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AUSTRALIAN ATOMIC ENERGY COMMISSION

RESEARCH ESTABLISHMENT

LUCAS HEIGHTS RESEARCH LABORATORIES

PROGRESS REPORT

FOR

NUCLEAR TECHNOLOGY DIVISION

SEPTEMBER 1980 - JUNE 1982

Chief: G.W.K. Ford

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FOREWORD

This report covers the work of Nuclear Technology Division, from its establishment in September 1980 to the end of June 1982. The Division was one of two new research divisions formed by amalgamation of the former Engineering Research, Physics, and Instrumentation and Control Divisions. Its broad objective was to investigate appropriate technologies for the conversion of the energy available from natural resources into forms which meet the energy needs of mankind. More specifically, the Division was to model and analyse the behaviour and performance of nuclear fission and solar energy devices, and to carry out selected experimental and theoretical scientific/engineering research and development activities aimed at the improvement of such systems.

Soon after the formation of the Division, the Commonwealth Government directed that work within AAEC not related specifically to nuclear activities should be transferred to CSIRO. This decision substantially reduced the staff of the Division, necessitating a complete review of its proposed activities.

In consequence, major areas of research and development in the Division have been reduced to two:

- Fission Technology: This includes neutronics and fluid heat transfer. Although primarily dedicated to the support of HIFAR reactor operation, a sound fundamental research component and the successful participation in selected international problem studies are essential to developing and maintaining state-of-the-art expertise.

- Fusion Technology: A small experimental program has been established to investigate the neutronics of fusion reactor blanket configurations. In parallel, the methods and data libraries developed for fission reactor core analysis have been modified and extended to allow their effective use in fusion reactor blanket problems. Comparisons of initial test results against a limited range of published experimental and calculated data have been encouraging. It is anticipated that eventually some effort will be directed to heat transfer problems in fusion reactors.

Some sections of this report refer to projects begun before the change of program. Others reflect the Division's first stages in the redirection of existing knowledge and skills to new activities, and give assurance that achievements for new programs will match those of previous programs, after a reasonable period of adjustment.

CONTENTS

1. FISSION TECHNOLOGY

1.1 HIFAR Physics

1.1.1 Computational models for HIFAR neutronics
(B. Harrington)

A standard set of calculational HIFAR models has been developed to meet the recurring need for reactor physics calculations on HIFAR. These models involve procedures for calculating quantities of interest for reactor operation, safety studies, etc.; input to codes is in the form of card image data sets on disk.

HIFAR models used earlier produced an almost flat flux across the core and very high excess reactivities, hence they could not be used with any degree of confidence. Extensive investigations showed that there was no single explanation for the inadequacies of these models but that, for satisfactory results, it was necessary to represent in some detail many features of the reactor, including all the reflector D_2O facilities.

The existence of both the UNED interactive editor [1] and the AUS modular scheme for reactor calculations [2] has made it feasible to have these detailed HIFAR models readily available 'at call'. The models are based on HIFAR operation program 251 (OP251) which was typical of the low-rig-burden HIFAR operations of 1978-79.

The following standard models have been developed and are available on the UNED data set BVH.HIFAR as the members:

(a) BVHIFXS - This is a five-group smeared cross-section preparation model. Fuel cross sections are produced as a function of burn-up for use in the models BVHIFRZ, BVHFXYA2 and BVHFXYE2.

(b) BVHIFRZ - This is a cylindrical diffusion HIFAR model. Region- and energy-dependent axial bucklings are produced for use in the models BVHFXYA2 and BVHFXYE2.

1. Cawley, R. AAEC/E580 [in press]

2. Robinson, G.S. [1975] - AAEC/E369

(c) BVHFXYA2 - This is a two-dimensional XY geometry, diffusion model of HIFAR, with average end-of-OP251 burn-up throughout the core.

(d) BVHFXYE2 - This is a two-dimensional XY geometry, diffusion model of HIFAR with the explicit end-of-OP251 fuel loadings for each fuel element.

(e) BVHIFCEL - This is an 'average' fuel cell calculation. The composition of the fuel is that of average fuel at the end-of-OP251 and includes fission products.

Representations of horizontal and vertical heavy-water facilities, voided safety-rod thimbles and rigs have been included in the models BVHFXYA2 and BVHFXYE2. The rig representation (accounting for 4.23 per cent in reactivity) was tuned to direct reactivity measurements, but all other reflector detail (accounting for 3.52 per cent in reactivity) was calculated. The calculated end-of-program reactivities of 0.09 per cent and 0.69 per cent for the models BVHFXYA2 and BVHFXYE2, respectively, compare well with the shut-down reactivity of 1.43 per cent observed in OP251. Limited comparisons with experimental fluxes indicate agreement to within 10 per cent. Even though these models have been developed for a specific operation program, they should be useful, with little or no change, for any current or future operating program.

1.1.2 Reduced enrichment studies
(G.S. Robinson)

Several studies related to the neutronic performance of HIFAR on lower enrichment fuel have been completed, and the AAEC has participated in the preparation of an IAEA guidebook on the conversion of D_2O research reactors to use low enrichment uranium.

A methodical benchmark calculation, based on a DIDO class reactor, is included in the IAEA guidebook. Comparison of preliminary results from various countries showed wide variation. The AAEC's results, based on the methods and models described in Section 1.1.1, have remained unchanged and satisfactory agreement has finally been achieved, largely because other laboratories adopted similar methods. Results for the key quantity, Δk_{eff} (the difference in the effective multiplication factor, k_{eff}, between 93 and 20 per cent enrichment) are:

	AAEC	Japan	Denmark	ANL	AERE
Δk_{eff}	0.0174	0.0219	0.0163	0.0207	0.0080

Neutronic calculations of the effect of lower enrichment fuel on HIFAR have also been submitted for inclusion in the guidebook. These calculations compare the performance of HIFAR, using 45 per cent and 20 per cent enrichment, with the current 80 per cent enrichment. Results encompass the required ^{235}U loading to match the cycle length, and the changes in neutron fluxes and safety-related parameters. The general conclusions, from the viewpoint of reactor physics considerations, were that 45 per cent enrichment would present no great difficulty, but 20 per cent enrichment (for which irradiation performance, etc. and commercial fabrication of fuel of the required uranium density have yet to be demonstrated) will require about 10 per cent increase in ^{235}U loading, with a significant degradation in performance through a reduction of about 15 per cent in the thermal neutron flux in the core irradiation positions.

1.1.3 Consequences of a loss of coarse control arm accident
(J.W. Connolly)

Reactivity control in DIDO class reactors is obtained by the operation of six 'signal arm' type, neutron absorbing blades. The angular position of these coarse control arms (CCAs) is varied by linear motion of a connecting rod between the drive mechanism and a point on the blade, thus inducing a torque about the pivot point of the blade. The position of maximum reactivity control of the CCA bank is at 34° from the vertical, and the normal range of operation is about 44° to 64°.

A consequence of this design is that the system is not fail-safe; if a connecting rod fractured, the blade would swing from its operational position to the vertical where its reactivity control would be almost zero. Such a loss of a CCA is the most demanding reactivity-addition accident which the protective system of DIDO class reactors must be designed to terminate safely. Protection against the consequences of connecting rod failure is provided, first by a halving time trip (since the initial movement of the affected arm decreases core reactivity), second by a doubling time trip, and third by an excess power level trip.

The reactivity gained by the core on the loss of a CCA blade depends on the reactivity controlled by the complete bank at the angular position when the blade fails. Analyses have shown that, unless this reactivity control is small, i.e. failure occurs at high critical angles, the excess power level trip cannot prevent core damage. Furthermore, under conditions of very low reactor power (low neutron source strength) and low critical angle, the halving time trip is too slow to arrest the transient that follows loss of a CCA blade. Administrative control of core reactivity and neutron source strength is thus relied on to prohibit operation with values of these parameters which would make the core vulnerable to a loss of CCA accident.

However, the above analyses have neglected the negative reactivity feedback response of the reactor to the energy released in the core during a transient. An extensive series of experiments in the USA during the 1960s, known as the SPERT program, clearly demonstrated that reactors can safely terminate excursions produced by deliberate reactivity additions above prompt critical. One such reactor investigated, SPERT BD22/24, had a core which was a close replica of the DIDO type and was demonstrated to withstand safely a reactivity injection of 0.02 ($\delta k/k$).

It was apparent that the ability to calculate the course of transients measured in BD22/24 would engender great confidence in the application of the same methods to postulated HIFAR reactivity accidents. Elucidation of the neutronics of core BD22/24 using the AUS code proved difficult, largely due to the lack of precise information on the core composition. Details which were unimportant in earlier studies on SPERT light water cores assumed considerable significance in the D_2O-moderated BD22/24.

These problems were, however, solved and reactivity feedback coefficients were obtained and used in conjunction with the previously developed nucleate boiling heat transfer model and a new representation of steam void formation. This allowed calculation of all the BD22/24 transients to reproduce the experimental results with considerable fidelity.

The same methods have now been used to study the consequences of the loss of a CCA blade in HIFAR, taking into account the angular dynamics of the broken arm and the manner in which it produces reactivity changes in the core. These reactivity changes were used in the reactor transient code ZAPP, together with reactivity feedback coefficients obtained from the AUS suite of neutronics codes, to calculate the time-dependent power following the breaking of a CCA blade. This power history was then converted to the corresponding reactor ion chamber current history by analogue computer tehniques and this was fed into actual reactor period meters and shutdown power amplifiers to obtain the times at which these units would open the reactor protection system guard-lines.

The results of these calculations show that for any initial condition within the limits permitted for HIFAR operational parameters, the self-shutdown characteristics of the reactor would safely terminate the initial power burst following loss of a central CCA blade, thus allowing a less demanding time response from the reactor protection system. It was also found that, contrary to prior belief, the shutdown amplifiers would produce a reactor trip before the period meters.

Power and temperature histories calculated for a particular set of conditions at the time of CCA failure are shown in Figures 1 and 2.

1.2 Verification of Reactor Codes

1.2.1 Group cross-section library
(G.S. Robinson)

The data for moderators and fission products in the main AUS library have been replaced by ENDF/B IV data processed using the AAEC version of the ORNL code XLACS2. With this code, temperature-dependent thermal data for the three important moderators (H_2O, D_2O and graphite) which joined smoothly on to the epithermal data were obtained. A copy of the modified XLACS2 code has been sent to Oak Ridge National Laboratory.

Cross-section data were generated for 155 individual fission products before reduction to 'pseudo-fission-products' schemes. Two such schemes were developed. For thermal reactors, a scheme with 45 individual nuclides and one pseudo-nuclide was adopted, but for fast reactors one pseudo-nuclide sufficed to represent reactivity effects.

The pseudo-nuclide group cross sections were formed from individual nuclide data weighted by typical atom densities obtained from simple pressurised water and liquid metal-cooled fast breeder reactor (PWR, LMFBR) models. The pseudo-fission-product yields from each fissile nuclide were adjusted individually to match the capture rates obtained with the full fission-product set. The resulting schemes differed from the full set by less than 2 per cent in total fission-product captures for reasonable ranges of

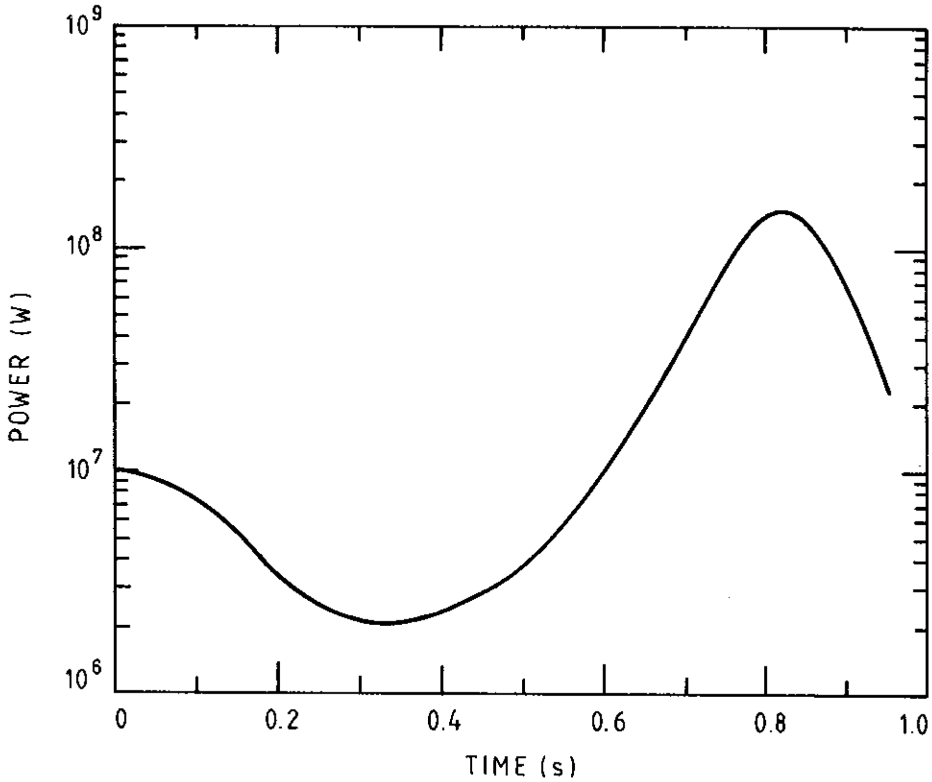

FIGURE 1. POWER TRANSIENT FOLLOWING FRACTURE OF CONNECTING ROD OF CENTRAL CONTROL ARM FOR CRITICAL ANGLE = 12° AND REACTOR POWER = 10 MW

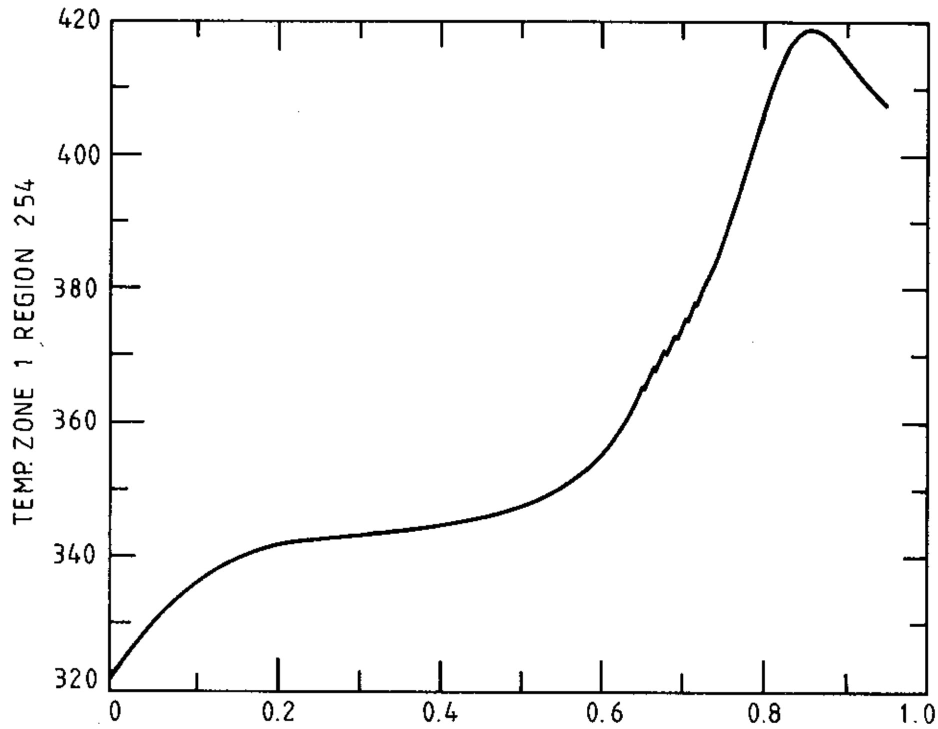

FIGURE 2. AVERAGE FUEL TUBE TEMPERATURE TRANSIENT FOLLOWING FRACTURE OF CONNECTING ROD OF CENTRAL CONTROL ARM FOR CRITICAL ANGLE = 12° AND REACTOR POWER = 10 MW

time, power density and nuclide composition.

1.2.2 OECD-NEA/CRP benchmark problem on LMFBR burn-up
(G.S. Robinson)

A solution to the OECD-NEA's Committee on Reactor Physics (OECD-NEA/CRP) benchmark problem on LMFBR burn-up was submitted. This benchmark problem involved calculation of changes of reactivity, breeding ratio, power distribution and void coefficient which occur in a large LMFBR during the first year of operation, and followed on from an earlier comparison for the initial core*. The calculations are for a precise specification of an RZ diffusion theory model of the reactor, so that differences between solutions are due to cross-section data and group data generation methods only. The AAEC calculations used ENDF/B IV data within the AUS code system.

The AAEC did not participate in the earlier exercise but, in the following table, AAEC results for the initial core are compared with those published:

Parameter	AAEC ENDF/B IV	Average of All Participants	'Best' Method using ENDF/B IV
k_{eff}	0.9943	1.0051 ± 0.0129	0.9930
Breeding ratio	1.403	1.392 ± 0.048	1.395
Sodium-void of inner core ($\delta k/k$)	0.0230	0.0212 ± 0.0025	0.0236
Central control rod worth ($\delta k/k$)	-0.0036	-0.0036 ± 0.0005	-0.0034

Preliminary results for some of the major burn-up parameters are compared in the table below. It can be seen that the AAEC results are in good agreement with the mean of all submitted results. The final results for this benchmark problem were presented and discussed at the OECD-NEA/CRP specialists' meeting in April 1982.

* Proceedings of OECD-NEA/CRP Specialists' Meeting on the International Comparison Calculation of a Large Sodium-cooled Fast Reactor, Argonne National Laboratory, USA, February 1978.

Parameter	Mean ± s.d.	AAEC
Total reactivity loss per cycle	0.0115 ± 0.0050	0.0054
Reactivity loss due to fission products	0.0183 ± 0.0030	0.0169
Breeding ratio at end of cycle	1.330 ± 0.014	1.339
Sodium-void worth at end of cycle	0.0286 ± 0.0024	0.0284

1.2.3 OECD-NEA/CRP shielding benchmark problem
(B.J. McGregor)

A calculation was submitted to the OECD-NEA/CRP shielding benchmark exercise on a PWR shield. The comparison of results from workers at a number of laboratories is valuable, both as a check on our ability to calculate such a shield, and also as a test of our recently produced 200 neutron-37 gamma-ray group cross-section library to be used for fusion neutronics work.

Results were submitted for the ^{54}Fe activation in the core barrel and pressure vessel cladding, the neutron damage production rate in the pressure vessel and the neutron and gamma-ray doses at the outside of the 195 cm thick concrete shield.

Results were obtained from six laboratories. Our neutron results were well within the range of values found by other workers. The gamma-ray dose at the outside of our shield was higher by 15 per cent than the next highest value. Overall, the agreement with workers from other laboratories is most encouraging.

An important outcome of this work was the realisation that the use of the old 128-group library over-estimated the fast neutron dose at the outside of a thick concrete shield, especially in the important 1-3 MeV energy range.

Another data check was performed on the ENDF200G library by calculating the neutron and secondary gamma-ray dose due to a neutron source at normal incidence to slabs of concrete 100 and 200 cm thick for both a fission and fusion neutron source. The calculations were in good agreement with previous results.

1.2.4 OECD-NEA/CSNI containment safety studies
(J. Marshall, P.G. Holland, W.B. Woodman)

The AAEC participated in containment analysis standard problem no.2 (CASP2) organised by the OECD-NEA's Committee on the Safety of Nuclear Installations (OECD-NEA/CSNI). An experiment was again performed on the large containment model at the Battelle-Institut, Frankfurt, but with an arrangement of the six compartments which was different from that in CASP1. The participants had to calculate the pressure and temperature transient distribution throughout the system; these calculations were then compared with each other and with the experimental results.

As previously, the AAEC used the computer code ZOCO V for the main study but also included a partial analysis using ZOCO VI, which had recently been received and commissioned. The most pertinent difference between these is in the calculation of heat transfer coefficient. ZOCO V appears to give the better result. A report on the results was distributed to all participants and the data sent to the 'lead' organisation GRS, Garching, Federal Republic of Germany. The AAEC was not represented at the workshop meeting at which the overall study was summarised and discussed. The AAEC analysis provided a reasonable representation of the measured transients, the pressure peak calculated for the blowdown receiving compartment being about 5 per cent higher than the peak measured (see Figure 3).

In 1980, all CASP participants were requested to report on the containment studies in their respective countries. As a consequence of this, the AAEC was further requested to submit details of the next proposed experiment for consideration as CASP3. On the basis of this information, the AAEC was then invited to be the lead organisation for CASP3, to perform the experiment, report on the results and then carry out the comparison of participants' analyses. This was agreed and the blowdown/containment rig was modified to improve the measurement of the blowdown flow and to divide the containment vessel into two compartments having an interconnecting flow path with flow measurement. The experiment was completed and a report on the results sent out by the end of 1981. The comparison study is in progress; six countries are expected to provide analyses.

The AAEC is also contributing an analysis of CASP3 using the code ZOCO V. It ws necessary to amend the code to permit reasonable representation of the experimental results. The method used for estimating the heat transfer

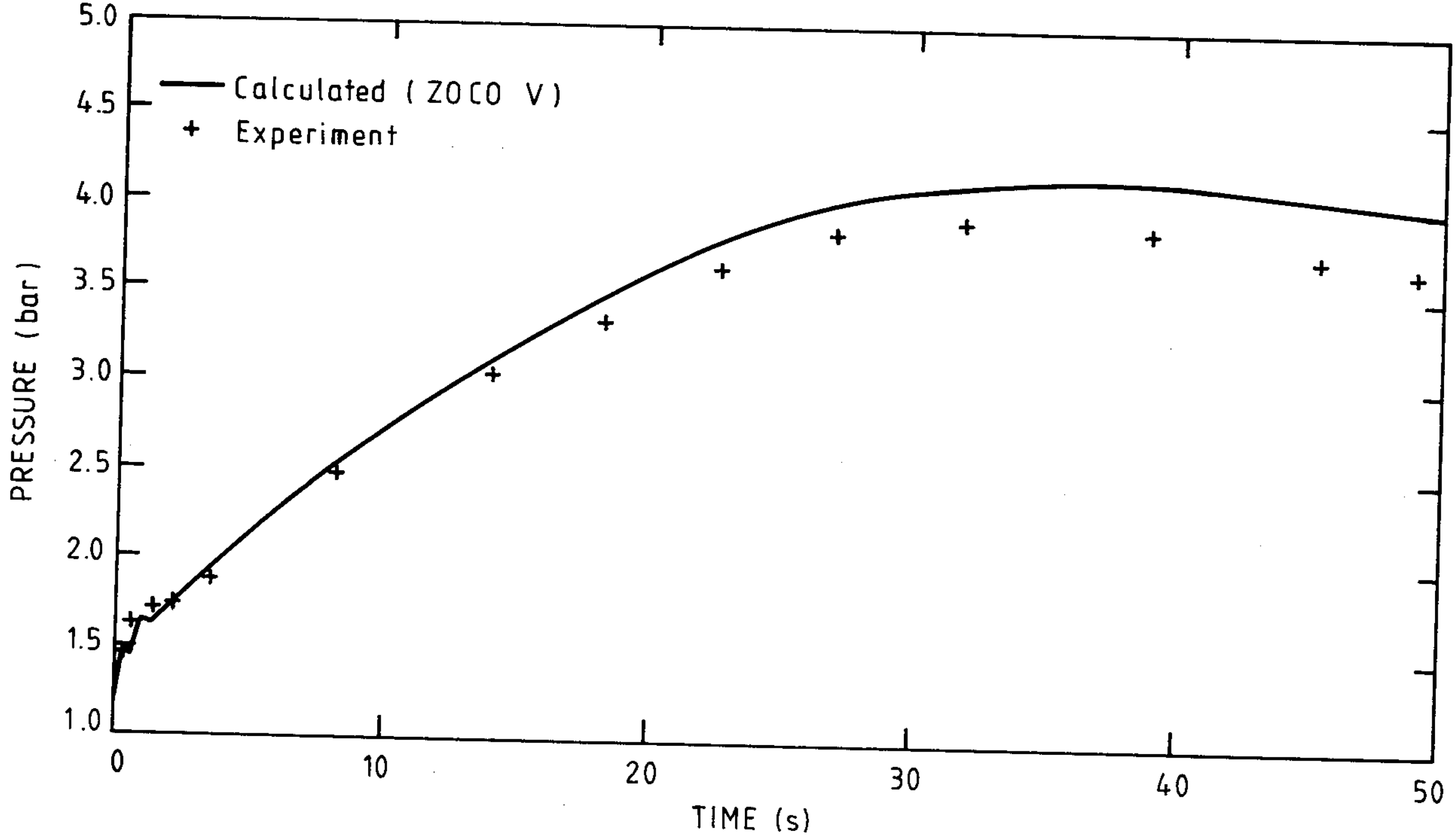

FIGURE 3. PRESSURE HISTORY: COMPARTMENT 4 IN CASP2

coefficient was inadequate for this situation, where heat transfer is a very important part of the pressure response because of the relatively small size of the compartments and the presence of steel rather than concrete walls. Based on the previous work, a method for evaluating heat transfer coefficient has been devised, containing functions of inflow power and the air-to-steam ratio in the compartment, which has greatly improved the analysis. A further modification, which enabled the sweeping of air out of the chamber ahead of the inflowing steam to be modelled, has given quite close agreement with the measured pressure transients. A report on this is being distributed to all members of the working group.

1.3 Reactor Thermohydraulics

1.3.1 Subchannel coolant flow in reactor fuel rod assemblies
(W.J. Green, J.D. Hooper, W.J. Crawford)

Further experimental studies of turbulence in developed single-phase flow through a square-pitch rod array, in which the rod pitch/diameter ratio was 1.107, were made at four Reynolds numbers: 22.6×10^3, 46.3×10^3, 133×10^3 and 207.6×10^3. For the three highest Reynolds numbers, there was little effect of Reynolds number on the scaling of the wall shear stress, on the mean velocity distribution, or on the Reynolds stresses. However, for the lowest Reynolds number, (i) there was a slight decrease in the magnitude of the Reynolds stresses, normalised to the local wall friction velocity, and (ii) mean velocity profiles were slightly below the logarithmic law of the wall distribution, indicating that the Patel calibration of the Preston tube used to measure the wall shear stress could be in error.

With a plate covering one half of the test-section flow area to create a complete blockage of one sub-channel in the two-sub-channel test section, flow recovery was investigated in the blocked sub-channel. The aims of this experiment were to measure the structure of the recovering flow downstream of the blockage, and to determine which parameters could be useful in diagnosing the presence of upstream blockage. Measurements were made $\sim$ 3.9,, 5.0 and 7.0 m (91 diameters) downstream of the blockage.

All Reynolds stress terms, wall shear stress distribitions, mean velocity profiles and secondary velocity components were measured in the unblocked and blocked sub-channels. High turbulence prevented reliable measurements closer than $\sim$ 4 m from the blockage.

Measurements of static pressure in the axial and azimuthal planes showed that the pressure distribution recovered relatively quickly. The mean velocity distribution was scaled in each case by the law of the wall (using the Patel constants). This was not expected, since (i) turbulence intensity was still high 3.9 m (50 diameters) from the blockage, and (ii) the flow structure was markedly different from that in Patel's calibration studies. At the 50-diameters position, the azimuthal turbulent mixing in the gap between rods was about twice that for the unperturbed flow. Some evidence of skew-induced secondary flows was provided by a rotated inclined hot-wire anemometer probe. This contrasts with the case for developed flow, which showed no evidence of secondary flow for the whole Reynolds number range.

Wall shear-stress distribution was the most sensitive indication of the blockage. At the maximum downstream position (91 diameters), Reynolds stresses and turbulent intensities reached levels typical of unperturbed flow, but the wall shear stress still showed marked asymmetry.

1.3.2 Non-equilibrium transient thermohydraulics
(A.W. Dalton)

Development of a computer code, for the calculation of power transients in a water-cooled nuclear reactor, was begun in 1979 and culminated in the code NAIADQ. This code was developed from the code, NAIAD, which is based on a coupled neutron-kinetics/hydrodynamics/heat transfer model, with point kinetics and one-dimensional thermohydraulics equations. Particular attention has been given to heat transfer from the fuel during rapid power transients. The model includes rapid propagation of a superheated temperature point into the coolant, and the assumption that vapour is generated within the growing superheated layer at a non-equilibrium rate.

Calculations from this code have been compared with available information on transients in experiments on the US SPERT II facility. These transients covered a wide range of conditions of pressure, flow and heat transfer. The code was able to provide good estimates of these transients and appears to be a promising means of investigating thermodynamic non-equilibrium in two-phase flow transients.

1.3.3 HIFAR emergency core cooling system
(J.W. Connolly, J.R.T. Rodd, V.W. Cornford)

A detailed assessment has been completed of the experimental work performed by J. Wolters (Julich) on the emergency core cooling system (ECCS) developed for the DIDO class FRJ-2 reactor. Basically, the same system is used for the ECCS of HIFAR. An independent assessment of Wolters' work and the effect of design differences between the ECCSs of HIFAR and FRJ-2 was required for a submission to the Regulatory Bureau.

As designed, the ECCS collects D_2O (lost from the primary circuit in the event of a circuit rupture) by means of a sump in the plant room floor. This D_2O is returned to the reactor aluminium tank by two pumps. The pumping capacity of each pump is sufficient to maintain the water level in the reactor tank above the arrays of holes in the fuel element shrouds. (A shroud is an unheated flow-tube surrounding an assembly of fuel tubes.) During emergency core cooling, jets of water from these holes strike the upper edges of the fuel tubes and form a cooling water film falling down the outer surfaces of the tubes.

The aim of this assessment was to identify factors which could impair or negate the heat removal capabilities of the system. It was possible to identify the coolant channel at greatest risk and provide quantitative values for the following critical factors:

(i) The minimum water level required in the reactor tank for the system to function.

(ii) The maximum allowable wave height on the free surface of the water.

(iii) The shutdown power to be removed from HIFAR as a function of leak rate.

(iv) The time delay before activation of the ECCS as a function of leak rate.

(v) The effect of departure from film flow of coolant falling over the fuel tube surfaces.

The major design weakness identified in the HIFAR ECCS was the method of delivering scavenged D_2O back to the reactor tank. Since the D_2O falls from above the free surface of the water, surface waves are generated. If only one pump were operational there is a definite probability that the height of these waves would exceed a critical value and that the emergency core coolant flow would be inadequate. With two pumps in operation and a higher water level, a greater wave height could be tolerated. We believe that the falling return flow must not generate waves of amplitude exceeding the critical value. However, to depend upon the availability of two pumps negates the redundancy principle intended in the present design. It may, therefore, be necessary either to modify the method of returning D_2O to the reactor tank (e.g. return below the surface) or to provide extra pumps.

1.3.4 Maximum permissible heat flux in HIFAR irradiation rigs
(A.G. Chapman, N.D. Hargreaves)

In the production of molybdenum-99 by the irradiation of uranium oxide in HIFAR, estimated heat fluxes at the surfaces of the irradiation cans exceed those permitted by the HIFAR Safety Document and Operating Manual. There is, however, no factual basis for the limits set. At the request of HIFAR Operations Section, the safe limit of the rate of heat generation in an irradiation rig was investigated.

The FLEX computer program was used to calculate that the existing burnout and flow stability margins for coolant flow through the reactor fuel element surrounding an irradiation rig were not affected by heat generation in the rig at the maximum contemplated heat generation rate. Calculated estimates of the rate and distribution of coolant flow around the cans were used with an empirical burnout correlation to predict the power level at which burnout would occur at the can surface. We then assessed the degree of uncertainty in the calculations, applied appropriate safety margins, and made a recommendation of maximum permissible heat generation rates, according to the number of cans loaded. Because of the complex form of the coolant flow passage, it is recommended that these estimates be regarded as provisional, until experimental work has been completed to measure burnout powers in a simulated irradiation rig.

This experimental work is in progress. The proportion of the reactor channel coolant flow diverted to a hollow fuel element irradiation rig with a perforated liner has been determined in a dummy fuel element assembly mounted in a water flow rig. Electrically-heated simulations of an irradiation can have been designed and manufactured, and burnout experiments with heated cans are proceeding. So far, the 'burnouts' that have been produced have been abnormal, the point of burnout being substantially upstream of the coolant outlet end of the can, indicating that flow disturbances are having an important influence, and casting doubt on the predicted can burnout powers.

1.3.5 Critical heat flux and post-dryout heat transfer
(W.J. Green, J.R. Stevens, D. Wassink)

The pressurised Freon-12 facility, was used to obtain experimental data on the thermal transient characteristics of uniformly heated tubes subjected to fast electrical power ramp inputs and internally cooled at low flowrates by Freon-12 in a two-phase state.

Analysis of the experimental data with the transient heat transfer code THETRAN has shown that the heat transfer concepts developed for uniformly heated tubes are also applicable to non-uniform heating conditions

An investigation is being made of heat transfer processes under slow dryout and rewetting conditions using high thermal capacity test sections.

Because of the sensitivity which has been found in post-dryout heat transfer analysis to the predicted value of the surface heat flux at which dryout will initiate, and since available critical heat flux (CHF) correlations are not reliable for Freon-12 over the range of conditions of interest, experiments have been performed to assist in the development of a CHF correlation which is more accurate than those currently available. Extensive overseas data have also been examined and utilised in the development of a general CHF correlation which is suitable for coolants in uniformly heated tubes.

A dimensional analysis approach was used with attention being paid to the concept that dimensionless groups may be interrelated, not only in a simple product relationship, but also as power functions of one another. Only a relatively small number of experimental Freon-12 data (a few hundred) were used to develop the correlation but, when tested extensively against many other Freon-12 data, it was accurate over a wide range of coolant conditions. With only very minor modifications, the correlation also compared well with over 7000 sets of water data. Excellent agreement between calculated and experimental CHF values was found for nearly all the water data in the pressure range 3.4 to 18 MPa. At higher pressures, agreement was less good. This was attributed to the surface tension and latent heat approaching zero as the critical pressure is approached; we therefore emphasise the need for precise knowledge of the local coolant conditions at high pressures. The correlation has also been favourably tested against nitrogen data demonstrating that it has a considerable degree of generality.

Very low flow CHF data (mass fluxes less than 300 kg s^{-1} m^{-2}) were used to indicate the presence of two distinct flow regimes analogous to single-phase flow, and to show that although the proposed CHF correlation is applicable to the vast majority of data, a different correlation is required for very low flow rates. The same analytical techniques have been applied to develop such a correlation, and to determine a means of delineating between the high and low flow regimes.

1.3.6 Two-phase flow characteristics
(D.R.H. Beattie)

In this project, established single-phase flow turbulence concepts are being extended to two-phase flows. Major developments during this period were as follows:

(a) Turbulence stability concepts were used to derive a criterion for the liquid film thickness of annular flows.

(b) The hydrodynamics of nucleating bubbles in boiling flow were examined in terms of shear and surface tension acting on bubbles. These concepts were extended to predict wall shear and average void fraction for boiling flows.

(c) Some 'anomalous' features of heat transfer crisis characteristics, namely, upstream crises and 'limiting quality' crises, were explained in terms of discontinuities in crisis characteristics caused by flow regime transition. A consequence of the explanation is that the flow qualities at the onset of 'anomalous' crises can be predicted from shear stress characteristics.

(d) For flow conditions where equations previously developed in the project were not supported by data, these concepts were applied to the annular flow situation in which the liquid film is confined to the viscous-dominated region near the wall.

(e) The two-phase concepts described above were applied to single-phase flows in roughened tubes. Although the analysis led to some conclusions which contradict accepted views on the influence of roughness, it provided a theoretical basis for the hitherto purely empirical 'Colebrook' equation for the characteristics of roughened tubes.

(f) A method of predicting wall shear in two-phase flow was developed, based on a homogeneous model in which the 'homogeneous' viscosity is replaced by a simple expression which approximates bubbly viscosity at low gas contents and annular flow viscosity at high gas contents. The method thus partially allows for flow pattern effects.

1.3.7 Neutron methods for two-phase flow measurements
(J. Marshall, H.J. Woodley, R.J. Blevins)

Work on neutron techniques for the measurement of void patterns in two-phase flow continued until mid-1981 when it was stopped because the effort was required for other work. Flow patterns in which polythene sections simulated the liquid water phase and air the vapour phase were used. The type of flow pattern could be selected, e.g. annular, stratified, or core (inverted annular) flow.

A suitable shielding and collimating container was designed and manufactured for tests on steam/water flows in a 5 cm bore pipe connected to the HIPPOCRENE rig.

1.4 Other Topics

1.4.1 Reliability studies
(E.R. Corran, H.H. Witt)

Reliability analysis work at AAEC falls into two categories - development of techniques and analysis of actual systems. In the current period, most of the available effort has been directed to a reliability analysis of the HIFAR ECCS. In 1980, a preliminary analysis had examined three possible ECCS configurations which were under consideration, and found that one had major reliability advantages. This finding influenced the subsequent design concept.

An extended analysis has been completed of the ECCS design submitted to the Regulatory Bureau for second-stage approval. Unlike the preliminary analysis, which examined only the parts of the ECCS which were the subject of modification, the extended analysis examined all parts of the ECCS barrier, such as the primary and secondary coolant loops and the power supplies, and considered possible human error in operation and maintenance, as well as hardware failures. Very careful evaluation was made of the data used and, wherever appropriate, HIFAR records were used as the basis for analysis.

The extended analysis revealed that the final design had potential shortcomings, which were eliminated from the design before construction. The extended analysis shows that the final design complies with the reliability goal required by the Regulatory Bureau.

An important aspect of study is the development of simple techniques for reliability analysis, suitable for use by non-specialist staff. As a general principle, the most suitable staff to carry out a reliability analysis are the engineers and senior technicians within the design team. They will be far more conversant with the system components than a lately arrived reliability specialist can hope to be. The most fruitful method of reliability analysis, therefore, would be for the specialist to provide suitable training for members of the design team, for those members to perform the initial analysis, and for the specialist to review their findings and advise on areas likely to cause problems. However, for this approach to succeed, the techniques used must be easily and quickly learnt and easy to apply. To this end, existing analysis techniques are reviewed, with the objective of selecting and developing methods that can be taught and applied simply. Although by no means complete, the techniques were used for part of the ECCS analysis.

1.4.2 Fundamental analysis of stresses in pipe bends
(J.F. Whatham)

A rigorous analytical treatment, using thin shell theory, was developed for calculating stresses in bends in circular pipes under in-plane and out-of-plane loading, and in non-circular pipes under presssurisation and in-plane bending. Derivation of exact solutions is of significant value for

(i) providing design data for flange-ended pipe bends under various loading conditions;

(ii) providing fundamental benchmark solutions for checking out numerical methods such as finite elements for practical situations; and

(iii) providing information on eigenvalues to assist (a) in developing approximate energy methods, and (b) in choosing finite-element patterns to reproduce essential features of the problem.

Following reports of this work in the literature, two overseas institutions requested relevant computer codes. The AAEC supplied a program package BENDPAC, containing all the codes, to the National Energy Software Center, Argonne; it is expected to be included in its list of available computer codes soon.

The programs were applied to the analysis of pipe bends in the HIFAR primary cooling circuit. The technique could also be applied to primary cooling circuits in power reactors.

2. FUSION TECHNOLOGY

2.1 Calculations for Magnetically Confined Plasmas

2.1.1 Antenna excitation of radiofrequency waves in plasmas

(B.E. Clancy, I.P. Donnelly (Applied Physics Division))

The excitation, propagation and damping of electromagnetic waves at frequencies around the plasma ion-cyclotron frequency are being investigated experimentally with the Sydney University TORTUS tokamak. The waves are excited by an oscillating current loop antenna surrounding the plasma. The motives for these experiments include the desire to provide efficient radiofrequency heating of plasmas in a fusion reactor regime. We therefore wish to be in a position to predict the energy deposition without experiment by simulating the experiments numerically.

To this end a set of computer programs is being developed to calculate the wave fields excited within a cylindrical vessel containing a cylindrical plasma and with a cylindrical conductor (to carry the current) placed at some position between the edge of the plasma and the perfectly conducting vessel walls. All fields and currents are taken to have a z-dependence given by the expression exp(ikz) for integral values of kR, where R is the major radius of the tokamak. The fields and currents thus found will then be terms in a Fourier series which will represent the situation in a toroidal vessel with a loop antenna. The dielectric tensor used at points in the plasma takes account of the toroidal magnetic field, the electron number density, temperature and collision frequency.

The numerical simulation in these circumstances reduces to a solution of Maxwell's electromagnetic field equations within the cylindrical vessel using appropriate boundary conditions at the vessel walls and at the current sheet. If θ is the poloidal variable, z the axial variable and t the time variable, the electric field and magnetic induction vectors have the forms

$$\underset{\sim}{E} \exp[i(kz + m\theta - wt)]$$
and
$$\underset{\sim}{B} \exp[i(kz + m\theta - wt)]$$

respectively, the two (complex) vectors having components in the r, θ, z directions. The Maxwell equations then reduce to four ordinary differential equations of the form

$$dE_\theta/dr = f_1[E_\theta, E_z, B_\theta, B_z, r, w]$$

$$dB_\theta/dr = f_2[E_\theta, E_z, B_\theta, B_z, r, w]$$

$$dE_z/dr = f_3[E_\theta, E_z, B_\theta, B_z, T, w]$$

$$dB_z/dr = f_4[E_\theta, E_z, B_\theta, B_z, T, w]$$

where f_1, f_2, f_3, f_4 are appropriate functions which depend linearly on their first four arguments. The components E_r, B_r are related to the other field components by simple analytic formulae.

Within the plasma, these differential equations admit of two linearly independent solutions which are finite at the origin; a computer program is being developed to find these solutions numerically as well as the simpler (analytic) solutions in the vacuum. Suitable matching of the boundary conditions then provides the required final solution.

For many frequencies and temperatures the numerical procedure is fraught with difficulties arising from the fact that one of the independent plasma solutions - probably associated with the 'electrostatic wave' - can exhibit a very rapid growth rate as the radial coordinate changes. This means that the two solutions developed numerically will not be truly independent and the solution method fails to produce sensible results.

A simpler (approximate) solution method has been developed which is not subject to these problems. In this method the electric field component E_z is taken as zero and two of the differential equations can then be ignored. The wave fields developed by this method (akin to those denoted as transverse electric waves in standard cavity resonator theory) produce results very similar to those developed from the complete solution where it applies - at least for frequencies above the ion-cyclotron frequency and for low frequencies at which the Alfvén wave resonance position is not in the plasma proper.

Work aimed at developing a satisfactory solution to the complete problem is continuing.

2.1.2 Numerical modelling of toroidal plasmas
(B.E. Clancy)

The particle and energy balance within magnetically confined toroidal plasmas is being studied by numerical simulation. A computer code SCORCH (for the AAEC's IBM3033S central computer) is being developed for this purpose.

The program calculates the losses and gains of energy and of particles, according to theoretical rules which describe the transport coefficients in the plasma (i.e. diffusion coefficients and thermal conductivities). Energy balance arguments then enable the code to calculate the rates at which ion and electron temperatures will change as a function of time. The plasma is assumed to be quiescent and the development of gross instabilities is ignored. The analysis is limited to the determination of volume-averaged electron and ion densities and temperatures. However, the analysis can be applied to specific geometrical shapes (e.g. toroidal) by assuming that the variation of plasma densities and temperatures with distance from the magnetic axis of the system can be described by simple algebraic functions, e.g. a parabolic distribution.

Collision and ionisation rates needed for the analysis can be determined from a comprehensive data library (compiled in collaboration with Applied Physics Division). This library can also provide data for line-radiation energy losses from partially ionised impurity ions which may be present in the plasma.

The original formulation of the work was based on neo-classical theory, but it is planned to modify the analysis to embody the empirical laws known as 'Alcator scaling'. These laws have been developed elsewhere from analysis and from the results of various tokamak experiments.

The code has been used to predict values for the various energy-loss terms in the Sydney University TORTUS machine during the plasma heating phase. The theoretical predictions will be compared with experimental findings when they become available.

The analysis has also been applied in a preliminary fashion to the Rotamak concept. This cannot be expected to be satisfactory until a version of the analysis can be developed which is not limited to magnetic field configurations appropriate to tokamak machines. This development is in hand.

The code is expected to have longer-term application to scoping studies of possible power reactor systems.

2.2 Fusion Blanket Neutronic Studies

2.2.1 Development of AUS modules and data library for fusion blanket calculations

(G.S. Robinson)

A new group cross-section library for fusion blankets has been prepared for use within the AUS code system. Previous AUS libraries have contained only neutron cross-section data but the new library, AUS.ENDF200G, also contains photon production and photon interaction data as well as kerma factor data for energy deposition calculations. The number of neutron groups has been increased from 128 to 200 by extending the energy range to 15.5 MeV and using lethargy divisions as fine as 1/32 at high energies. The number of photon groups is 37. The library currently contains the nuclides used in fusion blanket designs, plus some of the more important nuclides for general shielding calculations and fission reactor cores. Uranium isotope data are included but not plutonium. The fission and shielding data have been included to assist in data listing, but it is intended that this library alone will eventually be used for all neutronic calculations.

The library has been prepared from ENDF/B-IV using a number of locally written programs and programs obtained from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory. Treatment of neutron data has been previously described. The AMPX module SMUG was used for photon interaction data and the MACK IV code for photon production and neutron kerma data. The AMPX module LAPHNGAS, which has been used in a standard method of generating photon production data, was tried initially, but proved unsatisfactory for the weighting spectrum and the large number of thermal groups used. The use of MACK IV for photon production data enabled the generation of consistent photon production and neutron kerma data. A number of modifications to MACK IV were required. In particular, provision was made for various reactions to be kept separate to allow the output to be renormalised to XLACS cross sections, and to allow for resonance shielding.

The AUS cross-section data pool structure was generalised to allow the inclusion of photon data. Major modifications to the MIRANDA module were required to process these data. Because it is expected that other modules will work with cross-section data pools in which no distinction is made between neutron and photon data, they required little modification.

2.2.2 Fusion blanket neutronics experiments

(G.W.K. Ford, J. Marshall, A.W. Dalton, H.J. Woodley, W.J. Crawford, R.J. Blevins, C. Evans)

An essential part of a fusion power reactor is the surrounding blanket which provides shielding, a means for extracting heat from the system, and also the breeding of tritium through neutron interactions, e.g. with lithium. The last of these, tritium production, is the main topic of this project.

The fusion reactor blanket neutronics study became a substantial project of the Division in mid-1981 when it was agreed that the work should move to Building 53 (Critical Facility). Up to that time, the experimental work had been confined to commissioning an accelerator but, because of safety considerations, it could not be used to produce neutrons. Building 53 has a heavily shielded chamber which permits the neutron generator to be used with minimal safety restrictions.

In the subsequent period the following activities have been undertaken:

(a) the commissioning of a 14 MeV neutron generator;

(b) the acquisition, commissioning and calibration of neutronic measuring equipment;

(c) the selection of suitable techniques for measuring small amounts of tritium produced in a proposed blanket material; and

(d) fusion blanket experiments.

Neutron generator

A 14 MeV Activatron accelerator generating neutrons by a deuterium-tritium (D-T) reaction was available and considerable effort was spent in bringing it into operation. Various parts had to be replaced or modified. By the end of the period it could be operated at near to full rating in voltage and ion current. There are still problems in achieving the rated neutron output and these are being investigated.

Neutronics measuring equipment

The measurement of neutron flux is an essential part of most fusion neutronics experiments and work has been in progress, in cooperation with Applied Physics Division, to assemble and test equipment for this purpose. Adequate discrimination between neutron and gamma radiation has been achieved to enable neutron flux levels to be measured. The system depends upon detection by an encapsulated organic liquid scintillator and a discriminator system sensitive to the shape of the signal pulses. This system has been calibrated against established equipment, using the Applied Physics Division's neutron generator, and used to confirm and measure the suspected low neutron production from the Activatron generator.

Tritium measurement

Another essential measurement is that of the tritium production in lithium from neutron bombardment. A method has been developed, in conjunction with Materials Division, for the manufacture of pellets of lithium carbonate powder enclosed in aluminium containers (tritium detection samples). These will be used to obtain an indication of tritium production distribution throughout an experimental blanket assembly. Test samples have been irradiated and analysed by Isotope Division.

Fusion blanket experiments

So far, only preliminary experiments have been carried out on tritium production. An assembly of sacks filled with lithium carbonate was built to form a one metre cube into which the accelerator target was inserted; the accelerator was then run for several hours to generate neutrons and to irradiate the stack, which also included several of the tritium detection samples. Subsequently, some of these were analysed for tritium content. At the present time, the tritium production has been found to be very low, owing to low neutron production from the accelerator target. The reason for the low neutron output is being investigated.

2.2.3 Calculations for fusion blanket neutronics experiments
(B.J. McGregor)

Calculations were made for comparison with an analysis, carried out by Moshin and colleagues at Julich in 1980, of tritium production measurements in a lithium aluminate assembly.

The experimental arrangement consisted of a cubic (20 cm) central source region cm with an opening for an accelerator tube on one side. The source region was surrounded by a lead zone of side 40 cm, then a lithium aluminate zone of side 40 cm and a polythene zone of side 120 cm. Calculations were first done on a one-dimensional (spherical) geometry with ANAUSN and then with ANISN. A study was then undertaken with the Monte Carlo code MORSE using the correct experimental (cubic) geometry to assess the effect of the spherical approximation.

Calculated tritium production results are in good agreement with the analysis of Moshin and co-workers, though there are substantial differences between the calculations outside the tritium producing region. The MORSE results with a cubic geometry gave a tritium production rate from ^{6}Li that was lower than that for the spherical assumption near the source, and greater when remote from the source. The tritium production from ^{7}Li was similar in cubic or spherical geometry. For this well reflected system, production from ^{6}Li is the main source of tritium. These methods should be suitable for analysis of the blanket neutronics experiments now being undertaken at Lucas Heights. In this regard, preliminary calculations were made to estimate the tritium production and sodium activation rates to be expected in a bare lithium carbonate assembly.

An investigation was made of practical methods for using point detector estimation in the MORSE Monte Carlo code for fusion neutronics calculations, since the use of the standard routines requires extremely long computer runs to obtain reasonable statistical errors. A simple importance-sampling method of choosing whether to include the effects on a detector of any collision in the assembly substantially improved this application. The contribution of a collision to a detector is included with probability varying inversely as the square of the distance from the detector to the collision point. The weight of a chosen colliding particle is altered to compensate for the collisions that are not chosen. A number of methods were investigated for handling the effects of collisions very near the detector which make the estimate unbounded.

3. PUBLICATIONS AND CONFERENCE PRESENTATIONS

3.1 Journal Papers

Beattie, D.R.H. and Lawther, K.R. [1981] - Two-phase hydrodynamic experiments with axial flow through seven-rod clusters. Int. J. Multiphase Flow, 7:423-427.

Beattie, D.R.H. and Whalley, P.B. [1982] - A simple two-phase frictional pressure drop calculation method. Int. J. Multiphase Flow, 8:83-87.

Corran, E.R. and Witt, H.H. [1982] - Reliability analysis techniques for the design engineer. Reliability Eng., 3(1)47-57.

Green, W.J. and Lawther, K.R. [1981] - A flow boiling burnout correlation for water and Freon-12. Nucl. Eng. Des., 67:13-25.

Hooper, J.D. [1980] - Developed single-phase turbulent flow through a square-pitch rod cluster. Nucl. Eng. Des., 60:365-379.

Whatham, J.F. [1981] - Thin shell analysis of circular pipe bends. Trans. Inst. Eng., Aust., CE23:234-245.

Whatham, J.F. [1981] - Thin shell analysis of non-circular pipe bends. Nucl. Eng. Des., 67:287-296.

3.2 Reports

Beattie, D.R.H. [1981] - A smooth tube analogue of roughened wall thermohydraulics. AERE-R10194.

Chapman, A.G. and Carrard, G. [1981] - Compilation of experimental burnout data for axial flow of water in rod bundles. AAEC/E500.

Green, W.J. [1981] - Adaptation of a Freon-12 critical heat flux correlation to correlate water data from uniformly heated vertical tubes. Part I: Based on critical heat flux data for water at pressures of 3 to 14 MPa. AAEC/E532.

Green, W.J. [1982] - Adaptation of a Freon-12 CHF correlation to apply for water in uniformly heated vertical tubes. Part II: Based on CHF data for water pressures in the range 6-20 MPa. AAEC/E536.

Green, W.J. [1982] - An investigation of critical heat fluxes in vertical tubes internally cooled by Freon-12. Part II: The development of a critical heat flux correlation for uniformly heated tubes. AAEC/E528.

Green, W.J. and Jacobs, W.S. [1981] - THETRAN - A two-dimensional heat transport code for analysing power transients. AAEC/E507.

Green, W.J. and Stevens, J.R. [1981] - An investigaton of critical heat fluxes in vertical tubes internally cooled by Freon-12. Part I: Critical heat flux experiments with axially uniform and non-uniform heating and comparison of data with selected correlation. AAEC/E517.

Holland, P.G. and Marshall, J. [1981] - The use of ZOCO V in analysing the OECD/CSNI numerical benchmark problem. OECD/CSNI SINDOC(81)10.

Hooper, J.D. and Harris, R.W. [1982] - Hot wire anemometry technique for an automated turbulence measurement rig. AAEC/E516.

Marshall, J. [1980] - Containment safety research at the AAEC's Research Establishment, Lucas Heights. OECD/CSNI SINDOC(80)105.

Marshall, J. [1980] - AAEC: Proposed two-compartment containment experiment. OECD/CSNI SINDOC(80)219.

Marshall, J. and Holland, P.G. [1981] - AAEC containment experiment proposed for OECD/CSNI containment analysis standard problem. OECD/CSNI SINDOC(81)12.

Marshall, J., Holland, P.G. and Woodman, W.B. [1981] - OECD/CSNI containment analysis standard problem No.3. Experimental results. OECD/CSNI CASP3-1.

Turner, W.J. [1982] - MOVIES, plotting and printing of output from serial calculations. AAEC/E529.

Whatham, J.F. [1981] - The use of computer codes BENDEF and PRESEF. AAEC/M98. Amended June 1982.

Whatham, J.F. [1982] - The use of computer codes FLEXIN and FLEXOT. AAEC/M99.

Whatham, J.F. [1982] - The use of computer codes SHEREF, COILEF and TURNEF. AAEC/M100.

3.3 Internal Reports

Corran, E.R. and Witt, H.H. [1982] - A reliability study of the HIFAR ECCS. NT/TN426.

Woodley, H.J. and Blevins, R.J. [1981] - Measurement of voidage and flow using neutrons. ER/TN421.

3.4 Conference Papers

Beattie, D.R.H. [1980] - A point of inflection criterion for annular two-phase flow. European Two-phase Flow Group Meeting, Univ. of Strathclyde, Glasgow, Scotland, June.

Beattie, D.R.H. [1980] - An evaluation of two bubble-detachment models for two-phase flow. ANS-ASME Int. Topical Meeting on Nuclear Reactor Thermohydraulics, Saratoga, Wyoming, USA, October. NUREG/CP-0014, p.1343.

Beattie, D.R.H. [1981] - Hydrodynamic regime change effects on critical heat flux characteristics. European Two-phase Flow Group Meeting, Univ. of Technology, Eindhoven, The Netherlands, June.

Beattie, D.R.H. and Hanna, G.L. [1980] - The accurate measurement of flow rates in gas and wet steam systems using a radio-tracer technique. Proc. 8th National Chemical Engineering Conf., CHEMECA 80, August.

Beattie, D.R.H. and Lawther, K.R. [1980] - Experiments on air-water flows in an eccentric annular channel. Proc. 7th Australasian Hydraulics and Fluid Mechanics Conf., Brisbane, August.

Corran, E.R. [1981] - Reliability theory. Inst. Eng., Aust. Risk Engineering Symposium, Melbourne, October.

Corran, E.R. [1981] - Design of reliable control and protective systems. Inst. Eng., Aust. Risk Engineering Symposium, Melbourne, October.

Green, W.J. and Lawther, K.R. [1980] - An experimental study of dryout heat transfer in tubes by a temperature transient method. ANS-ASME Int. Topical Meeting on Nuclear Reactor Thermohydraulics, Saratoga, Wyoming, USA, October. NUREG/CP-0014, p.1092.

Green, W.J. and Lawther, K.R. [1981] - An investigation of flow boiling dryout transients. Proc. Third OECD-NEA/CSNI Specialists' Meeting on Transient Two-phase Flow, Pasadena, California, USA, March.

Green, W.J. and Lawther, K.R. [1982] - Application of a general critical heat flux correlation for coolant flows in uniformly heated tubes to high pressure water and liquid nitrogen. Seventh Int. Heat Transfer Conf., Munich, FRG, 6-10 September.

Hooper, J.D. [1980] - Measurement of secondary flows in rod cluster geometry. Proc. 7th Australasian Hydraulics and Fluid Mechanics. Conf., Brisbane, August.

Hooper, J.D., Crawford, W.J. and Hincksman, M.J. [1980] - Turbulent momentum interchange between connected flow passages. Proc. 7th Australasian Hydraulics and Fluid Mechanics Conf., Brisbane, August.

Ilic, V., Marshall, J. and Woodley, H.J. [1980] - Application of neutron techniques to the measurement of two-phase flow parameters. Proc. 7th Australasian Hydraulics and Fluid Mechanics Conf., Brisbane, August.