EUR 4470 e

COMMISSION OF THE EUROPEAN COMMUNITIES

EURATOM

PROCEEDINGS OF THE CONFERENCE ON ANALYSIS OF FEW ROD EXPERIMENTS IN REACTOR PHYSICS

JRC Ispra (Italy), May 12-14, 1969

edited by S. TASSAN

Directorate General Joint Research Centre
Reactor Physics Department
Experimental Neutron Physics

1970

Published by the Commission of the European Communities
Directorate General for Dissemination of Information
Centre for Information and Documentation — CID

Brussels, March 1970

CONTENTS

FOREWORD

The need to investigate expensive and not readily available fuel elements (such as Pu, U 233 - Th, etc.), together with an improvement in experimental and calculation techniques, have led to an increase in interest in so called "few-rod experiments"* during the last few years.

In these experiments the nuclear properties of a reactor lattice are inferred from measurements made using only a very limited portion of it (in the extreme case, only one fuel element).

In the "substitution" experiments the properties of the core under investigation are compared with the known properties of a reference core.

The extensive use in various laboratories of the "progressive substitution" technique for buckling measurements in thermal systems has produced considerable data, thus permitting an appreciable reduction in the number of test fuel elements used in the experiments (sometimes to five elements and in extreme case to a single element). The problem is to determine the possible extent of this reduction when the nuclear properties of test and reference regions are markedly different.

Progressive substitution methods are also used to measure differential reactivity effects, such as fuel-channel temperature and void coefficients.

In recent years increased attention has been devoted to "zoned-core" experiments, e.g. for the study of Pu-fuelled fast reactors. A delicate aspect of this technique is to define the minimum mass of the inner test zone which will meet the experimental requirements.

Differential reactivity determinations by single-rod substitutions have been successfully developed in calibrated form.

---

* Although, for fast cores, the term "few-rod" is incorrect, it is retained as a uniform definition of this type of experiment.

In these experiments the properties of the test fuel element are compared with the known properties of an element (the calibration standard) having the same geometrical, chemical and physical characteristics, except for a small change in the parameter whose reactivity effect is investigated, e.g. : isotopic composition, temperature, coolant composition and density, etc.

To this category belong measurements with synthetic Pu-U and irradiated fuel elements, as well as experiments set up to evaluate fuel-channel temperature and void coefficients.

In order to evaluate these differential effects with sufficient accuracy, sophisticated techniques, such as reactor oscillation, have been successfully developed.

A special type of calibrated measurement is the so called "void-zero-reactivity" method, developed for the evaluation of the k-inf of a lattice cell. Here the calibration standard is a void cell, which in suitable environmental conditions can be regarded as a medium with k-inf = 1.

In the analysis of these experiments the main problems come from the environment, which must be chosen in order to simulate the infinite lattice conditions of the test fuel elements.

Experiments where the nuclear characteristics of a fuel rod embedded in a moderator are directly inferred from measurement of the flux distribution around the rod, have received less attention due to the difficulties connected with the accuracy of the measurements and the extrapolation of the results to the case of a lattice cell.

The theoretical methods of analysis of the outlined experiments developed in recent years, are of a very different kind, ranging from those based on perturbation and correlation type models to very sophisticated ones. Either the "homogeneous" and "heterogeneous" approaches are being used.

The objective of the meeting is to review the status of development of research on thermal as well as fast reactor systems, i.e. : the most significant features of the experimental techniques (e.g. : accuracy and range of applicability), theoretical models elaborated for the interpretation of the experiments, comparison of results obtained under similar conditions in different laboratories; as well as to indicate the trends for future developments.

PROGRAMME

Monday, May 12th

9.30 - 10.30	Welcoming address (V.RAIEVSKI) Organization of Panel
10.30 - 12.30	Substitution experiments and zoned loadings (Chairman : R.PERSSON)
13.00	Lunch
14.30 - 17.30	Substitution experiments and zoned loadings (continued)

Tuesday, May 13th

9.30 - 12.30	Single-rod experiments (Chairman : B.LAPONCHE)
13.00	Lunch
14.30 - 17.30	Void-zero-reactivity experiments (Chairman : J.SANDERS)

Wednesday, May 14th

9.30 - 12.30	Conclusion. Comparative merits of experimental techniques (Chairman : R.RICHMOND)
13.00	Lunch
14.30 - 16.00	Visit of main facilities for few-rod experiments existing in the Center.

LIST OF PARTICIPANTS

1.	F.ACCINNI	CISE	Milano	Italy
2.	J.ASKEW	UKAEA	Winfrith	G.Britain
3.	T.AUERBACH	Eidg.Institut	Wurenlingen	Switzerland
4.	M.BOUCHARD	CEN	Saclay	France
5.	F.CASALI	CNEN	Bologna	Italy
6.	G.CASINI	Euratom CCR	Ispra	Italy
7.	A.GRÜN	Siemens	Erlangen	Germany
8.	W.HAGE	Euratom CCR	Ispra	Italy
9.	F.HELM	KFZ	Karlsruhe	Germany
10.	J.KOBAYASHI	Japan Atomic Energy Instit.	Tokai Mura	Japan
11.	P.KÖHLER	Eidg.Institut	Wurenlingen	Switzerland
12.	B.LAPONCHE	CEN	Saclay	France
13.	R.MARTINELLI	CNEN	Casaccia	Italy
14.	M.MEYER HEINE	CEN	Cadarache	France
15.	A.OKAZAKI	AECL	Chalk River	Canada
16.	L.PAPAY	MIT	Boston	USA
17.	R.PERSSON	AA	Studsvik	Sweden
18.	R.RICHMOND	Eidg.Institut	Wurenlingen	Switzerland
19.	J.SANDERS	UKAEA	Winfrith	G.Britain
20.	E.SOKOLOWSKI	AA	Studsvik	Sweden
21.	B.STURM	Euratom CCR	Ispra	Italy
22.	S.TASSAN	Euratom CCR	Ispra	Italy
23.	G.VELARDE	JEN	Madrid	Spain
24.	D.WARDLEWORTH	UKAEA	Winfrith	G.Britain
25.	R.YUMOTO	Japan Atomic Energy Instit.	Tokai Mura	Japan

SUMMARY OF SOME FEW-ROD EXPERIMENTS MADE AT THE CHALK RIVER NUCLEAR LABORATORIES

- A. Okazaki

I. Introduction

At Chalk River two types of few-rod experiments are made to obtain lattice parameters of heavy-water moderated lattices. In the first type the buckling is determined from critical substitution experiments in which the changes in moderator critical height are measured when one or more rods of the reference lattice are replaced by test rods. In the second type detailed cell parameters such as the neutron density distribution, conversion ratio, fast fission ratio, and neutron spectrum parameters are measured in a region containing seven or more test rods located at the center of a lattice of driver or reference rods.

Experiments were made with natural uranium rods whose buckling and other lattice parameters had been measured previously in critical cores containing a large number of these rods. To try to find the conditions for which the analyses are valid a wide range of fuel rod size, lattice pitch, and coolant were studied. Both uranium metal and UO_2 fuel rods, with fuel cross-sectional area ranging from 8 to 43 cm^2 were used. Heavy water, organic and air coolants were studied. The lattice pitches ranged from 18 cm to 32 cm.

The measurements and results are described in the paper[1] submitted to the panel and in an earlier report[2].

II. Substitution Measurements

The measurements were made in the critical facilities, ZEEP and ZED-2, with the reference rods listed in Table I. The ZEEP rods are natural uranium metal, 3.25 cm diameter, housed in 1 mm thick aluminum tubes. The CANDU rods are made of 19-element bundles of UO_2 pellets 1.42 cm diameter contained in zircaloy tubes.

The following natural uranium test rods were studied: 19 U(D_2O) (i.e. 19-element uranium metal bundles with D_2O coolant), 7 UO_2(D_2O), 7 UO_2(Air), 7 UO_2(HB40), 19 UO_2(D_2O), 19 UO_2(Air), 19 UO_2(HB40), 28 UO_2(Air), 28 UO_2(H_2O), and ZEEP. HB40 is a liquid organic with a carbon to hydrogen atom ratio of 18:22 and a density of about 1.00 g/cm^3. The volumes of the rod constituents are given in Table II.

In Reference Lattices I - III, up to 7 test rods were substituted for reference rods while in Reference Lattices IV - VIII, up to 25 rods were used.

The critical height changes were analyzed with a two-group heterogeneous line source-sink model using the computer code MICRETE[3]. In this model the rods are treated as line sources of fast neutrons and line sinks of thermal neutrons. The distribution of thermal neutrons and slowing down of fast neutrons are described by diffusion theory. The rod properties were calculated with a two-group cell code (POOOF). The outer radius of the reflector and the resonance escape probability, p, of the reference rods were adjusted in the MICRETE calculation to give the measured reference crtical height and buckling. Then p of the test

rods was adjusted to give the measured critical height changes upon substitution.

In principle the buckling of the test rods should be independent of the number of test rods substituted. For some cases in Reference Lattices III - VII this was obtained. However in general there is a dependence on the number of rods with the buckling approaching the correct value with increasing number of test rods. An empirical extrapolation method was used in which the bucklings were plotted as a function of $\frac{1}{N}$, where N is the number of test rods. For Reference Lattices IV - VIII, where both the test and reference rods were clusters and the pitches were greater than 20 cm, a straight line drawn through the points extrapolated to bucklings in good agreement with the correct values. The slope of the line decreases with increasing lattice pitch. Similarly for Reference Lattice III in the ZED-2 reactor the extrapolated value agrees in most cases with the buckling obtained previously by flux mapping in large cores. However in most of the cases of Reference Lattices I and II this empirical extrapolation does not yield the correct values. It should be noted that in these lattices the pitches are small, there is a large difference in the sizes of the test and reference rods, and there are large differences, up to 7 m^{-2}, in buckling of test and reference rods.

Measurements have also been made on 7-element UC rods as part of a joint Euratom - AECL program. These were made in Reference Lattices IV - VII and the bucklings showed a $\frac{1}{N}$ dependence.

It is felt that for lattice pitches greater than 20 cm and for test rods similar in size and buckling to those of the reference rods, the MICRETE analysis and empirical $\frac{1}{N}$ extrapolation method can be used to give bucklings to within 0.1 m^{-2}. An improved version of MICRETE which takes into account the finite size of the rods and allows the test and reference rods to have different slowing down areas, will be used to reanalyze these measurements.

III. Cell Parameter Measurements

If the neutron spectrum at the boundary of the central cell of the 7-rod test region is the same as that in a large core of test rods, then it is expected that the reaction rates measured in the central test rod should be the same as those measured in a full lattice of test rods. The Westcott spectrum parameters $r\sqrt{\frac{T}{T_o}}$ and T were determined at the central cell boundary position midway between the central rod and a nearest neighbor rod from In^{116}/Mn^{56} and Lu^{177}/Mn^{56} activity ratios. The measurements were made in Reference Lattices I - III. The neutron temperatures in all these lattices were the same as those obtained in full lattices showing that thermal equilibrium is established with seven test rods down to the 18 cm spacing. In the 18 cm lattice the epithermal parameter, r, at the central cell boundary was lower than the full lattice value. In Reference Lattices II and III where there is less mismatch in spectra of the test and reference rods, the r-values were the same as in the full lattices.

Measurements of the relative conversion ratio, fast fission ratio, $r\sqrt{\frac{T}{To}}$ in the fuel, and the flux distribution through the cell were made in the central cell of the seven 19 UO_2 (D_2O) rod test core in Reference Lattice I. The results were in reasonable agreement with the full lattice measurements.

These results show that cell parameters can be obtained from seven rod test cores even with the large mismatch in spectrum present in the 18 cm Reference Lattice I experiments. However it is clear that one should try to match spectra by choosing reference rods which are similar to the test rods in both size and fuel content.

REFERENCES

(1) A. Okazaki, "Few-Rod Experiments Made at the Chalk River Nuclear Laboratories", CRNL 282 (1969).

(2) A. Okazaki and D.S. Craig, "Determination of Lattice Parameters Using a Few Rods", AECL 2593 (1967).

(3) J.D. Stewart, J.M. Kennedy and Mrs. S.J. Cowley, "MICRETE: A G-20 Program for Calculation of Finite Lattices by the Microscopic-Discrete Theory", AECL 2547 (1966).

TABLE I

REFERENCE LATTICES

Reference Lattice	Reactor	Reference Rod	Lattice Configuration	Lattice Pitch (cm)	Number of Rods
I	ZEEP	ZEEP	Triangular	18	121
II		ZEEP	Triangular	22	85
III	ZED 2	CANDU	Triangular	22	121
IV	ZED 2	CANDU	Square	20	121
V		CANDU	Square	24	121
VI		CANDU	Square	28	97
VII		CANDU	Square	32	69
VIII	ZED 2	CANDU	Square	27.94	97

TABLE II

VOLUMES OF ROD CONSTITUENTS PER UNIT LENGTH (cm^3/cm)

	ZEEP	7 UO_2	19 UO_2	19 U	28 UO_2
Total	9.6	55.8	85.7	62.1	127.5
Fuel	8.3	30.6	29.1	25.7	42.7
Coolant	-	17.4	19.5	20.1	31.2
Al	1.1	6.7	11.7	13.4	16.2
Zr	-	-	4.4	-	6.1
Air	0.2	1.1	21.0	2.9	31.4

1. INTRODUCTION

Work carried out in Switzerland in the field of few-rod experiments is concentrated on the reactor PROTEUS which has been in operation at the Federal Institute for Reactor Research, Würenlingen, since January 1968. This reactor was designed to allow measurement of the balance of neutron absorption and production in a reactor lattice using minimal quantities of fuel and moderator. A simplified horizontal section of the reactor is shown in Figure 1. The lattice under investigation is constructed in a central zone 160 cm high and 120 cm in diameter and the neutron balance measurements are made in the central cell of this zone. Surrounding the central zone - both axially and radially - is a graphite-moderated driver zone which contains sufficient enriched uranium oxide fuel to drive the reactor critical. The reactor design is thus basically orientated towards the few-rod experiment since, even when the entire test zone contains fuel identical with that in the central test cell, the total amount of test fuel required is still an order of magnitude less than that needed to build a critical test lattice.

The reactor was constructed for initial use with D_2O-moderated test lattices and the experimental work to date has been restricted to lattices of this type. In parallel with this work a design study has been carried out for the conversion of the reactor to a mixed critical system with a central fast reactor zone surrounded by a thermal driver. It is expected that the reactor will begin operation in this form at about the end of 1969.

The Swiss Programme in the Field of Few-Rod Experiments

R. Richmond
Eidgenössisches Institut für Reaktorforschung
Würenlingen / Switzerland

2. SCOPE OF THE EXPERIMENTS

Experiments of the single-element type have been carried out using a test lattice containing an air-cooled cluster of 37 natural UO_2 rods, 13 mm in diameter arranged in a hexagonal array with 18 mm between centres (Figure 2a). The clusters were placed on a hexagonal pitch of 26 cms giving a total of 19 clusters in the test zone. The measurements were carried out in three stages with the following fuel configurations in the central column:

1) A cluster identical to the test lattice cluster.

2) A cluster containing the same fuel as the test lattice cluster but arranged in the ring geometry shown in Figure 2b. This fuel geometry is typical of the system cooled by boiling light water i.e. the Swiss UoV or British SGHW system. The rather small mismatch between the central cell and the test lattice cell in this case was introduced for the practical reason that the hexagonal clusters could be produced more quickly and economically than the ring-form cluster.

3) A central test cell of length 35 - 70 cm containing a ring-form cluster with the same basic geometry as the natural UO_2 cluster but with 10 mm diameter fuel rods containing 3 % U235. The axial buffer elements placed above and below the test cell contained the ring-form natural UO_2 cluster. Measurements on this test cluster were made so that the performance of proposed methods of analysis of single rod experiments could be investigated under conditions in which a considerable mismatch exists between test cell and test lattice. The test cell was used first with air coolant, then with H_2O coolant since the introduction of the H_2O produces a significant change in the resonance flux mismatch between test cell and test lattice, and finally with the H_2O coolant poisoned by boron so that the excess multiplication of the test cell was roughly the same as that of the test lattice. In the two latter cases the axial

buffer elements also contained H_2O as coolant.

The measurements carried out on these various fuel configurations were planned to give as complete a picture as possible of the balance of neutron absorption and production in the central test cell. In all cases the relative values of U238 capture, U235 fission and U238 fission in the test clusters were measured. In the case of the ring-form natural UO_2 cluster a null reactivity measurement was carried out by poisoning the lattice to unit K-infinity. The relative fuel reaction rates were also measured in the poisoned lattice. For the 3 % UO_2 fuel cases measurements were made of the reactivity worth of each test cluster.

In the following sections a description is given of each type of measurement and of the theoretical analysis of the results.

3. REACTION RATE RATIO MEASUREMENTS

3.1. Experimental Methods

The main effort has been devoted to the measurement of the ratio of U238 capture to U235 fission since these two reaction rates typically cover 80-90 % of total neutron absorption in a uranium fuelled lattice. The results to date have been obtained using a fairly standard approach which follows the general lines of the method proposed by Tunnicliffe (1963). The quantity measured is the relative conversion ratio (RCR) defined as the ratio of U238 capture to U235 fission in a lattice relative to the corresponding ratio in a thermal neutron spectrum. The method proceeds by irradiating foils of fuel composition in the rods of the test cluster and also in the PROTEUS thermal column. After irradiation the relative fission product γ-ray intensity from each foil for γ-ray energies greater than 600 keV is measured using NaI spectrometers. Simultaneous irradiation of depleted uranium foils in the fuel

allows the fraction of U238 fission product γ-ray to be determined and hence the U235 lattice-to-thermal fission ratio may be obtained. The corresponding ratio for U238 capture is derived from the relative intensity of the Np239 activity induced in the foils. This is determined by detecting the coincident emission of a 106 keV γ-ray and Pu239 K-X-rays (99.5 keV and 103.7 keV) which occurs in the Np239 decay. The same NaI detectors are used as for the fission ratio measurements. The foils used are relatively thick (1 mm) in order to reduce the variation in exposed foil surface - and therefore of U238 resonance capture-caused by accidental misplacement of a foil relative to the adjacent sections of the fuel rod. Since it is not possible to carry out coincidence counting directly from foils of this thickness, the foils are dissolved and appropriately diluted solutions are made up for coincidence counting.

The relative conversion ratio measurements were restricted in all cases to the central plane of the test cluster. Systematic errors in the measurement were in the range 0.7-0.9 % and random errors 0.4-0.5 %.

The intensity ratio of the fission product γ-rays from U238 and U235 is obtained in the course of the RCR measurement and this can be converted to the U238/U235 fission ratio by a calibration measurement using a double fission chamber. A calibration measurement of this kind is now in progress and this will allow fission ratios to be quoted for all the test clusters.

In the case of the 3 % UO_2 test cells the measurements of U238 and U235 reaction rates were supplemented by measurements of the relative reaction rates of Cu63, Au198 and Lu176 in the test cell and neighbouring cells. These measurements give additional possibilities of checking the predictions of proposed methods of analysis.

3.2. Analysis of the Results of the Relative Conversion Ratio Measurements

When the central cell is identical with the test lattice cells the neutron spectrum in this cell approximates very closely to that in a single-zone critical assembly with the composition of the test lattice. This was checked by means of a calculation for a homogeneous model of the system using the programme TWENTY GRAND (Tobias and Flower, 1962) with input data from METHUSELAH II (Brinkworth, 1966). This showed, for example, that with the test lattice poisoned to unit K-infinity the ratio of resonance flux (5.53 keV - 0.625 eV) to thermal flux lies within 0.1 % of the METHUSELAH II cell value over the volume of the test cell. This result can also be applied to the unpoisoned case where the greater reactivity of the test lattice will tend to improve the spectrum match. The results of the RCR measurement made in this case may therefore be compared directly with the predictions of a cell calculation for the test lattice. For this comparison we have used the codes METHUSELAH II and WIMS (Askew et al. 1966). The version of WIMS known as WIMSD was used.

For the analysis of the experiments in which the central test cell differed from the remainder of the test lattice cells we have followed the lines of the HPS scheme proposed by Askew and Pitcher (1967). The intention is to use the WIMS-based version of this method but this has been delayed by difficulties in the conversion of WIMS to run on a local computer and, as an interim measure, we have therefore used a version based on the codes METHUSELAH II and CRAM (Hassitt, 1962). In this method METHUSELAH cell calculations are carried out for the test cell and the test lattice cell and these provide flux and cross-section data for a one-dimensional CRAM calculation which has as input:

a) In the test cell - the average cross-sections in the various cell regions together with the cell edge fluxes of the test lattice cell.

b) In the test lattice region - the cell-edge-normalized test lattice cross-sections.

The calculation gives the perturbed fluxes $\Delta\Phi$, which are superimposed on the test lattice cell edge fluxes to obtain the total flux in each of the five METHUSELAH energy groups. The calculation is cut off at a boundary within the test lattice at which the condition $\Delta\Phi = 0$ is imposed. A preliminary calculation using the RIFIFI code (Amouyal et al. 1960) in the version by Zumbrunn (1961) was used to estimate the boundary radius which was set at 38 cm. It was found, however, that changes of $\pm$ 8 cm in this radius produced no significant changes in the test cell neutron fluxes within the $\pm$ 0.1 % error limits of the CRAM calculation.

The HPS type of calculation refers to a full length test cell and, although a test cell of this type was used for the natural UO_2 ring cluster measurements, the 3 % UO_2 cluster measurements were made primarily with a test cell 70 cm in length and, in this case, it was necessary to investigate end effects. This was done using the programme 2DF, a two-dimensional version of the programme 1DF which is essentially the same as DTF IV (Lathrop, 1965). The calculations were carried out in the S4 approximation and, in the majority of cases, two neutron energy groups were used with the intergroup boundary at an energy of 0.625 eV. The input data were derived from METHUSELAH calculations. The reactor representation used in the calculation is shown in Figure 3. The test cells investigated are shown in Table I. The results of the calculations are given in this table in the form of the fast/thermal flux ratio relative to the corresponding ratio at the centre of a full length test cell. The values are quoted in each case for a region 12 cm long at the cell centre and a region 6 cm long at the cell edge. In the case with no buffer fuel the surrounding D_2O layers were retained, corresponding to an experiment in which air-filled pressure tubes are placed above and below the test cell.

Table I: End Corrections in 3 % UO_2 Fuel Clusters

Test Cell		Buffer fuel	Flux Ratio	
Coolant	Length (cm)		Centre	Edge
Air	70	nat UO_2 Air	1.005	0.926
Air	35	nat UO_2 Air	0.988	0.920
Air	70	none	1.002	0.763
H_2O	70	nat UO_2 H_2O	1.012	0.975

These calculations indicate that, from the point of view of reaction rate measurements at the central plane of a 70 cm test fuel cluster, there is no particular advantage in using axial buffer elements. However, the change of spectrum ratio at the edge of the test cluster in the none-buffer case shows that, for reactivity measurements which involve the neutron balance throughout the test fuel, the natural UO_2 buffer elements serve a useful purpose. These elements were therefore used throughout the present measurements.

In the particular case of the 70 cm air-cooled test cluster a 4-group 2DF calculation was carried out to give a more detailed picture of the end correction. This gave the ratio of resonance to thermal flux at the centre of the test cell (relative to a full length test cell) as 1.008, corresponding to an RCR ratio of 1.003. The relative flux ratio at the edge of the test cell was 0.930.

As a check on the calculated values of the end correction the RCR measurement for the air-cooled cluster was made both in a 70 cm and in a 35 cm test cluster. This gave the result:

$$\frac{\text{RCR (70 cm)}}{\text{RCR (35 cm)}} = 1.001 \pm 0.006 \quad \text{(random error)}$$

It was concluded that the additional error in the measured RCR value resulting from end effects in the case of the 70 cm test cluster was not greater than 0.5 %.

The results of the RCR measurements are compared with the theoretical values in Table II. In addition to the METHUSELAH-CRAM values of RCR the table shows the values given by METHUSELAH and WIMS cell calculations with critical buckling spectra. In the WIMS calculations the PIJ routine was used. Some indication of the mismatch between test cell and test lattice cell is given in Table III which shows the values of K-infinity and of the cell-averaged resonance/thermal flux ratio.

Table II: Comparison of Experimental and Calculated Values of Relative Conversion Ratio

Test Cell	Cluster-Averaged Value of RCR in Test Cell				
	Measured	Calculated			
		METH-CRAM	Discrepancy (%)	METH.	WIMS
Nat UO_2 Hexagonal	1.585	-	-	1.544	1.502
Nat UO_2 Ring	1.575	1.549	-1.7	1.519	1.505
Nat UO_2 Ring (K-inf=1)	1.622	1.575	-2.9	1.547	-
3 % UO_2 Air Coolant	2.166	2.031	-6.2	2.108	2.133
3 % UO_2 H_2O Coolant	2.071	2.005	-3.2	1.937	2.025
3 % UO_2 Poisoned H_2O Coolant	2.168	2.151	-0.8	2.072	2.174

Table III: Comparison of Characteristics of Test Cell and Test Lattice

Test Cell	K-infinity		Resonance flux/thermal flux	
	Test Cell	Test Lattice	Test Cell	Test Lattice
Nat UO_2 Hexagonal	1.110	1.110	0.533	0.533
Nat UO_2 Ring	1.105	1.110	0.471	0.533
Nat UO_2 Ring (K-inf=1)	1.001	1.007	0.526	0.588
3 % UO_2 Air Coolant	1.608	1.110	0.727	0.533
3 % UO_2 H_2O Coolant	1.489	1.110	0.419	0.533
3 % UO_2 Poisoned H_2O Coolant	1.092	1.110	0.404	0.533

In considering the implications of the discrepancies between theory and experiment it is useful to note the values of the corresponding discrepancies obtained in measurements carried out at A.E.E. Winfrith by Briggs et al. (1968) on a range of full-scale lattices of the same general type as those used in the present measurements. These are shown in Table IV. It should be noted that Briggs et al. used the DSN routine in their WIMS calculations.

Table IV: Discrepancy between Theory and Experiment in Winfrith RCR Measurements

Coolant	Discrepancy (%)					
	METHUSELAH			W I M S		
	Max.	Min.	Mean	Max.	Min.	Mean
Air	-2.5	-1.0	-1.8	-3.5	-1.7	-2.9
H_2O	-1.5	0.0	-0.6	-2.2	-0.4	-1.5

It can be seen from Table II that, in the case of identical test cell and test lattice the METHUSELAH discrepancy (-2.6 %) is of the same order as that noted in the Winfrith experiments. For the test cells with the natural UO_2 ring cluster the environment correction amounts to only 2 % in RCR and, here again, the difference between METHUSELAH and experiment is in line with Winfrith experience. In the 3 % UO_2 test cell cases the agreement between theory and experiment with H_2O coolant (both poisoned and unpoisoned) is surprisingly good in view of the relative simplicity of the method of analysis. The break-down of the method in the air-cooled case is not unexpected, particularly since this cluster has a rather large ratio of coolant to fuel volume (1.8).

The reason for the poor performance of WIMS in the case of the hexagonal cluster (discrepancy - 5.2 %) is not at present clear. Published comparisons of WIMS predictions with experiment appear to be restricted to ring clusters but the effect of the geometry difference should not be significant, particularly since the PIJ routine was used. If the WIMS RCR value for the natural UO_2 ring cluster is subjected to the environment correction of 2.0 % given by the METHUSELAH-CRAM calculation the resultant value is only 2.6 % below the measured value and this is in line with the normal performance of WIMS.

4. NULL REACTIVITY MEASUREMENTS

4.1. Experimental Method

Null reactivity measurements were carried out using the standard approach of poisoning the test lattice until the exchange of the central test cell with a void cell produced no change in the reactivity of the reactor at which point, under ideal conditions, the K-infinity of the test lattice is unity. (Donahue et al. 1958). To date these measurements have been made only for the natural UO_2 cell with ring geometry.

PROTEUS is equipped with two features which are particularly useful for the performance of a measurement of this kind. The first of these is an arrangement for poisoning (and depoisoning) the lattice by dissolving boron in the moderator. We thus have a homogeneously distributed poison with an absorption cross-section simply related to neutron energy and this facilitates the analysis of the measurements. The second feature is a moveable central fuel column which allows the test cell to be exchanged with the void cell (an empty aluminium container 50 cm in length) in a period of the order of 10 seconds. This improves the accuracy of the reactivity change measurement. The moveable column is shown schematically in Figure 4. The method proceeds by measurement of the reactivity worth of the test cell for two boron concentrations chosen to give reactivity values close to zero. The exact concentration for null reactivity change is then obtained by interpolation or extrapolation from the measured values. In the present case the reactivity changes were determined from period measurements using standard techniques. The error in the reactivity measurement itself corresponds to an error of less than 0.1 % in the K-infinity of the test cell and the error arising from uncertainty in the boron content of the moderator is 0.2 % but the overall accuracy of the result is governed by the various corrections resulting from the departure of the test cell environment from the infinite homogeneous lattice condition in which a null reactivity change exactly corresponds to unit K-infinity. These corrections are discussed in the following section.

4.2. Environment Corrections

a) Driver effects

The TWENTY GRAND calculation mentioned in Section 3.2 indicated that corrections caused by variation of the neutron spectrum from the infinite lattice spectrum will be not greater than about 0.1 % in the K-infinity of the test cell.

b) Mismatch between Test Cell and Test Lattice

The METHUSELAH-CRAM calculations described in Section 3.2 showed that a correction of + 0.5 % must be applied to convert the measured value of the K-infinity of the test cell to the value which would be obtained in a test lattice of identical cells.

c) Heterogeneity Correction

Because the lattice is heterogeneous the introduction of a void cell allows neutrons to stream directly from the cell edge to the exposed surfaces of the buffer fuel elements. This alters the reaction rate distribution in the regions adjacent to the edges of the void cell and a correction must be made for the resultant reactivity change.

Calculations of the heterogeneity effect are currently in progress using 2DF in the S4 approximation with 4 energy groups and with METHUSELAH input data. The two alternative representations of the test region shown in Figure 5 are being used. In each case a zero current boundary condition is imposed at the outer boundary of the test lattice. The calculation is carried out in two stages, first with the buffer fuel in immediate contact with the edge of the void and then with a layer of aluminium representing fuel can end caps, pressure tube flanges etc. between fuel and void. This allows the fuel and aluminium components of the correction to be separated. The correction is determined by integrating the excess neutron absorption and production over the regions adjacent to the ends of the void cell. A preliminary estimate of the correction to be applied to the measured value of K-infinity gives + 0.87 % for

the fuel component and - 0.25 % for the aluminium component. These figures apply to the 50 cm void cell. Some additional measurements were made with a void cell 25 cm long and the heterogeneity effect will also be calculated in this case.
An important component of the heterogeneity effect which is not given by the 2DF calculation arises from the increase in U238 surface resonance capture caused by the exposure of additional fuel surface at the edges of the void can. An estimate of this effect has been made by representing the additional surface in an expression of the form $A+B\sqrt{S/M}$ for the resonance integral of the fuel cluster. The value obtained was - 0.5 %.
In view of the complexity of the heterogeneity effect and the fact that the surface resonance capture component cannot be accurately calculated an experimental check appears to be necessary. We are currently carrying out such a check by making a detailed survey of U238 capture and U235 fission rates in the end region of a buffer fuel cluster.

d) Neutron Absorption in Non-Lattice Components

A correction is required for the effect of neutron absorption in materials not normally present in the lattice cell. The most important of these is the aluminium wall of the void can and the correction was estimated in this case by attaching additional aluminium to the can wall so as approximately to double its weight and measuring the resultant reactivity change. This gave the correction to K-infinity as - 1.8 %. The corresponding corrections for various structural components of the test cell and for the nitrogen in the void cell were estimated as + 1.1 % and - 0.5 % respectively.

e) Location of Void Cell Boundary

The edge of the void cell should, in principle, coincide exactly with the boundary of the lattice cell. In practice a cylindrical void with the equivalent radius of the lattice cell was used.

This was not expected to lead to a significant error but a check was made by using a void cell with a radius 10 % greater than the equivalent lattice cell radius. This indicated a correction not greater than - 0.1±0.005 % in the measured value of K-infinity.

f) Moveable Fuel Column Effects

The moveable column could lead to a number of possible errors in the reactivity measurement arising from e.g. differences in the properties of the various lengths of buffer fuel, variations in the boundary conditions of the reactor produced by changing the position of the column etc. These effects were investigated by making an appropriate series of changes to the column which showed that the resultant correction to K-infinity was not greater than 0.02 %.

g) Total Correction

The corrections to the measured value of K-infinity may be summarized as follows:

Mismatch between test cell and test lattice:	+ 0.5 ± 0.2 %
Heterogeneity:	+ 0.1 ± 0.4 %
Absorption in Non-Lattice Components:	- 1.2 ± 0.4 %
Cell edge effect:	- 0.1 ± 0.05 %
Total:	- 0.7 ± 0.6 %

4.3. Analysis of the Results of the Null Reactivity Measurements

The analysis of the results proceeds by direct comparison of measured and calculated quantities and does not involve derived quantities such as the K-infinity of the unpoisoned lattice.

The result of the null reactivity measurement on the natural UO_2 ring cluster may be expressed in the form that for a boron concentration of 28.6 g/ℓ the K-infinity of the test cell is 1.000±0.006. The METHUSELAH value for the K-infinity of this cell

is 1.001. The measured error of - 2.9±1.0 % in the METHUSELAH value of RCR for this cell implies that, if the remaining quantities in the expression for K-infinity are correctly calculated, then the METHUSELAH value of K-infinity should be 1.012±0.004. This value is not entirely consistent with the result of the null reactivity measurement and the source of the compensating error in the METHUSELAH calculation which leads to a correct prediction of K-infinity is being investigated.

5. REACTIVITY CHANGE MEASUREMENTS

In the experiments on the 3 % UO_2 fuel measurements were made of the reactivity worth of the test cluster with air coolant and with H_2O coolant. Such a measurement characterizes the excess multiplication of the cluster and can be combined with the reaction rate measurements to build up an overall picture of the neutron balance. This fuel is so reactive, however, that a 70 cm cluster produced a reactivity change in PROTEUS of the order of 1.2 % which was too large to be measured with adequate accuracy. A measurement was therefore made of the reactivity worth of the H_2O-cooled cluster with the coolant poisoned by boron to reduce the reactivity change to approximately 0.1 %. In this case the main experimental error arose from uncertainty in the boron content of the coolant and corresponded to an error of the order of ± 0.5 % in the ratio of neutron absorption and production in the test cell. A subsidiary calibration experiment was carried out by determining the reactivity change produced by a small amount of boron dissolved in D_2O and contained in the pressure tube previously used to contain the fuel cluster. The analysis of the results of the reactivity measurements is currently in progress using the METHUSELAH-CRAM approach.

6. FUTURE DEVELOPMENTS

6.1. Thermal Reactor Lattices

In the thermal lattice work the immediate aim is to carry out the WIMS analysis of the measurements on 3 % UO_2 clusters. It is also planned to carry out a series of measurements using a test cluster of mixed plutonium-uranium oxide with a plutonium content of 0.8 %. Measurements will be made of the relative values of the U238 capture rate and the Pu239, U235 and U238 fission rates.

6.2. Fast Reactor Lattices

At about the end of 1969 PROTEUS will be converted for use in a programme of research on fast reactor lattices. A mixed critical design will be used in order to minimize the inventory of fissile material which is expected to be of the order of 100 kg. The design is shown schematically in Figure 6. The use of D_2O and graphite moderators in the driver as opposed to hydrogen was adopted so as to increase the size of the driver zone and allow the retention of the existing safety, shut-down and control rods. (An entirely graphite-moderated driver would not have been sufficiently reactive). The zones of the system working inwards from the driver are (1) a boron shield to exclude low energy neutrons from the fast zone (2) a 15 cm thick buffer zone of natural UO_2 which will have the same scattering properties as the fast zone and therefore produce effectively the same equilibrium spectrum (3) the fast zone, 50 cm in diameter and with a maximum height, including axial blankets, of 140 cm. The fast zone fuel will be in the form of 7 mm diameter rods.

Calculations carried out using the programme EDSN, which is a version of the WDSN programme (Francescon, 1963) with modified input and output facilities, indicate that the value of K-infinity

at the centre of the fast zone will lie within 1 % of the value in a critical fast reactor of the composition of the fast zone while individual reaction rate ratios will lie within 1.5 % of the fast reactor value.

It is planned to carry out measurements of reaction rate ratios at the centre of the fast zone together with neutron spectrum measurements using proton recoil counters. Null reactivity measurements will also be carried out. The moveable central fuel column will allow rapid extraction of irradiated foils and can also, in split form, be used to create the void required for the null reactivity measurements.

7. CONCLUSIONS

The results obtained to date on D_2O-moderated lattices have been subjected only to a preliminary analysis but some tentative conclusions may be drawn at this stage concerning the application of single-rod techniques in this type of lattice. In the first place the measurement of reaction rate ratios at the centre of the test cluster seems to have considerable advantages since this can give a fairly complete picture of the neutron balance in the cluster and is subject to very small end corrections for relatively short cluster lengths (of the order of 50 cm). The latter conclusion appears to be true even when no axial buffer elements are used. Measurements of the reactivity worth of the cluster, though giving an overall measure of the neutron balance are more subject to end effects and may not be sufficiently accurate in the case of highly enriched fuel clusters. The results of the preliminary analysis suggest that, in the case of liquid-cooled clusters it may be possible, even with a large mismatch between test cell and test lattice, to use the single rod measurements to check the predictions of a relatively simple assessment code.

References

Amouyal, A., Bacher, P., Lago, B., Mengin, F.L. and Parker, E. RIFIFI: Méthode de calcul analytique de la condition critique et des flux d'une pile à régions variées en théorie à deux groupes et à une dimension programmée pour le calculateur électronique Mercury. CEA 1398, 1960.

Askew, J.R., Fayers, F.J. and Kemshell, P.B. A general description of the lattice code WIMS. J. Brit. Nucl. Energy Soc., Vol. 5, p. 564, 1966.

Askew, J.R. and Pitcher, H.H.W. Concerning the analysis of single channel substitution measurements. J. Brit. Nucl. Energy Soc. Vol. 6, p. 260, 1967.

Briggs, A.J., Johnstone, I., Kemshell, P.B. and Newmarch, D.A. Further reactor physics studies for steam generating heavy water reactors. Part I: Uniform cluster lattices containing UO_2 or PuO_2/UO_2 fuel. J. Brit. Nucl. Energy Soc. Vol. 7, p. 61, 1968.

Brinkworth, M.J. Methuselah II. A Fortran programme and nuclear data library for the physics assessment of liquid moderated reactors. AEEW-R480, 1966.

Donahue, D.J., Lanning, D.D., Bennett, R.A. and Heinemann, R.E. Determination of K-infinity from critical experiments with the PCTR. Nucl. Sci. Eng. Vol. 4, p. 297, 1958.

Francescon, S. The Winfrith DSN Programme. AEEW-R273, 1963.

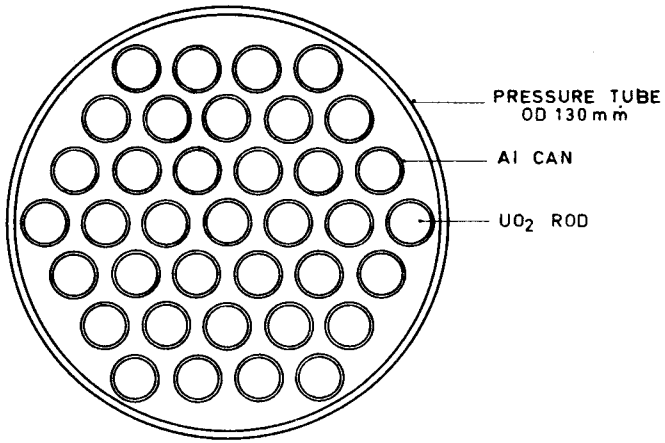
Hassitt, A. A computer programme to solve the multigroup diffusion equations. TRG Report 229(R), 1962.

Lathrop, K.D. DTF-IV, a Fortran-IV programme for solving the multi-group transport equation with anisotropic scattering. LA-3373, 1965.

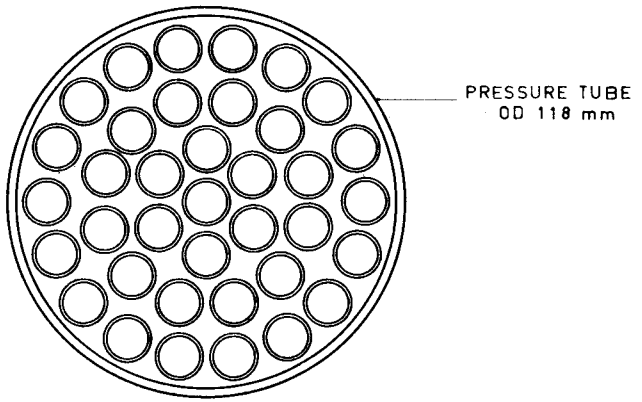
Tobias, M.L. and Flower, T.B. The Twenty Grand programme for the numerical solution of few group neutron diffusion equations in two dimensions. ORNL-3200, 1962.

Tunnicliffe, P.R., Skillings, D.J. and Chidley, B.G. A method for the accurate determination of relative initial conversion ratios. Nucl. Sci. Eng. Vol. 15, p. 268, 1963.

Zumbrunn, K. Der RIFIFI-Code. EIR Internal Report TM-PH-69, 1961.



a) HEXAGONAL GEOMETRY



b) RING GEOMETRY

Fig. 2 - NATURAL UO_2 CLUSTERS

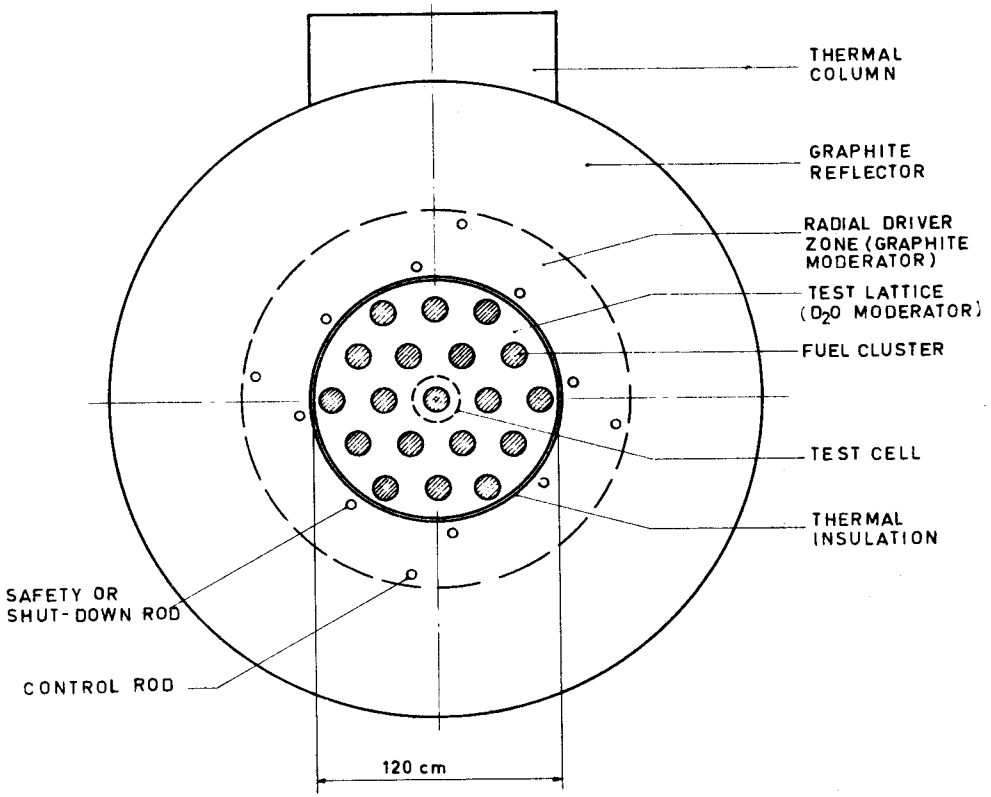


Fig.1-HORIZONTAL SECTION THROUGH THE PROTEUS REACTOR

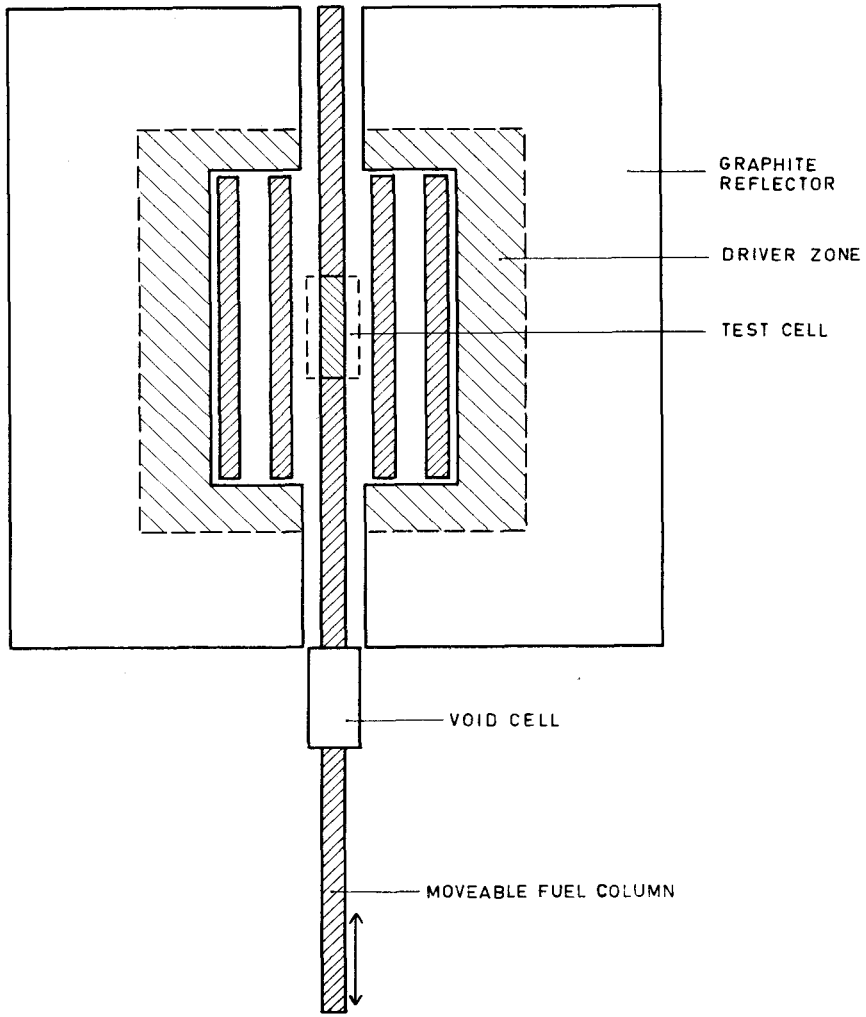


Fig. 4 - VERTICAL SECTION THROUGH THE PROTEUS REACTOR

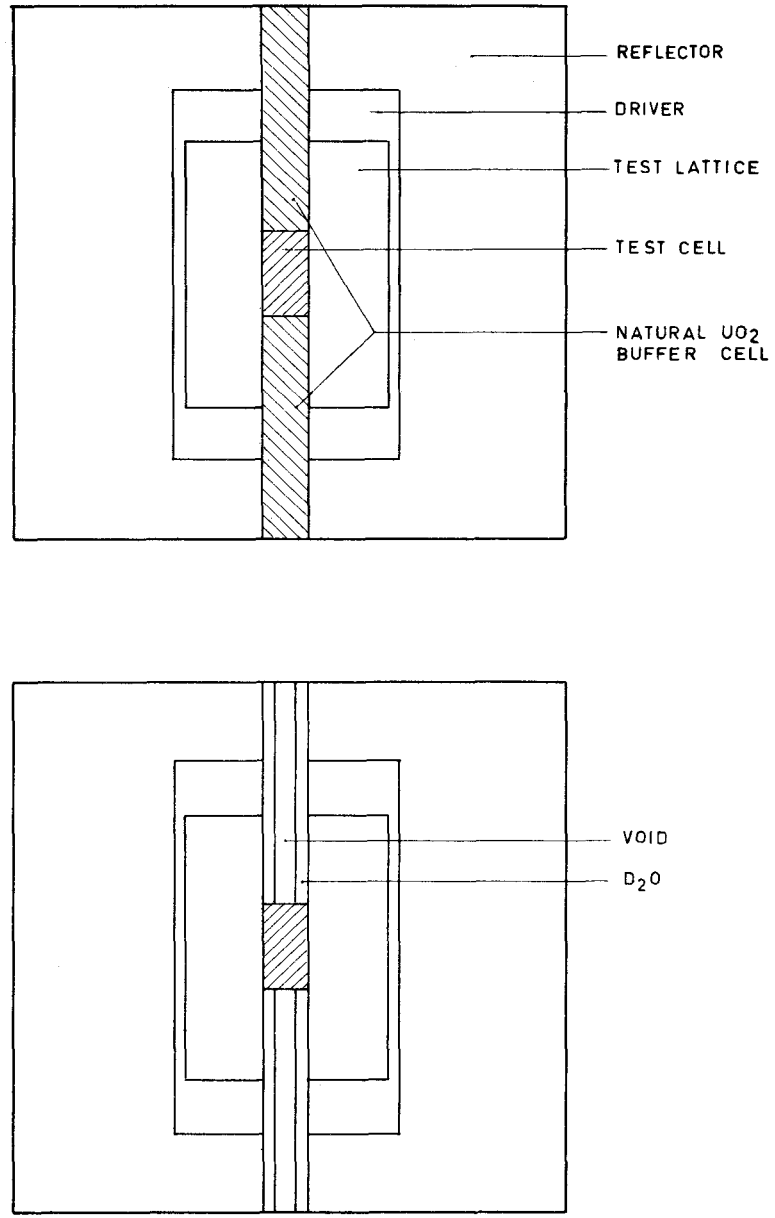


Fig. 3 - REACTOR REPRESENTATION FOR THE CALCULATIONS OF VOID CELL HETEROGENEITY CORRECTION

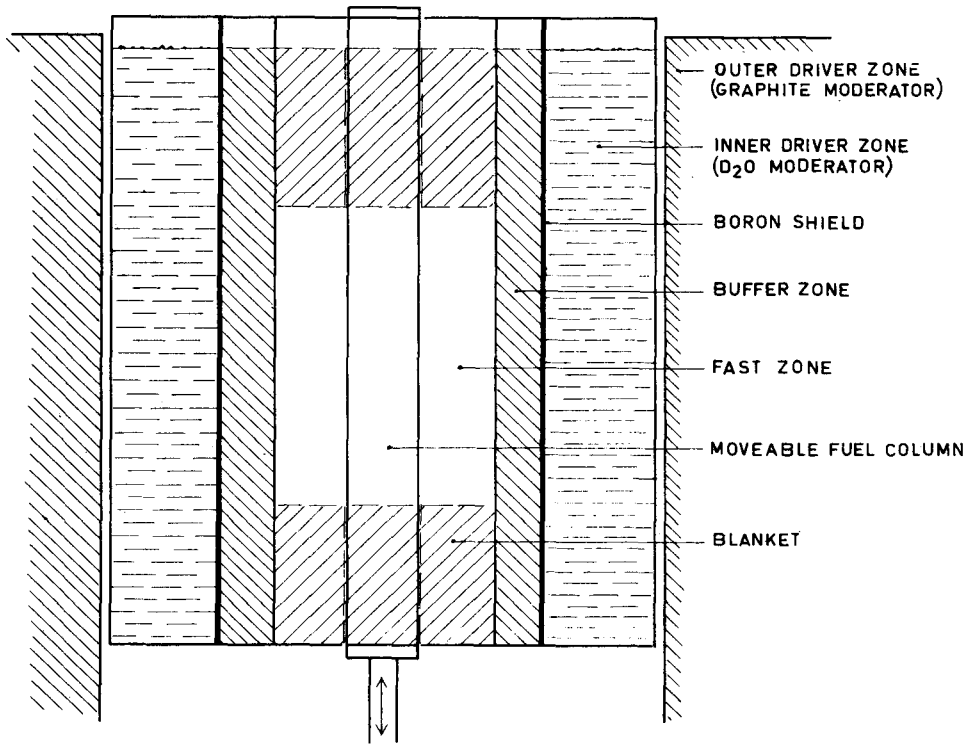


Fig. 6 - PROTEUS MIXED CRITICAL CORE

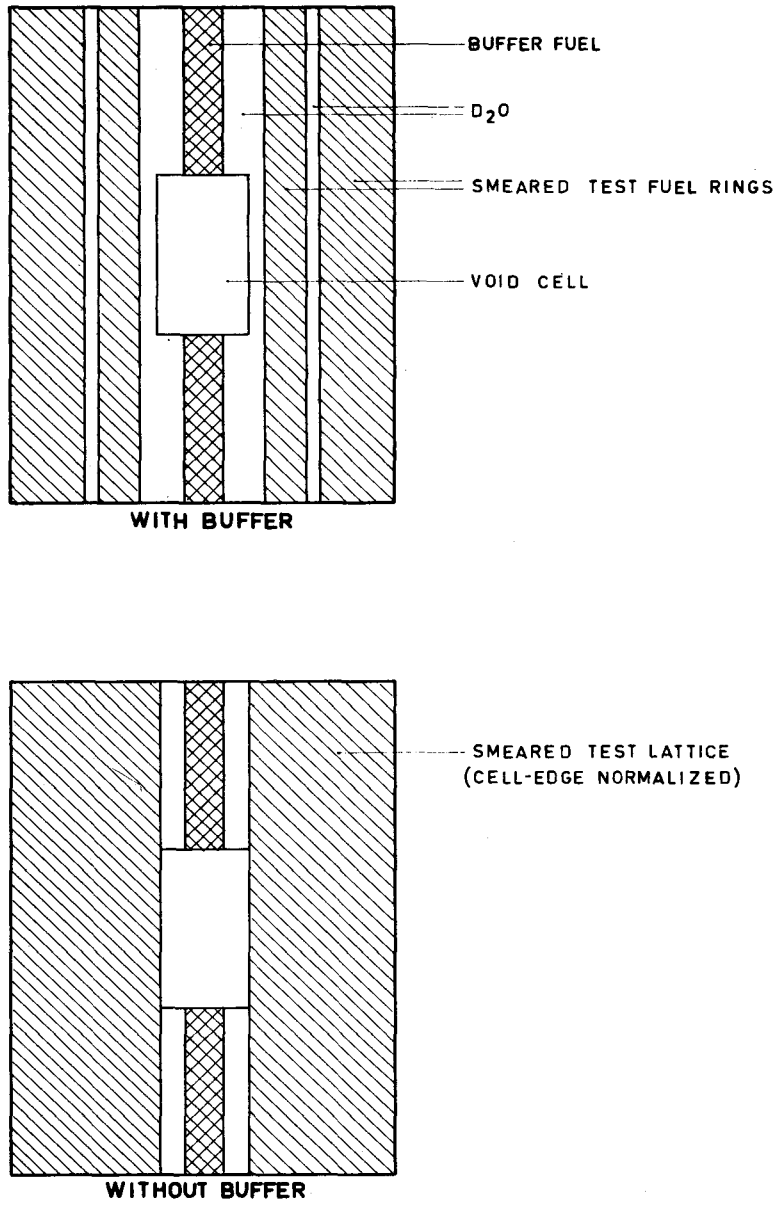


Fig. 5 - REACTOR REPRESENTATION FOR THE CALCULATIONS OF 3% UO_2 CLUSTER END CORRECTIONS

EXPERIENCES UTILISANT DE FAIBLES QUANTITES DE COMBUSTIBLE

PROGRAMMES ET METHODES AU C.E.A.

B. LAPONCHE - R. NAUDET

Essayer de réaliser des expériences mettant en jeu une quantité minimum de combustible a toujours pour motivation première un souci d'économie ; mais on peut avoir deux objectifs un peu différents : on peut vouloir faire, aux moindres frais et dans des conditions commodes, des études systématiques de milieux multiplicateurs ; mais on peut aussi chercher à atteindre plus particulièrement certaines propriétés de ces milieux, en ayant recours à des mesures différentielles.

Au C.E.A. on a très largement exploité ces deux catégories d'expériences : en particulier on a développé des méthodes de substitution progressive pour l'étude systématique des réseaux à eau lourde et à graphite, et on essaie maintenant d'étendre ces méthodes aux milieux à neutrons rapides. D'autre part un important programme consacré à l'étude des combustibles contenant du plutonium dans les réacteurs thermiques nous a amené à développer des méthodes de mesure différentielle ne mettant en jeu que des quantités très réduites de combustibles spéciaux.

On examine rapidement ci-après les méthodes qui ont été développées ainsi que les programmes déjà exécutés ou en cours.

I - SUBSTITUTION PROGRESSIVE DES ELEMENTS COMBUSTIBLES

a) Réseaux à neutrons thermiques

Les toutes premières expériences ont eu lieu en 1956, où on a introduit dans la pile G 1 quelques éléments combustibles de la pile G 2, ce qui a permis de prévoir la première criticalité de cette dernière pile avec une très grande précision.

A partir de l'année suivante, on a entrepris des mesures systématiques par remplacement dans la pile à eau lourde Aquilon. On étudiait uñ nombre restreint de réseaux de base, puis une très grande variété de géométries par substitution des éléments combustibles. Ce type d'expériences a été poursuivi dans Aquilon II à partir de 1960, jusqu'à l'arrêt définitif de cette pile en 1967. Plusieurs centaines de réseaux ont pu ainsi être étudiés.

Une exploration analogue des réseaux graphite a été menée dans l'assemblage critique Marius à partir de 1960, puis à partir de 1965 dans César où le graphite peut être chauffé.

Parmi ces mesures on peut accorder une place spéciale à celles qui ont trait au programme plutonium : alors que dans tous les autres cas on a comparé des géométries différentes, presque toujours en uranium naturel, on a comparé cette fois des réseaux de géométrie identique (barres d'uranium métal ϕ 29.2 mm) mais avec des compositions de combustibles différentes.

Le réseau de base étant en uranium naturel, on a testé :

- uranium légèrement enrichi, ou appauvri en U_5
- jeu 1 P : addition de 0.05 % de plutonium
- jeu 2 P : U appauvri à 0.22 % + 0.30 % de plutonium
- jeu 3 P : mêmes teneurs que 1 P, mais plutonium à 20 % d'isotope 240, au lieu de 6 %.

Les mesures ont été faites à la fois dans Aquilon où on a exploré une gamme de pas de réseaux (12 à 21 cm), et dans César où on a exploré une gamme de températures (20 ° à 450 °).

Enfin on peut noter que des mesures par substitution seront faites prochainement dans Eole, pile à eau lourde, qui comporte une zone nourricière et où on ne peut remplacer qu'un petit nombre d'éléments combustibles.

Méthodes d'interprétation

Plusieurs méthodes ont été développées. La plus ancienne et la plus employée consiste à utiliser des calculs à deux groupes, après homogénéisation et cylindrisation des zones référence et test. Il est nécessaire pour rendre compte correctement des résultats de recourir à un ajustement, car ce modèle traduit imparfaitement les effets de frontière. D'où la méthode de substitution *progressive* qui permet l'ajustement d'un paramètre caractérisant globalement les effets de frontière, par comparaison de résultats successifs. Cette utilisation combinée de plusieurs résultats est ce qui a fait au départ l'originalité de la méthode : à la limite on pouvait considérer qu'on procédait à une extrapolation presque purement expérimentale.

Cette méthode donne des résultats convenables, et elle a rendu de très grandes services ; on ne peut cependant espérer une très bonne précision que si les réseaux comparés sont suffisamment voisins et si la zone substituée n'est pas trop petite. Une analyse critique montre que pour faire de bonnes mesures :

- il faut que les coefficients de diffusion ne soient pas trop différents, c'est-à-dire que les conditions de propagation des neutrons ne soient pas trop perturbées au franchissement de la frontière (la méthode d'homogénéisation rend compte de manière imparfaite de la présence de canaux de dimensions ou propriétés diffusantes différentes).
- il faut que les facteurs p (absorption résonnante de l'uranium 238) ne soient pas trop différents, car les effets de frontière correspondants sont complexes, du fait de l'hétérogénéité, et parce que ces absorptions sont étalées dans un large domaine d'énergie, et on ne peut pas les représenter simplement.
- il est préférable que les laplaciens ne soient pas trop différents, c'est-à-dire que la taille critique ne soit pas trop modifiée par l'introduction des éléments testés, car alors certaines corrections ou sources d'erreurs expérimentales peuvent devenir importantes.

- enfin dans le cas d'éléments contenant du plutonium, pour lesquels on ne peut négliger les effets de spectre dans le domaine thermique, il faut qu'un paramètre global caractérisant le spectre (par exemple l'indice r : rapport entre la densité de ralentissement et la densité totale) ne soit pas trop différent d'un milieu à l'autre.

On peut améliorer grandement la précision des mesures en cherchant à *adapter* les milieux comparés selon les critères qui viennent d'être rappelés. Chaque fois qu'on l'a fait, cet effort s'est révélé extrêmement payant. Par exemple, on a testé dans Aquilon les premiers éléments de série du réacteur EL 4 en les comparant à des combustibles de caractéristiques apparemment extrêmement différentes : pour minimiser les sources d'erreurs, on a modifié la géométrie des tubes, et on a introduit des absorbants supplémentaires, de manière à rendre presque exactement égaux dans les deux milieux les coefficients de diffusion et les laplaciens ; on a pu faire ainsi des mesures précises comme l'ont confirmé des expériences ultérieures sur réseaux complets.

De même, dans le cas des mesures du programme plutonium, on a choisi des supports de même géométrie afin de conserver partout les mêmes conditions de propagation des neutrons, et la même absorption résonnante pour l'uranium 238 ; on a choisi en outre les compositions de manière que l'absorption globale reste du même ordre de grandeur, donc que l'indice spectral r soit peu différent d'une composition à l'autre. C'est grâce à ces précautions qu'on a pu obtenir une très bonne cohérence pour tout l'ensemble des mesures.

On peut d'autre part améliorer l'interprétation en utilisant un modèle plus élaboré. De ce point de vue, la méthode hétérogène est souvent nettement préférable. Cette méthode s'est révélée particulièrement adaptée à l'interprétation des expériences de programme plutonium, même en se contentant d'un outil assez simple (code HECTOR), parce que dans ce cas seuls varient les taux d'absorption et de production de neutrons. On peut en outre tenir compte de la forme des zones substituées mieux que par cylindrisation et également des petites différences éventuelles de composition isotopique ou de poids d'absorbants d'un barreau à l'autre.

Pour pouvoir utiliser la méthode hétérogène dans le cas général, il est nécessaire cependant de disposer d'outils plus perfectionnés. Nous mettons au point un code qui permettra de tenir compte correctement :

- des différences de propriétés de ralentissement des canaux,
- des différences de propriétés diffusantes (par introduction de termes dipolaires),
- d'effets spectraux (par utilisation de fonctions d'influence supplémentaires).

Ce code pourrait être utilisé en particulier pour interpréter les expériences Eole.

Nous pensons que la méthode hétérogène est celle qui a priori fournit les meilleurs modèles d'interprétation pour ce type d'expérience, tout au moins dans le cas des réseaux eau lourde et graphite dont l'hétérogénéité est en général assez grande. Avec une méthode très élaborée, il devait être possible d'interpréter avec précision des expériences où on ne substitue qu'un très petit nombre d'éléments combustibles, même si ceux-ci ont des propriétés relativement différentes de celles du réseau de référence.

Il faut cependant bien voir que si on utilise un modèle compliqué, on renonce par là même à caractériser simplement le milieu testé au moyen d'une propriété globale. Il faut donc choisir : si on ne dispose que d'outils assez rudimentaires, et si on désire en tout état de cause une interprétation simple, il faut absolument essayer de rapprocher, d'adapter les milieux comparés, et concentrer autant que faire se peut l'objet de la mesure sur une propriété particulière ; si les milieux sont franchement différents, et si la zone substituée est petite, on ne peut espérer de résultats précis que si on dispose d'outils suffisamment élaborés, et l'interprétation ne peut plus être très simple.

b) Milieux à neutrons rapides

Dans ce cas, les effets de frontière sont en général très complexes il n'y a plus, comme dans le cas des thermiques, un spectre de base qui est pratiquement le même de part et d'autre, et la nature et l'importance des échanges varient d'une manière mal connue en fonction de l'énergie ; en outre

les perturbations se propagent fort loin. Il n'est donc pas possible d'espérer, dans le cas général, tirer d'une expérience de substitution une grandeur globale caractérisant simplement le milieu testé.

Là aussi - mais d'une manière beaucoup plus stricte que dans les cas des thermiques - il est donc nécessaire que les milieux que l'on compare soient suffisamment voisins, ou pour mieux dire suffisamment adaptés, les critères étant finalement analogues à ceux qui ont été énoncés dans le cas des thermiques : il est important que les coefficients de diffusion (qui sont des fonctions de l'énergie) ne soient pas trop différents, et également que les laplaciens soient voisins, ceci pour minimiser les erreurs d'origine expérimentale ; mais il faut surtout que les spectres directs et adjoints soient suffisamment proches, ceci pour que les effets de frontière soient aussi faibles que possible et qu'on puisse donc les éliminer simplement. (On voit bien que si à la limite les spectres directs et adjoints sont les mêmes dans les deux milieux, un neutron franchissant la frontière donne lieu à des réactions qui ont la même importance, étant entendu que la disparition d'un neutron par effet de fuite est assimilée à une disparition par absorption ; d'où en définitive mise en évidence pour chaque milieu d'un paramètre global B^2, la distribution spatiale étant celle d'une théorie à un groupe).

Dans la pile Masurca, à Cadarache, on a fait des expériences dans lesquelles on remplace des réglettes d'uranium enrichi à 30 %, par des réglettes contenant 25 % de plutonium, le milieu diluant restant inchangé. Dans une première série le diluant était du graphite ; dans une deuxième c'est un mélange d'oxyde de fer et de sodium ; les expériences vont se poursuivre avec d'autres milieux. Il se trouve que chaque fois que l'on fait cet échange entre réglettes U_5 et réglettes Pu au sein d'un même diluant, on obtient des milieux qui non seulement ont des laplaciens très voisins et des coefficients de diffusion presque identiques, mais qui ont également des spectres qui ne sont pas très différents, de sorte que les effets de frontière restent relativement faibles.

Une méthode d'interprétation a été mise au point, basée sur un modèle dit "de synthèse". Dans ce modèle on fait comme si de part et d'autre de la frontière le spectre pouvait être assimilé à une combinaison linéaire des spectres caractéristiques des deux milieux. Pour représenter la distribution des neutrons il faut donc deux fonctions de l'espace, l'une figurant

la variation de la densité totale, l'autre celle d'un coefficient de "mélange" entre les deux spectres, ou ce qui revient au même l'une représentant le "fondamental" et l'autre une perturbation. Formellement le calcul est analogue à un calcul à deux groupes. On peut procéder comme on l'a fait autrefois pour les neutrons thermiques, c'est-à-dire ajuster sur une série de mesures un coefficient caractérisant globalement l'effet de frontière. Il faut noter que non seulement le formalisme est identique, mais le principe même de l'interprétation est semblable : on se donne a priori le spectre de la perturbation et les conditions de sa propagation, et on ajuste sa valeur absolue ; on admet donc que les effets de frontière sont suffisamment faibles pour qu'on puisse se contenter de les ajuster globalement, même si on les analyse mal.

Cette méthode, appliquée à Masurca, a donné de bons résultats par comparaison avec des mesures sur réseaux complets par cartes de flux ; les expériences sont présentées dans une communication de A. Meyer-Heine et P. Caumette.

Il est clair qu'on ne pourrait pas généraliser la méthode dans le cas où les milieux seraient nettement différents ; cependant dans ces limites, son intérêt pratique est considérable, puisqu'il est possible d'obtenir, à partir de mesures effectuées sur les milieux U_5, des renseignements sur les milieux Pu, en ne disposant que de quantités très limitées de plutonium.

II - OSCILLATIONS

Les oscillations représentent un cas limite des substitutions, la portion de milieu substitué étant dans ce cas très petite ; le recours à une technique qui met en jeu la cinétique n'est qu'un artifice destiné à éliminer un certain nombre de causes d'erreurs parasites donc améliorer la précision.

On pense souvent que l'interprétation est facilitée par la petitesse de la zone substituée : il n'y a plus a proprement parler de zone frontière où les propriétés sont intermédiaires entre celles des deux milieux ; l'échantillon se trouve dans un environnement qui est dominé par les caractéristiques du milieu non perturbé.

Nous pensons que cette manière de voir recouvre la plupart du temps une grande part d'illusions, et que pour faire de bonnes mesures, non seulement il est nécessaire de prendre des précautions analogues à celles décrites à propos des substitutions, mais qu'elles doivent être encore beaucoup plus strictes. En particulier, il est extrêmement difficile d'interpréter convenablement des mesures dans lesquelles interviennent des différences notables en ce qui concerne la diffusion des neutrons.

Réseaux à neutrons thermiques

C'est surtout dans le cadre du programme d'étude des combustibles contenant du plutonium qu'on a développé des techniques et effectué un nombre considérable de mesures. Par la méthode de substitution progressive on ne pouvait en effet étudier que quelques jeux de combustibles, alors qu'on voulait expérimenter une gamme beaucoup plus étendue de compositions isotopiques ; d'autre part il était essentiel pour ce programme d'étudier également les combustibles irradiés.

On a procédé comme pour les substitutions, c'est-à-dire qu'on considère une petite modification de composition (portant sur des noyaux à forte section d'absorption ou de production de neutrons) au sein d'un réseau dont la géométrie reste la même partout ; les conditions de propagation et ralentissement des neutrons ne sont donc pratiquement pas perturbées, et il en est de même pour l'absorption résonnante de l'uranium 238 ; d'autre part le spectre thermique est aussi peu perturbé que possible, et en tout cas on se place dans des conditions où il peut être calculé.

L'originalité de la méthode développée en France depuis 1962 réside dans la mesure de deux signaux indépendants. On exploite le fait que la substitution a des effets différents suivant la distance à laquelle on se place. Il est possible d'en déduire deux informations distinctes, qu'on peut traduire par exemple en termes de variations des taux d'absorption et de production de neutrons.

Pratiquement on oscille, au sein d'un réseau régulier, un train d'éléments combustibles dans lequel un élément est modifié. La variation de réactivité est annulée par un pilote automatique dont le changement de

position donne ce qu'on appelle le signal "global". On mesure par ailleurs la variation de la densité neutronique au voisinage immédiat de l'élément à mesurer par une chambre à fission annulaire ; cette variation est ce qu'on appelle le signal "local".

Par ailleurs les mesures sont doublement différentielles ; on utilise des éléments témoins comportant des additions connues d'uranium 235 ou de bore qui servent de calibration.

L'interprétation des mesures est faite par la méthode de l'"échantillon équivalent" qui consiste à définir un échantillon fictif dans lequel on aurait ajouté à de l'uranium naturel des quantités d'uranium 235 et de bore telles que les taux d'absorption et de production de neutrons soient les mêmes que dans l'échantillon réel.

Il est nécessaire d'effectuer une certain nombre de corrections et calculs interprétatifs : d'une part, il faut tenir compte d'effets parasites (tels que des petites variations de densité) et des perturbations introduites par les moyens de mesure (chambre annulaire). D'autre part on ne peut se dispenser, pour une interprétation fine, de faire un calcul détaillé en espace-énergie autour de la zone-test ; des méthodes ont été mises au point ; elles ne sont possibles que parce qu'on se place au sein d'un réseau par ailleurs régulier ; d'autre part l'existence d'une calibration et la méthode de l'échantillon équivalent permettent de limiter les calculs à une zone de très faible dimension entourant le train d'éléments combustibles.

C'est grâce à l'ensemble de ces précautions, et à une mise au point qui s'est révélée longue et délicate, tant sur le plan des techniques expérimentales que sur celui des méthodes d'interprétation, qu'on a pu obtenir un ensemble très cohérent de résultats et en tirer des conclusions précises sur les propriétés du plutonium.

Les compositions ayant fait l'objet de mesures ont été les suivantes :

Combustibles "reconstitués"

U nat + 0.05 % Pu	teneur du Pu en isotope 240 : 1 %, 3 %, 6 %	
+ 0.07 %	3 %,	20 %

appauvri 0.65 %	+ 0.05 % Pu	teneur du Pu en isotope 240 :	3 %, 8 %
	+ 0.07 %		3 %, 8 %
appauvri 0.45 %	+ 0.15 % Pu	"	8 %
	+ 0.20 %		8 %
appauvri 0.22 %	+ 0.25 % Pu	"	3 %, 20
	+ 0.30 %		3 %, 8 %, 20
	+ 0.35 %		3 %, 8 %

Les expériences ont été faites dans Minerve modérateur D^2O), Marius et César (graphite). Elles se poursuivent actuellement dans César à chaud.

Combustibles irradiés :

dans Minerve : combustibles EL 3
dans Marius : combustibles G 2 - G 3 - EDF 2.

Il est prévu de faire des oscillations de combustibles irradiés, en température, dans César.

Ces expériences font l'objet d'une communication de B. Laponche.

b) Milieux à neutrons rapides

Des oscillations ont été effectuées depuis deux ans dans Masurca, et dans Ermine (utilisation particulière de Minerve avec un coeur rapide entourée d'une zone nourricière à neutrons thermiques). Plusieurs types de mesures sont ou ont été mis en oeuvre :

- Mesures de réactivité dues à des variations de composition ou de dimension du combustible. La variation de réactivité est mesurée par un pilote automatique, et l'étalonnage réalisé au moyen d'échantillons d'enrichissement variable en U_5.

- "Coefficients de danger" : mesure de la variation de réactivité due à des corps n'entrant pas dans la composition du milieu.

L'interprétation est délicate parce qu'en général on modifie les propriétés de diffusion et ralentissement des neutrons.

- Mesure du α du plutonium (Ermine) : variation de réactivité + signal "local" donné par une chambre à fission à U^{238} placée au voisinage de l'échantillon oscillé. Etalonnage de la variation de réactivité et du signal local par des sources de californium.

- Effet Doppler (Ermine) : Mesure de la variation de réactivité entre un échantillon froid et un échantillon chaud placé dans un barreau de graphite. Le chauffage de l'échantillon se fait, à l'extérieur du coeur, par induction.

En ce qui concerne l'interprétation de ces mesures, aucune méthode autre que la comparaison de l'état perturbé à l'état non perturbé n'a été pour l'instant suffisamment élaborée. Nous pensons que beaucoup de travail reste à faire dans ce domaine, et que seules pourront faire l'objet d'interprétations précises les mesures suffisamment "propres" au sens indiqué plus haut.

III - EXPERIENCES $k_\infty = 1$

Ces expériences, appelées aussi "type PCTR" consistent à définir une modification du milieu (par exemple une addition d'absorbants) qui lui confère la propriété $k_\infty = 1$, c'est-à-dire laplacien-matière nul. On vérifie que cette propriété est réalisée par enlèvement d'une petite portion du milieu testé.

Cette méthode ne peut donner de résultats interprétables que si les dimensions de la zone testée (et donc modifiée) sont suffisamment grandes pour que les perturbations dues aux frontières aient complètement disparu à l'endroit où on effectue l'enlèvement. En général cette condition n'est pas vérifiée, et on cherche à y remédier en ajustant une zone "tampon" qui est censée minimiser les causes d'erreur. En outre, il faut que l'enlèvement d'une portion du milieu n'apporte pas par elle-même des perturbations, ce qui suppose en toute rigueur le milieu homogène et isotrope.

Au C.E.A. on n'a pas fait d'expériences de ce type en neutrons thermiques, mais on a analysé des expériences effectuées à l'étranger. On a constaté que la plupart du temps la validité des mesures est suspecte : la zone-test est presque toujours trop petite ; le critère d'ajustement de la zone-tampon est souvent insuffisant, et dans les milieux à neutrons thermiques, l'hétérogénéité est en général une source importante de difficultés.

On envisage d'effectuer des expériences $k_\infty = 1$ pour les milieux à neutrons rapides. Dans ce cas l'hétérogénéité est en général moins marquée, mais les perturbations de frontières ont des effets plus importants, et il faut donc être encore plus strict sur les dimensions du milieu, et la définition de la zone-tampon éventuelle. En fait il ne semble pas possible d'éviter de faire des calculs détaillés de perturbation, ce qui suppose que les conditions expérimentales soient suffisamment simples pour être accessibles au calcul. D'ailleurs dans le cas des milieux rapides, la condition $k_\infty = 1$ est éloignée des caractéristiques usuelles des réacteurs ; il s'agit donc de mesures de caractère fondamental, ce qui amène à choisir des milieux de composition simple.

Dans les expériences qui sont préparées pour Ermine les milieux comporteront seulement, outre U_8 et U_5 (ou U_8 et Pu_9), du graphite et du bore. On a l'intention de faire des mesures mettant en jeu des spectres très nettement différents.

CONCLUSION

Nous disposons au C.E.A. d'une longue expérience en ce qui concerne les mesures intégrales mettant en jeu une quantité plus ou moins réduite de combustible au sein d'un autre milieu. Nous pensons qu'il faut se garder d'une méfiance injustifiée à l'égard de ce type de mesure qui peut rendre d'énormes services, mais aussi d'un optimisme excessif basé sur un esprit critique insuffisant. On a toujours tendance à sous-estimer les difficultés d'interprétation, et tout le monde a cédé plus ou moins à la tentation de tirer des conclusions hâtives de coïncidences numériques ou de circonstances exceptionnellement favorables.

Nous pensons que dans ce domaine on ne peut pas faire n'importe quoi. Seules méritent d'être faites des mesures suffisamment "propres", où on a fait le maximum pour rendre compatibles les milieux en présence, et pour lesquels on dispose d'outils interprétatifs suffisamment élaborés. On ne consacre jamais assez de temps à la préparation des expériences. On ne saurait trop mettre en garde les expérimentateurs contre la tentation d'obtenir des résultats pour des résultats, sans avoir suffisamment réfléchi au parti qu'on pourra en tirer.

A Review of "Single Channel" Substitution Studies at A.E.E. Winfrith

J. R. Askew
J. E. Sanders

1. Introduction

Reactivity measurements involving the oscillation - or substitution - of single fuel elements in a zero-power reactor have been used at A.E.E. Winfrith for several years. Initially these were confined to thermal spectrum systems, but more recently the approach has been extended to include intermediate and fast neutron spectra. These experiments provide lattice parameters from far smaller amounts of test fuel than required for more conventional exponential or critical assemblies. In many cases the procedure is the only practicable way of measuring reactivity: for example in the case of burn-up studies on highly radioactive fuel taken from a power reactor.

The technique does, however, present rather severe problems of interpretation, particularly for heterogeneous thermal lattices. This factor has limited its role in reactor physics studies, and for several of the investigations reviewed here parallel experiments were made using sub-critical assemblies, uniformly fuelled, as a cross-check. Considerable progress has now been made in the interpretation, and these parallel experiments have served to confirm that it is possible to analyse well-designed single channel experiments to give basic tests both of nuclear data and of calculational models.

2. Thermal Reactor Studies

To date three major programmes of work on thermal reactor fuels have been carried out in the HECTOR facility; covering metal, oxide and carbon dispersed fuels. In each case both uranium and plutonium-enriched fuels were studied.

HECTOR is a graphite-moderated zero-power reactor driven by an annular region loaded with 80% enriched uranium. Inside this annulus is a central test region approximately 1 m in diameter in which different graphite moderated lattices may be loaded to provide the required environment for the fuel sample in the central channel. This region can be heated to 450°C by circulating CO_2. The reactor is very stable and high-precision measurements are possible without oscillation. A more detailed description of the reactor is available in Reference 1.

2.1 Uranium-Plutonium Metal Samples

The initial HECTOR programme comprised studies of single rods of plutonium-uranium metal fuel which were aimed at understanding the properties of Pu239 and Pu240 in situations similar to those of the U.K. gas-cooled power reactors. The approach was essentially a comparison against uranium fuel of identical geometry, whose properties were well understood from previous extensive work on exponential and critical assemblies.

The fuel samples were 30.5 mm diameter and generally 460 mm long. Two Pu/U ratios were studied, viz. 0.257% and 0.504%. The Pu240/Pu ratio varied from 3.20% up to 24.2%. A section of a fuel element irradiated to 4500 MWD/Te in one of the Calder Hall reactors was also included. The calibration elements were of identical geometry and of natural and slightly enriched uranium. The environment lattices were fuelled by slightly enriched uranium or 0.25% Pu/U fuel on a 216 mm pitch, at temperatures of 20° and 430°C.

The measurements comprised reactivity change and surface manganese foil activation rates for the fuel element inserted into the central channel of the test region. The analysis was based upon a multi-group cell calculation for a regular lattice of sample fuel, normalized to the observed manganese reaction rate at the sample. A simple spectrum correction was applied for mismatch between sample and environment: as the sample could be considered identical to the surrounding lattice epithermally, it was sufficient to apply a two-group correction deduced from the manganese flux determinations. The method and results are given in Reference 2, where the adequacy of the simple spectrum correction is demonstrated by the similar results obtained in different environments.

The results were interpreted in a relative sense, comparing uranium to plutonium enrichments. Combining all uncertainties, both experimental and theoretical (e.g. spectrum corrections) it proved possible to check k-inf for the plutonium/uranium fuels to an accuracy of approximately $\pm$ 0.7%, a considerable part of this arising from uncertainty in the absolute plutonium content of the fuel. Differences hot/cold were obtained to $\pm$ 0.3% equivalent to about $\pm$ $1.10^{-5}/^{\circ}C$.

Agreement between theory and experiment was very good for both low and high Pu240 fuels using the 2200 m/s value of eta for Pu239 in the TRACER code (2.088). This confirmed, and extended, an analysis (Ref. 3) of exponential assemblies by the same calculational method, experiments since re-analysed using the WIMS code with essentially similar results. In the case of the 4500 MWD/Te irradiated fuel sample, the measured heavy element isotopic content was assumed, so that the fission product reactivity could be separately deduced, confidence in the heavy element effects having been established using the synthetic Pu/U fuels. The fission products were calculated to hold down 4% in k-infinity and the measurements agreed with this prediction to well within the $\pm$ 0.7% overall uncertainty.

One set of these metal rod measurements has since been re-analysed using a multigroup correction for environment spectrum (Ref. 4). The same good agreement is observed using a WIMS lattice calculation, with Pu239 eta = 2.096, although a change to an absolute calibration enables one to observe an apparent "zero error" associated with over-prediction of resonance events in uranium-238: an effect which was not revealed in the earlier plutonium/uranium comparisons since all samples were effectively identical in the epithermal region.

2.2 Oxide Cluster Experiments

Following the studies on metal rods, the HECTOR programme was directed to 21-rod clusters of PuO_2/UO_2 fuel, the cluster geometry being essentially that of the Windscale A.G.R.

The compositions of the PuO_2/UO_2 clusters, and the UO_2 calibrating clusters were as follows:

Cluster Description	$\frac{PuO_2}{PuO_2 + UO_2}$ (%)	$\frac{Pu240}{Pu}$ (%)	$\frac{U235}{U}$ (%)
Low Plutonium Content	0.25	5	0.91
	0.4	3	0.71
	0.4	13	0.71
	0.4	26	0.71
High plutonium content	2.0	3	0.71
	2.0	13	0.71
	2.0	26	0.71
Uranium Calibrations	-	-	1.35
	-	-	1.8
	-	-	4.0

In addition to the synthetic fuels, similar clusters irradiated up to 10,000 MWD/Te in the Windscale A.G.R. were included in the programme, to provide information on fission product poisoning.

As the clusters represented substantial amounts of fuel, both the environment and finite length corrections were more important than in the metal rod studies and a more complicated procedure was needed, yielding essentially a few-group thermal spectrum correction. A typical example is shown in Fig. 1. In other respects, especially in the U238 content and geometry of samples being almost identical, and in a calibration against enriched uranium being the basis of reactivity determinations, the procedure was similar to that for the metal samples. A full description of the experiments and results is given in Reference 5.

Including all sources of uncertainty, including those associated with environment corrections, it was possible to measure k_{inf} to $\pm$ 0.5% for low plutonium clusters and $\pm$ 0.9% for high plutonium ones, and to obtain temperature coefficients to an accuracy of $\pm$ 1.0 and $\pm$ 1.7 . $10^{-5}/^{\circ}C$ respectively. It is noted in Ref. 5 that these accuracies are comparable with those obtained in the SCORPIO two-region exponential studies (Ref. 6) using much larger quantities of fuel. The uncertainties do not represent the ultimate accuracy of the technique: in fact the direct experimental errors are small. Improvements could be obtained by better knowledge of the fuel compositions, and by reducing the large mismatches encountered in this experimental programme, including that due to the use of short-length, unbuffered samples.

2.3 Dispersed Fuel Studies

The third series of experiments were made on single-rod fuel samples consisting of U235 and Pu uniformly dispersed in graphite, of six compositions:

UO_2 (7% U238) 1350 and 135 C/U ratio

PuO_2 (3% Pu240) 2000 and 200 C/Pu ratio

PuO_2 (24% Pu240) 2000 and 200 C/Pu ratio

Samples of three diameters, viz. 10, 30 and 45 mm were measured, all 350 mm long, and canned in stainless steel.

The samples in this experiment differed markedly from the environment, so that the spectrum correction methods developed for the metal rods and the clusters were not applicable. A preliminary analysis of the results has been made using the HPS method, thin copper samples being used to calibrate the reactivity scale. Whilst the results for the larger samples are consistent with expectation, both as comparisons between uranium and plutonium and also between environments for the same fuels, there appears to be a trend with sample size which is not yet understood. Further work on the interpretation of these experiments is in hand.

3. Null-Reactivity Measurements in an Intermediate Spectrum Assembly

The 'Null-Reactivity' technique has been used to obtain the k-infinity of small U235 and Pu239 intermediate spectrum systems (Ref. 7, 8), again loaded into HECTOR. The object of this work was to provide an integral test of nuclear data on the fissile nuclides in the 10 eV - 10 keV region.

The intermediate spectrum test region consisted of homogeneous fuel elements made from a uniform dispersion of U235 (or Pu239) in graphite with boron to give a k-infinity close to unity. The atom ratios in the fuel were chosen to provide a spectrum in which over 70% of fissions occurred between 10 eV and 10 keV with virtually no reactions below 0.5 eV. The test region was 460 x 380 x 380 mm in size and contained about 8 kg fissile material. It was surrounded by a heterogeneous buffer region of similar atomic composition which separated it from the thermal driver of the reactor.

A series of special homogeneous fuel samples with varying boron content could be substituted for a "void element" at the centre of the test region. Reactivity worths of these samples were measured by the doubling-time method to an accuracy of order $3 \times 10^{-7} \frac{dk}{k}$ The reactor sensitivity was such that this corresponded to $\pm$ 0.3% in the k-infinity of the fuel sample. The boron : fissile ratio giving null reactivity signal was then obtained by interpolation. Complementary measurements of the U235 and Pu239 fission rates and the boron absorption rate were made by inserting small fission chambers and BF_3 counters into the centre of the test region. These detectors were calibrated in a graphite thermal column, using the well-known effective thermal cross-sections. The combination of the unit k-infinity and the reaction rate values allows the capture-to-fission ratio ("alpha") of the fissile nuclide to be deduced from the neutron balance.

To obtain the corrections for spectrum and adjoint mismatch, perturbation calculations were made using the SCRAMBLE diffusion code in two-dimensions and with 10 energy groups. These showed that the k-infinity of a null-sample was 1.000 ± 0.002. Including all known sources of random and systematic error, the overall uncertainty in the k-infinity measurement was then approximately $\pm$ 1%. The calculations also showed that the reaction rate ratios required correction by not more than 3% to bring them to the infinite medium (zero-leakage) values. Once these infinite medium values have been established they may readily be compared with zero-leakage multi-group calculations for homogeneous media without further reference to the particular arrangement in HECTOR. As an example, the results of these experiments have been used as a sensitive test of the resonance region data for U235 and Pu239 in the library of the 69-group WIMS code. The comparison of the HECTOR measurement with WIMS prediction confirms the adequacy of the recommended U235 data (which has an epithermal alpha of 0.50) but is in marked disagreement with an earlier set (with an epithermal alpha of 0.67). In the case of Pu239, the measured k-infinity and reaction rates are in poor agreement with the predictions,

indicating the need for an up-dating of the library for accurate calculations of highly-undermoderated Pu239 systems.

4. Fast Reactor Studies

Two series of fast reactor measurements using the null-reactivity type of few-channel experiment have been made at Winfrith. The first series (in DIMPLE) were in the context of the study of the steam-cooled fast reactor and provided, in particular, important evidence on the capture-to-fission ratio of Pu239. The second series (in ZEBRA), which are still in progress, comprise "benchmark" integral experiments on a range of fast plutonium-fuelled lattices serving as a basis for the systematic adjustment of data sets for fast reactor calculations.

4.1 Experiments in DIMPLE

The experiments were carried out in a central test region simulating a plutonium-fuelled steam-cooled fast reactor (Ref. 9). The test region was loaded with 51 mm square plates of PuO_2/UO_2, UO_2, steel and polypropylene stacked inside steel tubes, to form a lattice 560 mm diameter, 740 mm high, and having a predicted k-infinity of 1.009. This array was surrounded by a UO_2 buffer region, and a steel filter region. The driver zone consisted of 3% enriched UO_2 fuel rods moderated by light water.

The measurement of k-infinity was made by removing from the centre of the test region a small central section containing 3 x 3 x 3 lattice cells, the resultant reactivity worth being determined by the change in the reactor doubling time. The adequacy of the spectrum and adjoint match at the centre of the test region was checked by transport calculations using the GMS code and diffusion calculations using the SCRAMBLE code. These indicated that the null-point corresponded very closely to the unit k-infinity condition. A null-point was not exactly obtained, the deduced k-infinity being 1.002 ± 0.005.

U238 capture, Pu239 fission and U238 fission in the central lattice cell were measured by foil activation, the foils being embedded in the appropriate plates to obtain the mean reaction rates. The counting techniques were based on those developed at Winfrith for work in thermal reactor lattices.

By combining the measured k-infinity and reaction rate data it was possible to deduce the capture-to-fission ratios in Pu239 from the neutron balance equation. The value thus obtained was 0.57 ± 0.13, compared to the prediction of 0.39 using the then current FD2 data set, adding further weight to the supposition that the values for Pu239 in the keV region were too low, since confirmed by differential measurements.

4.2 Experiments in ZEBRA

The ZEBRA techniques were developed from the DIMPLE studies. The plutonium fuelled test region lattice, which is about 620 mm diameter, 610 mm high is driven by a fast spectrum driver region fuelled by U235, both radial and axial drivers being provided. A plan view of the core is shown in Fig. 2. In a typical case, the test region contains 80 kg of plutonium, the driver 350 kg U235. The test lattice is built up from 51 mm square plates loaded into steel sheaths. Once the basic features of the test cell have been defined, the composition is adjusted until a near zero reactivity change is obtained on voiding a small central zone of the test region. So far four test regions have been studied containing plates of plutonium metal, natural uranium, graphite, stainless steel and sodium. Detailed reaction rate measurements are made in the null-reactivity lattice cell, and the neutron energy

spectrum is also determined by a combination of time-of-flight, recoil proportional counters, and Li6 semi-conductor sandwich spectrometers. This work will be described in a Paper to the Fast Reactor Meeting of the B.N.E.S. in June.

The k-infinity of the test region is obtained from the following expression:

$$k\text{-inf} = 1 + \Delta\varrho(\text{cell}) \quad S + \xi$$

where S is a calculated sensitivity relating reactivity and k-infinity, and ξ is the null-error arising from the finite size of the test region. ξ is a function of the product of the mismatch in flux and mismatch in adjoint between the infinite and finite systems and is a second order correction. It is obtained from 37 group perturbation calculations using diffusion theory on a spherical ZEBRA model with a range of assumed test regions. The ξ values for the first four test regions are less than 0.002. Provided k-infinity is close to unity (say $\pm$ .02), the error in calibration factor S is not important.

The basic series of experiments in eadh test region consists of separate measurements of the reactivity worths of fuel plates and fuel sheaths in test region zones comprising 1, 4 and 9 cells, i.e. in zones 54 mm, 108 mm, and 162 mm square and 1 or 2 cells high. The reactivities are measured by balancing the reactor using the fine control rod: the main source of experimental error arises from the drifts in the reactor state due to temperature variations. The sensitivity is such that removal of a single cell with a k-infinity of 1.01 gives a reactivity change of order 2.10^{-6}. The reactivity controlled by a cell has been found to decrease as the number of cells removed is increased and, moreover, varies with cell boundary. It was argued that this behaviour was mostly due to changes in the resonance shielding in the uranium surrounding the cell removed and this is confirmed by reaction rate measurements. The capture rate increases in uranium surfaces which were previously shielded by other uranium but which are now exposed to scattering material, and decreases in uranium surfaces previously exposed to scattering material, but which are now exposed to further uranium. By plotting the results as a function of the area of uranium, weighted either with the fraction of cavity surface which is scattering material, or uranium, whichever is appropriate, a reasonable correlation of the results can be obtained (Ref. 10) which enables them to be extrapolated to zero uranium area exposed per cell, as shown in Fig. 3. As might be expected, this extrapolation is least satisfactory for the results with two uranium boundaries, where the effect is most accentuated.

The uncertainties associated with the measurements of k-infinity in a typical case are summarised below. The overall uncertainty in k-infinity is $\pm$ 0.007. The target accuracy of $\pm$ 0.003 requires improvements in the cavity extrapolation technique referred to above.

Source	% k-infinity
Fuel Plate and Sheath Reactivity Measurements	0.3
Plate Mass Tolerance	0.1
Cavity Extrapolation Procedure	0.6
Uncertainty in Calculated Sensitivity Factor (S)	0.1
Uncertainty in Calculated Null-error (ε)	0.1
Total	0.7

5. Conclusions

Reactivity measurements using few-channel techniques are providing valuable integral data for thermal and fast reactors. Methods of interpretation have evolved which currently allow the k-infinity to be obtained to an accuracy of 0.5 - 1%, which is comparable to that found using much larger amounts of fuel in more conventional zoned-exponential experiments. More work is required in certain areas, for instance improved calculations of spectrum mismatch factors in thermal reactor experiments, and a more rigorous treatment of cavity size effects in fast reactor experiments with plate-fuelled cells.

General Reactor Physics Division, and
Fast Reactor Physics Division,
A.E.E. Winfrith,
Dorchester, Dorset,
U.K.

REFERENCES

1. TATTERSALL, R. B. et al. Reactivity Measurements in Plutonium-Uranium Fuel Elements in HECTOR. Experimental Techniques and Results.

AEEW - R 376 (1964)

2. BANNERMAN, R. C. et al. Analysis of the Results of Reactivity Changes in Measurements on Plutonium-Uranium Fuel Elements.

AEEW - M 441 (1964)

3. HARPER, R. G. An Analysis of Experiments on Graphite Moderated Assemblies Fuelled with Uranium, Plutonium and Uranium-Plutonium Metal.

AEEW - R 325 (1964)

4. ASKEW, J. R., PITCHER, H. H. W. Concerning the Analysis of Single Channel Substitution Measurements.

Journal of B.N.E.S., p. 260, July, 1967.

5. BANNERMAN, R. C. Analysis of Reactivity Measurements in HECTOR on Single Plutonium-Uranium Oxide Clusters.

AEEW - R 499 (1968)

6. KING, D. C. A Comparison of Measurements on PuO_2-UO_2 Fuelled Graphite Lattices with Calculations using the ARGOSY Method.

AEEW - R 466 (1966)

7. FOX, W. N. et al. Measurements of U235 Alpha in an Intermediate Energy Spectrum.

Transactions of Am. Nucl. Soc., 10, 231 (June 1967)

8. FOX, W. N. et al. An Experimental Study of the Integral Properties of U235 and Pu239 in an Intermediate Energy Spectrum, and Comparison with Theoretical Prediction.

AEEW - R 586 (1968)

9. FOX, W. N. et al. The Measurement of Pu239 Capture to Fission Ratios in Fast Reactor Lattices.

Journal of B.N.E.S., 6, pp. 63-79 (1967)

10. INGRAM, G., WARDLEWORTH, D. Private Communication.

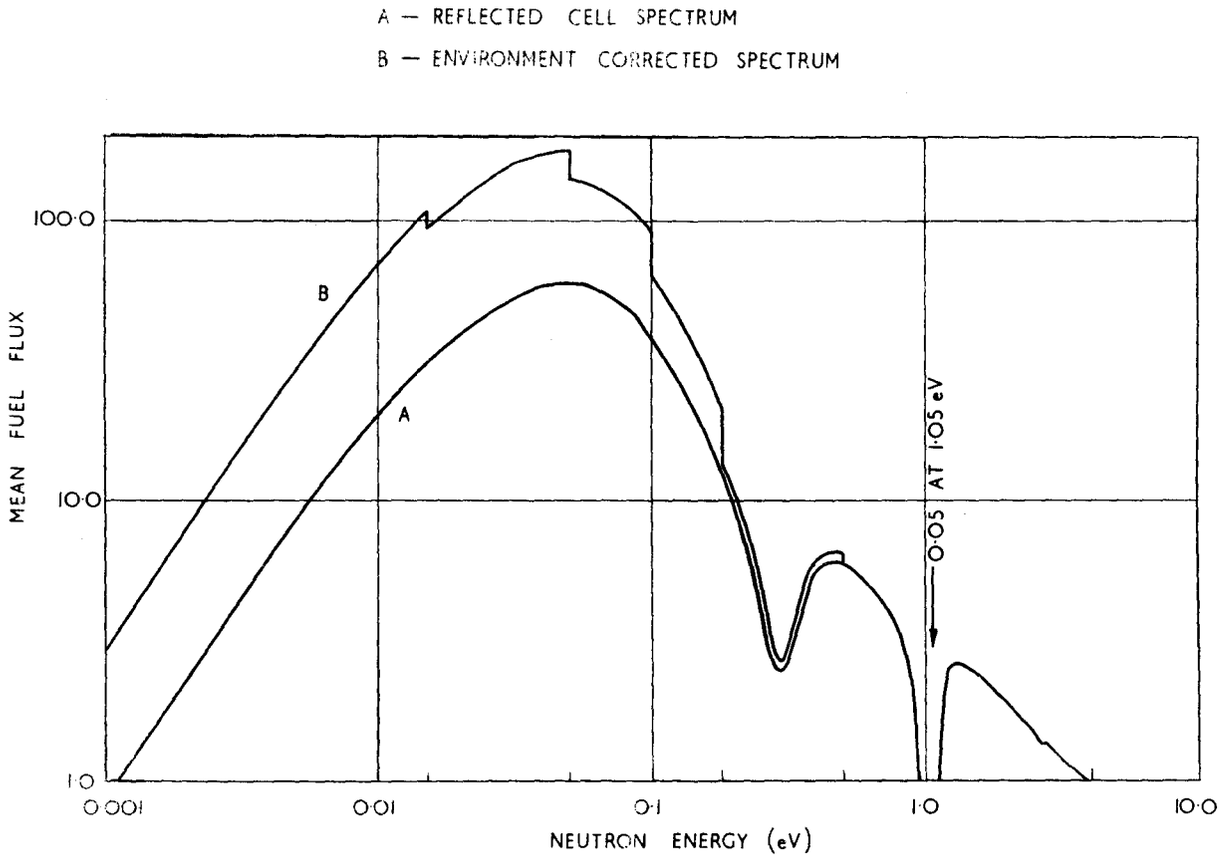


FIG. 1 SPECTRUM IN 2% PuO_2 (26% Pu 240) SAMPLE IN CORE 10 AT 22°C

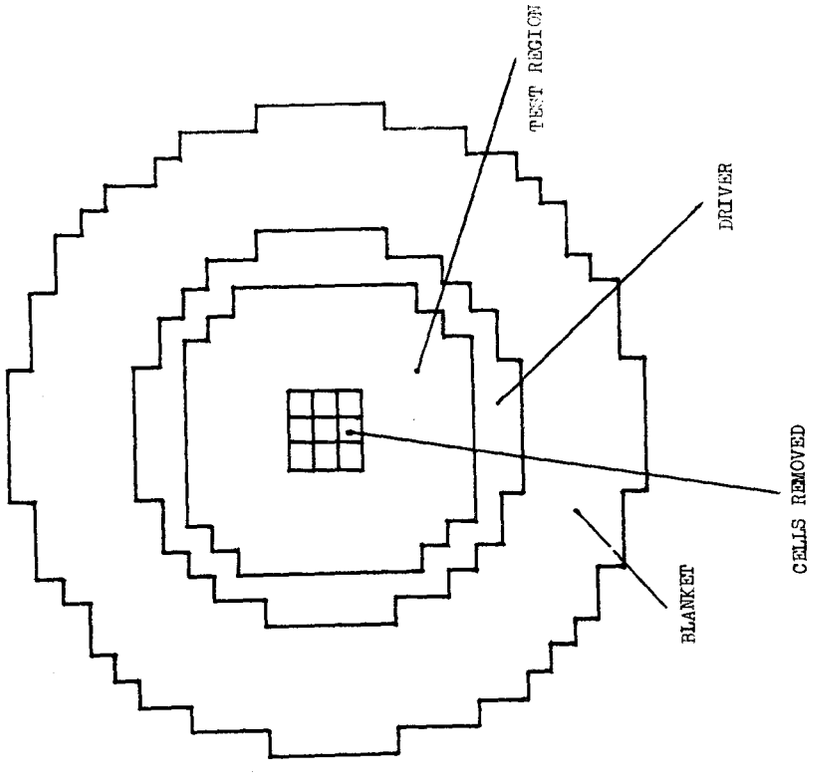


FIG. 2. PLAN OF ZEBRA LOADING FOR NULL-REACTIVITY MEASUREMENTS

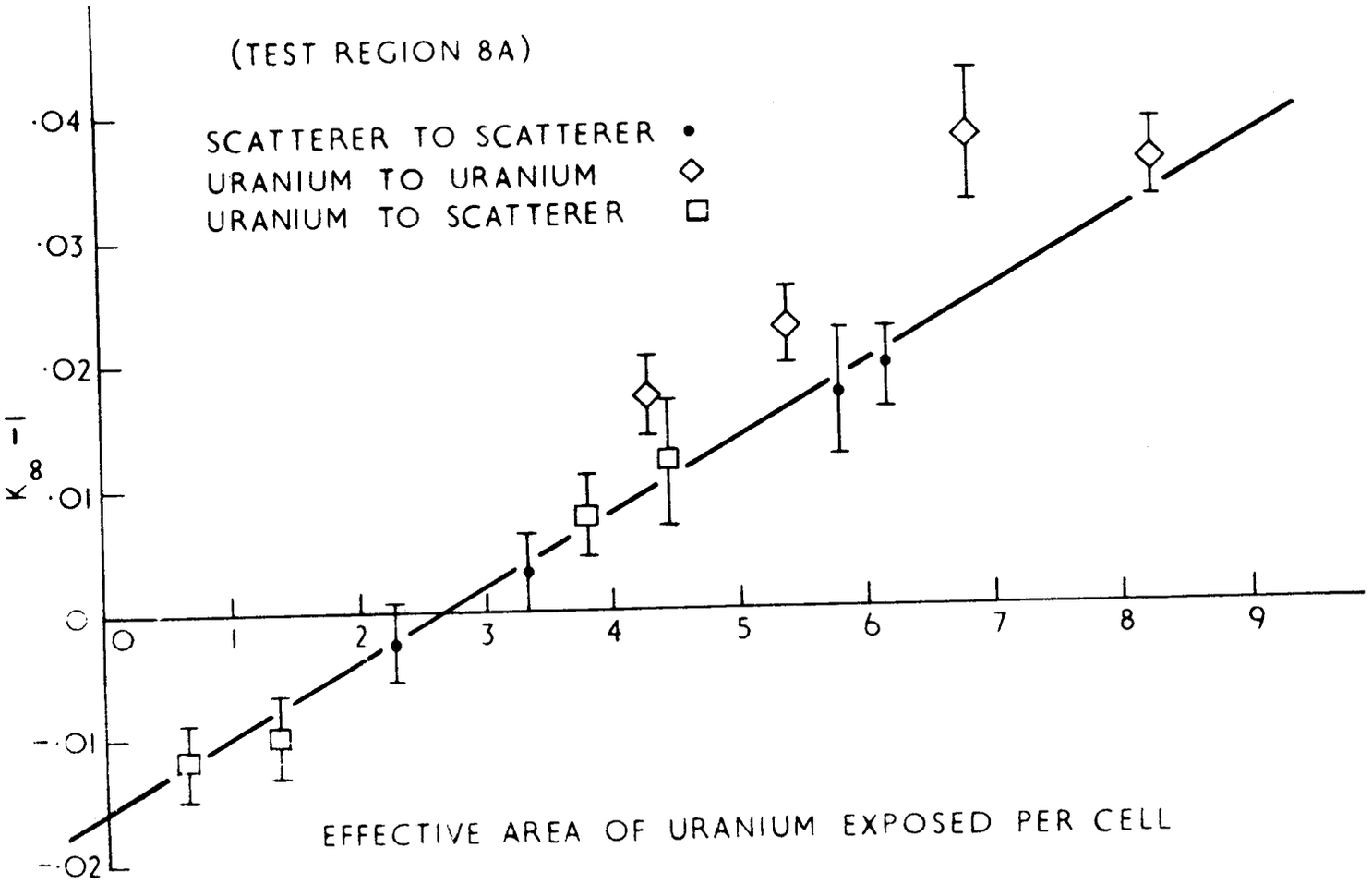


FIG. 3. CAVITY SIZE EFFECTS IN ZEBRA MEASUREMENTS

STATUS REPORT OF REACTOR PHYSICS IN ITALY

ON FEW RODS EXPERIMENTS

F. Accinni - F. Casali

1. Introduction

The main effort in thermal reactor physics in Italy is devoted, at present, to the investigation of D_2O moderated natural uranium fuelled lattices, in the framework of the Cirene project for a fog cooled power reactor, Tha main experimental activity has been concentrated on buckling measurements with substitution technique, zero reactivity and detailed parameters measurements.

At the same time experimental techniques and theoretical approaches were tried to deduce the heterogeneous parameters γ, η of large cross section fuel elements from measurements on a single rod embedded in a block of moderator.

The future programs will also deal with fuel elements containing low enriched uranium ($\sim$ 1%) and plutonium.

2. Buckling measurements

Buckling measurements have been performed in the ECO

facility at Ispra in the framework of a cooperation between CISE, CNEN and EURATOM.

Five CIRENE fuel elements, each constituted by five 19-rod, 50 cm long, natural UO_2 bundles were available. Twelve different configurations, three coolant densities at four lattices pitches, have been investigated.

The analysis of the results was done either by a simplified homogeneous calculation using the code MONOS (1) either by the heterogeneous method using the three-group code HETROIS (2). Table 1 lists the experimental and their comparison with PROCELLA (3) and PINOCCHIO (4) calculations.

Differences between buckling values deduced from one and five elements substitutions are always less than 0.1 m^{-2}.

In the case of the heterogeneous methods, the channel nuclear parameters, which enter as input data in the codes, are not very sensitive to the lattice pitch. Further, according to the three-group calculation, the contribution to the resonance absorption of a channel due to the other ones is better evaluated than in the homogeneous case. Then it is expected that the results should be better in this analysis as it is confirmed by the reduced discrepancy between one and 5 substituted rods.

In conclusion, we can take as most reliable the buckling values derived from the heterogeneous analysis of the 5-rod substituted cores.

An uncertainty equal to the maximum discrepancy between 1 and 5 substituted rods can be assumed as representative of the systematic error due to the theoretical model used in the interpretation. This assumption is certainly pessimistic because in the previous analysis (5) of substitution experiments with 1 to 25 substituted fuel rods, the discrepancy between the results with 5 rods and the "saturation" value was less than the difference between 1 and 5. As far as the experimental errors are concerned, assuming the accuracy of the measurements of the heavy water level being of the order of 1/10 mm, errors in the axial buckling differences are negligible in comparison with the errors in the measured B^2 of the reference core.

From the results presented here we can draw the following concluding remarks:

2.1 PROCELLA

a) The discrepancy between calculated and experimental buckling values is important for the configurations with no

coolant and with higher coolant density (equivalent to 500 + 1000 p.c.m.).

The overestimation of the buckling in the latter case can be explained with an overestimation of the thermal utilization factor, which was later evidenced in detailed measurements performed in the CNEN RB-1 facility, in Bologna. In the configurations with coolant density close to nominal (about 0.29 g/cm^3) for the CIRENE prototype reactor, a very satisfactory agreement was found.

b) The experimental and theoretical buckling trend versus lattice pitch are quite similar.

c) The experimental coefficient $(\Delta B^2/\Delta\rho)$, relevant to the coolant density influence, is about 25% higher than calculated. This result should be kept into account in design studies.

2.2 PINOCCHIO

a) The discrepancy between calculated and experimental buckling is significant only for the air cooled case (at maximum, differences in reactivity of the order of 600 pcm).

b) The trend of buckling versus lattice pitch is slightly flattened as compared to the experimental one.

c) The coefficient $(\Delta B^2/\Delta\rho)$ is sufficiently well calculated (differences of less than 5%).

For further details see reference (6).

3. Zero reactivity experiments.

A PCTR - type reactor, RB-1 (7), has been installed in Bologna in 1962.

After a set of experiments on graphite moderated-natural uranium lattices, the facility was modified to allow experiments on heavy water lattices.

A series of K_∞ measurements concerning Orgel lattices (UM/19 and UC/7) are reported in a Euratom paper presented in this panel. For UM/19 lattices also Pu bearing elements, with two enrichment in Pu^{239} and Pu^{240}, were studied. The activity on heavy water lattices continues now with CIRENE experiments; K_∞ and detailed parameters measurements were performed for nine lattice configurations.

K_∞ is derived from the experimental quantities, according to the expression

$$K_\infty = 1 + f' \frac{A^f \sigma_{ao}^{Cu} (g + rs)^{Cu} M_o^{Cu}}{A^{Cu} \sigma_{ao}^{f} (g + rs)^{f} M_f} \frac{n^{Cu}}{n^f} G^{Cu}$$

where:

f' in the "1/v utilization factor", defined as the ratio

betwen 1/v neutron absorption in the fuel and in the cell. It is deduced from intracell neutron distrinution measurements as will be later specified.

A^f, A^{Cu} are fuel and copper mass numbers.

$\overset{Cu}{ao}$, $\overset{f}{ao}$ are copper and fuel absorption cross-sections at 2200 m/s g,r, s are Westcott factors.

M_o^{Cu} is the zero-reactivity mass od copper as previously defined.

M_f is the mass of the fuel.

$\frac{n^{Cu}}{n^f}$ is the cell-boundary-to-fuel-average neutron density ratio (F').

G^{Cu} is the self-shielding factor for the copper sheets.

Assuming an error of 200 p.c.m. in the absolute value of f', 1600 p.c.m. and 300 p.c.m. in $\overset{Cu}{ao}$ and $\overset{f}{ao}$, 200 p.c.m. in M_o and 500 p.c.m. in F', an overall error of less than 200 p.c.m. in K_∞ can be estimed. This uncertainty must be increased by about 200 p.c.m. to take into account D_2O depletion during the measurements, the heterogeneity effects at the ends of the bundle and the eventual spectral mismatch.

In conclusion, an overall precision of about 400 p.c.m. can be assigned to K_∞ .

Bearing in mind that about half of this value depends

from cross-sections uncertainties, the reactivity differences between the lattices investigated can be quoted with an error of less than 300 p.c.m. - The results of K_∞ measurements for all the lattices are summarized in table 2.

A special attention has been devoted to the investigation of spectral mismatch effects and of heterogeneity effects arising from the perturbation of the neutron absorption in the bundles surrounding the central one, in the voided cell situation.

For what concerns the spectral effect, several spectral indices (Au CdR, Pu^{239}/U^{235} fission rate ratio, RCR) have been measured, in the central and in peripheral cells, to test the spectral matching at different energies.

A theoretical and experimental investigation of the error in K_∞ caused by a spectral mismatch, deliberately introduced by a difference in the coolant density in the central and in the "buffer" cells, showed that in the range on these experiments the uncertainty due to spectral effects in smaller than 100 p.c.m.

Heterogeneity effects were studied, from the theoretical side, performing 5-group calculations with the diffusion code ALCI-2 (ref. 9). In these calculations, a separate evalua

tion was tried of the "terminal effect", due to the perturbation in neutron absorption at the end of the bundles, adjacent to the central one, and of the "void effect" due to the perturbation of the neutron importance in the voided cell situation.

From the experimantal side, the distribution of the thermal neutron density and of resonance absorption was measured as a fonction of the axial distance from the ends of the adjacent bundles, with the central one inserted or removed.

Fig. 1 shows the axial distribution of the Relative Conversion Ratio, in the two cases, as measured at 30.2 cm hexagonal pitch, with 0.58 g/cm^3 coolant density.

From these measurements the difference in the average values of RCR at the ends of the bundles can be evaluated to be of 3.5% in a 2 cm lenght; the overall variation is then quite negligible.

Fig. 2 shows the axial distribution of the neutron density in the two cases, as measured with copper detectors.

These distributions have been used to evaluate, in a semi empirical way, the effect on K_∞ due to the "terminal effect".

For further details, see ref. 8.

The consistency of the results of K_∞ and B^2 measurements

for natural UO_2 CIRENE lattices is very good, and confirms the reliability of the zero-reactivity method also for large fuel elements.

Experiments in RB-1 facility are foreseen with a heated fuel element, containing natural UO_2 or Pu bearing UO_2, and with 0.91% and 1% enriched UO_2 clusters.

4. Single rod experiments.

Single rod experiments have been performed, in 1967, in the L-54 reactor of Centro Studi Nucleari Enrico Fermi (CESNEF) in Milano in the framework of a CESNEF-SNAM PROGETTI cooperation, under EURATOM contract, The theory of an heterogeneous exponential experiment, with a single rod in a block of moderator, was developed (10).

From the experimental side, thermal and epithermal neutron density distributions were measured. The heterogeneous parameters, which give the best-fit of experimental distributions, were determined in accordance with the theoretical model.

The efficacy of this method to predict criticality increases in the case of high-pitch lattices with large fuel elements.

The method works well, starting with fuel elements with

40-50 cm^2 cross section and average density abous 10 g/cm^3 for natural uranium.

A test with on ORGEL cluster gave the following results:

Without coolant	With organic coolant
$\eta = 1.347$	$\eta = 1.1251$
$\gamma = 4.191$	$\gamma = 2.613$

For details, see ref. (11).

A different approach, using a heavy-water facility (12) which rotates in front of a ghaphite column in the L-54 reactor, is now under consideration.

REFERENCES

1. C. Bruschi et alii

"MONOS: a code for the variational synthesis solution of two-groups diffusion equations in r-z geometry with the monoseparable flux approsimation" - CISE R-200 (1966)

2. J. Ligou - Private Communication

3. R.A. Bonalumi, F. Palazzi, G. Pierini

"The PROCELLA method for heavy water lattices" CISE-R-196 EUR-3921 e (1967)

4. L. Amyot, G. Casini, R. Cuniberti, C. Daolio,

"PINOCCHIO, the computer program for cell reactivity calculation" - EUR-4231 e (1969)

5. G. Casini - M. Paillon

"Buckling measurements with a reduced number of fuel elements" - J.N.E. 21-7-15 (1967)

6. F. Accinni et alii

"Buckling measurements in ECO with few CIRENE fuel elements". CISE-R283 (1968)

7. F. Casali, G. Zappellini

"Rapporto di sicurezza del reattore RB-1" RT/ING (66) 27

8. F. Accinni et alii

"K_∞ and detailed parameters measurements for UO_2/D_2O fog cooled lattices in RB-1 facility" CNEN Doc CEC(69)16

9. CEA R2747 (1965)

10. S. E. Corno

"Interpretazione teorica delle esperienze di moltiplicazione neutronica attorno ad un solo elemento di combustibile" . Eneria Nucleare 10 11, 12-11, 2(1963,1964)

11. S. E. Corno, L. Fava, V. Sangiust, S. Terrani.

"Determinazione dei parametri di cattura e moltiplicazione neutronica di grandi elementi di combustibile". Parte I e II CESNEF (1968)

12. F. Accinni et alii

"The design of a small assembly for heavy water lattice measurements" CISE-R-102 (1965)

Table 1

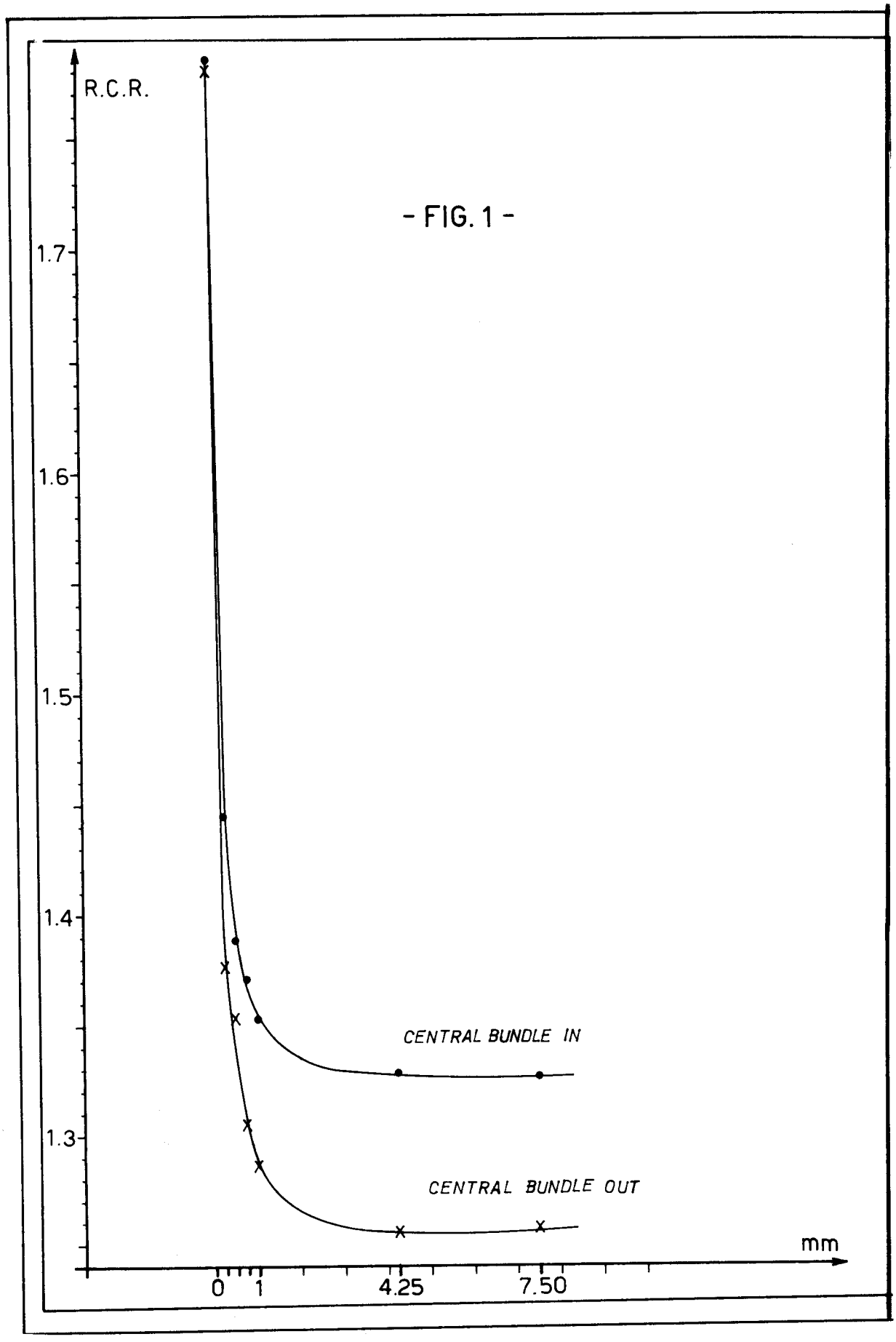
Comparison between experimental and calculated buckling values B^2 (m^{-2}) for the CIRENE UO_2/19 element

Type of element	Pitch	PINOCCHIO	PROCELLA	MONOS: 2 DIMENSIONS		HETROIS: 2 DIMENSIONS	
				1 element substituted	5 elements substituted	1 element substituted	5 elements substituted
UO_2/19/AIR	23.5	4.171	3.802	4.59	4.36	4.16	4.10
	25.5	4.589	4.247	4.81	4.64	4.52	4.46
	28.12	4.659	4.293	4.71	4.58	4.52	4.49
	29.0	4.603	4.224	4.55	4.45	4.41	4.38
UO_2/19/0.34	23.5	3.131	3.084	3.25	3.27	3.08	3.13
	25.5	3.403	3.416	3.44	3.46	3.36	3.40
	28.12	3.382	3.404	3.34	3.38	3.31	3.36
	29.0	3.313	3.333	3.20	3.25	3.19	3.26
UO_2/19/0.58	23.5	2.377	2.600	2.38	2.48	2.30	2.42
	25.5	2.573	2.850	2.54	2.62	2.55	2.62
	28.12	2.521	2.793	2.43	2.52	2.47	2.55
	28.5	2.494	2.765	2.30	2.42	2.41	2.52

Table 2

K_∞ values normalized to 99.60 % D_2O purity

Coolant density (g/cm^3)	0	0	0	0.34	0.34	0.34	0.58	0.58	0.58
Lattice pitch (cm)	27.4	30.2	32.2	27.4	30.2	32.2	27.4	30.2	32.2
K_∞ (experimental)	1.1432	1.1563	1.1627	1.1072	1.1151	1.11191	1.0818	1.0871	1.0906
K_∞ (PROCELLA)	1.1329	1.1510	1.1581	1.1000	1.1140	1.1192	1.0818	1.0926	1.0964
ΔK_∞ (x 10^5)	+1031	+531	+461	+720	+110	- 10	-	-550	-580


R.C.R.
1.7
1.6
1.5
1.4
1.3
CENTRAL BUNDLE IN
CENTRAL BUNDLE OUT
mm
0
1
4.25
7.50

- FIG. 1 -

SUMMARY REPORT ON SINGLE ROD PHYSICS RESEARCH AT MIT

Since January 1968 the Reactor Physics Project at MIT (1) has been carrying out experimental and analytical research with the primary objective of perfecting and applying single element methods for the determination of reactor physics parameters. This project and its predecessor, the MIT Heavy Water Lattice Project (2), under which many of the pertinent experimental methods were originally investigated, have been supported by contracts granted by the U. S. Atomic Energy Commission.

This work is based upon heterogeneous reactor theory, essentially in the formulation prescribed by Klahr et al. (3). Fuel elements are characterized by three parameters:

Γ = asymptotic thermal flux at fuel element surface per thermal neutron absorbed by rod,

η = fast neutrons emitted per thermal neutron absorbed in the fuel,

A = epithermal absorptions per unit slowing-down density.

Knowledge of these three parameters permits calculation of k_{oo} . The simplest of the hierarchy of approximate relations is, for a uniform lattice:

$$k_{oo} = \frac{\eta\,(1 - A/V_c)}{1 + \Gamma\,\Sigma_{am}\,V_m} ,$$

where

V_m = cross-section area occupied by moderator in a unit cell,

V_c = cross-section area occupied by total unit cell,

Σ_{am} = mean cross section for neutron absorption by moderator.

The material buckling can in turn be estimated from age-diffusion theory:

$$k_\infty = e^{B_m^2 \tau} (1 + L^2 B_m^2),$$

where the diffusion area, L^2, and age, τ, can be expressed in terms of the moderator properties Lo^2 and τo:

$$L^2 \approx Lo^2 \frac{V_m \Sigma_{am} \Gamma}{1 + V_m \Sigma_{am} \Gamma}$$

$$\tau \approx \tau_c \left(\frac{2 V_c}{V_c + V_m} \right)^2 .$$

The three parameters Γ, η and A can also be used as input to heterogeneous theory computer programs such as HERESY (3) for calculation of more complicated lattices.

Most previous work in the area of heterogeneous reactor physics has been concerned with methods for calculation of Γ, η and A. In the present work the emphasis is, instead, on experimental determination of these parameters using one (or a few) rod(s), and the concurrent theoretical and numerical work is primarily designed to help plan or interpret the experiments.

Previous work at MIT, summarized in Ref (2) has shown that η and A can be inferred from in-rod foil activation experiments. The present work has as its objective the determination of parameters by means of measurements made on the rod surface or in the surrounding moderator. This should obviate the need for cutting into rods containing plutonium and fission products, with the attendant contamination problem.

The current status of the work, scheduled for completion by September 30, 1970, is as follows. Methods have been developed for determination of η and Γ in single element exponential experiments. The parameter η is determined by measuring the cadmium ratio of gold foils in the moderator external to the fuel element; and the parameter Γ is determined by measuring the distance to the thermal flux peak in the moderator using bare gold foils. Work on determination of A is not as far advanced, but a feasibility study of a method using the ratio of gold to molybdenum subcadmium activities has been undertaken. Most of this work, together with a considerable amount of peripheral investigations, is reported in more detail in reference (1). Work has now progressed from the development to the applications stage. The major demonstration experiment being employed to assess the ultimate utility of the single element method involves measurements on 19 and 31 rod clusters of Type B Simulated Burned Fuel from the USAEC/AECL Cooperative Program. Full lattices of this same plutonium-containing fuel have previously been extensively investigated both

experimentally and analytically by the Savannah River Laboratory (4). Thus a definitive test of single element versus full lattice experiments should be possible. These experiments are now underway and preliminary results will be reported in the September 1969 annual progress report.

In addition to the above, work is also being carried out to extend the reactor physics applications of Ge(Li) gamma spectrometry. During the coming year two fuel pins will be obtained which have been irradiated to approximately 20,000 MWD/T in the Dresden reactor. Gamma spectroscopy and single element methods will be used to assess the past history and present properties of the fuel.

References

(1) Reactor Physics Project Progress Report MIT-3944-1, MITNE-96, September 30, 1968.

(2) Heavy Water Lattice Project Final Report MIT-2344-12, MITNE-86, September 30, 1967.

(3) C. N. Klahr et al., Heterogeneous Reactor Calculation Methods, NYO-2680, June 1961.

(4) Bauman, N. P., et al.,"Lattice Experiments With Simulated Burned Fuel for D_2O Power Reactors" DP-1122, Feb. 1968.

The Swedish approach to the analysis of few-rod experiments

R Persson

AB Atomenergi, Studsvik, Nyköping, Sweden

1. Introduction

Already in 1956 the progressive substitution technique was applied to buckling measurements in a D_2O-moderated exponential assembly in Sweden. In the beginning we used the simplest form of one-group perturbation theory for the analysis. However, we made a very important and successful modification by introducing an intermediate region, which takes care of transient effects between the test and reference regions [1]. The intermediate region came out later on as a consequence of a new cell concept, which places the fuel elements in the cell corners [2]. Parallel with this work using homogeneous reactor theory the development of heterogeneous programs started [3] [4]. In the first programs only two energy groups were used and the fuel elements were assumed to be line sources and sinks. But with more developed heterogeneous programs we are now able to make multigroup calculations taking the geometrical and physical properties of the fuel assemblies into full consideration.

The one-group perturbation theory approach was developed as a special program tailor-made for the buckling analysis of progressive substitution experiments with uniform lattices. The extension to other problems is not straightforward. The heterogeneous method is,on the other hand,directly applicable to composite lattices containing vacant positions, control rods and fuel elements of different types. This means that heterogeneous programs are appropriate for power project calculations - burning reactor with e.g. control rods and fuel shuffling - and it is therefore logical to check its performance on few-rod experiments in more or less complex lattices. If we are only looking for the material buckling of a uniform test lattice the two approaches are thought to complement each other, the homogeneous method being the simplest and cheapest one to apply.

Before we discuss in more detail the problems of analysis, it may be worthwhile to mention some important aspects on the experimental possibilities.

2. Experimental technique

The experimental facilities used for few-rod experiments at the Studsvik research center are presented in the table below.

Table

Name	Kind of facility	Year of start-up	Radius (m)	Temp. (°C)	Moderator
ZEBRA[1]	Expo.	1955	0.50	$\leq$ 80	D_2O
RO [2]	Crit.	1959	1.13	$\leq$ 80	D_2O
TZ [3]	Pressu-rized expo.	1964	0.80	$\leq$ 250	D_2O or H_2O

1) Not in use since 1966

2) Will be run only occasionally from the middle of 1969

3) Being modified to a critical facility (KRITZ) with expected start in June 1969

The pool of the R2-0 reactor has also been used for exponential experiments in H_2O with the R2-0 reactor as driving source. The variable-loading technique applied there may be regarded as one kind of the progressive substitution technique, but the number of fuel elements cannot be very few. We therefore restrict our discussion to experiments in D_2O-moderated assemblies, where we have got considerable experience.

Having a certain limited number of test elements at our disposal we have asked ourselves: "How shall we vary the experimental conditions in the most efficient way in order to get as much information as possible?" We have come to the following conclusions [5]:

- Use as many steps as possible in the progressive substitution.
- Do not be afraid of eccentric test regions. If you do not trust calculated eccentricity corrections, find them experimentally (if you need them).
- Vary the shape of the test region in order to find out as much as possible about the coupling effect between test and reference fuel.
- Place a single test fuel assembly in various radial positions of the reference core in order to investigate whether there are any significant differences in diffusive properties radially between test and reference fuel.

- In the case of loss-of-coolant studies, vary the void depth in the fuel assembly in order to find also the change in the axial diffusion coefficient.

Our experience tells us further that:

- An exponential assembly can be used successfully for substitution experiments, since differences in the axial buckling can generally be measured relatively accurately.

- The lattice pitches can be different in the test and reference regions. For instance in a square lattice the pitches can differ by a factor of $\sqrt{2}$.

- A central test position in a critical assembly allowing the fuel to penetrate the core axially is advantageous for reactivity measurements, e.g. of fuel gaps.

3. Problems of analysis

The perturbation method of analysis applied to the Studsvik substitution measurements is a one-group homogeneous approach backed up by two-group calculations [5]. The two-group results justify the introduction of a transition region. The success of the method must, however, be attributed mainly to the unconventional cell concept introduced. In this new cell the fuel elements are situated in the corners and thus transition cells and coupling effects appear quite logically. By correcting for the perturbed buckling (i.e. one-group flux) iteratively we increased the accuracy of the perturbation method still further. The numerical work is done on an IBM 360/30 computer by means of the program SIMBA (= Simple Iterative Method of Buckling Analysis) [6].

Our perturbation method of analysis (sometimes called the Persson method) has also been tried at the Savannah River Laboratory [7] and at Chalk River [8] [9]. The SRL results did show good agreement between the three different methods of analysis applied, viz. two-group theory, the Persson method and the heterogeneous code HERESY. The perturbation method was

also found to have the advantage of being easily applicable to all kinds of shapes of the test region. The first Chalk River conclusion reported in ref. [8] about the perturbation method was negative, but this was due to incorrect application of perturbed fluxes and diffusion coefficients. In a corrigendum [9] it is stated that "the Persson method gives as good agreement with the buckling obtained by flux mapping as does the MICRETE* method".

It may be worthwhile to notice the general definition of the one-group diffusion coefficient used in the perturbation method [5], viz.

$$\frac{D}{D_{ref}} = \frac{\sum_k (\bar{\psi}_k D_k)}{\sum_k (\bar{\psi}_k D_k^{ref})} = \frac{\sum_k (\bar{\psi}_k D_k^{ref})(D_k/D_k^{ref})}{\sum_k (\bar{\psi}_k D_k^{ref})}$$

where $\bar{\psi}_k$ is the flux of group k at the interface.
It means that when the group diffusion coefficients are equal, i.e. $D_k = D_k^{ref}$, we get $D \equiv D_{ref}$ regardless of the spectrum distribution.

The unconventional cell definition mentioned above with fuel assemblies at the cell corners moves the homogeneous method one step towards a more heterogeneous model. But due to difficulties in making cell calculations with fuel in the corners it is hard to find a direct application of the new cell concept to other problems than the progressive substitution measurements, where the cell parameters (buckling and diffusion coefficients) are adjusted to fit the experimental points.

The Swedish arsenal of heterogeneous programs has been described in several internal reports [10]. There are programs of different complexity: monopoles (MOP), dipoles (DIP), two or three dimensions (MOPZ), up to 10 energy groups (usually two or three), iterative "cell" calculations. The programs MOP, DIP and MOPZ get their inputs of cell data from the cell program FLEF by condensation of results with maximum 58 energy groups.

---

*) MICRETE is the heterogeneous program used by the Chalk River people.

A thorough systematic check of the heterogeneous programs on the great amount of experimental material available has not been done due to financial reasons. However, some comparative investigations of reactivity and change in reactivity when emptying coolant channels have been reported [11] [12]. These results indicate that FLEF + MOP applied to one substitution step gives k_{eff}-values closer to unity than does FLEF + + experimental buckling evaluated with SIMBA and valid for a full core of test elements. The loss-of-coolant reactivities show the same trend in favour of FLEF + MOP. This result is of course not unambiguous regarding the reliability of MOP or SIMBA due to possible compensating errors in FLEF and MOP. But it is encouraging in one respect, since it shows that the code system FLEF + MOP predicts reactivity effects of the investigated fuel rather well in the geometry of the RO reactor.* On the other hand, it was found [12] that FLEF-MOP results show a systematic trend in k_{eff} depending on the number of substituted elements. Such trends are eliminated in the SIMBA analysis by the extrapolation procedure to a full core.

4. Conclusions

In principle, heterogeneous theory should treat few-rod experiments more accurately than homogeneous theory is capable of, but in practice this is not necessarily so, because in both cases programs are developed to various degrees of approximation and sophistication.

Systematic trends in derived eigenvalues depending on the number of test fuel elements (or the shape of the test region) have to be eliminated before one can be quite sure of the reliability of a heterogeneous - or homogeneous - program in the analysis of substitution experiments.

---

*) We have the intention to apply the code system FLEF-MOP for calculations on the core of the Marviken power station and therefore we have also tried to simulate the Marviken core conditions as intelligently as possible in the RO reactor [13] [14].

References

1. Persson, R et al.
Exponential Experiments on Heavy Water Natural Uranium Metal and Oxide Lattices.
U.N. Int. Conf. Peaceful Uses of Atomic Energy. 2. 1958. Geneva. Proc. Vol. 12, Geneva, 1958, pp. 364-373

2 Persson, R, Wikdahl C-E, and Zadworski, Z
Critical and exponential experiments on 19-rod clusters (R3 fuel) in heavy water
Nukleonik 4, 1962, pp. 191-199

3. Jonsson, A and Näslund, G
Heterogeneous two-group diffusion theory for a finite cylindrical reactor
AE-57 (1961)

4. Pershagen, B
Heavy water lattice calculations in Sweden
Heavy Water Lattices: Second Panel Report, Vienna 1963.
IAEA, Vienna, 1963, pp. 305-342

5. Persson, R
Perturbation Method of Analysis Applied to Substitution Measurements of Buckling
Nukleonik, 10, 1967, pp. 163-177

6. Lindh, R and Persson, R
SIMBA, Simple Iterative Method of Buckling Analysis.
Computer program and formulas
AE-TPM-FFR-155 (1969) (Internal memo)

7. Baumann, N P et al.
Lattice experiments with simulated burned-up fuel for D_2O power reactors
DP-1122 (1968)

8. Okazaki, A and Craig, D S
Determination of lattice parameters using a few rods
AECL-2593 (1967)

9. Corrigendum to ref. [8]
AECL-2593 (Supplement) (1969)

10. Jonsson, A and Näslund, G
Description of "heterogeneous" programmes for macroscopic reactor calculations.

Part I. The solution of the diffusion equation
AE-RFR-559 (1967)

Part II.1 The determination of boundary conditions in the two dimensional case
AE-RFR-663 (1967)

Part III. The calculation of quantities derived from the basic solution
AE-RFR-625 (1967)

Part IV. Numerical methods
AE-RFR-610 (1967)

Part V.1 MOP-DIP
AE-RFR-653 (1967)

Part V.2 MOPZ
AE-RFR-690 (1968)

Part V.3 HETERO
AE-RFR-603 (1967)

Part VI. Notes on the use of the programmes
AE-RFR-693 (1968)

Part VII.1 Programme details: MOP
AE-RFR-615 (1967)

Part VII.3 Programme details: MOPZ
AE-RFR-623 (1967)

Part VII.3 Programme details: MOPZ
AE-RFR-689 (1968)

Part VII.4 Programme details: HETERO
AE-RFR-614 (1967)

11. Edenius, M, and Jonsson, A
Results from the lattice code FLEF compared with experimental information: Studsvik experiments on heavy water cooled and moderated cores
AE-RFR-729 (1968)

12. Edenius, M
Calculations on heavy water cooled and moderated lattices of Marviken type by use of the codes FLEF and MOP-MOPZ.
A comparison with Studsvik experiments
AE-RFR-749 (1968)

13. Apelqvist, G, Jonsson,A and Norinder, O
Analysis of measurements on heterogeneous cores with Marviken-RO clusters and 27-rod clusters of natural UO_2
AE-TPM-RFR-674 (1967) (Internal memo in Swedish)

14. Wikdahl, C-E
Results of RO measurements with the Marviken enriched fuel (36-rod clusters) and natural UO_2 fuel (31-rod clusters): reactivity void coefficient, internal form factor and gap effect
AE-FFR-152 (1968) (in Swedish)

Status Report about Activities in Germany Concerning "Analysis of Few Rod Experiments"

F. Helm

Institut für Angewandte Reaktorphysik
Kernforschungszentrum Karlsruhe

E.A. Grün

Siemens AG, Erlangen

Few rod or small zone experiments have been performed in Germany on a number of thermal and fast reactor cores. Essentially the activities may be divided into four groups:

1) Lattice experiments on the SAR at the reactor station Garching of the Siemens AG,

2) Measurements of k_∞ in various light water lattices in the AEG test reactor,

3) Substitution experiments and investigations in small zones on Assembly 3B of the SNEAK critical facility at Karlsruhe,

4) Experiments in a k_∞=1 zone of Assembly 5 of SNEAK.

The first two items concern experiments on thermal lattices which were performed by industrial groups in the years from 1961 to 1967. The last two points include more recent investigations on fast reactor cores. Some of the thermal experiments are described in ref. 1 and 2. Methods and results of the SNEAK-3 substitution experiments may be found in ref. 3 and 4 or in more detail in the accompanying report on "A Method of Evaluating Progressive Substitution Experiments for the Determination of Bucklings and Critical Radii". The evaluation of the measurements on SNEAK-5 is still in progress, results will be available in the near future.

1. Lattice Experiments with Few Fuel Elements

In the reactor station Garching of the Siemens AG a series of neutron physics measurements on small lattice zones was performed during the years 1961 - 1967. It was the aim of these measurements to test calculational methods which had been developed for the design of power reactors.

The measurements were performed in the Siemens-Argonaut-Reactor (SAR) which is a low-power research reactor. The test lattices were introduced as so-called lattice inserts into the interior of the annular core of the SAR. Together with the SAR-core they formed a critical system. The smallest lattice inserts only contained 7 cells of the test lattice.

The following lattice structures have been investigated:

1) A simple natural uranium lattice in graphite (1960) for testing the experimental method,

2) a natural UO_2 bunched lattice in D_2O (MZFR-lattice, 1962),

3) various ThO_2-UO_2 lattices (partially as bundles, partially pseudo-homogenous with different fuel-to-moderator ratios) in D_2O such as are under consideration for the thermal thorium breeder.

Measured were "microscopic" lattice data, that is lattice properties which may be found by measurements of reaction rates in a single cell. In detail these were:

Radial distribution of the neutron flux in several energy groups, derived from the activation of different detectors. The measured activations were compared directly with multigroup cell calculations.

2-group absorption ratios in the fuel in order to deduce the resonance absorption in the U238 or Th232, respectively. From these data resonance escape probabilities were calculated.

The test cell was the central cell of the insert. For the reaction rates measured in this cell to be representative of the reaction rates in a complete lattice that is, in a lattice of critical size, the neutron spectrum at the outer boundary of the test cell must be made equal to the equilibrium spectrum of the test lattice. It was a special characteristic of the arrangement that this adjustment of the neutron spectrum was

relatively easy, namely by variation of the thickness and the material composition of the surrounding adapter zone. As an experimental criteria for the correct adjustment the energy independence of the buckling B^2 measured in the lattice insert was used. It is essential for the practical applicability of the method that an exact adjustment of the spectrum is not necessary. In principle such an adjustment is impossible as the leakage losses in the insert are different from those in the complete lattice. It turned out to be satisfactory to make an approximate adjustment and to eliminate the influence of the remaining mismatch by small corrections. The correction factors are determined from the dependence of the bucklings on the energy.

As far as it was possible up to now to test the results there was a satisfactory agreement with values found in complete lattices.

The rather early measurements on the MZFR lattice gave the desired confirmation of the design calculations for the relatively complicated rod bundle.

The measurements on the thorium lattice gave information on the still little known resonance absorption in thorium rod bundle elements.

2. Measurements of k_∞ in Light Water Lattices

These measurements were performed from 1961 - 1966 in the AEG test reactor at Großwelzheim. This is an Argonaut-type test reactor with a maximal power of 1o Watts. The test zone consisted of a lattice insert surrounded by a spectral adapter region of similar composition. It was the aim of the experiment to determine the amount of boron poisoning which brings to unity the k_∞ in the lattice insert. For a thermal lattice one then gets the k_∞ for the unpoisoned lattice from the relation

$$k_\infty = \frac{p' \Sigma'_{ath}}{p \, \Sigma_{ath}}$$

p and Σ_{ath} are the resonance escape probability and thermal absorption cross section for the unpoisoned core, p' and Σ'_{ath} the same quantities for the poisoned core.

The following experimental procedure was used: First the light water in the test zone (lattice insert plus adapter) is poisoned by an estimated amount of HBO_3. By small variations in the fuel-to-moderator ratio and the poisoning of the adapter region one then adjusts the flux gradients at the boundary between insert and adapter region to zero. For this purpose the distribution of the thermal and epithermal flux was measured with gold foils having aluminium and cadmium covers respectively; the fast flux was determined with indium foils using the $In^{115}(n,n')$ reaction. After this adjustment the reactivity of the insert is measured against air. Several such zero-void measurements are performed with small variations of the boron content in the insert. The concentration which exactly yields zero-reactivity is then found by interpolation.

Typical measurements for a UO_2 lattice are described in ref. 2. Other measurements were performed in boiling water reactor type lattices with different fuel to moderator ratios and rod diameters. Finally k_∞ for a superheated steam reactor lattice with hollow cylindrical fuel elements was measured in the normal condition where the coolant channels in the fuel elements are empty and in the flooded condition where this channels are filled with water.

3. Zoned Experiments in SNEAK-Assembly 3B

The experiments on SNEAK assemblies 3A and 3B concerned the physics of steam-cooled fast reactor cores. The cores were approximately cylindrical with about 8o cm height and 45 cm radius. In assembly 3A the core had an essentially uniform composition and was fueled with uranium. Assembly 3B had a central plutonium fueled zone of 29.9 cm diameter, while in the outer core zone the uranium fueled composition of assembly 3A was retained. In a modification of SNEAK-3B the composition of the structural material was changed in a central core zone of 2o.59 cm radius by replacing steel with molybdenum and nickel thus simulating the presence of Inconel. Finally in a second modification the whole plutonium-fueled zone was poisoned with a mock-up of fission products.

The transitions from SNEAK-3A to 3B as well as from SNEAK-3B to the "Inconel" modification were performed as progressive substitution experiments.

The accuracies reached in these measurements were about o.oo2 B_r^2 for the change in radial buckling and o.5 cm for the critical radius extrapolated to complete substitution.

Further, in all the zoned cores mentioned neutron physics parameters such as CH_2-void and flooding effect, neutron spectra and material worths were measured. Since the zones always were large enough so that spectra and adjoint spectra were very close to equilibrium at core center these measurements do not really fall under the category of "Few Rod Experiments". Such discrepancies as appeared between measurements and calculations will most probably have to be resolved by improving cross sections and calculational techniques and have no particular relationship to the application of zoned techniques.

The experimental program for SNEAK-3B also included the measurement of the coolant (or CH_2) reactivity coefficient in an environment containing an increased Pu24o concentration. For this purpose a small zone in the core center was fueled with platelets containing Pu with a 22% atomic fraction of Pu24o. (The Pu in the normal fuel paltes contains only 8% of Pu24o.) Due to the limited supply of platelets with the higher Pu24o contents the average distance from zone center to boundary was only 12 cm. Calculations showed that under this conditions the CH_2 reactivity coefficient could be measured with an accuracy of 25%. Due to technical difficulties during the experiment only a qualitative evaluation was possible. It indicated that the influence of the Pu24o on the coolant loss reactivity (which was expected to change its sign from negative to positive) had been considerably overestimated. Also the absolute loss of reactivity upon introduction of the Pu24o was significantly overpredicted so that as a general result of the experiment the need was emphasized to thoroughly revise the Pu24o cross sections (ref. 5) which were used in the calculations.

4. Measurements in a k_∞=1 Zone of SNEAK-Assembly 5

This assembly consisted of a Pu-fueled cylindrical test zone (76 cm height and 35.9 cm radius) surrounded axially and radially by a uranium-fueled driver zone and a depleted uranium blanket. The aim of the experiment was to study the neutron balance by absolute measurements of the reaction rates of Pu239 (fission),U235 (fission),U238 (fission),and U238 (capture) in a non-leakage environment. If these measurements can be performed with sufficient accuracy it is possible to derive an energy averaged value for α(Pu) from the results.

The unit cell of the test zone was built in the following way:

n	graphite platelets, n = (7, 1o, 13)
1	platelet U(nat)
1	" PuO_2-UO_2
1	" U(depl.) .

A large component of graphite was chosen to soften the spectrum and thus make the measurement more representative of the energy range of interest (o.1 - 2o keV). The plutonium and uranium plates were grouped together in order to eliminate as far as possible strong spectral variations from the reaction rate fine traverses. The number (n) of graphite plates in the cell was increased in 2 steps in order to bring k_∞ closer to one. As a test for this condition zero void experiments were performed by removing five unit cells out of the center of maximal 9 elements. Since the steel of the reactor matrix could not be removed complementary steel worth measurements were also performed. As a test of the flatness of the flux distribution reaction rate traverses for U235 (fission) and U238 (capture) were performed with uranium metal foils. In the original configuration (with 7 graphite plates) k_∞ was, according to preliminary evaluations, several percent above one, in the final configuration (n=13) the deviation from unity was about halfed.

In the center of the final core (Assembly 5C, 13 graphite plates) the following reaction rates were measured:

U238 (capture) by metal foils using the coincidence counting method,
U238 (fission) by track foils, and small fission chambers,
U235 (fission) by the same methods as U238 fission,

Pu239 (fission) by small and large fission chambers.

In order to get an average value for the reaction rates the plutonium and uranium reaction of one unit cell in the core center was vertically split using half-platelets of the core materials. Foils were placed vertically in the resulting slot and also horizontally between the individual platelets. Small fission chambers were placed in a vertical channel provided through the center of the platelets. When large fission chambers were used a cavity was provided by removing several unit cells.

At the present state the measurements have been completed, however, the results are still being evaluated.

References

1) Riesch, G., "Messung des Multiplikationsfaktors k_∞ eines Wasser-UO_2-Reaktorgitters", Nukleonik 5, 1o4 (1963)

2) Dio, W.H., Grün, A.E. und Oldekop,W., "Experimentelle Bestimmung des p-Faktors von Reaktorgittern mit Hilfe des Argonaut-Reaktors" Nukleonik 5, 163 (1963)

3) Edelmann, M., Fischer, E.A., Helm, F. and Schröder, R., "Physics Measurements in the SNEAK-Facility on Steam-Cooled Fast Reactor Systems with Uranium and Plutonium Fuel", Proc. of the BNES Int. Conf. on The Physics of Fast Reactor Operations and Design, June 1969, to be published.

4) Helm, F., "Experiments on SNEAK Using the Method of Progressive Substitution for Critical Mass Determination in Fast Systems", Trans. American Nucl. Society 11, 595 (1968)

5) Abagjan, L.P. et al., German Translation of "Group Constants of Fast and Intermediate Neutrons for the Calculation of Nuclear Reactors" States Comittee for the Use of Atomic Energy in the USSR

ATATUS ACTIVITY REPORT ON FEW - ROD EXPERIMENTS IN JAPAN

Iwao Kobayashi

A serial critical experiment has been carried out at TCA of JAERI in cooperation with PNC, in order to get the basic informations concerning the utilization of plutonium enriched fuel in light water reactor. These experiments belong to the "FEW - ROD EXPERIMENTS". They are a) single rod experiment, b) 9 rods experiment and c) 7 × 7 lattice experiment. The experiments of a) and b) are almost finished and the analysis about a) is in progress. The experiments of c) started this January and is under way.

a) The purpose of the single rod experiment was to obtain the activation traverses in and in the vicinity of a PuO_2 - UO_2 fuel rod and to measure the equivalent reactivity worth of it in the inner moderator region of an annular core.

The measured PuO_2-UO_2 fuel elements were nine rods which included three kinds of PuO_2 enrichment (1.40, 2.06 and 2.52 w/o PuO_2-UO_2) and three kind of fuel diameters (9.5, 10.8 and 12.7 mmØ), and were fabricated by means of vibratory compaction. Boric acid was used to hard the neutron spectrum in the inner moderater region; 0, 1000, 2500 ppm B to H_2O for the reactivity measurement and 0, 2500 ppm B to H_2O for the activation measurement. Activation detectors were Dy-Al, Au, Lu, U and etc. with and without Cd filter. The fissile atom ratio of Pu-239 to U-235 in an irradiated fuel rod will be determined non-destructively by the reactivity measurement in the moderator region.

The 35 groups-THERMOS code is utilized for the analysis of the activation distribution. The calculated result over-estimates about 4 ~ 10% the neutron flux depression in the fuel as the ratio of the flux at the fuel surface to the center.

b) Power distribution in the 3× 3 test lattice of 2.5 w/o PuO_2-UO_2 and the driver region of UO_2 was measured by the gamma scanning method. This PuO_2-UO_2 assembly was scheduled to be irradiated in the HBWR in Norway.
Hence the purpose of this experiment was to estimate the relative power density in the assembly loaded in the D_2O moderated HBWR. It was impossible to make a mockup of the HBWR, the power distribution was measured parametrically on the cases the test lattice region was flooded and unflooded changing the surrounding water gap between 2.6 w/o enriched UO_2 driver region.
Power distribution was calculated by the PDQ-5 code using group constants obtained by THERMOS and HRG. Calculated results agreed in about 5% descrepancy with experimental values in the power distribution traverses in two regions of PuO_2-UO_2 and UO_2. The gamma activity yield of PuO_2-UO_2 and UO_2 was corrected by the double fission chamber technique.
c) 7 × 7 lattice experiment is in progress to obtain the two region critical configurations and the activation traverses in and in the vicinity of a test fuel rod in the center of the lattice. The test fuel rod for activation traverses is one of the rods used in section a); 2.5 w/o PuO_2-UO_2, 10.8mm dia., 84% T. D.* The fuel element in the lattice is, however, 3.4 w/o PuO_2-UO_2, 60% T.D. and 10.7 mm dia.** The driver region is 2.6 w/o enriched UO_2 of which volume ratio is 1.84. Volume ratio, V (H_2O)/V(fuel), in the lattice will be parametrically changed; 1.8, 2.0, 2.4, 3.0.
The THERMOS code is used to calculate the activation traverses. Criticality calculation will be performed by the two dimensional diffusion code KAK or PDQ using group constants obtained by THERMOS and HRG or UGMG.

In case of design calculation on the local power distribution in a fuel assembly of a power reactor, usually the symmetrical boundary condition of neutron flux is assumed. However, the experimental power distributions are obtained

---

* U-235 1.427×10^{20} atom/cm^3 Pu-239 4.784×10^{20} atom/cm^3
** U-235 1.062×10^{20} atom/cm^3 Pu-239 4.632×10^{20} atom/cm^3

in a finite system where the condition is not satisfied and can not be compared directly with the calculated values. In a zero power critical experiment especially the measurement must be performed in a fairely small core assembly since it is not economical to charge enough fuel materials to imitate a power reactor core entirely. Therefore a particular consideration will be necessary to compare the design values with the experimented.

The three kinds of method are taken into consideration for the direct comparison of the experimental values to the calculated. They are;

a) the calculated values assuming symmetrical boundary condition are corrected by the over all power distributions calculated with coase meshes including the whole experimental system,

b) the proper boundary condition for the fuel assembly in the experimental system are taken in the calculation, where the condition must be determined experimentally by measuring neutron flux distribution through the poison curtain, control rod follower and water slot, and

c) the measured values are corrected in terms of the geometrical buckling of the finite system.

The method of c) is proved experimentally for the light water lattices in TCA (light water critical assembly) at JAERI and applied to the test fuel assemblies for the JPDR.

A unique experimental method "VISTA Method" is proposed and now being under way to measure the fine neutron flux distribution in the light water lattice in SHE (Semi Homogeneous Experiment Assembly) at JAERI. In the VISTA method the experimental system concerning a neutron collision with other nuclides is expanded by a factor of ρ which is the diluted density of the system, and material densities (i.e. macroscopic cross sections) of each region are reduced by a factor of ρ . Then the space and time dependent phenomena is observed expanded in the system by the factor of ρ . These structural materials were fabricated by porous polystyrene and low density UO_2 pellets.

Summary Report on the Analysis of Few Rod Experiments at Ispra

by

G. CASINI

Introduction

In the last five years at Ispra a considerable effort has been devoted to the assessment of experimental methods for the evaluation of the nuclear properties of multiplying media. The investigation has been concentrated on well thermalized (heavy water) lattices, in different critical and subcritical assemblies: ECO (critical), EXPO (subcritical) and RB-1 (critical, in cooperation with CNEN). In order to have as clear a comparison as possible of the different measuring techniques, care has been devoted to using, as much as possible, the same fuel elements, heavy water, counting equipment and staff for analysis.
In all the studies, cluster-type fuel elements have been considered with different fuels (uranium metal, carbide, oxide) and number of rods (4, 7, 19). In the present summary report, the main emphasis will be devoted to two types of fuels; an uranium metal nineteen rod cluster (which forms the standard loading of ECO) and an uranium carbide seven rod cluster. The main features of these fuels elements are given in the appendix.

1. Substitution Experiments

The substitution experiments have been carried-out in ECO and EXPO. In ECO from 1 to 25 fuel elements have been replaced, whereas in EXPO the maximum number of substituted elements was 16.
The accuracy in determining the difference in critical water level in ECO was mainly influenced by the precision of the D_2O level meter ($\pm$ 0.1 mm), and, to a lesser extent, by the built-up of the long lived delayed photeneutron sources and local temperature variations in the moderator. The total experimental error was of the order of 1 pcm. The accuracy of the measured relaxation length and flux distribution was mainly affected by the neutron counting statics. Generally the relaxation length was measured with an accuracy of 0.2%.
For the last two years the analysis of these experiments at Ispra has been performed by heterogeneous methods. An improved version of the Swiss code SOS (ref. 1) has been written for the IBM-360 and used for the reactor calculations. The main features of this method are:

- finite radius of the fuel elements, with boundary conditions at the surface of them and monopole and dipole terms considered,
- diffusion theory in the moderator and
- three energy groups.

The calculation of the channel nuclear parameters has been done by the lattice code PINOCCHIO (ref. 2) on the basis that, for the case of an infinite lattice, they have to be consistent with the neutron balance evaluated by cell theory (ref. 3). For each measured core configuration the monopole thermal neutron production parameter ($\gamma_{13}(0)$) of the test fuel elements is adjusted to get agreement between experiments and theory.
In table I are given the ECO results for the carbide fuel elements 7/UC/25.2 and 7/UC/30.9 at different lattice pitches.

Table I - Fitted thermal neutron production parameters in ECO experiments

Fuel	Pitch (cm)	Number of substituted rods				
		1	5	9	13	21
UC/7/25.2	18 8	.11188	.11208	.11206	.11207	.11208
	23.5	.11082	.11052	.11045	.11046	.11047
	26.6	.10942	.10962	.10960	.10959	.10963
UC/7/30.9	23	.09359	.09362	.09351	.09352	---
	28	.09209	.09209	.09203	.09210	---

The results show that the maximum difference in the adjusted $\gamma_{13}(0)$, and thus in K-inf, never exceeds 2°/$_{oo}$ without having a systematic trend. This indicates that by the use of a suitable model for the interpretation, it is possible to strongly reduce the number of test fuel elements involved in critical substitution experiments even if the nuclear properties of the test and reference lattices are sensibly different.
In table II are given the results of relaxation length measurements in EXPO with the same fuel elements (7/UC/30.9). Replacements from four to sixteen fuel bundles have been considered.

Table II - Exponential substitution experiments (EXPO Uranium carbide fuel elements (7/UC/30.9)

Pitch (cm)	Number of substituted rods 4	12	16
23	0.09354	0.09370	0.09393
28	0.09250	0.9251	---

The fitted $\alpha_{13}(0)$ values are internally consistent but higher (up to 5°/$_{oo}$) than the corresponding values derived from the ECO measurements. The reason for this discrepancy, confirmed by other EXPO experiments, has not been completely explained. In all the experiments mentioned above a two dimensional method of analysis was used. A three-dimensional version of the Auerbach method, without the introduction of the dipole terms, has been applied to the analysis of substitution experiments in ECO with oxide fuel elements of CIRENE-type containing joints (ref. 4). A comparison between the results of two and three dimensional calculations has resulted in checks of the reliability of the homogenization models for dealing with joints in lattice calculations.
An experimental investigation to determine the temperature coefficient of the fuel element channel is underway in ECO. For this investigation a loop has been installed to heat up to nine central fuel channels of the pile. Reactivity measurements are carried-out for a fixed number of heated channels (1, 5, 9 successively) at different coolant temperatures. First results indicate that, from the analytical point of view, there are no difficulties in interpreting the measurements with a reduced number of heated channels due to the fact that in all the experiments the temperature of the moderator remains constant. The limitation arises in the experimental determination of the reactivity.

2. Single Rod Experiments

2.1. Oscillation experiments

Single rod measurements to investigate the nuclear properties of plutonium bearing fuel elements by the oscillation technique have been carried out at Ispra. The fuel elements were basically those of the standard ECO loading with different amounts of plutonium (ranging from 0.05 to 0.3%). Schematically the experiments consisted

of oscillating a 50 cm long section of the Pu-U fuel versus a natural uranium fuel section of the same geometry from an out-of-core to an in-core position according to a square wave function. In such a way the effect of plutonium in cluster-type fuelled lattices was obtained without the necessity of separating the uranium-238 absorption, which is strongly dependent on geometry and difficult to calculate. The experimental accuracy achieved was of the order of $\pm$ 0.05 x 10^{-5} equivalent corresponding to about twice the signal-to-noise ratio. The results of the experiments indicate that the main source of error arises from the uncertainties in the isotopic compositions while the accuracy of the oscillation technique is very satisfactory.
This is shown in table III where the experimental ($\Delta_{exp.}$) and composition ($\Delta_{comp.}$) errors are given together with the reactivity signals of the test samples. All values are referred to natural uranium in units of pcm.

Table III - Errors related to the oscillating experiments *

Pitch (cm)	Fuel	Exp.results (pcm)	$\Delta_{exp.}$ (pcm)	$\Delta_{comp.}$ (pcm)
18.8	Pu- II**	-16.7	.008	1.7
	Pu-III	+20.0	.010	0.3
	Pu- IV	-66.2	.011	1.5
23.5	Pu- II	-26.2	.014	2.4
	Pu-III	+31.2	.017	0.4
	Pu- IV	-102.5	.012	2.3

** Pu-II 0.21%U-235, 0.30%Pu (8.45% Pu-240)
Pu-III nat.uran., 0.05%Pu (25.5% Pu-240)
Pu-IV 0.23%U-235, I.26%Pu (27.8% Pu-240)

In the analysis the main problems, apart from the different composition between the test sample and the environment, are due to the fact that the amount of moderator associated with the test element was longer as compared to that of the boundary because of the presence of the oscillation tube. In order to deal with these effects the interpretation of the experiments has been carried out by a multigroup one-dimensional perturbation method set up at Ispra (code PERMULT)(ref. 5).

*(Note: the experimental errors quoted do not include the error in converting from the true experimental units of $\Delta k/k$ to pcm. The use of pcm in this and the following table are for convenience only).

The main hypotheses in the calculations are:

- the assumption that the flux perturbation due to the oscillating sample is significant only in the vicinity of the sample and therefore the calculation is limited to a macrocell surrounding the central experimental cluster,
- the use of an S_4- approximation without angular dependence for the calculation of the unperturbed flux and the adjoint in the macrocell and
- the use of the same number of energy groups (39) as in the code PINOCCHIO.

The usefulness of such an approach can be seen by comparing the reactivity worths calculated for the oscillating cluster in an infinite lattice to that calculated in the macrocell configuration. The results, shown in table IV, indicate that spectral differences in the two geometries lead to errors which increase with an increase in the plutonium enrichment.

Table IV - Reactivity worths for microcell and macrocell geometries at pitch 18.8 (pcm)

Test fuel	1: Microcell	2: Macrocell	1-2
Pu- II	- 3.01	- 7.62	+ 4.61
Pu-III	+22.88	+22.21	+ 0.67
Pu- IV	-56.49	-59.21	+ 2.72

Concerning the geometrical approximations in the PERMULT calculations, the effects related to the size of the macrocell have been found to be negligible for the lattice pitches investigated provided that a macrocell consisting of the central cluster and of the adjacent eight clusters is considered. As far as the schematisation of those eight surrounding fuel elements is concerned, they have been homogenized into an external annular ring at a distance from the center so as to have the same plutonium reaction rates in the test fuel element as in the case of a cell calculation (infinite lattice conditions).
To conclude, we think that the experimental technique and

the assumed method of interpretation are satisfactory for the analysis of this kind of experiment (with the main difficulties arising from the uncertainties in the isotopic content). In the next year it is planned to apply these methods to the analysis of 2% in U-235 or in U-233 enriched, thorium fuel elements.

2.2. Single rod imbedded in a moderator

A basic research program has been carried out by CESNEF-SNAM under contract with EURATOM on the possibility of evaluating the neutron absorption and multiplication parameters by flux mapping outside of the fuel for the case where a fuel element is imbedded in a moderator. This experiment has been performed in the thermal column of the CESNEF-reactor (Milano) and has used a fuel element of the ECO-type previously mentioned. Codes based on source-sink methods for reactor calculations have been set-up in order to correlate the experimental flux mapping in the moderator to the unknown values of the heterogeneous constants of the fuel. Difficulties connected with the specific design of the experiment (mainly concerned with the reduced heigth of the graphite stack) limited the accuracy of the experiment but it was shown that it is possible to get sufficient accuracy in the thermal production factor (ref. 6) with suitable measurement conditions.

3. Zero-Reactivity Experiments

In the field of zero-reactivity measurements for the evaluation of K-inf, EURATOM has gained considerable experience in the last few years through a cooperative program with CNEN of Bologna (Italy). The central part of the RB-1 graphite reactor has been replaced by a tank for heavy water measurements. The fuel elements (U/19/12 and UC/7/25.2) described in the appendix have been tested together with the plutonium bearing fuels mentioned in Chapter 2.
The application of the zero-reactivity technique to cluster-type fuelled lattices poses a certain number of problems which are related mainly to heterogeneous effects and to the replaced core fraction. From the experimental side, no particular difficulties have been found in obtaining the flattening and matching of the flux provided that the central test fuel elements is surrounded by eight fuel clusters of the same type. From the analytical point of view, the main difficulties arose in the evaluation

of the so-called "heterogeneity correction". In fact, the angular distribution of the neutrons changes when the test-cell section is replaced by void in the case of non-homogeneous media. This induces a perturbation of the flux distribution in the surrounding buffer and consequently a variation of the neutron production and absorption rates. A careful analysis of this point has shown that first order perturbation methods of the type proposed by Humbach and Holdekop (ref. 7) are not adequate for the evaluation of this correction for large void volumes (length of test cell about 50cm). For the lattices investigated, the correction has been evaluated by a two dimensional, S_4-code in a two-energy-group scheme. The results, shown in table V, indicate the importance of a proper evaluation of the heterogeneity correction in the analysis of zero-reactivity measurements.

Table V - Heterogeneity effect

Type of element	Pitch	K-inf	ΔK_{het} (pcm)
U/19/D	18.8	1.0902	690
	23.5	1.1279	960
	26.7	1.1334	1110
UC/7/D	23.5	1.0953	1020
	26.7	1.1090	1360

In order to limit the uncertainties connected with this effect, the thermal and resonance components of the correction now are evaluated experimentally from reaction rates for 1/v detectors and relative conversion ratio measurements in the fuel element sections adjacent to the void cell position (ref. 8).
In table VI are detailed the different contributions to the error in the K-inf measurements investigated by EURATOM.
The comparison between substitution and zero-reactivity techniques is not straight-forward because the derived quantity is the material buckling in the former type of measurements and is K-inf the latter type.
In table VII the results of such a comparison in terms of K-inf are given with leakage calculations performed by the two ISPRA lattice codes PINOCCHIO and HEROIC (ref. 9).

Table VI - Zero-reactivity technique: errors in K-inf (pcm)

Type of element	Pitch	Copper mass	Disadvan. factor	Heter. correct.	Total
U/19/D	18.8	115	150	140	280
	23.5	110	220	190	410
	26.7	120	220	220	470
UC/7/D	23.5	90	90	200	320
	26.7	150	220	270	410

Table VII - K-inf from different experimental techniques

Type of element	Pitch	RB-1	ECO	
			PINOCCHIO	HEROIC
U/19/D	18.8	1.0902	1.0906	1.0905
	23.5	1.1279	1.1313	1.1288
	26.7	1.1334	1.1412	1.1373
UC/7/D	23.5	1.0954	1.1034	1.0939
	26.7	1.1089	1.1034	---

Taking into account that the HEROIC method is more accurate than that used in PINOCCHIO the agreement between RB-1 (zero-reactivity measurements) and ECO (substitution measurements) is satisfactory. The zero-reactivity method has been extended to the investigation of the plutonium bearing fuel elements mentioned in paragraph 2. A theoretical analysis has shown that no sensible variations in K-inf (less than 150 pcm) can be attributed to the fact that the test cell is surrounded by natural uranium. Work is in progress to compare the results of the oscillation (ECO) and zero-reactivity (RB-1) experiments.

This report is based on work performed by:

A.BACCOLINI*, F.BEONIO-BROCCHIERI, G.BIRKHOFF
A.BOEUF, L.BONDAR, F.CASALI*, G.CASINI,
B.GARAGNANI*, V.GIACOBBI*, W.HAGE, H.HETTINGER,
H.HOHMANN, G.MARACCI, E.MACKE, M.PAILLON,
L.PAPAY, F.RUSTICHELLI, B.STURM, S.TASSAN, F.TOSELLI

* CNEN Bologna

Appendix

U/19/12/Diphyl reference fuel elements

Number of fuel rodlets (U nat.)	19
Radius of fuel rodlets	0.6 cm
Radius of canning (Al)	0.7 cm
Distance between rods	1.52cm
Internal radius of pressure tube (Al)	3.84cm
External radius of pressure tube	3.99cm
Internal radius of calandria tube (Al)	4.09cm
External radius of calandria tube	4.24cm
Organic coolant	DIPHYL (or DOWTHERM A)

Uranium carbide test fuel elements

	UC/7/25.2	UC/7/30.4
Number of fuel rodlets (UC)	7	7
Radius of fuel rodlets	1.26 cm	1.545 cm
Radius of canning (Al)	1.375cm	1.66 cm
Distance between rods	2.95 cm	3.57 cm
Inter.radius of pres. tube (Al)	4.55 cm	5.5 cm
Exter.radius of pres. tube	4.65 cm	5.7 cm
Internal radius of calandria tube	5.05 cm	
External radius of calandria tube	5.20 cm	
Organic coolant	DIPHYL	

References

1. T. AUERBACH and al.: The Heterogeneous Method and its Application - Third IAEA Conference 28/8/690, 1964

2. L. AMYOT and al.: PINOCCHIO, A Computer Programme for Cell Reactivity Calculations - EUR 4231 E. 1969

3. G. CASINI and al.: The Use of Heterogeneous Methods at ISPRA - BNES Conference on Physics Problems in Thermal Reactor Design - 1967

4. F. ACCINNI and al.: Buckling Measurements in ECO with Few CIRENE Fuel Elements - CISE-R-283, 1968

5. G.CASINI and S. TASSAN: Reactivity Studies on Plutonium - Uranium Cluster Fuel Elements in Thermal Lattices - Paper presented at the 11th EACRP meeting , London 10-15 Feb. 1969

6. S.E. CORNO and al.: Determinazione dei Parametri di Cattura e Moltiplicazione Neutronica di Grandi Elementi di Combustibile - CESNEF, 1967

7. W. HUMBACH and W. OLDEKOP: Zur Theorie des Reaktivitat-sbeitrages von Honlraumen und eines Verfaherens zur Bestimmung von K-inf - Nukleonik 1 (1959) p. 265

8. F. CASALI, F. ACCINNI and al.: K-inf and Deatiled Parameter Measurements for UO_2 Heavy Water Fog Cooled Lattices in RB-1 Facility - Report presented to the Panel on Analysis of Few-Rod Experiments-Ispra, May 1969

9. R. CUNIBERTI and C. DAOLIO: HEROIC, A Multigroup Cell Code for Cluster-Type Fuel Elements - EUR 4464 • (1970)

CHAIRMEN's SUMMARIES AND CONCLUSION

SESSION ON "SUBSTITUTION EXPERIMENTS AND ZONED LOADINGS"

(R.PERSSON)

The incentive for applying the substitution technique is the attractive possibility to determine critical-lattice parameters (usually material bucklings) from a strongly subcritical amount of expensive and rare fuel material. The advantage of a small amount of test fuel is more or less counter-balanced by

- an increased number of necessary measurements,
- special adjustements to get a suitable reference lattice, and
- a more complicated analysis.

The progressive replacement technique has been applied successfully in thermal reactor systems moderated with graphite or heavy water, where the transport mean free path is small compared to the lattice structure. The usefulness of the replacement method in fast reactor systems is not obvious. However, the substitution technique has been used in French and German fast critical assemblies for the determination of material bucklings. The results are interesting but maybe not quite conclusive.

If - owing to difficulties in the method of analysis - the substitution technique has to be limited to cases with good spectral matching, the theoretical situation is not satisfactory. Good spectral matching means that the result is insensitive for the way of extrapolation to a full-size core. Systematic variations in the final result depending on the size of the test region should not appear with a good theoretical approach, even if there is spectral mismatch. The experimental accuracy should be the limiting factor, that defines the minimum size of a useful test region. Swedish experiments indicate clearly that even single test fuel elements give adequate information, especially regarding the effective diffusion coefficient.

The German substitution experiments in a fast critical facility reported by Helm show that the diffusion coefficient cannot be determined by the conventional progressive replacement technique. The position of the test region has to be varied, if diffusion coefficients are to be studied. Different methods of analysis have been applied at different laboratories. There are heterogeneous and homogeneous models and different definitions of cells. The results reported here show that a heterogeneous program does not necessarily give more accurate results than models where the cells are homogeneized in one or another way.

SESSION ON "SINGLE ROD EXPERIMENTS"

(B.LAPONCHE)

During the session on single rod experiments the following points arose :

1. The usefulness of these experiments is obvious when one wants to know the neutronic properties of expensive fuel (such as plutonium-bearing fuel) or fuel elements with a unique composition (e.g. irradiated fuel). Both types of fuel element have been studied in different Countries.

2. There was little discussion of experimental techniques but it was agreed that one should minimize end effects as far as possible by the provision of axial buffer elements at either end of the fuel element under investigation.

3. If one is concerned with the overall balance of neutron absorption and production in the test element then a measurement of the reactivity change produced by the introduction of this element is adequate. It is useful, however, to supplement this measurement by a determination of all cell reaction rates of interest. Auxiliary measurements such as a reactivity calibration of the reactor with a $1/v$ absorber or integral measurements of neutron spectrum help to reduce errors of interpretation.

   It was emphasized that the most important source of error in measurements of this kind arose from uncertainty in the composition of plutonium-bearing fuel.

4. Separate measurement of the absorption and production of neutrons in a fuel element can be achieved by means of a method developed in France which involves a determination of the reactivity change produced by insertion of the element together with a measurement of the local variation of neutron density by a fission chamber technique. In these measurements a double calibration of the reactor is carried out using boron and U235. The method allows a measurement of η to be made without knowledge of the plutonium content of the fuel sample.

5. In the interpretation of the measurements the calculation of perturbation effects plays an important part. The HPS method calculates the perturbed flux in the sample and its environment. The calculation is confined to a central zone of the reactor, the effect of the outer part of which is represented by an appropriate boundary condition. The method appears to be applicable to a wide range of experimental situations.

   The "equivalent sample" method which has been developed only for the case of a "thermal" perturbation uses the double calibration of the reactor to confine the calculation to a small central zone (the central cell and the ring of adjacent cells) in which environment effects and local corrections are calculated.

6. In the extension of the "single rod" method to more difficult applications the increased importance of the environment correction may lead to errors because of the difficulty of determining whether a given discrepancy between theory and experiment arises from an incorrect calculation of the properties of the test cell or of the environment correction.

   It was agreed that it would be generally useful to separate the experiments concerned with the checking of fundamental data from those concerned with the study of environment effects.

   In the consideration of future programmes the current uncertainties concerning the best method to adopt in fast reactor work is illustrated by the fact that the "single rod" work plays an important part in the CEA programme but is not used by the UKAEA.

   In any case it was pointed out that the measurement of reaction rate ratios at the center of a minimum-size zone was of the greatest interest.

7. The type of experiment in which a single fuel element is placed on the axis of a thermal column was discussed only briefly since this measurement is based on a concept very different from that of the lattice experiments. The usefulness of comparative measurements in this case was stressed.

SESSION ON "VOID-ZERO-REACTIVITY EXPERIMENTS"
(L.SANDERS)

1. The value of the null-reactivity method for setting-up lattices of unit k-infinity has been shown by the work reported to this panel on thermal, intermediate and fast neutron systems.

2. The null-reactivity method has the advantage of providing an absolute value of k-infinity, whereas the single-channel reactivity technique has, up to now, been used only on a relative basis for comparing similar fuel elements. However, if the cell of interest is very reactive so that a large poison addition is needed, this advantage may be offset by the uncertainties introduced in relating the unpoisoned and poisoned systems.

3. Various criteria for the adequacy of the spectrum matching in null-reactivity experiments are in use. The corrections to the k-infinity determination are of a lower order than those to the individual lattice reaction rates and it appears that the matching criteria for good k-infinity measurements are not very stringent.

4. In heterogeneous systems, the detailed flux distribution on the exposed cavity surface is different from that before removal of the test cell and this effect introduces significant corrections to the k-infinity measurement. These corrections, which appear to be major contributors to the overall uncertainty, are currently evaluated by a combination of theory and experiment. Additional information is obtained if it is possible to vary the size of the test lattice section removed, since the surface correction depends on the cavity surface/volume ratio.

5. As in all reactivity measurements on small samples of a lattice, chemical and isotopic composition uncertainties can contribute significantly to the overall error and require careful evaluation.

6. On the basis of the reports presented to the Panel the overall errors on k-infinity in the current applications of the technique lie in the range 0.4 - 1%.

CONCLUSION

(R.RICHMOND)

1. Substitution measurements

In a preliminary discussion on methods of measurement it was generally (though not unanimously) agreed that the determination of material buckling, normally attributed to the "substitution" method, was not particularly useful. In the case of a large power reactor with small leakage, for example, K-infinity was the parameter of most importance. In addition the buckling was only useful as a parameter if energy independent and this implied measurement using a large sample. In particular it was clear that in the limit of a single rod substitution the quantity measured was not buckling but K-infinity. Similarly it appeared that the fast reactors substitution experiments did not measure buckling because calculated leakage parameters were included in the analysis. The remainder of the session was therefore devoted to discussion of the measurement of the balance of neutron absorption and production by "single-rod" and null reactivity experiments.

2. Single-rod experiments

2.1. Thermal reactor lattices

In the case of graphite - and D_2O - moderated thermal reactors the minimum sample required is, in fact, a single fuel element since flux depressions due to individual resonances in the fuel element spectrum are removed by moderator collisions and do not appear in the cell-edge spectrum. The analysis may therefore proceed without consideration of the differences in the detailed spectrum in two adjacent fuel elements. In practice it has been found that a length of about 50 cm of the fuel element is adequate for the measurement but that, in this case it is advisable to use axial buffer elements at either end of the test element.

The accuracy achievable will clearly depend on the detailed circumstances of the measurement and some typical figures were therefore given to illustrate the potential of the method.

A series of measurements carried out at Winfrith on graphite-moderated systems with plutonium-uranium metal rod and PuO_2/UO_2 cluster elements gave overall errors of the order of ±0.7% in k-infinity and $\pm 1 \times 10^{-5}/°C$ in lattice temperature coefficient. This work was carried out without axial buffers, the use of which would have improved the accuracy in the case of the oxide cluster experiments. Euratom measurements on plutonium-uranium metal rod clusters have been analysed in terms of the value of η for Pu239 and the errors correspond to an error of the order of ±0.5% in this quantity. A series of French measurements on single rods of plutonium-uranium metal were quoted as giving values of η_8/η_5 with an error of ±0.2% and values of fission and absorption cross-sections to ±1%.

It was emphasized that, in all cases except the French ratio measurement, a considerable part of the error arose from uncertainty in the plutonium content of the test elements. Any significant improvement of the accuracy of the single element measurements will be dependent on an improvement in the technique of chemical and isotopic analysis.

2.2. Fast reactor lattices

In order to carry out a measurement in a fast reactor lattice analogous to the thermal reactor single rod measurement it is necessary to use a fuel sample sufficiently large to create, in its central region, a spectrum with the correct microscopic structure. To date the minimum quantities of test fuel used in such investigations have contained of the order of 50 kg of fissile material but it was generally agreed that the minimum quantity of fissile material required would be of the order of 10 kg.

3. Null reactivity measurements

The panel generally agreed that the null reactivity measurement was of value in providing an absolute measurement of k-infinity which could be used in combination with reaction rate ratio measurements, to check nuclear data.

For this purpose an improvement in the currently quoted errors of 0.4 - 1% in K-infinity is desirable. This requires a reduction in the uncertainties relating to neutron streaming in the void in heterogeneous lattices or, alternatively, the use of homogeneous systems. As in the case of single-rod measurements, an improvement in the techniques of chemical and isotopic analysis is also necessary.

For the general measurement of neutron balance the single-rod type of measurement appeared preferable to the null reactivity measurement since, while achieving a similar accuracy in k-infinity, it required a significantly smaller amount of fuel and could also be carried out much more rapidly.

4. Theoretical methods

The main requirement in the field of theoretical analysis is for a method of calculation which can be used in the associated fields of (1) the effect of resonance neutron streaming in voids in heterogeneous lattices (2) the microscopic spectrum mismatch occurring at the interface between two fast reactor lattices. This would improve the accuracy of the null reactivity measurement and reduce the size of sample required for "single-rod" fast reactor measurements. It was suggested that differential Monte Carlo techniques might be applied in this field.

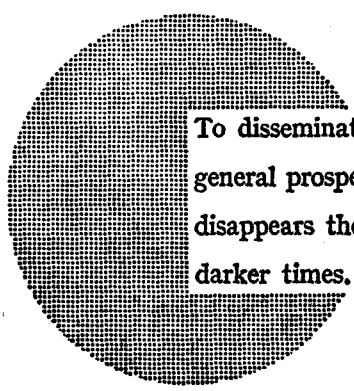
NOTICE TO THE READER

All scientific and technical reports are announced, as and when they are issued, in the monthly periodical **"euro abstracts"**, edited by the Centre for Information and Documentation (CID). For subscription (1 year : US$ 16.40, £ 6.17, BF 820) or free specimen copies please write to :

Handelsblatt GmbH
" euro abstracts "
Postfach 1102
D 4 Düsseldorf 1 (Deutschland)

or

Office de vente des publications officielles
des Communautés européennes
37, rue Glesener
Luxembourg



To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel

SALES OFFICES

All reports published by the Commission of the European Communities are on sale at the offices listed below, at the prices given on the back of the front cover. When ordering, specify clearly the EUR number and the title of the report which are shown on the front cover.

SALES OFFICE FOR OFFICIAL PUBLICATIONS OF THE EUROPEAN COMMUNITIES

37, rue Glesener, Luxembourg (Compte chèque postal N° 191-90)

BELGIQUE — BELGIË
MONITEUR BELGE
Rue de Louvain, 40-42 - 1000 Bruxelles
BELGISCH STAATSBLAD
Leuvenseweg 40-42 - 1000 Brussel

DEUTSCHLAND
BUNDESANZEIGER
Postfach - 5000 Köln 1

FRANCE
SERVICE DE VENTE EN FRANCE
DES PUBLICATIONS DES
COMMUNAUTES EUROPEENNES
26, rue Desaix - 75 Paris 15e

ITALIA
LIBRERIA DELLO STATO
Piazza G. Verdi, 10 - 00198 Roma

LUXEMBOURG
OFFICE DE VENTE
DES PUBLICATIONS OFFICIELLES
DES COMMUNAUTES EUROPEENNES
37, rue Glesener - Luxembourg

NEDERLAND
STAATSDRUKKERIJ
Christoffel Plantijnstraat - Den Haag

UNITED KINGDOM
H. M. STATIONERY OFFICE
P.O. Box 569 - London S.E.1

Commission of the European Communities
D.G. XIII - C.I.D.
29, rue Aldringer
Luxembourg

FB 165,— FF 20,55 DM 13,55 Lit. 2310 Fl. 13,45 US $ 3,50