NEACRP-L-201

27 JUIL 1978

NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD OF THE TWENTIETH MEETING

(TECHNICAL SESSIONS)

PETTEN, NETHERLANDS
6th-10th June 1977

Compiled by

BOB RICHMOND

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY
38, boulevard Suchet, 75016 PARIS

NEACRP-L-201

NUCLEAR ENERGY AGENCY COMMITTEE
ON REACTOR PHYSICS

SUMMARY RECORD OF THE TWENTIETH MEETING
PETTEN, NETHERLANDS, 6th-10th June 1977
(TECHNICAL SESSIONS)

Compiled by
BOB RICHMOND

OECD Nuclear Energy Agency
38 boulevard Suchet, 75016 PARIS

CONTENTS

TECHNICAL SESSIONS

1. NEW TOPICS

1.1. Gamma heating and coupled neutron-gamma calculations

US

LeSage introduced the paper NEACRP-L-181 which summarized the status of gamma heating measurements made at ANL using thermoluminescent dosimeters.

The basic problem was to relate the dose measured in a dosimeter to the dose in the reactor material of interest. This was achieved by putting the dosimeter in a uniform sleeve normally made of the material in which the gamma heating was to be studied. A theoretical relationship between the dosimeter dose and the sleeve dose was then established and this allowed the sleeve dose to be determined. The theory was checked by encasing Li7F TLD's in sleeves of several different materials, exposing them to constant gamma-ray fluence and checking that the calculated fluence was constant within experimental errors.

The absolute accuracy achieved in the measurements was about ± 5% and was limited primarily by uncertainties in the magnitude of the neutron sensitivity correction.

Typical results were presented for the case of ZPPR assembly 4. These showed good agreement between the measured and calculated shapes of the gamma-heating distributions in regions of constant composition (i.e. within the core or within the blanket). Difficulties occurred, however, in the case of a zone transition e.g. the calculated heating in the blanket was systematically low.

Maienschein introduced the paper NEACRP-A-304 which described the similar development of thermoluminescent dosimeters at ORNL over a period of several years.

Two sets of results were presented :

(1) Transmission of mono-energetic gamma rays (0.3 - 2.7 MeV) through thicknesses of up to 15 cm of iron was measured using CaF_2 and Li7F TLD's and an ion chamber. The expected values of energy deposition together with the spectral correction factors were calculated using DOT. Agreement within ± 3% was achieved between measured and calculated values and the spectrum correction factors were verified.

(2) Radiation heating in iron and stainless steel shields was measured at the TSF by staff of ORNL, ANL-Chicago and ANL-Idaho. In the case of the iron shield measurements agreement between all measurements and the DOT calculations was within ± 3% on an absolute basis. In the stainless steel case the agreement between the measurements was to within ± 4% on an absolute basis but there was poor agreement between the calculated and measured distributions. This resulted from uncertainties in the cross-sections determining transport in nickel and chromium.

The general conclusion was that measurements of gamma heating were currently achieving the requested absolute accuracy of ± 5%.

Japan

Hirota introduced the paper NEACRP-A-296 which described a study of fission rate and gamma ray heat generation made as part of the engineering mock-up experiment for MONJU. The measurements were made in a 90^{o} sector mock-up of MONJU built in the FCA VII-1 assembly. The spatial distribution of the gamma-ray absorbed dose was measured through the core, radial blanket and primary neutron shield using TLD's which were calibrated using a Co60 source in air. Interpolation (or extrapolation) to the Z_{eff} value of the medium then allowed the absorbed dose in the medium to be determined. Fission rate distributions were measured using micro-chambers.

In the analysis of the measurements the coupled 100-neutron and 20-gamma-ray group constants of each medium were generated by RADHEAT (Version 3) using basic data taken from ENDF/B-4 and from the POPOP4 library.

The calculated fission rate distributions agreed with the measured distributions within about 50% over the range from the core centre to the primary shield.

The calculated gamma-ray dose distributions were in excellent agreement with the measured distributions in the shield, where the medium dose could be obtained by Z_{eff} interpolation but were lower than the measured results by a factor of three in the core and blanket where extrapolation to the medium Z_{eff} was required.

UK

Campbell summarized the paper NEACRP-A-299 which gave an account of the measurement of gamma-ray energy deposition in ZEBRA. Li7F TLD's were used and calibration was carried out in a known Co60 gamma-ray field. The correction for neutron response was typically 15% with an associated uncertainty of 15% (1σ).

For the calculation of the cavity correction a Monte Carlo electron tracking programme known as PROCEED had been written. Testing of this programme had been carried out using a Co60 source housed in the centre of an iron cube of side 600 mm. A number of test materials including LiF, C, Fe, Eu_2O_3 and Ta were used. The photon spectra and dose rates at the centres of the various test materials were calculated using the MORSE code and UKNDL data. The calculated spectra were used as input to PROCEED. The C/E values showed excellent agreement between the PROCEED corrected measurements and the MORSE calculations and it was concluded that the measurements validated the PROCEED cavity correction within the range of the experiments. However, a study of the cavity correction carried out by Van Prooyen using mono-energetic sources with energies up to 8 MeV had shown that, in the high energy region, a significant discrepancy existed between experiment and the PROCEED results for the case of low Z cavities surrounded by high-Z walls. Further work was therefore needed to establish the cavity correction in the high energy region. In general, however, it appeared possible to attain a standard deviation of 5% on TLD measurements using current techniques.

France

Barré introduced the paper NEACRP-L-186 which summarized the current status of gamma heating studies in France. The work had been in progress for only a relatively short period and the experimental accuracy had not yet reached the value of ± 5% reported by other countries. In a fast reactor the main problems arose in the control rods and blankets where the contribution of gamma rays to the total heating was respectively about 34% and 23 - 33% as opposed to 10% in the fuel elements. The aim was to measure the gamma ray heating with an uncertainty of ± 20% (2σ).

Measurements had been carried out in a block of iron driven by the source reactor HARMONIE and also in the core and blanket of the critical assembly MASURCA and in steel bars and boron carbide rods inserted into the centre of the core.

In the iron block measurements ionization chambers were used and also TLD's calibrated using a Co60 source. Large discrepancies were observed between the measured and calculated results. The experimental errors were estimated as ± 8% plus a contribution due to uncertainties in the neutron correction term. The ion chamber and TLD results were consistent to within ± 5%.

The measurements on MASURCA (also made with both ion chambers and TLD's) were subject to large errors (typically ± 20%) due to uncertainties in the neutron correction and to the large variation of the spectral sensitivity of the TLD's in the boron carbide rods.

In the calculations of gamma-ray heating the gamma-ray sources were obtained using CARNAVAL III or CARNAVAL IV for the core and blanket regions and PROPANE for the shielding. The gamma-ray transport calculations were carried out using either ANISN (1D) or MERCURE (3D). A 21-group library of photon cross-sections was used. In the case of the iron block experiment the absolute C/E discrepancy was (-20 ± 15)%. For the measurements on MASURCA with a central steel rod the measured and calculated values of gamma heating in the rod and in the two core zones were in agreement within the experimental errors of ± 20% (2σ). In the case of a central boron carbide rod the MERCURE calculations gave an overestimate of about 30% while ANISN gave better agreement. In the blanket region ANISN underestimated the gamma heating by (40 ± 20)% the error being attributed to the neutron transport calculation of the gamma-ray source. Work aimed at the improvement of the calculation methods and the experimental techniques was now in progress.

In a general discussion it was noted that several countries were now in a position to measure gamma ray heating to the required accuracy of ± 5%. Campbell felt that some useful further work could be done and suggested that it would be useful to set up an experimental techniques intercomparison in a reference spectrum. It was agreed that this would initially be discussed bilaterally and then considered again at the next meeting of the committee.

1.2 Review of sensitivity analysis methods

US

Maienschein introduced the paper NEACRP-A-305 which summarized recent developments in sensitivity analysis in the US. These included :

(1) The development of the initial sensitivity theory for time-dependent, coupled decay and neutronics problems. Application of these methods was expected to be concentrated on decay heat, fuel management and actinide waste disposal problems in LWR's and LMFBR's.

(2) Work in the area of channel theory the application of which in complicated shield design problems was expected to be a major step towards a quantitative approach to shield optimization.

(3) Work in connection with fusion reactor shielding which involved the extension of sensitivity methodology to cover secondary energy and angular distribution effects.

(4) The development of multidimensional sensitivity codes.

(5) The development of higher order accuracy sensitivity theories.

(6) Work on cross-section data adjustment methods in conjunction with sensitivity theory and cross-section covariance data which had led to studies of the effects of integral experiments in the uncertainty analysis of shielding and core physics design problems.

In discussion Farinelli said that work was in progress in Italy on higher order sensitivity methods. These were being applied to transactinide build-up and burn-up, fusion neutronics problems, shielding and dosimetry. There was close co-operation with ORNL and ANL in this field. Rief mentioned that sensitivity analysis was being carried out at Ispra in 3 dimensions by correlated sampling in connection with a Monte Carlo problem.

This might be useful in the analysis of streaming experiments. Extensive studies had been carried out on the application of sensitivity analysis to calculations of actinide build-up. Barré said that no significant effort was devoted to the development of sensitivity analysis methods in France. Standard methods were used. Küsters said that the situation was similar in Germany where no methods development work was being carried out at Karlsruhe. At the University of Stuttgart sensitivity methods had been developed and were being applied to shielding problems.

UK

Campbell presented the paper NEACRP-A-298 which described a proposal, put forward at the meeting held in Vienna in October 1976, for the application of sensitivity analysis to the identification of data requirements for shielding.

It had been agreed at the Vienna meeting that the standard deviation on a calculation caused only by data uncertainties should be limited to one half of the target accuracy for a given parameter. In connection with the modelling of design problems it was recommended that sensitivity studies could be made with a one-dimensional model and that a single radial buckling independent of space and neutron energy could be used. Some details were given of the manipulations required to transform the SWANLAKE output into a form convenient for carrying out calculations on the effect of data uncertainties. Attention was drawn to the sparsity of data on correlations, the main existing sources of information being the compilation by Schmidt and the ENDF/B correlated error files which were now becoming available.

The paper then set out detailed recommendations on the procedures to be adopted for arriving at data requests in the shielding field. Equations were given to define the upper and lower bounds of the fractional accuracies on measured quantities. The minimum value was obtained by assuming complete correlation within each energy group and no correlation between the groups. The maximum value assumed total correlation. A "measurement sensitivity" (which gives the fractional improvement in the standard deviation of the predicted design parameter per unit fractional improvement in the standard deviation of the measured cross-section) was introduced to focus the attention of the evaluator or measurer on those cross-sections and energy ranges which could most profitably be investigated.

The application of the proposed procedures to a typical case (the Barré fast reactor shielding benchmark problem) was described.

The paper finally set out in detail the recommended recipe to be adopted in compiling a revised list of shielding data requests for inclusion in WRENDA. (This recommended procedure is also set out in the report on the Vienna meeting : NEACRP-U-75).

In discussion the degree of sophistication required in sensitivity studies was considered with the general conclusion that a relatively simple approach was adequate. The need to consider sensitivity to calculation methods as well as sensitivity to data was stressed. The fact that progress was now being made towards the identification of real data needs in the shielding area was felt to be very encouraging.

1.3 Heterogeneous fast core studies

UK

Campbell presented the paper NEACRP-A-302 which summarized the UK position on heterogeneous cores. The main UK interest in such cores arose because of the reduction they offered in the positive reactivity ramp rate produced by rapid sodium expulsion and the consequently improved prospect of containing the subsequent energy release. Other desirable features of the heterogeneous designs, e.g. generally improved breeding characteristics, had been noted, as also had undesirable characteristics such as the reduction in Doppler coefficient.

In planning the UK studies in this area three constraints were imposed in order to limit the very large number of possible designs. These were (i) only Pu/U fuel would be considered (ii) the sub-assembly would closely resemble that being tested in PFR to give a design basis for the conventional core of the CFR (iii) the overall diameter of core and breeder would be close to that of the reference CFR design. This had led to consideration of designs with (a) a single breeder annulus (b) two breeder annuli (c) a "cartwheel" design, i.e. a 6-sector core in which each sector had inner and outer core regions surrounded by a breeder region and (d) a design in which islands of breeder were immersed in the fuel region. Studies of the properties of these cores relative to those of the conventional reference design showed that all gave a substantial reduction in the maximum reactivity gain on loss of sodium (typically 2$ instead of 8$) and also in the voided fuel Doppler constant (typically -3×10^{-3} instead of -5×10^{-3}). Increases in breeding gain and mean enrichment were also noted. The cartwheel and double annulus designs performed similarly although the double annulus was loosely coupled neutronically and presented some control problems. The island core design used only one sub-assembly per breeder island and, as a result, showed a smaller reduction in sodium voiding reactivity than the other designs. Studies involving larger islands or a combination of larger islands with a single annulus might be considered.

The uncertainties in the prediction of the performance of the heterogeneous designs were significantly greater, at the present stage, than those of the conventional cores particularly since diffusion theory models had been used in the heterogeneous core studies. Experimental validation of the predictions was essential and one of the aims of the ZEBRA BIZET programme was to carry out initial studies of heterogeneous cores. It was expected that a core of this type would be loaded in ZEBRA at about the end of 1977.

US

Lesage presented the paper NEACRP-L-182 describing a US study on heterogeneous cores which had been primarily motivated by an interest in reducing the sodium void reactivity. Initial studies had been concerned with tightly coupled heterogeneous cores in which the breeding performance was equivalent to that of a homogeneous system with the same homogenized volume fractions. Studies of loosely coupled cores indicated an overall reduction of the sodium void reactivity arising from an increase in the leakage component coupled with a decrease in the spectral component. Most of this work related to cores with one thick internal annulus but some studies of cores with two internal annuli had also been made. The reduction in sodium void reactivity was typically from 7 \$ to 2 \$ and the Doppler coefficient was reduced typically from -9×10^{-3} to -5×10^{-3}. Increases in breeding ratio and in average enrichment also occurred, the latter leading to higher fuel compaction reactivities. Greater difficulties in the analysis of heterogeneous cores arose because of neutron decoupling of core zones, the need to use transport instead of diffusion calculations and the significance of gamma-ray heating in the internal blankets.

Netherlands

Bustraan summarized an investigation made at KEMA (Arnhem) into the improvement of the SNR-300 breeding ratio which could be obtained by introducing breeder assemblies into the core. An increase from 0.90 to 1.17 was achieved in the breeding ratio but this was accompanied by an increase from 30% to 75% in the average fuel element enrichment. Calculations of sodium void reactivity and Doppler coefficient had not yet been made.

Germany

Küsters commented that the current German view was rather the reverse of the UK and US view in the sense that a main aim of the fast reactor programme was seen, in Germany, as the conservation of uranium resources, i.e. the reduction of doubling time. There was a further incentive to improve the safety characteristics of the reactor if possible. He stressed that, although the heterogeneous designs reduced the sodium void reactivity they also led to higher fuel enrichment which produced higher fuel compaction reactivity and led to increased doubling times. At the present time it was not clear whether the introduction of a heterogeneous design would be beneficial.

Detailed physics studies of a variety of heterogeneous designs (see NEACRP-L-180, page 76) had given results very similar to those obtained in the UK studies. A more comprehensive investigation of heterogeneous cores was under way and this would include thermohydraulics and safety aspects.

France

Barré pointed out that the French position also differed from that of the UK and US since the aim was to optimize the heterogeneous core so as to achieve minimum critical mass and minimum doubling time. Two 1800 MWe heterogeneous reactors were currently being studied. The first of these had 5 annuli and the other had also 5 annuli but in addition a 15 cm blanket in the midplane. A comparison of the heterogeneous design with the conventional design showed that the breeding gain increased from 0.22 to 0.33 - 0.35, the doubling time decreased from 25 years to 16 years and the reactivity loss decreased from 13 to 4 pcm per day. Another useful feature of the heterogeneous core was that only one fissile enrichment was used. Full consideration was given in these studies to the safety aspects of the heterogeneous core but the main aim of the design was to minimize the doubling time.

In discussion LeSage stressed that, although the study reported in NEACRP-L-182 had been primarily aimed at the reduction of sodium void reactivity, other aspects of heterogeneous cores were also being examined.

US

LeSage presented the paper NEACRP-L-183 which summarized the results of a series of experiments performed on ZPPR to evaluate a radial parfait heterogeneous design (with a central breeder zone and 3 internal breeder rings) for the Clinch River Breeder Reactor. A total of 5 configurations were examined (ZPPR 7A - 7E) starting with a clean radial parfait configuration and including two distributions of simulated withdrawn control rods, representation of fissile build-up in the internal breeder regions by adding plutonium fuel to these regions and a core containing simulated natural B4C control rods. Analysis of the results of the measurements used ENDF/B - IV data and included both diffusion and S4 transport calculations.

Eigenvalue calculations indicated the increased importance of transport effects relative to conventional core designs and also showed significant differences between rz and xy calculations.

In the case of the radial reaction rate distributions both transport and diffusion calculations gave a tilt in the C/E values with the lowest values near the core centre and the highest in the outer core rings. This tilt was sensitive to adjustments of the U238 capture cross-section. In the comparison of overall reaction rates in the core and internal breeder regions transport theory gave significantly better results than diffusion theory. Special cross-section preparation procedures were needed in the case of U238 fission and this involved setting up a transitional cell with both blanket and core components and adjusting the core and blanket cross.sections with the spectrum obtained from this cell.

The C/E values for control rod worths were generally 5 - 10% below unity with the inner ring values a few per cent lower than those for the outer ring. In conventional cores the C/E values had been close to unity. The variation between rings was consistent with the observed C/E tilt for the fission rate distributions.

The analysis of sodium voiding measurements gave C/E values within 10% of unity while, for similar analyses of conventional cores, C/E had been in the range 1.2 - 1.4.

France

Barré introduced the paper NEACRP-L-184 which described the performance and analysis of experiments on a heterogeneous configuration in MASURCA. This consisted of a single fertile zone 30 cm in diameter inserted at the centre of a two-zone core with an overall diameter of 90 cm. The main parameters studied were : critical mass, reaction rate ratios, breeding gain, power distribution, sodium void effects and gamma-ray heating.

The original intention had been to analyse the measurements using the standard methods and data of Carnaval III or IV but it was found that improvements were required for the heterogeneous core analysis. The effect of using transport instead of diffusion theory was studied and it was found that the transport values of F8/F5 were 8 - 9% lower. In the case of breeding gain there was no significant difference between transport and diffusion theory in the internal fertile zone and the fissile zones but the transport value was 10% lower in the radial blanket.

In the preparation of cross-sections for the fertile and fissile zones the effects were examined of (a) allowing for leakage in the fertile zone (b) carrying out a calculation for a heterogeneous cell including both fissile and fertile zone components (as in the ZPPR study). It was found that the use of the heterogeneous cell led to a decrease of about 2% in the central value of the C8/F5 ratio but otherwise had negligible effect. Taking account of the leakage term increased F8/F5 by about 3% and C8/F5 by about 1.5%. The effect of these various improvements on the calculated breeding gain of the fertile zone was not significant.

Eigenvalue transport calculations showed good agreement with experiment in the case of Carnaval IV and this corresponded to the performance of Carnaval IV in the case of a conventional core. Central reaction rate ratios were generally calculated to within experimental errors using the transport method. The use of diffusion theory led to an overestimate of F8/F5 by 5 - 8%. Additional reaction rate ratio measurements were made at a point in the plutonium zone 11 cm outside the boundary of the central breeder zone. At this point the value of F8/F5 differed by 10% from its asymptotic value. In the case of F8/F5 and F9/F5 the C/E discrepancies were similar to those in the asymptotic spectrum but for C8/F5 there were significant differences between the C/E discrepancies in the two cases. A "measured" value of the breeding gain integrated radially over the midplane of the system was obtained from measured values of the distribution of C8, F8, F9 and F5 together with calculated values of the C9 and C5 distributions. Agreement between the calculated and "measured" values was within experimental errors in the fertile and fuelled zones. Sodium void effects were measured in the central fertile zone and its axial blanket and it was found that the calculated values were generally 30 - 40% greater than the measured values in the central zone and 50% in the axial blanket. This was in contrast to the conventional core results were the discrepancies were not greater than 20%.

A corresponding set of measurements on a core without the fertile central zone was nearing completion. It would be followed by a study of a core with a fertile central zone and one fertile ring and subsequently a core with several rings would be investigated.

In a general discussion the possibility of collaborative experiments was considered. It was generally felt that complementary experimental programmes would be more profitable than very similar programmes but that it might be useful to discuss the performance of collaborative experiments in particular cases with the aim of resolving specific C/E discrepancies. The US and UK members stressed that much further work would be required to obtain a basic understanding of the physics of heterogeneous cores and that their immediate programmes would be aimed at obtaining such an understanding although they would be related to realistic designs. Barré, on the other hand, expressed the French position that the experimental programme should be concerned specifically with the reference design. Hirota mentioned a Japanese experimental programme on a heterogeneous core which was scheduled to start at the end of 1977. Due to fuel shortage a basic study would be carried out with a single internal breeder on the middle plane of the assembly. The thickness of this zone would be varied and measurements made of the resultant changes in important lattice parameters.

1.4 Establishment of power peaking margins in power reactors

UK

Askew presented the paper NEACRP-A-303 which was concerned with the problem of determining peak fuel pellet ratings during normal reactor operation. Data sources could be classified as either continuous, e.g. electrical output of plant, flow and outlet temperature, fission chamber measurements or intermittent, e.g. flux scans, gamma scans of discharged fuel.

The sources of uncertainty in calculating pin rating were conveniently taken as uncorrelated and illustrative figures were given for the case of a hypothetical SGHWR. In this case individual measurements of channel power were available and these were smoothed over adjacent channels in order to improve statistics. The axial power variation was known at the start of life from axial flux scans made after refuelling but this knowledge deteriorated during irradiation. The pin-to-pin power variation in an assembly was assumed to be calculated, the methods being justified by data from zero energy experiments. The effect of power tilt in the pin and the fraction of heat generated in the fuel were calculated. Account was also taken of the effect of manufacturing tolerances.

Two problems arose in determining the distribution of peak rating. Firstly some of the distributions of uncertainty would be highly correlated between all high power assemblies and, secondly, the distribution of calculated pin powers which determined the population at risk had to be determined. The overall probability distribution was assumed to be normal since it was a combination of many components of similar magnitude but this would not be true in all cases.

In connection with the fitting of calculation to measurement UK studies currently included time correlation to improve the knowledge of power distribution. This should make it possible for example, to distinguish real overpower channels from instrument errors by the rate at which the power burned down during operation. It was expected that these methods would give greater confidence in the techniques used for rejecting measurements as inconsistent.

1.5 Sodium voiding

Japan

Hirota introduced the report NEACRP-L-189 which gave an account of an experimental study of the effect of simulated fission products on sodium void worth. These had been carried out in the FCA VII-2 assembly. Fission products were simulated by B_4C and also by the mixture "Fs" proposed by Schröder. These were built successively into a central zone of the assembly and the sodium void worth was measured in each case. Measurements were also made of the reactivity worths of natural and enriched uranium and of polythene. It was found that the central sodium void worth was affected significantly by the presence of the simulated fission products but that the two different simulants produced different changes in the void worth. The reasons for this had not yet been clarified. The worths of the other measured materials were not sensitive to the presence of fission products.

UK

Campbell introduced the paper NEACRP-L-300, which reviewed calculational methods and experiments in the sodium void area, and NEACRP-L-301 which dealt in more detail with the influence of heterogeneity.

The overall sodium removal effect was effectively the sum of the positive spectrum term and the negative leakage term and was therefore very sensitive to data and calculation methods. The voiding effect for a 1300 MWe conventional reactor was about 8$ while values of about 2$ were obtained with heterogeneous cores of 1300 MWe and conventional designs of 300 MWe. A target accuracy of ± 1.5$ (1σ) was suggested for a total effect of 8$ and ± 0.5$ for an effect of 2$. If only the voidage of flowing sodium were considered the effect would be reduced by up to 40%.

The paper gave a summary of completed and planned measurements of sodium void effects and concluded that these could be measured to within target accuracies. Measurements to date had, however, been restricted to zero energy cores of 300 MWe size and extrapolation to power reactor cores of 1300 MWe size was necessary. This involved allowance for the effects of increase of core size, progressive replacement of control rod absorber by fission product absorption, change of fuel temperature, change in plutonium isotopic composition and extrapolation from small void measurements to large void calculations.

The significance of heterogeneity in the extrapolation to the power reactor had been considered and, in the case of sodium void worth measurements in plate regions in ZEBRA, a least squares fit had indicated a correction factor of the order of +20% for the radial leakage component. It was clear that a correct assessment of streaming must be made in the analysis of this type of experiment. There were some doubts about the application of the Benoist theory and, in the UK, the discrete ordinates code WDSNST was being used to investigate the streaming problem. In the power reactor case the effect of streaming was to increase the leakage component by about 10%. The results of the calculations were, however, very sensitive to the choice of the cylindrical cell boundary condition. A proper treatment of the hexagonal cells was therefore required and the method of Köhler and Ligou might be appropriate.

Studies of fuel temperature effects showed that, for a 1300 MWe reactor, the void reactivity increased from 4.9$ at 300^{o}K to 7.5$ at 3300^{o}K. Burn-up studies had indicated that the most positive void effects would occur at the end of a fuel cycle. Japanese studies had indicated that the use of transport theory (S16) instead of diffusion theory increased the effect of whole-core voiding by about 20%.

Germany

Küsters presented the paper NEACRP-L-195 concerning the current status of sodium void effect prediction at GfK. This paper also stressed the problems of extrapolating from critical assemblies to power reactors particularly in relation to cell geometry, core size, plutonium vector and lumped and distributed absorbers. A large number of measurements had been performed in a number of SNEAK assemblies with a view to estimating the accuracy of prediction of the sodium void effect in planned power reactors.

Measurements of central void worths in regions containing plutonium metal plates with a Pu240 fraction of 18% showed large C/E discrepancies (up to 50%) compared with discrepancies of only about 10% when no plutonium was present. In the case of PuO_2 plates with a high Pu240 content a C/E discrepancy of 30% was observed. The reasons for these discrepancies were under investigation.

France

Barré introduced the paper NEACRP-L-188. This summarized the most important available sodium void measurements for conventional and heterogeneous cores. The main aims of the programme were (1) to determine the C/E discrepancies and to plan the future programme with a view to explaining these discrepancies and (2) to obtain a better understanding of the sodium void effect by detailed calculation of its various components and, in particular, to investigate the extrapolation from critical assemblies to power reactors. In the analysis of the experimental results it had been found that there were no significant differences between the Carnaval III and Carnaval IV schemes. The discrepancies between calculated and measured sodium void effects for the voiding of central zones in conventional cores were generally less than 20%. Greater discrepancies were found in the region of the core-blanket interface (both axially and radially) and also in the axial blanket. These were under investigation by means of anisotropic diffusion calculations.

An experimental programme was under way in MASURCA with the aim of checking the effects of high plutonium isotopes on sodium voiding. Plutonium loadings with respectiviely 8% and 18% Pu240 contents were being used and there was also the possibility of using a small zone containing plutonium with a

Pu240 content of 44%.

Details of the studies on sodium voiding in heterogeneous cores (where the C/E discrepancies were generally of the order of 30 - 40%) had already been given in the paper NEACRP-L-184 (Section B.1.3).

Japan

Inoué introduced the paper NEACRP-A-297 which presented a method for the prediction of sodium void reactivity worth. Attention had been given to a check of the Benoist anisotropic diffusion coefficient which was made by means of numerical studies using the TWOTRAN code. It was concluded that Benoist's method overpredicts the leakage components of the void reactivity worth. Following the introduction of modified leakage components and a number of other improvements a comparison of calculated and experimental values of sodium void worths in ZPPR Assembly 3 gave C/E values generally less than unity. A corresponding exercise for the MZA and MZB cores gave C/E values varying over the range 0.37 to 1.48. For the MZB-(2) assembly C/E also varied widely (from 0.20 to 1.33).

Studies of the sodium reactivity worth of MONJU indicated that the effects of the change to fuel pin geometry, the increase of fuel temperature to 3000^{0}K and the burn-up of core and breeder each produced increases of less than 10% in the void worth. The full insertion of regulating rods, however, produced a 30% increase in the void reactivity.

US

LeSage presented the paper NEACRP-L-193 which considered some aspects of the extrapolation of critical assembly sodium void data to the power reactor case. Sodium void worth measurements were carried out in plate lattices in ZPPR assemblies 2 and 5. The analysis (using ENDF/B-4 data) was carried out using directional diffusion coefficients to account for streaming effects. The use of Benoist diffusion coefficients in place of standard diffusion coefficients produced a substantial improvement in the (C-E) values for sodium void worths, primarily because the Benoist approach systematically increased the leakage term. Bias factors for extrapolating to the power reactor were obtained by making a least squares fit to determine leakage and spectral adjustment factors which would minimize (C-E). Bias factors of 0.84 were found for both spectrum and leakage terms when the Benoist diffusion coefficients were used.

In discussion of the sodium void effect there was general agreement on the importance of heterogeneity. Barré emphasized the hazards of applying to plate lattices the Benoist correction which had been developed for the pin lattice case but LeSage pointed out that a systematic Monte Carlo comparison by Wade and Gelbard had indicated that, although the Benoist treatment produced some biases the results were generally satisfactory. Campbell recalled the sensitivity of the UK WDSNST calculations to the choice of cell boundary conditions and stressed the need to use codes giving a correct treatment of the pin cell geometry. Richmond said that this was achieved by both the Köhler/Ligou and Eisemann methods which had, in fact, been shown to be in good agreement in the prediction of streaming effects in GCFR pin lattices. There were some limitations, however, in the Köhler/Ligou approach since, in its current form, it could not deal with material in the coolant regions. Askew mentioned the possible use of a characteristics method which would lend itself very easily to solving the problem of a cell with a hexagonal boundary.

In the area of high Pu240 lattices the good agreement between calculation and measurement obtained for experiments on ZEBRA and ZPR6 contrasted with the difficulties experienced with the interpretation of the German measurements on SNEAK.

It was generally agreed that it would be useful to have a review paper setting out the commmittee's views on the current status of the prediction of sodium void effects and Küsters undertook to prepare such a paper.

1.6 Miscellaneous Topics

France

Bouchard introduced the paper NEACRP-A-307 which gave the preliminary results of calorimetric measurements of decay heat carried out at Fontenay-aux-Roses during the period 1972 - 75 with the primary aim of validating the French fission product data bank. The measurements covered cooling times in the range 70 - 10^5 seconds and gave the decay heat values for U235, U233 and Pu239 after irradiation with thermal neutrons for periods in the range 100 - 10^5 seconds. The experimental errors were < 10% for cooling times greater than 150 sec. and 10 - 20% for times less than 150 sec.

Bouchard agreed to present the final report on this work at the next NEACRP meeting.

US

Maienschein presented the paper NEACRP-A-306 which gave, in graphical form, the results of recent ORNL measurements of fission product decay energy release rates for cooling times in the range 2 - 14,000 seconds after thermal neutron fission of U235. Irradiation times were in the range 1 - 100 seconds. Beta and gamma ray emissions were measured separately as spectral distributions using scintillation detectors.

The measured values of total energy release were significantly lower than the ANS 1971 standard. They overlapped results from other US laboratories within current error bars of ± 5% but there were some systematic differences. Agreement with

ENDF/B-4 predictions was satisfactory but partly fortuitous since there were significant discrepancies in the case of the individual beta and gamma components. Discrepancies were noted between the ORNL measurements and earlier US measurements for both beta and gamma components at cooling times of the order of 1000 seconds. It was hoped that final analysis of the ORNL results would indicate errors of ±3%.

Spain

Velardé summarized the paper NEACRP-L-187 which considered the various eigenvalues (e.g. the effective neutron multiplication factor per neutron generation, K, the multiplication factor per collision, γ , and the fundamental multiplication rate, α) and examined which of these could be most appropriately used for calculating a given property of a given reactor system. Recommendations were made for a wide range of reactor systems, both reflected and unreflected and covering both static and time-dependent calculations.

Germany

Küsters presented the paper NEACRP-L-196 which dealt with the measurement and calculation of neutron-induced gamma fields in iron. Two types of systems were studied: iron spheres and a square steel pile. A Cf252 neutron source or a Th228 gamma ray source (both absolutely calibrated) could be placed at the centres of these systems. Measurements of neutron spectra, U235 fission rates and gamma spectra were made (in absolute units) at the surface of the spheres and inside the steel pile.

Neutron spectra and fission rates were calculated in one dimension using the S_{16} approximation and 208 energy groups. For the gamma calculations an extended version of the gamma transport programme BIGGI 4T was used in spherical geometry. For the calculation of the neutron-induced gamma spectra an η-γ production matrix was produced from ENDF/B-IV data.

In the case of gamma leakage spectra from the iron spheres there was relatively good agreement between the measurements and ENDF/B-IV predictions. Similar agreement was obtained for gamma leakage spectra from spheres with a central Cf252 source. The measured and calculated gamma spectra within the steel pile with a central Cf252 source were also in generally good agreement. It was concluded that the gamma production cross-sections derived from ENDF/B-IV were reliable.

2. TOPICS CARRIED OVER FROM PREVIOUS MEETINGS

2.1 Problems related to burn-up and fuel cycle in fast and thermal reactors

Japan

Hirota presented the paper NEACRP-A-294 which dealt with the testing of the fission product data in JENDL-1 using integral data. An evaluation of nuclear data had been made for the 28 fission product nuclides which are most important for fast reactors and the reliability of the resultant group constants had been checked using integral measurements made on the STEK reactor. The results of an initial comparison had been given in JAERI-1248 but the comparison had now been repeated using revised values of the STEK flux and adjoint given by ECN. In the case of the mixed fission product samples the revised flux and adjoint values gave improved agreement between calculation and experiment. For the individual fission product nuclides the new flux and adjoint gave generally improved C/E values in STEK cores 500, 1000 and 2000.

France

Barré presented the paper NEACRP-L-191 which gave an account of the French experimental programme in the field of burn-up in fast reactors. The approach was the same as that used for studies of unirradiated cores, i.e. clean integral experiments followed by data adjustment. The main points to be considered were :

(1) Net plutonium production as a function of irradiation.

(2) Reactivity loss per cycle

(3) Other problems related to the fuel cycle e.g. fuel transport and reprocessing which require a knowledge of the quantities of actinides and fission products in the fuel as a function of irradiation and cooling time.

The experimental programme involved the study of higher plutonium isotopes and of fission products.

The investigation of higher plutonium isotopes was based on 3 types of experiment.

(a) Measurement of neutron balance using three batches of plutonium fuel with Pu240 contents of respectively 8%, 18% and 44%. The plutonium enrichments of each of the lattices were adjusted so that all had the same neutron balance. Buckling measurements were made by progressive substitution methods in MASURCA and by unit K-infinity measurements in ERMINE.

(b) Measurements of the fission rates of Pu240, Pu241 and Pu242 relative to the fission rate of U235 or U238 were made in all the lattices using micro-fission chamber techniques.

(c) Irradiation of separated plutonium isotopes in OSIRIS, RAPSODIE and PHENIX. These allowed the capture rates in Pu240 and Pu241 to be determined relative to the U238 capture rate with an accuracy of 3 - 4%.

The programme of measurements on fission products was mainly concerned with the measurement of the capture cross-section of the mixture of fission products present in irradiated fuel.

Pure fissile samples irradiated in RAPSODIE and PHENIX fuel elements irradiated for up to 3 cycles had been oscillated in ERMINE in 3 oxide fuelled lattices: a uranium lattice characteristic of SUPER PHENIX and 2 plutonium lattices characteristic of PHENIX and SUPER PHENIX respectively. Oscillation measurements were also made on corresponding unirradiated samples. After oscillation the samples were chemically analysed. Measurements on separated fission product nuclides had included oscillation in MASURCA and ERMINE, activation in ERMINE and irradiation in PHENIX.

A comparison of the results of the buckling and k-infinity measurements with the predictions of CARNAVAL IV showed agreement within the experimental errors. Good agreement between measurement and prediction was also obtained in the case of the PHENIX irradiation measurements of Pu240 and Pu241 capture and in the case of the various fission rate ratio measurements.

The results of the measurements on separated fission product nuclides made in France and also those made in STEK, FRO and CFRMF were used to adjust the capture cross-sections of the 34 nuclides representing 90% of the global capture. Present indications were that this adjusted set predicted global fission product effects to within 10 - 14%. The target accuracy for this quantity was 7 - 10%.

It was now considered that the higher plutonium isotope problem was solved but further work was planned in the fission product area.

Work in progress in the transactinides area involved evaluation studies on a large number of nuclides, measurement of integral cross-sections by irradiation in PHENIX and measurement of fission cross-sections by means of fission chamber measurements in MASURCA and ERMINE.

Barré briefly summarized the paper NEACRP-L-190 which dealt with the uncertainties related to a pseudo fission product. derived from the adjusted data of the individual fission product nuclides. The uncertainties in the predicted global fission product cross-section (which arose primarily from uncertainties in the cross-sections and yields of the individual fission products) were estimated as ± (13 - 16)%.

Barré summarized the paper NEACRP-L-185 which described the calculation scheme CARNAVAL IV. The main improvements relative to CARNAVAL III were:

(1) Improvement of calculation methods including the checking of a number of simplifying assumptions and the introduction of a new method for blanket calculations.

(2) Readjustment of clean core data using results from a total of 195 integral experiments. Specific adjustments were made in the case of the nuclides Fe, Cr, Ni, Pu238 and Am241. Following the readjustment the predicted parameters all lay within the experimental error brackets.

(3) The inclusion of integral burn-up data leading to the adjustment of the cross-section data for higher plutonium isotopes and fission products.

The new set gave reactivity predictions about 500 pcm lower than those of Carnaval III.

Japan

Hirota presented the paper NEACRP-A-195 which proposed the extraction of additional energy from used LWR fuel by feeding it to FUGEN reactors. The fuel would be reprocessed for removal of the fission products but without separation of the plutonium. Studies had indicated (1) that a satisfactory degree of burn-up, leading to a "once-through" cycle, could be achieved (2) that local power peaking could be kept within acceptable levels and (3) that fuel cycle costs could be competitive relative to the costs of a cycle using 1.5% enriched uranium fuel.

France

Bouchard summarized the paper NEACRP-L-194 which described the measurement of power distribution in irradiated LWR fuel assemblies using gamma-ray scanning. The aim was to obtain results complementary to those given by incore instrumentation in order to check the results given by the reactor calculations. The power distributions were obtained from measurements of the activity of La140 using calculated values of the gamma-ray transmission coefficients and the microscopic flux distribution in the sub-assemblies. The measurements were made in the storage pond during re-fuelling periods. Good agreement was obtained between the results of the gamma-ray scanning measurements and those given by the in-core instrumentation.

Germany

Introducing the paper NEACRP-A-310, which reviewed existing actinides data, Küsters said that he had received only two sets of one-group actinide cross-sections from members. These were the data used in the UK code FISPIN and the ORNL code ORIGEN for a typical fast reactor calculation. He had extended the review by introducing data from a number of additional sources. These included KFKINR, the Transactinium Nuclear Data Conference, ENDF/B-4 and a number of recent evaluations. A comparison of the FISPIN and ORIGEN data showed differences of more than a factor of two in the case of several nuclides. The higher Pu isotopes Pu240 and Pu241 differed by about 30% in the capture cross-section.

In the comparison involving the additional data the one-group cross-sections were obtained by collapsing over the spectrum of the LMFBR benchmark. This comparison was limited to Np237 and higher nuclides. A number of discrepancies in FISPIN, ORIGEN and KFKINR were indicated by comparison with the more recent data.

An indication of the influence of the observed data discrepancies on nuclide concentrations and on the radioactivity of an irradiated element after discharge was obtained by simulating the irradiation history of a typical fuel pin in RAPSODIE using ORIGEN. The ORIGEN data were then replaced successively by those of the other data sets and. in each case, ORIGEN was used to calculate the time behaviour of the nuclides in the fuel pin. Large discrepancies (i.e. factors greater than 2) were noted between the nuclide concentrations predicted by the different sets. These were not tolerable since target accuracies on these quantities were of the order of 20%.

In the thermal reactor area some gross inconsistencies had been observed in the 2200 m/sec. cross-sections in the ORIGEN LWR data library and it was felt that these might lead to large errors in calculations of the radioactivity of irradiated LWR fuel.

The general conclusion of the investigation was that the more recently available data must be introduced into the one-group cross-section sets.

France

Bouchard reported on a specialists' meeting on the separation and transmutation of actinides held at Ispra on March 16th - 18th, 1977 under the co-sponsorship of the IAEA and Euratom.

Three review papers had been given covering the chemical separation of actinides, the feasibility of actinide transmutation in reactors and the technological impact of actinides on the fuel cycle. A number of individual papers were also presented. It had been concluded that the incineration of actinides was feasible and that it would be more favourable in fast reactors than in thermal reactors.

During the meeting problems had arisen in the course of a discussion on reference calculations in the physics area. It had not been possible to reach agreement on the use of a reference reactor model and a reference one-group cross-section set. The NEACRP had therefore been asked to make a recommendation.

In discussion Küsters queried whether there had been any discussion at the Ispra meeting on the usefulness of actinide incineration. He reminded members that, at the Richmond conference in 1976, it was clearly concluded that the burning of actinides in reactors was not useful in the sense that it created more problems than it solved. Bouchard said that the Ispra meeting had concerned itself primarily with actinide separation. Only limited discussions had been held on incineration and these had made little progress because of the widely differing views expressed by different countries. Farinelli said that one conclusion of the Ispra meeting was again that incineration is only worthwhile if a separation efficiency greater than about 99.5% can be obtained.

In discussion of one-group actinide cross-sections it was generally agreed that it would be useful to investigate further the discrepancies indicated by Küsters review paper. Members agreed to calculate their one-group actinide cross-sections by condensing over the LMFBR benchmark spectrum and to send Küsters the results. Küsters agreed to review the data. It was also felt that it would be premature for the committee to make recommendations concerning the one-group actinide cross-sections at the present time and Tubbs agreed to inform the NEA of this view.

Hemmig suggested that a benchmark experiment related to actinide production would be useful and, after discussion, members agreed to consider this point and bring forward suggestions at the next meeting. Askew pointed out, in this connection, that it would be important to make full use of existing, non-classified information on actinide production in thermal reactors since this, in itself, would constitute benchmark information. Members agreed to bring forward any information of this kind at the next meeting.

2.2. Reactor physics problems related to LMFBR safety

Japan

Inoué presented the paper NEACRP-A-293 which dealt with the analysis of simulated melt-down and vapour explosion experiments in the MOZART MZB core which included the production of off-centre voids. For this purpose a modified diffusion theory was developed in which the effective longitudinal and radial diffusion coefficients of a cylindrical void were obtained from neutron current equations. A similar treatment was used to derive diffusion coefficients for channels containing dilute material. This modified diffusion theory was tested against TWOTRAN-II (using the S4 approximation) and satisfactory agreement was obtained. Comparison of the measured reactivity changes with those calculated by the modified diffusion theory gave C/E values ranging from 0.55 to 1.18.

US

LeSage introduced the paper NEACRP-L-192 which described the simulation of the successive stages of a Hypothetical Core Disruptive Accident on ZPPR Assembly 5. This included sodium voiding, clad redistribution and fuel redistribution.

In the steel slumping experiments only 29% of the steel could be moved, the remainder constituting the matrix and drawers. The steel in a central radial region of 30 subassemblies was moved axially from the central zone of the core to the outer zones. Diffusion theory calculations using ENDF/B-4 data predicted the resultant reactivity changes with C/E values in the region of 1.1 for both inserted and withdrawn control rods.

In the fuel slumping case the fuel in a central radial region of 18 sub-assembles was moved axially from the outer zones, which then had zero fuel density, to the inner zone, which then had double fuel density. This process was then reversed to give a zero fuel density inner zone and double density outer zones. For the fuel slump inwards the diffusion theory calculation gave a good prediction of the reactivity change which was further improved when streaming and transport corrections were applied (C = 1.77\$; E = 1.80\$). For the outward slumping the observed reactivity change was only -8.6¢ (with control rods withdrawn) while diffusion theory gave -56¢, reducing to -26¢ after streaming and transport corrections. With control rods inserted the outward slumping reactivity change was -23¢ while the predicted value (with corrections) was -37¢.

It was concluded that diffusion theory was not adequate for fuel slumping calculations but performed sufficiently well in the case of steel slumping.

3. NATIONAL PROGRAMMES

3.1 Review of recent activities, national programmes, evaluation work

The national reports prepared by NEACRP members were presented and discussed.

4. BENCHMARKS

4.1 2D LWR benchmark calculations

Askew presented a committee note which was a new copy of the paper issued last year (NEACRP-A-256/2) on the results of the benchmark exercise on BWR lattice cell problems. Some agreed corrections to the original document had been made and two new solutions had been added. These were :

(1) A second ABA solution (ABA/2) in which the results had been obtained from the CASMO code which contained an

improved model of the poison pins, and

(2) An additional AEA case which had only been applied to problem 2 and which used a characteristics method capable of a very refined representation of the cell geometry.

In the first problem (9-pin supercell with central burnable poison pin) the new ABA/2 solution lay between the AEEW (1) and JAERI solutions which had previously been considered to be the best solutions. In the second problem (small cell with cruciform rod) both new solutions lay between the AEEW(1) and JAERI results. In the remaining two problems the ABA/2 solution again lay very close to the "best" solutions. The new solutions had thus improved the situation and helped to define the best answer to within a range of about 1% in the eigenvalue. The results of the benchmark would therefore give a useful guide to anyone developing any other approximate method.

The paper describing this benchmark exercise would be published in the Annals of Nuclear Energy.

4.2 LMFBR Benchmark

LeSage informed the committee concerning the status of the LMFBR benchmark. Solutions had been received from the US, Japan, Italy, Switzerland and France and sheets had been prepared giving comparisons of the various calculated values of reactor parameters. Solutions from Germany and the UK were expected to be available in the near future.

The arrangements for the specialists' meeting to be held at ANL to evaluate the results of the benchmark exercise were discussed and the meeting dates were agreed as February 7th - 9th, 1978. It was agreed to invite the collaboration of IAEA in this meeting and the deadline for the receipt of any further solutions was extended to 1st November 1977 in order to encourage non-OECD countries to submit solutions. The meeting would include a review and intercomparison of the various solutions and working sessions on the evaluation of the results. The earlier benchmark problem set up by Baker would also be discussed. Details of the meeting are set out in the Information Note (NEACRP-A-311).

Tubbs agreed to write to the Chairman of CSNI asking him to designate an observer for the benchmark meeting and recommending the choice of a US observer. He also agreed to draft a letter for transmission to the IAEA by the Director of the NEA inviting non-OECD countries to submit solutions to the benchmark problem. LeSage agreed to distribute a report on the meeting to members in advance of the next NEACRP meeting.

4.3 Hydrogen entry benchmark

Solutions had been submitted by GfK (using KFKINR) and by EIR (using FGL4 and FGL5). A French solution (using Carnaval III) would be available shortly and there was the possibility of contributions from the US and Japan. Details of the benchmark had been sent to the NEA GCFR Co-ordination Group. A deadline date of 1st November 1977 was agreed for the receipt of further solutions and LeSage agreed to inform possible US contributors of this deadline. Larrimore suggested Russian participation and agreed to send details of the benchmark problems to possible Russian participants and to inform them of the deadline.

Küsters presented the paper NEACRP-A-309 which compared the results of the GfK and EIR calculations for 2 of the 7 lattice compositions constituting the benchmark. These were (1) a clean cold 300 MWe GCFR with no absorbers and (2) a 1000MWe GCFR at operating temperature with fission products. For the cold clean case the steam density coefficients ($dK/d\rho_{H_2O}$) were positive for all three data sets and had similar values. The absolute differences in K-eff values were more than 2.5% at low steam density values. For the hot poisoned case the steam density coefficent was calculated negative at low steam densities, becoming zero at a density of about 0.003 g/cm^3 and positive at higher densities. In this case the absolute magnitude of the steam density coefficient was quite different for the two data sets. In view of the K-eff differences it was suggested that participants should additionally quote k-infinity and leakage for the various cases solved. The general indication at this stage of the exercise was that the errors in the calculations of hydrogen entry effects were unacceptably large. Richmond recalled that this was in line with the conclusions of earlier work carried out at EIR.

Küsters agreed to circulate a review of the steam entry benchmark solutions to members by 1st April, 1978.

5. GENERAL

5.1 Compilations

Farinelli reported that the LWR compilation had been completed in draft and circulated to contributors for correction and to committee members (as NEACRP-A-290) for comments. The final version would be issued in September 1977 and would be given an NEACRP-L distribution. Members agreed to send comments to Martinelli by 1st September 1977.

5.2 Highlights of recent meetings of interest to NEACRP

Differential and Integral Nuclear Data Requirements for Shielding Calculations (Vienna, October 1976)

Campbell commented briefly on the meeting, an account of which was given in NEACRP-U-75. He noted that a supplementary paper by McCracken (NEACRP-A-298) had detailed the mechanisms for making further progress in this area (see Section B.1.2). The meeting report had included a proposal for a meeting, in Paris, of those who had collaborated in the NEA benchmark exercise. An attendance of 10 - 12 people was anticipated and it was suggested that the meeting should be held in November 1977.

In discussion of this proposal members expressed concern that, although large numbers of shielding meetings had been held during the course of the last few years, there had been relatively little tangible progress in the two main areas of data requests and benchmark experiments. Maienschein welcomed the McCracken proposal as representing the first real break-through in defining nuclear data requirements but emphasized that a considerable time would elapse before results became available. He also pointed out that effectively no progress had been made in the performance and analysis of the benchmark experiments during the past year. There was some feeling that the Paris meeting should concentrate more on data requests but Farinelli pointed out that it would, in any case, not be possible, at the time of the meeting, to produce revised request lists for iron and sodium since the necessary experimental data was not available. The meeting should therefore concentrate (as initially planned) on the theoretical analysis of the benchmark experiments using sensitivity studies. The committee agreed that the Paris meeting should be set up on this basis and Tubbs agreed to contact Butler in connection with the arrangements for the meeting.

The committee discussed the further proposal that a joint OECD(NEA)/IAEA specialists' meeting should be held in the autumn of 1978 with the aim of establishing specific nuclear data requests for shielding on the basis of sensitivity analysis. It was generally felt that the proposed timing of this meeting might be a little premature. It was therefore agreed that this date should be provisionally accepted but that a final recommendation concerning the date of the meeting should be made by the participants in the 1977 Paris meeting.

Fifth International Conference on Reactor Shielding

(Knoxville, April 1977)

Maienschein said that the Conference had been dominated by streaming problems, the most crucial of these being streaming in the reactor cavity which was a common central problem for almost all reactor types. Very useful discussions had taken place concerning the problems in the Calvert Ciffs PWR's, which have excessively high radiation levels resulting from reactor cavity streaming, and the similar problems experienced in the Japanese ship Mutsu. In both cases the problems had arisen from the use of obsolete shielding calculation methods.

Another point discussed was whether sensitivity analysis was so problem-dependent that it would not be very useful for determining shielding data requests lists since a different list would be generated for each different reactor problem. Views for and against this suggestion had been expressed.

In the area of fusion reactor shielding a major difficulty was that the shielding problems in test facilities were qualitatively different from those in planned power reactors. This arose because the low duty factors of the test facilities (which result from access requirements) would eliminate radiation damage so that the real shielding problems of the power reactor would not be tested.

Fast Neutron Fission Cross-Sections of U233, U235 and Pu239

(Argonne, June 1976)

Küsters said that the errors on the U235 and Pu239 fission cross-sections had been quoted as ± 3% and ± 6% respectively. Ratios could be measured with errors in the range 1 - 2 %. No definite conclusion had been reached at the meeting on whether it is better to concentrate on the measurement of the cross-section of a standard nuclide (e.g. U235) and determine other cross-sections relative to this standard or, alternatively, to make separate measurements of the cross-sections of the relevant nuclides. In future the aim would be to check the cross-sections in specific energy regions and not to make measurements over the full energy range. Küsters felt that no improvement in accuracy was likely in the near future unless very considerable effort and expenditure were devoted to the development of new techniques and that this was probably not warranted.

Maienschein commented that the ± 6% error on the Pu239 fission cross-section (ratioed to U235) was totally unsatisfactory at the present time. He also felt that the workshop sessions held during this meeting had been of very little value. Campbell pointed out that the requested accuracy for fission cross-sections was generally ± 3%. He expressed dissatisfaction at the inadequate accuracy of the Pu239 fission cross-section and felt that more work would be needed in this area.

5.3 Meetings planned or proposed

Second Specialists' Meeting on Reactor Noise (SMORN II) (Gatlinburg, September 1977)

Maienschein said that there had been a good response to the call for papers and that a tentative programme had been identified. One third of the time of the meeting had been reserved for discussion and there would also be a Review Session at the end of the meeting.

In discussion some members expressed the view that (bearing in mind that this was a specialists' meeting) the time allocated for discussion was too restricted and that fewer papers should have been accepted for presentation. Maienschein agreed to pass on to the organisers the message that time schedules should be strictly adhered to so that the presentation of papers did not encroach upon the discussion time.

Cross-Sections in Structural Materials (Geel, December 1977)

Böckhoff expressed concern that, although this meeting had been co-sponsored by the NEACRP, very few papers had been received from the reactor physics side. He reported that a Preliminary Announcement had been circulated in November 1976. A Programme Committee had been set up in January 1977 and had prepared a preliminary programme which would shortly be circulated. The first two days of the meeting would be devoted to the presentation of papers and workshop sessions would then be held.

Members expressed surprise that a detailed programme had been prepared in advance of an official announcement of the meeting and Tubbs agreed to issue a definite announcement and call for papers.

Technical Committee Meeting on Homogenization Methods (late 1978)

This 3-day meeting would be sponsored by IAEA. The NEACRP agreed to co-sponsor the meeting.

Proposed Specialists' Meeting on Cross-Sections of Relevance to Actinide Build-up (Sweden, Autumn 1978)

The NEACRP decided not to support this meeting since actinide cross-sections would be fully discussed at the NEA/IAEA Conference on Neutron Physics and Nuclear Data for Fission Reactors and other Applications to be held at Harwell in September 1978.

International Symposium on Fast Reactor Physics (1979)

The committee decided to co-sponsor this meeting together with the IAEA. Tubbs and Larrimore agreed to inform the NEA and IAEA respectively of this decision. Küsters and Barré agreed to draft the introductory paragraphs for the information sheet of the Syposium and send these to Larrimore (with a copy to the NEACRP Secretariat) by 1st August 1977.

5.4 Progress with the NEACRP Book on the Status of Fast Reactor Physics

Only the Japanese and US chapters had been circulated for comments. A number of members commented that, while the US chapter was extremely valuable, the sections dealing with accident sequences contained very sensitive material which should not be published openly in book form. Campbell pointed out that the content of these sections did not, in any case, fall within the scope of the NEACRP. LeSage agreed to remove these sections. A revised timetable was agreed for the production of the review book and members undertook to carry out various actions which are listed as items 41 - 50 of Annex 2.

6. OTHER ACTIVITIES

A tutorial session was held on the afternoon of 8th June followed by a visit to the ECN laboratories.

A dinner was offered by the host organization on the same evening.

ANNEX 1

LIST OF PARTICIPANTS

Members

ASKEW, J.R.	AEE Winfrith, UK
BARRE, J.Y. (Chairman)	CEN Cadarache, France
BUSTRAAN, M.	ECN Petten, Holland
CAMPBELL, C.G.	AEE Winfrith, UK
DURET, M.F.	AECL Chalk River, Canada
FARINELLI, U. (Vice Chairman)	CNEN Casaccia, Italy
HEMMIG, P.B.	USERDA Washington, USA
HIROTA, J.	JAERI Tokai-Mura, Japan
INOUE, T.	PNDC Jokyo, Japan
JIRLOW, K.	ABA Studsvik, Sweden
KUSTERS, H.	KFZ Karlsruhe, Germany
LESAGE, L.G.	ANL Argonne, USA
MAIENSCHEIN, F.C.	ORNL Oak Ridge, USA
RICHMOND, R. (Secretary)	EIR Würenligen, Switzerland
RIEF, H.	ESIS Ispra, CEC
VELARDE, G.	JEN Madrid, Spain

Observers

BOCKHOFF, K.H.	NEANDC, CBNM Geel, Belgium
BOUCHARD, J.	CEN Cadarache, France
LARRIMORE, J.	IAEA Vienna, Austria

Secretariat

JOHNSTON, P.	NEA Paris, France
TUBBS, N.L.	NEA Paris, France

ANNEX 2

LIST OF 'A' AND 'L' REPORTS PRESENTED AT THE TWENTIETH MEETING

NEACRP-A-292	"Review of fast reactor physics activities relevant to the LMFBR programme in PNC" T. Inoue, JAPAN
293	"Modified diffusion theory analysis of simulated meltdown experiments" Yuji Seki and Makoto Sasaki, JAPAN
294	"Test of FP data in JENDC-I by using the integral data" Y. Kikuchi and A. Hasegawa, JAPAN
295	"On the physics of LWR-FUGEN" T. Haga, M. Matsumoto, H. Kato, and S. Hattori, JAPAN
296	"A study on fission rate and gamma-ray heat generation in a fast reactor" K. Koyama, N. Sasamoto, M. Obu, S. Tanaka, Y. Furuta, H. Kuroi, and J. Hirota, JAPAN
297	"Study on sodium void reactivity worth predictional method" T. Kamei, T. Yoshida, K. Murakami, S. Iijima, JAPAN
298	"The application of sensitivity analysis to the identification of data requirements for shielding" A.K. McCracken, UK
299	"The measurement of gamma-ray energy deposition in the zero-power fast reactor ZEBRA with Li^7F thermo-luminescent dosimeters" A.D. Knipe, UK
300	"The sodium void effect: The influence of heterogeneity" M.J. Grimston and A.T.D. Butland, UK
301	"The sodium void effect: A review of calculational methods and experiments" A.T.D. Butland and M.J. Grimstone, UK
302	"Studies of low sodium void reactivity cores" C.G. Campbell, UK
303	"The determination of peak pellet rating" J.R. Askew, UK
304	"The thermoluminescent dosimeter measurement of gamma-ray energy deposition distributions in reactor shields" P.N. Stevens, C.E. Clifford, F.J. Muckenthaler, and W. Yoon, USA

305 "Recent developments in sensitivity analysis in the U.S."
E.M. Oblow, USA

306 "Fission-product energy release for times following thermal-neutron fission of ^{235}U between 2 and 14,000 s"
J.K. Dickens, J.F. Emery, T.A. Love, J.W. McConnell, K.J. Northcutt, R.W. Peelle, and H. Weaver, USA

307 "Fichier de données produits de fission"
FRANCE

308 "Proposal for GCFR-steam entry benchmarks"
E. Kiefhaber, GERMANY

309 "First comparison of some results obtained at Karlsruhe and Würenlingen for GCFR-steam entry benchmarks"
E. Kiefhaber, GERMANY

310 "Influence of various muclear data sets on the prediction of actinides in fast reactor fuel after discharge"
H. Küsters and M. Lalovic, GERMANY

(National progress reports NEACRP-L-180)

NEACRP-L-180

a. Australia	j. Japan
b. Belgium	k. UK
c. Canada	l. Austria
d. Denmark	m. Switzerland
e. Euratom	n. Norway
f. France	o. Sweden
g. Germany	p. United States
h. Netherlands	q. Spain
i. Italy	r. Finland

181 "Status of gamma heating measurements at ANL"
G.G. Simons, USA

182 "Potential and limitation of the heterogeneous reactor concept"
W.P. Barthold, J. Beitel, E. Khan, and C. Tzanos, USA

183 "Summary of physics studies of a heterogeneous LMFBR core in ZPPR"
USA

184 "Physics performance of heterogeneous fast reactor core concept studied in MASURCA", FRANCE

185 "Formulaire Carnaval IV"
J.P. Chaudat, A. Desprets, G. Langlet, and A. Filip, FRANCE

186 "Etude de l'échauffement du aux gammas"
D. Calamand, FRANCE

187 "A comparison of the eigenvalue equations in k, α, λ and γ in reactor theory. Application to fast and thermal systems, in unreflected and reflected configurations"
G. Velarde, C. Ahnert, and J.M. Aragones, SPAIN

188 Etudes de l'effet de vidange sodium"
F. Lyon and M. Martini, FRANCE

189 "Measurements of fission product effects on sodium void worth in fast reactor"
K. Koyama, H. Mitani, H. Kuroi, and J. Hirota, JAPAN

190 "Incertitudes sur un pseudo-produit de fission de la filiere rapide"
G. Langlet, P. Coppe, and J.P. Doat, FRANCE

191 "Etude de l' evolution neutronique pour la filiere a neutrons rapides au CEA: Programmes experimentaux et résultats"
J.P. Chaudat, G. Langlet, and L. Martin Deidier, FRANCE

88020047

L-192 "Simulation of an HCDA sequence on the ZPPR critical facility"
R.E. Kaiser, C.L. Beck, and M.J. Lineberry, USA

193 "On the extrapolation of ZPR sodium void measurements to the power reactor"
C.L. Beck, P.J. Collins, M.J. Lineberry, and G.L. Grasseschi, USA

194 "Mesures de la distribution de puissance par spectrometrie gamma sur des assemblages irradies de reacteur a eau"
M. Darrouzet, M. Bertuol, and A. Santamarina, FRANCE

195 "The present status of sodium void effect; predictions at Gfk"
F. Helm, GERMANY

196 "Measurement and calculation of ^{252}Cf-fission neutron induced gamma fields in iron"
S.H. Jiang, H. Werle, GERMANY

Committee notes (will not be distributed or numbered)

- Review of international solutions to NEACRP Benchmark BWR lattice cell problems (will be published)
  UK

- Draft of SMORN-II program
  USA

- Erratum to NEACRP-L-160
  FRANCE

- Report on the nineteenth meeting of NEANDC
  (NEACRP observer)

- CCDN progress report (SEN/COMP(77)2)
  NEA

- CPL progress report (SEN/PROG(77))
  NEA

- Shielding problems in fast reactors (with "Calculating the transport of neutrons and radiation")
  ITALY

- The effect of heterogeneities on the breeding ratio of an LMFBR (SNR-300)
  NETHERLANDS

- Reactor Centrum Nederland: Activities July 1975 - July 1976
  NETHERLANDS

- Council directive of 27 July 1976 (76/770/EEC)
  EEC

- Conclusions of IAEA consultants' meeting on Reactor Physics (May 1977)
  IAEA

- Draft agenda and notes on solutions to fast reactor benchmark calculations for ANL meeting
  USA

- Report of IAEA consultants meeting on reactor physics, 16-18 May 1977.
  a. Proposal of list of topics for 1979 symposium on fast reactor physics
  b. Technical committee meeting (late 1978) on homogenization methods - draft information sheet.
  IAEA

- Guidelines for organizers of NEACRP specialists meetings
  NEA

ANNEX 3

PRELIMINARY AGENDA FOR THE 21ST MEETING

Part A : EXECUTIVE SESSIONS

1. a. Participants in the meeting
   b. Committee membership
2. Adoption of the final summary record of the twentieth meeting
3. Adoption of the agenda of the meeting
4. Completion of actions arising from previous meetings
5. Activities of other bodies of interest to the NEACRP
6. a. Report of the NEACRP observer on the twentieth meeting of the NEANDC
   b. Designation of the NEACRP observer for the 21st meeting of the NEANDC
7. Distribution of NEACRP documents
8. Arrangements for the 22nd meeting of the Committee
9. Other business
10. Election of Committee officers

Part B : TECHNICAL SESSIONS

1. New Topics

1.1 Fuel cycles
1.2 Actinide production and burnup
1.3 Nodal and coarse mesh codes
1.4 Neutron damage
1.5 Streaming problems
1.6 Blanket physics

1.7 Miscellaneous topics

2. Topics carried over from previous meetings

2.1 Heterogeneous LMFBR cores

2.2 Establishment of power peaking margins

3. National Programmes

3.1 Review of recent activities, national programmes, discrepancies, evaluation work.

4. Benchmarks

4.1 Fast reactor benchmarks

4.2 Hydrogen entry benchmark

5. General

5.1 Highlights of recent meetings of interest to NEACRP

5.2 Specialists' meetings planned or proposed

5.3 Progress with the NEACRP book on the Status of Fast Reactor Physics

5.4 Other business