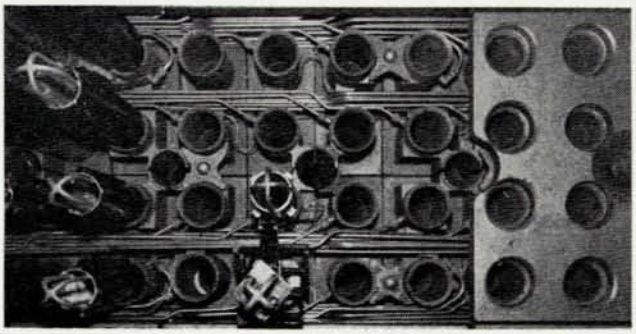

Fig 18 **Survey of reactor showing guides in position for relocating charge pan**

is a photograph taken during the survey and shows that the charge pan had been lifted clear of most of its locating charge pots and was supported on its four corners on adjacent charge pans. Lifting tackle was designed and manufactured which enabled the charge pan to be lifted clear of the reactor core, this operation being carried out remotely through the reactor standpipes. A full television and photographic survey of the area underneath this charge pan was then carried out. One of the photographs, Fig. 17, showed that one of the charge pots had been lifted clear of its graphite seat, and that some of the BCD pipes from adjacent channels were trapped beneath it. Further photographs showed that the BCD pipe from this particular charge pot was trapped between the corners of two adjacent charge pans. The quality of the in-pile photographs was such that on the original photographs the identification numbers painted on the charge pans during construction could still be seen after 2·5 years of reactor operation.

In order to replace the charge pan and charge pot correctly, it was first necessary to force the BCD pipes clear of the charge pot, replace the pot onto its original seat, and then replace the charge pan into position. A range of tools was designed and manufactured to carry out these operations. These included a tool to separate the BCD pipes and allow the displaced charge pot to be replaced over its channel. A further tool similar to a mandrel was used for locating other charge pots to their respective channels, holding them in position while further charge pans were lifted to give improved access. Fig. 18 shows the guides in position prior to replacing the charge pans.

Some of the photographs taken during the survey showed what appeared to be distortion of a BCD pipe, and it was decided that an integrity test on the pipework in the immediate area of the affected charge pan would be prudent. This test was carried out by blocking the BCD manifold in each charge pot with a balloon and reducing the pressure in the pipe concerned with a vacuum pump. If the vacuum condition was maintained, this was evidence that no pipe leaks existed. The final part of the integrity test was a flow check through the pipe to ensure that there was no blockage caused by tube distortion. None of the BCD pipes was found to be damaged.

At this stage a three-dimensional photographic survey of

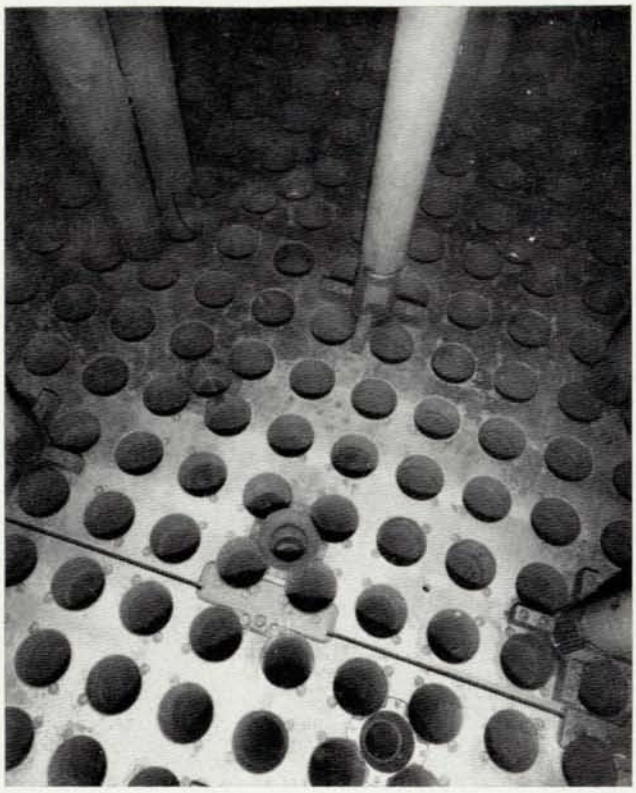

Fig 19 **Dungeness charge pans and standpipe assemblies**

the charge pan area was carried out to ensure that each charge pot was seated correctly. This survey was carried out for the CEGB by an organization which specializes in aerial survey techniques. Two cameras, specially mounted, photographed the charge pan area, and the resulting photographs were carefully analysed. Fig. 18 is one of the series of photographs taken at that time. A contour map of the charge pan area was made showing heights above or below a chosen datum level. This final survey confirmed that the charge pots were seated correctly, the charge pans were replaced and the reactor made ready to return to power. It should be emphasized that all these recovery operations were carried out from pile cap level at a distance of about 42 ft above the charge pans in the reactor, the only access being through the reactor standpipes.

Fuel element recovery operation at Dungeness reactor 1

During on-load refuelling operations on reactor 1 in March 1968, a fuel element was damaged whilst being discharged and the reactor was shut down for a recovery operation. A television survey was made inside the reactor to determine the exact location of the damaged fuel element. A recovery tool was then made up to retrieve the fuel element and, using television, the fuel element was successfully discharged. Afterwards a series of photographs was taken inside the reactor using the BNL flash camera to check the satisfactory completion of the operation, and Fig. 19 shows

one of these photographs. The charge pans and the vertical standpipe assemblies are clearly visible and fuel elements may be seen in some of the channels. Marks made on the charge pan during the operation may also be seen. On completion of the survey, the reactor was returned to power again.

CONCLUSIONS

The recovery and repair operations undertaken to date on the commercial reactors are most encouraging in that a wide variety of work has been successfully undertaken under conditions of high radiation levels and extremely limited access. Areas in the reactor which were thought to be in the category of 'fit and forget' are now accessible, and with careful planning and preparation a wide variety of work can be performed.

ACKNOWLEDGEMENTS

The Authors are indebted to the Central Electricity Generating Board for permission to publish this Paper. They also wish to thank the staffs at Bradwell, Dungeness and Trawsfynydd Power Stations and Berkeley Nuclear Laboratories for their assistance in preparing the Paper, and Messrs Pye TVT Limited and Mr R. Wright of APC Limited for providing data and photographs.

Previous measurements of a range of detailed reaction rates in SGHW lattices at ambient temperature have been extended to cover coolant temperatures up to 274°C. The measurements have been made in a single-channel, hot pressurized loop in the zero energy reactor JUNO, and have included the study of UO_2 fuel and of PuO_2/UO_2 fuel containing 8 kg/Te of plutonium with 24% Pu-240. In each case, the heated channel has been surrounded by 36 channels of similar cold fuel. The measurements have shown that the small discrepancies observed between measured cluster mean reaction rates and the METHUSELAH II and WIMS predictions are not significantly increased by the rise in temperature. The total reactivity effects of these discrepancies on the temperature defects are less than 0·5% for both methods of calculation and lie within the experimental uncertainties. The use of transport theory and the better spectrum models available in WIMS lead to improved predictions of cluster distributions of spectrum sensitive reaction rates in the hot condition.

Further reactor physics studies for steam generating heavy water reactors

Part 3: Coolant temperature effects in UO_2 and PuO_2/UO_2 fuels

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INTRODUCTION

THE study of a wide range of uniform and multi-zone steam generating heavy water cores has been described in previous papers.[1, 2] This work forms part of a series of reactor physics investigations in support of the 100 MW(e) prototype SGHW reactor which was brought into operation at AEE Winfrith during 1967 and is described in detail elsewhere.[3, 4]

2. During the preliminary planning of the experimental programme, the five-group diffusion theory code METHUSELAH[5] was used for a parametric survey of SGHW lattices in the region of interest. Calculations were carried out for a range of fuel and coolant temperatures, and it was shown that interactions between the effects of temperature and the associated density changes are small in SGHW lattices. The programme was therefore planned to investigate temperature and density effects in separate experiments. In the work described in Parts 1 and 2 of the series, the effect of coolant density variations in uniform and multi-zone cores was measured in some detail using D_2O/H_2O mixtures or polystyrene/H_2O mixtures to obtain a reduced effective coolant density. This work has shown that the METHUSELAH–AIMAZ[6] method of calculation incorporated in the PATRIARCH[7] group of design codes can predict average core void coefficients with good accuracy.

3. Under ideal conditions, temperature effects are determined by studying the physical behaviour of a critical system at a range of temperatures covering the operating conditions for the power reactor design. However, the plant required for such experiments is extremely expensive, particularly in the case of liquid coolants where pressures in excess of 1000 p.s.i. are required to suppress boiling. In addition, the engineering complexity required to ensure the safe and reliable operation of such plant makes it difficult to provide the rapid access to the fuel which is necessary for a large-scale experimental programme. The compromise adopted for the SGHW study at Winfrith was to measure temperature effects up to 280°C in a single-channel hot loop installed in the zero energy reactor JUNO, and to complement the information obtained in this way by a few whole core measurements in DIMPLE over a restricted range of temperature.

4. The design of the JUNO hot loop is described in the Paper. The validity of extrapolating from measurements in a single channel has been investigated by comparing the results of single and multi-channel calculations. Since the main purpose of this series of experiments was to validate current design methods, average cell constants were obtained from the METHUSELAH II code[8] but advantage

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was taken of the refinements available in the WIMS code,[9] particularly in the representation of the thermal spectrum, to provide alternative cell constants and to carry out multi-group investigations of important perturbing effects. The programme of experiments also included a systematic study of perturbations with the emphasis on cross-channel interaction effects and the small but significant differences between the hot loop and a standard fuel channel.

5. At each temperature, detailed reaction rates have been determined in representative fuel pins in the hot loop cluster, and these are used to comment on the theoretical predictions of individual phenomena, e.g. fast fission, resonance capture, etc. Finally, the effect of the reaction rate discrepancies on the computed reactivity changes are calculated.

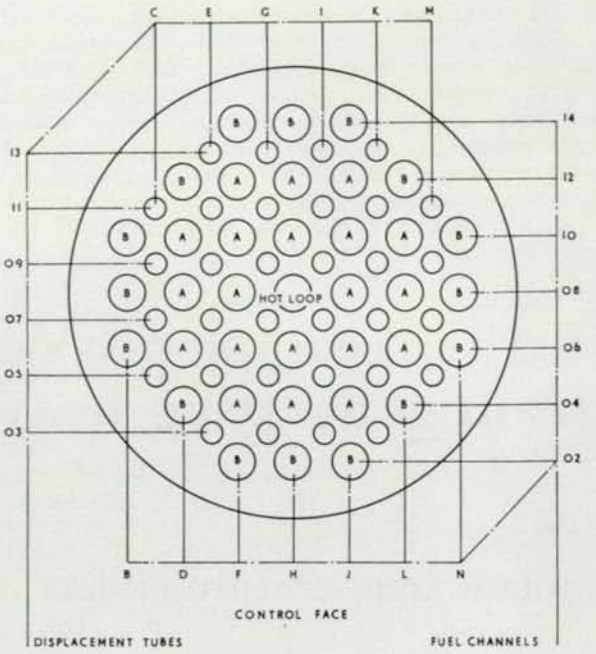
DESCRIPTION OF EXPERIMENTAL EQUIPMENT

Single-channel hot loop in JUNO

For the study of temperature effects, the core in the zero energy reactor JUNO consisted of 37 fuel channels arranged on a square pitch of 10·25 in., as shown in Fig. 1. The hot loop occupied the central position and was surrounded by 36 unheated channels containing identical or similar fuel clusters, chosen from the range of uniform cluster lattices described in an earlier paper in this series.[1] Because of the effects of thermal and near-thermal resonances, the problem of computing temperature effects is more severe in cores containing a substantial fraction of plutonium, and this aspect was therefore emphasized in planning the experimental programme. The range of fuels studied in JUNO is shown in Table 1. In each case, all 37 fuel channels contained 74 or 75 pins with a nominal fuel pellet diameter of 0·4 in. With the exception of the hot loop channel, aluminium pressure and calandria tubes 5·25 in. i.d. and 7·25 in. o.d. respectively were used to give a moderator to fuel volume ratio of 6·7 and a coolant to fuel volume ratio of 0·9. These volume ratios correspond closely to the corresponding ratios for the SGHWR prototype, namely 6·8 and 1·1 respectively.

7. The JUNO hot loop consisted of a single Zircaloy pressure tube placed in the centre of the JUNO core connected to an external process unit adjacent to the reactor. This unit circulated hot water through the central pressure tube, inlet water passing between the 75 pins to the bottom of the pressure tube and then returning to the outlet at the top via two channels made of 0·004 in. thick stainless steel and situated at either end of a cluster diameter (Fig. 4). A maximum coolant temperature of 274°C was attained. Boiling was suppressed by raising the system pressure by means of an external supply of nitrogen. The in-core section of the loop contained a single fuel cluster which was similar but not identical to the surrounding clusters, for reasons discussed below.

8. Access was provided to the fuel in seventeen of the 75 fuel pins. The top of the pressure tube was closed by a breech block (Fig. 2) which could be removed when the system pressure was equal to the atmospheric pressure, so providing access to the tops of the 17 fuel pins. Since the fuel stops 6 ft below the reactor top, these tubes were approximately 14 ft long, the fuel being contained in the lower 8 ft 6 in. Inside the tubes, the fuel pellets were encapsulated in 0·008 in. aluminium capsules about 2·4 in. long. Removal of the fuel capsules from these tubes was achieved by lowering a small bore tube down the fuel pencil until the lower end was in contact with the capsule; a vacuum was then applied and the capsule lifted from the core. Foil irradiations were carried out by loading foils into fuel capsules between fuel pellets.



FUEL COMPOSITION

EXPERIMENT	HOT LOOP	FUEL A	FUEL B
LOOP 1	1·35% UO_2	1·35% UO_2	0·8% PuO_2/UO_2 (10% Pu-240)
LOOP 2	0·8% PuO_2/UO_2 (10% Pu-240)	0·8% PuO_2/UO_2 (10% Pu-240)	1·35% UO_2
LOOP 3	0·8% PuO_2/UO_2 (24% Pu-240)	0·8% PuO_2/UO_2 (10% Pu-240)	1·35% UO_2

Fig 1 **JUNO core plan**

Table 1

	Hot loop		Remainder of core	
Number	U-235, %	Total Pu, %	U-235, %	Total Pu, %
1	1·35	—	1·35	—
2	0·40	0·80 (10% Pu-240)	0·40	0·80 (10% Pu-240)
3	0·40	0·80 (24% Pu-240)	0·40	0·80 (10% Pu-240)*

* In the case of item 3, sufficient fuel containing 24% Pu-240 was available to construct a single channel only.

Differences between hot loop and surrounding channels

One of the main objectives of the design of the hot loop channel was to make it as similar to the remaining 36 standard channels as possible (Fig. 3). However, because of the

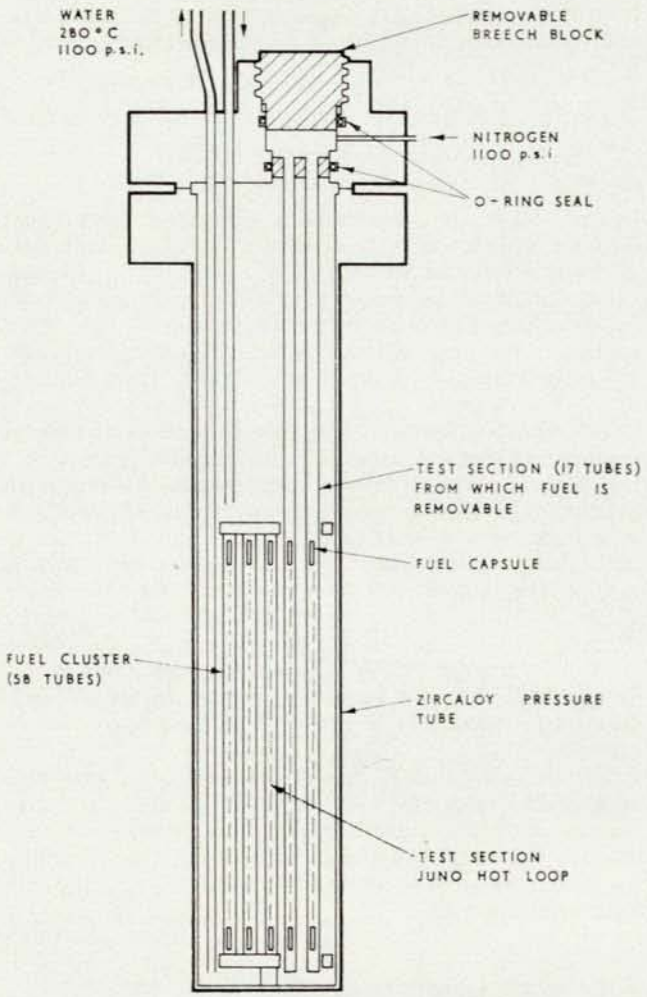


Fig 2 **JUNO hot loop**

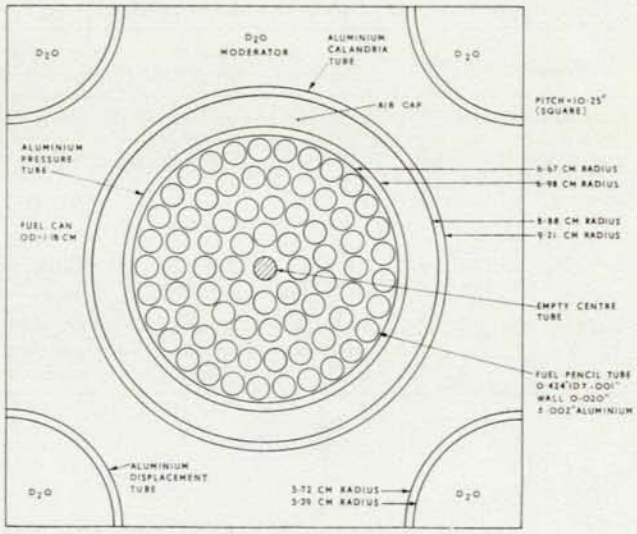


Fig 3 **Standard driver region channel in JUNO**

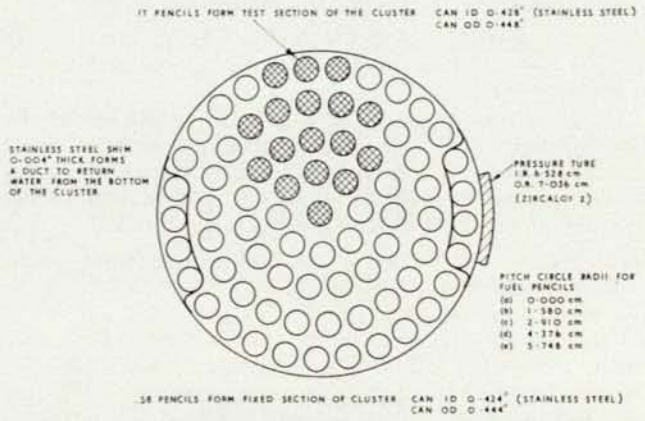


Fig 4 **Cross section of hot loop fuel cluster**

operating conditions in the loop, certain differences were inevitable, and these are listed and discussed below.

(*a*) *Pressure tube is of Zircaloy and is of diameter 5·14 in., compared with 5·25 in. in standard channels*

This choice of pressure tube diameter resulted from the decision to use an actual SGHWR pressure tube, which was available to the project. The difference in diameter is small and can be taken into account in the calculations. The thickness (0·125 in.) of the aluminium pressure tubes had been previously chosen to match the thermal neutron absorption of the 0·20 in. thick Zircaloy pressure tubes of the reactor. Since the proportion of neutrons captured by the pressure tube is in any case only $\simeq 2\%$, the error in calculating any small mis-match will be small.

(*b*) *Fuel cladding and cluster grids are of stainless steel, instead of aluminium as in the standard clusters*

In addition, two channels of 0·004 in. stainless steel are provided to guide the returning hot water to the top of the pressure tubes (Fig. 4). Provision of a return pipe for the flow out through the bottom of the reactor tank was not possible, since it was required that it should be possible to install and remove the loop from the reactor top, so as to avoid excessive delays in the reactor programme. Aluminium would not stand the conditions of the hot loop, so a choice had to be made between constructing the cluster from stainless steel or Zircaloy-2. The main reasons behind the choice of stainless steel were the existence of a proved conventional technology which presented no difficulties in the construction of a cluster suitable for the hot loop, and the high cost of Zircaloy-2 tubing which would have substantially increased the cost of the experimental programme. The thermal neutron absorption by the additional stainless steel is calculated to reduce k-infinity by 0·10, compared with the value for a similar cluster built from aluminium

Table 2: *Experimental cluster components*

Type of cluster	Fuel can	Cluster grids	Hot water channels
Standard channel	0·020 in. Al	0·062 in. Al	None
Experimental channel A	0·010 in. Al	0·062 in. Al	None
Experimental channel B	0·010 in. Al	0·062 in. stainless steel	None
Experimental channel C	0·010 in. stainless steel	0·062 in. stainless steel	None
Hot loop channel	0·010 in. stainless steel	0·062 in. stainless steel	0·004 in. stainless steel

components. This difference is clearly significant, and three additional clusters (Table 2) were therefore provided so that the calculations of the effects of each added stainless steel component could be checked in a separate experiment. These experimental clusters were simply constructed so as to be satisfactory for use in the cold condition only.

(*c*) *Outer diameters of the 17 fuel pins accessible from the reactor top were increased by 0·004 in. compared with the remainder of the core*

This represents a change of only 0·4% in the coolant to fuel volume ratio. Since fuel capsules had to be removed from the cores by means of a vacuum system, the extra clearance was essential to avoid the possibility of time-consuming jamming of fuel capsules in the tubes.

(*d*) *Fuel supplied for use in the loop was enclosed in aluminium capsules of a different design from those used in the outer regions of the core*

In the case of UO_2 fuel, the capsules in the hot loop were made from 0·008 in. aluminium compared with 0·001 in. aluminium foil in the remainder of the core. For PuO_2 fuel a sealed capsule was required and 0·008 in. and 0·004 in. thick aluminium was used in the loop and outer core channels respectively. The amounts of aluminium concerned are small and have been taken into account in the calculations by adjusting the average amount of capsule material associated with each fuel pin.

EXPERIMENTAL MEASUREMENTS

The JUNO hot loop measurements involved heating the central pressure tube of Zircaloy-2 to the operating temperature of the SGHWR (about 280°C) by means of circulating hot pressurized water through the pressure tube. Stable temperature conditions were easily obtainable in the loop, and loss of heat to the surrounding bulk moderator produced trivial increases of moderator temperature. The system provided well defined and easily controlled conditions, making it possible to make accurate measurements of the changes in detailed reaction rates associated with heating the loop. For each of the three fuels listed in Table 1, a full set of reaction rates was measured at ambient temperature and at the maximum temperature, some additional measurements being made at intervening temperatures. The following reaction rate ratios were determined:

(i) U-238/U-235 fast fission ratio (FR)

(ii) U-238 capture/U-235 fission relative to the same ratio in a thermal column. (RCR or relative conversion ratio)

(iii) Pu-239/U-235 relative to thermal column value (Pu/U)

(iv) Lutecium/manganese relative to thermal column value (Lu/Mn)

(v) Maximum/average U-235 fission rate in cluster.

In cases (i) to (iv) measurements were made in sufficient representative fuel pins to determine both the distribution of reaction rates across the cluster and the cluster average value. Additional evidence on temperature effects was obtained by repeating these measurements with the light water coolant in the loop replaced by a D_2O/H_2O mixture with the same slowing down power as H_2O at a reduced density of about 0·6 g/cm^3.

11. The techniques used for measuring reaction rates at temperature were the same as those described previously[10] for measurements at ambient temperature. Following an irradiation for the combined measurement of FR and RCR it was necessary to start counting within about 90 min to obtain adequate statistics from the depleted foil which is used in the double enrichment method[11] to differentiate U-238 and U-235 fission events. It was found in practice that the loop channel cooled from about 275°C to 40°C in 30 min, so that the nitrogen overpressure could be removed, thus making it possible to unbolt the breech-block mechanism shown in Fig. 2 and so gain access to the fuel capsules.

12. As in previous RCR measurements, about one third of the determinations were repeated by the alternative method involving chemical separation of fission products[12] to check the possible existence of systematic errors in the standard measurement of U-238 capture by the γ–X-ray coincidence method.

THEORETICAL METHODS

Introduction

The PATRIARCH[7] scheme of design codes for SGHW reactor systems uses the five-group diffusion theory code METHUSELAH[5] for producing average cell constants for input to the two-dimensional XY, two-group diffusion code AIMAZ,[6] which calculates core power distributions and reactivity. The JUNO core arrangement has been described in § 6 (Fig. 1) and consists of a central heated channel surrounded by similar cold channels. Measurements of detailed reaction rates are made in the hot loop at a range of coolant temperatures, and it is the primary purpose of the experiment to use these measurements to comment on the accuracy of METHUSELAH–AIMAZ predictions of temperature effects.

14. WIMS[9] calculations have also been carried out for each of the hot loop measurements and the results are compared with the METHUSELAH predictions and the measured values. The version of WIMS used for this work was WIMS I, which uses a capture to fission ratio (α_{25}) for U-235 of 0·5 above 0·5 eV and incorporates the reduction of 0·2 barns in the U-238 absorption resonance cross

Table 3: *Calculated interchannel effects in JUNO loop* $\left(\frac{JANUS-METHUSELAH}{METHUSELAH}\%\right)$

Region	Coolant in hot loop cell		FR	RCR	Pu/U	Lu/Mn	U-235 fine structure
Outer fuel pin	H_2O	23°C					+0·8
	H_2O	274°C					+0·7
	D_2O/H_2O	23°C					+1·2
	D_2O/H_2O	274°C					+1·1
Cluster average	H_2O	23°C	−2·2	−0·4	0·0	+0·1	
	H_2O	274°C	−3·3	−0·7	−0·1	+0·5	
	D_2O/H_2O	23°C	−1·7		−0·1	+0·4	
	D_2O/H_2O	274°C	−3·1	−0·1	0·0	0·0	

section as proposed by Askew.[13] The cell spectrum is calculated in 69 groups which are condensed to 18 groups for the transport solution using the Winfrith DSN code[14] with $S_n=4$. A Nelkin thermal neutron scattering model is used which predicts a diffusion length in pure water of 2·74 cm, which is in good agreement with experiment. For heavy water a Honeck model is used in the bulk moderator, but in the coolant recourse has to be made to the 'effective width' model as the former is only tabulated at room temperature. Experience at room temperature suggests that these two D_2O models should yield very similar results. Non-standard features of the hot loop cell have been taken into account in the same way as in the METHUSELAH calculations.

15. If the channel pitch were sufficiently large it would be possible to treat the hot loop results in isolation ignoring interchannel effects. However, preliminary calculations carried out for the JUNO lattice pitch of 10·25 in. showed that interchannel effects, although small, could not be entirely ignored. While the METHUSELAH–AIMAZ method of calculation has been shown to predict complex SGHW core power distributions with good accuracy,[4] it may not necessarily predict single-channel perturbations with the accuracy required for comparison with experiment, and it was therefore decided to investigate this aspect of the problem using the more flexible JANUS codes.[15] This critical investigation is described in § 16 *et seq.* and it is concluded that single-channel measurements are adequate to confirm the absence of significant discrepancies in the calculated temperature dependence of detailed reaction rates.

Channel interaction effects

A theoretical investigation of channel interaction effects in the hot loop was carried out by means of the JANUS 5 program. This computes fine structure and reaction rates in adjoining cells in an infinite chessboard array, leakage from cell A being transmitted to cell B and vice versa. Five neutron groups are used and ring smearing of the fuel pins can be used in place of the cell smearing method adopted in AIMAZ. The JANUS 2 code[15] treats the same problem with two neutron groups only. In both the JANUS codes, as in AIMAZ, cell average or boundary fluxes can be used to convert the METHUSELAH reaction rates into the cross sections required as input to the subsequent calculation.[4]

17. In the present calculation, one cell of the chessboard array was the hot loop cell with 1·35% UO_2 fuel and H_2O or D_2O/H_2O coolant; the other cell was typical of the slightly different surrounding driver region cells with the same UO_2 fuel and with cold H_2O coolant throughout. Since the hot loop in JUNO is surrounded on four sides by driver region cells, the JANUS chessboard representation simulates the real situation well, only diagonal interchannel effects being underestimated. The calculations should therefore be quite adequate to show whether interaction effects are important.

Table 4: *Validity of representation of water outlet duct*

Calculation method	FR	RCR	Pu/U	U-235 max/av.
WIMS (Extra steel in outer ring)	0·1465	2·143	1·361	1·415
WIMS (Extra steel uniform)	0·1464	2·141	1·361	1·417

18. The results of the JANUS 5 chessboard calculation are compared with a five-group METHUSELAH cell calculation in Table 3. Ring smearing is used in both calculations. These results show that the detailed reaction rates in the hot loop are affected to a small extent by the presence of dissimilar adjoining cells. The changes in these discrepancies as the loop is heated up to 270°C are of the same order or smaller than the uncertainties in the corresponding measurements (Tables 5–9). It is therefore concluded that the measurements of temperature effects on reaction rate determinations in the JUNO loop can be compared directly with METHUSELAH cell calculations which neglect interchannel effects. It is thus possible to avoid the very significant increase in computer time which would be involved in carrying out an AIMAZ whole core calculation for each hot loop experiment. The disadvantage of this approach is that the METHUSELAH predictions of individual reaction rates at a given temperature may be in error by amounts similar to those shown in Table 3, and are therefore not directly comparable with the measured values. However, this situation is acceptable since abundant evidence on the accuracy of METHUSELAH predictions of reaction rates in regular cluster lattices has been published previously.[1]

19. In the case of thermal fine structure, some additional experimental and theoretical evidence is available to confirm that interaction effects are relatively small. In a uniform

Table 5: *U-238/U-235 fission ratio*

Coolant	Fuel	Experiment	METHUSELAH	Δ METH	WIMS	Δ WIMS
Light water at 23°C	1·35% UO_2	0·0565 ± 0·0008	0·0546	−3·3%	0·0479	−15·2%
Light water at 274°C	1·35% UO_2	0·0565 ± 0·0009	0·0576	+1·9%	0·0509	− 9·9%
				+5·2 ± 1·9		+5·3 ± 1·9
Mixture at 23°C	1·35% UO_2	0·0566 ± 0·0005	0·0563	−0·5%	0·0496	−12·4%
Mixture at 274°C	1·35% UO_2	0·0580 ± 0·0007	0·0590	+1·7%	0·0523	− 9·8%
				+2·2 ± 1·5		+2·6 ± 1·5
Light water at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	0·2365 ± 0·0016	0·2302	−2·7%	0·2032	−14·1%
Light water at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	0·2631 ± 0·0014	0·2753	+4·6%	0·2420	− 8·0%
				+7·3 ± 0·9		+6·1 ± 0·9
Mixture at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	0·2435 ± 0·0017	0·2447	+0·5%	0·2120	−12·9%
Mixture at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	0·2636 ± 0·0014	0·2784	+5·6%	0·2440	− 7·4%
				+5·1 ± 0·9		+5·5 ± 0·9
Light water at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	0·2015 ± 0·0009	0·2058	+2·1%	0·1835	− 8·9%
Light water at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	0·2459 ± 0·0032	0·2456	−0·1%	0·2180	−11·7%
				−2·2 ± 1·4		−2·4 ± 1·4
Mixture at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	0·2097 ± 0·0012	0·2186	+4·2%	0·1906	− 9·1%
Mixture at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	0·2396 ± 0·0014	0·2491	+4·0%	0·2194	− 8·4%
				−0·2 ± 0·8		+0·7 ± 0·8

core fuelled with 0·8% PuO_2/UO_2 (Table 1) the maximum/average U-235 fission rate in the central channel was found to increase by only 2% when the density of the coolant in all the surrounding similar channels was increased from about 0·6 to 1·0 g/cm³. For the JUNO hot loop containing fuel 3 of Table 1, 7-group PIJ-WIMS cell calculations of thermal fine structure have been carried out with both light water and mixture coolants. Geometrical representation of the individual fuel pins is explicit in such a calculation but the effects of neighbouring dissimilar channels ignored. The results of these calculations agree with the measured maximum/average U-235 fission rates to better than 1% in both cases, thus confirming the trend previously observed in regular cluster lattices,[1] where channel interaction effects are not present. It is therefore concluded that in general, channel interaction effects on reaction rates measured in the hot loop are sufficiently small to be neglected when comparing measured temperature effects with theoretical predictions.

REACTION RATE MEASUREMENTS IN THE JUNO HOT LOOP

Introduction

The operating conditions in the hot loop channel made it necessary for the fuel cluster to be non-standard in various ways, as discussed in § 9 and illustrated in Figs 3 and 4. In the METHUSELAH and WIMS calculations for comparison with the measured reaction rates these special features were taken into account. The thermal absorption cross sections of samples of the materials used to fabricate all non-standard components were measured in the GLEEP reactor at Harwell. The methods of representing the new features of the cluster in the calculations are obvious with the following exceptions.

Stainless steel cluster grids

Since reaction rate measurements were made in a plane at least 10 cm from the nearest grid, the presence of the grids was ignored in computing reaction rates.

Larger diameter of accessible fuel cans

The increase in the diameter of these 17 fuel cans led to a reduction of 0·4% in channel coolant volume and was taken into account by using a correct cluster average value for all the fuel can diameters.

Hot water outlet ducts

These ducts of 0·004 in. stainless steel are located at opposite ends of a cluster diameter. The steel forming these channels was taken into account by increasing the thickness of the steel canning on the outer row of fuel pins. To check the validity of the procedure 18-group WIMS calculations using the WDSN option were carried out, first with the additional steel arranged as described above and then with the steel uniformly distributed among all the fuel cans. The effects on the calculated reaction rate ratios in the most sensitive outer row of fuel pins are shown in Table 4 to be quite negligible.

21. At ambient temperature and at the maximum loop temperature of 274°C, reaction rate ratios have been measured with H_2O in all channels and then with the hot loop coolant changed to a D_2O/H_2O mixture simulating H_2O at an effective coolant density of about 0·6 g/cm³. The results of these measurements are compared with METHUSELAH and WIMS predictions in the following sections. In each case, the observed discrepancies between the measured and predicted temperature effects are related to the associated changes in k-infinity by the method described in reference 16.

Table 6: *Relative conversion ratio*

Coolant	Fuel	Experiment	METHUSELAH	Δ METH	WIMS	Δ WIMS
Light water at 23°C	1·35% UO_2	2·004 ± 0·004	1·970	−1·7%	1·942	−3·1%
Light water at 274°C	1·35% UO_2	2·153 ± 0·004	2·138	−0·7%	2·098	−2·6%
				+1·0 ± 0·3		+0·5 ± 0·3
Mixture at 23°C	1·35% UO_2	2·138 ± 0·004	2·157	+0·9%	2·100	−1·8%
Mixture at 274°C	1·35% UO_2	2·261 ± 0·004	2·292	+1·4%	2·221	−1·8%
				+0·5 ± 0·3		0·0 ± 0·3
Light water at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	2·301 ± 0·020	2·237	−2·8%	2·238	−2·7%
Light water at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	2·619 ± 0·024	2·609	−0·4%	2·595	−0·9%
				+2·4 ± 1·3		+1·8 ± 1·3
Mixture at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	2·541 ± 0·022	2·511	−1·2%	2·462	−3·1%
Mixture at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	2·717 ± 0·024	2·807	+3·3%	2·746	+1·1%
				+4·5 ± 1·2		+2·0 ± 1·2
Light water at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	2·158 ± 0·019	2·117	−1·9%	2·149	−0·4%
Light water at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	2·484 ± 0·022	2·451	−1·3%	2·479	−0·2%
				+0·6 ± 1·3		+0·2 ± 1·3
Mixture at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	2·359 ± 0·021	2·359	0·0%	2·352	−0·3%
Mixture at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	2·579 ± 0·023	2·630	+2·0%	2·621	+1·6%
				+2·0 ± 1·3		+1·9 ± 1·3

U-238/U-235 fission ratio

The comparison of the mean fuel values of the U-238/U-235 fission ratios is shown in Table 5. The quoted experimental error represents the random component only. In addition there may be a systematic error of up to ±3% arising from the normalization of the measurement.[10] METHUSELAH overestimates FR by an average of 1·5 ± 2·7%, while WIMS underestimates it by about 10·1 ± 2·4% as observed in previous SGHW comparisons.[1] The evidence in Table 3 suggests that the theoretical values would be reduced by 2–3% if channel interaction effects were taken into account. On the variations of these discrepancies with temperature there is good agreement between the two methods of calculation. In the UO_2 and first PuO_2/UO_2 fuels, the predicted values show a larger rise with temperature than is observed, the average discrepancy being 5%. If we assume that this discrepancy is due to an error in the U-238 fission cross section, then it is equivalent to about 0·2% in reactivity; if we assume instead that the discrepancy is due entirely to an error in the removal cross section then the effect on reactivity is effectively zero. The predicted temperature effect for the third fuel containing 24% Pu-240 is in better agreement with experiment.

Relative conversion ratio (RCR)

The comparison of the mean fuel values of RCR is shown in Table 6. In addition to the random error quoted, a total systematic error of up to ±0·9% may be present.[10] WIMS underestimates RCR by an average of −1·2 ± 1·6%, thus providing further evidence that the reduction of 0·2 barns in the U-238 absorption cross section in the range 9·118 KeV to 4 eV is slightly too large. In every case both WIMS and METHUSELAH overestimate the increase of RCR with temperature. Taking the means of all the cases, METHUSELAH overestimates the change by 1·8 ± 1·5%, while WIMS overestimates it by 1·1 ± 0·8%; thus WIMS shows a small improvement over METHUSELAH.

24. For all cases an overestimate of 1% in RCR is equivalent to an error of about −0·3% in reactivity if the U-238 resonance integral is responsible. Thus the discrepancies are on an average worth 0·5% for METHUSELAH and 0·3% for WIMS, for the reactivity change on heating.

Pu-239/U-235 fission ratio

The comparison of the mean fuel values of normalized Pu-239/U-235 fission ratios is shown in Table 7. Additional systematic errors of ±1% may be present in the measured values.[10] In the UO_2 fuelled cluster, both WIMS and METHUSELAH predict the normalized Pu/U ratio to within 2·5% and the effects of temperature to within about 1% or better.

26. In the clusters containing plutonium the discrepancies are larger, particularly with the METHUSELAH calculation, which overestimates the Pu/U ratio in these eight cases by an average of 3·3 ± 2·7%. When considered in isolation, the measurement in 10% Pu-240 with hot mixture coolant appears anomalous since it is overestimated by 7·0% by METHUSELAH and by 4·4% by WIMS. However, the corresponding comparisons for the plutonium of higher 240 content show very similar trends with discrepancies of 6·1% and 3·5% respectively, thus suggesting that a real effect is present. The WIMS predictions of Pu/U ratios in these clusters are more accurate than the METHUSELAH values, as would be expected from the more detailed thermal spectrum representation available in WIMS. The predictions of temperature effects are similar

Table 7: *Pu-239/U-235 fission ratio*

Coolant	Fuel	Experiment	METHUSELAH	Δ METH	WIMS	Δ WIMS
Light water at 23°C	1·35% UO_2	1·276 ± 0·004	1·246	−2·4%	1·265	−0·9%
Light water at 274°C	1·35% UO_2	1·543 ± 0·005	1·522	−1·3%	1·544	+0·1%
				+1·1 ± 0·5		+1·0 ± 0·5
Mixture at 23°C	1·35% UO_2	1·303 ± 0·008	1·329	+2·0%	1·312	+1·6%
Mixture at 274°C	1·35% UO_2	1·506 ± 0·006	1·531	+1·6%	1·536	+2·0%
				−0·4 ± 0·7		+0·4 ± 0·7
Light water at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·217 ± 0·015	1·209	−0·7%	1·198	−1·6%
Light water at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·399 ± 0·015	1·421	+1·6%	1·400	+0·3%
				+2·3 ± 1·6		+1·9 ± 1·6
Mixture at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·212 ± 0·015	1·255	+3·6%	1·210	−0·2%
Mixture at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·301 ± 0·015	1·392	+7·0%	1·358	+4·4%
				+3·4 ± 1·7		+4·2 ± 1·7
Light water at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·194 ± 0·007	1·199	+0·4%	1·196	+0·2%
Light water at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·390 ± 0·007	1·422	+2·3%	1·401	+1·1%
				+1·9 ± 0·8		+0·9 ± 0·8
Mixture at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·177 ± 0·005	1·250	+6·2%	1·210	+2·8%
Mixture at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·319 ± 0·019	1·400	+6·1%	1·365	+3·5%
				−0·1 ± 1·5		+0·7 ± 1·5

in the two codes, both overestimating the spectrum hardening effect on the Pu/U ratio by an average of 1·9%. Such a discrepancy is worth about 0·3% in reactivity.

27. Discrepancies between METHUSELAH predictions and the measured changes in Pu/U ratio with effective coolant density have been discussed in a previous paper in this series.[1] It is interesting to note that this discrepancy persists throughout this range of measurements, METHUSELAH consistently overestimating the change from light water to mixture coolant by an average of 4·4 ± 1·0%. WIMS also overestimates this effect, but by only 2·5 ± 0·8%.

Lutecium/manganese activation ratio

The comparison of mean fuel values of the normalized lutecium/manganese activation ratio is shown in Table 8. Additional systematic errors of ±2% may be present in the measured values.[10] In all cases METHUSELAH underestimates the Lu/Mn activation ratio. This effect has been observed in previous SGHW lattices. Both METHUSELAH and WIMS agree in that they predict almost the same ratios for the two plutonium fuels, thus indicating that the Lu/Mn activation ratio is insensitive to Pu-240 content. This is confirmed by the experimental values, except for the hot mixture case where there is 5·5% difference between the results for the two plutonium fuels. The measured temperature effects are very consistently overestimated by WIMS by 4·5 ± 0·8% with the single exception of the hot mixture case. This evidence suggests that the experimental value for the plutonium fuel containing 24% of Pu-240 is in error and should be lower by about 3%. Omitting this point, METHUSELAH overestimates the heating effect by only 0·4 ± 1·8%, which is negligible in terms of reactivity.

U-235 fission fine structure

The comparison of the ratio of U-235 fission rate in the outer fuel ring to the mean cluster value is given in Table 9. METHUSELAH consistently overestimates the maximum/average U-235 fission rate in the cluster by a few per cent, as has been observed in previous SGHW lattices. The prediction of the temperature effect is very consistent, the overestimate increasing in every case by an average of 1·3 ± 0·5%. WIMS predicts fine structure more accurately than METHUSELAH in the cold conditions and the prediction for hot coolants is even better, since the temperature effect is underestimated in every case by an average of 1·2 ± 0·7%.

Reaction rate distributions across hot loop

Each of the cluster average values of the experimental reaction rate ratios quoted in the preceding tables is inferred from measurements in five representative fuel pins (Fig. 4). It is not possible to include all these results in a paper of this kind, but one set of typical reaction rate distributions is compared with METHUSELAH and WIMS predictions in Figs 5–9. In making these comparisons, interchannel effects on the cluster distributions should be taken into account, but it has been shown in § 16 *et seq.* that such effects are small, and they have therefore been neglected at this stage. In each case, the measurements were made at 274°C with PuO_2/UO_2 fuel containing 10% Pu-240 and with light water coolant. The measurements of FR and RCR are related to the U-235 fission distribution. In these cases, the measured U-235 fission distribution has therefore been used to eliminate this dependence in Figs 5 and 6. The experimental and theoretical values are normalized to unity for the cluster pin average value, except in Figs 7 and 8 in which normalized Pu/U and Lu/Mn values are plotted.

Table 8: *Lu/Mn activation ratio*

Coolant	Fuel	Experiment	METHUSELAH	Δ METH	WIMS	Δ WIMS
Light water at 23°C	1·35% UO_2	1·180 ± 0·020	1·120	−5·1%	1·187	+0·6%
Light water at 274°C	1·35% UO_2	1·629 ± 0·020	1·586	−2·6%	1·711	+5·0%
				+2·5 ± 2·1		+4·4 ± 2·1
Mixture at 23°C	1·35% UO_2	1·164 ± 0·020	1·154	−0·9%	1·204	+3·4%
Mixture at 274°C	1·35% UO_2	1·482 ± 0·020	1·475	−0·5%	1·614	+8·9%
				+0·4 ± 2·2		+5·5 ± 2·2
Light water at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·140 ± 0·020	1·082	−5·1%	1·132	−0·7%
Light water at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·498 ± 0·010	1·453	−3·0%	1·569	+4·7%
				+2·1 ± 2·2		+5·4 ± 2·2
Mixture at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·103 ± 0·020	1·089	−1·2%	1·123	+1·8%
Mixture at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	1·379 ± 0·020	1·327	−3·7%	1·453	+5·4%
				−2·5 ± 2·3		+3·6 ± 2·3
Light water at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·126 ± 0·020	1·083	−3·8%	1·138	+1·1%
Light water at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·516 ± 0·020	1·480	−2·4%	1·589	+4·8%
				+1·4 ± 2·2		+3·7 ± 2·2
Mixture at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·120 ± 0·020	1·097	−2·1%	1·134	+1·3%
Mixture at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	1·457 ± 0·020	1·361	−6·6%	1·480	+1·6%
				−4·5 ± 2·3		+0·3 ± 2·3

Table 9: *U-235 fission fine structure ratio of outer fuel ring to cluster mean*

Coolant	Fuel	Experiment	METHUSELAH	Δ METH	WIMS	Δ WIMS
Light water at 23°C	1·35% UO_2	1·3232 ± 0·0148	1·3730	+3·8%	1·3551	+2·4%
Light water at 274°C	1·35% UO_2	1·3253 ± 0·0116	1·3792	+4·1%	1·3287	+0·3%
				+0·3 ± 1·4		−2·1 ± 1·4
Mixture at 23°C	1·35% UO_2	1·3381 ± 0·0057	1·3684	+2·3%	1·3449	+0·5%
Mixture at 274°C	1·35% UO_2	1·3272 ± 0·0061	1·3779	+3·8%	1·3258	−0·1%
				+1·5 ± 0·6		−0·4 ± 0·6
Light water at 23°C	0·8% PuO_2 (10% Pu-240)	1·3751 ± 0·0020	1·4399	+4·7%	1·4273	+3·8%
Light water at 274°C	0·8% PuO_2 (10% Pu-240)	1·3952 ± 0·0026	1·4813	+6·2%	1·4227	+2·0%
				+1·5 ± 0·3		−1·8 ± 0·3
Mixture at 23°C	0·8% PuO_2 (10% Pu-240)	1·4018 ± 0·0055	1·4507	+3·5%	1·4210	+1·0%
Mixture at 274°C	0·8% PuO_2 (10% Pu-240)	1·4120 ± 0·0026	1·4801	+4·8%	1·4156	+0·3%
				+1·3 ± 0·4		−0·7 ± 0·4
Light water at 23°C	0·8% PuO_2 (24% Pu-240)	1·3621 ± 0·0013	1·4289	+4·9%	1·4204	+4·3%
Light water at 274°C	0·8% PuO_2 (24% Pu-240)	1·3772 ± 0·0021	1·4670	+6·5%	1·4156	+2·8%
				+1·6 ± 0·2		−1·5 ± 0·2
Mixture at 23°C	0·8% PuO_2 (24% Pu-240)	1·3792 ± 0·0021	1·4355	+4·1%	1·4133	+2·5%
Mixture at 274°C	0·8% PuO_2 (24% Pu-240)	1·3848 ± 0·0021	1·4652	+5·8%	1·4078	+1·7%
				+1·7 ± 0·2		−1·5 ± 0·2

31. Both METHUSELAH and the option of WIMS used to generate the majority of results quoted in the Paper employ a ring smearing technique to reduce each ring of fuel pins and associated coolant to an annular paste. The codes differ in their method of calculating hyperfine structure and also in the selection of ring boundaries for subdividing the coolant. It is not therefore surprising that they exhibit quite different error patterns in the prediction of ring-by-ring reaction rate distributions.

32. Fast fission is obtained in METHUSELAH by normalization of the U-238 fission cross section to the results of Monte Carlo calculations. Since this cross section adjustment is being used to compensate for the use of diffusion theory, METHUSELAH would not be expected

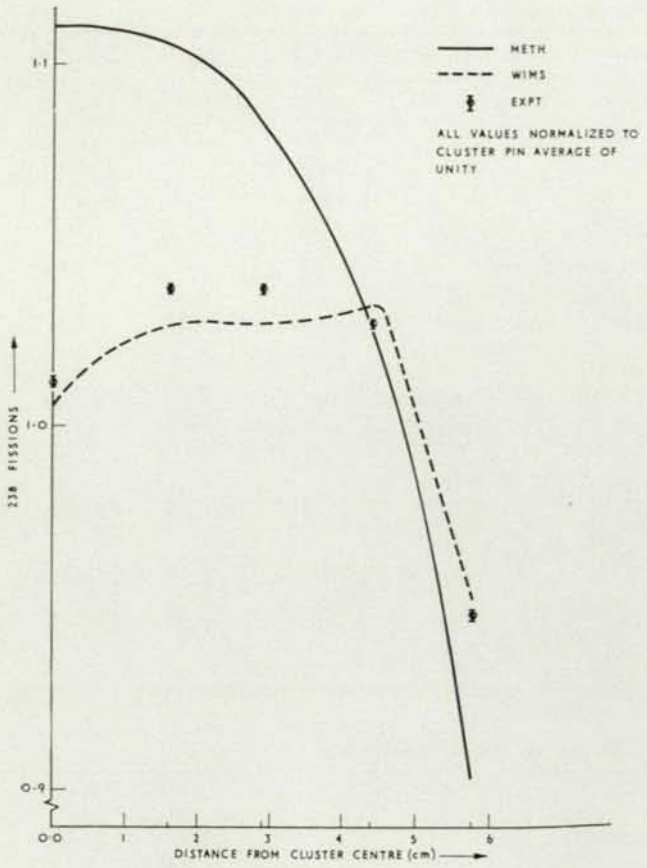


Fig 5 **Cluster distribution of U-238 fissions in PuO_2/UO_2 fuel 2 with H_2O coolant at 274°C**

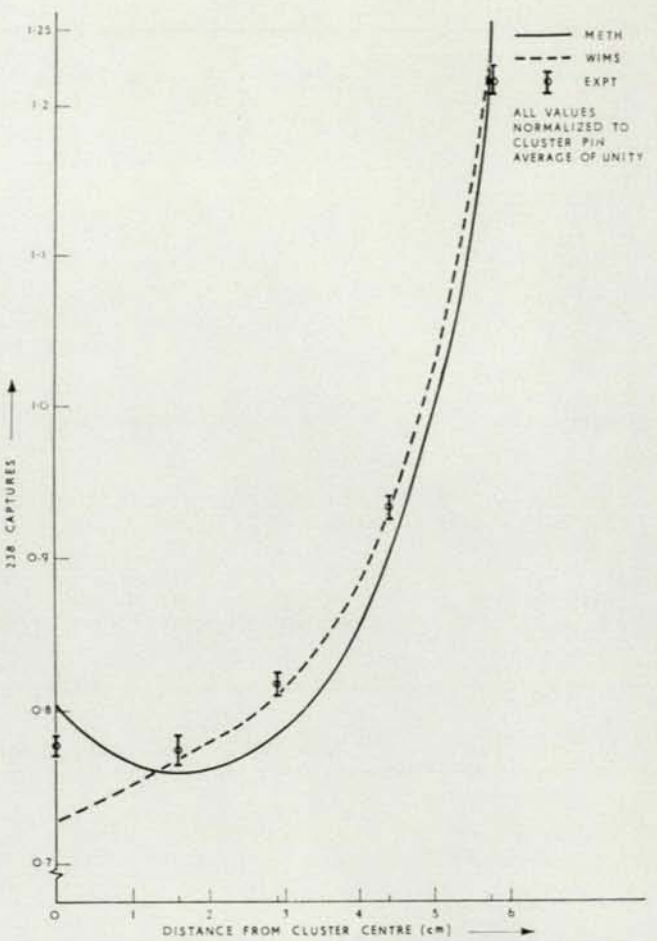


Fig 6 **Cluster distribution of U-238 captures in PuO_2/UO_2 fuel with H_2O coolant at 274°C**

to predict the U-238 fission rate distribution with high accuracy, and this is shown to be the case in Fig. 5. The plateau in the three intermediate rings is not reproduced and the error varies from a 10% overestimate at the central pin to a 4% underestimate of the outer ring. WIMS reproduces the shape of the curve very well and is within 1% at all points. In the case of relative conversion ratio, the variation of U-238 capture cross section across the cluster is reasonably well predicted by METHUSELAH, but again the WIMS prediction is better. Apart from the central pin, where it has a 5% underestimate, the latter code is within 1% of experiment in every ring.

33. The distribution of the spectral indicator reaction rates is a complex phenomenon and no one case can be regarded as typical. In room temperature lattices WIMS gives a good prediction of the spatial change of spectrum[1] but in the particular case illustrated here (Pu-239/U-235 and Lu/Mn in Figs 7 and 8) neither code is completely successful in the traditionally difficult task of predicting neutron spectra near a temperature discontinuity. It has been shown in § 29 that both METHUSELAH and WIMS predict the maximum/average U-235 fission rate well, but this is not a particularly sensitive test since the cluster average value is dominated by the outer rows of pins. Fig. 9 illustrates this point well and shows that in the case treated the METHUSELAH prediction of maximum/minimum ratio is 10% too large. WIMS gives much better agreement with experiment, except for the centre pin value where the prescription for defining the ring boundaries[1] seems to have led to an anomalous result.

EFFECTS OF THE REACTION RATE DISCREPANCIES ON REACTIVITY

The techniques of using reaction rates to determine *k*-infinity are described in detail in reference 16. This scheme has now been programmed as the LILLY computer code.[17] In this scheme three energy groups are used. Group 'f', the fast group, extends from 10 MeV to 5·53 keV and thus covers the range of the first two METHUSELAH groups. Group 'r', the resonance group, extends from 5·53 keV to 0·625 eV and is thus identical with the third METHUSELAH group. Group 't' is the thermal group which extends from 0·625 eV to 0 eV. The spectrum indicator measurement discrepancies, Pu-239/U-235 and Lu/Mn give a correction to the thermal component of reactivity; they are

Table 10: *Corrections to reactivity resulting from discrepancies between METHUSELAH and experimental reaction rates*

Coolant	Fuel	Δk-inf, %			k-inf and Δk_T		
		Group f	Group r	Group t	Total	Reaction rate	METHUSELAH
Light water at 23°C	1·35% UO_2	+0·13±0·06	−0·37±0·06	−0·22±0·05	−0·46±0·10	1·0706±0·0010	1·0752
Light water at 274°C	1·35% UO_2	−0·09±0·06	−0·14±0·06	−0·13±0·05	−0·36±0·10	1·0521±0·0010	1·0557
		−0·22±0·08	+0·23±0·08	+0·09±0·07	+0·10±0·14	−1·85±0·14%	−1·95%
Mixture at 23°C	1·35% UO_2	−0·02±0·04	+0·26±0·06	−0·00±0·09	+0·28±0·11	1·0782±0·0011	1·0754
Mixture at 274°C	1·35% UO_2	−0·08±0·05	+0·39±0·06	+0·06±0·06	+0·37±0·10	1·0600±0·0010	1·0563
		−0·10±0·06	+0·13±0·08	+0·06±0·11	+0·09±0·15	−1·82±0·15%	−1·91%
Light water at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	+0·09±0·03	−0·57±0·27	−0·08±0·18	−0·56±0·33	1·0933±0·0033	1·0989
Light water at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	−0·16±0·02	−0·21±0·27	−0·26±0·17	−0·63±0·32	1·0890±0·0032	1·0953
		−0·25±0·04	+0·36±0·38	−0·18±0·25	−0·07±0·46	−0·43±0·46%	−0·36%
Mixture at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	+0·07±0·03	−0·61±0·27	−0·42±0·18	−0·96±0·33	1·0812±0·0033	1·0908
Mixture at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	−0·06±0·02	+0·21±0·27	−0·76±0·17	−0·61±0·32	1·0758±0·0032	1·0819
		−0·13±0·04	+0·82±0·38	−0·34±0·25	+0·35±0·46	−0·54±0·46%	−0·89%
Light water at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	−0·09±0·02	−0·43±0·27	−0·18±0·09	−0·70±0·29	1·0197±0·0029	1·0267
Light water at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	+0·07±0·05	−0·47±0·27	−0·35±0·08	−0·75±0·29	1·0204±0·0029	1·0279
		+0·16±0·05	−0·04±0·38	−0·17±0·12	−0·05±0·41	+0·07±0·41%	+0·12%
Mixture at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	−0·05±0·02	−0·43±0·27	−0·83±0·06	−1·31±0·28	1·0092±0·0028	1·0220
Mixture at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	−0·03±0·02	+0·02±0·27	−0·85±0·22	−0·86±0·35	1·0083±0·0035	1·0166
		+0·02±0·03	+0·45±0·38	−0·02±0·23	+0·45±0·45	−0·12±0·45%	−0·57%

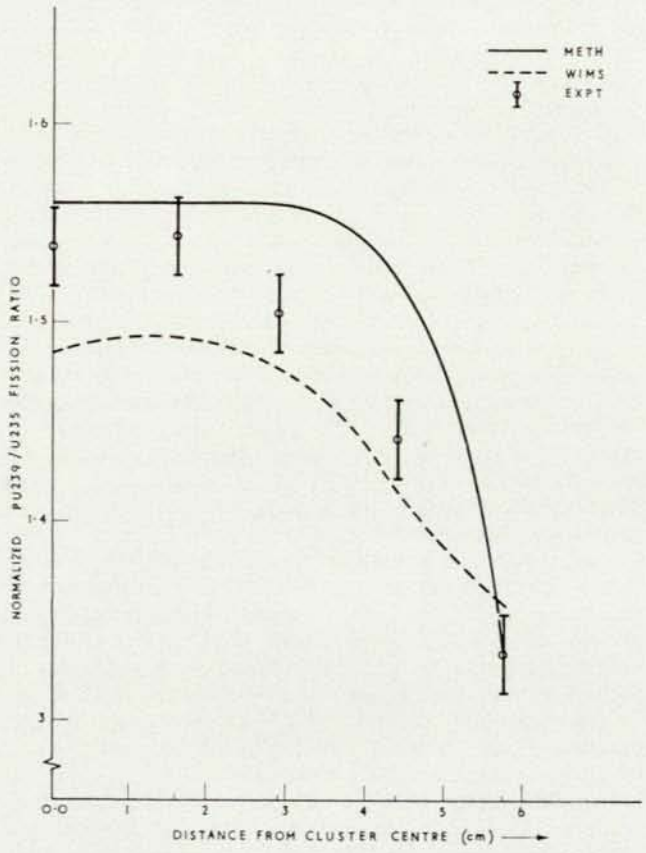


Fig 7 (above) **Cluster distribution of normalized Pu-239/U-235 ratio in PuO_2/UO_2 fuel 2 with H_2O coolant at 274°C**

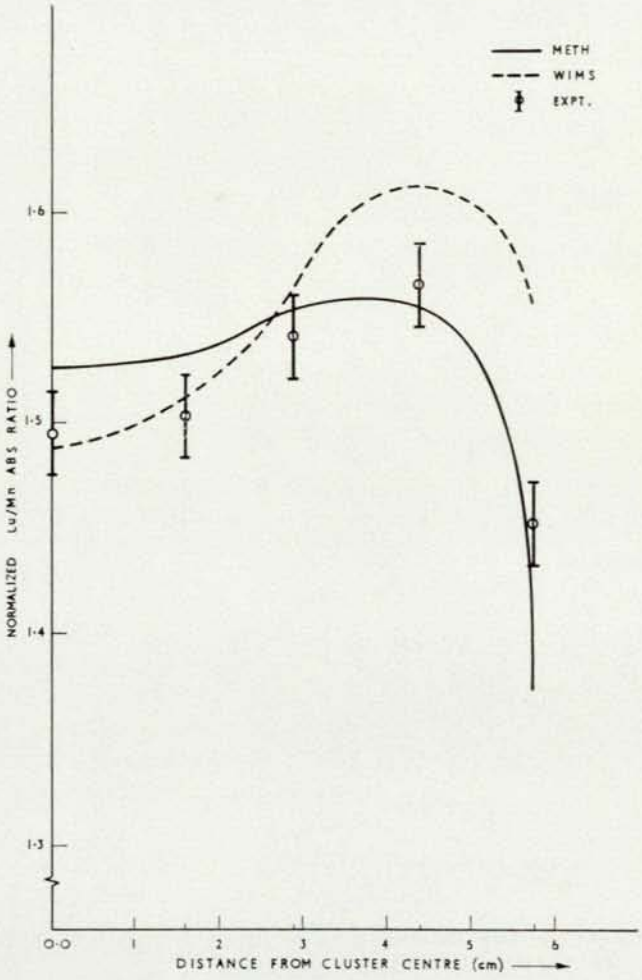


Fig 8 (right) **Cluster distribution of normalized Lu/Mn ratio in PuO_2/UO_2 fuel 2 with H_2O coolant at 274°C**

Table 11: *Corrections to reactivity resulting from discrepancies between WIMS and experimental reaction rates*

Coolant	Fuel	Δk-inf, %			k-inf and Δk_T		
		Group f	Group r	Group t	Total	Reaction rate	WIMS
Light water at 23°C	1·35% UO_2	+0·61±0·06	−0·83±0·06	+0·00±0·05	−0·22±0·10	1·0600±0·0010	1·0622
Light water at 274°C	1·35% UO_2	+0·42±0·06	−0·73±0·06	+0·01±0·05	−0·30±0·10	1·0436±0·0010	1·0466
		−0·19±0·08	+0·10±0·08	+0·01±0·07	−0·08±0·14	−1·64±0·14%	−1·56%
Mixture at 23°C	1·35% UO_2	+0·52±0·04	−0·57±0·06	+0·12±0·09	+0·07±0·11	1·0702±0·0011	1·0695
Mixture at 274°C	1·35% UO_2	+0·44±0·05	−0·66±0·06	+0·27±0·06	+0·05±0·10	1·0535±0·0010	1·0530
		−0·08±0·06	−0·09±0·08	+0·15±0·11	−0·02±0·15	−1·67±0·15%	−1·65%
Light water at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	+0·52±0·03	−0·55±0·27	+0·22±0·18	+0·19±0·33	1·1043±0·0033	1·1024
Light water at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	+0·33±0·02	−0·29±0·27	+0·05±0·17	+0·09±0·32	1·1062±0·0032	1·1053
		−0·19±0·04	+0·26±0·38	−0·17±0·25	−0·10±0·46	+0·19±0·46%	+0·29%
Mixture at 23°C	0·8% PuO_2/UO_2 (10% Pu-240)	+0·53±0·03	−0·82±0·27	+0·02±0·18	−0·27±0·33	1·0973±0·0033	1·1000
Mixture at 274°C	0·8% PuO_2/UO_2 (10% Pu-240)	+0·42±0·02	−0·18±0·27	−0·42±0·17	−0·18±0·32	1·0953±0·0032	1·0971
		−0·11±0·04	+0·64±0·38	−0·44±0·25	+0·09±0·46	−0·20±0·46%	−0·29%
Light water at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	+0·37±0·02	−0·19±0·27	+0·02±0·09	+0·20±0·29	1·0286±0·0029	1·0266
Light water at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	+0·52±0·05	−0·28±0·27	−0·11±0·08	+0·13±0·29	1·0338±0·0029	1·0325
		+0·15±0·05	−0·09±0·38	−0·13±0·12	−0·07±0·41	+0·52±0·41%	+0·59%
Mixture at 23°C	0·8% PuO_2/UO_2 (24% Pu-240)	+0·44±0·02	−0·40±0·27	−0·37±0·06	−0·33±0·28	1·0223±0·0028	1·0256
Mixture at 274°C	0·8% PuO_2/UO_2 (24% Pu-240)	+0·45±0·02	−0·05±0·27	−0·47±0·22	−0·07±0·35	1·0155±0·0035	1·0162
		+0·01±0·03	+0·35±0·38	−0·10±0·23	+0·26±0·45	−0·68±0·45%	−0·94%

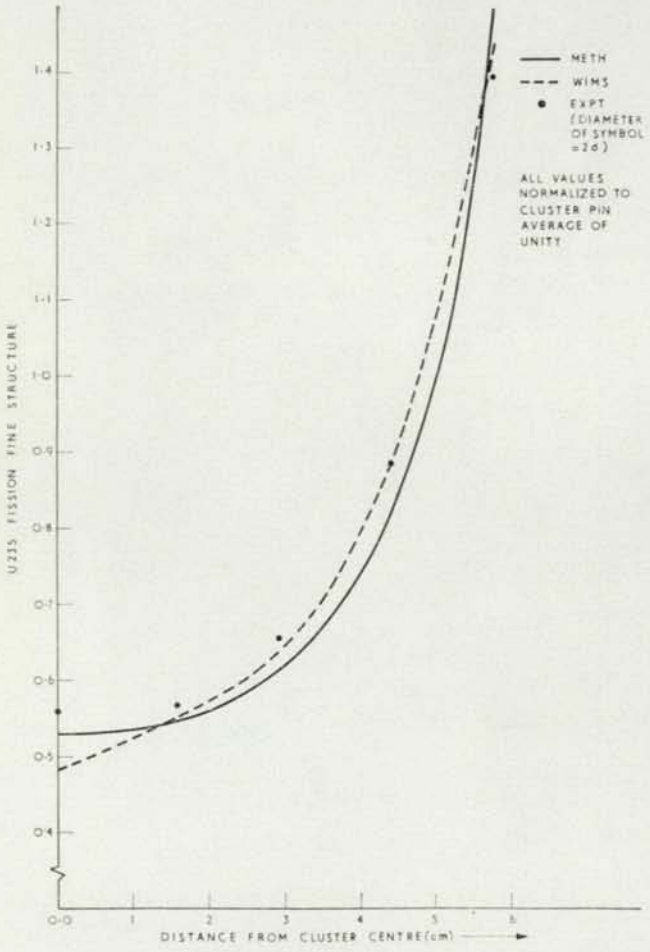


Fig 9 **Cluster distribution of U-235 fissions in PuO_2/UO_2 fuel with H_2O coolant at 274°C**

also used to make a thermal correction to the RCR and fast fission ratio results. The corrected RCR discrepancies are then used to give a correction to the resonance component of reactivity. The corrected fast fission ratio discrepancies are likewise used to correct the fast component of reactivity.

35. Table 10 shows the reactivity corrections obtained from the reaction rate discrepancies between METHUSELAH and experiment. The random and systematic errors arising in the measurement of these reaction rates have been discussed in reference 10. The uncertainties which have been attributed to the corrections to k-infinity in Table 10 are calculated from the random errors for the relevant reaction rate measurements. It is argued that in obtaining the differences between reaction rates at different temperatures, systematic errors largely cancel. With one exception the total correction to k-infinity is always less than 1%, and for the 12 cases studied the average total correction is $-0{\cdot}55\pm0{\cdot}48\%$. The temperature defect Δk_T is defined here by $\Delta k_T = k$-infinity (274°C) $- k$-infinity (23°C). The effects of the reaction rate discrepancies on Δk_T are very small, being less than 0·5% in every case. There is a trend for there to be a positive correction to the temperature defect, but this is scarcely significant when the experimental uncertainties are taken into account. The average of the six measured corrections is $+0{\cdot}14\pm0{\cdot}21\%$. This is shown in a different way by the measured and predicted values of the temperature defect in the last two columns of Table 10. In every case the METHUSELAH prediction agrees with the value of Δk_T inferred from the measured reaction rates to within the experimental uncertainties.

36. The corresponding comparisons with reaction rates predicted by WIMS are given in Table 11. The inferred total corrections to the WIMS k-infinity are even smaller than for METHUSELAH, the maximum correction being $-0{\cdot}33\pm0{\cdot}28\%$ for the case of the PuO_2/UO_2 cluster with high Pu-240 content at room temperature. The average of the six measured corrections to the temperature defect is

$+0{\cdot}01 \pm 0{\cdot}14\%$, and as with METHUSELAH the measured and predicted values of Δk_T agree to within the experimental values.

37. A comparison of Tables 10 and 11 shows that the values of Δk_T predicted by WIMS and METHUSELAH are in fair agreement, the mean discrepancy being $0{\cdot}33 \pm 0{\cdot}37\%$. Individual values of k-infinity at a given temperature, however, show larger discrepancies up to a maximum of $1{\cdot}4\%$. For UO_2 fuel, k-infinity (WIMS) is systematically lower than k-infinity (METHUSELAH) by an average of $0{\cdot}8\%$, while for the PuO_2/UO_2 fuel with low Pu-240 content the trend is in the opposite direction, k-infinity (WIMS) being the higher on average by $1{\cdot}0\%$. Similar trends were observed in the analysis of the uniform SGHW lattices in reference 1, where it was concluded that the trend with plutonium content could be attributed to the use of a low value (2·091) of η_0 for Pu-239 in the METHUSELAH II library. The library now in use incorporates revised U-235 data which reduce the discrepancy with WIMS predictions of k-infinity for UO_2 fuelled cores.

38. It is interesting to note that for the fuel containing a high proportion of Pu-240 the METHUSELAH and WIMS predictions of k-infinity are almost identical. This clearly indicates that differences in treating Pu-240 between the two codes are compensating for the above-mentioned effects relating to U-235 and Pu-239.

CONCLUSIONS

The single-channel hot loop has been used in an SGHW core in JUNO to measure variations of detailed reaction rates associated with an increase in coolant temperature from 23°C to 274°C. Three different fuels have been studied, ranging from enriched UO_2 to a PuO_2/UO_2 mixture containing 24% Pu-240. The results have been compared with theoretical calculations using the METHUSELAH and WIMS computer codes.

40. The measurements have shown that the small discrepancies observed between the measured and predicted cluster reaction rates in the cold condition are not significantly increased by the rise in temperature. The total reactivity effects of these discrepancies on the temperature defects are less than $0{\cdot}5\%$ for both methods of calculation and lie within the experimental uncertainties.

41. Temperature effects on cluster distributions of detailed reaction rates have also been compared with theory. The use of transport theory and the improved spectrum models available in WIMS lead to improved predictions for the hot condition.

ACKNOWLEDGEMENTS

The experimental work described in this Paper has been carried out by many members of Water Reactor Physics Division, and their help is gratefully acknowledged. Thanks are also due to Dr C. G. Campbell who was in overall charge of this work, for much helpful advice during the programme, and to Low Power Reactor Operations Group of General Operations and Technology Division, for their willing co-operation and efficient operation of the experimental plant.

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BOOK REVIEWS

Thorium fuel cycle

USA National Bureau of Standards, Springfield, 1968, 839 pp., $3

This book comprises a collection of 43 papers presented at the Second International Thorium Fuel Cycle Symposium at Gatlinburg, Tennessee in May 1966. The contents are in fact more comprehensive than the title suggests, since even aspects such as the operating experience of reactors using thorium fuel are covered. The papers are grouped as follows:

Thorium reactor concepts and physics (10 papers)
Thorium fuel-cycle economics (4 papers)
Thorium reactor fuel fabrication (7 papers)
Thorium fuel preparation and irradiation testing (7 papers)
Thorium fuel processing studies and demonstrations (8 papers)
Thorium-fuelled reactor operating experience (4 papers)
General (3 papers)

As may be seen from these section titles, most topical points of interest of thorium as a fuel are described, including work being carried out round the world. The general impression created is that thorium fuel technology is now sufficiently advanced for it to be compared with the uranium–plutonium fuel cycles currently in use. The quality of production of the book is high and at $3 it is very good value.

F.L.D.C.

Advanced Level Atomic Physics

G. M. MOSSOP

University of London Press, 1968, 30*s*.

At the present time university science faculties have difficulty in finding enough candidates of the right calibre. This is often attributed to the lack of good science teachers in schools, but the continuous stream of excellent textbooks from those who teach sixth forms suggests that it is in the junior forms where science teaching fails and that not enough potential scientists reach the upper part of a school.

The volume under review is the work of the head physics mistress at a girls' school which was sufficiently enlightened to give her a sabbatical term at Oxford to complete it. The text is a salutary reminder of the clarity with which topics of some sophistication may be explained to school pupils. Its combination of theoretical fact and experimental emphasis is a measure of the experience which the author brings to her task; its balance is calculated to encourage the average student, whilst the very good pupil may read beyond the formal syllabus about the structure of matter, high energy particles, electron waves, lasers and semiconductors.

These advanced topics form Part 4 of the book. The first three parts are devoted respectively to atoms and sub-atomic particles, the atom and its nucleus and the atom and its electrons. The division is both logical and successful in its presentation. The major topic of Part 1 is the electron, its properties as a particle and its practical function in electronic devices. Part 2 deals with radioactivity and nuclear physics and Part 3 with atomic structure, elementary quantum theory, spectroscopy and X-ray phenomena. Extensive subdivision of the subject matter together with ample use of illustrations allows a quick and informative reference by the reader to a particular topic but the sense of continuity is never lost. This is due to good editing, for texts as comprehensive as this can often make tedious reading.

Miss Mossop has written a very good book. It leaves the reviewer a little alarmed at the extent of the syllabus of what, after all, is only a part of one A-level subject.

H.J.P.

Low energy neutron physics

I. I. GUREVICH and L. V. TARASOV

North Holland Publishing Company, Amsterdam, 1968, 210*s*.

This work deals with those aspects of solid-state physics which can be investigated by the scattering of neutrons with energy less than 1 eV. It is a translation of the Russian original which appeared in 1965, and is divided into five parts. Part 1 occupies one third of the whole work and serves as an introduction to the subject. This part covers the fundamentals of neutron scattering in some detail and includes sections on coherent phenomena, inelastic and elastic scattering and the production of mono-energetic and polarized beams of neutrons. Part 2 deals with the nuclear properties of the neutron and describes its interaction with protons and electrons. Parts 3–5 describe the use of neutrons in the study of solid-state physics, and include sections on lattice vibrations, the dynamics of liquids including liquid helium, molecular dynamics, correlation functions and a short section on magnetic neutron diffraction. The whole work has been written with the experimentalist in mind, and the authors have obviously tried to present an up-to-date assessment of the current position in many branches of the subject. However, the value of this book is that it is a well designed, self-contained textbook rather than a mere review of the subject. In each section the relevant equations are derived from basic principles in a clear and simple manner, with the aid of many well-drawn and carefully thought out line diagrams and graphs, and there are also many references to scientific publications for further study. The publishers have further increased the usefulness of this work by including a comprehensive index to this edition.

The authors have made a good choice of topics to cover and have developed them well. My only general criticism is that the division of the book into five rather self-contained parts has led to a slightly discontinuous presentation of some of the work. This English edition is marred by many examples of lack of care and thought in translation and layout. In places the translation lapses into poor English, and on occasion what was clear in the Russian text becomes confused in the English. At some points, for example, the translator does not seem to be able to distinguish 'photon' from 'phonon'. Another disappointing feature is the way in which the references are arranged. Each of the five

parts into which the work is divided has its own list of references at the end of the book. Unless the reader happens to know which part of the book he is reading, he will be unable to discover which list contains the reference he is looking for. Also, the references have obviously not been checked, and most of the errors in the Russian edition appear in this English edition. Some of the names are obviously wrong and some references have the authors' initials omitted. The Russian editions of an author's work are often referred to when English translations have been available for many years. These demonstrations of lack of care are inexcusable, and it is particularly annoying that such a useful book should suffer in this way.

However, in spite of these unfortunate features, the book is well constructed and should prove very valuable to all experimentalists who use the scattering of neutrons as a tool of solid state physics.

B.C.H.

Correspondence

TRANSFER FUNCTION MEASUREMENTS

Sir,

We were very interested to read the article by Messrs Jeffers and Corpus on 'Transfer function measurements on the universities' research reactor at Risley' (*J. Brit. nucl. Energy Soc.*, 1968, **7** (2), 146–150). This is an excellent example of how it is possible with comparatively modest equipment to perform quite elegant experiments, provided these are planned with sufficient imagination. It hurt a little to read that the Authors were making a virtue out of a lack of the appropriate electronic equipment, but they clearly paid the price in terms of the hard work necessary to analyse their data.

Having had some experience in this field, we were rather surprised at the accuracy the Authors were claiming. We would be very interested to see an analysis of the errors to be expected from the experiment to support the $\pm 10\%$ accuracy quoted for both reactivity coefficients y_1 and y_2, since a brief look at the data and results appears to show some difficulties. Figs 1 and 2 of the Paper show that the responses H_1 and H_2 have the same variation in gain up to 2×10^{-1} rad/s and virtually the same phase up to 10^{-1} rad/s for the 5 cm/s coolant flow case used to derive the results. Fig. 7 shows that the reactor transfer function G_R at 200 kW coincides in magnitude with G_0 (for zero power) within measurement accuracy before reaching 2×10^{-1} rad/s and coincides in phase by 1 rad/s. Since the power coefficient H_R is formed by linear combination of H_1 and H_2, any distinction between y_1 and y_2 in the experiment must, it seems, rely on the difference in phase of H_1 and H_2 at between 10^{-1} and 1 rad/s. From the figures it seems reasonable to expect this distinction to come out best around 2×10^{-1} rad/s. At this frequency, $|G_R/G_0|$ is very near to unity and arg (G_R/G_0) is approximately $+10^\circ$ with an apparent experimental error of $\pm 1{\cdot}5^\circ$. But

$$\frac{G_R}{G_0} = 1 + H_rG_0$$

and the phase error range of $\pm 1{\cdot}5^\circ$ in 10° therefore appears as $\pm 15\%$ or ± 1 dB error in $|H_RG_0|$ and hence in $|H_R|$. This is reasonably consistent with the margin shown at 10^{-1} rad/s in Fig. 8. Similarly, the error in arg H_RG_0 is given approximately in radians by

$$\frac{\text{error in } |1+H_RG_0|}{|H_RG_0|}$$

and the $\pm 7^\circ$ shown in Fig. 8 is also reasonably consistent.

At 2×10^{-1} rad/s, H_1 and H_2 differ by just under 40° and, when weighted according to their coefficients, contribute roughly equally in magnitude. They can be added vectorially to give H_R. A rough sketch shows that an error region of H_R with $\pm 7^\circ$ on phase and $\pm 15\%$ on amplitude permits errors of very nearly 50% in y_1 or y_2.

At lower frequencies much larger 'exchanges' between y_1 and y_2 seem to be possible since vectors representing H_1 and H_2 tend to coincide in direction. At higher frequencies the magnitude of $|H_RG_0|$ diminishes with phase of G_R/G_0 and experimental error in estimates of H would appear to rise.

The coefficients obtained are, of course, the result of a statistical best-fit procedure and not just from results taken at one frequency. No details of this procedure are given in the Paper, but if 2×10^{-1} rad/s is near the position of best resolution between y_1 and y_2 it is not immediately obvious that the possible range of error is reduced to $\pm 10\%$ by the number of experimental points shown.

The Authors claim in their conclusions 'The results show that it is possible to obtain satisfactory results for these types of kinetic measurements with extremely simple apparatus, and indeed without the use of any electronic equipment'. Surely the Authors used an electronic computer for all their Fourier analyses and their best squares fit procedures. If they did all these on a hand-operated desk calculator, their perseverance is laudable.

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M. H. Butterfield
R. J. Cox

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Corrigendum

In the List of Members published in July 1968, the name **Victor, Rojkov** should read **Rojkov, Victor**

MEETINGS

Unless otherwise stated, meetings are at the Institution of Civil Engineers, Great George Street, London, SW1; 5 p.m. (tea) for 5.30 p.m.

1968

16 October
6.00 p.m.
Glasgow

JOINT BNES/IEE/I MECH E/ICE LECTURE*
Advances in gas-cooled reactor systems
by Dr Gordon Brown

16 October
7.00 p.m.
Leicester

JOINT BNES/I MECH E LECTURE
The design of sodium pumps for fast reactors
by L. Bowles

5 December
6.00 p.m.

BNES LECTURE
Future technology of gas-cooled nuclear power reactors
by R. D. Vaughan

1969

28 March
All day

BNES SYMPOSIUM
Safety and siting

30 April
5.15 p.m.
5.30 p.m.

BNES ANNUAL GENERAL MEETING
Meeting to be arranged

24–26 June

INTERNATIONAL CONFERENCE
Fast breeder reactor physics

10–11 July

BNES SYMPOSIUM
Model techniques for prestressed concrete pressure vessels

15 October
6.00 p.m.

SECOND COCKCROFT LECTURE
by Sir Stanley Brown

1970

8–9 April
Leicester

INTERNATIONAL SYMPOSIUM
Nuclear gas turbines

*** The attention of Members is drawn to this meeting, which will be held in Room 24 at the University of Strathclyde.**

MEETINGS OF INTEREST TO MEMBERS

1968

24 October **6.00 p.m.**	**I MECH E DISCUSSION*** Power station control—Is automation the preferred solution?
10–15 November **Washington, USA**	**INTERNATIONAL CONFERENCE†** American Nuclear Society in conjunction with AIF and AtomFair
3 December **All day** **Harwell**	**MATERIALS AND TESTING AT HARWELL‡**
5 December **All day**	**I MECH E SYMPOSIUM** Education and training of engineers in the nuclear industry*
12 December **All day**	**I MECH E SYMPOSIUM** Refuelling of gas-cooled reactors*
16–20 December	**ATOMIC SCATTERING RESONANCES‡**

1969

6 January **6.00 p.m.**	**I MECH E DISCUSSION*** Numerical estimation of reliability of engineering equipment
22 January **Harwell**	**MOISTURE MEASUREMENTS‡**
10–13 March **Idaho Falls**	**AMERICAN NUCLEAR SOCIETY** Hot laboratory participation in fast reactor programme†
26–28 March **Cambridge**	**CONFERENCE** Elementary particles‡
31 March–3 April **Manchester**	**NATIONAL ATOMIC AND MOLECULAR PHYSICS CONGRESS‡**

*** For further details apply to the Institution of Mechanical Engineers, 1 Birdcage Walk, London, SW1.**

† For further details apply to the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521, USA

‡ For further details apply to the Meetings Officer, The Institute of Physics and the Physical Society, 47 Belgrave Square, London, SW1.

Summary of forthcoming lecture

Future technology of the gas-cooled nuclear power reactor

R. D. Vaughan

The Paper is concerned with developments now current and planned in Europe and the USA for the further exploitation of the gas-cooled reactor. An attempt is made to assess which of these developments will be the most profitable and will therefore justify the most intensive effort over the next few years.

Work at the moment is concentrated on three fundamental items: first coated-particle fuel, which opens the way for reactors generating gas at temperatures of 800°C and above, and secondly, the graphite core, now being designed for complete replacement, along with the fuel, several times during the life of the reactor. The solution to this particular problem will have an important effect on the layout of the whole reactor unit.

The third item currently receiving close scrutiny is the thermodynamic cycle. Cycles using gas turbines can offer very high thermal efficiency or very compact layouts. Much work is needed, however, on the design and development of heat exchangers and control systems before large units of this type will be built.

This lecture will be given on 5 December 1968 at the Institution of Civil Engineers.

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