NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD OF THE TWENTY-EIGHTH MEETING (TECHNICAL SESSIONS)
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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT
NUCLEAR ENERGY AGENCY
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TECHNICAL SESSIONS

A complete list of all the papers presented at this meeting is given in Annex 2. In the technical sessions five new topics were introduced and seven topics were carried over from the previous meeting. Following an established practice, for each topic a committee member was assigned to prepare a draft summary and these summaries were reviewed before the close of the meeting.

1. New Topics

1.1 Status and Future Roles of 3-D Deterministic Transport Methods (S_n, Nodal, Finite Element, etc.)

Rapporteur: S. Matzuura

The ten papers on this topic covered various subjects. In the first (L-285, FRG) progress of the synthesis approach for the treatment of three-dimensional neutron transport was presented. Two-dimensional angle-independent trial-functions calculated with an S_n-code DIAMANTZ are synthesized in the third space-direction with solutions from a diffusion approximation. First comparisons with a three-dimensional S_n-code agree in the eigenvalue to 0.1%.

Progress was also presented on a three-dimensional nodal reactor analysis system based on equivalence theory and the burnup-corrected nodal expansion method. A unique feature of the system is the avoidance of any full core or quarter core fine-mesh calculations. In addition to these, the present status of development of 3D diffusion methods in hexagonal geometry was commented on. Recent effort is concentrated on new efficient perturbation capabilities for the modular code system IANS.

Paper A-704 (Sweden) discussed the recent version of POLCA-4 which is a modification of POLCA-3 nodal code for the calculation of the 3D core power distribution. During the development of the code, it was found that the albedos influenced the power distribution, and a sophisticated approach was necessary to find the appropriate albedos. The procedure was presented to determine the desired albedos.

Paper A-705 (Japan) described a 3D neutron transport calculation based on a variational principle. The solution algorithm uses a double finite element method, in which space and angle finite elements are employed. A combination of triangular and quadrangular prism elements is adopted to simulate the practical nuclear facility configurations as accurately as possible and six functions are adopted as tentative bases in the angular space to smoothly represent the angular distribution of the neutron flux. Two numerical examples on FBR cores are presented and compared with the two dimensional S_n code TWOTRAN-II. Other aspects for improvement are identified.
Paper A-706 (Japan) presented a fundamental outline of the 3D discrete ordinate code BERMUDA-3DN which solves a time-independent transport equation for three-dimensional, rectangular, multi-regional geometry using a direct integration method in a multigroup model. Numbers of spatial mesh points on x, y and z axes are extremely limited for computers available now, because angular discrete ordinates comprise 80 points for each of the spatial points. By this reason, the present version is a trial only to construct a logical structure.

Paper A-707 (Japan) discussed a 3D radiation transport code, TRISTAN. The code is based on the method of direct integration and the usual concept of group theory and uses group-angle transfer matrices from the double-differential cross section data instead of the Legendre polynomial expansion. Several techniques, such as separation of radiation into two components and conservation of radiation, are adopted from the standpoint of numerical accuracy. Furthermore, two new techniques, stratification of the angular mesh and a kind of forward-adjoint coupling, are introduced to enhance its applicability to practical shielding analysis. The validity of TRISTAN has been verified by good agreement with the Monte Carlo code MCNP and with the experimental values in the analyses of the JRR-4 streaming experiments.

Paper A-708 (Japan) presented the derivation of equations and numerical calculations for the X-Y-Z three-dimensional transport code TRITAC, developed on the basis of the discrete ordinates method with the diffusion synthetic acceleration (DSA) technique. In TRITAC, Larsen's scheme is extended to three-dimensional problems, and the DSA method is applied not only to the inner iteration but also the outer iteration. The code was validated by comparing numerical calculations with TWOTRAN-II, and the CPU time was reduced by a factor of 3 compared with calculations with the rebalance method.

Paper A-709 (Japan) gave the calculational method of the three-dimensional discrete ordinates transport code TRITAC in hexagonal-3 geometry, developed for the analysis of LMFBRs with hexagonal right prismatic subassemblies. Numerical calculations for a simple test problem gave good agreement with results extrapolated from TWOTRAN-II.

Paper A-710 (Japan) discussed the accuracies of two conventional methods to calculate three-dimensional transport corrections, which are based on the 2D RZ model and the combined XY, RZ and RY model, respectively. 3D transport corrections for keff control rod reactivity worths, neutron spectra and power distributions are calculated for the ZPPR-9 10A, 13A, 13B/1 and 13C cores, and are compared with results of the 3D and 2D transport codes TRITAC and TWOTRAN-II. The combined 2D model predicted the 3D transport effects with good accuracy.
Paper A-711 (UK) presented the recent development of the transport code MARC/PN, which solves the multigroup transport equation in most geometries of interest using finite element or finite difference techniques. The code is based on the method of Legendre expansion. Two recent developments are described. The first consists of a treatment allowing for anisotropic scattering, and the second modification is a link to the interactive graphics package GEOMOD/SUPERTAB. Several numerical example show the effects due to the order of the PN expansion, number of energy groups, and spacial dimensions.

In the final paper, A-712 (France), the 2D$n$ transport code BISTRO and its performances were presented. At present, three methods have been implemented: SLOOR (successive even and uneven line overrelaxation) accelerated by optimal W, ADI (alternating direction implicit method) accelerated by effective precalculated W, and SIM (strongly implicit method) accelerated by the conjugate gradient method. The effectiveness of the Chebychev method was identified. Numerical examples are presented for RACINE 3 and SUPERPHENIX and compared with the results obtained by the DOT code.

In the discussion, it was pointed out that relevant test problems should be selected to clarify the effectiveness of 3D codes, especially due to ray-effects and heterogeneity. Problems concerning convergence characteristics, energy scheme and I/O treatment were also discussed. It was agreed that as the topic was being continuously investigated in member countries, it should be retained on the agenda for next year.

1.2 Application of Spatial Kinetics to Reactivity Measurements

Rapporteur: L.G. LeSage
Papers A-713 to A-717

These session contained five papers, three of which discussed the evidence of and the correction for spatial and space-time effects in fast reactor reactivity measurements. The first paper (US, A-713) discussed the reactivity techniques used at ZPPR. Clear evidence of space-time effects was evident in ZPPR-13C, a large, decoupled fast critical assembly with fuel zones containing different fissile isotopes.

The second paper (USSR, A-714) discussed reactivity measurements in the USSR on critical assemblies and on the BN-600 power reactor using both an inverse kinetics (IKM) and a pulsed alpha technique. Results from measurements on BN-600 using a digital reactimeter (based on the IKM technique) were given. Efficiency corrections were shown to improve consistency in the measured reactivity. The measurements on the heterogeneous RACINE 1F Core at MASURCA. Significant spatial effects were observed and there was some evidence of small space-time effects.
In another paper (US, A-715) the Cf-252 source-driven neutron noise analysis method of Mihalczo was discussed in detail. It was shown that this method is applicable to the measurement of $k_{eff}$ of a wide variety of systems, including many non-reactor applications. Efficiency corrections are not required; however careful placement of the detector relative to the neutron source is required.

The final paper (France, A-716) described a new method for measuring control rod worth for PWRs based on the evaluation of ex-core detector signals along with the use of a feedback model and weighting corrections. The method was shown to be very fast and convenient while remaining reasonably accurate. The method should be applicable to other types of reactors.

The committee noted that the reactivity measurement methods described in the final two papers are interesting developments that should have many useful applications. The data presented in the first three papers again showed the importance of efficiency corrections for reactivity measurements in large fast reactors. Space-time effects are much less important, and only appeared clearly in ZPPR-13C which contained special zones of different isotopic loading.

1.3 Resolution of Local Heterogeneous Effects due to Geometry in Fast Reactors

Rapporteur: J.M. Stevenson
Papers A-718 to A-720

The session contained three papers on very different subjects.

Paper A-718 (US) presented a summary of measurements in the ZPRs over about 10 years which were related to the central worth discrepancy. Two different types of measurements were used; samples in the plate structure and cylindrical samples in a tube whose axis was perpendicular to the plates. Calculational effort which includes corrections for variations in flux and adjoint has brought into satisfactory agreement all the measurements with foils in the plates and next to the plates. Difficulties were found with the cylindrical samples where differences of several percent still exist. The central worth discrepancy has largely been resolved, while admitting that some of the measurements with cylindrical samples must be discounted. It was noted that a very similar situation has been found for ZEBRA.

Paper A-719 (Japan) considered measurements in an axially heterogeneous fast reactor core in FCA and their analysis. There is significant support in Japan for such a design which, because of the low reactivity loss, requires reduced control rod worths which may be advantageous in an earthquake situation. The FCA mock-up had a disc-shaped internal breeder, 20 cm. thick in a 90 cm. high core with radial driver zones. Only axial measurements were therefore relevant. The expected axial power flattening was found, but there
were significant C/E discrepancies for reaction rates in the internal and axial breeders, even after transport corrections. H. Kuesters suggested that all heterogeneous designs were now out of favour but it was noted that two of the reference designs in the USA were still (radially) heterogeneous.

Paper A-720 (France) described the method which has been adopted in France to allow for the internal heterogeneity of control rods with a view to reducing the rather large errors in calculated rod worths in LMFBRs. Earlier studies had shown that the worth error associated with smearing the composition over the whole subassembly area was approximately 10% and was dependent on the radial position and axial insertion. The method chosen was an extension of that of Rowlands to produce effective homogenised cross sections over the subassembly area by conserving reactivity via perturbation formulae using real and adjoint fluxes. These homogenised cross sections were obtained for a central rod in a 1D cylindrical model. Tests of these cross sections were then made in a 2D XY quarter-plan simulation of SPX1. The actual values found for the heterogeneity effect on rod worths using a detailed rod representation and transport theory for a central rod and two rod rings were approximately 9% and agreed within 0.3% of those obtained using the homogenised cross sections. Although diffusion theory was not able to reproduce the effects for the heterogeneous representation, it did when using the homogenised cross sections, after correcting for mesh effects.

The chairman remarked that this topic had been retained for the presentation of further CADENZA analysis. It was noted that the US had a paper and that there was some discussion in the UK National Report.

The meeting decided to keep the topic for next year, possibly for more CADENZA results and for the study of the heterogeneity effect of secondary shutdown rods.

1.4 Application of Neutron Noise in Reactor Systems (Excluding Mechanically Induced Noise)

Rapporteur: L.G. Lesage

Paper A-721

Only one paper (US, A-721) was contributed for this agenda topic. In this paper, Bennett summarizes the results of $\beta_{eff}$ measurements in six different ZPR fast critical assemblies over a period of years. A two-detector coincidence experimental method is described. The $\beta_{eff}$ results and their C/E values based on ENDF/B-4 and 5 calculations are presented for the six assemblies with cores with U-235 fuel, Pu-239 fuel, and cores with mixtures of U-238 with either U-235 or Pu-239. The C/E results are consistent with a small increase in nu-delayed for U-238 over the current ENDF/B-5 value.
1.5 Special Applications of Gamma and Neutron Source Modelling (Medical, Therapy, Beam Tubes, etc.)

Rapporteur: L.U. Lesage
Paper A-722

The single paper in this agenda topic (US, A-722) described a Monte Carlo method, based on the VIM code, for calculating the neutron flux emergent from a reactor beam tube. The Monte Carlo calculation is limited to a small region of the reactor near the head of the beam tube where nearly all of the emergent neutrons originate. As a result of a calculation using this method the head end of the beam tube in one experimental reactor will be moved slightly further from the core to increase the ratio of thermal to fast flux in the beam without significantly reducing the thermal flux.

2. Topics Carried Over From Previous Meetings

2.1 Monte Carlo Whole Core Models

Rapporteur: J.R. Askew
Papers A-723 to A-725

Paper A-723 (US) reported the state of CADENZA and ZPPR analysis, relating especially to the pin/plate discrepancy. The consensus view was that a small discrepancy (0.3 – 0.6%) remained in both sets of experiments when analysed using deterministic methods, though Takeda had obtained a solution showing good agreement and the reason for the different result in his analysis was still being sought. The observed discrepancy was at a level which might be ascribed to combined uncertainties of composition, nuclear data and calculational models. Monte Carlo studies did not yet rule out the last contribution, perhaps the most important missing evidence being a Monte Carlo solution for the two CADENZA cases which showed the biggest difference.

LeSage commented that, although not presented here, there is at least one Monte Carlo (VIM) calculation of the CADENZA pin and plate cores. These Monte Carlo results showed about the same pin/plate discrepancy as the deterministic calculation.

Stevenson noted that supplementary measurements in which thinner and thicker plutonium plate configurations were used had not been well predicted by calculation. Kuesters noted that similar measurements had been performed in SNEAK and that he would seek to bring them forward. The observation suggested that resonance shielding in the plutonium metal might be a contributory factor, the difference reflecting either modelling or basic data errors.

It was agreed that the issue was no longer one of urgent practical importance, but was such as to warrant a continuing effort to resolve.
Paper A-724 (UK) reported Monte Carlo calculations using MONK and WIMS B1 data for a 7% enriched benchmark lattice. Comparison with a previously reported 3% case gave confidence in the leakage model and, with reaction rate measurements, suggested that the U-238 capture cross section used was a little too low. Golinelli commented that he had similar results for UO$_2$/H$_2$O lattices, and that a thermal resonance in U-238 had been postulated to fit both multiplication and temperature coefficient results. The UK Monte Carlo model had recently been enhanced by provision of a subgroup model, obviating the need for prior lattice calculation of resonance shielding.

Paper A-725 (UK) carried forward the discussion of bias in eigenvalue and standard deviation estimation in a Monte Carlo calculation by citing an extreme case of a small, supercritical sphere in a large, subcritical store. Standard powering techniques systematically underestimated both variables, even when positive steps were taken to make sure that the sphere was sampled. A 'superhistory powering' technique, in which particles were tracked for several generations before renormalization, was described and shown to remove the problem. Rief drew attention to the work of Dubl on this general problem, and commented that the use of the source multiplication matrix facility of KENO would at least have identified the existence of a problem to the user.

Two papers were tabled for information prior to publication. Rief, Gelbard, Schaefer and Smith reviewed Monte Carlo techniques for analysing reactor perturbations. The special version, KENO-EUR used in the analysis was under test by CSS Garching and would later be distributed. The second paper, by Rief, was a draft contribution to a proposed CRC handbook on uncertainty analysis, edited by Y. Ronen and was entitled Monte Carlo Uncertainty Analysis.

The committee noted the important progress which was being made in this field, but concluded that the topic should not remain on the agenda for the next meeting with its present emphasis.

2.2 Physics Problems of Tight Pitch Lattices

Rapporteur: P. Wydler
Papers A-726 to A-731 and L-286

Three papers (A-726, A-727 and A-731) are related to the first phase of the High Converter PWR experiments carried out in the PROTEUS reactor at Würenlingen.

The first paper, A-726, from Switzerland, gives a review of lattice calculations using various methods and data sets. The comparisons of C/E's for $k$-infinity reaction rate ratios and the $k$-infinity void coefficient suggest that commonly used LWR calculational tools do not meet all requirements, at least not for the entire range of moderator voidage and plutonium enrichment investigated in the PROTEUS experiments.
A Swedish study of the PROTEUS lattices using the CASMO Code is reported in paper A-731. C/E's for the reaction rate ratios and the heterogeneity factors associated with the two rod nature of the lattice are given for two different data libraries. The performance of the libraries and the effects of data adjustments in the epithermal and fast energy range are assessed in the paper.

A Japanese paper, A-727, deals with a generalized treatment of the Dancoff factor for infinite arrays of multi-region cells including absorber lumps with different absorber concentrations. A sequence of validation checks using a Monte Carlo code showed that in the case of the heterogeneous PROTEUS lattices the proposed formalism allows the effective resonance cross sections to be calculated more accurately than an equivalent single rod lattice model.

Paper A-730 gives an overview of the French activities in the field of the High Converter PWR. In the past twelve months a first phase of the ERASME programme at the reactor EOLE has been successfully completed using a large test lattice with a fuel-to-moderator ratio of 0.51. Included are measurements of k-infinity, the conversion ratio, the fission rates of the principal heavy isotopes and the void coefficient as well as absorber reactivity studies. In general, the measurements were found to be in good agreement with predictions based on a modified version of the APOLLO code. In the near future the ERASME programme will be extended to other fuel-to-moderator ratios. Complementary measurements of captures in heavy isotopes will be made in the framework of the ICARE programme. This programme involves the irradiation of doped fuel pellets in the reactor MELUSINE at Grenoble, followed by (destructive) isotopic analysis. An integral measurement of captures in the fission products will be carried out in the reactor MINERVE at Cadarache using the sample oscillation technique.

A Japanese paper, A-729, describes another experimental programme which will be carried out at the FCA facility at JAERI. In the first phase the assembly will be loaded with an enriched uranium fuelled test zone, and in the second phase with a plutonium fuelled test zone. The paper gives results of a preliminary analysis of the characteristics of these test zones.

A further paper from Japan, A-728, presents results of a control rod study. An interesting result is that in a High Converter PWR spectrum the strong resonance absorber Hf has a smaller control rod worth than B4C. The paper gives some interesting correlations between the control rod worth and particular lattice parameters.

The last paper, L-286, from the Federal Republic of Germany gives a comprehensive review of a KfK validation of High Converter PWR procedures. It is shown that the "modified FBR" calculational method and the one-dimensional collision probability program WEKCPM adequately predict the k-effective values measured in the SNEAK assemblies 128, 12F1 and 12F2. The accuracy for the moderator void reactivity is comparable to that for the void reactivity in fast
reactors. Furthermore, the KARBUS burn-up system adequately predicts k-infinity, k-effective and the conversion ratio for a wide range of fast and thermal lattices, including the k-infinity of the PROTEUS-LWHCR Core 1. In the presentation Kuesters concluded that the KFK data and calculational methods fulfill the current needs for High Converter PWR design studies.

In the general discussion it was noted that both the SNEAK and the ERASME experiments are in good agreement with their respective calculations. Since the ERASME and the PROTEUS programme are continuing and new experiments are about to begin in FCA, the committee felt that, in a more general form, the topic should reappear on the agenda of the next NEACRP meeting.

2.3 Physics Modelling of Neutron Sources (Research Reactors, Spallation Sources, etc.)

Rapporteur: P.M. Garvey

The first paper (A-732) from the US described a steady state fission reactor under consideration as a new facility for the neutron research community. This reactor similar in concept to HFIR but with a D2O reflector would provide a peak thermal neutron flux in the reflector of $5 \times 10^{15}$ n.cm$^{-2}$s$^{-1}$ at a power of 200 MW. This paper describes the types of facilities that would be serviced by the reactor, the range of possible activities supported, the core design and potential costs. In December 1985 a workshop is scheduled to be held at the NBS, Washington to further discuss user needs and potential facilities.

The second paper (AECL-8841) referenced in L-285 (Canada) describes a neutronics model under development as an operational code for the NRU reactor at CRNL. This reactor has a highly heterogeneous core supporting a wide range of experimental facilities. The model is a fine mesh, two group, three dimensional diffusion code and also uses the discontinuity factor formalism. Lattice parameters are generated using the WIMS-CRNL code with albedos where appropriate. The model has been validated against experiments in the zero power research reactor ZED-2 for specific situations.

A spallation neutron facility planned for SIN was described in paper A-733 from Switzerland. This facility, using the excess beam from the SIN cyclotron, consists of a Pb/Bi target surrounded by moderators and beam tubes. The paper describes the facility and various experiments to establish its characteristics. A surrounding Be sleeve was shown, contrary to calculation, to slightly depress the peak neutron flux.

Paper L-287 from FRG provides a very complete review of the characteristics of various kinds of non-fission reactor intense neutron sources either in operation, planned or conceived. The data needs for such sources are also discussed.
The potential application of a spallation neutron source as a fusion materials facility was described in paper L-288 from the JRC. Its characteristics were intercompared with FMIT, showing a factor of 10 improvement in dpa.

The last paper A-734 from the US described the application of the VIM Monte Carlo code to the calculation of time dependent fluxes in the IPNS facility. The main purpose of the calculation was to evaluate the neutronics of the system for an enriched uranium target that would realize a factor of 3 gain in source strength.

2.4 Physics Modelling of Fusion Blankets

Rapporteur: G.E. Whitesides
Papers A-735 to A-739, A-741 and L-285

The seven papers discussed in this session covered several aspects in the development, analysis and use of fusion blanket systems.

The Japanese reported on two experimental programs. The first (A-738) involved a series of experiments to measure reaction rates, such as tritium production rates, and time-of-flight experiments to measure angle-dependent neutron spectra. The second (A-737) involved a benchmark experiment on tritium breeding in a lithium sphere. Both sets of experiments have been analyzed and reasonably good agreement between experiment and computations was shown.

The US representatives reported (A-741) their analysis of several of the data which had been generated in the first Japanese program listed above. This work is not yet complete and only preliminary results were reported.

The Japanese representatives also reported studies (A-735) to evaluate methods of increasing the tritium breeding in a fusion blanket and a study (A-736) of the effect of cross section uncertainty on tritium blanket breeding calculations. By using materials with \((n, 2n)\) reactions, which are mixed with \(\text{Li}_2\text{O}\), it was demonstrated that tritium breeding can be substantially increased. The cross section sensitivity study revealed that the major uncertainty would be produced by the structural steel in the system.

The FRG representative discussed (L-285) their fusion blanket work and reported the development of a new transport code which will use doubly-differential cross sections rather than the usual Legendre Polynomial expansion.

The JRC representative presented a paper (A-729) which gave data on activation of the first wall of a fusion reactor and the subsequent data needed to assess problems in removing, transporting and storing these components.
In the discussion following the presentations the following conclusions were reached:

- Discrepancies between experiments and calculations had been shown which demonstrated the value of the experimental programs.

- Some of the discrepancies could have resulted from a lack of maturity in analyzing this type of experiment and possibly will improve with time.

- We are still years away from actual need for much of this data.

- Due to the continuing experimental programs, and the preliminary nature of much of the analysis, this topic should remain on the NEACRP program.

2.5 Calculation of Structural Reactivity Feedback Effects such as Bowing during the Normal Operation of Reactors

Rapporteur: M. Salvatores
Paper A-740

The only new information provided to the committee is contained in a paper (A-740) from France, and concerns PHENIX reactivity coefficients. The major point of interest is that no evidence of a correlation of reactivity loss evolution with bowing was found, in particular due to the competing contributions to the reactivity loss caused by bowing, fuel pin elongation, increase in burn-up and the presence during the time in which the evolution of reactivity was recorded, of a fertile subassembly in the core.

Although the members felt that at present the topic should not be carried over, the subject is of continuing interest and new work in the field, in particular from the US, will be eventually considered at future meetings.

2.6 Advanced Fuel Cycles for Thermal and Fast Reactors

Rapporteur: C. Golinelli
Papers A-742 to A-745 and L-285

Eight papers were presented at this session. Two of them dealt with a new concept for the Breeder Reactor (US).

Papers A-742 and A-743 show its advantages. The main option is to use a metallic fuel (for instance a uranium-plutonium-zirconium alloy). The other important points are the burnup extension and a small core.

The consequences are:

- a simplification of the fabrication process: A simple fuel rod contains only two or three 20-inch-long fuel pins obtained by melting the alloy.
- easier and more economical reprocessing.
- a better heat transfer between fuel and coolant.
- the inherent safety is enhanced.
- an increased conversion ratio.

A demonstration is in progress in the core of EBR-11.

Canada presented three papers (AECL-8703 and 8839 and a CNS paper, referenced in L-285, Canada).

The current generation of CANDU reactors have been designed for the natural uranium fuel cycle. With a low enrichment uranium fuel or with a mixed oxide fuel, the burnup can be considerably extended, with consequent potential economical benefits. This strategy would change the neutronic parameters, particularly the neutron flux distribution. Its effect requires the redeployment of the reactivity devices or the use of different fuel management schemes. In the checkerboard fuelling scheme the axial flux and power distribution has a shape similar to that for natural uranium fuelling.

The last three papers concern plutonium recycling in thermal reactors.

Paper A-744 (Japan) describes the ATR which is designed to use MOX fuel. ATR is a heavy-water moderated, boiling light water cooled reactor. The FUGEN prototype has been in commercial operation since March 1979. A critical experiment (DCA) allows the investigation of the characteristics of plutonium utilization. Improvement of the WIMS Nuclear Data Library has occurred through isotopic analysis of spent fuel and by microparameter experiments.

The $k_{ef}$ calculations for FUGEN are in good agreement with experiments. A specific study shows the effect of variation of the isotopic composition on burnup.

In Germany (L-285) plutonium recycling in LWRs has been evaluated. A mixture of the multi-recycled and the first generation plutonium characterizes a cycle in equilibrium. Also cited are the investigations of KWU on thorium utilization in PWRs.

The French decision to recycle plutonium in PWRs is discussed in paper A-745. It has been decided to recycle some 8 tons in 1988, increasing to 85-100 tons after the year 1992. A manufacturing facility (MELOX) is planned at Marcoule. An agreement between Belgo-Nuclaire and Cogema has been signed (COMMOX). The production of MOX fuel will be some 35 tons/year at Deassel and 10-15 tons at Cadarache.
A demonstration program is being prepared. The first loading is planned for 1988. When MELOX is in operation, 11 PWRs will be reloaded with one third of the recharge every year. The benefit is that natural uranium consumption and SWU requirements are then decreased by some 8%.

2.7 Studies of Reactivity Loss due to Burnup in FBRs

Rapporteur: K. Shirakata
Papers A-746 to A-748

Paper A-746 from Japan presents measurements and analyses of the burnup reactivity and its breakdown by nuclide for the JOYO MK-II core. The effect on burnup reactivity due to the time difference between U-238 capture and Pu-239 production was observed, and was taken into account in the analysis. After this correction good agreement is obtained between calculation and experiment. This effect is predicted to amount to about 0.1% $\Delta k/k$ for a 1000 MWe-size LMFBR.

Paper A-747 from UK presents the reactivity history of PFR during the period 1975 to 1984. The main purpose of this paper is to identify and investigate any long term unexplained reactivity changes which might have occurred during the life of PFR. The rates of reactivity loss during this period had C/E values of -0.92. The calculated and measured balance points at the start of each reactor cycle have also been examined and found to be in good agreement with a systematic overprediction of the calculated $k_{eff}$. This overprediction was consistent with that expected from analysis of a PFR mock-up in ZEBRA. No measurable unexplained reactivity effects have come to light during this investigation.

Paper A-748 from France discusses the source of inconsistencies between separate fission isotope cross section assessments and lumped fission product integral tests, and also discusses the model improvements for LMFBR design calculations. Dependence of the uncertainty on burnup and several effects due to inelastic scattering cross section, Pu isotope yields and gaseous and volatile FPAs have been investigated. An experimental study of Cs migration at high burnup is underway on PHENIX irradiated fuel and the results will be available in 1986.

The topic has been well reviewed in the last two years and will be dropped next year.

3. National Programs

Reports on the reactor physics activities in the NEA member countries were summarized and discussed. The USSR also reported on their FBR reactor physics activities during the past year. The full reports will be consolidated into a report (L-285) to be issued by the NEA secretariat.
4. Benchmarks

4.1 Radiation Shielding Benchmark

Paper L-289, identifying adjustments to the Fe cross sections, was presented by Rief. A previous action (12) on Rief to arrange a meeting on the Radiation Shielding Benchmark was amended to early 1986 (see Action 6). A further action was placed on Rief that a review of the Stuttgart meeting on shielding be distributed (see Action 2).

4.2 Criticality of Fuel Undergoing Dissolution

Progress was reviewed by Whitesides. Seventeen critical experiments have been chosen as benchmarks, some of which contain either B or Gd in solution or Hf in solid form. The results to date are encouraging but a more severe test will be for non-critical situations. Further problems, covering fuel dissolution, will be available by June 1986.

4.3 Heat Transfer in Transport Flasks

Progress in this area was reviewed by Whitesides. The problems to date have been relatively simple with radiation the dominant mechanism. The results have been with one exception in good agreement.

Three further problems have been proposed, each with a further complication. For these problems there are only five participants, France, Italy, Sweden, UK and USA.

The question was raised as to whether the data could be stored by the NEA Data Bank (see Action 13).

4.4 Shielding in Transport Casks

The status of this benchmark (L-290) was reviewed by Kuesters. For the first phase, three simple problems have been defined for which solutions have been requested by 1st March 1986. A meeting will be arranged for June 1986 to discuss the results (see Action 7). The second phase problems will be more complicated.

4.5 Noise Analysis

In response to Action 6 of the 27th meeting a Task Force Report (A-699) on the State-of-the-Art of Reactor Noise Analysis was tabled. This report covers several aspects including the state-of-the-art, benchmarks, and the formation of a data library. The committee heartily complimented the Task Force on this report and recommended that it be given wider distribution.

The report stimulated much discussion. It was generally felt that practical application of this technique still has to be effectively demonstrated in order to generate wider support of the Utilities.
Currently the investigations are normally conducted by outside specialists and also specialized equipment is required. The main issue now concerns how the interest of utilities can be stimulated, especially as the level of research in the national laboratories is low. The possibility of using the IAEA's Coordinated Research Program as a means to stimulate interest was raised. Three actions (8, 9, 10 and 11) were assigned to enable the state-of-the-art report to be published as an NEA document.

A meeting of utilities was held in the USA concerning loose parts monitoring. There was concern that use of the technique might result in premature reactor shutdown. Hitachi have also made various measurements and should be encouraged to publish the results. An action (12) was placed on members concerning the archiving of such information by the NEA Data Bank.

4.6 Reactivity Scale and Central Worth Benchmark

There has been no further action due to the departure of the key person from ANL.

4.7 Treatment of the Unresolved Resonance Region Intercomparison

A paper (A-698) on this topic had been previously distributed. A specialists meeting was held in December 1984 where it was agreed that the benchmark should be restricted to an algorithm that treats the unresolved region. It was also recommended that the resolved region be extended to higher energies.

4.8 Intercomparison of Reaction Rate Techniques

The IRMA intercomparison of reaction rate techniques has been documented as A-750. Eight teams from seven laboratories took part in this exercise which used the MASURCA reactor. The situation in general was fairly good. Some further recalibration is being undertaken and there will be a final meeting in March 1986. The participants felt that it was a very successful exercise and shown the value of such intercomparisons.

4.9 Local Heterogeneous Effects due to Geometry

The CADENZA experiments, relating to heterogeneous effects in plates, are already in benchmark form. However, although there is much experimental information, analysis to date has not led to a clear explanation of the differences.

4.10 Archiving of Benchmark Data

It was generally agreed that it would be beneficial to have benchmark data stored by the NEA Data Bank in a form that would allow easy use (see Action 13).
4.11 PWR U-236 Benchmark

A draft of the final report will shortly be sent to the participants and it is anticipated that the final report will be issued in early 1986 (see Action 1).

5. General

5.1 Highlights of Recent Meetings of Interest to NEACRP


A summary (A-698) of this meeting had previously been distributed. This item was also addressed under item 4.7.

APS Nuclear Data Meeting, Santa Fe, USA, May 1985.

Aspects of interest to NEACRP were discussed where appropriate in the agenda.

5.2 Future Meetings of Interest to NEACRP

ANS Topical Meeting on Advances in Reactor Physics and Safety, Saratoga Springs, USA, September 1986.


5.3 Other Business

In response in Action 9 of the previous meeting, Dr. C.J. Allan of CRNL has been in contact with several people to assess potential interest in a follow-up meeting to the 1983 Specialists' Meeting in In-Core Instrumentation. There seems to be sufficient interest for a meeting to be held in late 1987/early 1988. A final decision will be made at the next NEACRP meeting. (See also Action 14.)
ANNEX 1

LIST OF PARTICIPANTS

Delegates

For Canada

Mr. P.M. Garvey

Scientific Secretary

For Japan

Mr. S. Matsuura
Dr. K. Shirakata

For the USA

Dr. P.B. Hemmig
Dr. L.G. LeSage

Mr. C.E. Whitesides

For the countries of the European communities and the European Commission acting together.

Dr. M. Rief
Dr. M. Salvatores
Dr. C. Golinelli
Dr. H. Kuesters
Dr. R. Martinelli
Mr. R.J. Heijboer
Dr. J.R. Askew
Mr. J.M. Stevenson

(CEC)
(France)
(France)
(F.R. of Germany)
(Italy)
(Netherlands)
(United Kingdom)
(United Kingdom)

For the other European countries of the OECD

Dr. M. Rief
Mr. K. Jirlow
Dr. P. Wydler

(Spain)
(Sweden)
(Switzerland)

Nuclear Energy Agency

Mr. J. Rosen
Dr. L.G. de Viedma

Secretariat

Dr. P. Nagel

Observers (all sessions)

Dr. E. Fort
Mr. F.A. O'Hara
Dr. A. Abramov
Dr. Yu. Kazansky

(NEANDC)
(IAEA Secretariat)
(IAEA)
(IAEA)

Apologies for absence were received from Dr. McCullock (Australia). Following an established rotation Mr. Jirlow (Sweden) also represented Denmark, Finland and Norway. The delegates for Spain and Switzerland also represented Portugal and Austria, respectively.
ANNEX 2

NEACRP DOCUMENTS PRESENTED AT THE 28th MEETING

"L" DOCUMENTS

L-285 National Reports

L-286 C.H.M. Broeders
Validation of Calculational Procedures for the Design of
Light Water Tight Lattice Reactors (LWTLR) with Epithermal
Spectrum.

L-287 S. Cierjacks
New Intense Neutron Sources and Related Nuclear Data
Needs.

L-288 W. Kley and G.R. Bishop
The JRC Proposal for a European Fusion Reactor Materials
Test and Development Facility.

L-290 An FRG-Proposal for an International Intercomparison of
Codes for Radiation Protection Assessment of
Transportation Packages.

L-289 R.D. Bachle
Adjustment Results of Iron Cross Sections on the Basis of
EURACOS and ASPIS Integral Measurements.

L-291 S. Brandes
Core Physics Tests of THTR Pebble Bed Core at Zero Power.
P. Bernard, D. Fry, D. Stegemann, and H. van Dam
State-of-the-Art on Reactor Noise Analysis

NEA Data Bank Activity Report, November 85.

A Glance at NEA's Programme of work in the Nuclear Safety Area.


O. Norinder
A Method for the Determination of Albedos for Nuclear Reactor Codes.

T. Fujimura, Y. Nakahara and M. Obara
Three-Dimensional Multi-Group Neutron Transport Calculations by Double Finite Element Method.

T. Suzuki, A. Hasegawa and T. Ise

T. Ida, Y. Oka, S. Kondo and Y. Togo

M. Bando, T. Yamamoto, Y. Saito and T. Takeda

Y. Saito, M. Bando and T. Takeda

T. Takeda, Y. Sasaki and Y. Saito
Three-Dimensional Transport Correction in Fast Reactor Core Analysis.

J.K. Fletcher
Recent Developments of the Transport Theory Code MARC/PN.

C.J. Cho, G. Palmiotti, J.M. Rieunier and M. Salvatores
BISTRO. A Fast Two-Dimension $S_N$ Transport Code for $k_{eff}$ Calculations.

S.C. Carpenter, S.B. Brumbach and R.W. Goin
Spatial Kinetics Effects on Reactivity Measurements in ZPPR.
Yu.A. Kazansky, V.A. Lititsky, I.P. Matveenko and A.G. Shokodko
Some Methods of Taking Into Account Spatial Effects of Reactivity Measurement.

\(^{252}\)Cf Source-Driven Neutron Noise Analysis Method
J.T. Mihalczo, W.T. King and E.D. Blakeman.

J. Cray
Control Cluster Efficiency Measurements in PWRs by Analysing Power Signals During a Rod Drop.

J.P. West and J.C. Gauthier
Some Evidence of Space Kinetics Effects in Rod-Drop Experiments at MASURCA.

R.W. Schaefer
Local Heterogeneity Effects on Small Sample Worths.

S. Iijima et al.
Experimental Study of Large Scale Axially Heterogeneous LMFBR Core at FCA Assembly XII-1.

G. Palmiotti and D. Riou
A Method to Take into Account Control-Rod Heterogeneity Effects.

K.F. Bennett and R.W. Schaefer
Effective Beta Measurements on Uranium and Plutonium Fast Reactor Mockups.

R.M. Lell
Beam Port Detector Response Calculations for the R2 Reactor.

P.J. Collins
The State of CADENZA and ZPPR-12 Analysis.

A.F. Course
Prediction of K-Effective and Leakage for DIMPLE Assembly SO3/20/A Using the Monte Carlo Computer Code MONK.

R.J. Brissenden and N.R. Smith
Further Studies of the Prescott Interaction Problem using Superstage Powering in MONK6.4.

R. Chawla
A Review of Lattice Calculations for the PROTEUS-LWHCR Phase I Experiments.

Y. Ishiguro and K. Kaneko
A Generalized Dancoff Factor for Applications to HCFWR in Complex Lattice Arrangement.
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Characteristics of Plutonium Utilization in ATR.

Plutonium Recycling in the French PWRs.

T. Ikegami and N. Mizoo
Burnup Characteristics of JOYO Mk-11 Core.

D. J. Lord
The Reactivity History of PFR during Period 1975 to 1984.

N. Karouby-Cohen
Uncertainties and Model Improvements for the Definition of a Lumped Fission Product for LMFBR Design Calculations.

S. G. Carpenter
Status of Reaction Rate Calibrations at ZPPR.

W. Scholtyssek and G. Granget
Intercomparison of Reaction Rate Measurements Techniques in MASURCA.

R. M. Westfall et al.
TMI-2 Criticality Studies: Lower-Vessel Rubble and Analytical Benchmarking.