NEACRP-L-230

NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD OF THE TWENTY-FIRST MEETING

(TECHNICAL SESSIONS)

JAERI, JAPAN

6th-10th November 1978

Compiled by

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CONTENTS

TECHNICAL SESSIONS

1. NEW TOPICS

1.1 Actinide production and burn-up

France

Bouchard introduced the paper NEACRP-L-213 which dealt with the production and burn-up of actinides in LWR's and fast reactors. The effects of the actinides on the reactivity of the power reactor were generally small (less than about 1 %) although this resulted from compensating effects. In the transport and reprocessing area the main problems resulted from the neutron activity of curium. Uncertainties in this area arose from inadequate knowledge of cross-sections and of the yields of (α, n) reactions. In the fuel fabrication area problems arose with the recycling of fissile material and gamma-ray emission was of particular importance.

French experimental work in the actinides area followed three main lines:

(1) analysis of spent fuel and irradiated nuclides

(2) physics experiments in critical assemblies

(3) checks of neutron and gamma-ray activities.

In the irradiated fuel analysis area a large number of measurements had been made on various types of fuel. In the LWR case there were some significant discrepancies in the C/E values for actinide concentrations: especially in the case of the curium isotopes. Corresponding adjustments had been made to the data library. Analysis of separated nuclide irradiations in PHENIX had given values of the capture cross-sections of several actinides with fairly good accuracy.

Physics experiments on critical assemblies included the measurement of the fission cross-sections of Am241 and Pu238 in fast reactor lattices. It was planned to carry out activation measurements of the capture rates of U236, Np237 and Am243.

Work on the checking of neutron and gamma emission rates included measurements on plutonium samples of various isotopic compositions and on irradiated samples.

USSR

Bobkov presented the paper NEACRP-A-335 which described the use of perturbation theory to assess the accuracy of actinide nuclear data required for fast reactor burn-up calculations. Actinides ranging from U232 to Cm244 were taken into account and sensitivity coefficients were calculated for two reactor cores using the perturbation sensitivity code PERS. These were in reasonable agreement with the coefficients given by Gandini (CNEN, RT/FI (77)1). Detailed data on achieved and required accuracies were presented and this indicated that, in general, the current accuracies of transactinide nuclear data do not provide the required accuracy for fast reactor burn-up calculations.

Japan

Hirota introduced the paper NEACRP-A-318 in which a sensitivity analysis of the build-up and burn-up of actinides in fast reactors was described. The code EIGENS, which had been developed for this analysis calculated the sensitivity coefficients for actinide production by solving the nuclide chain equations and their adjoints using the eigenvalue method and then combining these solutions with the time-dependent generalized perturbation technique.

Sensitivity analyses were carried out for two 1000 MWe fast reactors whose plutonium fuel came respectively from a BWR and the blanket of an FBR. These studies confirmed the usefulness of the eigenvalue method in solving the nuclide chain equations and also showed that the sensitivity coefficients for actinide production were strongly dependent on the type of plutonium fuel used. The results of the sensitivity analysis for the fast reactor fuelled with BWR plutonium indicated that priority should be given to improving the accuracy of the decay constants of Pu241 and Cm242, the capture cross-sections of Am241, Am243, Pu242 and Np237 and the fission cross-sections of Am241, Pu242, Np237 and Cm244.

USA

Hemmig presented the paper NEACRP-A-341 which described US studies of actinide formation and burn-up. The aims were firstly to characterize the neutronics describing actinide production and burn-up by conversion to fission products during operational exposure of power reactor fuel and secondly to understand the separation of actinides from one another and from fission products. The results of neutronic transmutation studies showed that substantial actinide depletion was possible, in principle, giving equilibria between production and fission utilization. There were, however, significant practical complications and a number of major issues were still under consideration including special fast reactors optimized for actinide utilization. These featured hard neutron spectra (average energy about 0.2 MeV) to maximize the fission-to-capture ratio and Am241 and 243 reductions of 60 % and 40 % respectively were indicated after a three-year burn-up cycle. Studies on the chemical partitioning of long-lived actinides from radioactive waste were in progress at ORNL and involved both experimental and calculational activities. This work was expected to produce flow sheets which had been at least partially verified by benchscale experiments. Maienschein reminded the committee that, in discussions at previous meetings, the feasibility of actinides partitioning had been in doubt. He felt that the situation had now changed significantly since the ORNL work (on separation by cation exchange chromatography) had now largely demonstrated the feasibility of partitioning.

Switzerland

Richmond introduced the paper NEACRP-A-339 which advocated the use of a uranium/plutonium/neptunium fuel cycle. The proposal was that the large amounts of Np237 produced in operating LWR's should be recycled in a GCFR with a neptunium-to-plutonium ratio of unity at start of life. In this case the total power produced by fission in the Np237 itself and in the Pu238 produced by neutron absorption in the neptunium amounted to 25 % of the total power of the reactor. An additional possibility was to mix the Pu238 with normal plutonium used in the fabrication of LWR fuel elements. This would increase the proliferation resistance of the fuel because of the increased neutron emission caused by spontaneous fission of Pu238 and (α, n) reactions with oxygen. The calculations showed that a 300 MWe GCFR could use the Np237 from about twelve 1 GWe LWR's and could produce sufficient Pu238 to denature the fuel from 2-3 LWR's. A further point of interest was that the build-up of plutonium in the GCFR reduced the rate of loss of reactivity so that the reactor could be operated with a reload interval corresponding to the maximum fuel burn-up.

In discussion several members commented that their studies had shown that the use of Pu238 to denature LWR fuel elements was ineffective because of its rapid burn-up.

Belgium

Debrue presented the paper NEACRP-A-331 which described experiments intended to produce data on the Pu241 and Am241 cross-sections. Measurements were made of the variation of the K-effective of a plutonium-fuelled core in the VENUS critical facility over a period of about ten years. About 80 % of the reactivity loss resulted from the formation of Am241 and it was estimated that the capture cross-sections of this nuclide in the thermal energy range could be estimated with an accuracy of $\pm$ 10 %. Some difficulties in the analysis of the experiment arose because of differences in the composition of UO_2 fuel batches used in the outer zone of the reactor.

UK

Askew tabled a note (in response to action 24 of the 20th meeting of the committee) giving the UK one-group actinide cross-sections averaged over the central spectrum calculated by FGL5 for the NEACRP LMFBR benchmark. He drew attention to the adjustment factors that had been applied to the differential cross-section data and, in particular to the factors in the range 1.5 - 2.1 applied in the case of the Am (n, γ) reactions. Further americium capture measurements made in ZEBRA were now in course of analysis.

Germany

Küsters reported on physics investigations in the field of thermal and fast reactor fuel cycles with main emphasis on the comparison of radioactivity inventory, thermal power, neutron and gamma radiation levels and criticality control in the out-of-pile stages of LWR, HTR,LMFBR and some alternate fuel cycle concepts. A review of the various technical problems together with an assessment of the rôle of nuclear data uncertainties in fuel handling, reprocessing and waste had been given at the Harwell Conference.

Euratom JRC Ispra

Rief reminded members of the substantial programme on actinides and waste disposal which was in progress at the JRC Ispra. Recent work had concentrated on burn-up calculations in connection with the recycle of actinides (together with diluents) in FBR fuel rods. The code ORIGEN was used to calculate the variation of linear power rating and burn-up rate as a function of irradiation time. This work had included the comparison of the ORIGEN one-group cross-sections with those given by a wide range of other sources. (This is given as Table 3 of the Ispra-Activities Report: NEACRP-L-202e).

Rief introduced the report NEACRP-A-342 which described the activity in progress at Ispra in connection with the monitoring of plutonium-contaminated solid waste streams. A general theory of radiometric assay had been elaborated and applied to the monitoring of the waste streams by passive gamma and passive and active neutron techniques.

Canada

Duret stressed the Canadian interest in the thorium fuel cycle whose behaviour was expected to be significantly different from that of the natural uranium cycle. High accuracy was required in the cross-section data since conversion ratios were close to unity. The effects of resonance shielding were significant (up to 15-20 %) but their influence on reactivity was small, partly because of the well thermalized neutron spectrum and partly as a result of compensating effects.

Italy

Farinelli mentioned detailed calculations of the possibility of neutronic burn-up of actinides in an LWR and an FBR. A rather realistic approach had been adopted, featuring the use of two-dimensional calculations and a detailed representation of two existing reactors: the Karlsruhe BWR and PHENIX.

In a general discussion of the actinides question the committee agreed that the most stringent requirements for data accuracy arose in the field of actinide transmutation. The main question at issue was whether currently available data was sufficiently accurate to allow adequate evaluation of the physics aspects of actinide burn-up. It was generally felt that the best existing data, which have been systematically adjusted to take into account integral data, are now reasonably adequate for this purpose. It was noted, however, that attention should be drawn to the existence of a number of older data sources (some based on purely differential data) which do not provide the accuracy required for actinide transmutation studies.

Hirota and Inoué agreed to prepare a document setting out the NEACRP's view of the current position concerning actinide production and burn-up and to make this available before the Second Advisory Group Meeting on Transactinium Isotope Nuclear Data (Cadarache, May 1979).

1.2 Neutron damage

Japan

Hirota introduced the paper NEACRP-A-319 which dealt with the use of the code GRAPE to simulate defect configurations in molybdenum with bcc structure and with the extension of the code to deal with the diamond structure and the hexagonal graphite structure so that defects in silicon and graphite could be treated. The extended code allowed the treatment of such phenomena as the anisotropy of the displacement energy and the motions of interstitials.

Belgium

Debrue presented an information note (NEACRP-A-333) concerning dpa calculations. He pointed out that dpa values are strongly dependent on the choice of model but that the ratios of dpa values calculated with different models are rather insensitive to the shape of the neutron spectrum. When selecting a model and data one should note that only the neutron energy dependence of the displacement cross-sections is of prime importance and it was important that the same conventions should be used for the analysis of irradiation data obtained in a materials testing reactor as for the utilization of these data by the power reactor designer. It was also important that test data which had involved irradiations of a given material in a number of different reactors over a period of years should all be reported in a consistent way.

US

Hemmig introduced the report NEACRP-L-220 which described work on neutron radiation damage to structural materials carried out at HEDL. This damage was generally sensitive to neutron energy and, since the test spectra in which data on radiation effects were obtained were normally different from the spectra to which components were to be exposed, procedures were needed for taking spectrum differences into account when correlating and applying the test data. The adopted procedure involved the combined use of two different types of computer calculation: (1) damage function analysis in which the computer was used to solve iteratively a set of integral equations in order to deduce the energy dependence of specific types of radiation damage and (2) the atomistic simulation of defect production and annealing in a metal. Both of these techniques were under active development.

Germany

Küsters presented two papers: NEACRP-L-209 and L-211. The first of these was concerned with a research project on the structural integrity of reactor components. The aim of this work was to quantify the safety margins against brittle fracture and generally to clarify the effects of neutron damage. Work described in the paper included (1) a comparison between experimental measurements of two-dimensional flux distributions in a 350 MWe PWR with distributions calculated by DOT3 which showed good agreement (2) two-dimensional calculations in (r, θ) geometry of the displacement rate distribution in a 1300 MWe PWR benchmark which indicated that about 80 % of total displacements were produced by neutrons with energies greater than 1 MeV. (3) a sensitivity study on the correlation of radiation effects between surveillance sample positions and the pressure vessel wall (4) prediction of radiation-induced changes of the properties of materials used in reactor structural components.

The second paper described benchmark experiments carried out on the reactor KAHTER for the determination of fast neutron flux. Foils of Rh 103 were irradiated in a number of axial and radial channels in the reactor and the results were compared with the predictions of the programme system GAMTEREX. There was good agreement between the measured and calculated fast flux values in the reactor core but, in the reflector, the calculation overestimated the Rh 103 reaction rates by about 10 %. This indicated that the irradiation damage in an HTR reflector was less than the predicted values. It was planned to carry out further measurements in an attempt to resolve this discrepancy.

Küsters additionally reported on work carried out at IKE Stuttgart to obtain calculational means for predicting the safety of pressure vessels. Samples were irradiated in research reactors and power reactors and the neutron spectra in the samples were calculated. Since these spectra differed from the spectra in the pressure vessel it was necessary to develop a transformation method to correlate sample damage with pressure vessel damage.

1.3 Blanket physics

France

Barré introduced the paper NEACRP-L-215 which described recent work at CEA in the field of blanket physics. The main points under investigation were "reflector savings", power distribution, plutonium production and the shielding role of the blanket.

The influence of the blanket on critical mass was determined by measuring the fundamental mode critical buckling and deriving the "reflector savings" from the critical size of the assembly. Analysis of a number of MASURCA and SNEAK cores with UO_2-sodium blankets showed good agreement between the measured values of reflector savings and the values predicted by CARNAVAL IV calculations. Transposition of these results to the case of a large power reactor indicated a bias factor of the order of $(250 \pm 200) \cdot 10^{-5}$ $\Delta k/k$ confirming the adequacy of the CARNAVAL IV system for problems of this type.

The agreement between calculated and measured power rate and reaction rate distribution in the blanket was not fully satisfactory, especially at the inner and outer boundaries. It was aimed to improve this situation by (1) giving closer attention to the treatment of heterogeneity (2) taking account of streaming effects and (3) taking leakage in the blanket into account when processing neutron cross-sections. Space-dependent cross-sections would be used in the interface zone between core and blanket.

The experimental programme of blanket studies on MASURCA would continue and it was also planned to carry out a further series of experiments on TAPIRO in the framework of the CEA-CNEN agreement.

Switzerland

Richmond reported on measurements of reaction rate distributions and small sample reactivity worths made in the core and depleted UO_2 blanket of a rodded GCFR lattice in the critical assembly PROTEUS. The axial distributions of the Pu239 and U238 fission rates and of the U238 capture rate had been measured from the core centre to the outer edge of the blanket. The results of the measurements were used to check the predictions of two-dimensional, 10-group diffusion theory calculations. The U238 fission rate distributions was well predicted but the calculated capture rate distribution was systematically about 2 % low in the blanket relative to core centre. The Pu239 fission rate distribution was well predicted through the core-blanket interface but progressively under-predicted with increasing depth in the blanket. The analysis was being repeated using two-dimensional transport theory. The results of measurements of the reactivity worths of a variety of small samples in the blanket were being analysed using exact perturbation theory.

1.4 Streaming and sodium void problems

UK

Askew presented the paper NEACRP-A-325 which summarized work aimed at establishing practical methods of deriving streaming-corrected diffusion coefficients for use in sodium-cooled fast reactor calculations. Both pin and plate geometries were considered. The general approach was to apply "streaming corrections" to the homogeneous diffusion coefficient for each of the principal directions of the lattice. An important feature of the lattices under consideration was that they contained void regions extending infinitely in two dimensions and that, in the case of buckling-independent treatments, such as those of Benoist, the diffusion coefficients parallel to such regions became infinite. The methods chosen for study therefore included buckling-dependent approaches. The codes studied included:

- The one-dimensional discrete Sn code WDSNST, which calculated buckling-dependent diffusion coefficents

- The two-dimensional discrete Sn code DOT which could be used to obtain radial diffusion coefficients for plate cells

- DIFFAX, a Würenlingen code developed to calculate diffusion coefficients for GCFR pin cells. The axial coefficient was calculated from the buckling-dependent, Köhler-Ligou formula and the radial coefficient from the buckling independent formula of Eisemann and

- DSLABA, which evaluates the buckling-dependent Köhler-Ligou formula for the radial diffusion coefficient of a slab cell.

The various codes were compared by calculations on a variety of plate and pin cells. In the case of plate cells it was concluded that the parallel diffusion coefficient could be calculated by WDSNST or DSLABA, which were in good agreement, while Benoist's formula, though accurate for a sodium-flooded cell, overestimated the diffusion coefficient correction for a voided cell. For the plate cells the streaming correction perpendicular to the plates could normally be neglected.

For the pin cells both radial and axial corrections were needed and these could be calculated by DIFFAX. Benoist's method underestimated the streaming corrections for fast reactor pin cells.

In the case of a power reactor cell typical of PFR or CDFR (Commercial Demonstration Fast Reactor) the axial corrections were of the order of 1.5 % (flooded) and 3 % (voided) with the radial corrections about half as great. The corresponding streaming correction to the maximum positive sodium void effect in CDFR was - 4 % out of a total heterogeneity correction of - 12 %.

Outstanding problems were (1) streaming effects could be very sensitive to simplified cell modelling procedures so that checks against reference calculations for three-dimensional models (e.g. Monte Carlo calculations) were needed and (2) the calculations has been carried out in broad groups and their validity should be tested by comparison with fine-group calculations.

France

Barré introduced two papers: NEACRP-L-219 and L-217. The first of these described the results obtained for diluent rods during the course of an experimental study of the SUPER-PHENIX control rods carried out in the "PRE-RACINE" core of the critical assembly MASURCA. Particular attention had been given to the effects of the differences in size and composition between the SUPER-PHENIX rods and the PHENIX rods which had been studied in an earlier series of measurements.

The results of the measurements were analysed by means of two dimensional diffusion calculations using CARNAVAL IV cross-sections. Large C/E discrepancies were noted in the case of diluent rods with high sodium and low stainless steel contents. A comparison of C/E values for sodium followers and stainless steel sub-assemblies obtained in the PRE-RACINE programme and in the earlier PHENIX-based programme showed similar discrepancies except for the PHENIX R3 core. This arose as a result of differences between the R3 and PRE-RACINE axial blankets. It was concluded that a complete analysis of the results of this programme could provide bias factors for both core zones of SUPER-PHENIX and these results were being used to define the SUPER-PHENIX critical enrichment. Since the bias factors were still significant further efforts were being made to improve the calculation methods.

The second paper (L-217) gave an analysis of sodium void measurements carried out in the PRE-RACINE programme in MASURCA. Voided cell cross-sections had been used in all the voided zones and directional diffusion coefficients had been generated by the Benoist method. Two-dimensional diffusion calculations were carried out using the 25-group CARNAVAL cross-section set and zone voiding reactivity changes were determined by eigenvalue differences and by first-order and exact perturbation theory. The experimental measurements were mainly concerned with the study of sodium voiding effects in a heterogenous core with a central blanket. The voiding was carried out in eleven substitution steps. The C/E discrepancies were small at the centre of the internal blanket but increased towards the edge of the core in both axial and radial directions. The use of directional diffusion coefficients led to a systematic improvement but did not remove the discrepancies. The influence of Pu240 content on the sodium void effect was checked by changing the Pu240 fraction from 8 % to 18 % in a 30 cm diameter zone. This showed that the C/E discrepancies given by CARNAVAL IV increased with increasing Pu240 content.

Germany

Küsters introduced the paper NEACRP-L-210 on the current status of sodium void reactivity predictions. This was largely based on previous NEACRP discussions but the results of some additional investigations had been included. The paper had been presented at the Gatlinburg Conference. Topics covered by the paper included reactor physics aspects of fast reactor safety, the target accuracy for the sodium void effect and the theoretical assessment of the sodium void effect. Under the latter heading a discussion was given of the sensitivity of the sodium void reactivity effect to nuclear data uncertainties. Analysis of the NEACRP LMFBR benchmark had indicated that basic nuclear data uncertainties could lead to errors of about ± 15 % in the estimation of the sodium void effect. Further topics covered in this area included computational errors and transport effects in whole core calculations, the treatment of control rods, cell heterogeneity and neutron streaming. It was concluded that there was an uncertainty of about 20 % in the prediction of the overall heterogeneity effect (including streaming) and that this led to an uncertainty of about 5 % in the total void reactivity effect. Temperature effects were briefly described and, in this case also, the uncertainty was taken as 20 % leading to an additional uncertainty of 5 % in the net sodium void effect. Consideration of burn-up effects had included a re-investigation of the effects of higher plutonium isotopes. The analysis of earlier SNEAK measurements with proper treatment of spectrum effects etc. had reduced a discrepancy of a factor 2 to about 25 %. It had also been shown that a similar discrepancy arising in the measurement of Pu240 effects in the PRE-RACINE programme could be correspondingly reduced.

In conclusion the overall uncertainty in predicting the effect of voiding the inner core zones of a power reactor was estimated as ± 25 % (1σ) on the assumption that appropriate calculation methods were used.

The Secretariat agreed to reproduce and distribute the sodium voiding status report and Küsters agreed to assist the Secretariat with the preparation of a cover note to be sent to the CSNI with a copy of the report.

UK

Askew presented the paper NEACRP-A-234 which outlined the fast reactor calculation methods available in the UK and described the application of these methods to the analysis of ZEBRA and SEFOR experiments. In the case of the investigation of subassembly heterogeneity effects in the SEFOR Doppler experiments three different levels of approximation were used in the treatment of heterogeneity i.e. (1) a cylindricalized sub-assembly (2) an improved sub-assembly representation in pin cluster geometry and (3) the inclusion of a boron carbide rod heterogeneity effect using a broad group treatment and a more detailed representation of the cell. Comparison of the magnitude of the various heterogeneity effects showed that the fuel pin effect was the most significant. With all corrections applied the C/E for SEFOR-1 and SEFOR-2 were respectively 0.94 ± 0.14 and 1.00 ± 0.14.

A similar investigation was made for a typical UK fast reactor design consisting of sub-assemblies with relatively massive wrappers. Calculations indicated that the fuel pin and wrapper effects were of similar magnitude in this type of design. The total heterogeneity effect was 5 % of the whole core Doppler constant. The influence of neutron streaming on the flooded core Doppler effect was found to be small because the change in the neutron diffusion coefficient due to streaming was small in the flooded core and because the leakage component was a small part of the total Doppler effect. The influence of the subassembly heterogeneity on the sodium void reactivity in a typical fast reactor had been studied using (1) a fuel pin cell with external sodium and wrapper material and (2) a homogenized fuel pin cell with the wrapper represented explicity. The combined effect of all heterogeneity corrections on the total maximum positive sodium void effect was about 2 %.

US

Till introduced the paper NEACRP-L-225 which reviewed the current status of US work carried out during the past 3 years on the Monte Carlo based validation of the ENDF/MC^2-II/SDX cell homogenization route. The homogenization procedure involved three steps: (1) modelling from 3D to 1D (2) the MC^2-II/SDX generation of the 30-group cross-sections for cell-averaged reaction rates and (3) either the Benoist or Gelbard formulation for anisotropic, cell-averaged diffusion coefficients. The paper described the detailed series of steps involved in testing against the high precision Monte Carlo calculation of the original 3D cell. When the Gelbard formulation for the diffusion coefficient was used the homogenization procedure reproduced very closely the correct result.

1.5 Nodal and coarse mesh codes

Japan

Inoué presented the paper NEACRP-L-204 which described an improved few group coarse mesh method for the calculation of three-dimensional power distributions in fast breeder reactors. This method had been developed by modifying Askew's one-group method so that it could be easily incorporated into conventional diffusion codes. It employed modified macroscopic cross-sections including group dependent corrections for coarse meshes. The improved method had been used as the basis of the three-dimensional diffusion code ICOM.

Test calculations had been carried out for the prototype fast reactor MONJU using three different control rod patterns. The results given by the improved coarse mesh method were compared with those of a fine mesh calculation (based on the CITATION code) and a conventional coarse mesh calculation. The improved method gave values of K-effective within 0.01 % of those given by the fine mesh calculation while the conventional coarse mesh calculation gave values differing by up to 1.1 % from the fine mesh values. Similar results were obtained in the calculation of control rod worths and power distributions with the improved coarse mesh method giving effectively the same results as the fine mesh calculation. The computing time for the improved method was, however 8 times shorter than that required for the fine mesh case.

Hirota introduced the papers NEACRP-L-207, L-205 and L-206. L-207 described the development of a three-dimensional neutron diffusion code series using the leakage iterative method which had been chosen because it eliminated the problem of estimating neutron leakages from a sub-region which occurs in the case of nodal and coarse mesh methods. In the leakage iterative method the reactor is divided into several layers along the z-axis and into several rectangular channels across the xy plane. A parallelepiped formed by a channel and a layer is called a block. A fine mesh difference approximation technique is applied only to the channels and layers so that it is not necessary to calculate the neutron fluxes at all fine mesh points in the core and the computing time is correspondingly reduced. The neutron leakage from a block is calculated by a fine mesh difference approximation and the numerical error due to discretization is therefore minimized.

Descriptions were given of the various codes developed using the leakage iterative method.

L-205 described a static core performance simulator for light water reactors known as SCOPERS-2. This code was generally based on the FLARE code but contained a number of modifications and improvements including a generalized nodal equation and migration kernel. Three-dimensional power distribution calculations for the reactor of the Japanese nuclear ship MUTSU were carried out using SCOPERS-2 and the results were in good agreement with those of corresponding CITATION calculations.

L-206 described an application of the finite element method to the solution of the three-dimensional neutron diffusion equation. A code known as FEM-BABEL incorporating the successive over-relaxation method and the use of a finite element mesh generator had been developed for this purpose. Testing of this code has been carried out by comparing its performance with that of the conventional finite difference CITATION code for (1) a homogeneous cubic reactor case that could also be solved analytically (2) a two-zone reactor with two energy groups and (3) a modified version of the IAEA three-dimensional PWR benchmark problem. In the first case the FEM-BABEL and CITATION solutions for K-effective both lay within 0.01 % of the analytical solution. For the second problem the K-eff values given by FEM-BABEL and CITATION agreed to within 0.2 % to 1.2 % depending on the mesh size but the computing time was shorter for FEM-BABEL. For the third problem it appeared that FEM-BABEL gave a better performance than CITATION in the calculation of power distributions in the region of the core-reflector interface.

It was concluded that the three-dimensional finite element calculations were acceptable from the point of view of precision and computing costs.

UK

Askew introduced the paper NEACRP-A-340 which described the use of the "kernel" method to calculate PWR power distribution on a coarse mesh. The basis of this method is a detailed solution of the spatial fission source in a uniform array of pins or channels for unit source in the pin or channel at the origin. This distinguishes it from the most usual application of heterogeneous theory in which the base Green's function is that for a pure moderator. Variations in fuel absorption cross-section are dealt with by "re-starting" neutrons at thermal energies. The base solution is obtained by Monte Carlo methods and includes both space and energy effects.

The method had been applied to a PWR with the aim of producing a coarse mesh model from which individual pin powers could easily be constructed. As a result of the smoothing influence of neutron migration it was possible to do this using a low resolution description of the fission source in the reactor. It was therefore envisaged that a coarse or nodal solution would be used for core follow studies and that a fine pin representation would follow using the earlier whole reactor solution as a fixed source.

Testing of the method was carried out by using it to calculate the channel power distribution in the Beznau I reactor. A simple two-dimensional kernel calculation was carried out with one cell per channel. The root mean square deviation between the kernel values and measured values was 2.5 %, similar to the discrepancies given by a two-group diffusion calculation with four meshes per channel. A further test involved the calculation of power distribution in an infinite chequer board lattice with zone enrichments of 1.80 % and 2.78 % U235. The results of the kernel calculation were compared with those given by 6-group homogenized diffusion theory and reasonable agreement was obtained between the two methods. These preliminary studies suggested that the kernel method might be more than competitive with few group, coarse mesh corrected or finite element techniques.

Germany

Küsters presented the paper NEACRP-A-238 which described the solution of the multigroup diffusion equation using a coarse mesh method based on local flux expansion within the individual nodes. The use of quadratic and cubic expansions led to the codes QUABOX and CUBBOX respectively. These codes had been tested successfully on the IAEA LWR benchmark. A particular feature of this work was that, in addition to the solution of three-dimensional power distributions, it also included the use of thermohydraulic models to represent correctly the influence of feedback effects from changes in the coolant conditions. This would allow the analysis of spatial effects in transients. It was also planned to introduce heterogeneous nodes so that the material properties within the nodes could be changed. This would allow burn-up effects to be studied.

Küsters reported on the three-dimensional finite element code DIFGEN produced at the University of Stuttgart. This code could be used to solve the neutron diffusion equation in two and three-dimensional non-orthogonal geometries. Either coarse or fine mesh calculations could be carried out according to the detail required. The basic elements were triangles and, in the third dimension, these could become triangular prisms or tetrahedra. The code had been tested successfully against the IAEA two- and three-dimensional LWR benchmark problems and also against HTR and FBR benchmarks.

Küsters also described work on homogenization and group condensation in coarse mesh calculations carried out at KWU. A self-consistent system of differential equations and boundary conditions for the determination of group parameters had been derived. The theory was based on the postulate that the integral reaction rates, average fluxes and average leakages were conserved. A unique feature of this theory was that flux continuity between adjacent homogenized regions was no longer postulated and the flux discontinuity was represented by an additional equivalent factor known as the heterogeneity factor. For the description of the interaction between rectangular assemblies it was necessary to use the directional dependence of the diffusion coefficients and also of the heterogeneity factors. The method has been checked against the BWR 2-group problem defined by Henry and Worley and the numerical solution of the equivalence equations reproduced the integral quantities within five digits.

USSR

Bobkov summarized the topics covered by a collection of USSR reports which he offered to hand over for circulation to committee members. In view of the practical difficulties of copying the full collection of reports it was agreed that those specifically concerned with nodal and coarse mesh codes would be copied and distributed to interested members.

Belgium

Debrue described a nodal code MERCATOR, which had been developed by Belgian utilities for use in fuel management. This was a two dimensional code in xy geometry and with two energy groups. It was known as a "modal nodal" code because the space dependence of the flux was described according to the various modes of the flux distribution. It had been tested against the Michelsen benchmark problem with excellent results.

France

Bouchard said that there had been no recent developments in France in the field of nodal and coarse mesh codes but reminded the committee that 1D and 2D finite element codes were included in the NEPTUNE code system which had been in use for some years.

US

Till presented the paper NEACRP-L-226 which described a nodal method under development at ANL for multidimensional, multigroup fast reactor analysis using diffusion theory. This method gave acceptably accurate results for realistic LMFBR problems with considerable savings relative to finite difference methods. The method could also produce accurate and computationally efficient solutions for LWR configurations such as the IAEA benchmark problem.

Following a general discussion on the topic of nodal and coarse mesh methods it was agreed that the performance of such codes was of particular importance to reactor operators and that it would be appropriate for the NEACRP to set up a specialists' meeting on this topic with emphasis on rapid methods for core follow studies. Askew agreed to make arrangements for this meeting to be held in Paris towards the end of 1979.

1.6. Fuel cycles

Italy

Farinelli introduced his paper NEACRP-A-337 which gave a preliminary evaluation of the technical aspects of INFCE. He pointed out the difficulty of separating technical and political issues in INFCE studies. This mainly resulted from the problem of quantifying the criteria used to evaluate fuel cycles. On the question of proliferation-resistance INFCE had been useful in giving an understanding that a scale of proliferation-resistance did not exist. In some cases one might say that a given cycle was more proliferation-resistant than another but, in general the evaluation would be subjective or largely dependent on local circumstances and would also be time-dependent, e.g. in the past it was thought that isotopic separation was not feasible but it was now clear that it would be relatively easy to start with a mixture of 20% U233 in U238 and produce weapons grade U233 by centrifuge.

Concerning the timescale of the INFCE studies it was generally agreed that technical remedies for proliferation were required during the next 20-25 years but very few of the new fuel cycles that were being proposed could have a large scale application on this timescale. The discussion had therefore been largely concentrated on the two original major lines: (1) the US position which advocates the once through cycle in LWR's on the assumption that there will be no shortage of uranium in the next 20 years and (2) the European and Japanese approach, which is more pessimistic concerning uranium reserves (and especially the free availability of uranium), and is therefore based on the reprocessing of LWR fuel and the use of plutonium to start up fast breeders. These two strategies were effectively the only ones to emerge at the end of the INFCE discussions. Future possibilities might be that the US would propose improved once-through cycles and that the Europeans would improve the proliferation-resistance of their reprocessing strategies. Farinelli presented NEACRP-A-336 which considered whether a global optimization of nuclear strategies was possible and concluded that this was not the case since the variety of criteria for selecting a strategy, their variation in time and space, the difficulty in their quantitative definition and the absence of many of the required data made long-term scenarios completely unreliable. Evaluation of the future rôles of fusion and hybrid reactors would have to be based on different criteria such as the importance of diversification of energy sources.

Canada

Duret introduced the paper NEACRP-A-330 which compared the long-term behaviour of a denatured uranium/thorium cycle with that of a normal thorium cycle using highly enriched uranium as topping material. The reactor calculations were carried out using the code WIMS. Calculations of fuelling histories for both fuel cycles showed that most nuclides reached their equilibrium concentrations relatively rapidly but that the concentrations of U234 and U236 continued to increase beyond the 15th recycle generation. Calculations of uranium utilization indicated that a natural UO_2 once-through cycle would use 160 tons of natural uranium per GWe installed per year while the denatured-uranium/thorium cycle and the highly enriched uranium-thorium cycle required respectively 40 tons and 20 tons. The most important result of the calculations was that both denatured and highly enriched uranium-thorium cycles required uranium with a U235 content greater than 20% for use as topping material. This implied that either cycle could only be used in connection with a safe international fuel cycle centre (IFCC).
If, in the absence of IFCC's the required fissile topping

could only be provided in denatured form then the denatured uranium/thorium cycle would rapidly approach the uranium cycle because of progressive replacement of Th232 by U238. The denatured cycle was not, therefore, particularly attractive.

Switzerland

Richmond presented the paper NEACRP-A-338 which described measurements in progress on a series of thorium-bearing fast reactor lattice configurations in PROTEUS. Measurements were made of the reaction rates Th232 (n,γ), (n,f) and (n,2n) U233 (n,γ), and U238 (n,γ) and (n;f) relative to the Pu239 fission rate. Configurations examined to date were (1) a benchmark lattice of PuO_2/UO_2 rods which gave a check on infinitely dilute thorium data (2) a lattice in which one third of the PuO_2 rods were replaced by ThO_2 rods to give a check on the shielded thorium data, and (3) a column of ThO_2 at the centre of the PuO_2/UO_2 lattice which was the first of a series of heterogeneous configurations intended primarily to give a check on calculation methods. Further measurements would be made in an axial ThO_2 blanket and in a central column and axial blanket of thorium metal.

The results of the measurements to date had been compared with the predictions of calculations based on ENDF/B-4 data. For the two lattice configurations C/E values of 1.00 - 0.99 were obtained for the Th232 capture rate and the U233 fission rate. For the Th232 (n,2n) reaction rate the C/E values were in the range 0.87 - 0.94 but this poor prediction may have arisen because the (n,2n) reaction occured only in the top two groups of the 28-group structure used in the calculations. The U238 capture measurements indicated the usual ENDF/B-4 overprediction of about 5%. For reaction rate measurements made on the axis of the central column the C/E values were generally within 1-2 % of the lattice values. Measurements of radial reaction rate distributions through the ThO_2 column were analysed using spatially dependent cross-sections which were prepared by detailed modelling of cells extending through the interface region. These calculations predicted the shape of the thorium capture, thorium fission and plutonium fission distributions within experimental errors of 1-2 % but, for U238 capture the calculated distribution showed errors of the order of 6-10 %.

US

Till introduced the paper NEACRP-L-224 which gave initial results of integral measurements made in ZPPR assemblies 8A and 8C. The former of these was a reference plutonium-uranium oxide heterogeneous configuration and the latter was identical except that thorium was loaded into the central blanket. Reaction rates were measured in the reference case relative to Pu239 fission and the results were analysed using calculations based on ENDF/B-4 data. For U235 fission the C/E values were of the order of 1.03 while, for U238 capture, values of the order of 1.09 were obtained, corresponding to those quoted in the previous paper (A-338). In the case of Th232 capture the C/E values were consistently high in the fuel zones giving values of the order of 1.08. The Th232 capture results in the blanket zones were questionable because the cross-sections were averaged over the asymptotic blanket spectrum. The corresponding results for assembly 8C gave a C/E value for Th232 capture (relative to Pu239 fission) in the thorium blanket of 1.033 ± 0.008. Measurements of the reactivity worths of materials substituted at the centre of the central blanket led to C/E values of the order of 0.9 for both assemblies. These low C/E values were linked to the underprediction of the central flux. Measurements of Doppler reactivity changes indicated C/E discrepancies of 20 - 30 % for uranium oxide and thorium oxide samples in both assemblies. Measurements of sodium void reactivity showed that, in the central blanket, this was 6% lower in the thorium case than in the reference system. The calculated value, however was 20% lower. The reasons for this discrepancy were under investigation. The measurements on thorium-bearing lattices were continuing.

Germany

Küsters described studies related to alternative fuel cycles and reactor concepts in progress at KWU. Topics investigated were: (1) Potential improvements to the once-through fuelcycle for light water reactors (PWR). (2) Light water reactor (PWR) with Th-U-cycle and uranium recycle. (3) The pressure vessel type heavy water reactor on the natural and slightly enriched uranium once-through cycle. (4) The conceptual design of a quasi-homogeneous pressurized heavy water reactor for operation on a closed Th-U233 fuel cycle.

Küsters also described KWU work on plutonium recycle in LWR's. The demonstration of commercial plutonium recycling had begun in 1972 following work on the design and fabrication of mixed oxide fuel assemblies and the irradiation of these in a number of KWU reactors. Fuel and core performance in the demonstration programmes had been very satisfactory. Sufficient evidence had been obtained to allow large-scale use of recycle fuels in LWR's. Design studies for large-scale commercial recycling had involved the investigation of various recycling modes including cores with both uranium oxide and mixed oxide fuel assemblies and "plutonium burners" in which all assemblies were of mixed oxide with a single plutonium enrichment.

Netherlands

Debrue introduced the paper NEACRP-A-332 which reported on the Dutch Fission Product Nuclear Data Project. The purpose of this project was to obtain neutron cross-sections for the prediction of the effects of fission products in fast power reactors. For this purpose neutron cross-sections were evaluated and, from these, 26-group constants were calculated and adjusted to fit integral measurements performed at STEK (Petten) and CFRMF (Idaho). Recently, adjustments had been applied also to evaluated point cross-sections. The project was performed in the framework of a collaborative agreement between Germany, Belgium and the Netherlands on fast breeder reactor development. Results to date indicated that the contribution of integral measurements to the present status of fast fission-product capture cross-sections was particularly important. For most nuclides the requested accuracy could only be reached by using these data in an adjustment process. The contribution of STEK integral data was especially significant.

France

Bouchard presented the paper NEACRP-L-214 which described work that had been in progress for some years on the development of low enriched UO_2 plate fuel (Caramel) for research and test reactors. This was intended to replace (for reasons of non-proliferation) the highly enriched uranium fuel normally used for such reactors. In producing the Caramel fuel the aim had been to maintain the same performance level and the same safety and reliability standards as in existing reactors without substantially increasing the fuel costs. The Caramel fuels used slightly enriched uranium because of the high density of the uranium oxide. The reactivity control capacity of the core was adjusted to the burn-up so that the cycle length remained the same as in current reactors, the average burn-up being limited to 30'000 MWD/t. The enrichment was maintained below 10 % and this led to conflicting requirements since (1) an economic fuel cycle cost required high reactivity which implied thick plates and wide water channels and (2) high performance required thin plates and narrow water gaps. The appropriate compromise between these conflicting requirements varied from one reactor to another. The research and development work associated with the Caramel programme also included studies for low and medium power land-based plants and merchant naval propulsion plants.

Bouchard introduced the paper NEACRP-L-218 which described the application of fuel history data to reprocessing plant input inspection. The paper pointed out that the use of simplified burn-up calculations in conjunction with a limited amount of data on the history of the irradiated fuel made it possible to predict the composition of the fuel to a very good approximation. Combination of these data with the results of destructive and non-destructive measurements made on fuel samples on entry into the reprocessing plant provided chacks on the accuracy of the predictions of fuel composition. Particular attention had been given to the use of the gravimetric balance method of determining plant input uranium and plutonium masses. Checks carried out during the reporcessing of LWR fuel at the La Hague plant had confirmed the accuracy of this method. As a result it would be possible to reduce the number of more complicated analyses (e.g. estimation of neodynium) and hence to reduce the costs of the determinations of input material balance.

In discussion it was generally felt that the situation in the field of isotope correlation techniques required some clarification and that it would be appropriate for the NEACRP to give further consideration to this area of work in the general framework of safeguards studies. It was agreed to include "Use of isotopic correlation techniques for the determination of fuel exposure history" as a new topic on the agenda for the 22nd meeting of the committee.

1.7. Miscellaneous topics

Netherlands

Debrue presented the paper NEACRP-A-343 which described studies of the neutronics characteristics of the Belt Screw-Pinch Reactor. Transport calculations had been performed in the toroidal belt geometry using albedos and absorption data obtained from a one-dimensional Sn calculation. For this purpose the code ANISN had been modified to produce the version ANISN-ALB and a new code known as FURNACE had been developed to perform multiple reflection calculations in toroidal belt geometry. Initial experience with these methods showed that they could give a good insight into the neutronics and photonics of a toroidal fusion reactor.

Germany

Küsters mentioned investigations of a number of CTR blanket designs which used beryllium as a neutron multiplier to obtain an adequate breeding ratio in systems with low lithium and tritium inventories. Measurements had been made of the neutron multiplication factor of beryllium and the measured results were about 25 % lower than the calculated values. This cast serious doubts on the suitability of beryllium as a neutron multiplier in CTR blankets.

USSR

Bobkov introduced the paper NEACRP-A-329 which gave an outline description of the adjusted data set OSCAR 76. This was based on a set of evaluated integral experiments and a number of evaluated nuclear data files. Group cross-sections for U235, U238, and Pu239 fission and capture were taken from the UKNDL file and used as a basis for the adjustment. Cross-sections of structural materials were obtained from Nulcear Data Centre evaluations. The remaining group cross-sections and all self-shielding factors were taken from the BNAB-20 data set. The differential data were adjusted using the results of the integral experiments. The data set had been checked by calculations on about 100 integral measurements and details of the C/E deviations were given in the paper. These were generally of the order of 1-3 % but larger discrepancies (C10-12 %) occurred in the case of F8/F5 and in the reactivity worth ratio of B10 to U235. The data set had been used for power reactor calculations and had given good predictions of critical mass, reactivity coefficients, sodium voiding and breeding ratio.

Kazansanskij tabled a paper (in Russian) describing measurements on critical assemblies with unit k-infinity. The first of these was a uranium metal assembly on which measurements were made of k-infinity and the reaction rate ratios C8/F5, F8/F5 and F5/F9. Corrections had been applied for heterogeneity and for the presence of structural materials. Agreement between measured and calculated results had been generally within experimental errors. A greater discrepancy arose in the case of the ratio F8/F5 but this was reduced following correction for fission chamber wall effects. Similar measurements (including neutron spectrum determinations) had been made on a UO_2 assembly with unit k-infinity. He offered to make data on this assembly available as the basis of an international benchmark exercise in which other memebers would carry out their own cell calculations of the unit k-infinity lattice. The committee welcomed this offer and Kazanskij agreed to send the data to the Secretariat for distribution to members.

Bouchard mentioned similar French measurements on a unit k-infinity UO_2 lattice and agreed that he would also send the appropriate data to the Secretariat for distribution.

UK

Askew presented the paper NEACRP-A-323 which discussed the derivation of spatially averaged cross-sections suitable for use in transport theory whole reactor calculations. The spatially averaged cross-sections were derived from transport theory calculations for a cell or supercell in which the fine structure was represented. The method was to average the cross-sections for the region to be homogenized so that the reactivity of the cell or supercell was the same for the fine structure representation and the homogenized region representation. This was achieved by spatial averaging with the neutron flux calculated for the fine structure representation cell and the adjoint neutron flux calculated for the homogenized region cell. The method was applied to the homogenization of control rod fine structure for use in reactor calculations. A simple cylindrical representation of a control rod consisting of an inner absorber region surrounded by a coolant and structural material region was homogenized, the homogenization calculation being made for a supercell containing the control rod region surrounded by core material.

2. TOPICS CARRIED OVER FROM PREVIOUS MEETINGS

2.1. Heterogeneous LMFBR cores

Japan

Inoué introduced the paper NEACRP-A-320 which described a comparison of the properties of a 1000 MWe homogeneous core with those of a selection of 12 heterogeneous cores. The homogeneous core had the same fuel pins, plutonium vector and linear heat rating as the prototype fast reactor MONJU. In the heterogeneous cores the fuel and control rod assemblies were the same as those of the homogeneous core and the core volume was equal to that of the homogeneous core. The 12 heterogeneous cores were dividided into two equal groups: those with internal axial blankets and those with internal radial blankets. Calculations were made of the values of breeding ratio, doubling times and sodium void reactivity, control rod reactivity worth and thermohydraulic characteristics for the various cores under investigation. On the basis of this data it was concluded that the heterogeneous cores giving the best breeding performance were (1) a core with an internal axial blanket about 30 cm thick and (2) a core with several thin internal radial blankets. A comparison of the characteristics of these two cores with those of the homogeneous core showed that the heterogeneous cores had greater breeding ratio, shorter doubling time, larger plutonium inventory, lower sodium void reactivity, lower burn-up reactivity, smaller control rod reactivity worths (when the same control rod pattern was used for both types of core) and a larger coolant flow rate.

Hirota introduced the paper NEACRP-L-203 which described an experimental study made on FCA Assembly VII-3 which had an internal blanket at the mid-plane of the cylindrical core. Mesurements were made of criticality, sodium void worths, sample reactivity worths, fission rate ratios and reaction rate distributions for integral blanket thicknesses in the range 20-40 cm and for varying U238 and plutonium contents of the blanket.

The measurements of the axial distribution of sodium worth indicated a trough in the region of the interface between the core and the internal blanket and also showed that the void worth at the centre of the blanket was more positive than that in the core. The sodium void worth was not sensitive to the composition of the internal blanket.

The void worth in the internal blanket increased with increasing internal blanket thickness but the void worth in the core zone was reduced by about 35 % as the internal blanket increased from 20 cm to 40 cm.

Diffusion theory calculations based on the JAERI Fast Set Version II were carried out for the case with an internal blanket thickness of 30 cm. Good agreement was obtained between calculation and experiment for the axial distributions of Pu239 fission rate and Pu sample worth. The calculation tended to overestimate the U238 fission rate and to underestimate U235 fission and U238 capture rates in the internal blanket relative to the corresponding reaction rates in the core zone. The axial distribution of the sodium void worth was fairly well predicted but there was a discontinuity of more than 20 % in the C/E value at the interface of the internal blanket and the core zone.

Barré presented the paper NEACRP-L-215 which indicated the current status of work aimed at improving the characteristics of fast breeders. Initial parametric studies concerned with optimizing the rate of introduction of fast breeders were aimed at the reduction of fuel inventory and had been carried out using cores of the SUPER-PHENIX type with the constraint that the fuel pellet diamteter, subassembly size and core height were all fixed. These studies had, in particular, confirmed an earlier result that the use of 3 enrichment zones rather than 2 was not worthwhile because the reduction in fuel inventory was only about 3% and this did not compensate for the additional problems related to fuel fabrication and management.

Studies of heterogeneous cores, also aimed initially at the reduction of critical mass, had included a core with an axial internal blanket and a core with two internal radial blankets. Neither core contained control rods. Initial analysis using one-dimensional cylindrical calculations indicated that a heterogeneous core with a fertile to fissile sub-assembly ratio of 0.16 had a critical mass 3% greater than a corresponding heterogeneous core but a 20 % shorter doubling time because of the improved breeding gain which was 0.07 greater than the homogeneous core value.

Further studies would involve two-dimensional calculations, control rod location studies and investigaton of sodium void effects.

Belgium

Debrue mentioned work being carried out at Belgonucleaire in collaboration with Interatom in connection with the design of the 1300 MWe SNR2 reactor. Studies were made of the adjustment of the lay-out of breeding zones to ptimize the power distribution and consideraton was also given to plutonium inventory, core size and breeding properties. In these studies the fraction of fertile material in the core was held to 25-30 % of total core volume and a start-of-life composition with no control rods was assumed. Comparison with a corresponding homogeneous case indicated that the heterogeneous system had a plutonium inventory 12% greater, a breeding ratio 5% greater and a doubling time 8% shorter.

Safety studies had suggested that the behaviour of the two cores was similar from the safety angle.

<u>US</u>

<u>Till</u> introduced the paper NEACRP -L-223 which gave a comparison of C/E values obtained for a series of conventional and homogeneous assemblies in ZPPR using consistent methods and data. The assemblies investigated had similar critical masses. Analyses of the results was carried out using rz diffusion theory and ENDF/B Version IV data in 28 energy groups. After the application of corrections for streaming, transport and a number of minor effects the k-eff C/E values for all assemblies were of the order of 0.98.

Extensive reaction rate distributions for Pu239, U235, and U238 fission and for U238 capture were measured in all assemblies. In assemblies 7A and 7B, which had the most heterogeneous fuel distributions, the U235 fission rate showed a "flux tilt" in the sense that C/E values ranged from about 1.00 in the centre to 1.04 in the outer core ring. The application of a variety of corrections succeeded only in halving this tilt. Some studies were made of the effect of cross-section alterations on the shape of the flux distribution in assembly 7A. It was concluded that the major sensitivities were to U238 capture and inelastic scattering which gave respectively three times and five times greater effects in the heterogeneous case than in the homogeneous case. It would therefore be possible to postulate changes in these data which would account for the residual flux discrepancy. In assemblies 7A and 7B the U238 fission rates varied by as much as a factor of two between adjacent fuel and blanket subassemblies. Diffusion theory calculations gave relative errors of 25% and the use of transport theory reduced the differences to 11%. Improved multigroup cross-section processing which treated coupled core and blanket cells reduced the discrepancy to 2%. A study of the ratio of U238 capture to Pu239 fission in the various regions of the heterogeneous assemblies gave C/E values of the order of 1.1 in the fuel regions and 1.07 in the internal blankets. The use of transport instead of diffusion theory made little difference to these discrepancies. Studies of control rod worths indicated C/E values of the order of 1.08 for the homogeneous assemblies and 1.00 for the heterogeneous. Application of a variety of corrections to the calculated control rod worths generally improved the C/E values. Measurements of sodium void reactivities showed that, for zones near the centre of assembly 7A, the positive component of sodium worth was reduced relative to the corresponding values in homogeneous cores.

Till presented the paper NEACRP-L-222 which summarized a detailed set of control rod interaction experiments carried out in heterogeneous cores in ZPPR, The interactions in heterogeneous cores were much greater than these in the corresponding homogeneous cores. In a particular case the value of two control rods in a ring changed by 340% according to whether the remainder of the ring were empty or filled with rods. All of these very large interaction effects were well predicted by routine diffusion theory calculations.

In a general discussion members stressed the continued importance of heterogeneous core studies and it was agreed that this topic would be included on the Agenda of the next meeting of the Committee.

2.2. Establishment of power peaking margins.

Japan

Inoué presented the paper NEACRP-L-208 which described the development of a computer programme for optimal control rod programming of a BWR with the aim of maximizing cycle length under the various operational constraints. A particular feature of this approach was that a realistic three-dimensional model of the core was used. The scale of the problem was reduced by dividing it into two stages: (1) an inner loop optimization to determine the control rod positions that minimizedthe mean square difference between the power distribution and the target distribution at each burn-up step and (2) an outer loop optimization to modify the target power distribution so as to minimize the residual control rods at the end of cycle. Test calculations carried out on a reference core indicated core lifetimes longer than those obtained using Haling's method.

France

Bouchard introduced the paper NEACRP-L-221 which indicated the current status of a PWR protection system and described proposed improvements. The existing system was a ΔT system which protected the core from overpower conditions and DNB. A trip was initiated when the power level equalled a power limit which was a function of temperature and pressure and a parameter depending on the difference between the power operation in the upper and lower halves of the core. The latter parameter (known as the axial offset-AO) was measured by out-of-core detectors, each having two vertical sections. Nuclear design calculations were used to produce a 3D power distribution and to correlate the parameter AO with the total power peaking factor and the enthalpy rise hot channel factor. Calculations were carried out for normal operation, including normal transients, and an envelope was drawn over the "calculated" points to represent an upper bound envelope on local power density as a function of vertical position in the core. This was an input to the LOCA analysis which had to satisfy the corresponding safety criteria, Class II events such as control rod malfunction and operator errors were also simulated.

Improvements to the protection system had been made possible by the use of out-of-core detectors with several vertical sections giving a more detailed picture of the axial power distribution and microprocessors which could deal with complex algorithms on a short response time. The improved system allowed the actual power distribution to be used to determine the DNBR on an on-line basis. It enhanced the level of safety in the sense that, by the use of more accurate instrumentation it allowed the protection system to include the effect of adverse power shapes much more effectively.

In discussion the importance of this topic was emphasized and it was agreed that it should remain on the agenda for discussion at the next meeting of the committee. Members agreed to make efforts to obtain the maximum number of contributions to this discussion.

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. LMFBR Benchmark

Till tabled a number of documents relating to the LMFBR benchmark exercise. These were the papers NEACRP-L-227 and A-344 plus four short synopsis documents ("Radial Distributions", "Central Control Rod Worth", "Central Worth Discrepancy", "Proposed NEACRP Comparison Calculations") together with "Plots and Tables of Contributed Solutions".

Till summarized the main conclusions of the report L-227 which were as follows:

(1) There were large differences between the radial fission rate and worth distributions given by the various solutions. These distributions appeared to be sensitive to small variations in the k-infinity difference between inner and outer core zones produced by the specified enrichment difference between these zones. In general the adjusted sets showed much smaller k-infinity differences between the two zones and a correspondingly greater flux drop at the edge of the inner zone. The flux drops predicted by the various national solutions ranged from 0.75 to 0.9. It was also noted that participants whose solutionsdisagreed on the 1250 MWe benchmark had each obtained good agreement with experiment on 300 MWe size critical experiments.

(2) There were very large differences (amounting to a 45% spread) in the various calculated values of the worth of a single central control rod. These differences were not well correlated with the calculated values of central boron worths.

(3) In general there were strong correlations between k-eff, k-infinity, breeding ratio and reaction rate ratios but the correlations between central worths, control rod worth, leakage and safety parameters were weak.

(4) There were large variations in the structural material cross-sections and in the corresponding capture rates. These appeared to be due as much to the processing methods as to the basic data files.

(5) In the adjusted data sets the integral parameters had considerably less scatter than in the non-adjusted sets but the scatter in the calculated cross-sections was about the same for both adjusted and non-adjusted sets.

Till introduced the paper A-344 which aimed to take the cross-sections from FGL5, CARNAVAL-III and ENDF/B-IV and to relate the differences in integral parameters to the differences in the cross-sections. Relative to ENDF/B-4 the adjusted sets gave k-eff values about 3% higher and values of the maximum to minimum inner core power ratio 8-11 % higher. Control rod worth relative to fuel was 21% higher for FGL5 and 34% higher for CARNAVAL-III. Detailed calculations of the changes in k-eff given by the listed cross-section differences showed that these accounted for about 80% of the k-eff difference between ENDF/B-4 and FGL5 for the benchmark calculation. Similar calculations carried out for ZPPR-2 (350 MWe size) and ZPPR9 (700 MWe size) indicated that sensitivities for power distributions and control rod worths for Assembly 9 were larger than those for Assembly 2. This should also be the case for some other properties such as central reactivity worths. The sensitivities for Assembly 9 approached those of the 1250 MWe benchmark so that only a small extrapolation of the integral data would be needed. Analysis of ZPPR-9 and BIZET experiments would be carried out to test the performance of data libraries in large core situations.

A particularly puzzling discrepancy concerned the differences in leakage and diffusion coefficient between the various solutions. The current situation was that smaller diffusion coefficients were predicted in the solutions with higher leakages. This totally inconsistent situation was being further investigated.

Following discussion members agreed to continue with their detailed analyses of the sources of discrepancies in the benchmark and to bring forward their results at the next meeting of the committee where this topic would be included on the Agenda.

Küsters agreed to present the paper NEACRP-L-227 at the next meeting of the CSNI and the Secretariat agreed to inform him of the time and place of this meeting.

The committee expressed its appreciation of the excellent work carried out by ANL in the analysis of the benchmark results.

Askew presented the paper NEACRP-A-322 which gave a comparison of neutron diffusion theory codes using the SNR 300 benchmark model. An earlier investigation had compared a variety of diffusion codes but all except one of these had been mesh-edged finite difference (MEDF) codes. The current paper took the original results and added calculations performed by the UK codes TIGAR and SNAP, both of which used mesh-centered difference formulae (MCDF). All the calculations with the MCDF codes TIGAR, SNAP and CITATION showed good agreement in k-eff when this was calculated for TIGAR and SNAP at fine mesh using a neutron balance edit. The errors in the TIGAR and SNAP calculated values for k-eff for the two dimensional benchmark problems were found to exhibit a good linear relationship with the square of the mesh interval. Similar relationships existed for the other codes used in the benchmark although there was some evidence of rounding errors in the fine mesh results quoted in earlier work for the MEDF code DIXY. When the k-eff values were extrapolated to correspond to an infinite mesh the results obtained using the MCDF codes TIGAR, SNAP and

CITATION were found to agree well with those given by the MEDF codes HEXAGA II, TRIBU and DIXY. This agreement indicated that no significant errors arising from the finite difference approximation but independent of mesh spacing were present in these calculations.

In discussion Rosén mentioned that it would be useful to Data Bank users if documents such as A-322 could be given an L distribution. Askew agreed to investigate whether his paper could be issued on an L distribution and inform the Secretariat.

4.2. Hydrogen entry benchmark

Küsters reported that little additional material had been received since the previous NEACRP meeting and that it was not felt to be appropriate to organize a specialists' meeting to discuss the results of this benchmark. Richmond agreed to contact Kiefhaber in connection with a review of the results.

4.3. Follow-up Dynamic Benchmark Exercise

Küsters proposed that the committee should set up a dynamic benchmark exercise as a follow-up to the benchmark discussed at the joint NEACRP/CSNI specialists' meeting held at Munich in 1975. He prepared a proposal during the course of the meeting (see Annex 6) and this was accepted by the committee. Küsters said that he would appreciate assistance from members in organizing the benchmark. He agreed to circulated to members a list of participants in the previous benchmark and members agreed to check what assistance could be provided.

The benchmark was further discussed in connection with the IAEA activities report (see Section 5.3).

5. GENERAL

5.1. Highlights of recent meetings of interest to NEACRP

SMORN II (Gatlinburg, September 1977)

Since the proceedings had already been published and since a number of NEACRP members had attended SMORN II, the meeting was not further discussed.

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

It was expected that the proceedings of this meeting would be published shortly. The Secretariat agreed to ask the organizer of the meeting (Böckhoff) to distribute to members a summary of the conclusions of the meeting. It was also agreed that, in general, the organizer of a specialists' meeting sponsored (or co-sponsored) by the NEACRP should submit a report on the conclusions of the meeting to the next session of the NEACRP before making the proceedings public. Johnston confirmed that this point was being included in the "Guide lines for organizers of NEACRP specialists' meetings".

NEA Conference on Neutron Physics and Nuclear Data for Reactors and other Applied Purposes (Harwell, September 1978)

Küsters expressed the view that the meeting had been overloaded with review papers which had amounted to about 50% of the total number of papers. He also felt that the organizers of the meeting had not paid sufficient attention to the conclusions of earlier meetings on the requirements for nuclear data. Most of the data requirements expressed at Harwell had already been put forward at earlier conferences and effectively no new requests had emerged.

NEACRP Workshop on Iron Shielding Benchmark Analysis (Paris, December 1977)

Rief reported that the meeting had concentrated mainly on the iron and sodium benchmark measurements, contributions had been made by Japan, Switzerland, Euratom, Italy and the UK. The Japanese delegate presented the results of recent iron and sodium measurements and concluded that two-dimensional calculations were needed to obtain satisfactory agreement between calculated and measured fluxes. No sensitivity analysis was presented. The Swiss contribution included a full listing of the results of PROTUES shielding measurements in iron and stainless steel together with their theoretical interpretation by 1 D and 2 D Sn calculations. The calculated results were in reasonable agreement with the measured values for penetration depths up to about 60 cm. The UK contribution was a preliminary paper by McCracken setting out his approach on data adjustment and giving some initial adjustment results obtained using ASPIS measurements. He also gave a detailed identification of shielding data requirements. On the Euratom side the first results of experiments on EURACOS were presented.

5.2. Specialists' meetings planned or proposed

Third Specialists' Meeting on Reactor Noise (SMORN III)

It was agreed that the SMORN III meeting would be held in Japan in 1981. The Secretariat agreed to write to the IAEA informing them of the meeting and inviting IAEA participation and also to write to the CSNI inviting them to co-sponsor the meeting.

International Symposium on Fast Reactor Physics (Aix-en-Provence) September, 1979)

This Symposium was initially discussed in connection with the IAEA Activities report (see Section A.5.3.).

The Secretariat agreed to distribute the IAEA Information Sheet on the Symposium to NEACRP members with an explanatory cover note.

Specialists' Meeting on Homogenization Studies (Lugano, November 1978)

Discussion concerning the organizational problems of this meeting is noted in Section A.5.3.

NEANDC Specialists' Meeting on the Cross-Sections of Fission Product Nuclei (Bologna, 1979)

The committee felt that it would be useful to have a statement from the organizer of this meeting giving the current status of fission product data evaluation and indicating what changes have taken place since the previous meeting on this topic (Petten, 1977). Fuketa agreed to transmit this request to the Chairman of the NEACRP before the date of the Bologna meeting.

NEANDC Specialists' Meeting on the Capture Cross-Sections of Important Fissile and Fertile Isotopes (Argonne, 1980)

The committee felt that, at the date proposed, a small working party meeting would be more appropriate than a meeting with substantial participation from data users. It was also felt that, as in the case of the Bologna meeting, it would be appropriate for the organizer to compare the status of these data with that at the previous meeting on this topic.

Specialists' Meeting on Calculation of 3-Dimensional Rating Distributions in Operating Reactors (Paris, November 1979)

Following discussion of nodal and coarse mesh methods (see Section B.1.5.) it was considered appropriate to set up a specialists' meeting on this topic. Askew agreed to make the necessary arrangements.

5.3. Progress with the NEACRP Book on the Status of Fast Reactor Physics

The Shielding chapter was available in its final form and the revised version of the Blanket Physics chapter had also been received. The remaining four chapters were not yet available. It was agreed that individual chapters would be published separately as they became available and subsequently combined to produce the book. A revised timetable for the production of the book was agreed and members undertook to carry out the various actions listed as items 42 to 50 of Annex 2.

5.4. OTHER ACTIVITIES

The committee visited the JAERI and PCN laboratories on the afternoon of 8th November.

The host organization offered a reception on 6th November and a dinner on 8th November.

ANNEX 1

LIST OF PARTICIPANTS

Members

DEBRUE, J.	CEN Mol, Belgium
DURET, M.F.	AECL Chalk River, Canada
RIEF, H.	ESIS Ispra, CEC
SILVENNOINEN, P.	TRCF Helsinki, Finland
BARRE, J.Y. (Chairman)	CEA Paris, France
BOUCHARD, J.	CEA Cadarache, France
KUESTERS,H.	KfK Karlsruhe, Germany
FARINELLI, U. (Vice Chairman)	CNEN Casaccia, Italy
HIROTA, J.	JAERI Tokai, Japan
INOUE, T.	PNC Tokyo, Japan
RICHMOND, R. (Secretary)	EIR Würenlingen, Switzerland
ASKEW, J.	AEE Winfrith, UK
HEMMIG, P.B.	USDOE Washington, USA
MAIENSCHEIN, F.C.	ORNL Oak Ridge, USA
TILL, C.	ANL Argonne, USA

Observers

TSUKADA, K.	NEANDC
FUKETA, T	NEANDC
KAZANSKIJ, Ju. A.	USSR
BOBKOV, Ju. G.	USSR

Secretariat

ROSEN, J.	NEA Paris, France
JOHNSTON, P.	NEA Paris, France

ANNEX 2

PROPOSALS FOR THE CONTINUATION OF SECU ACTIVITIES

1. The SECU activity is important and useful and the results so far obtained are encouraging.

2. In principle, it would be very desirable to collect and disseminate the users' experience on applications, improvements, new versions and variations, implementation on different computer environments for all the codes that are serviced by the Data Bank, or at least for those of more common use.

3. This generalization of SECU, even as a long range objective, cannot be implemented for all areas and for all codes with the same depth, since the most effective servicing requires a good knowledge of the code and an appreciation of the physical problem by the bank, as demonstrated by the past SECU experience.

4. It is therefore proposed to direct the SECU activity along two parallel lines: on one side the continuation of intensive studies of the most widely used codes in some selected areas; on the other side, an attempt to generalize a lower level SECU activity to all codes by an automated or semi-automated procedure.

5. Also in consideration of the transient produced by the transfer of the former CPL to Saclay, it is recognized that new subject areas should not be added to SECU for extensive studies, but a vigorous follow-up of the two areas of shielding and cross-section processing codes should be pursued. In addition to this, the code RELAP should be SECUed as a first step into the safety area.

6. The feasibility of generalized collection and dissemination of users' experience should be explored by looking at the possibility of computer-assisted tracking of code requesters and users, of selective dissemination of experience relative to the various codes, of the requests of users' experience etc.

7. It is recommended that the members assist in having the network of NDB liaison officers work efficiently in prompting the release to the NDB of nationally produced codes of major interest, and in feeding back to the Bank the experience gained in using their and others' codes.

8. The NDB has expressed interest in a more extensive advice from the NEACRP. It is proposed that NEACRP appoints some of its members to discuss in detail the problems and the programs with the NDB staff, and prepares a report with specific proposals for the next meeting of NEACRP.

ANNEX 3

FOLLOW-UP DYNAMIC BENCHMARK

At its 21st meeting, NEACRP decided to set up a dynamic benchmark problem as a follow-up of the benchmark discussed in the 1975 NEACRP/CSNI Specialists' meeting held at Munich, W.Germany.

The 1975 benchmark was aimed at exploring the range of applicability of numerical techniques in dealing with the solution of space-time dependent transients for LWRs and FBRs in 1d, 2d and 3d transients with fixed group cross-sections and prescribed simple feedback treatment for heat transfer and Doppler reactivity.

The specification of the benchmark has to be carried out in co-operation with industrial institutions in order to assure participation. Furthermore CSNI has to be approached in order to obtain full support from this committee.

Basically, the benchmark should start from the layout and the introduction of the transient as formulated in the previous benchmark, subject to minor changes. But neither the cross-sections nor the heat transfer should be prescribed, rather the models used in the various laboratories should be applied. The aim is to discuss the range of the applicability of these models and to locate further points for improvement in coupled neutronic/ thermohydraulic calculations.

It is proposed to bring forward a specification of the benchmark to the next meeting of NEACRP in 1979.

Küsters will send the participation list of the 1975 meeting to members; members will check the interest in their countries in participating in the exercise and will inform Küsters on their findings. A partner, hopefully from Western Europe, is requested to help Küsters actively in finalizing the benchmark specification in spring 1979.