

NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD OF THE THIRTIETH MEETING (TECHNICAL SESSIONS)

HELSINKI, FINLAND
14-18 September 1987

Compiled by
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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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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 five 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 by the other members of the committee.

1. New Topics

1.1 Recent Results from Operating Reactors

Rapporteur: F.N. McDonnell
Paper L-297

In the FRG section a comparison between theory and experiment, performed at KfK, for the activation of PWR-element and end-pieces showed a fairly good agreement. In addition the theoretical assessment at KfK of Cl¹⁴ formation in a PWR fuel pin was found to be in satisfactory agreement with experiments. Recent work by KWU has confirmed that the insertion of MOX and ERU in both PWR and BWR is feasible on an industrial scale.

Thermal hydraulic tests of 9 x 9 BWR fuel assemblies showed that the design meets the demands imposed on advanced BWR fuel (KWU). A BWR Fuel Channel Model with low calculation time and memory requirements, called FLOT, was described (KWU).

Monte Carlo calculations for the activation rates of incore detectors of the KWO (PWR) reactor agree with measurements to between 0 and 5% (IKE-Stuttgart). Application and use of JEF-1 Data for description of the burn-up behaviour of PWR fuel was also described by KfK.

The French section described recent results from SUPERPHENIX which show that the calculated mass of the first criticality core was in excellent agreement with experiment. This required the use of sophisticated transport methods to account for streaming and heterogeneity effects.

The observed 10% discrepancy in control rod worth is partially attributed to basic data uncertainties, besides control rod heterogeneity effects. This points out the difficulty of extrapolating the E/C values, observed in critical assemblies, to power reactors, and the need to use a parametrical approach to critical experiments. Shielding experiment results show excellent agreement for the E/C value of the secondary sodium activation.

In the Canadian section recent experience at Ontario Hydro operating CANDU stations was described. CANDU reactors have two independent shutdown systems. The first consists of solid rods while the second consists of a poison injection system. A satisfactory simulation model was developed to analyse this second system.

Typical load cycling in CANDU stations will consist of reducing power to 50% in 30 minutes, holding at that level for a few hours, then returning to full power over a period of six hours. A simulation was carried out for the CANDU-600 and for actual load-following operations performed at Ontario Hydro stations.

1.2 Calculations and Measurements of Void Coefficients on Thermal and Epithermal Lattices

Rapporteur: M.J. Halsall
Papers A-825 to A-831

This session was devoted almost exclusively to papers arising from the Chernobyl accident.

Paper A-825 (Japan) described a series of calculations made to compare PNC methods with results for Chernobyl published by the USSR. The methods used were embodied in a comprehensive scheme of codes covering neutronics and thermal hydraulics. WIMS-ATR, for the lattice calculations, used data adjusted to give good agreement with Fugen measurements. Parameters calculated included reactivity coefficients, isotopics and power distribution, and generally agreed well with the USSR data.

Paper A-826 (Japan) followed A-825 in describing calculations of possible improvements to the Chernobyl operating procedures. Rods initially inserted 1.2 metres into the core begin to bite more rapidly, and 80 such rods inserted at 0.4 m/sec would have been sufficient to prevent a serious excursion. The insertion of 80 rods improved the positive void coefficient to a value that was demonstrated to be tolerable.

Paper A-827 (Japan) described a series of cell calculations made by JAERI. Separate cell calculations were made for fuel and absorber to get data for a 16-channel supercell. From this model were derived discharged fuel compositions and reactivity coefficients in good agreement with those published elsewhere.

Paper A-828 (Italy) examined the implications of Chernobyl for the safety of Cirene (40 MWe HW/BLW pressure tube reactor). Lattices of interest were mocked up in RB-3, a D_2O moderated, D_2O -graphite reflected critical facility. The void reactivity effects obtained by changing tie-rod materials, enrichment, and simulated coolant density were measured by critical height adjustments. Calculations of the effects were accurate to within about 3%.

Paper A-829 (CEC) described KENO-EUR, a version of KENO-IV for perturbation calculations. Among other schemes, algorithms had been developed using correlated tracking techniques for representing the removal of absorbers, and independently, the removal of scattering materials. (Perturbation techniques are being developed independently at Winfrith by Brissenden and Hutton.)

Paper A-830 (Canada) described an assessment of lattice characteristics affecting void reactivity in CANDU lattices using the Chalk River version of WIMS. These studies considered changing V_m/V_f ratio, the number of fuel pins, the enrichment and introducing coolant displacers. Of these the favoured option for further study was the displacement of some of the inner fuel (probably the inner 7 of 37 elements) by a graphite cylinder. This would reduce the void effect at some small penalty in achievable burnup. WIMS plus 3D calculations had been shown to agree well with reactivity and reaction rate measurements made in ZED-2.

Paper A-831 (Canada) emphasised the importance of spatial effects in the analysis of Chernobyl. An axial flux distribution, deduced from information on spatial burnup distribution, showed a large peak near the bottom of the core. Insertion of the graphite follower at the bottom of the control rods served only to displace water from a region of high importance, and contribute positively to the reactivity transient. It was noted that calculations subsequently made in the UK substantiated the conclusion that the rather slow insertion of totally withdrawn rods had exacerbated the transient.

There had been some speculation about whether the point kinetics model would be adequate for modelling the Chernobyl power transient. However, the importance of detailed flux and control rod modelling indicated in A-831 suggested that more spatial detail was necessary.

1.3 Uncertainties in Reactivity Feedback Coefficients in Fast Reactors

Rapporteur: P.B. Hemmig
Paper A-832 to A-835

Paper A-832 (UK) presented results of several reactivity feedback measurements on the PFR from 1974 to 1985. Attempts to separate the individual components of feedback are discussed. Generally good agreement is obtained between measurements and predictions. Changes in the power coefficient of reactivity feedback were larger than expected and these were attributed to changes in the fuel to coolant heat transfer coefficient and changes in the mode of fuel expansion with power level.

Paper A-833 (France) discusses the measurement of reactivity feedback coefficients for Super-Phenix and the main sources of uncertainties in these measurements. Good agreement between mean theoretical and experimental values are noted, except for the core temperature rise coefficient. Studies are continuing to explain this discrepancy. It is concluded that calculations used in the safety analysis of Super-Phenix are essentially validated.

Paper A-834 (USA) presented a simplified analysis of uncertainty propagation in inherently controlled ATW events. Using the natural grouping of effects related to fuel temperature and coolant temperature, expressions were derived for loss of flow and loss of heat sink conditions, which show strong attenuation of error propagation. It is noted that partial error cancellations occur when the same phenomena, such as Doppler, contribute to positive reactivity on power reduction and negative reactivity on increasing core temperature.

Paper A-835 (USSR) discusses the main factors which determine uncertainties in the reactivity coefficients of fast reactors and the difficulty of developing accurate reactivity feedback models. Measurements in BN-350 and BN-600 are discussed. Several changes related to operating conditions are noted. These are generally predicted with acceptable accuracy. The limitations of passive safety effects for BN-350 and BN-600 are discussed, as well as the conditions necessary to assure passive safety in larger oxide fuelled reactors.

1.4 Reactivity Effects of Fuel Fragmentation in Light Water Cooled Reactors

Rapporteur: M. Rajamäki
Paper A-836

One paper was contributed from Finland for this topic. The paper described the reactivity effects caused by geometric fragmentation of fuel and by simultaneous cooling of fragments. A series of LWR cases and three speculative scenarios for the Chernobyl accident are considered. Calculations are carried out with the LWR cell burn-up code CASMO-HEX.

Fragmentation is described by the increasing number of the equal fuel pieces with decreasing diameter and cooling is considered to occur as quasi-stationary. Relative movement of the fragments and the coolant is taken into account by varying the fuel coolant ratio. Although the analysed cases include some hypothetical features the multiplication factor receives such large increases, up to 10 \$, that more realistic and more detailed studies concerning this item are justified.

1.5 Data Testing in the USSR

Rapporteur: I. Slessarev
Paper A-838

Paper A-838 (USSR) describes the results of Monte Carlo calculations (MCU-code, USSR) for critical assemblies with the following compositions: (U235, U238, H₂O); (U235, U238, D₂O); (Pu239, U, H₂O); (U233, Th232, H₂O). Most of them are acknowledged as benchmarks. The following data sources were used:

- U233, U235 - ENDF/B-V library for thermal neutrons,
- U238, Pu239, Pu240, Th232 - USSR files.

The group constants and subgroup parameters for all isotopes were taken from ABBN library set.

In the case of the heterogeneous assemblies with H₂O the calculation underestimates k_{∞} by about 0.5%. This results from the group approximation in the energy range above the U238 fission threshold.

The calculated data for the MIT assemblies using the MCU-code with ENDF/B-V constants differ strongly from the experimental results. The authors are prone to think that the experiment is the reason here, not the calculations.

The analysis of the results obtained in the calculations carried out has shown that the level of knowledge about the neutron constants of U233, U235, U238 and Th232 isotopes satisfies principally the requirements for the accuracy of calculations of present day thermal power reactors. The cross-section of Pu239 requires refining in the range of resonance energies including the first resonance.

Mr. Hemmig noted that there was variation between data sets for Pu239 and, in the light of new experimental data, a new evaluation is being carried out for ENDF/B-VI.

2. Topics Carried Over from Previous Meetings

2.1 Integral Validation of Recent Delayed Neutron Data

Rapporteur: M. Salvatores
Papers A-840 and A-841

Paper A-840 (France) summarises the results of the analysis of several integral β_{eff} measurements, performed at ANL and SNEAK in fast critical facilities. The analysis indicated that somewhat high delayed neutron data for both U238 and Pu239 (higher than the Tuttle data) are needed to get calculation/experiment agreement in U238/Pu and U238/U235 systems. However, in the case of systems with a high content of iron, the C/E is greater than one, indicating possible effects of iron cross-section uncertainties on the adjoint flux shape. Finally, it was noted that one of the most representative experiments (SNEAK Assembly 9C2), did show a C/E ≈ 0.93 , inconsistent with most of the results for the other assemblies. It was noted that for this experiment the error bars were fairly significant. A new measurement in a large, Pu-fuelled core, is planned at MASURCA.

Paper A-841 (USSR) indicated a good agreement between a summation calculation of the total yield of delayed neutrons from individual contributors, and global macroscopic measurements. The summation calculation results have been used to derive 6-group delayed neutron constants. The group constants for U235, were compared to other group constants, but when used in experiment analysis, no definite evidence was found in favour of one particular set.

As a comment to the relatively large spread of C/E observed in integral β_{eff} experiments, Prof. Condé remarked that the NEANDC recently dropped the delayed neutron data item from the list of items which should be addressed with priority by that committee, in view of the results of the Birmingham specialists' meeting. After discussion, to resolve the potential inconsistency between integral and differential data in this field, it was suggested that Prof. Rudstam, who will summarise the conclusions of the Birmingham meeting, should include in his summary the best recommended delayed neutron yields and spectra. These data will be used to reanalyse the integral β_{eff} measurements. The topic will be then carried over to the next meeting.

2.2 Validation of Fission Product Data (in particular for Thermal Reactors)

Rapporteur: M. Darrouzet
Papers A-837, A-847 and A-848

Paper A-847 (France) concerns the studies of fission products in light water reactors. A new multigroup fission product cross-section set for the APOLLO code, has been generated. Data are issued from the JEF-1 evaluation. On the other hand an important program to validate the total

effect of fission products is in progress. The measurements of PWR spent fuel samples and test sample reactivities are obtained by oscillation in several lattices (with UO_2 or UO_2/PuO_2 fuel rods).

The two other papers concern fission products in fast breeder reactors.

In the French paper (A-848), the effects of the migration of gaseous fission product (Br, Xe, I and Cs) must be noted (this effect is important for the calculation of the reactivity loss during a cycle in the large core) in the same way as the work to validate the JEF cross-section of each important fission product, in particular on Pd106 and Sm151. A new pseudo fission product has been produced and will be used together with the new migration model, in the analysis of reactivity loss/day in Super-Phenix.

The Russian paper (A-837) presents the reactivity measurements of Rh103 and Pd105 samples in BFS 49-4 and 49-2. The results are compared with the measurements of the same isotopes performed in STEK 500 and STEK 1000 and with calculated values. Good agreement is obtained but it is important to note the problem of extrapolation to a sample of "zero" thickness.

2.3 Physics Aspects of Design Innovation to Increase Inherent Safety for Fast and Thermal Reactors

Rapporteur: L.G. LeSage
Papers A-839 and A-852 to A-856

Paper A-852 (Japan) describes design studies of a 1000 MWe loop-type LMFBR with mixed oxide fuel. The objective of the studies was to assess the inherent safety performance of the design under ATWS conditions (specifically the unprotected loss-of-flow accident). The studies indicated that the sodium temperature could be maintained below boiling if certain conditions were met. These included an initial control rod insertion of 0.2m, an internal chimney design to enhance control rod drive line expansion, and a pony motor with a capacity of 20% of rated flow to maintain flow during the transient.

Studies in the USSR of a new sodium cooled fast reactor concept with in-assembly heterogeneity are described in paper A-839. The core (designated the IAH-type) consists of a regular lattice of mixed oxide and fertile metal fuel pins in each subassembly. The ratio of fissile mixed oxide pins to fertile metal pins is 2:1. The advantages of the design include good breeding properties (due to high fertile density in the core), the possibility of a small reactivity swing with burnup and the consequent TOP related safety advantages, an increased refuelling interval, a lower than expected difference in power difference between oxide and metal fuel pins due to the high absorption of gamma energy in the metal pins, and lower swelling of structural materials due to decreased neutron fluence. A number of other possible advantages of the IAH concept as well as the possible problems of reprocessing the IAH subassemblies were discussed.

In paper A-856 (USSR) a method of determining reactor parameters in order to optimize reactivity effects for safe and efficient performance is described. The program utilises a three dimensional multigroup diffusion

calculation of the reactor system, and is based on a direct variational method. Coefficients of sensitivity used in the method are obtained with the aid of generalised perturbation theory.

In the US the fast reactor program has emphasised passively safe designs and the current focus of the US design effort is on relatively small (100 to 400 MWe) plants. The achievement of passive or inherent safety features is relatively easier in smaller plants. Paper A-853 (US) examined the trends in these inherent safety features vs. plant size in order to determine if the features could be retained in the larger size LMFBR's. A simplified quasi-static reactivity balance approach was used in the analysis. It was concluded that the inherent safety features were retained with only small degradation in large metal fuelled LMFBR's. Because of the higher fuel temperature in large oxide LMFBR's it is more difficult to retain the inherent safety features and design changes such as fuel derating may be required.

Paper A-854 (US) describes a simplified mechanical structural model of core subassembly bowing for a limited free bow core restraint design. The solution is for a single core subassembly at the core blanket interface and an analytical solution is obtained. The total radial core movement contains components due to grid plate temperature increase, temperature rise across the core (which is directly proportional to the radial temperature gradient across a subassembly), and core restraining ring temperature increase. The primary value of the model is as a design tool in showing the functional dependence of core radial dilation and lockup on the key core mechanical design parameters such as load pod stiffness and height of the core above the grid plate.

Paper A-855 (US) presents a new simplified design method for calculating core radial bowing reactivity feedback changes. The current standard design method, called the triangular homogenization scheme (THS) is shown to have a bias which tends to under predict the reactivity values. The new method, called the corrected triangular homogenization scheme (CTHS) makes a linear correction for this bias. The two methods were validated by comparison with more accurate 2D and 3D calculations in which the geometry was explicitly represented. The CTHS method was shown to be in much better agreement with the benchmark problem. Additional investigations were made of the approximations in the benchmark problems. Changes in subassembly gap streaming were shown to contribute 5-10% to core radial expansion, while the other effects considered were relatively smaller.

2.4 Fusion Blanket Experiments

Comparison of Measurement and Calculations

Rapporteur: Y. Kaneko
Papers A-857 to A-860, L-297

Paper A-857 (Japan) discusses a new series of neutronics integral experiments which have started at FNS as the second phase of the fusion blanket engineering benchmark experiment program which is a collaborative activity between JAERI and USDOE. A closed-geometry configuration is

adopted, in which the 14 MeV neutron source point and the blanket test region are surrounded by a spectrum-matching enclosure. The blanket test region is now a single-zoned Li_2O breeder, to which will be added a beryllium neutron multiplying layer. Measurements of the tritium production profile and other neutronic parameters have continued.

Measurement of the detailed spatial variation of the low energy neutron spectra within the blanket assembly of the closed geometry is one of the most important items of the collaborative program between JAERI and USDOE. This measurement requires that the detector and electronics package is compatible with a small access hole, and that the total amount of material is held to a low value to minimise perturbation to the neutron spectrum. Therefore, a small proton recoil counter unit has been developed for this purpose as described in Paper A-860 (US).

A cross-section sensitivity-uncertainty analysis code, SUSD was developed. The code calculates sensitivity coefficients for one and two dimensional transport problems based on the first order perturbation theory. The uncertainties of tritium breeding ratio, fast neutron leakage flux and neutron heating were analysed on various types of blanket concepts - see Paper A-858 (Japan). Two design concepts are studied: a helium cooled ceramic blanket and a blanket with Pb-Li eutectic as breeder material and coolant. The potential and problems of an aqueous salt solution are also discussed.

A general transport code GANTRAS, making full use of double-differential cross-sections, is under development. The one-dimensional module, ANTRA I, had been documented. This code has been used for a comparative study of the neutron multiplication of a 14 MeV neutron source in a spherical lead assembly, using the EFF-1 file. ANTRA I and the Monte Carlo Code MCNP agree very well on the neutron multiplication in lead. Comparison with experimental results from Takahashi did indicate that the measurements may be too high, also compared to the experiment by Hansen (L-297, Federal Republic of Germany contribution).

Measurements of tritium production rates (TPR) have been conducted in integral neutronic experiments on model fusion blankets at several organisations. In these experiments, it is required to determine TPRs in absolute values. An examination on the accuracy of TPR measurement on unified basis is very important. From this view point, a benchmark experiment can be justified. The initial proposal expressed in Paper A-859 (Japan) was discussed, and the following plan was adopted.

1. The benchmark experiment will comprise two stages. The first will be the comparison of the tritium production rate measuring technique using Li-containing samples which have been irradiated in a reference assembly. The second will include comparison between other experimental methods as well as with respect to the intensity determination of the 14 MeV neutrons injected into the reference assembly.
2. The initial plan will be revised through the discussions between JAERI and LOTUS groups. Other candidates for participation are also welcome to comment on the revised plan.

3. The revised plan should be submitted to the next NEACRP meeting, and then, the irradiation experiment will be carried out.
4. The final report should be submitted to the 1989 NEACRP meeting.

2.5 Reactor Physics Issues Related to Intermediate Spectra Reactors
(experiments, burn-up related problems, eventual design features)

Rapporteur: H. Klüsters

Papers A-842 to A-846, A-849 to A-851, A-861 and L-297

The preliminary results of the NEACRP-HCLWR benchmark were presented in Paper A-849 (Japan).

Sixteen solutions had been submitted from thirteen organisations. At present, the following conclusions can be drawn:

- A relatively large discrepancy is found in k_{∞} . The discrepancies are 3~5% with burn-up in the 0% voided case, while the largest discrepancy is 8% in the voided cell.
- The discrepancy of conversion ratio is larger than that for k_{∞} , and the largest discrepancy is 10%.
- The k_{∞} discrepancy is mainly caused by the difference between Pu239 production rates.
- The conversion ratio is influenced by the discrepancy between the reaction rates of both fertile (U238 and Pu240) and fissile (especially Pu239).
- Discrepancies are also found in the reaction rates of Pu241 and Pu242. For Pu241, treatment of resonance energy region seems to be important. The self-shielding effect of the 2.67 eV resonance of Pu242 is considered to be the cause of the difference in the absorption rate. A calculation by the SRAC system shows that this shielding effect results in more than 1% $\Delta k/k$.
- The major variation in burn-up reactivity loss is strongly influenced by the difference in Pu241 reaction rates, though those for Pu239 and fission products in all are also of some importance.
- Neutron absorption by Pu242 plays an important role in the void reactivity change. The shielding effect of Pu242 contributes at least 1% $\Delta k/k$ to the void reactivity. The production rate of Pu239 affects greatly the behaviour of void reactivity especially in the higher voidage state. The difference in void reactivity is caused, in some cases, by variations in U238, Pu240 and Pu241 reaction rates. At higher burnup stages the absorption rate of all fission products is also important for the void reactivity discrepancy.

Some revisions from various contributors are underway. The deadline for the corrected results is 5 January 1988. The Committee recommended that a specialists' meeting should be held in April 1988 at the Data Bank in Paris.

Papers A-850 (France) and A-851 (Switzerland) were only briefly discussed. They contain additional submissions to the HCLWR benchmark.

In A-844 (FRG) some investigations on the nuclear data set used for HCLWR reactors are described. Introduction of JEF-1 group data into the KEDAK-4 group set (isotope by isotope) showed especially the effect of the U238 and Pu239 capture data on the results for k_{∞} and the conversion ratio. KEDAK has still too high capture data for U238 compared with the more recent evaluations of JEF. Furthermore, a study was performed on the influence of the weighting spectrum on the results, especially for the voided case, by using a PWR and a HCLWR weighting spectrum. The results show that differences up to 15% occur in the void reactivity for these spectra; the preparation of a group set with a weighting spectrum for the voided case is underway. Finally, JEF was applied to the experimental results of the PROTEUS experiment case 7. In using isotope-dependent fission spectra, both KEDAK-4 and JEF-1 show excellent performance for k_{∞} , KEDAK-4 overestimates C_8/F_9 by 3%, JEF underestimates this ratio by 4%. F_5/F_9 , F_8/F_9 and F_1/F_9 are satisfactorily reproduced by JEF.

Paper A-842 (France) describes the experimental program for the qualification of undermoderated Pu/H₂O lattices in EOLE, MINERVE and MELUSINE. Theoretical results, obtained with APOLLO and the new CEA-86 cross-section library, composed from cross-sections out of various nuclear data libraries, are very satisfactory.

In Paper A-843 (Japan) some results of the analyses for a HCLWR in FCA are reported. The calculated k_{eff} and k_{∞} are in very good agreement with experiment. The measured moderator voidage reactivity effect was also well predicted by theory.

An uncertainty evaluation for the coolant void worth in HCLWR is described in A-845 (Japan). It was found that especially that the present uncertainties in nuclear data for the unresolved resonance region in U238 gives the main effect.

In A-846 (Switzerland) comparisons of calculated and measured neutron balance components are reported for the 7.5% fissile-Pu reference test lattice of the PROTEUS-LWHCR Phase II programme, both wet (with H₂O) and dry (100% void). Special experimental techniques have been developed and applied, particularly for k_{∞} , and the range of directly measured reaction rate ratios has been extended. For the two cell codes tested, i.e. WIMS-D/1981 library and KARBUS/KEDAK-4, specific shortcomings have been identified - the new measurements being found to be significantly more representative and accurate than the earlier Phase I experiments.

The k_{∞} void coefficient for the Phase II reference lattice between 0 and 100% void has been found to be qualitatively different from those assessed for the earlier Phase I test lattice (it is positive). Consideration of the individual void coefficient components show this to be largely a consequence of the more LWHCR-representative fuel rod diameter and

plutonium isotopic composition of the fuel currently being used. Results of control rod studies conducted for the Phase II reference lattice - both wet and dry - serve to illustrate the efforts being made towards the investigations of special power reactor features.

Paper A-861 (USSR) describes the calculation of a detailed neutron space-energy distribution in intermediate reactors. The code SPEKTR can describe the spatial and resonance heterogeneity in tight lattice cells, also for fast reactor application. By using a subgroup treatment, an accuracy corresponding to an S_{40} to S_{60} approximation is obtained. For the FCA-XIV assembly, good agreement is obtained for criticality, and for the breeding ratio SPEKTR compares favourably with the Japanese SRAC-solution.

Finally, some comments on the design characteristics of large HCLWRs were taken from the FRG contribution to Paper L-297. KWU outlines the strategy to improve present days Convoy-PWRs by tightening the lattice. As a yardstick, KWU is primarily interested in a very tight lattice to study the physics and design performance. At KfK theoretical fluid dynamic studies for a large tight lattice reactor showed that the homogeneous and heterogeneous designs are almost comparable. Both physics and fluid dynamic investigations indicate that the tight lattice should be widened to guarantee a negative void reactivity for all burn-up stages and to guarantee the desired reactor power.

3. National Programmes

Reports on the reactor physics activities in the NEA member countries were summarised 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-297) to be issued by the NEA Secretariat.

4. Benchmarks

4.1 Shielding Benchmarks

Dr Salvatores reported on the Specialists' Meeting held at the NEADB in October 1986. The experimental benchmarks, which had been collected and discussed at a previous meeting, were used to validate basic data, in particular the new JEF-1 file. The conclusions are described in the summary (Paper A-823). Major points relate to the need of some revision of JEF-1 iron data at high energy, and to the overall good performance of JEF-1 data for sodium. More information is expected on water and carbon data.

During the meeting there was discussion on the possibility of a Task Force to help the setting up of a databank of the shielding benchmarks (specifications, experimental data, modelling options etc), together with appropriate calculational routes). This Task Force was set up as described in the section on the NEA Data Bank activities.

4.2 Criticality of Fuel Undergoing Dissolution

Dr Hemmig reported preliminary results of the Criticality Working Group studies of fissile material in solid form surrounded by fissile material

in solution, as discussed in Paper A-863. The need for these analyses arises in connection with transport accidents, reactor core melt accidents and in the reprocessing of fuel elements by acid dissolutions. There are no measurements for these types of systems; however 18 critical systems with similar characteristics were chosen for calculation. Benchmark 13, for example, included 451 UO_2 rods enriched to 4.3wt% U235 in 2.28cm triangular pitch lattices. The rods were immersed in uranyl nitrate solutions with 0.148gm Gd/litre. Benchmarks 19, 20 and 21 are hypothetical assemblies of 2.5% enriched UO_2 spherical pellets in borated water or borated water slurries. The volume fraction of the pellet to the slurry, the boron concentration and the fraction of the total UO_2 in the slurry are varied.

A wide divergence of results were obtained for k_∞ and k_{eff} solutions for the measured benchmarks. Results were generally divided into two groups, those that used the SCALE system nuclear data and those that used other data libraries. Benchmark 20 results indicated spreads of 20 to 25% in k_∞ and variations up to 16% in k_{eff} . No firm conclusions have been reached regarding reasons for these discrepancies. It was suggested that an MCNP analysis might help clarify the effects of resonance region cross-section treatments. Dr. Küsters accepted an action to try and initiate such a calculation at KfK.

4.3 Heat Transfer in Transport Flasks

Mr. Glass presented a draft version of the Final Report on this Benchmark (Paper L-299). A specialists group was convened in May 1985 to define a standard thermal problem set which could be used to evaluate codes used in cask analysis and design. The six chosen problems cover the relevant areas of cask thermal response from fuel assembly simulation through cooling media phase change to fin heat dissipation. The problems also address conditions arising during normal transport (steady state) and those that occur when the cask is exposed to a fire - like (transient) environment ($800^{\circ}C$ for 30 minutes).

The problems, working from fuel assembly response to global cask response involve,

- (i) a simulated horizontal fuel pin array in a gas environment during normal transport,
- (ii) fuel surrounded by sodium which undergoes phase change during a fire,
- (iii) thermal stratification and pressure build-up in a water-filled cask during a fire,
- (iv) a heat source with conduction through a monolithic cask wall and convective cooling at the surface during normal transport,
- (v) fin response during a fire,
- (vi) a multiple layered cask in a fire environment with a non-axially symmetric thermal shield.

One solution had experimental results and one had an exact analytical solution. The other four had neither.

Seven participants from five countries took part. Agreements with experiment, the analytical solution and between participants were within 10% for predicted temperatures in $^{\circ}\text{C}$. The final report should be issued before the end of 1987.

4.4 Shielding in Transport Casks

Dr. Nagel presented the status report (Paper A-864) following the second meeting of participants in May 1987. The first three problems (1a, 1b, 1c) have been examined in detail. They are simple cylindrical casks with a cast iron wall and bottom, and a steel lid, two with dry fuel and one with wet fuel, assumed to give homogeneous sources through the cavity. One of the casks with dry fuel had a polythene annulus around the cast iron flask.

Agreement between calculated surface dose rates from neutrons and gamma rays and attenuation factors had improved from up to factors of ten to factors of about two. Factors which had contributed to the improvements and may contribute to the remaining discrepancies are:

- a) errors in the interpretation of the specification of the problems
- b) mesh intervals
- c) multigroup averaging of data
- d) over-simplification of the geometry
- e) ray effects in the discrete ordinate codes.

A third meeting is proposed for February 1988 at which the results for problems 2 to 6 will be discussed.

Results from measurements on three flasks had been proposed as benchmarks. The NEACRP recommended that one should be chosen rather than more than one, thus avoiding a spread of solutions.

4.5 Noise Analysis

The completed analysis of the benchmark tests on artificial and actual anomaly noise data were to be presented at the SMORN V meeting in Munich in October 1987.

4.6 Reaction-Rate Comparisons in MASURCA (IRMA)

The meeting to discuss the final results had been delayed because of an inconsistency between ANL results for U238 capture when French or American foils were used. An exchange of foils is taking place to try and resolve the situation. Other results were generally in good agreement. The final meeting is now planned for Spring 1988.

4.7 The Pin and Plate CADENZA Assemblies

Mr. Stevenson presented the Final Report (Paper L-300) on these calculations. The best calculations from all eight solutions gave positive C-E discrepancies for the k-values of the mixed-oxide fuel pin cells relative to the plutonium metal fuel plate cells in the normal (sodium-containing) assemblies, varying from 0.0015 to 0.0082d ρ .

Extra calculations in simple geometries showed a variation between participants of 0.003d ρ from the difference in the homogenised cell compositions between the pin and plate cells. Significant variations were also found in the spatial cell heterogeneity effects and streaming for both cells, arising from nuclear data and data processing methods, and from the detailed methods which attempt to represent the 3D nature of the cells. The largest variation was for the spatial heterogeneity of the plate cell where the standard deviation about the mean for those models which have a 3D or psuedo 3D model was $\pm 0.0013d\rho$.

The mean reactivity discrepancy of 0.0047d ρ is just over three times the standard uncertainty of 0.0015d ρ . There may be systematic errors from cross-section data and attempts to represent the plate cell in 3D.

Some participants had analysed supplementary experiments. Analysis of the voided cores and element replacement confirmed the pin-plate discrepancies. Analysis of plate cells with mixed-oxide fuel rather than metal fuel (and ANL analysis of the ZPPR-12 assemblies) suggested that the pin-plate discrepancies are partly associated with the change in geometry and partly with the change of fuel type.

Analysis of reactivity effects from changing the plate cell heterogeneity were somewhat contradictory, maybe confirming that there are problems in the data and data handling methods for dealing with cells containing the plutonium metal plates.

While the reasons for the discrepancies are not fully understood, it is suggested that (1σ) uncertainties of $\pm 0.004d\rho$ for the plate cells with plutonium metal fuel and $\pm 0.002d\rho$ for plate cells with mixed-oxide or mixed-metal fuel and mixed-oxide pin cells should be assumed to arise from heterogeneity effects alone.

Mr. Stevenson added that, after publication of L-300, further information had been received from one participant and an addendum would be issued.

4.8 Calculation of Fission Product Data in a Thermal Reactor

This was an exercise to calculate the depletion of a homogeneous mixture of hydrogen and U235 (200:1) in order to compare contributions to reactivity (k-infinity) from fission products. Optional extensions were a reduced H/U235 ratio (20:1) and Pu239 instead of U235.

Ten submissions had been received and the results were summarised in A-866 and A-867. Many different data libraries had been used and the results showed a wide spread of k-infinity values. In addition, several submissions were incomplete.

Before an attempt is made at analysing the results, copies of the submissions so far received will be sent to the participants for checking and possible revision.

4.9 Calculation of Reactor Characteristics in HCLWR's

This benchmark had already been considered under the topic 2.5.

5. Future Meetings of Interest to the NEACRP

SMORN-V. Symposium on Reactor Noise. Munich F.R. of Germany. October 1987 (NEA/IAEA)

Advisory Group Meeting on Nuclear Data for Calculation of Reactivity Coefficients, IAEA, Vienna. December 1987 (IAEA/NDS)

International Conference in Nuclear Data for Science and Technology, Mito, Japan. May 1988 (JAERI, PNC, JNS)

Technical Committee Meeting on Reactor Fuel Burn-up Determination. Argentina. June 1988 (IAEA)

Specialists' Meeting on In-Core Instrumentation and Reactor Core Assessment. Cadarache, France. June 1988 (NEA)

International Shielding Conference. Bournemouth, UK. 12-16 September 1988 (NEACRP)

Reactor Physics Topical Meeting. Jackson Hole, USA. 18-21 September 1988 (ANS)

Specialists' Meeting on the Application of Critical Experiments and Operating Data to Core Design via Formal Methods of Cross-Section Data Adjustment (tentatively, Jackson Hole, USA. 22-23 September 1988) (ANL)

Specialists' Meeting. Methods for Reactor Physics Calculations of Control Rods in Fast Reactors. Winfrith, UK. December 1988 (IAEA)

Activation cross-sections for Fission and Fusion Applications. ANL, USA. Spring 1989 (NEANDC)

Advisory Group on Possible Modifications of Fuel Assemblies' Design and Materials to Improve Safety in Off-Normal and Accident Conditions (Undated, IAEA)

Man-Machine Interface in the Nuclear Industry (Undated, IAEA)

ANNEX 1

LIST OF PARTICIPANTS

Delegates

For Australia	Dr. D.B. McCulloch	
For Canada	Dr. F.N. McDonnell	
For Japan	Dr. K. Shirakata Dr. Y. Kaneko	<u>Vice Chairman</u>
For the USA	Dr. L.G. LeSage Dr. P.B. Hemmig	<u>Chairman</u>
For the countries of the European Communities and the European Commission acting together	Dr. H. Rief Dr. H. Klusters Dr. M. Darrouzet Dr. M. Salvatores Dr. R. Martinelli Prof. H. Van Dam Dr. M.J. Halsall Mr. J. M Stevenson	(CEC) F.R. of Germany (France) (France) (Italy) (Netherlands) (United Kingdom) (United Kingdom) <u>Scientific Secretary</u>
For the other European countries of the OECD	Dr. M. Rajamäki Dr. P. Wydler	(Finland) (Switzerland)
Nuclear Energy Agency	Dr. P. Nagel Dr. E. Sartori	<u>Secretariat</u> <u>Secretariat</u>
<u>Observers</u>	Prof. H. Condé Mr. M.J. Crijns Dr. I. Matveenko Dr. I. Slessarev Mr. R.E. Glass	NEANDC (IAEA Secretariat) (IAEA) (IAEA) (USA)

Apologies for absence were received from Dr. Caro (Spain) and Dr. Maienschein (USA). Following an established rotation, Dr. Rajamäki (Finland) also represented Denmark, Norway and Sweden. The delegates for the Netherlands and Switzerland also represented Belgium and Austria, respectively.

ANNEX 2

NEACRP DOCUMENTS PRESENTED AT THE 30TH MEETING

"L" Documents

- L-297 National Progress Reports
- L-298 M. Salvatores
Highlights of the JEF Project Activities (September 1986 -
September 1987)
- L-299 Standard Thermal Problem Set for the Evaluation of Heat Transfer
Codes used in the Assessment of Transport Packages
- L-300 M.J. Grimstone, J.L. Rowlands, J.M. Stevenson
Final Report on the International Comparison of Calculations for the
CADENZA Assemblies, The Pin Plate Benchmark

"A" Documents

- A-818 List of CSNI International Standard Problems
- A-819 Reproduction and Distribution of the Documents of the NEACRP
- A-820 Steering Committee for Nuclear Energy: Renewal of the Mandate of the
NEACRP
- A-821 NEA Data Bank Activity Report 1986-1987
- A-822 Report by the Secretariat on Recent Activities of the CSNI of
Interest to the Committee on Reactor Physics
- A-823 Summary of the NEACRP Specialist Meeting on Shielding Benchmarks
OECD, Paris 13th-14th October 1986
- A-824 Report on Relevant IAEA Activities to the NEACRP
- A-825 T. Wakabayashi et al.
Analysis of the Chernobyl Reactor Accident (I). Nuclear and Thermal
Hydraulic Characteristics and Follow-up Calculation of Accident
- A-826 T. Wakabayashi et al.
Analysis of the Chernobyl Reactor Accident (II). An Examination of
the Improvement Measures concerning the Accident of Chernobyl Power
Plant
- A-827 K. Tsuchihashi, F. Akino
An Analalysis of Reactivity Coefficients of the Chernobyl Reactor by
Cell Calculation
- A-828 A. Grossi, A. Vanossi
Measurements of Void Reactivity Effects in CIRENE Mock-up Fuel
Channels at RB-3

- A-829 H. Rief
A Guide to the Use of "KENO-EUR" A Code for Perturbation Analysis in Multiplying Systems
- A-830 J. Griffiths, P.G. Boczar, M.T. van Dyk
An Assessment of Lattice Characteristics that Influence Coolant Void Reactivity in CANDU-type Lattices
- A-831 P.S.W. Chan et al.
The Chernobyl Accident. Multidimensional Simulations to Identify the Role of Design and Operational Features of the RBMK-1000
- A-832 D.J. Lord et al.
Review of Prototype Fast Reactor (PFR) Reactivity Feedback Coefficient Measurements 1974-1987
- A-833 P. Bergeonneau, M. Salvatores, M. Vanier
Uncertainty Analysis on the Measurement and Calculation of Feedback Effects in LMFBR. Application of SUPER-PHENIX-1 Start-Up Experiments
- A-834 D.C. Wade
A Simplified Analysis of Uncertainty Propagation in Inherently Controlled ATWS Events
- A-835 I.A. Kuznetsov, I.P. Matveenko
Uncertainties in Coefficients of Reactivity of the BN-Type Fast Reactors
- A-836 F. Wasastjerna
On the Reactivity Effects of Nuclear Fuel Fragmentation with Reference to the Chernobyl Accident
- A-837 S.M. Bednyakov
Integral Experiment Analysis on Refinement of Rhodium-103 and Palladium-105 Capture Cross-Sections in Fast Critical Assemblies
- A-838 E.A. Gomin et al.
Analysis of Experiments on Critical Assemblies for Testing the Constant Support of Reactor Calculations
- A-839 V.V. Orlov et al.
The Concept of Fast Sodium Power Reactor with In-Assembly Heterogeneity
- A-840 A. D'Angelo, M. Salvatores
Comparison Between Calculated and Experimental Effective Beta Results on ZPPR and SNEAK Fast Facilities
- A-841 L.G. Manevich et al.
Calculation of Integrated Characteristics of Delayed Neutrons
- A-842 M. Darrouzet et al.
Neutronic Experimental Program in France for the Qualification of Unmoderated PWR Calculation

- A-843 T. Osugi
Results and Analyses for FCA Phase-1 Experiment on High Conversion Light Water Reactor
- A-844 C.H.M. Broeders, A. Mateeva, H. Küsters
Some Special Methodical and Nuclear Data Investigations for Tight Lattice PWR-Cores (HCLWR and PROTEUS)
- A-845 Y. Yamaguchi, T. Takeda
Uncertainty Evaluation of Void Reactivity Worth in High Conversion Light-Water Reactors
- A-846 R. Seiler et al.
Investigation of the Void Coefficient and Other Integral Parameters in the PROTEUS-LWHCR Phase II Programme
- A-847 L. Martin-Deidier, J. Krebs
Amélioration de la Connaissance de la Capture des Produits de Fission dans les Réacteurs à Eau
- A-848 L. Martin-Deidier, M. Salvatores
Recent Data and Method Improvements for the Pseudo-Fission Product Cross-sections Assessment in a LMFBR
- A-849 H. Akie, Y. Ishiguro, H. Takano
Preliminary Report of HCLWR Cell Burn-up Benchmark Calculations
- A-850 P. Chauchepirat
CEA Contribution to the HCLWR Burn-up Benchmark Proposed by NEACRP
- A-851 J. Stepanek and P. Vontobel
EIR Results for the HCLWR NEACRP Burn-up Benchmark Obtained using EIR Version of DANDE System and JEF Library
- A-852 K. Yamaguchi, S. Ohta, H. Endo
Inherent Safety Performance of a Mixed-Oxide-Fuelled 1000 MWe Loop-Type LMFBR under ATWS Conditions
- A-853 D.C. Wade, E.K. Fujita
Trends vs Reactor Size of Passive Reactivity Shutdown and Control Performance
- A-854 T.J. Moran
A Simplified Model of Core Thermal Dilation
- A-855 P.J. Finck
A Technique for Computing Bowing Reactivity Feedback in LMFBR's
- A-856 P.N. Alekseev, I.S. Slessarev, M.V. Trunov
Optimisation of Reactivity Effects in Fast Reactors
- A-857 T. Nakamura
Fusion Blanket Engineering Benchmark Experiments - JAERI/USDOE Collaborative Program Phase II

- A-858 K. Furuta, Y. Oka, S. Kondo
Error and Uncertainty Analysis of Neutron Transport Calculation
- A-859 Y. Kaneko, T. Nakamura, H. Maekawa
On the Proposal of International Comparison of the Measuring
Techniques for Tritium Production Rate Used for Fusion Neutronics
Experiment
- A-860 E.F. Bennett
Neutron Spectrum Measurements with Proton Recoil Counters at FNS
- A-861 A.G. Morozov, I.S. Slessarev
Simulation of Detailed Space - Energy Distribution of Neutrons in
Intermediate Reactors
- A-862 EURACOS Iron and Sodium Benchmark Analysis.
A Comparison of JEF-1 and BMCCSI Cross-Sections in Deep Penetration
Experiments
- A-863 G.E. Whitesides
Draft Report on Criticality Benchmarks (and memo)
- A-864 A. Avery
Summary of the Second Meeting on the NEACRP Intercomparison of Codes
for the Shielding Assessment of Transportation Packages
- A-865 S.G. Carpenter
Status of IRMA
- A-866 M.J. Halsall
Thermal Fission Product Benchmarks
- A-867 P. Chaucheprat
CEA Contribution to the Thermal Fission Product Benchmark Proposed by
NEACRP
- A-868 The Third Meeting on the Status of Reactor Physics in the Nordic
Countries
- A-869 Proposed Specialists' Meeting on the Application of Critical
Experiments and Operating Data to Core Design via Formal Methods of
Cross-Section Data Adjustment
- A-870 Future Management of the WIMS Group of Codes at Winfrith

Documents for Information

- G. Perlini et al.
Construction of a Neutron Deep Penetration Sodium Shielding Mock-up
- G. Perlini, S. Acerbis
Neutronic Spectrometry Measurements in Sodium