NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD
OF THE TWENTY-SIXTH MEETING
(TECHNICAL SESSIONS)

OAK RIDGE, USA
17th-21st October 1983

Compiled by

Peter GARVEY

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P.M. Garvey

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PART B: TECHNICAL SESSIONS

A complete list of all the papers presented at the meeting is given in
Annex 2. In the technical sessions, six new topics were introduced and
six 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
closing of the meeting.

1.   New Topics

1.1   Three-Dimensional Transient Models

Rapporteur: J.M. Stevenson
Papers A-573 and A-574

Paper A-573 described the MAGIK program which has been used to
investigate transients in AGR and Magnox reactors. It uses 3-D
multigroup diffusion theory for the neutron behaviour and has a
detailed heat transfer model. The neutronics calculation is based
upon the quasi-static approach in which the flux time dependence can
be factored into a slowly time-dependent shape function and a more
rapidly time-dependent amplitude function. The effects of
temperature changes on cross sections are taken into account and
other reactivity variations with time can also be allowed for. The
basis of the code is that different variables are calculated at
different time intervals. The general flux amplitude is calculated
most frequently, followed by the amplitude of the delayed neutron
precursors, reactivity changes, temperature effects and the relative
distributions of the flux and precursors. This gives a compromise
between accuracy and computing time. Some validation of the code
against other codes and experimental results has been carried out.

Paper A-574 discussed a one-dimensional method to carry out the
calculations of power transients from the ejection of a control rod
out of a PWR reactor vessel. A large number of these calculations
are required for various reactor conditions. The main difficulties
in the 1-D model concerned representation of feedback effects and
flux redistributions. It was found that both can be obtained from
static three-dimensional calculations at various power levels. The
average cross sections to represent each axial slice in the 1-D model
at each stage of a transient are deduced via the flux-weighted
average fuel temperature. (Linear interpolation of cross sections
with temperature was found to be conservative for the k-value and
therefore for the transient.) The peak fuel temperature is finally
obtained from the highest average temperature during the 1-D
transient using an appropriate static 3-D calculation.

The Chairman commented that there were only two papers on this topic
and suggested that every group who needs to calculate PWR
rod-rejection transients has their own code, that is sufficient for
the required application. There might, however, be specific problems
for which a more detailed thermal hydraulic interaction is required.

Maienschein reported that the Mathematics and Computing Section of the ANS had a working party considering such problems and that some previous problems have been documented in ANL-7416. There is still interesting work to be done, in particular analysis problems when phase changes occur.

1.2 Primary Circuit Modelling

Rapporteur: J.R. Askew
Papers A-581 to A-583

The three papers in this session addressed themselves to different aspects of thermal hydraulics modelling of the primary circuit.

Paper A-581 describes the results of using the COMMIX-1A code on a natural circulation experiment conducted in EBR-II. The basic transient studied was a flow reduction from 100% to 1% at low reactor power. Although the code is capable of 3-dimensional modelling, a 2-dimensional approach was used to economize in computer time. Despite the known sensitivity of the modelling to details of the thermal structures, and the use of an averaged permeability in the flow distribution baffle, good agreement was obtained even in the reversed flow regime. LeSage reported that Gelbard was advising on possible speeding up of the code.

Paper A-582 reports a benchmark exercise for the qualification of PWR pressuriser models, comparing models developed by CEGB, EDF, ENEL, and EPRI, for a range of transients for which experimental data is difficult to obtain. The differences in predictive capability are related to the modelling differences, the effects being maximized by the open-loop nature of the tests. Attention is drawn to the sensitivity, in cases where two-phase mixtures are present, to heat transfer modelling at the steam/water interface. This is especially true where the mixture is expelled through the relief valves. It is suggested that the benchmarks chosen are good tests of model performance.

Paper A-583 describes a digital simulator for PWRs. This is a simplified model capable of running at real time on a VAX-11/78 computer. Although a control desk representation is provided, the simulator is not intended for operator training but rather for research on consequences of accidents and on recovery procedures. The present version centers on a 900 MW(e) three-loop plant, though an improved model for a 4-loop plant is under development.

Discussion centered on the problems of adjusting models to reproduce the observed behaviour of real plant and in obtaining data for components, such as pump characteristics, for abnormal conditions. The possible use of artificial intelligence methods was raised and some work at ORNL using LISP was reported to be at an early stage.

1.3 Advanced Fuel Cycles

Rapporteur: H. Kusters
Papers A-584 to A-588

This session dealt with two aspects:

   a) U- and Pu-recycling in LWRs (papers A-585, A-586, A-587), and

   b) Investigations on a high conversion tight lattice PWR with mixed oxide fuel (papers A-584 and A-588)

a) Pu-Recycling in Thermal Reactors. (A-587)

This paper contains a brief contribution published in 1980 and a more recent one (Nov. 1982), which describes the present achievements in, and plans for, Pu recycling of Kraftwerk Union (KWU) in the FRG.

KWU has irradiated almost 10,000 pins with MOX fuel in their reactors. In none of these tests has normal reactor operation been restricted by the MOX fuel assemblies. It has been shown that MOX fuel rods and standard uranium rods behave identically with regard to changes in outer diameter and the growth in length of the fuel rods during burnup. Maximum local burnups of 40 GWd/t have been reached for MOX fuel. As far as physics aspects are concerned, KWU has verified the calculated power density distribution, the burnup behaviour, and the reactivity work of control rods in MOX fuel assemblies (fresh and irradiated). Postirradiation isotopic analysis shows a highly satisfactory agreement between theory and experiment for the plutonium nuclides concentrations. The decrease in reactivity worth of multiple recycled Pu (due to the progressive burnup of Pu-239 and buildup of Pu-240 and Pu-242) can be accounted for easily by increasing the enrichment of the subsequent recycling generations.

The paper also touches on questions related to the manufacture and reprocessing of MOX fuel assemblies. Economically an advantage is being deduced for Pu-recycling in thermal reactors, depending on the value of Pu, the price of natural uranium and the price for MOX fuel fabrication.

The present plans are to recycle plutonium in LWRs only to that amount which is not to be used for fast breeder programs with FRG participation.

The discussion showed that similar conclusions have been reached in other European countries. The aspects of Pu recycling in thermal reactors are completely understood. Important parts of the theoretical analysis have been satisfactorily verified by experiments.

Uranium-Recycling in LWRs. (A-585, A-586)

Uranium recycling in thermal reactors has also been investigated in some countries, e.g., in Canada, France and FRG. As in the case of Pu-recycling, the results obtained are very similar. The French contribution, A-586, describes that the multiple recycling of uranium is limited because of the buildup of the absorbers U-236 and U-234. Additional difficulties arise from the buildup of Tl-208 and Bi-212 as daughter nuclides from the decay of U-232. These nuclei emit high energetic gammas (2.6 MeV), which requires additional shielding in fuel

factories, to avoid this penalty during fuel element fabrication, the cladding and assembly has to be done immediately after reprocessing.

The conclusion of the French study, which is in agreement with the results of investigations in other countries, is that uranium should only be recycled once.

Because of a request from the Chairman of the Working Party on Nuclear Fuel Cycle Requirements to NEACRP, a simple benchmark on the special aspect of U-236 recycling should demonstrate that the prediction of the U-236 concentration after one recycling step is internationally in agreement (paper A-585). This simple benchmark will be set up by France.

b) <u>Investigations on High Conversion Tight Lattice PWRs.</u>

In paper A-588 from FRG, recent publications on this subject are collected. These papers contain information on the motivation to investigate high conversion tight lattice PWR configurations ($V_M/V_F \sim 0.5$) in the U-Pu fuel cycle mode. Physics and thermal hydraulic studies for four concepts are described: the homogeneous design, two modular heterogeneous designs with movable or fixed seed positions, and a ring core design similar to what is investigated in heterogeneous fast reactors to improve breeding and coolant loss reactivity effects. A variety of about 60 critical assemblies (fast and epithermal) have been calculated. The agreement in the criticality prediction is satisfactory for the purpose of a consistent analysis in a feasibility study. An important constraint of the investigation is to guarantee a sufficiently negative coolant void reactivity feedback so that normal PWR licensing procedures can be applied. To study this effect, operational transients without scram have been performed with different coolant density reactivity feedback parameters in a homogeneous tight lattice core. From these studies it appears that only in a widened lattice can a sufficient negative feedback be guaranteed, with still an acceptable conversion ratio of about 0.9.

In paper A-584 the experimental investigations in the PROTEUS reactor of EIR Wurenlingen are described. These studies concentrated on measurements of reaction rates and the coolant void reactivity effect using H_2O, Dowtherm (simulating H_2O void of 42.5%) and air (100% void) as "coolant" material. For the clean core with a fissile-Pu enrichment of 6% the void effect was measured to be negative. The comparison with standard thermal reactor calculational methods and their data libraries, WIMS-D and EPRI-CPM, has shown discrepancies which are very much greater than generally encountered for LWR lattices. To a large extent this discrepancy may be due to nuclear data, which for WIMS-D were established in 1961. Further information can be found in A-588.

In the discussion it was pointed out that improved calculational methods are in progress in FRG to study the sensitivity of the void effect on methods more deeply; as far as nuclear data are concerned, the latest version of the KEDAK library is used in these investigations.

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The NEACRP encourages further research in this area to clarify especially the physics aspects of a tight lattice PWR design.

1.4 Prediction of Pin Rating

Rapporteur: H. Neltrup
Papers A-589 and A-590

In paper A-589, the precision of pin power prediction from coarse mesh solutions was examined by help of a reference solution obtained by a super assembly calculation covering an entire quarter core from a BWR reactor. Apart from acting as reference, this solution also provided the information normally obtained from coarse mesh solutions such as mean assembly power and appropriate boundary parameters for the assembly. Two methods of obtaining pin power distribution from this information were compared: the normalization method and the superposition method. In the first, the heterogeneous pin power distribution obtained in the assembly with nonleakage boundary conditions is normalized by mean assembly power. In the second, the pin power distribution is obtained via a best fit to the assembly boundary conditions by a set of precalculated base solutions. These were obtained with zero leakage on three sides of the assembly and a given J/ϕ shape on the third, multiplied by an eigenvalue, to obtain criticality for the assembly.

The investigation showed that very large pin power errors (up to 50%) could arise in the homogenization method, whereas with the superposition method using only 8 base functions, the maximum pin power error could be kept within 8%. In the second part of A-589, the minimum pin power error during burnup when burning in assemblies with nonleakage boundary conditions was examined. The conclusion was that if the target value of 5% error was to be obtained, burnup has to take place in a realistic leakage spectrum.

In A-590 an investigation was carried on along lines very similar to the ones in the first part of the preceding paper. However, a third method, the flux-lupe, was also considered. Furthermore, the interface with two typical coarse mesh methods, the nodal coupling and the finite difference, was treated. In connection with the latter, the effect of heterogeneity factors introduced by Koebke was assessed. With the superposition method, the set of assembly boundary parameters as well as the eigenvalue determination was different from the ones in A-589 but generally the trend from this paper was confirmed.

1.5 Fine Structure of Energy Deposition During Operation

Rapporteur: J.M. Stevenson
Paper A-591

Paper A-591 described heating calculations for an equilibrium model of PFR. A three-dimension (tri-z in a 60° sector) model was used. Neutron and gamma transport were modelled by diffusion theory in 37

and 13 groups respectively, and the neutron heating by scatter reactions was correctly treated. Supplementary calculations showed that the errors associated with the standard mesh (6 triangles per subassembly and an axial mesh of ~90 mm) or with the use of 6 groups for the neutron calculation (using condensed cross-sections obtained with selected zone spectra) are acceptably small. The errors associated with assuming that all heat is dissipated at the point of reaction are also small for all locations except for non-fissile /non-fertile subassemblies where significant underpredictions are found. The error is a factor of 5 for the central guide tube.

It was observed that the correct prediction of total heating in a fast reactor subassembly is important for fixing the coolant flow with implications for temperature rise and bowing effects.

It was reported that heating calculations for the French fast power reactors assume that all the heat is deposited at the reaction point for all subassemblies except specials. For each of these, supplementary transport calculations for neutrons and gammas are carried out in two dimensions, with the relevant subassembly at the center.

The uncertainties in gamma heating from the gamma source data were discussed. These arise largely from the photon density and spectrum and are estimated as ~ 20%. The uncertainties in experimental determinations of gamma deposition rates were considered and the discrepancies between different techniques noted. The Chairman suggested that papers on the state of the art of the measurements of gamma deposition and associated problems should be invited for the next meeting.

1.6 <u>Prediction of Rating Distribution in Large FBR Cores through Burnup</u>

Rapporteur: M. Salvatores
Papers A-575 to A-580

Six papers were presented, covering broadly two subjects, namely (a) simplified models to calculated power distributions in large LMFBRs taking into account burn-up effects, fuel management requirements, and reactor operation results and (b) experimental data in large cores of critical facilities, relevant to the prediction of power distribution.

Concerning the first subject, the French papers (A-575 and A-576) presented the perturbation method used in the fuel management and optimization code SUPERCAPHE, which will be used for SUPERPHENIX, and in particular its validation against reference calculations. The second paper dealt with improvements to that perturbation method being presently developed. The UK papers (A-577 and A-578) presented respectively a simplified model to take into account control rod movements during the cycle of PFR and the method used to predict S/A power variation during the cycle of a large commercial fast reactor. The first paper showed that acceptable errors result in ignoring the movement of the control rods during a cycle; the second demonstrated the importance of representing the relative insertion of the inner

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and outer rings of control rods and their influence in optimising the radial form factor.

Concerning the second subject, the two remaining papers (A-579 and A-580) reported the evidence for a significant radial dependence of the C/E values on reaction rate distributions in large cores (both homogeneous and heterogeneous) of the ZPPR program. Discrepancies at core edge were reported both by the Japanese and by the US groups; in general, it is observed an increase of the C/E values up to 6% at the outer core edge relative to the core center after having applied correction factors to take into account transport, mesh, cell asymmetry etc. effects. Sensitivity studies were performed which indicate that a part of the explanation of the discrepancy can be eventually traced back to cross-section data.

In the course of the discussion, LeSage indicated that the use of point Monte Carlo in the experiment analysis could only slightly reduce the discrepancy with respect to a standard (transport corrected) deterministic calculation (from 8% to ～5% in a specific configuration). It seemed then that more than one cause should be looked for (basic data, core calculation methods, gradient effects in cell calculation, etc.) and that further experimental evidence from other laboratories for the same type of problem would be very helpful. In fact, both the ZEBRA and MASURCA teams have been observing flux tilt effects in larger critical configurations.

Moreover, since similar C/E space behavior has been reported in the case of reactivity worth, it seemed of interest to further investigate the correlation of the two effects.

2. Topics Carried Over from Previous Meetings

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

Rapporteur: J.R. Askew
Papers A-595 and A-596

Dr. Whitesides summarized papers A-595 and A-596. This activity arose because of difficulties in predicting criticality of large arrays. The working group had been asked to examine three questions:

1. What is the safe value of k-effective?
2. Is there a computational problem in dealing with large arrays?
3. Do mixed packages lead to worse conditions than aggregates of similar packages?

They had concluded, in respect to the second question, that the observed discrepancies (which ranged from 3-6% in reactivity in the USA and 3-10% in Japan; both in the direction of the calculation being less multiplying) were unlikely to be due to calculational problems.

Concerns had been identified with the experimental data, especially with the chemical analysis of systems with dissolved fissile

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material. It might be that more experiments were needed; at least a critical appraisal of existing information would be required.

The possibility of the working group considering dissolver criticality problems in some general (non-proprietary) geometry was under discussion.

Rapporteur: M. Rief
Papers A-592 to A-594, A-597 and A-598

In paper A-592 the main characteristics of MONK6, developed from MONK5, were presented. They include improved sampling schemes for eigenvalue calculations and detailed (8500) point data input. The ultimate aim of this development was a Monte Carlo code capable of predicting k_{eff} to within 1% for all U and Pu systems.

This goal could be reached for almost all U-systems with MONK6.1. Pu and mixed Pu-U systems were still overpredicted which led to a further improvement of data, such as the refinement of thermal scattering models, self-shielding in the unresolved resonance region, substitution of fissile material cross sections by recent evaluations (JEF), etc. These changes are realized in MONK6.3. It produces substantially improved k_{eff} values for many more critical systems (including the ones containing Pu) leaving, however, discrepancies of 3-5% for the mixed U-Pu assemblies and the gadolinium poisoned ones.

In paper A-593 k_{eff} values from VIM continuous energy Monte Carlo calculations are compared with results from sophisticated deterministic methods for a variety of ZPR assemblies. In some cases plate versus pin biases are also established. The analysis shows a small (0.3 - 0.5%), but systematic overprediction of k by the deterministic methods. In the discussions, no agreement could be reached if further investigations could remove the reported discrepancies which might result from the different treatment of the nuclear input data by the two methods used.

An eventual benchmark exercise on this topic was aired during the discussion, but it was concluded that too few laboratories could tackle the problem in an adequate manner.

Paper A-594 describes TRIMARAN, a new French Monte Carlo code belonging to the TRIPOLI family, mainly developed for 3-D criticality safety analysis. It uses the TRIPOLI geometry capabilities and a multi-group cross-section scheme. Its main virtue is computational speed which makes TRIMARAN a candidate for routine calculations and parametic studies. To reduce the variance of k_{eff}, a combination of three different estimators is used (i.e., neutron balance by collision and track length estimation, and production per batch).

A generalized formulation of two different Monte Carlo perturbation algorithms is presented in paper A-597 for the collision and the track-length estimator. They are based on correlated sampling and a second order Taylor series approach. In a synopsis their mutual advantages are discussed, especially in view of uncertainty behaviour. A promising scheme for their application in eigenvalue (criticality) problems is included, together with an example of

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calculating the coolant temperature coefficient in a D_2O test reactor.

Paper A-598 described the application of Monte Carlo to fuel burnup problems in a 1-D slab. Results are compared with those of equivalent deterministic calculations. Estimates of Monte Carlo uncertainties were quoted and it was also mentioned that the sampling procedures in use are self-stabilizing.

The calculation included cases where the distribution of k_{oo} in the slab was made to vary both linearly and quadratically with irradiation. One example simulated the withdrawal of a control rod. In a rather extreme case of rod movement, a calculation was completed in which a rod was partially inserted or withdrawn at alternate depletion steps. This study could open the door to an area where the potential of Monte Carlo might become very beneficial in the future.

2.2 Out-of-Pile Production of Fissile (and fusile) Material

Rapporteur: F.C. Maienschein
Papers A-599, A-600, L-267, and L-268

In paper A-599, Bartine of ORNL summarized views resulting from a brief study of possible accelerator breeders. Although long-term R&D is required for careful evaluation, progress is noted in relevant accelerator development, which is carried out for other purposes. A concept with a sodium-cooled target region containing ternary metal spheres (6% Pu, 16% U, 78% Th) indicated a net fissile production of 6 kg/d with a large initial fissile inventory. To be cost effective, fissile material values must be high and power must be sold. Fission-suppressed blankets appear to lead to much higher costs for the fissile material produced.

Garvey stated that Canadian studies indicate that intermediate fissile loadings, with no net power production, appear to be desireable. Assuming that LMFBRs remain more expensive than water-cooled reactors, accelerator breeders supplying fuel to many such reactors could be less expensive than an all-LMFBR system.

In paper L-267 from Japan, the yield and spectra of neutrons from high-energy charged particles are considered. Low-energy neutrons, with an evaporation peak at $\sim$3 MeV, are well predicted. Intermediate-energy neutrons are presumed to arise from the pre-equilibrium nucleus. Decrease of the effective nucleon-nucleon interaction within the nucleus appears to improve agreement of predicted and measured intermediate-energy spectra while decreasing neutron yields. Further calculations for larger targets are planned.

The LOTUS Fusion-Fission Hybrid Project in Switzerland is described in paper A-600. This integral test of blanket neutronics will verify methods and data, with first operation planned for 1984. Fusion-fission hybrids may be attractive because they offer the possibility of very large fissile production rates per unit power compared to a breeder reactor. An intense 14-MeV neutron source (5×10^{12} n/s), stepwise addition of blanket components in a

slab geometry, and a variety of detectors for measuring reaction rates, tritium production and neutron spectra are planned.

For fusion programs, plans must include breeding tritium. Paper L-268 reports measurements of tritium production-rate distributions in simulated blanket assemblies in the FNS at JAERI. The purpose is to test data and methods used in neutronics design. The intense 14-MeV neutron source at FNS has been analyzed with fairly good agreement between measurements and calculations. Measurements with Li_2O assemblies were made with three detection techniques (Dierckx method, LiF TLD, and liquid scintillation counting). Differences among methods and calculations somewhat exceed expected errors but studies continue. Young's evaluation for the $^7Li(n,n'\alpha)^3T$ cross section is clearly preferred.

In general summary, it appears that studies of out-of-pile fissile production are of rather specialized interest. In contrast, tritium-breeding studies for fusion programs appear widespread, and further sharing of information by the committee about experimental techniques and results should be helpful.

2.3 Calculational Methods for Evaluating Control Rod Effects in FBRs and Their Validation (in Rod Reaction Rates and Lifetimes)

Rapporteur: K. Shirakata

Four papers were presented and discussed: A-601, A-602, A-603, A-604, and two papers, submitted to Nuclear Science and Engineering, were distributed: A-605 and A-606.

Paper A-601 from the U.S. describes measurements of reaction rates in and around simulated control rods at ZPPR. Measurements were made using U-235 foils and TLDs. The overall influence of inserted control rods on the reactor fission distributions in ZPPR-3/1B was well predicted with 6-group diffusion calculations in xy and xyz geometry. Reaction rates in the natural boron pin-type control rods were predicted with S_4 transport theory within 5%, when normalized to experiment outside the rod. Measurements of axial U-235 fission rate distributions in and adjacent to fully enriched boron pin control rods in ZPPR-11A were preliminarily analyzed with diffusion theory within an error of 20%. Axial distributions adjacent to fully inserted control rods in ZPPR-11D were well predicted with 28-group data.

Paper A-602 from Japan describes a homogenization method for control rods, considering the neutron leakage effect. An effective homogenization method, which preserves the integrated reaction rates in a heterogeneous super-cell calculation, has been extended to treat off-center control rod channels in FBRs. An albedo at the super-cell surface was combined with collision probabilities to treat the neutron leakage. The method was applied to 1-D off-center rod worth calculations. As a result, discrepancies of off-center control rod worths from the reference calculations were very much improved.

Paper A-603 from France describes the boron capture in boron carbide rodlets irradiated in PHENIX, a comparison of calculation and

experiment. For SPXI design purposes, 90% B-10 enriched carbide rodlets were irradiated in PHENIX in the frame of the so-called PRECURSAB irradiation program. Calculations performed with SPX1 design models, lead to a E/C + 1.1 $\pm$ 0.3 for the mean value of the maximum B-10 consumption. The large uncertainty is associated with the uncertainty in the exact position of the irradiated boron carbide sample used in the isotopic analysis.

Paper A-604 from France describes a sensitivity approach to study control rod worth uncertainties. The systems which have been considered are a large LMFBR of the SUPERPHENIX type and the MASURCA critical configurations PRE-RACINE and RACINE. Sensitivity profiles both in energy and in space for several kinds of control rods in these systems are investigated. In conclusion, sensitivity studies are helpful in analysing the possible influence of data uncertainties on control rod worth values and in defining future integral experiments.

In the subsequent discussion, the following comments were made. The effect of global flux gradience on CR worth might impact the prediction of CR worths, particularly in the case of heterogeneous cores. The consideration of the effect will mitigate the C/E bias of CR worth observed in the large FBR cores. The results indicate that reaction rates inside and near rods are predicted reasonably well but additional data and calculations would be very helpful to further evaluate this issue.

In summary, the general picture on this topic is not very much changed from that of the last year, that is, the CR worths are fairly adequately predicted using the present methods, the problem being the impact of the present uncertainties in defining bias for future large cores. The problem of prediction of reaction rates inside and near CR rods will be discussed next year in the item on power distribution in large FBR cores.

2.4   Reactor Physics Modelling of Distorted Cores

Rapporteur: H. Kusters
Papers A-607 to A-609

Paper A-609 is a summary report, submitted to Nuclear Science and Engineering, on calculations and those measurements in SNEAK, which were discussed last year. Paper A-607 was written in fulfillment of an action to forward the discussion of the 25th meeting on distorted LMFBR cores to CSNI. As an additional remark, there exists a worldwide consensus that transport theory (S_4) is able to describe reactivity effects in idealized distorted core configurations reliably. This was a result of a SIMMER-II workshop held in Karlsruhe in July 1983. Paper A-608 describes recent measurements in SNEAK in a Pu-test zone. In comparing theory and experiment, most of the results confirm the conclusion drawn at the last meeting. The preliminary analysis of small slump-out and large slump-in configurations, however, shows unexpected discrepancies. Both cases will be re-analyzed.

In general, it was felt that at least for idealized geometries, reactivity effects of distorted core configurations can be described to an acceptable degree of accuracy. Large uncertainties in the theoretical analysis of unprotected accidents in LMFBRs may be connected with the non-neutronic modelling of the accient sequences.

2.5 Intercomparison of Reaction Rate Measurements in Fast Reactors

Rapporteur: L.G. LeSage
Papers A-610 and A-611

There were two papers in this session, one (A-611, US) dealing with the results of reaction rate intercomparisons in the US and UK, and the second (A-610, France) describing an upcoming international reaction rate intercomparison at Cadarache.

Preliminary results from a US/UK reaction rate intercomparison were presented at the NEACRP meeting in 1982. In Paper A-611, the standard ANL (US) calibrations for the three reaction rates, U-235 fission, Pu-239 fission, and U-238 capture are examined. The data are presented in tables showing how much the ANL reaction rate results would be altered if an alternative calibration were selected, such as the UK (ZEBRA) intercomparison results. For U-235 fission, there is good agreement among the alternative calibration methods. For U-238 capture the evidence suggests that the US experimental results should be increased by about 1%, which will bring them into much better agreement with both the ZEBRA result and a recent mass spectrometer measurement. A difference of 2.5% between the US and UK Pu-239 fission results is indicated. This difference, which is considerably larger than expected, is so far unexplained.

Paper A-610 describes the IRMA program. This is a planned international intercomparison of key reaction rates to be conducted on MASURCA at Cadarache in May and June 1984. Seven laboratories are currently planning to participate in IRMA. They include: ENEA, Casaccia, Italy; CEN-SCK, Mol, Belgium; KFK, Karlsruhe, FRG; UKAEA, Winfrith; CEA Cadarache, both MASURCA and ERMINE teams; and ANL (West), US.

In the course of the discussion it was decided that these inter-comparisons represent the state of the art in fast reaction rate measurements and should be brought to the attention of the data community since there may be implications on nuclear data or thermal reaction rate results.

3. National Programmes

Reports on the reactor physics activities in the NEA member countries were summarized and discussed. The full reports will be included in a consolidated report (L-265) to be issued by the NEA Secretariat.

4. Benchmarks

4.1 Noise Analysis Benchmark

Three physical benchmark problems have been defined and sent to the initial participants for comments. A questionnaire to determine the number of expected participants for each of the three problems

(PWR, BWR, and FBR) has also been issued. It is planned that an analysis of all contributions will be ready for the SMORN-IV meeting in October 1984.

4.2 Intercomparison of Cell Heterogeneity Effects in Pin and Plate Geometries

A Specialists' Meeting was held at AEE Winfrith in June 1983 to discuss the seven solutions provided by six countries (Japan(2), USA(2), UK, France, FRG and Italy). A progress report (A-614) was distributed and Stevenson gave a status report. The discrepancy between calculations and experiment for the pin - plate core reactivity difference lies in the range +0.15 to +0.82% δk, where the experimental uncertainty is $\pm0.15\%$ δk. The calculated pin-plate reactivity differences were separated into their several components in order to better understand the causes of the variation among the calculations. However, although these analyses better identified where the differences were occuring, the reasons for the discrepancies were not established. Recommendations have been made for further analysis and measurements to help resolve this problem.

LeSage briefly discussed paper A-615 in which good agreement had been obtained for plate geometry between Monte Carlo and deterministic methods.

Two other papers (A-616, A-617) were distributed that described the methodology being used in this benchmark calculation.

4.3 Benchmark on Interactive Effects in Gadolinium Poisoned Pins in BWRs

The main results and conclusions of the status report, A-567, were presented by Wydler. There were significant differences among the solutions and the target accuracy had not been achieved. Even for the simple unpoisoned pin cell there were large differences. The impact of such discrepancies in the power distribution within an assembly might, however, be sufficient from an operational standpoint.

A decision as to whether to continue this benchmark exercise would be based upon there being sufficient interest from the participants.

Hemmig brought to the attention of the committee that two papers on this topic were to be presented at the November 1983 ANS meeting and that information on Gd poisoned PWR assemblies is to be found in CEND-397.

4.4 PWR Multi-Dimensional Kinetics Benchmark

As experimental information for a PWR benchmark was not available from EPRI, it was not possible to procede.

4.5 Reactivity Scale and Central Worths Benchmark

A proposal (A-623) for such a benchmark was discussed. It was felt that a simpler benchmark would be more appropriate due to the large computational effect required for the one proposed. Participants were invited to offer comments on the proposal.

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4.6 Radiation Shielding Benchmark

Two documents, A-618 and A-619, were distributed. Rief described the results of the PWR shielding Benchmark, documented in A-618. He stated that agreement among the several contributions was significantly improved as compared to the situation at the 1976 Vienna meeting. Larger differences were still obtained and these must be due to the processing codes as a common data base was used. The main conclusion was that to meet target accuracies for neutron damage and activation reactions in the pressure vessel, adjustments need to be made to the iron inelastic cross section.

As a possible benchmark, members were asked to identify potentially suitable experiments.

4.7 LMFBR Burnup Benchmark

Salvatores stated that the USSR (Obninsk) had provided a contribution and that this would be included in the final report.

4.8 PWR U-236 Benchmark

A Benchmark to identify the accuracy of calculating the impact of U-236 in a PWR pin cell is to be specified by France. This topic was discussed in Section B 1.3.

5. General

5.1 Highlights of Recent Meetings of Interest to NEACRP

-- Sixth International Conference on Radiation Shielding, May 1983, Tokyo, Japan. This was a very successful conference. A summary report of the meeting was issued (A-620).

-- Specialists' Meeting on In-Core Instrumentation, October 1983, Halden, Norway. A verbal report on this meeting was given by Martinelli who also offered to provide a written summary. There were 50 participants and 24 papers were presented covering a wide range of aspects. The general opinion of participants was that it had been a most useful meeting. The proceedings will be available in several months.

-- Specialists' Meeting on the Intercomparison of Cell Heterogeneity Effects in Pin and Plate Geometries, June 1983, Winfrith, UK. A discussion of this meeting is to be found in section 4.2.

5.2 Future Meetings of Interest to NEACRP

-- Specialists' Meeting on Reactor Noise - SMORN-IV, October 1984, Dijon, France. Publicity for this meeting was distributed, which will also be sent to the L-report distribution list.

-- Specialists' Meeting on the Treatment of Resonance Data

The need for such a meeting was previously discussed during the Executive Sessions. It was agreed that there was a value to such a meeting and Stevenson agreed to be responsible for it.

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It was also felt that it could be advantageous to hold a further Specialists' Meeting in conjunction with the International Conference on Nuclear Data to be held in April 1985 in Sante Fe, USA.

- ANS Topical Meeting on Reactor Physics and Shielding, September 1984, Chicago, U.S.A.

ANNEX 1

LIST OF PARTICIPANTS

Delegates

For Canada	Mr. P.M. Garvey	Scientific Secretary
For Japan	Dr. T. Asaoka	
	Dr. K. Shirakata	
For the USA	Dr. P.B. Hemmig	
	Dr. L.G. LeSage	
	Dr. F. Maienschein	
For the countries of the European Communities and the European Commission acting together.	Dr. M. Rief	(CEC)
	Dr. M. Salvatores	(France) Vice-Chairman
	Dr. C. Golinelli	(France)
	Dr. H. Kusters	(F.R. of Germany)
	Dr. R. Martinelli	(Italy)
	Dr. J. Askew	(United Kingdom) Chairman
	Mr. J.M. Stevenson	(United Kingdom)
For the other European Countries of the OECD	Mr. H. Neltrup	(Denmark)
	Dr. P. Wydler	(Switzerland)
Nuclear Energy Agency	Dr. L.G. de Viedma	
	Dr. P. Nagel	Secretariat
Observers (all sessions)	Dr. F.G. Perey	(NEANDC)

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

ANNEX 2

NEACRP DOCUMENTS PRESENTED AT THE 26th MEETING

"L" Documents

L-265   Reactor Physics Activities in NEA Member Countries

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

L-267   K. Tsukada and Y. Nakahara
         The Yields and Spectra of Neutrons from Cascade Process in the
         High Energy Spallation Reaction.

L-268   H. Mackawa, K. Tsuda, T. Iguehi, Y. Ikeda, Y. Oyama, T. Fukumoto,
         T. Schi and T. Nakamura
         Measurements of Tritium Production-Rate Distribution in Simulated
         Blanket Assemblies at the FNS.

"A" Documents

A-571   NEA Data Bank Activity Report, October 1983.

A-572   NEANDC Activity Report

A-573   J.K. Fletcher and M.A. Perks
         MAGIK. A Computer Program to Investigate Transients in AGR and
         MAGNOX Reactors.

A-574   R. Lenain
         Ejected Rod Calculation, Kinetic Axial Modelisation.

A-575   R. De Wouters and F. Flamembaum
         Power Distribution Prediction for a Large Commercial L.M.F.B.R.

A-576   G. Palmiotti and M. Salvatores
         Perturbation Methods for Power Distribution Prediction.

A-577   D.J. Lord and T.D. Newton
         The Effects of Simplified Control Rod Modelling on Neutronics
         Calculations for the Prototype Fast Reactor.

A-578   R.E. Sunderland
         The Prediction of Sub Assembly Powers throughout the Fuel Cycle
         of a Commercial Fast Reactor.

A-579   Y. Kato, M. Yamamoto, K. Shirakata and T. Takeda
         Radial Dependence of C/E Value in JUPITER-I Core Analysis.

A-580   P.J. Collins
         ZPPR Data Relevant to the Prediction of Power Distribution in
         Large LMFBRs through the Burnup Cycle.

A-581   W.L. Baumann, W.T. Sha and H.M. Domanus
         Thermo Hydraulic Transient Simulation of LMFBRs using COMMIX-IA
         Computer Code.

A-582   IFAC WORKSHOP
         Modelling and Control of Electric Power Plants.

A-583   P. Bernard, C. Bonnett, J. Chinardet, F. Ducamp, S. Nisan and
         J. Romeyer - Dherbey
         SALAMANDRE: A Digital Simulation for Night Water Pressurized
         Reactors.

A-584   R. Chawla, K. Gmuer, H. Hager and R. Seiler
         LWHCR Moderator - Voidage Experiments.

A-585   J.R. Askew
         Recycle of Uranium-236.

A-586   C. Golinelli
         The French Neutronics Studies Concerning the Uranium Recycle.

A-587   O. Beer and P. Schmiedel
         Technical and Economic Aspects of Plutonium Recycling.

A-588   H. Kusters
         Investigations on a Tight Lattice PWR in the Federal Republic of
         Germany.

A-589   F. Niscen
         The Superposition Method, a New Approach to Pin Power
         Determination.

A-590   M. Hamasahi, W. Kawamura and T. Takeda
         Comparison of Pin Powers in Thermal Reactors Calculated from
         Coarse Mesh Methods.

A-591   G. Reddell and A.T.D. Butland
         Distributed Heating Calculations for a Fast Reactor; Application
         to the PFR Equilibrium Core Model.

A-592   P. Hogue, G. Walker, R.J. Brissinden and D.E. Bendall
         The MONK Code and its Validation from Data.

A-593   R.D. McKnight
         New Developments in the Critical Eigenvalue Assessment.

A-594   G. Erment
         TRIMARAN: A Three Dimensional Multigroup Pl Monte Carlo Code
         for Criticality Studies.

A-595   G.E. Whitesides
         Status of CSNI Working Group on Standard Problem Exercises on
         Criticality Calculations for Spent Fuel Transport Cashs and
         Arrays of Fissile Material Packages.

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A-596   J.T. Thomas
Difficulties with Experiments Involving Arrays of Fissile
Solutions.

A-597   H. Rief
Generalized Monte Carlo Perturbation Algorithms for Correlated
Sampling and a Second Order Taylor Series Approach.

A-598   A.F. Course, M.J. Halsall and J.L. Hutton
Monte-Carlo Depletion Calculation in a Slab.

A-599   D.E. Bartine and J.O. Johnson
Comments on Out-of-Pile Production of Fissile Material.

A-600   L. Green, P.A. Haldy, A. Kumar, C. Sahraoui and K. Gmur.
The LOTUS Fusion-Fission Hybrid Project.

A-601   P.J. Collins
Measurements of Reaction Rates In and Around Control Rods at
ZPPR.

A-602   S. Ono and T. Takeda
A Homogenization Method of Control Rods with Neutron Leakage
Effect.

A-603   G. Humbert, G. Palmiotti and C. Tournier
Boron Capture in Boron Carbide Rodlets Irradiated in PHENIX: An
Experiment to Calculation Comparison.

A-604   G. Palmiotti and M. Salvatores
A Sensitivity Approach to Study Control Rod Worth Uncertainties.

A-605   H. Giese
Control Rod Worths and Interactions in Fast Reactors.

A-606   G. Humbert, K. Kappler, M. Martini, G. Norvez, G. Rimpault,
B. Ruelle, W. Scholtyssek and A. Stanculescu
Parametric Studies for the Heterogeneous Core Concept in the
Framework of the PRERACINE and RACINE Experimental Programs.

A-607   H. Kusters
Validation of Neutronic Calculations for Distorted Core
Configurations Arising in Accident Situations of LMFBRs.

A-608   F. Helm and G. Henneges
Reactivity Effects of Fuel Rearrangement in Fast Reactor Rod
Bundles.

A-609   F. Helm, G. Henneges and W. Maschek
Measurements and Computation of the Reactivity Effects of
Accident-Caused Core Distortions in Liquid Metal Fast Breeder
Reactors.

A-610   W. Scholtyssek
Intercomparison of Reaction Rate Measurement Techniques in
MASURCA.

A-611   D.W. Maddison and S.G. Carpenter
Intercomparison of Reaction Rate Measurements on Fast Reactors.

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A-612   H. Kusters and S. Pilate
         The Present Accurary of Physics Characteristics of Unirradiated
         Fast Reactor Cores.

A-613   S.-O. Lindahl
         Summary of the Meeting "Status of Statie Reactor Calculations in
         Nordic Countries".

A-614   M.J. Grimstone, J.L. Rowlands and J.M. Stevenson
         Progress Report on the International Comparison of Calculations
         for the CADENZA Assemblies (the Pin-Plate Benchmark).

A-615   ANL
         Intercomparison of Cell Heterogeneity Effects in Pin and Plate
         Geometries.

A-616   M.J. Grimstone and J.L. Rowlands
         Collision Probabilities for Plate Geometry Cells with an
         Approximate Treatment of Plate Edge Regions.

A-617   M.J. Grimstone and J.L. Rowlands
         A Method of Calculating Collision Probabilities for the Pin Cell
         Minicalandria Used in Fast Reactor Critical Facilities.

A-618   G. Hehn
         Results of the NEA Shielding Benchmark.

A-619   G. Hehn, R.-D. Bachle, G. Pfister and M. Mattes
         Adjustment of Neutron Multigroup Cross-Sections to Integral
         Experiments.

A-620   T. Asaoka, T. Hyodo, T. Suzuki and S. Kikuchi
         The Sixth International Conference on Radiation Shielding. A
         Brief Overview.

A-621   H.F. McFarlane, S.G. Carpenter, P.J. Collins, D.N. Olsen and
         S.B. Brumbach
         Experimental Studies of Radially-Heterogeneous LMFBR Critical
         Assemblies at ZPPR.

A-622   K.S. Smith and R.W. Schaefer
         Recent Developments in the Central Worth Discrepancy.

A-623   K.S. Smith and R.W. Schaefer
         Proposal for an International Comparison Calculation of Sample
         Reactivities.