NEACRP-A-560
NEANDC-A-171

NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD OF THE TWENTY-FIFTH MEETING

KFK, KARLSRUHE, F.R. GERMANY
13th-17th September 1982

Compiled by
P. WYDLER

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT
NUCLEAR ENERGY AGENCY
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PART A: EXECUTIVE SESSIONS

1. Participants and Committee Membership

The list of participants is given in Annex 1. Drs. Asaoka and Shirakata replaced Drs. Hirota and Inoue as Japanese delegates. Dr. Butland replaced Dr. Campbell as one of the UK delegates, and Dr. P. Nagel replaced Dr. D.M. Johnson as secretary for the NEA. Dr. S. Cierjacks was welcomed as NEANDC observer.

The Committee expressed their pleasure that two IAEA technical observers, Drs. Kazanskii and Khodarev, could attend, but regretted strongly that the IAEA secretariat as such was not represented.

2. Summary Record of the 24th Meeting (NEACRP-A-492)

The summary record was accepted unchanged.

3. Agenda for the Meeting

The agenda was adopted without modification.

4. Completion of Actions from Previous Meetings

All actions from the previous meeting were completed or were discussed under other items of the agenda.

5. Activities of Other Bodies of Interest to NEACRP

NEA Data Bank

A report on the activities at the NEA Data Bank (NEACRP-A-502) was presented and discussed. In particular, attention was drawn to the shielding version of the TRIPOLI code. In collaboration with the French CEA and with the help of an expert representing the Radiation Shielding Information Centre at Oak Ridge National Laboratory, the Data Bank undertook extensive testing and benchmarking of this code.

In response to a request made at the last NEACRP meeting, a report on standardization and portability of computer programs (NEACRP-A-509) was presented by J. Rosén. In particular, it was noted that the CCCC format standard used to interface files for cross section data does not have enough reaction types for certain fission and fusion problems. An extension to the CCCC standard has therefore been proposed by the Los Alamos National Laboratory.

The Data Bank has acquired a Fortran 66 verifier and is awaiting the release of Fortran 77 verifiers from the U.S. National Technical Information Service and from the TOOLPACK project. H. Rief noted that ISPRA does have an untested Fortran 77 verifier which has been supplanted by an IBM compiler/verifier.

Committee on the Safety of Nuclear Installations (CSNI)

A report on the activities of the CSNI working groups was distributed to the Committee. CSNI was restructured in 1982 into five principal working groups, focusing mainly on water reactor safety.

The detailed report on the Standard Problem Exercise on Criticality Codes for Spent LWR Fuel Transport Containers (CSNI Report No. 71), issued by a CSNI group of experts on nuclear criticality safety computations, was requested to be distributed to the Committee members. The aforementioned group of experts is currently responding to an IAEA request to extend the work to cover criticality calculations for shipment of nuclear fuel in other forms. (See also part B, section 2.7 of this summary record.)

Working Party on Nuclear Fuel Cycle Requirements (WPNFCR)

The WPNFCR inquired as to whether NEACRP could contribute to updating the characteristics of actual and advanced reactors as used in the 1982 Yellow Book (entitled "Nuclear Energy and its Fuel Cycle; Prospects of 2025") with a view to improving the next issue, foreseen for 1985.

Specific topics of interest include: fuel performance improvements for existing reactors (such as higher burnup), implications of U-236 on burnup and recycling of LWR fuel, developments in advanced reactor types and their fuel cycles.

In the discussion P.M. Garvey stressed that specific topics in reactor physics which have not been well covered in the past should be identified such as the U-236 problem. The Committee decided to include points covering advanced fuel cycles and recycling of uranium and plutonium in the agenda for the next NEACRP meeting.

IAEA Section of Advanced Nuclear Power Technology

Activity reports were distributed on the IAEA International Working Group on Fast Reactors (IWGFR) and the International Working Group on Nuclear Power Plant Control and Instrumentation (IWGNPPCI). A short list of meetings planned in the framework of activities of the Advanced Nuclear Power Technology Section was also distributed.

6. Matters Related to NEANDC

Since no NEANDC meeting had been held since the last NEACRP meeting, the Committee mainly discussed the new High Priority Nuclear Data Measurement Requirements for the Reactor Programmes (NEACRP-A-500/ NEANDC-A-156) compiled by J. Rowlands. The relevant discussions were summarized by S. Cierjacks as follows:

It was pointed out that a U.S. high priority listing for reactor applications had recently been provided. The Committee agreed that the new list (with some inevitable compromises) reflected the most urgent nuclear data needs for the fission reactor programmes in the NEA member countries. The list takes into account the different views emphasized in some countries on the importance of integral data on the requested accuracies. (The accuracies quoted in the document refer consistently to one standard deviation.)

The high priority requests were defined to be of immediate importance for current reactor physics work, implying that improved nuclear data should be available on a short term basis, i.e. within the next 3 to 5 years. A somewhat different philosophy applies to a few high accuracy requests, such as for U-235 fission in the range 1 eV to 1 keV, Pu-239 fission from 1 eV to 1.5 MeV, etc., which may not be met within the same short time scale because of existing limitations in experimental techniques, national budgets or available manpower. In these cases the Committee wanted to ensure that a permanent effort is made in the NEA member countries to reach the requested accuracies in the framework of a desirable ultimate aim.

For the collaboration between the NEACRP and the NEANDC the following procedure was suggested:

(1) After having reviewed and endorsed the new high priority request list, the NEACRP expects to receive detailed comments from the NEANDC on the pertinent data measurement activities, their status and the short term achievability of the requests.

(2) The NEACRP is then prepared to review at its annual meetings the high priority request list. In particular, the Committee will provide guidance, if high priority requests are not sufficiently pursued in member countries.

Following the discussion of the high priority request list, two documents were tabled by J. Debrue, dealing with fast neutron flux measurements using neutron dosimeters. The first paper, NEACRP-A-507, gives an overview of the development of the Nb-93m decay measurement technique at SCK/CEN Mol. The second paper, NEACRP-A-508, relates to LWR pressure vessel surveillance dosimetry and gives an account of an interlaboratory comparison of fluence neutron dosimeters in the framework of a start-up measurement programme at the pool-side facility (PSF) of the ORR reactor at Oak Ridge. In both papers the need for more accurate Nb-93 (n,n') cross section data is emphasized.

Finally, S. Cierjacks invited the Committee to help NEANDC in identifying contributors on data requirements for fission reactors for the planned volumes of the monograph series on Neutron Physics and Nuclear Data in Science and Technology. The subjects of these volumes will be (1) neutron radiative capture, (2) fast neutron induced particle emission cross sections, (3) neutron total cross sections, (4) fast neutron scattering and (5) neutron detectors.

The next NEANDC meeting will be held during the week of 27 September 1982 at Chalk River. At this meeting P.M. Garvey will act as the NEACRP observer.

7. Creation of a Joint European/Japanese Evaluated Neutron Data File

J. Rosén reviewed the status of the work done at the NEA Data Bank on assembling and testing the joint evaluated neutron data file (JEF). There are currently some fifty isotopes completed and another ten are near completion.

H. Küsters reported on the Second Meeting of the Scientific Co-ordinating Group of the Joint Evaluated File Project, held at Antwerp. It was agreed at that meeting that access to the JEF library would be restricted to Data Bank member countries; some sections of the file might be considered for general release after the benchmark testing. The next meeting of the co-ordinating group will focus on specifications of benchmarks for testing the file.

(The Summary Record of the Second Meeting of the Scientific Co-ordinating Group of the Joint Evaluated File Project has been distributed by the NEA secretariat as document NEACRP-A-558.)

8. Future of the Committee

The extensive discussion on the question of the renewal of the mandate of the Committee was summarized by the Chairman as follows:

The Committee reviewed some of its activities which were seen as important. It was noted that in the technical area covered by the NEACRP there is no parallel committee of IAEA and that many of the questions of technical judgement on the need for conferences or specialists' meetings are decided by NEACRP on behalf of a wider community.

Because of its small size and expert representation, the Committee was seen as an efficient way of identifying outstanding technical problems and resolving discrepancies. Examples of such topics were the difference between U.S. and UK assessments of uncertainty in predicting decay heat, and intercomparisons of effective delayed neutron fractions which affect the interpretation of central worth experiments on small samples.

Technical issues of this kind underlie safety studies and it was seen to be important that there was a full resolution of them in an international forum.

The Committee, through its discussion and its national progress reports (which are published and have a wide circulation) also plays an important part in identifying new issues and in informing NEA member countries of the current state of the art on a more timely basis than is possible by the transmission of the results of completed work through normal scientific publication procedures.

It was also noted that the work of the Committee generates valuable feedback to reactor physics programmes which are isolated, in particular for the smaller countries of the OECD.

The emphasis of the technical work of the Committee changes as problems are progressively resolved. Of recent years, more attention has been paid to issues related to the safe and efficient operation of thermal power reactors and to validation of the methods required for the assessment of fuel storage and reprocessing plants. An example of this type of work is the new topic proposed for the 26th meeting on the modelling problems of reactor simulators.

Also, interest is growing in the areas of large heterogeneous cores for fast breeder reactors, out of reactor production of fissile material, and neutronics of non-critical systems (blanket studies).

The Committee concluded unanimously that a renewal of the mandate should be sought.

9. Arrangements for the 26th Meeting of the Committee

The next meeting will be held at the Oak Ridge National Laboratory, USA, from 17th to 21st October 1983. A preliminary agenda for the meeting is given in Annex 2.

10. Election of Committee Officers

Dr. J. Askew (United Kingdom) was asked to remain as Chairman for one more year. Dr. M. Salvatores (France) was elected Vice-Chairman and Dr. P.M. Garvey (Canada) Scientific Secretary.

PART B: TECHNICAL SESSIONS

A complete list of all the papers presented at the meeting is given in Annex 3. In the technical sessions five new topics were discussed. Further topics were carried over from previous meetings. Following an established practice, for each topic a Committee member was assigned to prepare a draft summary, and these summaries were reviewed before the closing of the meeting.

1. New Topics

1.1 Validation of Criticality Methods, Especially in Geometries Appropriate to Reprocessing Plants

Rapporteur: J. Debrue
Papers A-522 to A-526

The topic addresses criticality in connection with the handling of fissile material outside the reactor. Five papers from the F.R. of Germany, the UK, Japan, France and Belgium were submitted.

Paper A-523 from Germany reviews the physics and the main technical aspects related to storage, reprocessing and shipment of reactor fuel. The aspects concerning criticality control are emphasized, and the respective requirements for nuclear data and/or calculational methods are discussed.

For the interim storage of spent fuel assemblies in boron containing steel boxes under water, there are no further data requirements. However, the configuration of the assemblies being different from any in-pile fuel arrangement, the applicability of standard reactor codes has to be verified using, for example, the more sophisticated Monte Carlo and/or deterministic transport methods.

In the reprocessing plant, a limitation of the fissile concentration is required for the dissolution of low burnup fuel and for the subsequent uranium/plutonium separation. Subcriticality conditions can be improved using gadolinium as a homogeneous neutron poison in the solution and hafnium as a heterogeneous neutron absorber in the dissolver tank and the extraction columns. From the reactor physics point of view, further integral experiments with heterogeneous Gd, Hf and Eu absorbers are recommended. These should be carried out for both simple and complex geometries, in order to provide a basis for checking nuclear data and the methods used to describe the complicated geometrical configurations. Effects to be included are the moderation and the reflection of neutrons by the aqueous solution and the surrounding concrete walls.

From the discussion, it appeared that several national programmes either already include (France, Japan, UK, USA), or will include (Germany), experimental criticality studies on systems representative of the situations encountered in storage ponds and/or reprocessing plants. These experiments will probably fill the gaps which presently exist in the experimental data base needed for validation purposes. Specific actions by the Committee to stimulate further work are therefore not felt to be necessary.

Two papers, one from the UK (A-522), the other from Japan (A-525), deal with the validation of criticality codes on the basis of a large number of published experimental results.

The UK Monte Carlo criticality code MONK has been in use since the early 1960's and is periodically updated. The point energy cross sections are essentially taken from the UKNDL without further adjustment. The first version of MONK6 is now being tested for uranium, plutonium and mixed systems covering a range of moderation and reflection. The conclusions to date are summarized as follows:

(1) The only significant discrepancy in the calculation of uranium systems appears for low enriched fuel pins in low enriched solutions, with or without gadolinium. In these cases k_{eff} is underestimated by about 5 %.

(2) Fast plutonium systems are well calculated. For moderated plutonium systems, the reactivity is overestimated by 1 to 2 %.

(3) Agreement between calculation and experiment is not so good for mixed uranium/plutonium systems where k_{eff} is overestimated by several per cent in some cases.

Reasons for the discrepancies are being investigated but cannot be clearly identified at present. However, improvements could result from the utilization of a thermalization model better suited to the Monte Carlo calculation. It is noted that the description of the systems (geometry, composition) is sometimes inaccurate or incomplete, in particular for the old experiments.

Paper A-525 describes a new criticality safety evaluation technique. The authors start from an experimental data base consisting of more than 700 critical assemblies. The computer code system JACS, developed specially for criticality evaluations, was used to group these into 10 classes, according to the neutron leakage/absorption ratio, the relative importance of the thermal, epithermal and fast neutrons, and the type of fissile material. In each class a histogramme of the number of assemblies versus the calculated k_{eff} value reflects a systematic bias with respect to the measured value (k_{eff} = 1) and an associated k_{eff} uncertainty (spread). On the basis of this information the YENMA code, which is included in the JACS system, evaluates the criticality of the system in terms of a probability. A criterion establishing the subcriticality was selected and tested on the 700 benchmark cases. In summary, the judgement that a system is subcritical is performed by the code itself instead of imposing , for example, $k_{eff} < 0.95$ on a single calculation.

Measured and calculated k_{eff} values reported in this paper differ by a few percent, these discrepancies being of the same order as those mentioned in the UK paper (A-522). Although the utilization of group cross sections adjusted for reactor calculations would lead to much better agreement at least in the benchmark cases close to reactor lattices, it appears from the discussion that the utilization of pointwise cross sections with appropriate self shielding treatment in the Monte Carlo criticality calculation is more desirable.

Paper A-524 from France describes the CRISTO III experiments. These are designed to qualify the criticality calculations of a dissolver tank in a reprocessing plant poisoned with gadolinium. In the experiments 3 % enriched UO_2 rods are loaded in the central test region of a two-zone system. Two lattice pitches were investigated, the first one corresponding to the most close-packed square lattice feasible, the second one being representative of large PWR fuel assemblies. In these lattices the reactivity equivalence between boron and gadolinium in solution was determined, thus providing the information needed to qualify calculational methods for gadolinium systems. Analysis of the close-packed lattice results and a comparison with APOLLO calculations indicates that the thermal gadolinium cross sections are being underestimated in APOLLO by 5 %.

Generally, criticality calculations for irradiated fuel storage ponds are performed with fresh fuel compositions. An increase in storage capacity of the pond would be achieved however, if the burnup of the fuel could properly be taken into account. This would require, in particular, specific validation of the calculational methods used, as well as a means for measuring burnup for each fuel assembly entering the storage area. An evaluation of the gain in capacity is made in paper A-526 from Belgium. In the paper the criticality aspects of PWR assemblies in an infinite square pattern are considered.

The SCALE code system was used to perform two k_{eff} calculations as a function of the assembly pitch respectively for fresh fuel and fuel with a burnup of 33 GWd/t. The 123-group cross section set included in this system was condensed to a 23-group set to reduce the size of the problem. The main fission products were represented explicitly, and the composition of the irradiated fuel was determined using the CASMO code.

A typical result is that the storage capacity could theoretically be increased by a factor of about 1.7, assuming that the burnup is uniform in the axial direction of the assembly. The k_{eff} value of a single burnt fuel assembly in water was calculated to be about 0.67, a figure of interest when discussing the applicability of subcriticality monitoring techniques (cf. topic 2.3).

In the discussion several comments were made concerning the evolution of the fuel and fission product composition over the storage period, the axial variation of the burnup, the trend to increase burnup in the future and the measures to be taken in view of a premature unloading of the fuel from the reactor.

1.2 Rating Distribution and Reactivity Effects of Gadolinium Poisons in Thermal Reactors

Rapporteur: H. Neltrup
Paper A-510

One paper from Norway was submitted under this topic. It describes the modifications introduced in the FMS code system in order to treat fuel assemblies containing Gd_2O_3 bearing fuel pins. Cross sections for the poisoned pin cell are generated using the THERMOS collision probability routine for a super cell consisting of the poisoned cell surrounded by a homogeneous mixture representing the eight nearest neighbour cells. The cross sections are normalized to the flux on the surface of the poisoned cell. In order to conserve the reaction rates in the subsequent RECORD assembly calculation the THERMOS cross sections are transformed by an X-Y diffusion theory calculation extending over the same pin cells as the THERMOS calculation. The highly absorbing isotopes Gd-155 and Gd-157 are depleted up to a given value of the time-integrated flux, after which a constant residual gadolinium absorption cross section is used.

The methods have been verified against cold, critical, beginning of cycle measurements in the Dodeward reactor and against gamma scans on low burnup gadolinium fuel from the Mühleberg reactor. The local power distribution within a fuel assembly as predicted by RECORD is compared with gamma scan measurements on assemblies from the 2nd cycle of the Quad City reactor. In summary, the paper reports fair agreement between calculation and experiment, with differences within 5 %.

In the subsequent discussion some uncertainty became apparent regarding the state of the art and the need for further investigations. It was decided to return to these issues when discussing the gadolinium benchmark (cf. benchmarks, section 4.6). Attention was drawn to work going on in the PWR field by referring to the national activities report of the F.R. of Germany (cf. NEACRP-L-258), to the Combustion Engineering report CEND-397 and to the international gadolinium programme proposed by Belgonucléaire and CEN Mol.

1.3 Out-of-Pile Production of Fissile Material

Rapporteur: T. Asaoka
Paper A-527

Paper A-527 from Japan reports an evaluation of computational models for spallation and fission reactions in the energy range 50 - 1000 MeV. The models are those incorporated in the NMTC/JAERI code, used for feasibility studies of the accelerator breeding and transmutation concept. It is shown that proton and neutron non-elastic and fission cross sections calculated using the level density parameters of Il'inov et al. agree well with the experimental data.

In the discussion it appeared that experimental data related to the accelerator breeder concept are scarce, but relevant measurements are presently being carried out in various laboratories including, for example, CRNL (cf. NEACRP-L-258, Reactor Physics Activities in Canada) and KfK Karlsruhe (cf. S. Cierjacks et al., Proceedings of the Antwerp Conference 1982 on Nuclear Data for Science and Technology, pp. 383-386).

Another method for producing fissile material is to make use of the fusion-fission hybrid reactor concept. Attention was drawn to the fusion-fission hybrid blanket experiments to be performed in the zero-power facility LOTUS at the Swiss Federal Institute of Technology at Lausanne.

The Committee recognized a fairly wide interest in the out-of-pile breeder concepts, especially from the viewpoint of the long term fuel cycle, and therefore agreed to pursue activities in this field.

1.4 Calculational Methods for Evaluating Control Rod Worths in FBRs and their Validation

Rapporteur: A.T.D. Butland
Papers A-511 to A-516

Paper A-511 describes an analysis, by KfK, of subcritical rod worth measurements made in the UK Zebra single annular core BZD/2, which formed part of the BIZET UKDeBeNe collaborative programme. The multi-group cross sections used in the analysis were prepared using the cell program KAPER for the core regions and the control rod singularities, the latter involving a super cell approach with transport corrected diffusion coefficients. The whole-reactor calculations were performed in full plane X-Y geometry with a single axial buckling derived from the reference (all follower) array. The C/E ratios range from 0.96 to 1.02, which is up to 7 % lower than the range found in the earlier conventional assemblies BZA and BZB, and up to 6 % lower than the range found in the "salt and pepper" assembly, BZC. The paper suggests that this may be due to the use of X-Y geometry and the heterogeneous nature of the BZD assembly. Three dimensional X-Y-Z calculations are planned.

Paper A-514 is a fairly wide ranging review of UK calculational methods and experimental validation, with a discussion of potential fast reactor absorber materials. The currently recommended calculational method uses a super cell model to prepare the absorber rod cross sections, with an adjustment to the flux-averaged rod cross sections to preserve reactivity. Streaming in the low density channels is modelled by Bonalumi's approximation to Benoist's formulation, with an axial buckling correction added. The design methods use diffusion theory in triangular-Z geometry to calculate the whole-reactor neutron flux, which has been shown to give rod worth to within 2 % of an extrapolated fine mesh transport theory calculation. The design methods also use 6 or 9 energy groups, which introduces an error of about 2 % on rod worth.

The paper discusses the prediction of rod worths and reaction rate predictions in the MOZART assemblies, the Zebra PFR mock-up and the BIZET assemblies of conventional design. It is concluded that the currently recommended methods can predict the reactivity worths of both single boron carbide rods and arrays of rods to $\pm$ 5 %. Rod interaction effects are well predicted as are the relative worths of partially inserted rods. The effect of increasing the enrichment of the boron carbide rods is predicted to within the uncertainties on the measurements, the use of the homogenization procedure being important in removing trends with enrichment reported in the past. Transport theory is shown to be important in the analysis of rod reaction rate measurements, but discrepancies of more than 5 - 10 % remain in, for example, U-235 fission rates within the rod, although the analysis of these measurements has not yet used the currently preferred homogenization procedure.

The survey of potential absorbers reports small sample reactivity measurements and considers how these results must then be interpreted in the light of the absorber density and volume fractions possible in a realistic rod design, plus the resonance shielding effects important with some absorber materials. The work shows europium to be a useful alternative to boron carbide, although it was reported that the UK have no plans at present to pursue this possibility any further.

Paper A-512 establishes correction factors for the standard French calculational methods used for the prediction of boron carbide rod worths in power reactors. The standard method consists of hexagonal-Z diffusion theory in 6 energy groups with 7 mesh points in the subassembly plane (mesh edge diffusion theory) and a homogeneous model for the calculation of the rod cross sections using CARNAVAL IV data. The correction factors introduce

- reductions of up to 3 % to allow for the coarse energy group structure,
- reductions of up to 8 %, depending on rod position, to allow for the use of diffusion theory,

- reductions of up to 1 %, depending on rod position, to correct the transport corrected result for the use of a finite mesh size,
- reductions of up to 13 %, depending on rod position, to correct for the use of a homogeneous model for the generation of cross sections.

A total uncertainty of ± 13 % (2σ) is placed on the final prediction, largely made up of a contribution derived from experimental results.

In fact, the method has been tested against measurements made in the PRE-RACINE and RACINE experimental programmes in MASURCA, considering both natural and enriched boron carbide rods. The (E-C)/C results range from -1 ± 2 % to -10 ± 6 %, the largest discrepancies occurring when sodium channels are present, which could reflect effects related to neutron streaming.

Paper A-516 compares the calculated and measured worths for a boron carbide control rod absorber in Phenix, the calculational method being that discussed in paper A-512. The measurement was for an articulated SAC (complementary shut down) rod at the centre of Phenix. The correction factor applied to the standard calculation was 22 % giving a final (E-C)/C of +2 ± 8 %, where the 8 % uncertainty quoted arises solely from the measurement, the calculational uncertainties being however much smaller.

It was reported at the meeting that a recent measurement in Phenix for a single Super-Phenix type rod gave similar agreement with calculation.

Paper A-513 reports the development of a new Japanese homogenization method for boron carbide rods and assesses its effectiveness by the analysis of control rod experiments carried out in ZPPR-10A. The homogenization method consists of a super cell flux averaging method to determine the rod cross sections, followed by an iterative adjustment to the rod cross sections in order to preserve the integrated reaction rates in each energy group. The use of the adjustment procedure improved the rod worth predictions by 2 - 4 %, depending on the pin layout but not on the boron enrichment. Using this procedure together with a transport whole-reactor flux calculation, the rod worths were predicted to better than 1.5 %.

Paper A-515 summarizes the measurements and associated analyses for a large number of boron carbide absorber rod measurements in ZPPR-3/1B to ZPPR-11F. The calculations were performed using the standard U.S. design method, which involves coarse group diffusion theory in two dimensional geometry, with axial bucklings and no special treatment of rod heterogeneity. The C/E's for the rod worths range from 0.95 to 1.15 using ENDF/B-IV data for all cross sections, but with the delayed neutron yields and emission spectra used in the calculation of β_{eff} being taken from ENDF/B-V. When the standard calculational method is corrected for finite mesh size, the use of diffusion theory, the neglect of neutron streaming and heterogeneity effects, and the use of a coarse energy group scheme with an axial buckling the C/E's average to about 1.1. Rod worth profiles and rod interaction effects seem to be well predicted by the standard method, but only in three dimensional geometry. It was reported at the meeting that the standard method has given a significant C/E difference (up to 10 %) for heterogeneous and homogeneous core designs. However, this comparison did not include results from a large heterogeneous design, these not yet being available. It was stated that this C/E difference may be due to problems in the cross section preparation methods used for the heterogeneous cores.

During the general discussion it was concluded that rod worths were adequately predicted using the latest homogenization methods available in the UK, Germany and Japan, although there might be some evidence for a systematic difference between conventional and heterogeneous cores. Rod interaction effects were also well predicted. It was felt however, that reaction rates inside and near absorber rods are not adequately predicted and require further study. The situation may improve once the latest homogenization techniques are included in the analysis. It was agreed that the available nuclear data appeared to be adequate for the prediction of rod reactivity worths.

It was agreed that future discussion should consider

- reaction rate predictions in and near absorber rods, with particular reference to predictions of rod lifetime,
- the variation in rod worths and reaction rates through the fuel cycle,
- any evidence for the adequacy of rod worth predictions arising from power reactor measurements,
- heat deposition in absorber rods.

1.5 Reactor Physics Modelling of Distorted Cores

Rapporteur: H. Küsters
Papers A-517 to A-520

Since a fast reactor core is not in its most reactive configuration, redistribution of the core materials (fuel, steel, sodium) during an accident has the potential to induce a severe power excursion. Paper A-518 describes the accident sequence resulting from a loss of flow incident (As regards possible energetics, this accident can be shown to cover accidents caused by other initiators). The paper outlines the physics modelling of the various accident phases by computer codes such as SAS3D and SIMMER-II.

It is essential to validate these codes against benchmarks and experiments. Particularly the neutronic parts of the codes can be tested to a large extent by simulating distorted core configurations in critical assemblies. Examples are configurations corresponding to strong fuel compaction or local material dilution due to slumping. In some accident scenarios molten steel (clad material) may move to the cooler axial blankets where it may freeze and form massive blockages. Such steel relocations can also be simulated in critical assemblies.

Papers A-517, A-519 and A-520 describe experiments and their analysis in the SNEAK-12 and FCA VIII-2 assemblies. The results show that the reactivity effects associated with cavities, streaming channels and the redistribution of steel can be predicted to an accuracy of 10 % or better, if (S_4) transport theory is used.

The contributions concentrate on experiments related to the redistribution of fuel and the formation of asymmetric fuel compactions near the lower core/blanket interface, simulating molten pools of fuel. In addition to the reactivity effects, fission rate traverses for U-235 and U-238 were measured in these distorted core configurations, with the aim to check whether theory is able to appropriately predict the power distribution. A partly-voided reference core was chosen, and the analysis of core distortions was performed relative to this standard.

The results can be summarized as follows: Although calculational methods and data problems associated with the analysis of distorted core configurations are rather complex, the experimental results can be predicted to an accuracy of 15 to 20 %, provided that transport theory (e.g. S_4P_o) is used and special precautions are taken to describe neutron streaming and cell heterogeneity effects in the compacted regions. Normal diffusion theory gives discrepancies of about 50 %. The analysis shows that, in accident simulations, reactivity changes at least can be predicted to a reasonable accuracy. The fission rate traverses are predicted to within 10 % of measured values. It should be kept in mind however, as pointed out in previous discussions of the Committee, that major uncertainties may nevertheless arise in the non-neutronic parts of the accident simulation, although very convincing progress has already been made with the introduction of codes like SIMMER.

From the discussion it appeared that no significant reactivity insertion mechanisms have been identified for thermal reactors.

Because this topic is essential for the safety assessment of fast reactors, the Committee decided that further progress should be reported at the next NEACRP meeting. The Committee on the Safety of Nuclear Installations (CSNI) was invited to comment on the accuracy requirements for physics related parameters used in whole core accident analyses.

2. Topics Carried over from Previous Meetings

2.1 3D Transport Solutions for Reactor Calculations with Emphasis on Acceleration Techniques, Including Those Used in Diffusion Codes

Rapporteur: P.M. Garvey
Papers A-528 to A-532

The five papers contributed to this topic by the UK (A-528, A-529), France (A-530, A-531) and Japan (A-532) deal with improvements leading to cost reductions in 2D and 3D transport and diffusion calculations.

As many calculations require transport methods to give the required degree of accuracy, attention has been given to faster methods of solving the transport equation. Paper A-528 describes a formalism which allows methods developed for the diffusion equation to be used. The equations to be solved involve the directional currents and fluxes at the mesh boundaries, the relation between these two quantities being found from the standard transport equation using a uniform, isotropic source within each mesh. This calculation is repeated at each outer iteration. An application of the method to a slab has given a significant cost reduction as compared to a DSN solution as well as acceptable accuracy. Generalization to two and three dimensions has been assessed. A 2D code using the buckling iteration approach has been implemented with satisfactory results in some, but not all, cases. This work complements the previously described spherical harmonics method at AEEW.

Paper A-529 describes the calculation of various reactivity coefficients and other parameters for the Winfrith SGHWR with the MONK Monte Carlo code using an 8 group data base generated from WIMS. This data had previously been used in the 3D diffusion code SNAP. Use of the code allowed the full geometrical detail of the core to be simulated, which was important at low moderator levels. Good agreement was obtained with experimental data. This calculation showed that it is now reasonable to apply Monte Carlo codes to full core calculations. Future such calculations would use a point rather than a group data base.

Recent improvements in the computer software field have been applied to the 2D/3D diffusion code CORA-1. Paper A-531 shows that a cost reduction by a factor of about 5 may be obtained in a 2D problem. A one dimensional flux synthesis method is used to give the starting flux in a 3D calculation. The same software development has led to a cost reduction by a factor of 8 in the synthesis calculation. Further developments have included an exploration of the possible advantages of the finite element method. The authors think that in three dimensions the method could have a cost advantage over the finite difference approach. The

CORA-1 code has also been implemented on the CRAY-I computer, and this has resulted in a cost reduction by a further factor of 10.

3D diffusion calculations are required, for instance, in the calculation of the reactivity coefficients of fast reactors. An alternative way to reduce the cost of such calculations is to minimize the number of groups. In paper A-530 it is shown that 3 and at times 2 groups together with a bilinear weighting method are sufficient to calculate the spatial sodium void coefficient for various control rod configurations. Adjoints determined for an intermediate control rod distribution are sufficient for the weighting procedure. These calculations are compared with the flux weighting method where more groups are required and a lower accuracy is obtained.

The final paper, A-532, describes a finite element program which solves the neutron transport equation in three dimensions based upon a Legendre polynomial expansion of group to group transfer. Acceleration methods have been applied which led to a significant reduction in the number of outer iterations.

In conclusion, advances in computer software and hardware, improved group collapsing techniques and acceleration methods have led to a significant saving in cost in the solution of two and three dimensional problems. Further assessments of gains to be obtained in these areas are to be anticipated due to the need to run substantial numbers of calculations of this kind.

2.2 Beta and Gamma Decay Heat Measurements for Fast and Thermal Reactors, Particularly for Pu-239

Rapporteur: R. Peelle
Papers A-533 to A-536

Four papers contributed by the UK, the USA and Japan describe new data on fission product decay heat or consider the present status.

Paper A-533 gives new data of Taylor, Murphy and March on observations of beta and gamma decay energies from U-235 and Pu-239 in a fast reactor spectrum. The beta-ray data include observations taken as soon as 15 - 20 s after the end of irradiations. The results are compared with summation calculations from the UKFPDD-1 and -2 decay data files. Beta emission appears consistent with the UK calculation, while both low- and high-resolution gamma measurements starting $\sim 10^4$ and $\sim 10^3$ s after shutdown show greater emission than that calculated. Measurements following thermal fission are planned.

Paper A-536 by Akiyama and An gives gamma-ray energy release rates from fast fission in U-238, Th-232 and natural uranium. Energy release is nearly consistent with the JNDC file for U-238 except near 10^3 s after a burst. A similar but more definite situation exists for Th-232. However, somewhat different summation results from other data files are also shown. (Note that a conference paper by these authors combines these data with those released earlier to this Committee, and shows direct comparisons to the data of Murphy and to that of Dickens et al. for fission by thermal neutrons. Cf. paper by Akiyama and An, Proceedings of the Antwerp Conference 1982 on Nuclear Data for Science and Technology, pp. 237-244.) Beta-ray data for the non-fissile nuclides do not seem to be accessible.

In paper A-534 M.F. James continues his previous analysis of decay heat integral results relative to those of the UKFPDD-2 decay data file. Ratios of heat release for the various experiments are also compared to the UK summation calculations. The apparent real differences among the U.S. measurements as well as contrasts with the UK calculations are emphasized to reach the conclusion that too-small uncertainties should not be assigned for fission product decay power in the face of these differences among experiments.

In paper A-535 Peelle and Dickens review the status of decay power estimates now used or proposed for use in the United States. The authors note differences among the U.S. measurements, but also the quantitative agreement between the thermal-spectrum data of Dickens and the fast-spectrum data of Akiyama and An (except near 10^3 s for U-235). The significant differences among summation calculations (on a fission burst basis) are noted. (See also paper by Duchemin et al., Proceedings of the Antwerp Conference 1982, pp. 249-256.) The present near-consensus U.S. view is restated that practical decay power calculations for relatively short times after fission can now best be obtained by careful combination of integral and differential data. For U-238 however, little data exist. Tables are given in paper A-535 comparing several data sources for U-235 and Pu-239.

The Committee in discussion noted the various data differences at the 5 - 10 % level, the expected final writeups of experiments in Japan and the UK, new plans for experiments in the UK, and the present general recognition that differences among the various fission product decay data files affect summation calculations at the $\sim$10 % level at least for fission bursts. It was concluded that scientific inquiry processes would likely lead to some clarification. The Committee generally agreed that short term fission product heat data cannot now be considered precise, but may be known to $\sim$5 % for long exposures. The Committee expressed the wish to be cognizant with future developments. To facilitate the achievement of economic gains associated with narrower uncertainties in after-shutdown decay power estimates, the Committee asked that experimenters communicate with each other to search for explanations of differences and that the NEANDC be made aware of the continuing importance of valid fission product yield and decay data even for cases not accessible to direct measurement.

2.3 Subcriticality Reactivity Monitoring in Out-of-Pile Configurations; Reactivity Monitoring by Noise and Pulsed Neutron Methods, Non-Destructive Testing of Burnup, Isotopic Correlation

Rapporteur: L.G. Le Sage
Papers A-537 to A-539

Paper A-537 from the USA and paper A-539 from France discuss the isotopic correlation method for verifying the fissile material input balance to the reprocessing process. Paper A-538 from Belgium is concerned with the use of pulsed neutron techniques for subcriticality monitoring of fuel in storage ponds.

Paper A-539 presents results obtained from applying isotopic correlation methods at the La Hague reprocessing plant. Data from three reprocessing campaigns from 1978 to 1980, including more than 400 dissolution batches, are used in the study, and several experimental comparisons are discussed in the paper. Isotopic ratios measured at the input to the reprocessing process agree well with calculated ratios using the discharge burn-up quoted by the reactor operator, the agreement being somewhat better for PWR than for BWR fuel. Burnup values deduced from the (measured) isotopic ratios also agree well with the quoted burnup values, but with a systematic discrepancy of approximately 3 %. This could possibly be due to an error in the value taken for the mean energy release per fission. Finally, measured values for the Pu/U ratio, the uranium input mass and the plutonium input mass agree well with the respective values deduced from the correlations. It is noted that the total discrepancy in plutonium mass for all the campaigns is only 0.3 %.

Paper A-537 deals with the application of the isotopic correlation method based on detailed analysis of the Zion 2 PWR. The emphasis is on safeguards applications. The importance of including the plutonium isotopes in the correlation method in order to obtain unambigious verification of the initial enrichment and the burnup of the sub-assemblies input to reprocessing is discussed. It is noted that some isotopic correlations depend on whether averages are performed over short axial segments or the full height of a subassembly, implying that in these cases full subassemblies but not individual segments should be dissolved for the method to be applicable. A prototype computer for in-field implementation of the isotopic correlation technique, for instance by an IAEA safeguard inspector, is also described in the paper.

In summary, the feasibility of using isotopic correlation methods for verifying the input fissile balance has been established. Recent work on the isotopic correlation method has emphasized the need to derive the basic correlations from core burnup calculations, since many of the correlationsare non-linear. Previous work had attempted to obtain linear fits to measured isotopic ratios with only limited success.

Credit may be taken for fuel burnup in the analysis of fuel storage ponds, if it can be verified by measurement that the burnup of each fuel assembly entering the pond is higher than the design limit. In paper A-538 a pulsed neutron technique for measuring fuel assembly subcriticality is discussed. Both the reactivity ρ and the prompt neutron decay constant α are determined. It is concluded that using this technique k_{eff} values as low as 0.75 can be measured in a reliable way. Since k_{eff} for a burnt subassembly is likely to be less than 0.75, the objective of the pulsed neutron method would possibly only be to verify that the burnup of the fuel entering the storage pond has exceeded a specified value. The advantages of the pulsed neutron method over burnup determinations by gamma spectrometry are also discussed.

In discussion it was noted that subcriticality monitoring by neutron noise using the californium source technique (cf., for example, J. Mihalczo, Transactions of the ANS, December 1981, vol. 39, p. 517, and J. Mihalczo et al., Transactions of the ANS, June 1982, vol. 41, p. 588) is a useful alternative to the pulsed neutron method. Attention was drawn to the work on remote identification of materials at the U.S. National Bureau of Standards (cf. R.A. Schrack et al., Proceedings of the Fifth International Conference on Quantitative NDE in the Nuclear Industry, San Diego, May 1982, p. 158) and to a new review of reactivity monitoring methods presently being undertaken in France.

2.4 Intercomparison of Reaction Rate Measurements in Fast Reactors

Rapporteur: A.T.D. Butland
Papers A-540 to A-544

Paper A-540 describes integral measurements made in three Zebra assemblies of the reactions of interest to primary circuit activation in a fast reactor. The measurements were made relative to Pu-239 fission rates by irradiating small samples of single important chemical elements and of typical steels in core and breeder spectra and by subsequently determining the build-up of the reaction products with a calibrated Ge-Li spectrometer. The measured reaction rates were compared with calculations made using the standard MURAL/FGL5 route to determine the neutron flux and evaluated differential cross sections for the reactions of interest. The calculated neutron fluxes are given, which will facilitate comparison with new future evaluations of these reaction cross sections, except where sample resonance shielding effects are important. The work recommends factors to bias future calculations of the neutron reactions considered, together with uncertainties which are significantly less than those associated with the steel corrosion process itself.

Paper A-541 considers the importance of electron transport in fast reactor heat deposition calculations. At present the standard UK calculational route assumes that all the photon energy is deposited at the point of photon interaction, whereas an electron cascade is triggered, so that some of the energy is deposited elsewhere. This may be important in fine structure calculations, such as the estimation of heat deposition in the relatively thin subassembly wrappers necessary when calculating subassembly irradiation induced distortion effects. The paper notes that the Monte Carlo program MCBEND will be extended to model electron migration in such geometries. It also notes that programs already exist to model electron migration in the interpretation of TLD measurements (the PROCEED program) and in the calibration of TLD's (the TIGER program), where the effects are significant.

Paper A-543 describes the measurement of three reaction rate ratios (F8/F5, F5/F9, C8/F9) during the UK Zebra BIZET programme by teams from AEE Winfrith, CEN Mol and KfK Karlsruhe. Each team used its own measurement techniques, and the measurements were made at the centre of BZA and BZB, The measurements were all finally expressed as cell averaged values for easy comparison with calculations performed with the cell programs MURAL (UK) and KAPER (KfK). The individually measured values are in good agreement with the mean results. The C/E values generally differ from unity by more than one standard deviation, the largest difference being four standard deviations for F8/F5 computed using KAPER.

Paper A-542 describes a similar measurement intercomparison between ANL and AEE Winfrith, the measurements covering the same ratios and being made at Winfrith during the Zebra CADENZA programme. The results presented are preliminary, but show the ANL and AEEW measurements to be in agreement to one standard deviation for F8/F5 and F5/F9 and to be discrepant by two standard deviations for C8/F9. Further analysis of the measurements is underway, including a comparison of the measured absolute reaction rates and a re-appraisal of the calibration data.

Paper A-544 presents a comparison by ANL of the gamma counting and mass spectrometer methods for measuring capture in U-238, the former being the technique used by ANL in the intercomparison reported in paper A-542. The method was to irradiate U-238 metal foils and then to dissolve them creating two samples, one for gamma counting of Np-239, and one for mass spectrometer analysis of Pu-239. The mass spectrometer measurement gives higher values than the gamma counting by about 2 %. The paper also reports a radiochemical determination of the Np-239 activity, which showed good agreement with the gamma counting method.

During the discussion it was noted that if the spectrometer results in paper A-544 were used, the discrepancy between the C8/F9 measurements reported in paper A-542 would be reduced to about one standard deviation. It was also noted that there was a difference of about 2.5 % in the ANL and AEEW absolute measurements of the Pu-239 fission rate, the ANL result being the larger. This is being investigated further, and if resolved, could bring the ANL and AEEW results for C8/F9 still closer together.

The Committee felt that the reaction rate intercomparisons were proving to be very useful. It was noted that the French proposed to have a similar intercomparison in MASURCA in the near future, involving themselves and ANL. It was suggested that AEE Winfrith, KfK Karlsruhe, CEN Mol and possibly Japan should take part. This will be considered by the various national representatives.

2.5 Delayed Neutron Data and Reactivity Scales, Beta (Eff) Data, Central Reactivity Worth Discrepancies

Rapporteur: P. Hemmig
Papers A-545 to A-548

Paper A-545 by H. Nakamura and N. Tsuji recommends bilinear weighting for energy group collapse. This paper also cites the ZPPR-9 and -10 measurements of material worths and the k_{eff} measurements in the ZPPR-10B approach to critical as supporting their choice of the Tuttle/Saphir/Keepin delayed neutron data set.

Paper A-548 by J.M. Stevenson summarizes a wide range of plutonium worth and kinetic measurements in the Zebra facility over the past 10 years. The C/E ratios obtained are consistently in the range 0.97 to 1.06, which is well within the range of uncertainty in the reactivity scale.

Paper A-546 by K.S. Smith and R.W. Schaefer provides new analyses of the ZPR-6 diagnostic cores. The C/E material worth ratios obtained range from 0.99 to 1.05. These ratios are much reduced from previous measurements at ANL due in large part to the improved treatment of the intracell adjoint flux distribution.

Paper A-547 by L.G. Le Sage outlines a possible benchmark for intercomparing reactivity worth calculations.

The Committee in discussion noted the considerable recent progress in this area and proposed the following actions to develop further understanding:

(1) Participants in the LMFBR burnup benchmark problem should recalculate β_{eff} using the most recent delayed neutron data and transmit their results to M. Salvatores for inclusion in the burnup benchmark report.

(2) Members should send their comments on the reactivity worth benchmark proposal to L.G. Le Sage in time for preparation of the benchmark specifications.

(3) P.M. Garvey and S. Cierjacks should convey to the NEANDC the basis of NEACRP interest in more accurate U-238 delayed neutron yield information.

2.6 Heterogeneous Cores, New Physics Related Information

Rapporteur: M. Salvatores
Papers L-260, A-549 to A-552

Five papers were presented on this topic, three from the USA, one from Japan and one from France. The U.S. and the French papers are mainly related to experiments performed in fast critical assemblies and their interpretation. The subject is also discussed in the UK national activities report (cf. NEACRP-L-258).

Paper L-260 from Japan (which was already presented at the Lyon 1982 ENS/ANS Topical Meeting on LMFBR Safety) gives a comparison, from the physics point of view, of the safety performance of "equivalent" homogeneous and heterogeneous large fast power reactor cores. The conclusions of the paper confirm the indications given in a previous NEACRP document (A-484) that heterogeneous core configurations have potential advantages over homogeneous configurations.

In the U.S. papers points related to sodium void assessment in heterogeneous cores are particularly emphasized. Paper A-550 outlines the procedure followed to define bias factors and uncertainties for CRBR sodium void coefficients by region, taking into account different voiding sequences. The bias factors were obtained from a large number of ZPPR experiments in a core of the same size as that of the reference reactor. Both critical experiments and design calculations were made using the same calculational methods and data. In general, the two-component bias factors (for leakage and non-leakage) do not exceed $\sim$1.25 for most situations, this being a clear indication of good performance of the standard calculational methods for predicting sodium void reactivity effects.

More information related to critical mass and reaction rate ratios is given in paper A-552. The data confirm that homogeneous and heterogeneous core configurations have similar trends as regards C/E values.

Paper A-549 reports preliminary results of gamma heating measurements on a heterogeneous core configuration in MASURCA. Different experimental techniques (ionization chamber, TLD of different types: LiF, Al_2O_3, $CaSO_4$) are used in the framework of an intercomparison between several laboratories with the participation of CEA, CEN Mol and UK specialists. The preliminary C/E comparisons reported show reasonable agreement and no evidence for major difficulties.

Finally, paper A-551 reports on a recently started new ZPPR programme on large heterogeneous core configurations, carried out under a co-operative agreement between the U.S. and Japan. Fissile core volumes will be in the range 4100 to 4300 l. A first configuration, ZPPR-13A, reached criticality in August 1982. For this core, bias factors obtained from previous experiments enabled criticality to be predicted with only a very small residual discrepancy. It was pointed out that there are preliminary indications for space-time effects on the scale of several seconds.

2.7 Miscellaneous

In a short note, A-553, R. Peelle informed the Committee of the status of the CSNI activities on nuclear criticality safety computations. The transport flask criticality benchmark proved to be very useful in highlighting and resolving specific problem areas. The results have meanwhile been documented in CSNI Report 71, and five new problems centering on Class I and II containers for fresh fuel were defined.

In response to an action from the previous meeting, L.G. Le Sage presented a brief review, A-554, on the use of low enriched fuel in research reactors. The (non-proliferation orientated) objective of the programme is to develop the technology to permit research and test reactors to be converted to low enriched (<20 %) fuel without significant penalties in core performance and fuel cost. So far, the programme has been very successful. Uranium/aluminium fuels with a higher density than those used hitherto have been developed and successfully tested. New fuels (uranium silicide and others) may offer the possibility of even higher densities. Core physics and safety parameters of reactors converted to this fuel have also been investigated with the aim of demonstrating that existing limits in the safety analysis reports can still be met. Attention is drawn to the handbooks on test reactor design with lower enriched fuel which are being prepared for the IAEA.

Dr. Y. Kazanskii presented a paper, A-555, on the reactor physics experiments performed during the BN-600 start-up. The paper gives a detailed account of the approach to critical and describes measurements of control and safety rod worths, various reactivity effects, and the power distribution. The reactivity effects include the coefficients of temperature and power as well as effects related to cover gas pressure, primary pump speed and fuel burnup. Calculations based on the BNAB-70 data set are generally in good agreement with the measurements, the variations being within the limits evaluated earlier during a zero-power reactor programme for BN-600.

Finally, a Japanese paper, A-556, was distributed describing a study of a shielding calculation technique using the albedo S_N method. The method has been implemented in the DOT 3.5 code and applied to a test problem with a saving of nearly 40 % in computing time. The accuracy of the method is discussed in the paper.

3. National Programmes

Reports on the reactor physics activities in the NEA member countries were summarized and discussed at the meeting. The full reports will be included in a consolidated document (NEACRP-L-258), reproduced and distributed by the NEA Secretariat.

4. Benchmarks

4.1 LMFBR Burnup Benchmark

The draft proceedings of the Specialists' Meeting on the LMFBR Burnup Benchmark Calculations held at Cadarache in April 1982 (NEACRP-A-504) were presented by M. Salvatores.

A comparison of the solutions led to some interesting discussions. Larger variations than expected were observed when comparing the one-group pseudo fission product cross sections, and it became apparent that for their production the authors had used quite different procedures, including specific adjustments of the cross sections in some cases. Non-negligible contributions to the standard deviation of the total reactivity loss over the cycle due to differences in the burnup and build-up of the heavy isotopes were identified and analysed.

The final version of the proceedings will be ready by the end of the year after comments from the participants have been received and incorporated.

4.2 BWR and PWR Multi-Dimensional Kinetics Benchmark

Having been made aware by the Kjeller research centre of the Peach Bottom 2 turbine trip data, the Committee felt that these BWR data would be appropriate for code testing. It was decided that those interested in a comparison should proceed independently with the calculations. A specialists' meeting was no longer felt to be necessary. H. Neltrup and the Secretariat were asked to identify the relevant literature and distribute the references to the Committee members.

In response to an action from the previous meeting H. Küsters had investigated whether Kraftwerk Union could supply data for a PWR benchmark and co-ordinate the exercise. Unfortunately, this did not prove to be possible. In a last attempt the Committee asked P. Hemmig to contact EPRI once again, express to them the continued interest of the NEACRP in a PWR multi-dimensional kinetics benchmark, and invite them to provide the necessary data.

4.3 Noise Analysis Benchmark

The final record of the first stage of the computational benchmark conducted in conjunction with SMORN-III is currently being prepared. Three institutions in Japan, France and the Netherlands had agreed to pursue the benchmark into a second phase with an analysis of a physical benchmark. A description of the physical benchmark will be prepared by France.

4.4 Radiation Shielding Benchmark

The results of the shielding benchmark calculations had been presented and discussed at a specialists' meeting held in Paris in July 1982.

The draft proceedings of the LMFBR shielding benchmark calculations (NEACRP-A-503) were distributed to the Committee. In the discussion it was noted that the variations in the data were reduced considerably when comparing with the results presented at the Vienna 1976 meeting. The remaining discrepancies can be attributed mainly to different cross section preparation methods. A final version of the proceedings will be completed in early 1983 after comments have been received from the participants.

A draft report on the first results of the PWR shielding benchmark (NEACRP-A-506) was also distributed to the Committee. On the one hand, good progress has been made with predicting fast neutron fluxes, this being of particular importance for neutron damage calculations. On the other hand, the results indicate persisting difficulties with the thermal data processing, leading to considerable differences in the predicted gamma fields and neutron dose rates.

In the discussion on experimental benchmarks H. Rief introduced a status report on the EURACOS iron and sodium deep penetration experiments (NEACRP-A-505). The preparation of a benchmark based on EURACOS data is underway. Analyses of iron deep penetration measurements performed within the ASPIS and EURACOS programmes have led to new multigroup cross section adjustments (cf. NEACRP-L-259). In this context H. Rief was asked to arrange for the revised EURLIB-4 library to be transmitted to the Winfrith Shielding Group.

The Committee agreed that a decision to continue with another shielding specialists' meeting focusing on uncertainty analysis should await the outcome of the Sixth International Conference on Radiation Shielding, to be held in Tokyo in May 1983.

4.5 Intercomparison of Cell Heterogeneity Effects in Pin and Plate Geometries

The benchmark specifications (NEACRP-A-445) were issued in early 1982 by A.T.D. Butland who will also distribute some supplementary data by November 1982. The six countries (France, the F.R. of Germany, Italy, Japan, the UK and the USA) which have expressed their interest to take part in the intercomparison were invited to submit their solutions by the end of January 1983. The contributions will be analysed at AEE Winfrith, and a specialists' meeting on the results of the benchmark will be organized by A.T.D. Butland in June 1983.

4.6 Benchmark on Interactive Effects of Gadolinium Poisoned Pins in BWR's

A preliminary comparison of results from six different organizations (NEACRP-A-521) was presented by P. Wydler. An additional solution from Italy had been received only recently.

Larger variations than expected were observed particularly for the k_∞ of the reference cell. In view of the unexplained discrepancies the Committee invited Dr. Maeder to pursue the exercise and to identify additional information required to understand the source of the differences. It was indicated that France may also have a contribution to the benchmark problem. Further contributions should be submitted by December 1982.

5. General

5.1 Highlights of Recent Meetings of Interest to NEACRP

- Specialists' Meeting on the LMFBR Burnup Benchmark Calculations, April 1982, Cadarache, France

  See benchmarks, section 4.1

- Meeting of the Working Group on the Shielding Benchmark Exercises, July 1982, Paris, France

  See benchmarks, section 4.4

- Specialists' Meeting on Reactor Noise - SMORN-III, September 1981, Tokyo, Japan

  The proceedings of the meeting, published by Pergamon press, were distributed in the middle of 1982. T. Asaoka presented an overview of the meeting to the Committee (cf. NEACRP-A-557).

  There is significant interest in the development of better noise analysis techniques and, in particular, the application of reactor noise technology to nuclear power plants is growing in many countries.

  Four countries had formally proposed to host the next meeting on reactor noise (SMORN-IV) in about three years time (Australia, France, Italy and Switzerland). Since the French offer included a joint EDF/CEA participation in the organization of the meeting, the Committee agreed that the involvement of the French utility company would be an advantage. The safety related applications aspects of reactor noise analysis would thereby be emphasized, in particular the qualitative surveillance and diagnosis on nuclear power plants. The Committee therefore endorsed the French proposal.

5.2 Future Meetings of Interest to NEACRP

- Specialists' Meeting on 3-D Transport Solutions for Reactor Calculations

  The Committee noted that too few papers were presented on this topic and that the problem was still at an early technical development stage. It was felt therefore, that there would be no benefit in organizing a specialists' meeting at this time.

- Specialists' Meeting on In-Core Instrumentation, 1983

  It was reported that this meeting originally planned for 1982 had to be postponed because of a conflicting IAEA meeting including the same topic. The Committee felt that the NEA specialists' meeting could now be arranged for Autumn 1983 and asked members to nominate national representatives to an organizing meeting, including a representative of the Halden Project. M. Salvatores agreed to chair the organizing meeting.

- Sixth International Conference on Radiation Shielding, May 1983, Tokyo, Japan

  A technical programme committee meeting will be held during the week of 20 September 1982.

ANNEX 1

LIST OF PARTICIPANTS

Delegates

For Canada	Dr. P.M. Garvey	
For Japan	Dr. T. Asaoka	
	Dr. K. Shirakata	
For the USA	Dr. P.B. Hemmig	
	Dr. L.G. Le Sage	
	Dr. R. Peelle	(substituting for Dr. F. Maienschein)
For the countries of the European Communities and the European Commission acting together	Dr. J. Bouchard	(France) Vice-Chairman
	Dr. M. Salvatores	(France)
	Dr. H. Küsters	(F.R. of Germany)
	Prof. R. Martinelli	(Italy)
	Dr. J. Askew	(United Kingdom) Chairman
	Dr. A.T.D. Butland	(United Kingdom)
	Dr. H. Rief	(CEC)
	Mr. J. Debrue	(Belgium)
For the other European countries of the OECD	Mr. H. Neltrup	(Denmark)
	Dr. P. Wydler	(Switzerland) Scientific Secretary
Nuclear Energy Agency	Mr. J. Rosén	
	Dr. P. Nagel	Secretary
Observers (all sessions)	Dr. S. Cierjacks	(NEANDC)
	Dr. Y. Kazanskii	(IAEA)
	Dr. E. Khodarev	(IAEA)

Apologies for absence were received from Dr. McCulloch (Australia) and Dr. R. Caro (Spain). Following an established rotation Mr. Neltrup (Denmark) also represented Norway, Finland and Sweden, and Mr. Debrue (Belgium) also represented the Netherlands.

ANNEX 2

PRELIMINARY AGENDA FOR THE 26TH MEETING

Part A: Executive Sessions

1. a. Participants in the meeting
   b. Committee membership

2. Adoption of the final summary record of the 25th 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 NEACRP

6. Matters related to NEANDC

7. Activity report on the creation of a Joint Evaluated File of neutron data (JEF)

8. Arrangements for the 27th meeting of the Committee

9. Other business

10. Election of Committee officers

Part B: Technical Sessions

1. New Topics

1.1 Three dimensional transient models

1.2 Primary circuit modelling

1.3 Advanced fuel cycles:
Thorium cycle, high conversion thermal reactor cycles, long burnup cycles in thermal reactors, uranium and plutonium recycle in thermal reactors

1.4 Prediction of pin rating:
Validation of coarse mesh methods for estimating pin power, rating distribution and reactivity effects of Gd poisons in thermal reactors

1.5 Fine structure of energy deposition during operation

1.6 Prediction of rating distribution in large FBR cores through burnup

2. Topics Carried over from Previous Meetings

2.1 Validation of criticality methods, especially in geometries appropriate to reprocessing plants

2.2 Out-of-pile production of fissile material

2.3 Calculational methods for evaluating control rod effects in FBRs and their validation (in rod reaction rates and lifetime)

2.4 Reactor physics modelling of distorted cores

2.5 Beta and gamma decay heat measurements for fast and thermal reactors, particularly for Pu-239

2.6 Intercomparison of reaction rate measurements in fast reactors

3. National Programmes

3.1 Review of recent activities and national programmes

4. Benchmarks

4.1 Noise analysis benchmark (second stage)

4.2 Intercomparison of cell heterogeneity effects in pin and plate geometries

4.3 Benchmark on interactive effects of gadolinium poisoned pins in BWRs

4.4 PWR multi-dimensional kinetics benchmark

4.5 Reactivity scale and central worths benchmark

4.6 Radiation shielding benchmark

5. General

5.1 Highlights of recent meetings of interest to NEACRP

- Sixth International Conference on Radiation Shielding, May 1983, Tokyo, Japan

- Specialists' Meeting on In-Core Instrumentation, October 1983, Halden, Norway

- Specialists' Meeting on the Intercomparison of Cell Heterogeneity Effects in Pin and Plate Geometries, June 1983, Winfrith, UK

5.2 Specialists' meetings planned or proposed

- Specialists' Meeting on Reactor Noise - SMORN IV, Autumn 1984, France

5.3 Other business

ANNEX 3

NEACRP DOCUMENTS PRESENTED AT THE 25TH MEETING

"L" Documents

L-258 Reactor Physics Activities in NEA Member Countries

Australia	Japan
Austria	Netherlands
Belgium	Norway
Canada	Spain
Denmark	Sweden
Finland	Switzerland
France	United Kingdom
F.R. of Germany	United States of America
Italy	JRC Ispra

USSR (Annex)

L-259 G. Hehn, R.D. Bächle, G. Pfister, M. Mattes, W. Matthes
Adjustment of Neutron Multigroup Cross Sections with Error Covariance Matrices to Deep Penetration Integral Experiments

L-260 K. Suzuki, K. Miyagi, K. Aoki, T. Inoue, N. Ohtani
A Study on the Potential Safety Advantage of Large Heterogeneous LMFBRs

"A" Documents

A-502 NEA Data Bank Activity Report, September 1982

A-503 G. Palmiotti and M. Salvatores
Draft Proceedings of LMFBR Shielding Benchmark Calculations

A-504 G. Palmiotti and M. Salvatores
Draft Proceedings of the NEACRP Specialists' Meeting on the LMFBR Benchmark Calculation Intercomparison for Fuel Burnup

A-505 H. Rief
Integral Shielding Benchmarks; Status of the EURACOS Iron and Sodium Deep Penetration Experiments

A-506 G. Hehn
Draft Proceedings of PWR Shielding Benchmark Calculations

A-507 H. Tourwe
Fast Neutron Flux Measurements with the $^{93}Nb(n,n')$ Reaction - Brief Review of the SCK/CEN Activities in this Field

A-508 H. Tourwe, F. Kam, A. Fudge, W.L. Zijp, W. Mannhart, A. Thomas
Interlaboratory Comparison of Fluence Neutron Dosimeters in the Frame of the PSF Start-Up Measurement Programme

A-509 Standardization and Portability: NEA Data Bank Activities September 1981 to August 1982

A-510 H.K. Naess, T. Skardhamar
The FMS Treatment of Gadolinium Burnable Poison in Light Water Reactors

A-511 H. Giese, F. Helm
BIZET Programme; Analysis of Subcritical Rod Worth Measurements in the Single Annular Core BZD/2

A-512 C. Giacometti, G. Humbert, G. Palmiotti, M. Salvatores
Control Rod Calculation Methods and Uncertainties for a Power LMFBR Design

A-513 T. Takeda, K. Tanimoto, T. Yamamoto
Homogenization Method of Pin Rods in ZPPR-10A Assembly

A-514 J.L. Rowlands, M.J. Grimstone
Fast Reactor Control Rod Calculation Methods and Validation Experiments

A-515 H.F. McFarlane, P.J. Collins
Calculation Methods for Control Rods in LMFBRs

A-516 G. Humbert, R. Petiot, P. Coulon
A Control Absorber Rod in Phenix; Comparison of Calculated and Measured Worths

A-517 F. Helm, G. Henneges
Critical Experiments on the Reactor Physics of Distorted Cores

A-518 W. Maschek
Reactor Physics Modelling of Distorted Cores

A-519 M. Nakano, H. Tsunoda
Analysis of Fuel Slumping Experiment on FCA Assembly VIII-2

A-520 E. Wachi, T. Takeda, T. Sekiya
Effect of Cell Calculation Model on the Analysis of Fuel Slumping Experiment

A-521 C. Maeder, P. Wydler
Burnup Calculations for a BWR Lattice with Adjacent Poisoned Fuel Rods

A-522 G. Walker, R.J. Brissenden
MONK 6.1 Validation

A-523 H. Küsters
General Considerations on Physics and Technical Aspects for Storage, Reprocessing, and for Shipment of Reactor Fuel

A-524 A. Santamarina, B. Nouveau
The CRISTO III Experiment

A-525 Y. Naito, J. Katakura, M. Yokota
A Study on a Nuclear Criticality Safety Evaluation Technique

A-526 G. Minsart
Parametric Reactivity Study of Compact PWR Fuel Storages Including Fuel Burnup Effect

A-527 Y. Nakahara
Evaluation of Computational Models for Spallation and Fission Reactions Used in an Accelerator Breeding and Transmutation Analysis Code

A-528 C. Budd
Some New Approaches to the Efficient Solution of the Transport Equation

A-529 J.L. Hutton, P.B. Kemshell, K. Robichaud
Monte Carlo Calculations of the WSGHWR Using the Group Data Version of MONK

A-530 M. Cosimi, C. Giacometti, G. Palmiotti, M. Salvatores
Group Collapsing Strategy for 3D Design Calculations

A-531 C. Giacometti, J.C. Estiot, G. Palmiotti, C. Grondein, G. Le Cardinal, M. Ravier
Recent Developments and Improvements in the Code System for the Neutronic Design of Fast Breeder Reactors at CEA

A-532 T. Fujimura, Y. Nakahara, M. Matsumura
Iterative Solution of the DFEM Algorithm for the Three-Dimensional Neutron Transport Problems

A-533 W.H. Taylor, M.F. Murphy, M.R. March
Beta and Gamma-Ray Decay Energy from Fragments from the Fission of U-235 and Pu-239 in a Fast Reactor

A-534 M.F. James
A Brief Survey of Experimental and Theoretical Data on Fission Product Decay Heat from U-235 and Pu-239

A-535 R.W. Peelle, J.K. Dickens
Fast Reactor Short-Term Fission-Product Decay Heat and its Uncertainty

A-536 M. Akiyama, S. An
Measurements of Gamma-Ray Energy Release Rates Following Fast-Neutron Fission of U-238, Th-232 and Natural Uranium

A-537 P.J. Persiani, Kalimullah
Isotope Correlations for Safeguards Surveillance and Accountancy Methods

A-538 J. Debrue
Subcriticality Monitoring by Means of the Pulsed Neutron Technique; Measurements on Single PWR-Type Subassemblies

A-539 A. Giacometti, R. Girieud
Application of Calculated Isotopic Correlations in the Determination of the Input Balance at the La Hague Reprocessing Plant

A-540 M.F. Murphy, J.M. Stevenson, W.H. Taylor
Comparison of Measured Neutron Reaction Rates in Fast Reactor Steels with Values Calculated Using FD5 Data

A-541 A.D. Knipe
The Importance of Electron Transport in Reactor Heating Calculations

A-542 D.W. Maddison, G. Ingram
ANL/AEEW Comparison of Reaction-Rate Ratio Techniques in ZEBRA

A-543 R. Böhme, B.L.H. Burbidge
A Comparison of Central Reaction-Rate Ratio Measurement Techniques in BIZET Cores

A-544 J.M. Gasidlo, D.W. Maddison, R.J. Armani, J.A. Morman, S.G. Carpenter
A Comparison of Counting and Mass Spectrometer Methods for Measuring U-238 Capture

A-545 H. Nakamura, N. Tsuji
On the Reactivity Scale for Large LMFBR Cores

A-546 K.S. Smith, R.W. Schaefer
Recent Developments in Small Sample Worths

A-547 L.G. Le Sage
Benchmark Intercomparisons of Reactivity Worths in an LMFBR-Type Plate Critical Assembly

A-548 J.M. Stevenson
Comparison of Reactivity Scales Based on Kinetics Measurements and Plutonium Fuel Worths in Eleven ZEBRA Assemblies

A-549 D. Calamand
Gamma Heating Measurements in a Heterogeneous Core Configuration

A-550 H.F. McFarlane, C.L. Beck, H. Henryson II
Sodium-Void Reactivity in a Heterogeneous LMFBR

A-551 M.J. Lineberry
U.S. Fast Critical Assembly Developments

A-552 M.J. Lineberry, H.F. McFarlane, P.J. Collins
Physics Assessments of LMFBR Integral Parameters

A-553 R.W. Peelle
Note on CSNI Activities on Nuclear Criticality Safety Computations

A-554 L.G. Le Sage
Draft Notes on the Reduced Enrichment Research and Test Reactor (RERTR) Program

A-555 Y.A. Kazanskii et al.
Investigation of Physical Characteristics During the BN-600 Reactor Start-Up

A-556 M. Kawai, J. Itoh
A Study on Shielding Calculation Technique Using the Albedo-S_N Method

A-557 J. Hirota, Y. Shinohara
The Third Specialists' Meeting on Reactor Noise - A Brief Overview

ANNEX 4

LIST OF ACTIONS

1.	Secretariat	Distribute the NEACRP summary record to NEANDC and vice versa.
2.	All members	Nominate by the end of October 1982 representatives to an organizing committee to set up a specialists' meeting on in-core instrumentation.
3.	Le Sage	Distribute a review on the use of low enriched fuel in research reactors.
4.	Wydler	Distribute the chapter on critical assembly descriptions of the defunct "Review of Topics in Fast Reactor Physics" book.
5.	Secretariat	Keep members informed of IAEA (in particular IWGFR) activities.
6.	Secretariat	Distribute CSNI report on transport flask criticality.
7.	Secretariat	Obtain and distribute NEANDC comments on the high priority data measurement request list (NEACRP-A-500).
8.	All members	Direct comments on whether the high priority data measurement request list satisfies the priority requirements to Mr. J.L. Rowlands at Winfrith.
9.	All members	Assist NEANDC in identifying contributors on data requirements for fission reactors for the planned volumes of the monograph series on "Neutron Physics and Nuclear Data in Science and Technology".
10.	Küsters Askew Secretariat	Invite CSNI to comment on the accuracy requirements in the physics modelling of whole core accidents in fast reactors.
11.	Secretariat	Obtain and distribute the French and Dutch second stage noise analysis benchmark problems and distribute the Japanese one.

12.	Hemmig	Express the Committee's continued interest in the PWR multi-dimensional kinetics benchmark to EPRI and invite them to provide the data for the exercise.
13.	Rief	Ask Dr. G. Hehn of IKE to send the revised EURLIB-4 library to Dr. J. Butler at Winfrith.
14.	Neltrup Secretariat	Distribute the references to documents on the U.S. Peach Bottom 2 turbine trip.
15.	All members Secretariat	Submit solutions for the pin and plate heterogeneity benchmark to Mr. J.M. Stevenson at Winfrith by the end of January 1983.
16.	Butland	Issue the supplementary data for the pin and plate heterogeneity benchmark and organize a specialists' meeting in June 1983.
17.	Wydler	Invite Dr. C. Maeder to continue the BWR poisoned lattice cell benchmark problem and to identify additional information required.
18.	All members	Submit additional solutions for the BWR poisoned lattice cell benchmark to Dr. C. Maeder at Wuerenlingen by December 1982.
19.	Secretariat	Communicate to NEANDC the data library problems associated with beta and gamma fission product decay heat.
20.	All members	Report plans for using the new ANSI-77 Fortran standard.
21.	All members	Recalculate β_{eff} for the clean core of the LMFBR burnup benchmark problem and send results to Dr. M. Salvatores at Cadarache.
22.	All members	Send any comments on the outline specification of the reactivity worth benchmark to Dr. L.G. Le Sage at Argonne.
23.	Le Sage	Formulate specifications for the reactivity worth benchmark for the next meeting.
24.	Askew	Provide final input to Secretariat for the renewal of the NEACRP mandate.