NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD
OF THE THIRTY-SECOND MEETING
TECHNICAL SESSION

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9-13 October 1989

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The Thirty-Second NEACRP meeting was hosted by the Argonne National Laboratory. A technical visit was arranged to the Fermi National Accelerator Laboratory. The visit took place on Wednesday, October 11 and was particularly interesting and timely.

In the Technical Session, 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 by the other members of the Committee.

1. New Topics

1.1 Recent Concepts to Reduce the Sodium Void Reactivity in Fast Reactors

Rapporteur: David C. Wade
Papers: A-1005, A-1014, and A-988


The US paper was addressed at metal fueled reactors of two sizes, 450 MWth and 900 MWth. The goal was to qualify the tradeoffs among sodium void worth reduction and resultant changes in burnup control swing, breeding gain, fissile inventory, core diameter, and core volume. Each data point in the tradeoff curves represented a legitimate design in terms of peak linear heat rating; peak discharge burnup and fluence and core pressure drop; all cases were based on Hex-Z equilibrium cycle depletion studies with the sodium void worth quoted at EOC for all flowing sodium in core + internal blankets + upper axial blanket and reflector. Three classes of design change were discussed:

(a) composition changes at fixed core layout--encompassing changes in steel, Na, and void volume fractions and addition of Be0 and of B4C;
(b) changes in height to diameter ratio at fixed assembly design—and for pancaked cores a side study of changes in P/D ratio in the assembly design, changes in axial blanket thickness and/or axial blanket to absorber substitution; and

(c) changes in core layout—encompassing axial vs. radial heterogeneous vs. homogeneous layouts, annular cores, and coupled cores.

The conclusions of the study were that:

(a) Na void worth can be reduced to near zero or even made negative—but the result will be a nonfavorable change in one or more of the other performance parameters considered.

(b) There is no universal best way to reduce sodium void worth because the relative importance of the several other performance changes will depend on the local, current institutional and economics environment.

The value of the study lies in its quantification of the tradeoffs available for the reactor size range (450 to 900 MWth) and fuel choice (metal) which was considered.

The paper from the USSR was addressed at both medium and large reactors (~500 and ~1600 MWe) fueled with mixed oxide with a goal to hold burnup control swing near zero while reducing Na void worth to a negative value so as to contribute to the broader goal of a high degree of reactor passive self protection. The concept investigated made use of a homogeneous layout of ductless assemblies having driver mixed oxide pins and breeder metal pins commingled in the same assembly—in a 2 to 1 ratio. The steel volume fraction was reduced to the ~5% range by virtue of the removal of ducts; the sodium volume fraction was varied to as high as ~60% by virtue of increasing the pitch to diameter ratio up to 1.7. The core height was varied between 150 and 100 cm. And the radial and axial blanket and reflector thickness were varied parametrically.

The conclusions of the study were that:

(a) a negative sodium void worth and zero burnup control swing could be achieved simultaneously at both power ratings considered.

(b) This was achieved by use of the combined effects of the several design means available—including:

i. structural material reduction (by using ductless assemblies)

ii. core height reduction

iii. pitch to diameter ratio optimization

iv. thin axial blankets

Moreover, the study quantified the impacts of the various changes taken individually. It was remarked that at a refined design stage Pb or Pb/Mg coolant might be introduced as a way to reduce the
negative impact on breeding gain of designing to negative void worth while also benefiting from the high boiling point and non-flammability of these coolant choices.

The paper from France had three goals; to assess the size of uncertainties in void worth computation which are likely to be irreducible, to investigate the variation among several reactor designs of sensitivity of void worth to fissile, fertile, steel, and oxygen content changes and to independently investigate the design directions suggested by the previous (USSR) paper. The assessment of uncertainties in sodium void worth predictions lead to a conclusion that uncertainties would never fall below 1$. The study to achieve negative void worth using the ductless mixed driver/blanket assembly design with high pitch to diameter ratio was able to reach negative void worth in the 500 MWth size by going to 90 cm core height with no axial blanket and use of UO$_2$ fertile pins in place of U metal ones. At the 1500 MWe core size, these measures dramatically reduced the void worth, but it remained positive at -900 pcm.

In the discussion following the papers, the following observations were expressed regarding the institutional environment regarding tradeoffs to reduce Na void worth: in the USSR, void worth likely must be non-positive; in the US it should be suitably small; in Europe and in Japan sodium void worth is not as dogmatic an issue, and the consideration of large cores so as to achieve economic feasibility is a dominant issue. It was generally agreed that the underlying goal for reactor physics activities worldwide is not sodium void worth per se but is one of overall safety and that even with a negative void worth, design to avoid a persisting power to flow mismatch is an overriding goal.

It was the consensus of the committee to retain the agenda item for the next meeting and to consider enlargement of its scope beyond the strict confines of reactor physics to include the broader issues of safety and economics goals.

1.2 Evaluation of the Uncertainty in FBR Burnup Reactivity Swing

Rapporteur: M. Salvatore

Three papers were discussed (A-970, A-989 and A-1015). Paper A-970 deals with a specific aspect of the burn-up reactivity swing in metal fueled FBRs. In fact, in this type of reactors it is important to know the reactivity effects due to the larger fuel swelling inherent to metal fuel irradiation. The paper describes the results obtained for a 300 MWe core. The reactivity effect due to fuel axial elongation was found to be of the order of 1$.

The uncertainty on such value was estimated to be of the order of 30% (-30). This uncertainty, important in the assessment of the near-zero burn-up reactivity swing requirements, has to be compared with the uncertainty on the reactivity loss due to isotope composition changes. Work done in the US on that subject indicated the possibility to greatly reducing that uncertainty using integral experiments. Paper A-989 describes the experimental programs available in France for that very purpose. The paper indicates that in particular the use of high accuracy irradiation experiments (pure
sample or characterized fuel fill irradiation) can allow a substantial reduction of uncertainties. The same paper also indicated that the use of a simple relation between the reactivity loss, the internal breeding gain and the pseudo fission product cross section, gives the possibility of a simple and unambiguous definition of the reactivity loss uncertainty and an application has been given for the cases of the PHENIX and SUPERPHENIX reactors.

During the discussion on this topic (that will be kept for one more year on the committee agenda) the US indicated that EBR-II irradiation data are expected to be used to reduce the uncertainties of the reactivity loss in the case of metal fueled cores.

The paper A-1015 from Obninsk (in Russian) describes a study of burnup reactivity loss and breeding gain for two models of the BN-1600 and BN-800. The essential changes of sensitivity coefficients to nuclear cross sections for specific characteristics are investigated throughout the whole reactor burnup period. The main sources of uncertainties in the calculations are identified.

1.3. Comparison of the Reactivity Feedback Properties of Nitride and Metal FBR Fuels

Rapporteur: K. Jirlow

During this session 4 papers were presented:

- Two papers comparing core design parameters and performance parameters from Japan (NEACRP-A-971 and A-972).
- One paper comparing core behaviour during ATWS transients for large oxide and metal fueled core (SUPERPHENIX type) from France.
- One paper from Belgium on the use of Np-237 to reduce the burnup reactivity in LMFBR cores.

NEACRP-A-971 investigates new fuels for FBR cores; i.e., metal, carbide and nitride fuels, in comparison with the oxide fuel from the viewpoint of nuclear characteristics of the core. The three new fuels have advantages for future FBR cores because of their high thermal conductivities, high contents of heavy metals, and high linear heat rates.

Comparisons of physics parameters between cores with the four kinds of fuel are summarized below:

i. Neutron spectrum is harder for the new fuel cores - metal giving the hardest and nitride and carbide giving slightly softer spectra than metal.

ii. Burnup reactivity loss is decreased to nearly zero for the advanced fuels.

iii. Breeding ratio is raised by about 0.2 for the new fuel cores compared to the oxide core.
iv. The sodium void reactivity and the Doppler coefficient for the metal fuel core are worse than those for the oxide core from safety aspects.

The second paper NEACP-A-972, also from Japan, studies the characteristics of metal, carbide and nitride fueled cores. In order to compare the performance for these designs on a comparable basis a set of fixed common conditions defining total power (1000 MWe), cycle length (1 year), residence times (3 cycles), inlet and outlet core temperatures has been adopted. Peak linear heat rates have been chosen consistent with thermal properties of the different fuels. Further reasonable restrictions on burnup reactivity swings and subassembly pressure drops are adopted. On this common basis core designs with the three types of fuel have been evaluated. The performance parameters for each design have then been compared with the parameters of a reference oxide satisfying the same common design basis. Among the important differences relative to the oxide core we note the lower core heights, the smaller fissile inventories and the higher breeding gains. Comparing the three advanced fuels to each other we find that carbide and nitride have similar parameters but that metal has slightly less favorable parameters.

The French paper NEACP-A-991 gives a general description of the dynamic behavior of two large FBR cores with oxide and metal fuel, respectively, during unprotected transients caused by loss of flow and control rod withdrawal. The two cores are comparable in the sense that the core design, power and burnup levels were those of SUPERPHENIX. The smeared fuel density in the oxide pins is about 85% and only about 70% in the metal pins. The so defined metal core is a low enriched low-rated core with high sodium void reactivity compared to a more optimized metal core.

Comparing the slow LOF transients (50 sec flow half time - total pump stop in SPX) we find that higher sodium outlet temperatures are reached in the metal core in the early phase of the transient (up to 200 sec) but after 200 sec outlet temperatures become much higher in the oxide core. In fact, the margin to boiling is small for the oxide core but large for the metal core. Assuming that the inadvertent withdrawal of control rods means a reactivity insertion of 5 pcm/sec up to 500 pcm this slow TOP gives rise to a power peak of about 165% of nominal power in the metal core but only to a peak of 118% in the oxide core. Thus a greater risk of fuel failures is indicated at slow TOPs for metal cores than for oxide cores.

The last paper NEACP-A-999 investigates the effects of adding Np-237 to the fuel in FBR cores. The incentive is to decrease the burnup reactivity by means of the relatively fast conversion of Np-237 to fissile Pu-238. Calculations for SUPERPHENIX and SNR2 confirm this effect and it is found that a 3.5% Np-237 addition to the heavy metal decreases the yearly burnup reactivity from 2.5% Δk to 1.6% for SNR2.

1.4 The Reactor Physics of Gas Cooled Thermal Reactors

Rapporteur: Y. Kaneko

The paper A-1002 reports the new plan of the critical experiments in the framework of an IAEA Coordinated Research Program on "Validation of Safety Related Reactor Physics Calculations for Low Enriched HTR's"
at the PSI PROTEUS facility. The experiments are designed to supplement the experimental data base and reduce the design and licensing uncertainties for small- and medium-sized helium-cooled high temperature reactors using low-enriched uranium (LEU) fuel pebbles. The experiments are scheduled to begin early in 1991.

The paper A-973 reports the outline of the critical experiments which are being done at VHTRC in order to verify the nuclear design accuracy of HTTR (High Temperature Engineering Test Reactor). The initial core named as the VHTRC-1 core was made by loading fuel rods which contained fuel compacts of the coated particles of the 4% enriched uranium.

The following measurements were carried out:

1. critical mass
2. reactivity worths of HTTR workup control rods and burnable poison rod
3. neutron flux distribution
4. temperature coefficient of reactivity
5. kinetic parameter $\theta_{eff}/A$

The agreements of the experimental results with the calculated ones using the SRAC code system are satisfactorily good except for $\theta_{eff}/A$.

The paper A-974 describes the feasibility design study of the annular core graphite moderator gas-cooled reactor as a high flux reactor. By choosing optimal values of the core-reflector geometrical parameters and moderator-to-fuel atomic density, a maximum thermal neutron flux of $10^{15}$ cm$^{-2}$ s$^{-1}$ can be achieved in the inner reflector.

The paper A-975 describes the analysis on the effect of fuel loading schemes on the passive safety features of the modular pebble-bed type high-temperature reactor against depressurization accident involving loss of helium forced circulation. The maximum core temperatures attained following the accident were much lower for the infinite velocity multipass scheme than for the One-Through-Then-Out (OTTO) scheme.

Discussions are made on what should be done on the gas-cooled thermal reactors. It was pointed out by some members that the experimental studies are still helpful to obtain evidence for the prediction accuracy on some safety-related core characteristics, although the nuclear data and calculational method have substantially been improved; in any case a satisfactory modelling is mandatory. The Committee agreed that the topic on GCR should remain on the agenda.

1.5 New Trends and Impact from Tight Lattice Experiments on Reactor Designs

Rapporteur: P. Wydler

Paper A-976 from Japan describes axially heterogeneous HCLWR concepts with a double flat core and multiple stacked cores. The concepts have the advantage of achieving a high conversion ratio and burnup while maintaining a negative void reactivity coefficient. The double
flat core design leads to a relatively large core diameter, but with a multiple stacked core this diameter can be reduced. Power peaking effects in these types of cores can be reduced by admixing gadolinium to the blankets.

Results of a study of the neutronic characteristics of a Th/U-233 fueled HCLWR with an axially heterogeneous core are reported in another Japanese paper, A-979. The main results are that the void reactivity coefficient is always negative and the excess reactivity can be controlled using B-10 enriched boric acid as a chemical shim. With a volumetric moderator-to-fuel ratio of 0.4 an average conversion ratio of 0.95 can be achieved. Higher conversion ratios appear to be feasible and, as was mentioned in the discussion, such reactors would, therefore, have a good performance as client reactors for Pu-Th-fueled fast reactors.

Paper A-977 reports results from critical experiments with plutonium-fueled cores in FCA at JAERI, Japan. The measured data (\(k_{\text{eff}}\), \(k_{\infty}\), reactivity worths and reaction rate ratios) are analyzed using the SRAC code system and the JENDL-2 data file. The calculations agree fairly well with the measurements except for the reaction rate ratios and \(k_{\infty}\) in the cores with higher void fractions. For a moderator void fraction of 95%, \(k_{\infty}\) is overestimated by about 2%. To resolve these discrepancies, further analyses are necessary.

Tight pitch lattice experiments were also performed at the Kyoto University Critical Assembly (KUCA) in Japan. In these experiments natural uranium plates were located adjacent to enriched uranium-aluminum alloy plates. Paper A-978 is mainly concerned with assessing the validity of the Dancoff factor method used in the SRAC code system for this type of heterogeneous cell. It was found that the performance of the Dancoff factor method varies with cell geometry but that a collision probability method due to Tone gives good results for all cell patterns used in the KUCA experiments.

Paper A-1003 from Switzerland discusses the transferability of experimental results from the PROTEUS-HCLWR cores to tight-lattice PWR design using a general perturbation theory approach. In particular, the paper presents sensitivity profiles for a PROTEUS model which represents the driver regions as a fixed source. This model has some decided advantages for sensitivity studies of driven critical systems. The analysis confirms that the neutron spectrum conditions at the centre of the test zone are close to fundamental mode conditions and that the experimental data is transferable to power reactor design.

In France an extensive experimental programme has been undertaken to validate data and methods for HCLWR design studies. Paper A-990 reports experimental and theoretical results from the ERASME and the associated MORGANE programme. The ERASME programme involves comprehensive measurements of integral parameters for plutonium-fueled test lattices at the centre of the EOLE driver core. The measurements agree satisfactorily with calculations based on the French assembly code APOLLO and the CEA-86 library. The results of a sensitivity analysis will be used to further improve the design accuracy. In the MORGANE experiments the captures due to the fission products were measured globally by oscillating irradiated samples in a tight lattice at the centre of the experimental reactor MINERVE. The analysis showed that the reactivity loss due to the fission products can be
From the now available information, the Committee drew the conclusion that plutonium-fueled High Conversion Light Water Reactors can achieve conversion ratios between 0.8 and 0.9 depending on the detailed design.

1.6 Miscellaneous Topics

Rapporteur: R. T. Jones

Of the above papers, the first three were from Germany, A-1031 was from Canada, and the final three were from the USSR. A wide range of topics was covered.

Paper A-996 describes the calculation of cross-sections for slow neutron scattering by liquid hydrogen and deuterium. The cross-sections have been prepared in ENDF-6 format which makes them immediately useful in existing calculational schemes. Members of the Committee expressed interest in obtaining the ENDF-6 format cross-sections for use in cold-source design projects.

The second German paper (A-997) describes a comparison of JEF-1 with ENDF/B-IV and V in calculations of two cycles of a PWR. With regard to reactivity the JEF-1 results were systematically higher than those from ENDF-B-IV/V. For power distributions, the results from the two data sets agreed within 2%.

The final German paper (A-998) describes calculational methods to treat cavities in three-dimensional diffusion calculations, particularly for the space between the core and top reflector, through which the control rods pass, in pebble bed HTGRs. Transport theory methods for treating cavities in three dimensions are discussed.

The paper from Canada (A-1031) concerns a comparison of measured reaction-rate ratios in (U-233,Th)O2 fuel bundles in a CANDU-type lattice with calculations by WIMS-AECL using a data library derived from ENDF-B/V. The Th-232 capture/U-233 fission ratio was over-estimated by the calculation by up to 4%.

Paper A-1037 from the USSR is a review of activities in the area of data libraries and processing codes for fast-reactor physics. Names of codes and libraries are given.

The second paper from the USSR (A-1038) describes measurement and calculation of cores in a series of critical assemblies designated BSF-52. Three cores are being studied: a reference single zone core and two radially heterogeneous cores. Comparison of measured k-effective, fission rates, and control rod worths with diffusion theory calculations are given. The most obvious problems are in the calculation of U238 fission rates in the radially heterogeneous cores.

The final paper (A-1013), in Russian, describes 1-D calculations of reaction-rate ratios measured in the BN-350 reactor. The idea is to use this to compare different data libraries and possibly as a basis for data adjustment. In the results presented the largest discrepancy is for the fission ratio Pu-241/U-235.
2. **Topics Carried Over from Previous Meeting**

2.1 **Engineering and Physics Aspects of Transuranium Burning by Reactors and Accelerators**

Rapporteur: P. Hemming  
Papers: A980, 981, 982, 1006

These papers describe options for converting long lived isotopes into short lived isotopes via fission, capture, and spallation processes in reactors or accelerators designed to produce useful energy and more acceptable waste forms.

In paper 980, the transmutation rates of TRU elements is calculated for a metal fueled fast breeder reactor. For TRU fuel loading of 15%, it is noted that an LMR can burn the minor actinides produced by 19 LWRs of comparable size. Retention of 15% rare earths in the fuel reduces the TRU burnup efficiency by about 5%.

Paper 981 reports on the design of an accelerator driven spallation system which can transmute about 250 kg of TRU elements annually at a thermal power of 769 MW and a proton beam of 20 mA and 1.5 GeV. R&D requirements include target design with Na or Pb/Bi cooling and development of advanced accelerator technology. Development of the first 10 MeV section of such an accelerator has been started. Spallation integral experiments are also planned using lead and uranium targets.

Paper 982 reports systematic studies of TRU transmutation rate and burnup reactivity variations in a 1000 MWE LMFBR system. It is found that use of MOX fuel containing 5% TRU can transmute the TRU inventory produced by 5 LWRs of comparable size. It is also noted that TRU elements are effective in reducing burnup reactivity variations and can thus reduce concerns about control rod runout margins.

Paper 1006 analyzes the core physics aspects of two LMR fuel cycle concepts - one an actinide storing system, the other an actinide burner. The integral fast reactor concept is noted to provide considerable flexibility with regard to increasing, maintaining, or decreasing the actinide inventory while concurrently producing electric power.

In the general discussion of these concepts, it was noted that the IFR operation is tolerant of the rare earth equilibrium content throughout the fuel cycle and pyroprocessing can be designed to provide fission product waste with very low actinide content. Development of the latter capability is a part of IFR processing studies in progress at ANL.

A major issue regarding use of LWR spent fuel is whether one can reduce the costs and improve the efficiency of recovering fissile actinides in forms readily useable in the IFR fuel cycle. Several processes are being considered in the U.S. which require various degrees of development and testing. Another issue is whether partitioning of the fission product waste stream provides cost savings in geological storage and/or other types of storage systems which could be specially developed for the partitioned products.
It was noted that the physics aspects of actinide burnup in nuclear reactors have been of interest to the NEACRP for many years and appear reasonably well understood. Because several ongoing actinide burnup studies were identified which could impact future fuel cycle and reactor design activities, the NEACRP indicated a desire to review this topic again at its next meeting.

2.2 Gamma Energy Deposition

No further work was presented under this topic.

2.3 Identification of Factors Affecting Local Stability in LWRs and HWRs

Rapporteur: M. Darrouzet

This session gathers four papers concerning instability phenomena in BWRs (A-985, 1018, 1019, and 1022), a paper concerning the analysis of azimuthal irregularities in the flux distribution on a PWR (A-994), and a paper on the improvement of TRIAD 3 code for NRU reactor simulation (A-1032).

The first paper (A-985) studies the local and global oscillations in the Italian BWRs in particular the out-of-phase oscillations observed (for the first time) in the Caorso reactor. This problem of out-of-phase regional oscillations is analyzed and discussed in some detail. On the basis of an analysis of the neutronic and process signals recorded during the event, it is suggested that a global contribution to the power oscillations may derive from oscillatory motions of the coolant in the pressure vessel lower plenum. As it looks as though some mechanisms leading to out-of-phase core power oscillations are not fully understood or are difficult to simulate accurately by standard calculational tools, work on the subject is continuing.

The second paper (A-1018) concerns the measurements performed in Oskarshamn 3. During natural circulation tests, decay ratios, natural frequencies and oscillatory modes have been measured by the noise analysis and pressure perturbation. At about 50% of the power, the core instability limit was reached and limit-cycle flux oscillations developed and were recorded. The oscillations were "out-of-phase" usually across a core diagonal. Channel flow noise measurements have been evaluated in order to compare the influence of different fuel types on the stability margin.

Paper A-1019 concerns stability investigations performed at Forsmark 1 (BWR). A detailed analysis of phenomena is given and a comparison between results of Forsmark 1 and Forsmark 2 is reported. The high value of the decay ratio of Forsmark 1 (that imply that the core cannot suppress oscillations fast enough and small perturbations can cause scram) is, may be, due to the simultaneous presence of different fuel (8x8, 9x9). This paper underlines the importance of having an on-line stability surveillance system.
Paper A-1022 examines a TVO-1 oscillation incident which took place in 1987 and analysis possible remedies. Calculations have been performed with Ramona-3B and TRAB BWR transient analysis codes and the consequences and the actions in order to limit the problems in the future are discussed. Sensitivity studies show for example that decreasing the fuel gap conductance has a destabilising effect.

Paper A-994 concerns PWRs. During the first core start-up of the PWR reactor Paluel, azimuthal irregularities had been observed. This phenomenon has been confirmed by measures using control rod clusters. After inquiries, no error in the loading was found. Then neutronic analysis have been performed and have shown that the deviation was due to a variation of the water blade between baffle and fuel edge in two quadrants. A detailed mechanical analysis has shown that the blade was due to the bowing of the rods related to the fabrication method. Actions have been conducted in order to correct these phenomena.

The simulation code system (A-1032 paper) used for NRU research reactor at Chalk River Nuclear Laboratories was sufficiently accurate for the reactor many years but results have been less satisfactory as the reactor has become more and more heterogeneous. Recently various options for improving the code were studied, including modelling the axial dimension, an improved representation of the lattice split and discontinuity factors. The new modular code system is now in a commissioning phase and the results are ameliorated.

After discussion, it was felt that the topic should be kept on the committee agenda, since possible new contributions can come from the US and the Scandinavian countries.

2.4 Physics Methods in Fuel Accountability

Rapporteur: Leo LeSage
Papers: A-992, A-1010

Paper A-992 from France reviewed both active and passive methods for nondestructive characterization of nuclear material. The emphasis was on the newer active methods under development. The paper addresses methods for characterization of spent reactor fuel, measurement of fissile material concentrations, and measurement of nuclear waste material. The active methods all utilize some type of neutron source (e.g. 252 Cf or 14 MeV pulsed source), and the sensitivity limits associated with each of the active methods are discussed. Paper A-1010 (from the US) discussed the application of the isotopic correlation technique (ICT) developed at ANL to the ESARDA Reprocessing Input Verification (RIV) Working Group's benchmark exercise. Data from the COGEMA facility for the Obrigheim reactor assemblies was used in the exercise. The results indicated that anomalies intentionally introduced into the COGEMA data were detected and their magnitudes were estimated to reasonably good accuracies. These results tend to confirm the applicability of ICT methods to reprocessing input verification.

Discussion on this agenda topic agreed that the methods described were not conceptually new; however, the application of these methods to new problems is becoming increasingly important. Some recent specific accountability problems were discussed:
Evaluation of the fissile content of drums filled with fabrication waste material.

Evaluation of fuel assembly shipping containers (either new or spent fuel) especially with regard to verifying the number of assemblies in the container.

It was agreed to keep this topic on the agenda for next year's meeting, with a focus on the applications of these methods to real accountability problems.

3. Benchmarks

3.1 Radiation Shielding Benchmark Data Base

Rapporteur: D. T. Ingersoll
Papers: A-1020, A-1033

An effort had been initiated at the prior NEACRP meeting to develop a data base containing existing shielding benchmark experiments in a computer-readable form. The purpose of the project is to provide archival of this important data resource and to facilitate the use of the data for validation of future shielding methods and data. As the initial step in the effort, A. K. McCracken prepared a proposal for the structure and operation of the data base (summarized in A-1020). Several committee members felt that his proposal did not properly address the original purpose of the data base and was overly ambitious in its scope. It was decided that D. Ingersoll would review McCracken's detailed proposal and prepare an alternative proposal. This new proposal will be distributed to other NEACRP members for comment and will be presented at the next meeting in 1990.

Paper A-1033 provided the draft of a new shielding benchmark problem prepared by P. Miller. The benchmark resulted from difficulties related to the energy weighting of multi-group constants as identified in an earlier NEACRP sodium/stainless steel benchmark problem. E. Sartori will distribute the draft benchmark to interested members. Comments are to be forwarded to P. Miller so that a revised benchmark specification can be prepared for the next NEACRP meeting.

3.2 Criticality of Fuel Undergoing Dissolution

Rapporteur: M. Salvatores
Papers: A-1028, A-1029, and A-1030

Recent results on the NEACRP standard problem exercise on criticality codes for dissolving fissile oxides in acids have been reported (paper A-1028). The results for the calculational benchmarks relevant to fuel dissolvers had been found to be highly discrepant and a detailed physical analysis has been undertaken to understand the origin of the wide spread. Paper A-1029 gives the details of the physical analysis that has been performed at CEA and which allowed to understand the spread among the results as a consequence of unsatisfactory treatment of the double heterogeneity effect on the self-shielding. Point Monte Carlo calculations performed at CEA, UK and Stuttgart (paper A-1030), confirmed the results of the physical
analysis indicated in A-1029. The working group will meet in May 1990 to discuss the final report. Future work should address irradiated fuel and in particular fission product treatment in criticality studies. This committee should try to make available experimental data in that field to the working group.

3.3 Shielding of Transport Casks

Rapporteur: D. T. Ingersoll
Papers: A-1004 (A-961)

Paper A-1004 provided an interim progress report on the intercomparison of shielding codes for transport casks. Solutions to the six computational benchmark problems are still being received and a final report is expected at the next NEACRP meeting. A recent contribution from PSI, Switzerland, supported earlier results which indicated significant azimuthal variation of the surface dose correlated with the presence of polyethylene shield rods in one of the problems.

Paper A-961 (distributed prior to the meeting) describes a detailed experimental transport cask benchmark. Due to the complexity of the TN-12 shipping cask specifications, solutions for this benchmark are not expected for some time.

3.4 Noise Analysis

Rapporteur: E. Sartori
Paper: 968

The specification for an artificial noise analysis benchmark was prepared and distributed to potential participants. The aim is to improve methods for detection of anomalies in noise signals.

Fifteen groups from ten countries expressed the intention to participate.

Analog and/or digital tapes were distributed to participate including a complete description of the tasks to be executed.

A final report, the committee recommends, should be presented at the next NEACRP meeting.

3.5 HCLWR Benchmarks

Rapporteur: M. J. Halsall
Paper: A-1024

E. Sartori reported progress on the analysis of the PROTBUS-LWHCR Phase 1 double cell experiment, cores 1-6, and also on a revision of the earlier theoretical problems; this revision consisted of a specification of the fission product compositions to remove uncertainties associated with their calculation. The JAERI-VIM point Monte Carlo results are considered to be a good standard for comparison.
A number of submissions had been withdrawn from the earlier input - these were codes not regarded as state-of-the-art for intermediate spectrum applications - and as a consequence the original spread of \( k \)-values of around 4\% had been reduced to 1\%.

A report by Bernnat et. al., will be presented at PHYSOR in Marseille.

Discussion centered around the desirability of extending the benchmarks to calculate temperature coefficients. The conclusion was that the present results should be fully documented. It was not considered worthwhile to await MCNP results from PSI as data generation and code modifications were necessary to produce a useful accuracy.

### 3.6 Measurement of Tritium Production Rates

**Rapporteur:** Y. Kaneko  
**Papers:** A-1021 and A-1007

The paper A-1021 reports the international comparison on experimental techniques for the measurement of tritium production rates in fusion blanket neutronics experiments. Two 14 MeV neutron source facilities were used for this purpose, FNS at JAERI-Japan and LOTUS at EPEL-Switzerland. Eight teams from six countries participated in this program which was undertaken to evaluate the state of accuracy expected from current measuring methods. A simple-geometry experimental assembly composed of breeder material served as the test bed at each facility, in which Li-containing samples from the participants were irradiated in an exactly identical neutron field. The tritium amounts produced in the samples were determined by the participants using their characteristic ways by using liquid scintillation counting method. Tritiated water of blind concentration was also distributed and measured to make a common reference. Although not all the data have lined up, interim results revealed differences among the data beyond the target accuracy expected for fusion neutronics experiments. Further efforts are required to find out the causes of the discrepancies, and then assess the accuracy in determining tritium production in very small amounts.

The paper A-1007 reports the US contribution to the international comparison. The US assaying method differs somewhat from those of the other countries. Thin metallic foils are encapsulated in aluminum. After irradiation, the capsules are digested in a furnace, then sparged with carrier hydrogen to be converged into THD and collected by cold-trapping. Direct redistillation from the cold-trap to a counting vial, to which a scintillator cocktail is then added allows typical yields of 97\% that can be directly measured. Concerning future needs, accurate tritium-breeding measurements in ITER blanket modules will be more difficult for samples that rely on wet-chemistry extraction.

Next step of the international comparison is approved as follows:

**Action 1:** JAERI and LOTUS ask some questions to each participant in order to find the causes for the dispersion of the measured results.
Action 2: The certified HT0 standard with high accuracy such as NBS tritiated water standard should be delivered for analysis.

Action 3: After the causes of the dispersion are found through Actions 1 and 2, irradiation experiments should be repeated.

3.7 Three-Dimensional Neutron Transport Benchmark

Rapporteur: H.W. Rief
Papers: 983, 1008, 1023, 1017, 1016, 1034 and 1036

Takeda and workers have proposed four 3-D neutron transport eigenvalue problems to test computational tools presently being used. In all four cases it was requested to deal with strong geometrical heterogeneities.

Fifteen groups expressed their intentions to participate in the exercise. Out of these ten groups submitted their contributions in time for the deadline of Sept. 30, 1989. Two NEACRP members promised that their missing contributions would be submitted soon.

A brief analysis of some of the solutions revealed that quite a variety of methods had been employed, such as flux-synthesis, 3-D SN, nodal transport and diverse Monte Carlo Codes.

3.8 Validation of Delayed Neutron Data

Rapporteur: M. Salvatores
Paper: A-987

Following the discussions held at previous meetings, the CEA has prepared a proposal (paper A-987) for a series of experimental benchmarks to be performed in MASURCA at Cadarache, starting mid 1991. The benchmark is intended to provide high accuracy experimental data for $\beta\text{eff}$ in three critical assemblies (two PuO$_2$/UO$_2$ fueled and one UO$_2$ fueled) to investigate in a systematic way major isotope contributions to $\beta\text{eff}$. The European partners of the Fast Reactor Cooperation (FRG, UK and France) will participate but participants from other countries are welcome, both to participate to the experimental work or to the analysis for the experiments. This benchmark activity is coupled to a priority task defined in the frame of the international cooperation in data evaluation (see A.5.1), for differential delayed neutron data assessment and validation.

Thermal reactor $\beta\text{eff}$ measurement which are available (Japan) or that will become available (France), will complement the experimental data base for data validation.

3.9 Thermal Reactor Fission Product Benchmark

Rapporteur: M. J. Halsall
Paper: A-995

This paper was presented as a final report with only minor modifications from last year (Paper A-921).
The problem was to calculate k-infinity as a function of burnup for homogeneous mixtures of hydrogen and U-235 or Pu-239. Fission product discrepancies were difficult to assess, being masked by surprisingly large discrepancies in the clean start-of-life k-infinities.

The paper contains extensive details of individual fission product and global fission product poisoning, with tables and graphs of data from 10 submissions. The following conclusions were drawn:

- the discrepancies were much larger than anticipated, but it is difficult to separate the various possible causes.
- the benchmark was not an ideal way to compare fission product data. A better approach might have been to ask submissors simply to extract appropriate spectrum averaged cross sections from their libraries.
- individual submissors would draw their own conclusion about the accuracy and adequacy of the methods they had used.

3.10 Radial C/E Trends in Large FBRs

Rapporteur: Leo LeSage
Paper: A-1009

Paper A-1009, prepared jointly by representatives from France and the US, defined a proposed benchmark problem to evaluate the radial dependence of control rod C/E values in large FBRs. The benchmark will be based on ZPPR Assembly 13A. As reported at previous NEACRP meetings, calculations by US and Japanese analysts of large heterogeneous cores in ZPPR have indicated a radial dependence of control rod C/E values, while this dependence has not been observed in other analyses (e.g., French analysis of SUPERPHENIX). The proposed benchmark was approved. Peter Collins of the US (ANL-West) will be asked to distribute the detailed assembly specification and to compile the results. The specifications should be distributed in early 1990, results should be submitted by June 1990, and a summary report prepared for the 1990 meeting of the NEACRP. Additional phases of this study will be considered after results of this initial phase have been evaluated.
ANNEX 1

LIST OF PARTICIPANTS

Delegates

For CANADA

McDONNELL, Dr. Frank N. Scientific Secretary

For JAPAN

KANEKO, Dr. Yoshihiko SHIRAKATA, Dr. Keisho Chairman

For the UNITED STATES

LeSAGE, Dr. Leo HERMMIG, Dr. Phil INGERSOLL, Dr. Daniel T.

For the countries of the European Communities and the European Commission acting together

KUESTER, Dr. Heinz (F. R. of Germany) DARROUZET, M. Michel (France) SALVATORES, Dr. iMassimo (France) MARTINELLI, Dr. Renato (Italy) D’HONT, M. Pierre J. (Belgium) CARO, Dr. Rafael (Spain) HALSALL, Dr. Michael J. (United Kingdom) RIEF, Dr. Herbert W. (CEC)

For the other European countries of the OECD

JIRLOW, Mr. Class M. (Sweden) WYDLER, Dr. Peter (Switzerland) Vice-Chairman

Nuclear Energy Agency

SARTORI, Dr. Enrico Secretariat

Observers

CRIJNS, Mr. Martin (IAEA Secretariat) GRUPPELAAR, Dr. Harm (Netherlands) JONES, Dr. Richard T. (Canada) MATVEENKO, Dr. Igor (IAEA) SMITH, Dr. A. B. (NEANDC) SLESAREV, Dr. Igor (IAEA) WADE, Dr. David C. (USA)

Apology for absence was received from Dr. G. S. Robinson (Australia). Following an established rotation Dr. Jirlow (Sweden) represented also Denmark, Norway and Finland. The delegate for Switzerland, Dr. Wydler, also represented Austria.

On October 9th, the following members of the NEANDC participated in the joint NEACRP/NEANDC session: Dr. C. Dunford (USA), Dr. R. McKnight (USA), Mr. C. Nordborg (NEA), Mr. J.L. Rowlands (UK), Dr. M. Sowerby (UK), and Dr. Y. Kikuchi (Japan)