

NUCLEAR ENERGY AGENCY COMMITTEE ON REACTOR PHYSICS

SUMMARY RECORD OF THE NINETEENTH MEETING (TECHNICAL SESSIONS)

CHALK RIVER, CANADA
21st-25th June 1976

Compiled by
BOB RICHMOND

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TECHNICAL SESSIONS

1. TOPICS CARRIED OVER FROM PREVIOUS MEETINGS

1.1. Short-lived fission product decay properties

Campbell introduced the paper NEACRP-A-257 which summarized the current position of fission product decay data.

Recent experiments in this field included measurements of gamma heating in U235 by Dickens (ORNL). Calculations using ENDF/B-4 data were in good agreement with these measurements for cooling times up to about 1000 seconds. The new experimental results were however discrepant with the results of earlier measurements made at short cooling times. Both the new experiments and the ENDF/B-4 data gave values for the U235 gamma heating that were lower than the ANS standard (by up to 10%) and the evaluation of Perry et al (by up to 20 %) for cooling times less than 100 secs and infinite irradiation. It therefore seemed desirable to make a careful evaluation of the theoretical and experimental data before full reliance could be given to the more recent results.

In the beta heating area some recent measurements had been made at Winfrith of total beta heating from fast fission of U235 and Pu239. Following an irradiation of 10^5 secs in ZEBRA the decay had been followed for a period of 18 months and this work was continuing. A preliminary comparison of the results with those of McNair et al and with calculations based on the data of Tobias showed that, for U235, agreement with calculation was good up to about 2×10^6 secs but that, at longer cooling times, the experimental values were higher by about 40%. The new results were rather higher than those of McNair and also higher (by 7-10%)

than the evaluation of Perry et al. For Pu239 the calculated values were greater than the experimental ones by about 20 % for cooling times from 20 secs to 500 secs and about 10 % from 10^3 to 10^4 secs. Above 2×10^4 secs the agreement was good.

Calorimetric measurements of total decay heat for thermal fission of Pu239 had been made by Fiche at Fontenay-aux-Roses. For a single fission C/E was 1.13 for a cooling time of 100 secs and about 0.9 for $10^3 - 10^5$ secs. The calculated values were derived from the data library of Devillers et al.

In the data library area much effort had been devoted in many countries to the improvement of decay data libraries for fission products. A large amount of new data had been produced and the speed with which it could be processed could be greatly improved by international collaboration in this field. The general adoption of the ENDF/B-4 format in order to facilitate data comparison was encouraged.

Calculation of sensitivities had been made by Devillers et al. for the French library and by Schmittroth for ENDF/B-4. For short cooling times ($10 - 10^5$ sec) and a burst of irradiation the estimated uncertainties were between about 15% (1 SD) and 5 % for U235 and rather higher for Pu239. Calculations of total decay heat made using different libraries differed by less than these limits but the differences were greater for the separate beta and gamma contributions.

In a general discussion on this topic it became apparent that further experimental and evaluation work was in progress in a number of member countries and it was felt that co-ordination was needed in order to avoid duplication of effort. Members agreed to send to the Secretariat by 1st October 1976 statements on the work in this area being planned in their respective

countries. These statements would include the results of experimental work recently performed. The Secretariat agreed to distribute copies of these statements to members by 15th October.

Bustraan agreed to remind the IAEA of the usefulness of the benchmark exercise on fission product decay data proposed at the Bologna meeting.

1.2. Intercomparison of fission chamber standards

Campbell introduced the paper NEACRP-A-262 which gave a brief history of intercalibration exercises using fission chambers. These involved:

- (a) Intercomparisons, carried out about ten years ago, using parallel plate chambers from Winfrith, Idaho and Aldermaston containing U233, U235 and Pu239. The results were consistent to an accuracy of about 1% and mass calibrations were accurate to 1-2 %.
- (b) Subsequent work on ZEBRA to establish the technique of absolute fission rate measurement.
- (c) A comparison of the ZEBRA parallel plate chambers with cylindrical chambers from MASURCA. This was rather inconclusive and gave an unresolved discrepancy of 4-5 % for the U238/U235 fission ratio.
- (d) A further comparison, made as part of the "Scherzo-556" benchmark exercise, which involved measurements on a standard, unit k-infinity lattice at Winfrith, Karlsruhe, Fontenay and Cadarache. The measured values of the U238/U235 fission ratio were in good agreement (standard deviation 0.8 %).
- (e) A check on the validity of low geometry alpha-counting techniques in which U235 deposits were exchanged between ZEBRA

and Geel and between ZEBRA and Würenlingen. Both cases gave agreement within 0.3 % for the deduced alpha-disintegration rates.

(f) An indirect comparison involving the NISUS facility at London University and the $\Sigma\Sigma$ Group at Mol from which a satisfactory agreement between ZEBRA and NBS calibrations of Pu239 and U238 deposits could be inferred.

The paper concluded that, for Pu239, U238 and U235 the results were consistent within a standard deviation of one percent and that there was no particular incentive to carry out further work on these nuclides. Further work would, however, be needed in the future to extend the intercomparisons to the higher isotopes of plutonium and to americium and curium.

In discussion it was generally agreed that the position concerning Pu239, U238 and U235 fission rates was satisfactory but that further work was required in connection with the fission rates of americium, curium, the higher plutonium isotopes and Pu238 and also the U238 capture rate. There was, however, no urgent need for this work. Campbell agreed to prepare a proposal for setting up a "Scherzo-556" core for intercomparison of these reaction rates. This work could then be undertaken when a suitable opportunity arose.

1.3. Summary of NEACRP views on fast reactor breeding assessment.

Introducing the paper NEACRP-A-263 Barré pointed out that it summarized the breeding situation as it was in 1975 and did not take into account developments during the current year.

For an oxide-fuelled LMFBR in the "classical" two-zone design the global breeding gain (GBG) ranged typically from 0.10 at

250 MWe to 0.25 at 2000 MWe. (The global breeding gain was defined as the net equivalent Pu239 balance in atoms in the reactor per fission in the reactor.) The current GBG target accuracy was ± 0.03 . The uncertainties in the basic nuclear data would lead to errors in the calculated GBG of the order of ± 0.1 for a typical 1200 MWe design. This was well outside the target accuracy.

Improvements in the accuracy of prediction of the GBG were made via integral measurements and three approaches were used:

- (i) The mock-up approach in which a critical assembly was constructed to represent closely a power reactor at a given stage of life and reaction rate measurements were made which allowed the GBG to be determined. A typical result, obtained on a mock-up of the Clinch River reactor in ZPPR4 Assembly 4, gave a C/E value for the breeding ratio of 1.076 ± 0.041 for calculations performed using ENDF/B-3 data. The mock-up approach had the problems that it was difficult to produce a really representative mock-up and that the C/E values corresponded only to a specific configuration.
- (ii) Parametric analysis of the neutron balance and data adjustment. In this approach neutron balance parameters were measured in a wide range of critical assembly configurations and the results were used to adjust the basic nuclear data so as to improve the prediction of the integral parameters. This approach had the advantage that it could be applied to a wide range of power reactor lattices. The accuracy of GBG prediction varied from ± 0.03 at start of life to ± 0.04 at end of life.
- (iii) Analysis of the isotopic composition of fuels irradiated in power reactors. It was expected that measurements of this kind in progress on PHENIX would reduce the error on the end of life GBG to ± 0.03 .

Methods of improving breeding performance included the use of carbide or nitride fuels which were said to give breeding ratios of up to 1.5 and the gas-cooled breeder which would give a breeding ratio of about 1.35. The use of the mixed oxide LMFBR in the heterogeneous version with internal breeding zones could lead to doubling times of 11 years compared with 22 years in the classical version.

After discussion it was agreed that members would send comments on the paper to Barré not later than 1st September and that Barré would then send the final version of the paper to the Secretariat who would reproduce and distribute it as an NEACRP-U document.

1.4. Neutronic problems in fusion reactors

Italy

Farinelli introduced a series of reports.

The paper NEACRP-A-265 summarized the geometrical model and material compositions used in benchmark calculations for the JET machine operating with a DT plasma. The calculations, which were carried out at Jülich, Harwell and Casaccia, were intended to determine the radiation dose rates resulting from neutron activation of the various components of the assembly. A simple infinite cylindrical model of JET was used.

The results of the calculations were given in the paper NEACRP-A-282 (Chapter V of the document EUR-JET-R5). These showed good agreement between the dose rate values predicted by the three establishments. The predicted dose rates indicated that maintenance or repair work on JET would have to be carried out using remote handling techniques.

The scope of this work was then extended to include 2D transport calculations (using DOT) at Casaccia and 3D Monte Carlo calculations at Jülich. The Casaccia work was described in the paper NEACRP-A-266. The calculations were carried out in (R,Z) geometry using the S_8/P_3 approximation. The calculated neutron flux distributions were very similar to those given by the earlier 1D calculations. This was believed to be the first application of a 2D transport code to a toroidal geometry.

The paper NEACRP-A-271 gave a brief account of a sensitivity analysis related to the 1D benchmark model of JET. Transport calculations were performed in the $S_{16}P_3$ approximation with 29 energy groups and the sensitivity calculations were based on generalized perturbation theory. Sensitivity coefficients for selected reactions were quoted. The work was still in progress.

The paper NEACRP-A-268 examined the possibility of using a plasma focus machine as a neutron source for cross-section measurements by the activation method. It was found that the Frascati machine investigated could produce a neutron flux of 4×10^{17} n/cm²/sec using a DT mixture. It was hoped to begin activation measurements in 6-12 months' time.

The paper NEACRP-A-269 was concerned with the prospects for hybrid fusion-fission systems. These systems made use of the fact that fusion was neutron-rich while fission was energy-rich. By surrounding a fusion zone by a fission blanket containing uranium and plutonium (and also lithium for tritium breeding) it was possible in principle, to obtain useful power and also to breed fusion and fission fuel. This concept made it possible in principle, to produce a positive energy balance by coupling a sub-Lawson plasma with a subcritical fission blanket.

Farinelli expressed the view that the possible incentives for such systems were not such as to justify further investigation of them at the present time. He discussed this in terms of the advantages generally attributed to these systems, i.e.:

- (a) 'They have safety and environmental characteristics better than those of pure fission or pure fusion systems'.—There would certainly be no hazard from a reactivity excursion but the main problem in the fission reactor case was, in fact, the loss of coolant accident and this would also be the case in the hybrid system. The safety problems might well be greater in the case of the hybrid.
- (b) 'The hybrid systems breed new fissile fuel at a greater rate than pure fission systems'.—This would be of advantage only if fission reactors could not be produced with doubling times equal to that of electricity demand. In fact the required doubling times could be obtained by the development of, for example, a gas-cooled fast reactor, which would surely involve less problems than the development of a hybrid fission-fusion system.
- (c) 'The hybrid systems can be used for burning actinides produced in fission systems'.—This was not a valid point since it currently appeared that neutron burning of actinides was worse, from an environmental point of view, than their disposal in appropriate geological formations. In any case the main problem concerning actinide burning was that the inefficiency of re-processing was such that the wastes generated made the advantages of neutron burning extremely questionable.
- (d) 'The hybrid systems could anticipate the advent of pure fusion systems by relaxing some of the plasma physics and technological requirements'. — This was also a questionable point since it was likely that Lawson's criteria would be relaxed by a factor of only about five. The question was whether this relatively

small gain was not totally outweighed by the considerable problems involved in the development of the hybrid system.

In a discussion on this latter paper Askew expressed general agreement with Farinelli's view in connection with hybrid systems intended to produce power and also to breed fission and fusion fuel. He felt, however, that there were prospects for a laser fusion device intended primarily as a plutonium producer and therefore having a wholly U238 blanket. This would be fairly frequently reprocessed so as to maintain a low plutonium content.

Euratom

Farinelli introduced the paper NEACRP-A-270 which described studies of the radioactivity and afterheat of FINTOR as a function of time after shut-down. ENDF/B-4 data was used in conjunction with the code ANISN and POPOP. A comparison was made with the corresponding quantities for a fission reactor and this indicated that, if tritium were taken into account, the radioactivity per kWt in the fusion reactor was greater than that in the fission reactor for cooling times between 6 months and 20 years.

Spain

Velarde presented the paper NEACRP-L-166 which described theoretical studies of the neutronics of laser-driven fission-fusion systems. The case considered was a micropellet consisting of 0.2 g plutonium (with a Pu239 content of 95%) surrounded by a layer of Li6D. This would be compressed to a supercritical state by means of a pulse from a 5 MJ laser which would increase the density by a factor of about 200. The resultant micro-explosion would produce a burst of about 10^{20} neutrons. The confinement time would be about one nano-

second. The energy produced by fission would be about 10^9 J per pulse.

Japan

Hirota introduced the paper NEACRP-L-165 which described the measurement of radiation heating in a lithium assembly simulating the blanket of a fusion reactor. The assembly was driven by a source of DT neutrons. The combined use of Li6F and Li7F thermoluminescent detectors allowed the greater part of the radiation heating to be measured. The experimental errors were in the region of ± 35 %.

The calculated responses of the thermoluminescent detectors were obtained using the code system RADHEAT. The 42-group neutron transport cross-sections were generated by SUPERTOG from the ENDF/B-3 library and the 21-group gamma-ray transport cross-sections were calculated by GAMLEG-JR. The gamma-ray production cross-sections were given by POPOP4. Neutron and gamma-ray fluxes were calculated using ANISN in the P_5S_8 approximation and the group responses of the thermoluminescent detectors were calculated from the neutron and gamma response functions. Large errors in the calculated gamma-ray fluxes were expected because of the inadequacy of the data in the POPOP4 library.

In the case of the bare lithium assembly the measured and calculated values of the detector responses were in reasonable agreement in relation to the experimental error and the accuracy of the gamma-ray data. For the assembly with a graphite reflector the measured and calculated values were in fairly good agreement for the Li7F detector but there was a large discrepancy in the case of the Li6F detector. This could be attributed to an error in the calculation

of the low energy neutron flux arising from an inaccurate treatment of the anisotropy of inelastically scattered neutrons.

Netherlands

Bustraan presented the paper NEACRP-A-267 which gave an account of blanket design studies for the European Tokomak. In the chosen design beryllium was used for neutron multiplication and moderation and the amount of lithium required for tritium breeding was reduced by using a thin breeding zone operating in a soft neutron spectrum. In this case only the $\text{Li6}(n,\alpha)\text{t}$ reaction made a significant contribution to the tritium breeding ratio. Calculations using ENDF/B-3 data gave a lower breeding ratio than that predicted by calculations based on ENDF/B-1. This was because the more recent set predicted a lower neutron multiplication from the $\text{Be}(n,2n)$ reaction. The studies showed, however, that the required breeding ratio of 1.20 could be achieved by appropriate positioning of the lithium within the beryllium zone of the blanket and by the use of a Li6 enrichment of 90 %. A major drawback of the design was the large beryllium inventory of about 10^6 kg. The study indicated a need for more accurate data on the $\text{Be}(n,2n)$ reaction.

USA

Maienschein introduced the paper NEACRP-A-264 which gave a review of fusion reactor neutronics in the US.

In the nuclear data area a US CTR Neutronics Working Group had identified maximum priority requirements for the measurement of secondary neutron emission spectra and gas-production cross-sections for Li7, B11, C, Fe, Li6, Al, Mo, Ni and Cr. Correlated uncertainty files were needed for all significant CTR materials. It also

appeared necessary to extend the upper energy limits of data files to about 30 MeV in order to analyse radiation damage studies using D-Li sources. Data available up to 15 MeV were being assembled into a new multigroup set (171 neutron groups, 36 gamma groups) which was currently being validated and was expected to be available by the end of 1976. Benchmark testing of nuclear data was under way at the University of Illinois using a 14 MeV source. This work had raised questions concerning the higher energy iron cross-sections.

Calculation methods appeared to be generally adequate. The finite element code TRIDENT incorporated a triangular (R,Z) mesh capability and a toroidal geometry preprocessor was being developed for use with this code.

Studies of the sensitivity of the tritium breeding ratio to nuclear data uncertainties had identified the $\text{Li7}(n,n'\alpha,t)$, $\text{Be}(n,2n')$ and $\text{F}(n,t)$ cross-sections as contributing the greatest uncertainties (of a few percent). A study of materials for the first structural wall of a CTR indicated stainless steel 304 as the preferred material. Studies of the effect of nuclear data uncertainties in iron and carbon on the heating and radiation damage in the toroidal field coil of a Tokomak had indicated uncertainties of the order of 100 %.

In the systems design area the two machines of most interest were the Tokomak Fusion Test Reactor (TFTR) and the Tokomak Experimental Power Reactor (TEPR). The TFTR would be built at Princeton to study plasma physics. It was expected that air and steel activation would give the primary limitation on personnel access and that radiation damage to test equipment would also be a limitation. The design of the TEPR was still in a very early stage. Multi-dimensional neutronics studies had indicated the great importance of neutron penetration through streaming paths such as neutral

beam injectors. This could give rise to flux increases by factors of 10^4 to 10^6 at the toroidal field coils. These studies showed, however, that the penetrations had only a small effect on the tritium breeding ratio.

Germany

Küsters presented the paper NEACRP-L-177. Measurements had been made of neutron spectra at various radial positions in a lithium metal sphere with a 14 MeV neutron source. Time-of-flight methods and proton recoil techniques were used. Calculations of the neutron spectra were made with a 1D Sn code using the extended transport approximation for the treatment of anisotropic elastic scattering. ENDF/B-3 data were used. Good agreement between measured and calculated spectra was obtained when the measured neutron source spectrum was used in the calculation.

Studies were made of the sensitivity of tritium production in the sphere to uncertainties in the cross-sections of the $\text{Li6}(n,\alpha)t$ and $\text{Li7}(n,n'\alpha,t)$ reactions. These indicated uncertainties in the tritium breeding ratio of 0.9 % and 15 % arising from the Li6 and Li7 reactions. Similar studies made by Tobias and Steiner for the case of a large fusion reactor blanket model gave corresponding figures of 0.34 % and 15 %. This indicated that the cross-section sensitivity of tritium production is strongly dependent on the blanket configuration.

Measurements of the tritium production rate in the lithium sphere were made using Li_2CO_3 probes and the results showed some deviations from the calculated values. The possibility of experimental error was being investigated.

General Discussion

In a discussion on sensitivity analysis it was generally agreed that uncertainties in the tritium breeding ratio arising from cross-section uncertainties did not present major problems. The KFK conclusion that the cross-section sensitivity of the breeding ratio depended strongly on the blanket configuration was noted but Maienschein pointed out that US sensitivity studies had given the same results for three significantly different blankets. It appeared that variations between different blankets were less important than the difference between a blanket design and the KFK sphere. The situation concerning heating and radiation damage in the toroidal field coils of a Tokamak was less satisfactory and the US studies had shown uncertainties of the order of 100 % arising from uncertainties in the scattering cross-sections of iron and carbon.

It was generally agreed that existing calculation methods were adequate for dealing with fusion reactor problems.

In a discussion on the accuracy of gamma-ray production cross-sections Barré said that measurements of gamma ray heating in an iron block had led to C/E values of the order of 0.5. Küsters described measurements of gamma-ray flux, made in an iron block on the SUAK facility, which had given C/E values of the order of 0.5 when analysed using ENDF/B-4 data. Hemmig said that corresponding US measurements had given C/E discrepancies of only 10-15 % for ENDF/B-4 analyses of iron block experiments. He suggested that the larger discrepancies in the German case might have originated in the processing of the ENDF/B-4 library data. Maienschein suggested that the problem might arise from errors in the calculated neutron distribution.

It was agreed that future results on fusion would be included in

the national activities reports.

2. NEW TOPICS

2.1. Problems related to burn-up in thermal and fast reactors.

Belgium

Buistraan introduced the paper NEACRP-A-278 which gave a review of the joint Belgonucléaire-Mol programme on plutonium recycling in LWR's. An extensive programme involving both experimental and theoretical work had been in progress for over 15 years. The experimental programme included critical assembly measurements in VENUS and irradiations in a number of power reactors. The results were used to check the validity of the neutronics codes and data. The calculations used a code package based on the PANTHER cell calculation code. This scheme allowed the properties of plutonium fuel configurations to be calculated with the same accuracy as that obtained in the case of uranium fuel, for example, in the case of power distribution, the discrepancy between calculation and experiment was typically of the order of 5-7 %. The irradiation experiments had shown that the mixed plutonium-uranium oxide fuel elements could satisfy operational requirements.

In discussion it was generally agreed that C/E values for power distributions in Pu/U lattices did not differ significantly from those in uranium lattices. There was some feeling that the C/E discrepancies in the Belgian paper could be reduced by the use of a more advanced calculation technique.

France

Bouchard presented the thermal reactor aspects of the paper NEACRP-L-157 which dealt with problems related to burn-up. The French approach was to create a system of codes and data for the calculation of all necessary burn-up parameters and to test the system by using the results of critical experiments and of measurements on irradiated fuel elements.

An extensive programme of fuel irradiations had been carried out with subsequent determination of the global capture cross-section of fission products, power distribution within the fuel rods and heavy nuclide concentrations (the latter including Pu238, Cm242 and U232). The various measured quantities were generally well calculated but a 5 % discrepancy between the measured and calculated values of the Pu239/U238 concentration ratio was difficult to explain and might be related to the discrepancy between the measured and calculated temperature coefficient in LWR's.

In a discussion of the C/E discrepancy on the concentration ratio Askew remarked that this did not appear in the corresponding work in the UK. He said that, if the appropriate information on the APOLLO cell calculations could be made available, he would be prepared to carry out a calculation using UK methods so that the results could be compared with those given by APOLLO. It was agreed that this should be done by direct contact between Askew and Bouchard. Küsters asked to be informed of the outcome of this comparison.

Barré presented the fast reactor sections of the French paper. The main problems related to the reactivity loss per cycle, the variation of the global breeding gain and the production of transactinides. A combined experimental and theoretical programme had been set up to deal with these problems and this work was

closely linked to the corresponding work in the thermal reactor area.

In the reactivity loss area work was being carried out on the effect of higher plutonium isotopes involving (a) measurements of buckling and k-infinity in plutonium-bearing lattices in which the Pu240/Pu ratio varied from 8% to 45% (b) fission rate ratio measurements (c) reactivity worth measurements on samples with various isotopic compositions (d) irradiation of samples in RAPSODIE and PHENIX. Analysis of the irradiation experiments using Carnaval version 3 had given good agreement between calculation and experiment and, in particular the capture cross-sections of Pu239 and Pu240 relative to U238 were predicted to an accuracy of 2%. The data obtained in the plutonium measurements would be included in a re-adjustment to produce Carnaval version 4 which was scheduled for completion in November 1976.

In the fission product area (which was mainly of importance in relation to reactivity loss per cycle) measurements had been made of the capture cross-section of global fission products by oscillating irradiated fuel samples in three different lattices in ERMINE. Measurements were also made on the more important individual fission products using oscillation, activation or irradiation techniques. Following data adjustment it was expected that the capture in global fission products would be predicted with an accuracy of $\pm 7\%$ (2σ).

The variation of global breeding gain was being followed in an extensive irradiation programme on PHENIX which would give the variation of heavy isotope composition from the first cycle up to 60,000 MWd/Te. The main results up to the third cycle were already available and the blanket results were expected to be available by the end of 1976.

In studies of transactinide problems (both in-pile and out-of-pile) fission ratios were measured using fission chambers. It was considered important to estimate the build-up of Pu238 with reasonable accuracy because of the problems it would produce at the reprocessing stage.

It was therefore proposed to make capture measurements for this nuclide, and also for other important transactinides by means of an irradiation in PHENIX. The neutron emission rate of curium would also be measured.

In discussion Barré stressed the need for better evaluated data on the inelastic scattering cross-sections of fission products. The present data could lead to errors of the order of 10% in the derivation of capture cross-sections from the results of oscillator measurements. A list of fission products for which these data was required had been produced at the Bologna meeting. Cross agreed to inform the NEANDC of the NEACRP's continued interest in these data.

Italy

Farinelli summarized the paper NEACRP-A-279 which described the development of a generalized perturbation theory applied to the nuclide compositions in a reactor lattice so that the variable considered was the composition of the fuel. The time evolution equation was then written in terms of the fuel composition vector and a perturbation equation was derived and used to calculate perturbed quantities. The method had been applied to burnup in a fast reactor and to the calculation of actinide production.

Germany

Küsters introduced the paper NEACRP-L-175 which reviewed the IAEA

advisory group meeting on transactinium nuclear data held at Karlsruhe in November 1975. The papers presented at the meeting had indicated that the most stringent requirements for trans-actinide cross-sections in the reactor area are given by the effect of these nuclides on core performance. Target accuracies for the actinide data were generally in the range 20-50 % and these targets could currently be achieved in most cases. A particular exception was Am241 in which case the targets for thermal reactors were 10% for both capture and fission and, for fast reactors, 5% for capture and 15% for fission. One of the main current problems in setting target accuracies was that the concentrations of the various nuclides were critically dependent on the fuel cycle scheme adopted. For example, the hazard potential of a PWR with plutonium recycling was about ten times greater than that of the corresponding reactor without recycling and such variations were much greater than uncertainties arising from the cross-section data. The conclusion of the meeting was that the uncertainties in transactinide cross-sections did not currently play a dominating role.

The PWR burn-up code GELS, produced by the Geesthacht Research Centre, was distributed at the meeting.

US

Maienschein said that US work on burn-up included the development of a 3D depletion code which was an extension of VENTURE. This was expected to be available from the Argonne Code Centre by the end of 1976. In the fission product area CSWG was working on the incorporation of substantially more detail into the next version of ENDF and to make comparisons with the results of integral fission product data measurements. Improved actinide data for thermal reactors was also being incorporated into ENDF. Consid-

eration was being given to investigating the sensitivity of depletion calculations to nuclear data uncertainties.

In a discussion concerning the various versions of the ORIGEN library, which contains one-group actinide data for fast reactor calculations, it appeared that there were considerable uncertainties as to which one-group data were being used by individual member countries and also as to which were the currently recommended values. In a move to clarify this situation members agreed to send to Küsters details of the one-group actinide data which they were currently using in fast reactor calculations and Küsters agreed to make a critical review of these data. Consideration would then be given, at the next meeting of the committee, to the setting up of a benchmark exercise.

Euratom

Farinelli presented the paper NEACRP-A-283 which gave an example taken from a series of parametric studies on the results of applying various nuclear strategies in the period from 1980 to 2030 together with a study of the sensitivity of the results to the values of the parameters used.

In a final discussion it was agreed to carry over the topic of burn-up problems to the next meeting of the committee.

2.2. Prospects for thermal breeders

US

Introducing the paper NEACRP-A-273 Till said that the work described resulted from an increased US interest in possible back-up

systems for the LMFBR. A study had been made of the possible impact of thermal near-breeders in a growing power system which was a simplified model of the US power economy in which the nuclear capacity doubling time increased from 5 years in 1985 to 30 years in 2050. The code ALPS had been used to compare cumulative uranium requirements in a number of scenarios. Very favourable near breeder characteristics were chosen, i.e. those of a CANDU reactor on thorium cycle with a conversion ratio of 1.0. The first set of cases examined were:

- 1: LWR only
- 2: LWR + Near-breeder (1995)
- 3: LWR + LMFBR (1995)
- 4: LWR + Near-breeder (1995) + LMFBR (2015)

The only constraint on the introduction of the near breeder and LMFBR was plutonium availability. Case 3 was the only one which tended to level off the cumulative uranium requirements at values consistent with estimated US high grade reserves.

A similar series of calculations was carried out for the case of a CANDU-based power economy and gave qualitatively similar results except that the uranium requirements for all scenarios were substantially less than those for the LWR-based economy because of the much better plutonium production of the CANDU reactor. The results indicated however that, with the assumed power growth rates, it would be absolutely essential to introduce the LMFBR at about the year 2000 in order that cumulative uranium requirements remain consistent with US high grade reserves.

Another set of system studies (Hatcher, Banerjee et al, ANS Meeting, Nov. 1975) had assumed an initial nuclear capacity doubling time of 5 years through 2000, 11 years from 2000 to 2030 and no further growth after 2030. Under these assumptions

the combination of a normal converter and a near breeder gave better uranium utilization than the LMFBR. This was because, in the initial rapid growth phase, the uranium requirement was determined solely by the specific inventory of the reactor so that a reactor with low inventory was favoured while, in the zero growth phase a self-sufficient reactor was as good as a breeder.

The general conclusion was that, in the case of system studies of this kind, quite plausible differences in input assumptions could lead to diametrically opposed conclusions. A useful comparison between studies having differing assumptions could, however, be made in terms of the relation between the doubling time of energy demand and the doubling capability of the breeder system.

Germany

Küsters introduced the paper NEACRP-L-169 which proposed a combination of two types of pebble bed HTR's: (1) a conventional type fed with 93% U235 and thorium and having a conversion ratio of 0.76 and (2) a near breeder with a conversion ratio of 0.97 in which the bred U233 was recycled. The U233 from the conventional reactor was used as make-up for the near breeder. In the equilibrium condition the U233 produced by one conventional reactor was sufficient to provide make-up for ten near breeders. The production of the fuel inventory for one near breeder required seven years full power operation of the conventional reactor. The two types of reactor had the same fuel element and core dimensions and also the same power densities and the same control systems and, as a result, a given reactor could be changed from one cycle to the other during full power operation. This gave useful flexibility for coping with variations in power growth rates. A study covering the Debenelux area had shown that the cumulative uranium ore requirements of this combined HTR system

would be approximately the same as that of an LWR-LMFBR combination for a period of 8-10 decades.

2.3. Calculations of control rod worths

UK

Campbell presented the paper NEACRP-L-162 which described the performance and analysis of two series of control rod measurements made in ZEBRA core 12 which contained about 800 kg of fissile plutonium. The first series was the MZC programme of measurements using MONJU mockup rods and the second used the PFR/CFR natural boron mockup rods.

The MONJU mockup rods consisted of a 19 pin cluster, the diameters of the absorber pins being 11.1 mm for boron and 13.1 mm for tantalum. The pins were contained in a stainless-steel, sodium-filled calandria 103 mm square. These mockup rods each occupied the space of 4 ZEBRA elements. Measurements were made for rods containing natural boron, 30%, 80% and 90% enriched boron and tantalum. In addition a measurement was made for a rod containing 37 pins (19 pins of 90% enrichment and 18 pins of 80% enrichment). Measurements were made for single rods at the core centre and at four radial positions. Measurements were also made for arrays of up to four rods.

The PFR/CFR mockup rods also contained 19 absorber pins but the pin diameter was 18.9 mm. The 19 pin cluster was contained in a hexagonal calandria which was filled with sodium. This hexagonal calandria fitted into an outer square calandria which replaced a 3x3 array of ZEBRA elements. The outer calandria contained mixed PuO_2/UO_2 fuel pins. Measurements were made with single rods at the core centre and at two radial positions in the inner core and with arrays of 2 and 3 rods.

The critical-balance technique was used to determine the rod worths, additional fuel elements at the outer core/blanket boundary being loaded to compensate the control rod absorption. All reactivities were related to the worth of central or distributed plutonium.

The standard method of analysis used to interpret the experiments was XY geometry diffusion theory. The number of energy groups and mesh points, and the treatment of axial leakage were chosen so that each approximation introduced an error of less than about 3%. This resulted in the choice of 9 energy groups, a mesh area of 27 mm x 27 mm in the core and group and zone-dependent axial bucklings to represent axial leakage.

The standard method of treating the fine structure of the flux in a control rod (within the square box) was to approximate it by a concentric cylindrical model, surrounded by a homogeneous region of core material. The flux distribution was calculated using MURAL and collision probability methods and equivalent homogeneous cross-sections were obtained for the control rod region.

The analysis by the standard method gave C/E values (for absorbing rods referred to follower or follower referred to fuel) that were greater than unity. The C/E values for boron carbide rods at the core centre increased with enrichment, varying from 1.04 for natural boron to 1.14 for 90% enriched. The value for tantalum was 1.10. The accuracy of rod worth predictions appeared to improve as rods were moved towards the edge of the core. This was the result of a compensating error in the prediction of flux distributions. The C/E values for multiple rod arrays were consistent with those for single rods, implying good predictions of rod interactions for this size of core. An important point was that sodium follower worths were not well predicted (C/E = 1.15)

Additional calculations were made to study the influence of

approximations in the standard methods (diffusion versus transport theory, mesh size, group structure) and modelling of the rods (smeared versus pin geometry). It was concluded that these more sophisticated treatments would predict a wide range of control rod arrangements in a prototype-size core to within a standard deviation of $\pm 7\%$.

Inoué said that an analysis of the results of the ZEBRA measurements on the MONJU rods had been carried out using diffusion theory calculations with transport correction. The C/E values lay in the range 0.95-1.05. There was no indication of a variation of C/E with control rod position or with the number of control rods.

France

Bouchard presented the thermal reactor section of the paper NEACRP-L-159. The calculation of control rod worths and power distribution perturbations in an LWR proceeded in two stages:

- (1) Calculation of the 99-group cross-sections of each type of sub-assembly by the 1D collision probability code APOLLO which gave the cross-sections of the homogenized assembly and the heterogeneous cross-sections of each individual cell.
- (2) Whole core calculations using diffusion theory in 2 or 4 energy groups.

The calculation methods were checked against the results of measurements made on critical assemblies and on one power reactor and all calculated reactivity effects agreed with the measured values within experimental errors. Reaction rate distributions were generally calculated within $\pm 5\%$. The greatest discrepancy (at the core edge) was less than 10%. The accuracy of the predictions

was considered to be satisfactory.

In discussion it was generally agreed that control rod calculations for thermal reactors did not present significant problems.

Barré introduced the section of the French paper dealing with the programme on fast reactor control rods. The aim of this programme was to achieve an accuracy of $\pm 5\%$ (2σ) in the prediction of control rod worths, $\pm 3\%$ on the maximum local power variation and $\pm 2\%$ on the integrated axial power in each sub-assembly.

Two methods of calculation were used: (1) the design method based on 6-group, diffusion theory in one or two dimensions and using CARNAVAL version 3 data and (2) the "reference" method using 25 groups and 2D or 3D calculations.

A large programme had been carried out on central control rods and, for boron carbide rods, a systematic 5% overestimate of the reactivity worth was found. In the case of stainless steel the overestimate was 5-10%. The most important problem occurred in the case of the sodium follower for which the reactivity overestimate varied with stainless steel content, ranging from 40% with zero steel to 5% at 100% steel. The SUPER-PHENIX follower was designed to have only 7% of steel so that this discrepancy would cause considerable problems. Anisotropic diffusion calculations using the code DIFHET were leading to some improvement in this situation. In the case of a central depleted uranium rod there was a discrepancy of 10% in a uranium core and 6% in a plutonium core. For excentric rods the conclusions were generally similar but the discrepancy varied from 27% at the core centre to 18% at the edge. For absorber rods the discrepancy (6-7%) did not vary with radius. Interaction effects between two excentric rods were overestimated by up to 6%.

Interaction effects were expected to lead to considerable problems in the case of SUPERPHENIX in which 24 rods in 2 or 3 rings would be used

In the case of power distribution it was found that, if the rod reactivity worths were correctly calculated, then the calculation of the perturbations produced by the rods presented no major problems and discrepancies were small even in the case of strong rod interaction.

The control rod programme would continue through 1977 and would include work on the "heterogeneous" concept.

In a discussion of target accuracies Barré said that the French target of 5% (2σ) for the prediction of control rod worths had been set to allow a reduction of the number of rods in the core. This would be important in the case of SUPERPHENIX where of the order of 24 rods would be required. From the point of view of safety alone it would have been adequate to adopt a target of $\pm 10\%$. Campbell said that the UK target was $\pm 5\%$ (1σ) but this was linked to the proposed use of natural boron rods. If enriched boron were considered then the target would be relaxed since the rod capacity could be adjusted by changing the enrichment.

Concerning the choice of rod composition, Barré, Campbell, Hemmig and Küsters said that there was no current intention, in their respective countries, to depart from boron carbide although in some cases (US and UK) back-up studies of other materials had been made.

Sweden

Hellstrand introduced the paper NEACRP-A-261 which described the

treatment of cruciform control rods in CASMO, a multigroup transport code for burnup calculations in LWR's. In this code the absorption in the cruciform rod was calculated by the collision probability routine CROCOP using a realistic representation of the rod as an array of cylindrical absorbers. Comparisons of the results of CROCOP calculations with those of calculations using slab representations of the absorber showed that the latter overestimated the blackness. CASMO had been tested in power reactor applications and also in comparison of the measured and calculated worths of control rods in KRITZ where it was found that the reactivities of systems with and without a central control rod were calculated with similar accuracy.

France

Barré presented the paper NEACRP-L-161 which reported on the specialists' meeting on control rod measurement techniques held at Cadarache on 21st and 22nd April, 1976. The meeting had been attended by about 30 people including representatives from industry and from electrical utilities. This mixing of people from different disciplines had been an extremely useful feature of the meeting.

The aim of the meeting had been to review experimental techniques for the measurement of control rod reactivity worths and of the power distribution perturbations produced by the rods. Many contributions concerning reactivity worths had been presented and had led to fruitful discussions but, in the power distribution area, little information had been presented and the discussions had been lacking in depth.

It had been agreed that, for safety purposes, the target accuracy for the prediction of the reactivity worths of control rods

was ± 10 %. For design optimisation, however, it would be useful to predict the worths to ± 5 % with the aim of being able to reduce the number of rods. The need to have a continuous analysis of the reactivity level of a power reactor (both in operation and after shut-down) led to a target of less than ± 5 % for the control rod reactivity worth gradient.

In the power distribution area few comments had been made on target accuracies. For fast reactors accuracies of ± 3 % on maximum local power and ± 2 % on integrated axial subassembly power had been mentioned.

It had been agreed that experimental techniques used for control rod measurements on power reactors must:

- (1) use the standard instrumentation of the plant
- (2) be simple and quickly executed
- (3) give results rapidly without the need for additional calculations.

Measurement techniques in thermal power reactors were still largely based on simple point kinetics analysis and this did not seem to be adequate except for low power measurements. Only a few comments had been made concerning detailed spatial analysis. Measurements of rod worths under power conditions involved non-linear feedback effects and none of the presentations described techniques which took these effects into account. Further work was needed in this area.

For fast power reactors point kinetics techniques were also used and simple perturbation techniques had been developed for predicting control rod worths for 300 MWe reactors. It would now be necessary to extend these techniques to deal with the more complex problems of commercial-sized fast reactors.

The importance of defining the accuracy requirements for measurements of control rod worths and power distributions in operating power reactors was stressed.

Measurements of control rod worths in critical assemblies were well established and followed two main lines:

- (1) Control rod insertion compensated by increasing the core radius
- (2) Subcritical techniques, notably the source multiplication method with corrections for spatial dependence.

A US presentation had proposed a more accurate subcritical method using source multiplication calibrated by an improved rod-drop technique and this might partially replace the core radius variation technique which was very time-consuming.

Noise techniques though applicable to critical facilities were not considered useful for power reactors.

Barré recommended that a further meeting should be held at about the beginning of 1979 and that efforts should be made to ensure a large participation by power reactor operating staff.

It was agreed that members would send comments on the paper NEACRP-L-161 to Barré by 1st September and that he would then send the final version to the Secretariat who would reproduce and distribute it.

Germany

Küsters presented the paper NEACRP-L-171 which reviewed the methods and data used in control rod worth calculations. The most widely used methods were 2D diffusion theory, 3D diffusion synthesis,

2D transport and Monte Carlo. During the past few years the improvements in the calculation models, and in the data sets used, had resulted in a reduction in the C/E discrepancies from about 25 % in 1967 to about 10 % at the present time. In general no major problems were experienced in connection with control rod interactions but special care was needed in the treatment of followers.

A preliminary assessment of some very recent rod worth measurements in SNEAK 9C had shown a C/E discrepancy of 29 % for 90 % enriched B_4C rods but it was felt that this might be due to the use of an inadequate geometrical model in the calculation. A similar discrepancy in the case of EU_2O_3 rods was probably attributable to nuclear data errors.

In a final discussion on the control rod topic it was agreed that it should be taken again at the 1978 meeting.

2.4. Homogenization problems in thermal and fast reactor calculations

UK

Campbell introduced the paper NEACRP-A-285 which summarized the methods currently used for the treatment of heterogeneity in fast reactors and indicated the following limitations of these methods.

- (a) The use of simple one dimensional geometry cell representations for more complicated geometries. Effects not treated in such approximations were:
- (i) a pin in a cell with a hexagonal boundary
 - (ii) a simultaneous treatment of the pins in a sub-assembly and the sub-assembly wrapper plate
 - (iii) in zero power critical assemblies which used plates, the plate compositions were usually smeared over the lattice

area, thus effectively reducing the density of canned plates and neglecting edge effects and the effect of the steel box containing the plates.

- (b) Anisotropic leakage effects were usually neglected. Some work had been done on the effects of streaming in low density channels in the interpretation of sodium voiding experiments in zero power critical assemblies and streaming along coolant channels and between subassemblies. However, the methods were not well established and not routinely available. More work was needed to establish the accuracy of the available methods.
- (c) Anisotropic scattering effects might be important, particularly in the calculation of the effects of water or hydrogenous material. Methods of calculating these effects in multi-region cells had to be developed.
- (d) For cells near boundaries, and, in particular, in the regions of breeder zones, the usual simple treatments of leakage by a macroscopic absorption cross-section equal to B^2D could be quite erroneous, particularly in the zero power reactors with breeder zones containing thick plates of uranium and diluent material. The effective buckling could be quite strongly energy dependent and this should be taken into account.

US

Till introduced the papers NEACRP-A-276 and 277 which described the comparison of the results of VIM calculations with those of SDX and MC²-II calculations. This work had been carried out to validate the SDX and MC²-II procedures for ZPR plate cells.

An initial check on the energy collapse procedures was made using a zero leakage, homogeneous model. This showed excellent overall

agreement for k-infinity and reaction rate ratios except for C8/F9 for which SDX and MC²-II were in agreement but were both low relative to VIM. This was due to inadequate treatment by VIM of the unresolved U238 capture.

The exercise was then repeated using a zero leakage, heterogeneous 1D slab model and excellent agreement was again obtained except for the previously noted C8/F9 discrepancy.

The check was then extended to include cases with leakage and, for the homogeneous and 1D slab cases, good agreement ($\sim \pm 1\%$) between VIM and SDX was obtained for k-infinity and mean squared chord lengths.

A comparison of the results of a VIM calculation of a 3D unit cell with the results of a standard solution for a corresponding 1D slab model showed that the standard procedures underpredicted leakage to an extent which could lead to an error of about 1% in the prediction of k-eff for a typical GCFR cell. This was apparently because the procedure used to produce the 1D cell model smeared too much of the steel from the tops and bottoms of the matrix tubes into the void slots.

Japan

Hirota presented the paper NEACRP-A-260 which described a measurement of the ratio of parallel to perpendicular diffusion coefficients in a plate lattice in the sector zone of FCA Assembly VII-1. The measurements was based on a determination of the reactivity change produced by changing the orientation of the plates by 90°. There was some disagreement between the results of the measurements and the values calculated by Benoist's method.

Germany

Küsters introduced the paper NEACRP-L-170 which discussed a number of homogenization problems:

- (1) The influence of grain structure in HTR fuel.

A collision probability approach had been developed to deal with the doubly heterogeneous situation arising from the use of fuel particles embedded in graphite spheres in the HTR pebble-bed reactor. There was a significant difference between the resonance integrals given by this method and those obtained by smearing the fuel through the graphite sphere. The results given by the collision probability method were in satisfactory agreement with experiment.

- (2) Heterogeneity effects in fast-thermal test loops.

Experiments and calculations had been carried out to assess heterogeneity effects in Mol experiments involving irradiation of bundles of highly enriched fuel rods in the 10 cm diameter channel of the central graphite zone of the reactor ARK. The bundles each consisted of a central rod surrounded by a ring of 12 rods with a 1.64 cm radius. The spatial variation of reaction rates in a bundle was satisfactorily calculated by WIMS and the Karlsruhe Monte Carlo code KAMCCO but, in the case of RABBLE, there was a considerable discrepancy which had not yet been explained.

- (3) Resonance phenomena near boundaries in fast systems.

The method used for calculating reaction rate distributions in the region of the core-blanket interface had been improved by taking into account (1) the spatial dependence of the source density and (2) anisotropic elastic scattering. This improved treatment gave excellent agreement with experiment.

After discussion of the papers Campbell agreed to prepare a statement on the current status of the treatment of heterogeneity in reactor calculations.

2.5. Power reactor noise

US

Maienschein introduced the paper NEACRP-A-280 which summarized US activities in power reactor noise analysis. The application of noise analysis methods had recently contributed to a better understanding of LWR problems such as instrument-tube vibrations, core-barrel motion and loose-parts detection. These applications had led to the commercialization of noise methods. The US Regulatory Commission had now accepted noise analysis as a useful tool in assessing the safety of operating plants. EPRI was sponsoring noise analysis research and also considering sponsoring a noise analysis information centre and data bank.

Specific US activities in the noise area were listed as:

(1) LMFBR

Detection of coolant boiling within sub-assemblies

Detection of coolant flow-blockages

Detection and location of failed fuel elements via delayed neutron release to coolant.

Surveillance of spent fuel reprocessing plants

(2) LWR

In-service measurement of void fraction.

Measurement of time-dependent two-phase flow.

(3) General

Instrumentation integrity and in situ response time testing.

Automated detection of plant performance anomalies.

Monitoring for excessive flow-induced vibrations.

Monitoring for loose parts in the reactor primary system.

Detection of cracks and leaks in primary piping, steam generators and condensers.

Continuous monitoring of rotating machinery.

France

Bouchard presented the paper NEACRP-L-158 which dealt with the application of noise analysis to PWR's. The use of such methods had increased significantly in the past few years as a result of improvements in the apparatus and advances in the field of signal analysis. Current experiments were aimed at monitoring the correct functioning of the reactor and at the detection and diagnosis of faults.

Bouchard also summarized the French noise analysis programme in the fast reactor field. Work on the analysis of temperature noise in a sodium loop had been described in a paper given at the Risley meeting (January, 1976). The development of a noise-based general surveillance system for fast reactors (the 3DC system) had been described in a paper to the 1975 Paris Conference. Measurements on sodium boiling had been carried out in RAPSODIE using both neutron noise and acoustic noise detectors and local sodium boiling had been detected. Further experiments of this type were envisaged.

Holland

Buistraan reported on Dutch activities in the noise field (see NEACRP-L-155h) which covered temperature noise in sodium and reactor noise in LWR's.

The sodium noise measurements were made in heated rod bundles simulating LMFBR fuel elements. In a recent experiment a partially blocked 28-rod bundle was used and the recirculating flow pattern was clearly observed.

Noise measurements had been made on the 470 MWe PWR at Borssele using neutron detectors and pressure transducers. These had indi-

cated movements of the core and vertical fuel movements.

Similar measurements had been made on the 58 MWe BWR at Dodewaard and these were currently being interpreted.

Germany

Küstlers presented a series of Karlsruhe reports under the number NEACRP-L-174.

Measurements had been carried out in the sodium-cooled thermal reactor KNK to check the possibility of using noise methods to detect flow anomalies (L-174d and e). It had not been possible, from these investigations to decide whether the methods could achieve the high accuracy required to give early detection of a local blockage or the short response time required in the case of a large flow blockage which might propagate rapidly through the core.

The experience gained in this work led to the development of two methods for in situ testing of neutron and temperature instrumentation in a power reactor (L-174c). These methods used the inherent fluctuations of the quantities being measured as dynamic test input to the whole signal channel. They could therefore be used during normal reactor operation. Neutron instrumentation testing was based on the prompt jumps in reactor power following single reactivity steps produced by the control system during normal operation to maintain the power at a constant level. The signal response to single steps could be clearly identified. Missing a prompt jump in the signal would indicate a significant failure. The testing of temperature instrumentation required a different procedure and the relationship between temperature and power noise was used. It was found that the ratio of the maximum

value of the cross-correlation function between neutron and temperature noise signals was a suitable quantity for testing the temperature instrumentation of the reactor.

A new method for the detection of sodium boiling in LMFBR's (L-174b) was based on the assumption that the boiling produced fluctuations of the neutron flux within a restricted frequency range. A resonance-type increase in the power spectral density of neutron noise signals was therefore observed. The paper discussed the criteria relating detection sensitivity, false alarm rate and response time and showed that, for a prototype reactor of the type of SNR 300, local and integral boiling could be detected by this method.

A method of measuring sodium flow velocity (given in the paper NEACRP-L-168) depended on the measurement of temperature fluctuations by two thermocouples spaced at a known distance apart. Cross-correlation of the thermocouple signals allowed the transit time of the temperature fluctuations (and therefore the sodium velocity) to be determined provided that appropriate allowance was made for heat exchange effects.

A measurement of local steam bubble velocities in BWR fuel elements (NEACRP-L-173) had been carried out in the Lingen BWR. Three SPN detectors, whose vertical positions could be varied over the core height, were used and cross-correlation analysis of the signals from the detectors gave the bubble velocity. Good agreement was obtained between measured and calculated values.

Küsters summarized the areas in which further work would be carried out at Karlsruhe as (1) sodium boiling detection (2) measurement and interpretation of power reactor noise signals (3) instrumentation (4) data processing.

Sweden

Hellstrand said that Sweden had a substantial programme in the noise area involving measurements in a number of power reactors. To date these had largely used in-core detectors but an extension to include surface detectors (e.g. flow meters) was envisaged. The aim was to include noise measurements as an integral part of the surveillance systems of power reactors.

In a discussion on SMORN 2 it was agreed that it would be appropriate to hold this meeting in October 1977 and that an attendance of about 80 would be appropriate. Maienschein agreed to ask Booth to arrange the meeting on this basis. The Secretariat agreed to contact the CSNI Secretariat, the IAEA and the International Working Group on Reactor Control and Instrumentation in connection with the arrangements for SMORN 2.

2.6. Review of reactor physics problems related to LMFBR safety

GERMANY

Küsters introduced the paper NEACRP-L-172 which outlined the role of reactor physics in the analysis of fast reactor safety.

The accident situations of particular current interest were:

- (a) Local failures and the possibility of failure propagation
- (b) Unprotected whole core accidents caused by pump coast down accompanied by failure of the safety system or by reactivity insertion.

Local pin failures could not be excluded because of the very large number of pins (of the order of 10^5) in a fast reactor core. The important question was whether such a failure would cause the release of fuel into a coolant channel leading to a local blockage

and local sodium boiling. Failure to detect the blockage could lead to propagation between sub-assemblies producing a major accident condition. Research objectives in this area included early detection of local failures, determination of the timescale of blockage growth, effects of local boiling and local fuel coolant interactions on the subassembly structure and proof that a subassembly accident is unlikely to lead to core destruction.

In the case of whole core accidents the research objectives were summarized in relation to the most important aspects, i.e.:

Fuel Pin Failure

It was important to obtain a reasonably accurate prediction of the mechanism of fuel pin failure in a power transient because of its sensitive effect on subsequent accident sequences. Important aspects were (1) upward movement of fuel in the central pin cavity leading to negative reactivity feedback if the pin fails in the upper part of the core (2) mode and amount of fuel expulsion into the coolant channel initiating fuel-coolant interactions and sweep-out (3) mode and amount of fission gas release possibly resulting in a violent disassembly mechanism.

Little experimental evidence was available on the mode of pin failure.

Fuel-Coolant Interactions (FCI)

Experiments to date with UO_2 and sodium had shown no explosive interaction except in the case where a few grams of sodium were injected into molten UO_2 , this being the reverse of the situation in the reactor. It was however extremely difficult to demonstrate that an explosion would not occur in any circumstances in the course of a whole core accident.

The incoherent onset of FCI in subassemblies with different burn-

up would have a considerable effect in reducing the void reactivity ramp rate.

Fuel Movement

Fuel sweep-out with its associated negative reactivity feedback could produce a very important reduction in the severity of this accident sequence but only if it occurs rapidly enough to forestall the positive reactivity effects of voiding. It is therefore important to have a means of measuring the rate of fuel sweep out in experiments. Neutron hodoscopes had been investigated (e.g. at the TREAT facility) but experience to date had not been too satisfactory.

Fuel Slumping in Pump Coast-Down Accidents

Up to the present the boiling phenomena occurring in this accident had been treated theoretically only in one-dimensional geometry. In fact the onset of boiling within a sub-assembly was radially incoherent. This would modify the void reactivity pattern and have a very important effect on the course of events. The incoherence of fuel slumping in assemblies with different burnup was also of importance. The aspect of this problem needing further experimental investigation was clad motion leading to partial or complete blockage of the coolant channels.

Recriticality

In the case of recriticality after a primary excursion there were no clear boundary conditions for any deterministic predictions of the course of the accident. Experimental support was also lacking at the present time. It was hoped that increasing knowledge of the dynamic behaviour of molten materials would help to reduce the uncertainties to acceptable limits.

The above consideration of accident situations indicated that the current uncertainties in fast reactor safety analysis are dominated by the non-neutronic characteristics of the accident sequences, i.e. material dynamics, thermohydraulics and thermodynamics.

In the reactor physics area the most important requirements were the prediction and verification of:

- Doppler reactivity effects in the pre-disassembly and disassembly phase of a nuclear excursion
- Sodium void reactivity distribution in a complex core situation
- Reactivity effects of fuel and clad motion
- Control rod worth distribution
- Space-time dynamics behaviour, especially in the case of large reactors which are more decoupled than the prototype reactors.

The accuracy of prediction of the first four items was now such that, in almost all cases, it satisfied the requirements of the safety analysis.

One of the most important problems was to determine the permissible degree of simplification which could be used in the physics analysis since the sophisticated methods normally used in static analyses would be prohibitively expensive in a safety analysis.

Special attention must be given to the reactivity effects of:

- control rod movements if calculated by first order perturbation theory
- local sodium boiling (e.g. near the core-blanket interface in a pump coast-down accident)
- gross material movement in the disassembly phase.

These problems would become even more important in the case of large reactors and an important task of reactor physics was to provide relevant core physics data for large reactor designs aimed at the improvement of safety characteristics and breeding potential. In this connection it should be borne in mind that heterogeneous designs which minimize the sodium void reactivity effect do not necessarily remove the main safety problems. This is because core compaction or fuel slumping, if it occurred, would dominate the excursion behaviour and largely remove the advantage of a low sodium void coefficient.

In conclusion it was stressed that primary attention should be devoted to the development of reliable instrumentation and safety systems.

US

Till presented three papers. The first of these, dealt with the extrapolation to the power reactor of sodium void measurements made in a critical assembly. Data presented in the paper indicated the improvement in the agreement between calculated and measured values of sodium voiding worths which was produced by using Benoist directional diffusion coefficients instead of the homogeneous values. This improvement was due primarily to the increase in the magnitude of the calculated leakage term. Calculations showed that this leakage change was due almost entirely to the increased difference between diffusion coefficients in the normal and voided states and not to the changes in flux and adjoint gradients.

Preliminary results were presented of an extrapolation to the power reactor case using a least squares fit to determine leakage

and non-leakage components which would minimize (C-E) for groups of critical experiments. In this case the calculations included no streaming corrections. The adjustment factors were of the order of 0.8 for the non-leakage components and 1.1 for the leakage components. The use of the Benoist diffusion coefficients would produce a greater change in the leakage components than that required by the fitting procedure. It therefore seemed that the Benoist correction led to an overestimate of the measured effect.

The second US paper, NEACRP-A-274, described the simulation of a Hypothetical Core Disassembly Accident (HCDA) sequence which was carried out as part of the CRBR engineering mockup series in ZPPR assembly 5. Static representations were made of the significant steps in an HCDA sequence in order to check the safety analysis calculations. The sequence included sequential voiding of sodium, slumping of steel and slumping of fuel.

The third paper, NEACRP-L-163, described experimental studies of GCFR safety physics parameters in ZPR9. The reported measurements comprised the reactivity worths of small samples of nuclides of interest, helium depressurization worth and U238 Doppler reactivity worth at core centre, and the reactivity effects of accidental steam entry. The most interesting result of this work was that, while the worths of small central samples of water and polythene foam were calculated as negative (using ENDF/B-IV data), the measured values were fairly strongly positive. A series of measurements were made of the worths of polythene foam spread through a central zone of the reactor. Analysis of the measurements using ENDF/B-IV data showed that the worths were underpredicted at low polythene densities and overpredicted at high densities. Measurements of a small central sample of helium indicated that the positive effect of helium depressurization was about 70 % of the calculated value.

SWITZERLAND

Richmond summarized the studies on GCFR steam entry which were being carried out on the reactor PROTEUS. (This work is described in the Swiss activities report: NEACRP-L-155 m.) Measurements were made of the effect of steam entry (simulated by polystyrene) on neutron spectrum and reaction rate ratios. The observed changes in these quantities were accurately predicted by calculations using FGL4 data. Measurements of the reactivity worth of polystyrene in a central zone of the reactor gave positive values as in the ANL experiments. Analysis of the results using FGL4 showed that the worth was underpredicted when the central zone was surrounded by a "dry" lattice and overpredicted when the surrounding lattice also contained the steam simulant.

An attempt had been made to get some feel for the errors arising from data uncertainties in calculating the effect of steam entry in an operating power reactor. This was done by comparison of results given by different data sets and also by considering the individual errors on the roughly equal and opposite terms (change in k-infinity and change in leakage) making up the total reactivity change. These checks had indicated errors of the order of $\pm 2\%$ at a steam density of 0.015 g/cm^3 as a result of which it was, in many cases, not possible to predict even the sign of the steam entry effect.

In discussion Campbell commented that, for the proper treatment of distributed hydrogen in sodium-cooled reactors, the current UK 37-group set was considered inadequate in the lower energy region and some changes were contemplated, Till pointed out the consistency between the ZPR9 and PROTEUS results in the sense that the calculated hydrogen worths changed from underprediction to overprediction as the amount of hydrogen in the system was increased and Barré suggested that the progressive substitution technique might usefully be applied to the experimental study of steam entry.

The proposal by Kiefhaber for a GCFR steam entry benchmark exercise was discussed and Richmond agreed to send details of this proposal to those members who had expressed an interest (i.e. Till, Campbell, Hellstrand and Barré).

UK

Campbell introduced the paper NEACRP-A-281 which identified those safety problems requiring reactor physics information. The Doppler coefficient, for example, must be known to an accuracy of about $\pm 15\%$ not only over the operating temperature range but also over the temperature range appropriate to accident situations. The UK view was that much further work was required to achieve this accuracy and this was in contrast to the view, expressed in the German paper, that such an accuracy had been achieved.

In the case of core perturbations it was important to check the range of validity of the point kinetics approximation and to know in what circumstances space-time kinetics must be used.

Pessimistic assumptions concerning a sodium voiding accident indicated the addition of about \$8 of reactivity to a 1300 MWe LMFBR. This would lead to very high reactivity ramp rates and to core disassembly accidents which would be very difficult to contain. These problems could be greatly reduced if the sodium void effect could be limited to about \$2. It was for this reason that studies were being made on designs with annular or modular cores for which maximum sodium void effects of this magnitude were predicted. In the case of a maximum sodium void effect of \$8 an accuracy of $\pm \$1.5$ would be acceptable. However, if the effect were reduced to \$2 then this should be known to $\pm \$0.5$ and this accuracy had not currently been achieved. Here again the UK view differed from that expressed in the German paper.

A further requirement for reactor physics information related to the change of the core structure from heterogeneous to homogeneous which would occur after sodium voiding. It was important to know the magnitude of the reactivity change which would occur when the streaming paths were closed.

In the calculation area it was necessary to set up approximate models and to show that the use of these in accident codes could be justified. The aim was to limit the additional uncertainties due to these approximations to about $\pm 10\%$. It was of particular importance to be able to calculate reactivity changes following large displacements of core materials.

Sub-assembly accidents leading to local voiding presented particular difficulties in the case of off-centre incidents. Experimental measurements had shown that calculated predictions were currently inadequate in these cases. In the case of less severe incidents it was necessary to have methods for calculating the reactivity effects of elastic displacement of core components.

The determination of the state of sub-criticality of an LMFBR core was another problem area since, for example, the removal of an irradiated fuel element containing a large fission neutron source and its replacement by a more reactive element with a smaller source was very difficult to interpret in reactivity terms. It was very important to develop a simple and direct method of following the reactivity changes.

JAPAN

Inoue presented the Japanese review paper (NEACRP-A-258). In connection with the sodium voiding accident there were many associated phenomena such as fuel-coolant interaction and fuel slumping whose effects could not be adequately calculated. The

aim was to develop analytical models for solving these problems and then to attack the void reactivity itself. Experiments would be made to determine the effect of control rods, fission products and core temperature on the void reactivity. The analytical treatment had included the use of modified diffusion coefficients calculated by the Lelitin method.

In large fast reactors it would be more important to study space-dependency than in the prototype. The space dependence of the effects of localized accidents and the optimization of detector positions had been investigated and the strongest space-dependent effects were found in peripheral zones of the core.

The Doppler coefficient must be known to an accuracy of $\pm 15\%$ and must be measured at temperatures appropriate to accident conditions.

FRANCE

Introducing the French review paper (NEACRP-L-160) Barré stressed that neutronics problems were not of first order importance at the present time. An improvement in accuracy was, however, desirable in the areas of Doppler feedback, sodium void reactivity effect and the reactivity effect of fuel motion in core disassembly. Characterizing a hypothetical disruptive accident by the total energy yield E the long term aim was to reduce to 15% the uncertainty in E arising from all neutronics parameters.

Doppler feedback

Current estimated uncertainties in Doppler coefficient calculations with CARNAVAL III ranged from 15% at 1500°K to 25% for temperatures greater than 3500°K . The average value was taken as 20% and this would produce a 15% uncertainty in the energy yield.

It was concluded that an improvement by a factor 2 was required in the accuracy of the Doppler coefficient.

Sodium void reactivity effect

The initial reactivity ramp was determined not only by the excess reactivity due to sodium voiding (a neutronics problem) but also by the rate at which the voiding takes place (a thermohydraulics problem). The uncertainty in the latter parameter is about twice as great as that in the former. The total uncertainty would have to be not greater than 15 % (for calculations covering the whole core and the full fuel lifetime) in order to limit the uncertainty in energy yield to 15 %. Present uncertainties were of the order of 30-40 %. Further experimental or theoretical work was therefore required in this area with the aim of improving the present accuracy by a factor 3.

Negative reactivity insertion rate during disassembly

This parameter was concerned only with that part of the energy release which occurs after the onset of core disassembly. In the SUPER-PHENIX case this fraction was typically 40 %. The reactivity insertion rate was currently calculated by applying perturbation theory to the intact core. The main requirement in this area was to determine the uncertainty in insertion rate resulting from this method of calculation and, if it were significantly greater than 20 %, to develop an improved method.

General Discussion

Doppler

It was generally agreed that the Doppler effect was required with an accuracy of ± 15 % and that considerable further work would be needed to achieve this. It was accepted that the main un-

certainties in the energy release in a disruptive accident lay in the non-neutronic area but felt that an improvement in the Doppler accuracy would make a useful reduction to the overall uncertainties.

There was some divergence of view concerning the feasibility of extrapolating Doppler coefficient measurements (restricted to 2000 °C) to obtain values characteristic of accident temperatures in excess of 3500 °C. The general feeling was that such an extrapolation was not meaningful but the UK view was that it could be made with reasonable confidence.

The time separation between the Pu239 and U238 Doppler effects was not felt to have an important effect on the accident sequence.

Sodium void effect

It was felt that the required accuracy for the sodium void coefficient could not be simply stated since it would depend on the value of the coefficient in any particular design and on the maximum permissible void reactivity. It did not, however, appear meaningful to try to place a value on this latter quantity at the present time. It was agreed that the study of designs giving low sodium void reactivity was desirable but that their usefulness would be questionable if the energy release were dominated by other effects such as core compaction and fuel slumping.

Material motion

The difficulties of dealing with the relatively large core rearrangements produced, for example, in a pump coast-down accident were emphasized. The migration of the steel cladding, which would produce a positive reactivity effect, was rather more difficult to deal with than fuel movement. There was currently little experimental evidence in the area of material motion and, where results were available there were often large discrepancies

between prediction and measurement especially in off-centre situations. Further experimental work was under consideration in the US and the UK and, in particular, the US was studying a "Dispersive Mechanism Test Facility" which would be intended for investigation of the problem of fuel behaviour during core melt-down.

It was generally agreed that further reactor physics work was required in relation to the reactivity worth of the movement of core materials.

Conclusion

It was felt that the discussion had given a useful clarification of the current situation and that, although there were some differences of opinion on individual points, there was an overall consensus of opinion on the work required in the safety area. Küsters agreed to prepare a summary of the discussion and circulate it to NEACRP and CSNI members.

3. NATIONAL PROGRAMMES

3.1. Review of recent activities, national programmes, evaluation work

The national reports prepared by NEACRP members were presented and discussed.

4. BENCHMARKS

4.1. 2D LWR benchmark calculations

Askew presented the papers NEACRP-A-256/1 and 2 which reviewed the results of the NEACRP BWR benchmark exercise. The problems set were:

- (1) A 9-pin supercell with a central burnable poison pin.
- (2) A "Mini-BWR" with differential U235 enrichment, water gaps and a cruciform control rod.
- (3) A 7x7 pin BWR lattice cell with several U235 enrichments and poisoned fuel pins.
- (4) An 8x8 pin BWR lattice cell with uranium, plutonium and poisoned pins, one empty pin position and a cruciform control rod.

In each case six-group macroscopic cross-sections were provided for each material since the exercise was intended solely as a check on calculation methods. Eight sets of solutions had been received from six countries.

A variety of calculation methods had been used:

- (a) The JAERI solution and the Winfrith "AEEW(1)" solution both used the same explicit collision probability routine to represent the pins in their true geometry. These could not, however, be considered as reference solutions since only a coarse spatial mesh could be used. In the AEEW(1) case a correction for the coarseness of the mesh had been attempted and it was therefore considered as the best solution for the purpose of discussion. The difference between the JAERI and AEEW(1) results (generally about 1 % in k-effective) could be taken to give some measure of the residual error in the calculational model.

- (b) The AEEW (2) and RISØ (1x1) solutions both used essentially one-mesh-per-pin diffusion theory and, as would be expected, the results fell close together in most cases.
- (c) The ABA and EIR solutions both used a synthetic type of calculation and, here again, the results were generally close together.
- (d) The RISØ solution with 4x4 mesh regions per pin would be expected (on the basis of earlier experience) to give the worst solution and this was generally the case.

In the case of problem (1) the predicted k-eff values varied over a range of 5%. The AEEW(2) and RISØ (1x1) solutions lay within 1% of the "best" value while the ABA and EIR solutions were about 2% low and the RISØ (4x4) was 3-4 % low. The results for problem (3) were very similar with the various solutions falling in the same order. The wide spread of results in these cases was considered disappointing since the problems were relatively simple and the same data were used in all cases.

The two cases with cruciform control absorbers showed different behaviour. Problem (2) which had a very thin black absorber, gave particular difficulties and the spread of k-eff values was again of the order of 5%. In this case EIR and RISØ (4x4) gave better results than AEEW (2) and RISØ (1x1) gave the worst result with a k-eff overprediction of 3-4 %. In problem (4) the spread of k-eff values was reduced and all results except RISØ (4x4) lay within 1%. The ordering of the various predictions, however, differed from that in problem (2) and it seemed prudent to assume that the improved agreement could have resulted from cancellation of errors.

An outstanding question was whether the JAERI and AEEW (1) methods

could be accepted as giving an adequate standard to aid in the improvement of other models or whether a more definitive solution would be needed.

In discussion it was agreed that it would be reasonable to wait for a period of 2-3 months before publishing the results of the benchmark so as to allow any further contributions to be included. It was also agreed that the final version of the report should be submitted for publication in Nuclear Science and Engineering or the Journal of Nuclear Energy provided that the individual contributors were in agreement.

4.2. Fast reactor benchmark

Till introduced the paper NEACRP-A-272 which set out the detailed specifications, prepared by ANL, for the fast reactor benchmark. He said that it had not been possible to meet the committee's original proposal to complete the benchmark exercise before the Chicago Safety meeting and it was now suggested that the results of the calculations should be discussed at a specialists' meeting.

The proposal was based on a simplified version of an optimized design for a typical 1250 MWe oxide-fuelled, sodium-cooled fast breeder reactor. A two-zone cylindrical model was used as a reference core and homogeneous compositions were specified for each region. Further detailing, e.g. the introduction of cell calculations, could, if necessary, be introduced in a second stage of the exercise.

The main aim of the exercise was to determine the influence of the differences in the various current data sets on the calculated reactor physics parameters of the reference core and on

the safety related physics parameters in several variants of the reference configuration.

In discussion members expressed some concern at individual features of the proposal (e.g. omission of fission products, choice of a temperature of 1100 °K for all regions etc.) but it was generally agreed that such simplifications were appropriate for the first stage of the exercise and that a more detailed treatment could be considered in a second stage. The majority of members agreed to participate in the benchmark exercise.

It was agreed that solutions should be sent to Till by 17th March 1977. Till agreed to analyze the the results and arrange a specialist meeting to be held in December 1977. The Secretariat agreed to invite the CSNI to send an observer to the meeting. Campbell agreed to invite Dr. R.D. Smith to approach the Russian member of the International Working Group on Fast Reactors with a view to obtaining a Russian solution of the benchmark.

Küsters invited members to participate in the German 2D and 3D fast reactor benchmark exercise for a 300 MWe fast breeder reactor which was detailed in the paper NEACRP-L-167. This benchmark was intended primarily for methods testing. Solutions should be sent to G. Buckel at KFK.

5. INFORMATION ON MEETINGS RECOMMENDED BY NEACRP

Sensitivity studies and shielding benchmarks (Paris, October 1975)

Farinelli reported briefly on the meeting, an account of which was given in the paper NEACRP-L-150. He felt that this had been a high point in the current series of shielding meetings and that

it had ensured useful discussions at the Vienna meeting in October 1976. A new feature of the Paris meeting had been the setting up of fast and thermal reactor benchmarks for sensitivity calculations. A sodium-cooled fast reactor benchmark was described in NEACRP-L-152 and a PWR benchmark in NEACRP-L-151.

On the experimental side the results of the first four iron benchmark experiments had been given in NEACRP-U-73. It appeared that the comparison of the results of the various experiments would be relatively straightforward since, in most cases, a one-dimensional representation (with imposed transverse buckling) was sufficiently accurate.

The Secretariat agreed to ensure that the full proceedings of the meeting would be given an NEACRP-A distribution.

6. OTHER ACTIVITIES

A tutorial session was held on 23rd June when various aspects of the AECL plutonium fuel programme were described. The session was followed by a visit to the Plutonium Laboratory.

A dinner was offered by the host organization on 22nd June.

ANNEX 1

LIST OF PARTICIPANTS

Members

ASKEW, J.R.	AEE Winfrith, UK
BARRE, J.Y.	CEN Cadarache, France
BUSTRAAN, M.	ECN Petten, Holland
CAMPBELL, C.G.	AEE Winfrith, UK
DURET, M.F.	AECL Chalk River, Canada
FARINELLI, U. (Vice Chairman)	CNEN Casaccia, Italy
HELLSTRAND, E.	AB Atomenergi, Studsvik, Sweden
HEMMIG, P.B.	USERDA Washington, USA
HIROTA, J.	JAERI Tokai-Mura, Japan
INQUE, T.	PNDC Tokyo, Japan
KÜSTERS, H (Chairman)	KFZ Karlsruhe, Germany
MAIENSCHEIN, F.C.	ORNL Oak Ridge, USA
RICHMOND, R. (Secretary)	EIR Würenlingen, Switzerland
TILL, C.E.	ANL Argonne, USA
VELARDE, G.	JEN Madrid, Spain

Observers

BOUCHARD, J.	CEN Fontenay-aux-Roses, France
CROSS, W.	NEANDC, AECL Chalk River, Canada

Secretariat

ROSEN, J.	NEA Paris, France
TUBBS, N.	NEA Paris, France.

ANNEX 2

LIST OF 'A' AND 'L' REPORTS PRESENTED AT THE NINETEENTH MEETING

- NEACRP-A-256/1 "NEACRP Benchmark BWR Lattice Cell Problems"
J.R. Askew, UK
- 256/2 "Review of International Solutions to NEACRP
Benchmark BWR Lattice Cell Problems"
M.J. Halsall, UK
- 257 "A State-of-the-Art Commentary on Fission
Product Decay Data"
M.F. James, UK
- 258 "Review of Fast Reactor Physics Activities
Relevant to LMFBR Programme in PNC, Japan"
T. Inoue, JAPAN
- 259 "Some Reactor Physics Problems Related to LMFBR
Safety"
K. Aizawa, T. Inoue, JAPAN
- 260 "A Study of Anisotropic Diffusion Effect in
Plate Lattice Assembly"
K. Shirakata, T. Iijima, K. Yamagishi, JAPAN
- 261 "Treatment of Cruciform Control Rods in CASMO"
Malte Edenius, SWEDEN
- 262 "Intercomparison of Fission Chamber Standards"
J.E. Sanders, UK
- 263 "Summary of Current NEACRP Views on Fast Reactor
Breeding Assessment"
J.Y. Barré, FRANCE
- 264 "Brief Review of Fusion-Reactor Neutronics in
the U.S."
F.C. Maienschein, US
- 265 "Benchmark Calculation of DT Fusion Machine
ITALY
- 266 "Bidimensional Calculation of Neutron Flux for
JET Outline"
Edmondo Pedretti & Luciano Tondinelli, ITALY

- 267 "Neutronic and Photonic Studies on Fusion Reactor Blankets with Low Lithium and Tritium Inventories"
K.A. Verschuur, H. Brockmann, NETHERLANDS
- 268 "On the Possibility of Using a Plasma Focus Machine as Neutron Source for Cross-Section Measurements by the Activation Method"
R. Abbondanza, E. Pedretti, ITALY
- 269 "On the Prospects of Mixed Fusion-Fission Energy Systems"
Ugo Farinelli, ITALY
- 270 "Radioactivity and Afterheat of FINTOR and Related Problems of Maintenance and Waste Disposal"
EURATOM
- 271 "Sensitivity Analysis of Cross-Section Uncertainties Related to the JET Machine"
ITALY
- 272 "Specifications for an International Comparison Calculation of a Large Sodium-Cooled Fast Breeder Reactor"
UNITED STATES
- 273 "Some Comments on the Impact of Thermal Near Breeders on Uranium Utilization"
C.E. Till, Y.I. Chang, UNITED STATES
- 274 "Simulation of an HCDA Sequence on the ZPPR Critical Facility"
R.E. Kaiser, C.L. Beck, UNITED STATES
- 275 "On the Extrapolation of ZPR Sodium Void Measurements to the Power Reactor"
C.L. Beck, P.J. Collins, M.J. Lineberry, G.L. Grasseschi, UNITED STATES
- 276 "Validation Results for the SDX Cell Homogenization CODE"
D.C. Wade, UNITED STATES
- 277 "Model Problems to Validate the MC²-II/SDX/Benoist Procedures for Generating Cell-Average Broad Group Cross Sections and Anisotropic Diffusion Coefficients in ZPR Plate Cells"
UNITED STATES
- 278 "Belgian Plutonium Recycle Program"
H. Bairiot, J. Debrue, BELGIUM

- 279 "New Developments in Generalized Perturbation
Methods in the Nuclide Field"
A. Gandini, M. Salvatores, L. Tondinelli, ITALY
- 280 "Overview of U.S. Activities in Noise Analysis
For Nuclear Power Reactors"
R.C. Kryter, UNITED STATES
- 281 "The Reactor Physics Problems of LMFBR Safety"
A.R. Baker, UNITED KINGDOM
- 282 "Some Suggestions Relating to Safety Oriented Work
For NEACRP"
F.J. Fayers, UNITED KINGDOM
- 283 "Effects of Fast Breeder Characteristics on
Consumption and Expenditures Related to Electric
Power Generation. An Assessment for the European
Community "
G. Graziani, C. Zanantoni, EURATOM
- 284 "Neutron-Induced Activation in JET" (Chapter V
from document EUR-JET-R5)
- 285 "Some Problems in the Calculation of Fast Reactor
Heterogeneity Effects"
J.L. Rowlands, A.T.D. Butland, UNITED KINGDOM

NEACRP-L-155

a	Australia	j	Japan
b	Belgium	k	UK
c	Canada	l	Austria
d	Denmark	m	Switzerland
e	Euratom	n	Norway
f	France	o	Sweden
g	Spain	p	United States
h	Netherlands	q	Germany
i	Italy		

(NEACRP-L-156 - Changed to NEACRP-A-263

"Summary of Current NEACRP Views on
Fast Reactor Breeding Assessment"
J.Y. Barré, FRANCE

- 157 "Problèmes Lies à l'Irradiation Neutronique dans
les Reacteurs Thermiques et Rapides"
J.P. Chaudat, M. Darrouzet, FRANCE
- 158 "Etudes de Fluctuations Neutroniques dans les
Centrales Nucléaires à Eau Pressurisée"
P. Bernard, J.C. Carré, FRANCE
- 159 "Calculs des Antiréactivités des Systèmes de
Contrôle pour les Réacteurs de la Filière à
Eau Légère et de la Filière à Neutrons rapides"
J. Bergeron, Ph. Hammer, FRANCE
- 160 "Review of Physics Problems Related to Fast
Reactor Safety"
A. Khairallah, FRANCE
- 161 "Specialists' Meeting on Control Rod Measurement
Techniques: Reactivity Worth and Power Distribution"
J.Y. Barré, FRANCE
- 162 "The Analysis of Control Rod Experiments in Zebra
Core 12 and Implications for Power Reactor Calculations"
A.M. Broomfield, M.D. Carter, P.J. Collins,
J. Marshall, J.L. Rowlands, UNITED KINGDOM
- 163 "Experimental Studies of GCFR Safety Physics
Parameters in the ZPR-9 Critical Assemblies"
S.K. Bhattacharyya, R.B. Pond, E.M. Bohn,
L.G. LeSage, UNITED STATES
- 164 Re-numbered as NEACRP-A-285

- 165 "Response Distributions of ^6LiF and ^7LiF Thermo-
luminescence Dosimeters in Lithium Blanket
Assemblies"
H. Meakawa, J. Kusano, Y. Seki, JAPAN
- 166 "Neutronics of Laser Fission-Fusion Systems"
SPAIN
- 167a "A Fast Reactor Benchmark Problem in Two and
Three Space Dimensions"
G. Buckel, R. Froehlich, K. Kufner, B. Stehle,
GERMANY
- 167b "Activities Concerning the INR Fast Reactor
Benchmark Problem"
G. Buckel, K. Kufner, GERMANY
- 168 "Sodium Flow Velocity Measurements by Correlation
of Thermocouple Signals"
J. Benkert, C. Mika, K.-H. Raes, D. Stegemann,
GERMANY, J.E. deCarlos, I. Melches, A. Perex-
Navarro, F. Verdaguer, B. Zurro, SPAIN
- 169 "Near Breeding Thorium Fuel Cycle in the Pebble
Bed HTR"
E. Teuchert, H.J. Rutten, GERMANY
- 170 "Homogenization Problems in Thermal and Fast
Reactors"
H. Küsters, GERMANY
- 171 "The Calculation of Control Rod Worth"
H. Giese, F. Helm & G. Henneges, GERMANY
- 172 "Reactor Physics Aspects Related to LMFBR Safety"
H. Küsters, GERMANY
- 173 "Experimental Determination of Local Steam Bubble
Velocities in BWR Fuel Elements"
D. Ceelen, P. Gebureck, D. Stegemann, GERMANY
- 174 "Reactor Power Noise Activities in Germany"
a. "A Simple Analysis Method for Measuring Time Power
Spectral Densities and Coherence Functions in a
Large Frequency Range"
W. Văth,
b. "Detection of Sodium Boiling in LMFBRs by Monitoring
Neutron Noise Signals for Oscillatory Components"
J. Ehrhardt

NEACRP-L-174

- c. "Two On-Line Methods for Routine Testing of Neutron and Temperature Instrumentation of Power Reactors"
M. Edelmann
- d. "Comparison of Flow and Temperature Signals at Subassembly Outlet with respect to the detection of Flow Anomalies"
P. Hoppe, F. Mitzel
- e. "Comparison of In- and Ex-Core-Neutron Detector Signals Measured in the Sodium Cooled Reactor KNK I"
P. Hoppe, F. Mitzel

175 "Results and Main Conclusions of the IAEA-Advisory Group Meeting on Transactinium Nuclear Data, held at Karlsruhe, November 1975"
H. Küsters, M. Lalovic, GERMANY

NEACRP-L-177

"Determination of Neutron Spectra and Cross-Section Sensitivity of Tritium Production in a Lithium Sphere".
U. Fritscher, F. Kappler, D. Rusch, H. Werle,
H.W. Wiese, GERMANY

PRELIMINARY AGENDA FOR THE TWENTIETH
MEETING

Part A : EXECUTIVE SESSIONS

1. a. Participants in the meeting
b. Committee membership
2. Adoption of the final summary record of the nineteenth meeting
3. Adoption of the agenda of the meeting
4. Completion of actions
5. a. Activities of Centres of interest to the NEACRP
b. Designation of NEACRP observers to the next meetings of these Centres
6. Designation of NEACRP observer to the 20th NEANDC meeting
7. Arrangements for the 21st meeting of the Committee
8. Other matters
9. Election of Committee officers

PART B : TECHNICAL SESSIONS

1. New Topics

- 1.1. Gamma heating (including TLD) and coupled neutron-gamma calculations.
(Euratom, US, UK, Germany, France)
- 1.2. Heterogeneous fast core studies (including thermohydraulic aspects). (France, UK, Germany, US)
- 1.3. Review of sensitivity analysis methods
(Italy, US, France, Euratom, Germany)
- 1.4. Sodium voiding (UK, France, US, Germany)

1.5. Establishment of power peaking margins in power reactors. (UK, France, Germany)

1.6. Miscellaneous topics

2. Topics carried over from previous meetings

2.1. Reactor physics problems related to LMFBR safety (US, Germany, UK, France)

2.2. Problems related to burn-up and fuel cycle in fast and thermal reactors. (France)

3. National Programmes

3.1. Review of recent activities, national programmes, discrepancies, evaluation work

4. Benchmarks

4.1. Fast reactor benchmarks

4.2. Hydrogen entry benchmark

5. General

5.1. Compilations

5.2. Highlights of recent meetings of interest to NEACRP

5.3. Any other business
