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NUCLEAR ENERGY AGENCY COMMITTEE
ON REACTOR PHYSICS

REACTOR PHYSICS ACTIVITIES IN OECD COUNTRIES

JUNE 1973 - MAY 1974

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY

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The Organisation for Economic Co-operation and Development (OECD), which was set up under a Convention signed in Paris on 14th December, 1960, provides that the OECD shall promote policies designed:

— *to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;*

— *to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development;*

— *to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.*

The Members of OECD are: Australia, Austria, Belgium, Canada, Denmark, Finland, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Japan, Luxembourg, the Netherlands, New Zealand, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States.

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA) on the adhesion of Japan as a full Member. NEA now groups eighteen European Member countries of OECD, Japan and Australia, with Canada and the United States as Associated countries. The Commission of the European Communities takes part in the work of the Agency.

The objectives of NEA remain substantially those of ENEA, namely the orderly development of the uses of nuclear energy for peaceful purposes. This is achieved by:

— *assessing the future role of nuclear energy as a contributor to economic progress; and encouraging co-operation between governments towards its optimum development;*

— *encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to health and safety, radioactive waste management and nuclear third party liability and insurance;*

— *forecasts of uranium resources, production and demand;*

— *operation of common services and encouragement of co-operation in the field of nuclear energy information;*

— *sponsorship of research and development undertakings jointly organised and operated by OECD countries.*

In these tasks NEA works in close collaboration with the International Atomic Energy Agency, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

REACTOR PHYSICS ACTIVITIES IN OECD COUNTRIES

This document is a compilation of the reports on reactor physics activities during the period June 1973 - May 1974, presented at the Seventeenth Meeting of the Committee (held at CEN Cadarache, France in June 1974).

(Next Austrian progress report will cover the period June 1973 to May 1975. No progress reports were received from Belgium, Iceland, Ireland, The Netherlands and Turkey.)

REACTOR PHYSICS ACTIVITIES IN CANADA

M.F. Duret

Research Reactors

The NRU reactor is in the final stages of rehabilitation. The new calandria vessel has been installed and leak tested. Startup is expected in early August. A program of improvement of the WR1 reactor is in progress. This involves the replacement of steel pressure tubes with Zircalloy-II tubes, and replacement of uranium oxide fuel with uranium carbide fuel. The extra reactivity obtained allows the neutron flux to be raised and reduces the operating cost. The NRX reactor continues to be used for fuel testing and operation during the year has been uneventful.

Power Reactor Program

The four Pickering units have operated well during the year, establishing new records of availability. The average capacity factor of the 4 units was about 85% during the year.

Both NPD and Douglas Point are operating normally. NPD is being used as an experimental reactor to a large extent, Douglas Point continues to supply steam for the Bruce Heavy Water Plant.

The Gentilly reactor has been shut down during the year. Advantage has been taken of this fact to add 6 absorber rods and 9 flux detectors to the core to improve the operating characteristics during refuelling operations at full power.

The construction of Bruce "A", a 3000 MW station, is on schedule, and the first unit is expected to be in service September 1975. There are plans to uprate the thermal power from this station by about 13%. The extra power will be used to provide steam for a heavy water plant. The same site has been chosen for Bruce "B", a second 3000 MW plant, and the Pickering site will be used for a second 2000 MW station.

Experimental Reactor Physics

The experimental reactor physics program is dictated to a large extent by the Canadian power reactor program and thus involves measurements of interest to this program.

Examples of measurements of this sort include

1) In-core detectors
2) Absorber rod design
3) Fuel design

The in-core detectors investigated were cobalt and vanadium which are primarily neutron detectors and platinum which is primarily a γ-ray detector. Because the response of these detectors is sensitive to local neutron and γ-ray fluxes their signals change appreciably during an on power refuelling operation in their vicinity and a series of measurements were made to investigate this behaviour for the different detectors.

The second area of investigation involved a series of measurements to determine the absorption in a variety of materials fabricated into various shapes. By increasing the surface area of a black absorber, the absorption can be increased, reducing the number of elements required to provide a given reactivity depth.

The fuel for CANDU reactors is in the form of short bundles and may require in some cases a fission gas plenum. A program of measurements to determine the flux peaking effects for plenums of different sizes and positions in the bundle will be started soon. In addition, buckling measurements, cell fine structure measurements and void effect measurements using the commercial Bruce reactor fuel will be made when the fuel becomes available. These measurements will complement those obtained using substitution techniques on similar 37 element bundles.

A third aspect concerning fuel design has been investigated experimentally using the substitution technique. Calculations predict that a significant reduction in the reactivity release on loss of coolant can be obtained without a large loss in rated power by replacing the central rods in a bundle with a void. Measurements have confirmed these predictions.

Preliminary construction work is still being done in preparation for two new experimental programs. One involves dynamic measurements with two-phase absorbers, a system initially proposed in Italy. Most of the loop hardware is now ready and the present schedule calls for commissioning in ZED-II this summer to allow the program of measurements to start in September. The second program, which is being done in collaboration with Italy, is to determine temperature dependent bucklings for fuel containing plutonium. The fuel is being fabricated in Italy.

In Canada a 7 channel hot pressurized loop for installation in ZED-II is being built. One leg of the loop has been built and is being tested out of pile. Present plans forsee the measurement program commencing late in 1974.

Nuclear Data Evaluation

A status report on fission product thermal yield data presented at the IAEA Power on Fission Nuclear Data, held in Bologna November 1973, has been issued as an AECL report (AECL-4704).

Analytical Assessment Work

Two major assessment studies have been in progress during the year.

A study concerned with the design of a heavy water moderated boiling light water cooled reactor using plutonium enriched fuel will be completed this summer. The original idea that this design would show capital cost savings has been borne out, although detailed cost estimates are still to be completed. Areas of interest in reactor physics concern the accurate determination of the void coefficient, the reactivity released on complete voiding and the specification of the optimum fuel management scheme. A number of experiments have been done to determine voiding reactivity effects and the consistency of measurements and predictions has improved considerably during the year. The range of uncertainty is now sufficiently small that it will not affect the cost estimates to a large extent.

The second area of assessment work involves the use of thorium in CANDU reactors. This study has been in progress during the past year and a considerable part of the effort has gone into developing the calculation methods to be used. The lattice codes used are LATREP and WIMS and both have been compared with available thorium data to ensure that predictions of the codes are adequate for this assessment study. Development of the WIMS code was done in collaboration with the UK. A fuel management program FUMANCHU has been written to deal with the fuel management problem using thorium. Fuel management with thorium fuel is complicated because of the long half life of Pa^{233} which makes the reactivity and fission cross section sensitive to the flux. Preliminary results from this study are expected this summer.

Reactor Plant Dynamics

A large hybrid analogue digital computer has been ordered and delivery is expected in November 1974. This system will be used primarily for dynamic simulations of future CANDU power plants.

Recent Reactor Physics Activities in Denmark

by

B. Micheelsen and H. Neltrup

In connection with a Scandinavian Benchmark Problem calculations have been performed on UO_2 - light water lattices containing 2, 4 and 8 W% PuO_2 for which measurements and calculations have been published in Nuclear Technology, Vol. 15, August 1972.

k_{eff} was calculated both 0-dimensionally by help of experimental bucklings and in 1-dimensional diffusion theory. Values between 0.97 and 1.005 were found with a general tendency towards increasing k_{eff} values with growing lattice pitch.

Cross sections were generated from the 1968 version of UKNDL. Switching to the 1973 version for $^{239-240-241-242}Pu$ gave a general increase of roughly 2% both in k_{inf} and k_{eff}.

A burn up study for an advanced type of BWR-fuel box containing several enrichments and a number of gadolinium poisoned fuel rods has been completed with the box code CDB paying special attention to the power shape during burn up.

The flat power shape claimed by the designer was confirmed. The study further revealed that extreme care has to be taken in chosing gadolinium concentration since the poisoned pins have a tendency to peak after the poison is burned away.

Pseudo fission product group cross sections calculated from the PSFP programme are now used in the CDB-program. In order to get the right spectrum particularly by large void content, the extra water outside the shroud has to be introduced in the PSFP assymptotic cell.

Work has been done to introduce high order difference equations instead of low order difference methods in D-3 diffusion theory calculations. A one group version is operational and preliminary tests indicate increased speed although very dependend on the type of problem.

A multigroup version, which is under way, will certainly not be able to displace approximate methods such as nodal theory and flux synthesis but might be useful as a slightly faster and more reliable back up programme relative to programmes using low order difference methods.

The development of a fuel management programme optimizing over several fuel cycles or the entire reactor life is taken up.

The multicycle programme is based on an iterative process, starting with a rough zero dimensional linearized model. In the present state the optimization is done by linear programming. The interaction in the fuel combination obtained by this sequence is investigated in more detail during burn-up yielding improved parameters to the linear programme, which performs re-optimization leading hopefully to stable limit values. So far the detailed investigation has taken place only in fully reflected rectangular configurations. Usefull empirical relations between k_{inf}-distribution and power distribution have been found, when proper mixing is made for obtaining flat power shape.

A more systematic approach to the problem of administering burnable poison has been taken up.

The subject has much in common with multidimensional fuel management. By help of a programme developed to that purpose the poison cross section in two regions as function of time necessary to maintain a suitable power distribution during burn up is found.

The poison cross sections found in each region corresponds to a certain quantity of poison to be distributed in the fuel box so as to maintain both the internal power shape as well as the correct time dependence of the absorber cross sections of the box.

By use of concentration dependent selfshielding factors the time dependence is fitted by least square methods to the time dependence found in the overall calculation.

So far this has only been carried out in two dimensions, but three dimensional treatment by use of the nodal flux calculating routine from the three-dimensional dynamics programme ANDYCAP is being prepared.

The three-dimensional BWR-simulator ANDYCAP which describes the reactor core in details, has been implemented on the B 6700 computer and a number of improvements has been carried out in order to speed up the code. A number of transients has been studied.

REACTOR PHYSICS ACTIVITIES AT JRC-EURATOM/ISPRA
June 1973 - May 1974

G. Casini

NUCLEAR DATA

The main part of the work was concentrated on (n, γ) cross-section libraries. Using the nuclear data from ENDF/B-3 and POPOP-4 a (n, γ) library for 27 nuclides (100 neutron and 19 gamma energy groups) has been prepared (SL-E2) together with auxiliary processing codes for computing the cross-sections of mixtures and producing the data in the form adapted for MORSE and DOT.

CODAC, a code for generating group cross-sections from the ENDF/B data library for use in the Ispra Monte Carlo code TIMOC-72, has been updated to read ENDF/B3 and ENDF/B4 data. On request of several users the code has been given together with TIMOC-72 to the NEA computer library for distribution. The new version for IBM-360/175 of the multigroup fast and resonance processing codes GAND-GAF-GAR has been also transmitted to CPL-NEA for distribution.

The preparatory work to set-up an experimental investigation of the iron integral capture cross-section in the energy range 1keV - 100 keV has been almost completed. The starting of the measurements, which will be performed in the RB-2 critical facility of Bologna, in collaboration with CNEN and AGIP-Nucleare, is scheduled by the month of June 1974.

Apart from the construction of the components of the central zone of the facility, including a highly enriched buffer of MTR-plates and a central region with a mixture of $^{235}UO_2$ and graphite microspheres, some points related to the accuracy of the experiment have been investigated, namely :

- the homogeneity of the test medium. A number of homogeneity tests by an air-jet mixer have been carried out for binary (graphite-coated UO_2 and graphite-iron) and ternary (graphite-coated UO_2-Iron) mixtures. In this last case a satisfactory homogeneity was achieved by a two-step procedure, mixing first coated UO_2 and C microspheres then adding iron and increasing the air-jet pressure. Finally the quality coefficient of the mixture was found in all cases less than 2%.

- the effects on the experimental error related to the non-infinite medium conditions of the test-zone. The correction to be applied to the null-reactivity results has been analysed by a two-dimensional approach at CNEN-Bologna (code ROMPO). In the case of test zone without iron, corrections of the order of 1000 pcm were found whereas these values are considerably increased (up to 3.000 pcm) for the case of test mixtures with iron.

These results have been introduced in a revised overall error analysis of the experiment. For the case of $N_C/N_U = 150$ and $N_{Fe}/N_C = 0.6$ an expected error of about 5% in the iron capture rate was evaluated.

REACTOR SHIELDING

The ESIS (European Shielding Information Service) activity has been oriented to the study and development of shielding codes from the point of view of efficiency, ease of operation and accuracy. MORSE has been provided with a number of calculation facilities resulting in a standard version called MORSE-E, ready for use. MORSE-E enables the user to choose among several built-in source geometries ; it calculates volume particle fluxes and reaction rates ; standard deviations are also computed. MORSE has been linked to the S_n-two dimensional code DOT through the code DOMINO. This code transforms the angular DOT-fluxes into normalized probability distributions, furthermore prepares source data for MORSE.
An improved version of SABINE (number 3) has been produced. The input data have been simplified : in particular the coefficients of the polynomial fit of the source distribution are now calculated by the code itself. The temperature dependance of capture

has been taken into account ; gamma-ray fluxes can now be calculated for points which need not to be equidistant. Two new libraries have been included. The report is ready for distribution. At the beginning of 1973 a new NEA-CPL activity called SECU (Service on Experience of Code Utilization) has been initiated. Following the request of the NEA-CPL, ESIS has offered its collaboration for undertaking this study. The following programmes have been selected as standards, because of their wide use and newness :

1-dimensional SN	- ANISN
2-dimensional SN	- DOT-2
Group Monte Carlo	- MORSE, TIMOC
Point Monte Carlo	- UNC-SAM-2
1-dimensional Removal-Diffusion	- SABINE
2-dimensional Removal-Diffusion	- ATTOW
Point Kernel	- MERCURE-3

In order to obtain as many details as possible from the authors and/or users of these programmes, (e.g. present status, improvements, strengths and weaknesses, typical applications etc.) a questionnaire has been disseminated to people concerned in the NEA countries, United States, Canada and IAEA countries. Once filled out and sent back to the NEA-CPL, these questionnaires have been studied by ESIS, the shielding groups of Fontenay-aux-Roses and of the Stuttgart University ; the work has been concluded during the first months of 1974.

An integral check of the gamma production cross-sections of iron has been carried out in the EURACOS facility of the ISPRA-I reactor. The mock-up configuration consisted of a set of layers of iron forming a block put in front of the neutron source. A lead filter was placed between the converter and the iron assembly to remove direct fission gamma production from the source. The neutron spectrum in the epithermal range has been measured by means of the triple foil method (sandwich detectors) In the energy region up to 10 kev He-3 proportional counter measurements were also carried out, in collaboration with the University of Hannover. Gamma dose rates were recovered by thermoluminescout dosimeters. The comparison between He-3 spectrometer and the sandwich neutron results looked satisfactory. The measured gamma dose rates have been compared to the SABINE

results. Along the iron block the slope is practically the same ; a discrepancy in the absolute values is found.
A Shielding Data Bank (S.D.B.) is being set-up to speed-up access to and retrieval of shielding information. The bank is subdivided into various classes, such as bibliography, experiments, cross-sections etc... In a first phase, the data bank handles only bibliographic items delivered by the weekly scanning of periodicals, reports and books. The retrieval system used is deviced by the SIMAS system set-up by COPIC. This structure, together with a thesaurus of key-words makes possible the retrieval of a specific item. The issues of the ESIS-Newsletter issues have proceeded on schedule : four Newsletters have been edited, with specific contributions on the work in progress from :

- Hochtemperatur-Reaktorbau, at Mannheim
- GAAA : Groupement Atomique Alsacienne Atlantique, at Plessis-Robinson
- INTERATOM, at Bensberg
- RCN, at Petten
- Japanese Shielding Community.

BLANKET PROBLEMS IN FUSION REACTORS

A modular code system has been set up to calculate the main nuclear parameters in blanket, shielding and magnet. Fig. 1 shows a layout of the method as well as of the nuclear input data and outputs. The neutron and gamma transport calculations are performed in one-dimensional geometry, in 100, 21 energy groups respectively. P-3 approximation is taken for the scattering anisotropy. The neutron energy group structure is that of GAM-II. As far as nuclear data libraries are concerned, the effort is oriented towards a reference to ENDF/B-3. However, it has so far been necessary in some cases to take other nuclear data sources, namely :

- (n, γ) production cross sections from POPOP-4 for the isotopes which are not included in ENDF/B-3 ;
- the neutron KERMA-factors (apart from boron and lead) are still from AVKER, due to the fact that the code MACK has only very recently been made available at Ispra ;

- the atom displacement production rates for stainless steel have been calculated from DORAN compilation, whereas for copper the UK-NDL Library was taken as the nuclear data source of ARTUS-X. The possibility of adapting RICE (based on the ENDF/B-1 file) to ENDF/B-3 is being considered at Ispra.

This calculation scheme is quite expensive in computing time. A preliminary effort is planned to optimize the neutron energy group structure both for tritium production and magnet shielding calculations. For shielding parametric studies, the possibility of replacing the S-n approach with more simplified models has been investigated. The SABINE code has been found attractive for this purpose.
The (n, γ) libraries now in the SABINE code are not adapted for CTR-calculations, due to the fact that no inelastic gamma sources are included. Work is in progress to replace this library by one produced by POPOP-4.
The reliability of SABINE for magnet shielding calculations in fusion reactors has been checked by means of a comparison with ANISN. The results of this comparison for a typical magnet shielding arrangement, are shown in Fig. 2. It appears that SABINE overestimates the neutron attenuation up to the magnet boundary : this corresponds to an error in the total thickness of about 10%. For this configuration the computing time of the SABINE calculation is 4.4 minutes on IBM-370/165 (including γ -calculation) which has to be compared to about 1 hour in the ANISN calculation.

REACTOR DYNAMICS
A long term effort has been undertaken to improve the methods of integration and achieve better accuracy and savings in computer time and memory of the COSTANZA-series of codes. The spline method has been tentatively selected for this purpose.

A one dimension programme based on the spline functions approximation has been made. This solves the time dependent diffusion equations, in two energy groups, in slab and cylindrical geometry, for homogeneous and heterogenous boundary conditions. This method gives a much better accuracy than the finite difference method with a lattice of the same number of points. Comparison calculations were made also for cores having inner regions with different diffusion coefficients. The problem of the extension of the method to two dimensions has been tackled, and work on the subject is being pursued in collaboration with the"Université Libre de Bruxellés".

The code MATTEO for subchannel analysis of BWR rod bundles in steady state and transient has been developed in collaboration with GE. The model is based on Zuber-Findley's approach to two phase flow analysis and on an original approach to subchannel mixing that takes into account geometrical effects for the vapour drift among neighbouring subchannels formulated as a diffusion process against a velocity dependent potential. A new model for subcooled boiling is included in the formulation. An analysis of the complete series of GE experiments on nine rod bundle has been carried-out, leading to a better specification of the free parameters in the model's correlations. The code is fully operative and has been released to the CETIS code library for general availability

Some improvements have been introduced in COSTANZA-RZ code for use on BWR problems. These concern a neutron diffusion kinetic calculation scheme, leading to smaller round-off errors, a better fitting for water properties in the range 30 to 100 bar and a better definition of the numerical constants in the two phase flow multiplier correlations.

FUEL CYCLES

A theoretical investigation of correlations among heavy isotopes has been pursued. Fuels irradiated in light water power reactors have been considered ; the dependence of the correlations on several reactor parameters (fuel enrichment moderator-to-fuel volume ratio, cladding material, reactivity control method)has been analyzed ; the study concerns a wide range of possible lattices of pressurized reactors and some specific examples of boiling reactor lattices. Analytical expressions have been provided for the correlations whenever possible ; comparison has been made with experimental data concerning the fuels of the reactors : TRINO VERCELLESE, YANKEE ROWE, DRESDEN 1 HUMBOLDT BA.

The work on correlations involving fission products has been initiated. Some comparisons with experimental data referring to TRINO VERCELLESE reactor fuel have been performed.

The effect on the correlations of errors in the fission yields and in the absorption cross sections of the fission products has been evaluated for a particular fuel type of pressurized water reactor. The precision required for these data, in order to obtain a reasonable accuracy in the calculation of fission product concentrations, has been evaluated.

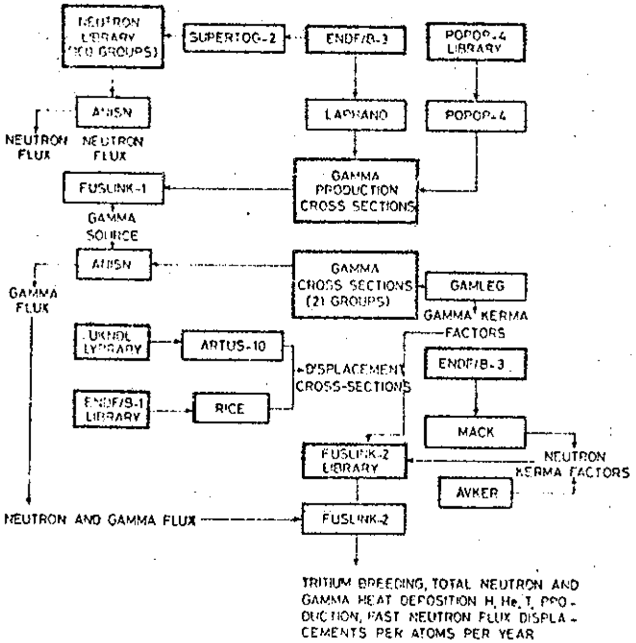

Fig. 2 - CALCULATION SCHEME

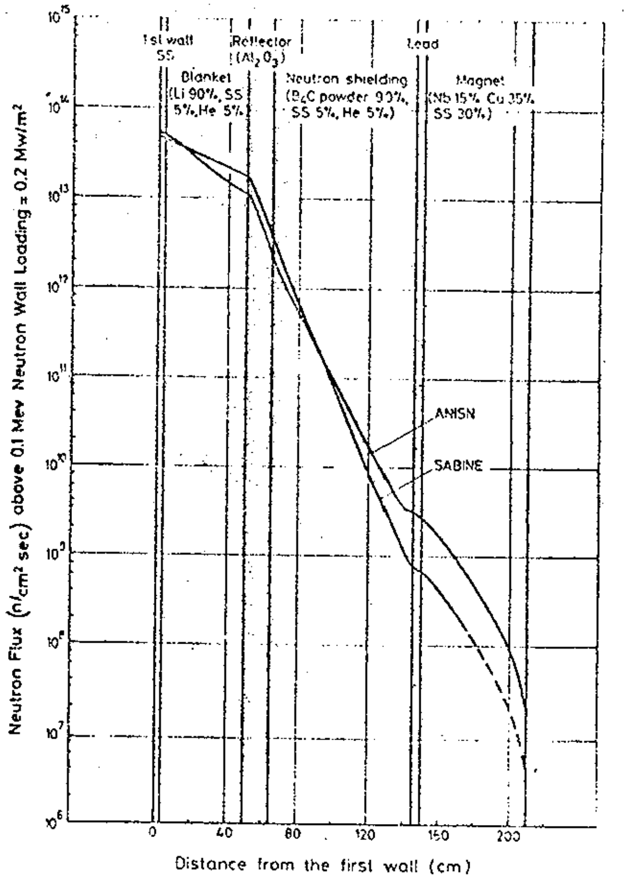

Fig. 2 - FAST NEUTRON FLUX (> 0.1 MEV) COMPARISON ANISN SABINE

REACTOR PHYSICS ACTIVITIES IN FRANCE

JUNE 1973 - MAY 1974

1. GENERAL

From June 1973 up to now, two main events have drawn reactor people attention in FRANCE : first the acceleration of the electricity nuclear programme based on LWR reactors, second the successful results of the fast reactor programme with PHENIX start-up and power rise to nominal power.

Due to the still-increased oil crisis, an acceleration of the 1973 PEON Committee nuclear programme based on LWR reactors has been decided : plans are now to set up 13000 MWe for 1980 and 50000 MWe for 1985 : 13 plants will be ordered in 1974 and 1975. Only two firms will build these nuclear stations : first CGE receiving up to now 2 BWR 6 orders (995 MWe - operational in 1980), second FRAMATOME in charge of the construction of all the other PWR plants. The first PWR power plant SENA (250 MWe, in cooperation with BELGIUM) is now running normally on the 4th cycle.

The second one TIHANGE (870 MWe, in cooperation with BELGIUM) is expected to supply power at the end of this year. First 900 MWe PWR plants (FESSENHEIM 1 and 2, BUGEY 2) are expected to be operational in 1976.

Only five years after the beginning of the construction and with only a few month delay on the initial schedule, PHENIX went critical on August 31th 1973. The plant was hooked up to the grid on December 13th 1973 and reached full power on March 12th 1974 without any serious problem. Although the final check will only be the one billion KWH stage, the PHENIX really encouraging first results lead to propose to countries interested in fast reactors a 450 MWe plant derived from PHENIX.

Now the French fast breeder programme is focused on SUPER-PHENIX plant (3000 MWth) extrapolated from PHENIX design, to be located at CREYS-MELVILLE on the RHONE river near LYON. The final decision to build this plant would have to be made at the end of this year, construction could start next year.

The RAPSODIE-FORTISSIMO irradiation facility continued to give during this period full satisfactory results : for the first time a 100 % load factor during one run (six weeks) has been achieved, 60 % experimental subassemblies are loaded in the core, 1 billion KWh equivalent have been supplied.

2. LIGHT WATER REACTOR PHYSICS

Experimental work in this field has been carried on in 1973-1974, mainly following three directions :

2.1/ HETEROGENEITY EFFECTS

Some experiments has been realized in AZUR, the light water critical facility located at the CADARACHE Centre. They allowed us to check the validity of calculation methods presently used for the evaluation of power distributions and conversion factors in assemblies including strong heterogeneities like water holes, control rods, burnable poisons,...
A complementary programme will be achieved in MINERVE at FONTENAY-AUX-ROSES (MELODIE experiments) with a fuel typical of PWR 17 x 17 type assemblies. This programme will include a study of gadolinium as a burnable poison.

2.2/ BURN UP STUDIES

Mass spectrometry measurements and physical studies of burnt fuels have been intensified in order to test quickly the calculations of changes in isotopic compositions for PWR type reactors. These measurements are carried on fuels irradiated up to 35 000 MWd/T and coming from several reactors, mainly from the SENA PWR.

Oscillations of irradiated fuels are also being made with samples taken in three assemblies of the SENA reactor unloaded respectively in 1971, 1972 and 1973 with burn-up going from 5000 up to about 25 000 MWd/T. These measurements, achieved in SENA type lattices, simulated with fuel rods belonging to the Nuclear Energy Centre of MOL (BELGIUM) and nor-

/mally

used for the VENUS experiments, allowed us to check the calculation of reactivity changes due to the fission products.

2.3/ PLUTONIUM RECYCLE

A first set of experiments, achieved in AZUR in the beginning of 1974 concerned small blocks of mixed oxide UO2-PuO2 loaded in assemblies with UO2 fuels. Power distributions in uranium and plutonium have been measured particularly in the close vicinity of the boundary.
A more complete study, including reloading configurations with whole plutonium assemblies will be undertaken in MINERVE at the end of 1974.

2.4/ THEORETICAL ANALYSIS

Integral neutron measurements may - since we now have very accurate codes as APOLLO /15/ - provide significant information on nuclear constants. Their analysis performed in this mind is the logical outcome of important experimental research done on reactors /16/.

A systematic analysis of B^2 measurements in various types of thermal uranium lattices allow us to obtain a few simple conclusions :

- use the new evaluated fission cross-section of ^{235}U (585 b at 2200 m/s) ;
- decrease (0.7 b from the value calculated with the UK library) the effective resonance integral of ^{238}U;
- use the directly measured age values of H_2O, D_2O and graphite. Then a very good agreement between calculation and experiment is observed /17/.

The analysis now in progress for B^2 measurements in plutonium lattices and isotopic composition of irradiated fuel elements in water reactor seems to prove the P.RIBON's evaluation of plutonium to be quite satisfactory.

3 . HTR THERMAL REACTORS

Since 1970, CEA has undertaken a comprehensive programme on HTR fuel lattices. The fuels are of oxide particulate type 3.5 % enriched. Reference studies have been achieved at room temperature conditions in MARIUS critical facility.

In 1973, a large amount of work was devoted to temperature dependence studies dealing with plutonium reaction rates in small samples using the two signals oscillation method. For this goal, a large HTR reference zone (Ø 1.4m, H = 1.4m) was built in CESAR facility which can be heated up to 400°C by hot CO_2 circulation.

In this core, a first set of measurements proved that asymptotic spectrum conditions were achieved allowing to perform buckling measurements. The aim of this analysis, still in progress, is to derive accurate k_{∞} values as a function of temperature and hence to check temperature coefficient calculations.

In a second step, small mixed UO_2-PuO_2 samples (~ 200g of fuel) have been oscillated in the centre of the core as well as calibrating samples made of known quantities of U235 and boron. The measure of global (reactivity) and local (flux depression) signals for each sample enable to derive the values of Pu reaction rates related to U235 and boron ones using the "equivalent sample" theory. These measurements have been made at 20, 200 and 400°C.
In the same manner, we intend now to measure in cold conditions by almost the same method the reactivity perturbation due to some samples irradiated up to 50 000 Mwd/T from a CEA irradiation experiment in DRAGON.

Five samples of this type are available and it is foreseen to get from the analysis a precise information on fission product capture.

All these analysis are based on the fact that the actual heavy metal content of each sample is very well known. To fulfil this assumption, a special effort has been made in this field of chemical and isotopic analysis. Accuracies better than 1 % are currently achieved for irradiated fuels. These methods have been developped in SACLAY.

Since a few months, CEA's interest has shifted towards thorium cycle fuel. At the moment, a new experimental programme is foreseen concerning critical experiments in GAI type lattices.

4 . FAST REACTOR PHYSICS

A comprehensive report on fast reactor physics activities in FRANCE was presented at the TOKYO symposium in october 1973 /8/ . At the LONDON BNES conference in March 1974, the outstanding physics problems raised by PHENIX start-up and power operation and by 1200 MWe design were discussed /9/.

The two major points from June 1973 concern first, PHENIX start-up and power rise to nominal power, second, the 1200 MWe studies.

4.1/ PHENIX

Systematic physics tests have been performed at PHENIX start-up from approach to criticality in August 1973 till full power operation. Furthermore, from standard fuel burn-up and specific sample or pin irradiations, a lot of physics informations concerning breeding gain, reactivity loss or power variation due to burn-up and reaction rate ratios for uranium , plutonium, americium and some separate fission product isotopes will be obtained.
Detailed presentations of these start-up measurements have been made previously /1/; /10/, /11/,/12/, /13/. Only noticeable topics will be mentionned.

a) Due to the excentric position of the detectors and of the auxiliary source, the analysis of the counting rate variation during the approach to criticality remains difficult. A counter central position would certainly be better.

b) The measured critical mass value is inside the error bars of the calculated one :

$$\frac{E-C}{C} = -1.4 \pm 1.5 \text{ \% in Keff } (2\sigma)$$

The sources of errors in the prediction are now clearly identified and the remaining discrepancy is lower than 0.5 % in Keff. The largest error comes from the 9 diluent (Na 40 % v/o - SS 60 % v/o) and 6 control rod hole (Na) predicted reactivity worths that were announced not enough negative : altogether these 15 subassemblies represent about 8 % in Keff.

c) Core and blanket subassembly reactivity worth measurements at several positions confirm the calculations. Main discrepancies with the predictions come from diluent subassembly reactivity worths measured at 5 positions on a radial traverse. The bias factors for these diluent calculated worths were interpolated linearly versus the volumic SS content from full-Na and full SS rod measurements performed in MASURCA : the true curve was not linear.

d) B4C rod worths calibrated from the rod-drop and discontinuous run methods agree with the predictions in the error bar

e) Radial and axial power distributions in the three zone core and blankets were measured either from ^{239}Pu, ^{235}U and depleted uranium foil irradiations (900 foils) or from pin γ scanning (171 pins). The two methods agree and no large discrepancy from calculations has been observed. Furthermore, reaction rate ratios have also been measured in the oxide fuel with special pins. 700 Mn, Au, Na foils were used to measure reaction rate traverses for shielding purposes.

f) Isothermal temperature coefficients measured from 150°C up to 430°C and power coefficients are on the whole correctly calculated. Thus, at the present stage of the analysis, it seems that Doppler effect is well predicted.

g) In spite of difficulties for defining the real inlet temperature due to thermocouple positions, the outlet temperature map, measured for each subassembly with two thermocouples, agrees with the predictions, so the thermal balance. The results allowed to go to full power without exceeding the cladding temperature or linear power limits.

4.2/ CRITICAL FACILITIES

/ MASURCA /

A/ The programme was devoted to control rod interaction studies for 1200 MWe plant /6/ /14/. In order to analyse the elementary phenomena, five rodded configurations were studied either with full Sodium rod or with B4C rod : three configurations with one rod at several radial positions, two configurations with two rods either symmetric (positive interaction) or dissymetric (negative interaction). The uranium core 4B" contained two zones : innerzone R3 E = 15 %, outerzone R2 E = 30 %. Fissile inventory was about 850 kg.

Reactivity worths and power distributions were measured for each rodded configuration. A complete power distribution measurement was performed with an excentric B4C rod to check the validity of a new method based on the heterogeneous theory, allowing to reconstitute the power distribution for each point in the core from only three power traverses.

Due to the PHENIX diluent rod results, a rush programme was performed at the centre of the core to study the rod reactivity worth variations versus the stainless steel volumic content with seven Na-SS compositions.

Finally before going to the planned higher Plutonium isotope programme, reactivity measurements by oscillation technique were performed for structural material and separate fission product isotopes

/ ERMINE /

The programme dealt first with fission product studies second with higher plutonium isotope problems.

The first phase of the fission product programme has been performed with pins irradiated in RAPSODIE FORTISSIMO /7/ : up to a 9 % burn-up.

Oscillations of irradiated pins, fresh pins coming from the same batch, and standard pins have been performed in the uranium core R3. Chemical analyses are now waited to get the final results. The second phase of the FP programme dealing with pure Pu or ^{235}U pins irradiated in FORTISSIMO or with PHENIX pins will take place in 1975.

The higher Plutonium isotope programme uses three Pu mixed oxide fuels with variable isotopic compositions between 8 % and 45 % Pu240.

The first phase dealing with four "$K_{\infty} = 1$" cores using PuA (8 % Pu240) and B (45 % Pu40) has been carried out at ERMINE.

The second phase using PuC (20 % Pu240) in the same lattices will take place this year at ERMINE. The third phase based on the substitution technique will be carried out in 1974 and 1975 in MASURCA.

/ HARMONIE /

The reactor was mainly used during this period for calibrations and standard for PHENIX start up experiments. The future programme concerns shielding experiments.

/ IRRADIATED FUELS /

The results of sample irradiations in FORTISSIMO confirmed the previous conclusions obtained on RAPSODIE for the capture rate of uranium and plutonium isotopes in hard spectra /3/ /7/. The same irradiation experiment is now loaded in PHENIX core 1. Main emphasis was placed this year on FP sample analysis.

At the CADARACHE VAN DE GRAAFF, a TOF path is now beeing installed : the programme of spectrum measurements in subcritical lattices will start next year.

4.3/ THEORETICAL WORK

The main emphasis has been put on PHENIX start-up. So the synthesis of the whole experimental results /2/ on cell parameters, control rod, blankets,... have been performed using the CARNAVAL III system /5/ and the transposition to PHENIX reactor has been carried out. In the same time, the first transposition to SUPER-PHENIX is obtained.

For PHENIX, an operation and fuel management code CAPHE, based on analytical models and perturbation theory /11/ has been made operational to the plant.

From the standard point of view, the experimental parameters of a very simple pure uranium "$K_\infty = 1$" lattice, named SCHERZO, obtained from measurements performed at SNEAK, ZEBRA, ERMINE and HARMONIE has been proposed as an international standard for fast cross section set tests /4/.

A major effort is placed on control rod interaction problems for SUPER-PHENIX not only from the theoretical point of view, synthesis code, or diffusion coefficient in diluant rods for example, but also from the general analysis of the whole experimental results available in MASURCA and SNEAK.

A comprehensive synthesis of the blanket programme dealing with reflector savings, blanket power distribution and Pu production has been obtained /2/.

A general cross section set for fission product separated isotope in the energy mesh of the CARNAVAL III system is now available (188 isotopes) and used for the analysis of integral experiments in ERMINE. This cross section set together with the CEA library of FP parameters (yields, ...) is now used to generate pseudo FP for burn up calculations.

A simplified model to calculate neutron attenuation in Fe-Na media used for shielding have been optimized for the 1D transport ANISN code with 26 group, S4, P1 and large mesh approximations; the results obtained are in good agreement with the results coming from a reference method, with large computing time save. This model is used to analyse the neutron propagation experiments performed on HARMONIE for several Na-Iron volumic compositions /3/. The model will also be used to analyse the shielding PHENIX start-up experiments together with more sophisticated methods : DOT transport 2D or TRIPOLI MONTE CARLO 3D. An adjustment of the cross sections of this model from the discrepancies on neutron propagation experiments is also planned, based on sensitivity studies from generalized perturbation theory.

5. REACTOR PHYSICS FOR SAFETY PROGRAMME

To support the extensive light water reactor and fast reactor safety programmes carried out at CABRI in the frame work of a wide international cooperation, an experimental programme aiming at defining the main neutronics properties of the experiments planned in the power reactor has been performed at the EOLE liquid moderated critical facility.

Two mock-up of the PHEBUS (LWR) and SCARABEE (FBR) loops have been studied. The first mock-up contains at the centre , a PWR pin cluster, hight water moderated, inside several structural tubes. That loop is surrounded by a driver core, light water moderated, fuelled with hightly enriched plates (MTR type). The second mock-up consists with 7 or 19 pins in a central fast core zone surrounded with the same driver core.

The main neutronic parameters measured are : critical size, control rod reactivity profiles, reactivity worths of materials or void effect, power distribution in the core and the loop, temperature coefficients and γ heating.

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Physics Activities in the Federal Republic of Germany

Compiled by
H. Küsters

G E N E R A L

The 1974-budget of the Ministerium für Forschung und Technologie (BMFT) for nuclear energy amounts to a total of about 3496 Million DM /1/.
The support for reactor development is about 269 Million DM, in addition 37 Million DM are given to reactor safety research and general nuclear safety technology.
An interesting review on the manpower in Germany's nuclear industry and research centers was published recently /2/. Some data are reproduced in the following tables.

Manpower in Nuclear Industry (1.4.1973)

Firms	Manpower	Working in : Production	Research and Development	Administratio
Reactor Manufacturers	7575 (37,8%)	4229	2404	942
Component Manufacturers	6570 (32.8)	4688	804	1078
Firms engaged in Fuel Cycle	2021 (10.1%)	1041	523	457
Reactor Utilities	2085 (10.4%)	1629	98	358
Manufacturers of Measuring Instruments	1375 (6.9%)	834	255	286
Isotope Laboratories	402 (2%)	183	141	78
Total	20028	12604	4225	3199

Manpower in Nuclear Research Centers

Center	Academics	Graduate Engineers	Other Personal	Total
Ges. für Kernforschung mbH (GfK), Karlsruhe	717	253	2432	3402
Kernforschungsanlage Jülich GmbH (KFA), Jülich	697	231	2600	3528
Deutsches Elektronen-Synchrotron (DESY), Hamburg	159	94	807	1060
Institut für Plasmaphysik GmbH (IPP), München	202	44	757	1003
Hahn-Meitner-Institut für Kernforschung (KMI), Berlin	147	14	290	451
Ges. für Kernenergieverwertung in Schiffbau und Schiffahrt mbH (GKSS), Hamburg	83	82	370	535
Ges. für Schwerionenforschung mbH (GSI), Darmstadt	56	23	129	208
Total	2061	741	7385	10187

NUCLEAR RESEARCH CENTER KARLSRUHE

1. Analysis of Critical Assemblies

The critical assembly SNEAK-9A was built with the main purpose to study the effectiveness of the controlsystem of SNR 300 and to check the prediction of the power distribution for various control rod insertion patterns. The SNR 300 geometry was approached as closely as possible. However, because of the insufficient Pu supply at SNEAK, only enriched U was used as fuel /3/. The calculational procedure was based on techniques used normally at Karlsruhe. The 3 dimensional power shapes were obtained with the diffusion synthesis code KASY.

The main results can be summarized as follows:

a) Criticality prediction is possible within about 0.5%. If control rods are present a special transport correction has to be applied to get results consistent with those for clean cores. This holds especially when a large part of the rod positions is filled with follower material.

b) Generally power distributions are predicted within 2%. However in axial traverses near partially inserted rods the shift of the power maximum is underestimated by calculations. This leads to local deviations of as much as 5%.

c) Subcriticality measurements by different methods (source jerk and inverse multiplication) agree within 1% for small reactivities ($\Delta k \lesssim 0.01$), for large ones (Δk = 4%) the discrepancy reaches about 8%. Calculations and experiments agree within about 10%.

The two following figures give the core cross section of SNEAK-9A2 and typical deviations of calculated power distributions from measured ones.

SNEAK-9C also follows the SNR program. The main objectives are:

- Starting with a 1 Zone Uranium assembly (SNEAK-9C1) in several steps a mixed (U-Pu) oxide core will be reached (SNEAK-9C2) by sector substitution.

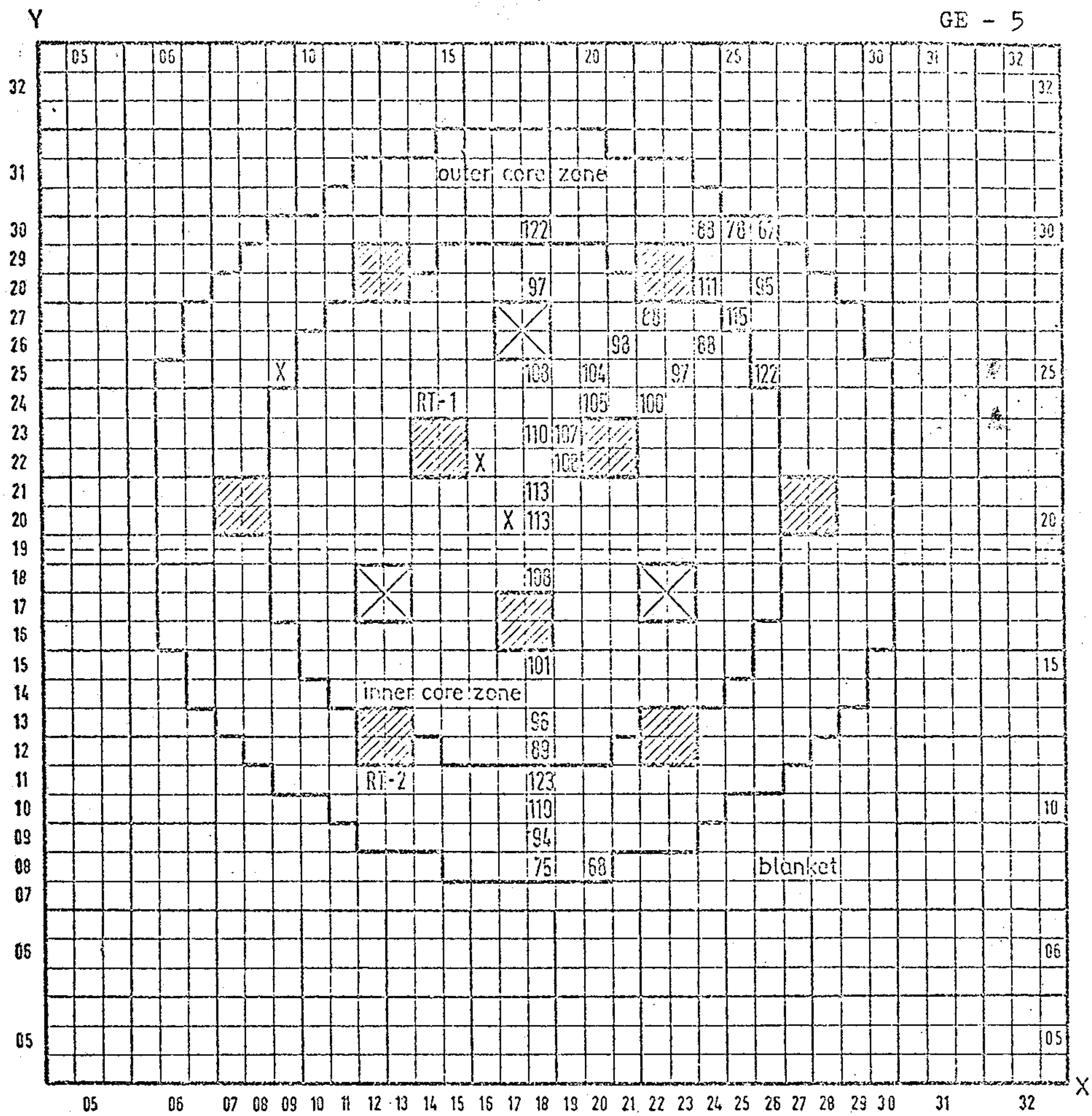
- Measurement of reaction rates of higher Pu-isotopes and of ^{241}Am and ^{242}Am.

- A detailed investigation of the influence of Pu with high ^{240}Pu content on reaction rates, and sodium void effect.

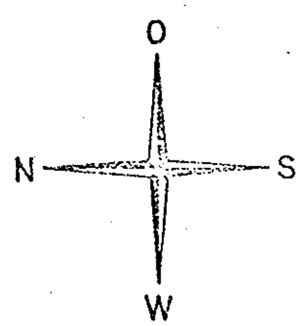
Measurements in SNEAK-9C-1 are under way. Reaction rate ratios are quoted in the following table for the midst of a 20% enriched U-platelet in the core center.

	σ_{f8}/σ_{f5}	σ_{c8}/σ_{f5}
Experiment	0.0466 ± 1.5%	0.128 ± 2%
Calculation (KAPER and KFK INR-Set	0.0429	0.125
C/E	0.92	0.98

An investigation of the heterogeneity effect in sodium void reactivity measurments was presented at Tokyo /4/. The paper demonstrates a reasonable understanding of the void effect on fast critical assemblies. Further investigations on the effect of control rods and ^{240}Pu on the sodium void coefficient are necessary before a reliable extrapolation can be made to



SNEAK - 9A 2 Core Cross Section



Simulated SNR control rod

Simulated SNR safety rod

X Position of axial traverse measured with SNEAK-platelets

Fission chambers (axially integrated fission rate is quoted)

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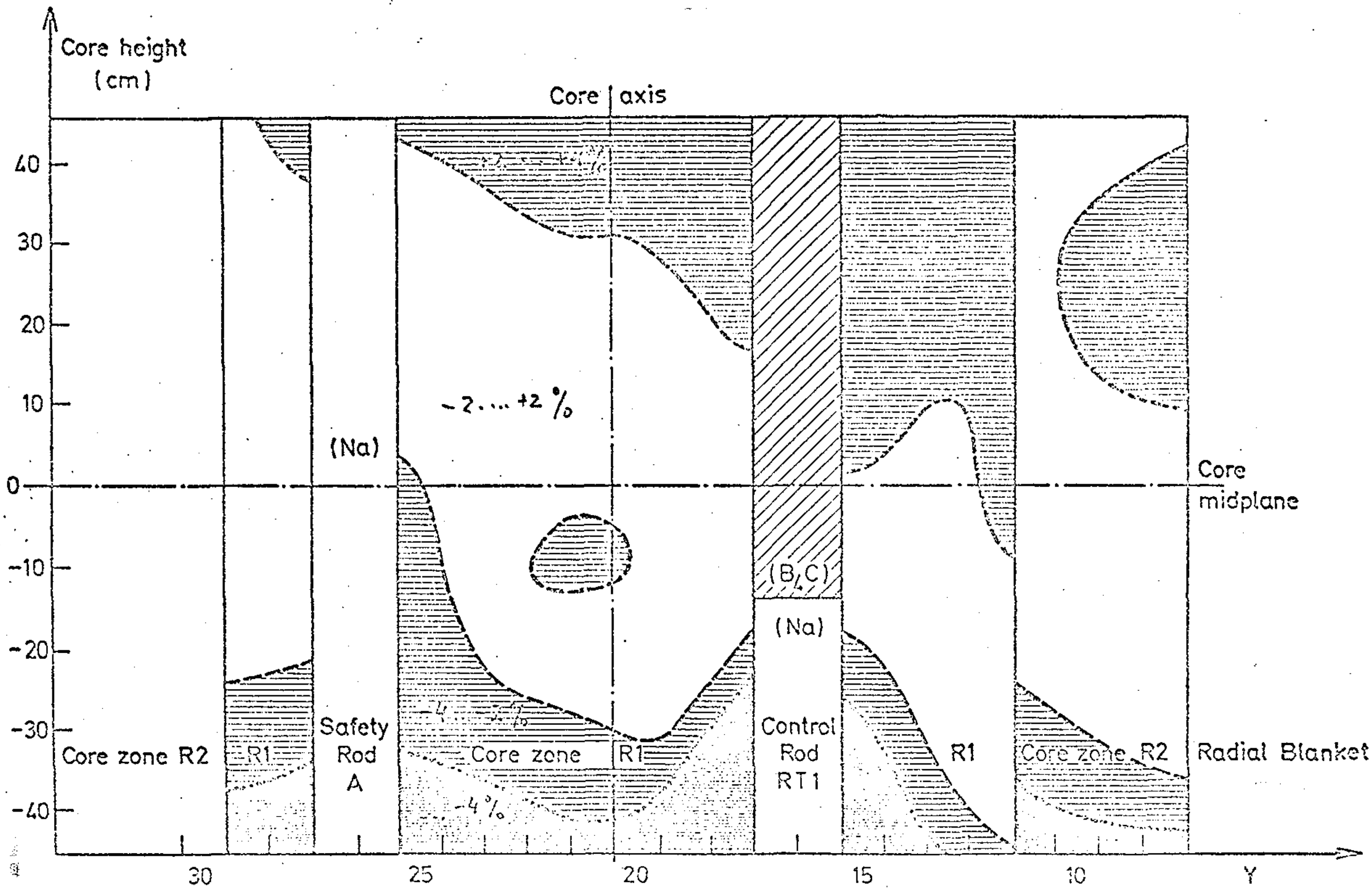


SNEAK - 9A2, Third Core (58-20)

Relative Deviations (C - E) / E (in %) of Power Density : ------ ± 2 % ; -·-·-· ± 4 %

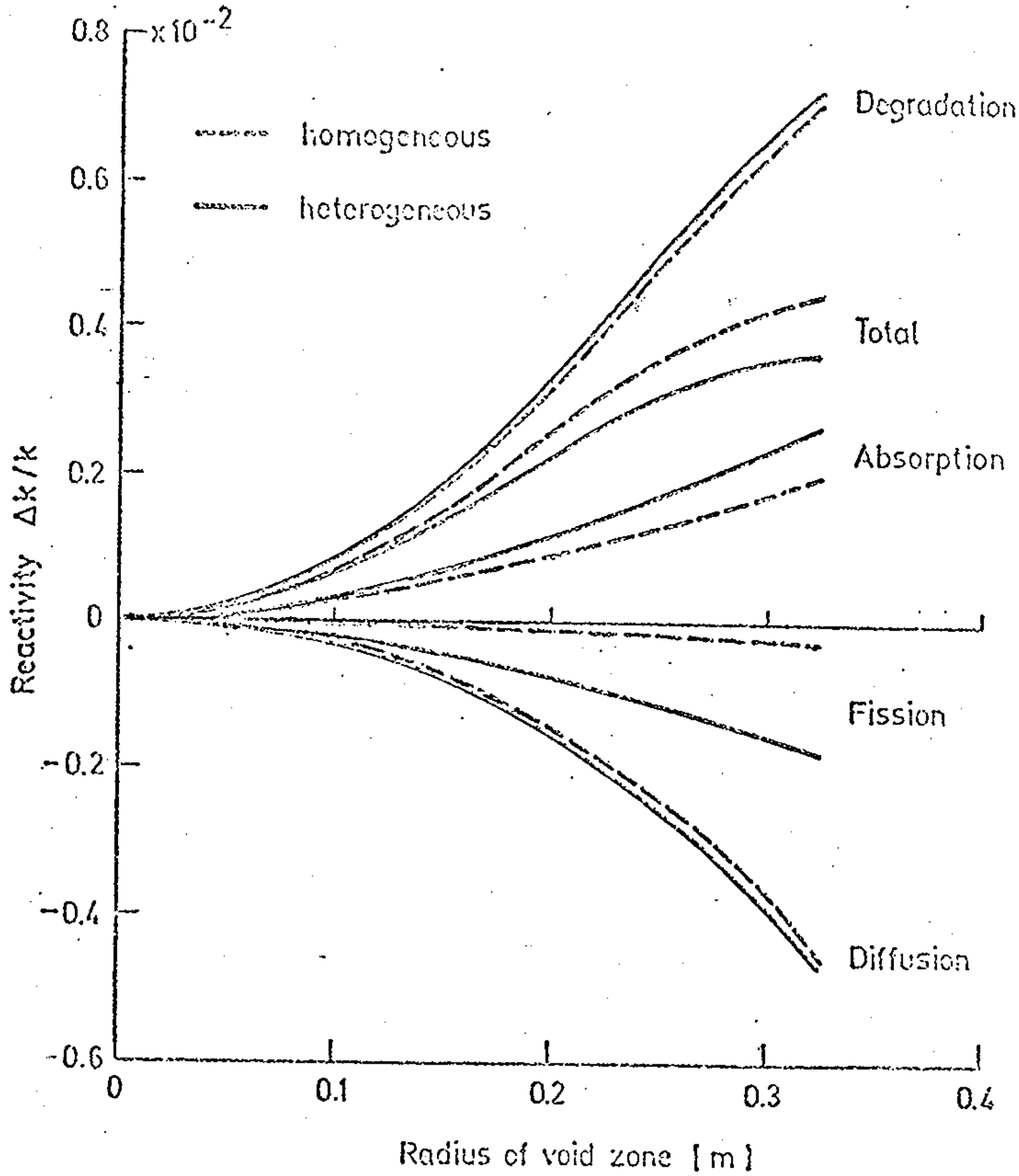
large fast power reactors. The following figure gives the various contributions to the void effect in SNEAK-9-B.

A separate study on the influence of neutron streaming in some Na-void patterns for a simplified 2d geometry of a 300 MWe LMFBR was done /5/. The results are listed in the following table:

Voided regions / Change in criticality	Corresponding to about max. void reactivity	1. Core Zone	1. Core Zone and half of upper blanket	1. Core Zone and both adjacent axial blankets	1. and 2. Core Zones	As before except last row in each core zone plus half of the adjacent axial blankets	Core and axial blankets
$(\Delta k)_{Hom}$ [%]	+13.9	+10.7	+9.1	+6.8	+1.8	+3.7	-3.9
$(\Delta k)_{Het}$ [%]	+13.4	+10.0	+7.9	+5.1	-1.2	+1.9	-8.1
$\Delta k_{Hom} - \Delta k_{Het}$ [%]	0.5	0.7	1.2	1.7	3.0	1.8	4.2

It can be seen that neutron streaming reduces criticality only for relatively large void regions. Since no control and shut down rods are considered, in reality a further reduction will occur, since due to radial streaming a larger fraction of neutrons will penetrate into non voided regions containing absorbing materials.

At the subcritical facility SUAK simple assemblies of iron and uranium blocks as well as a sodium-iron assembly have been investigated /6/ with respect to nuclear data uncertainties to be deduced from stationary and time dependent neutron spectra. Following conclusions can be drawn:



Sodium Void in SNEAK - 9B

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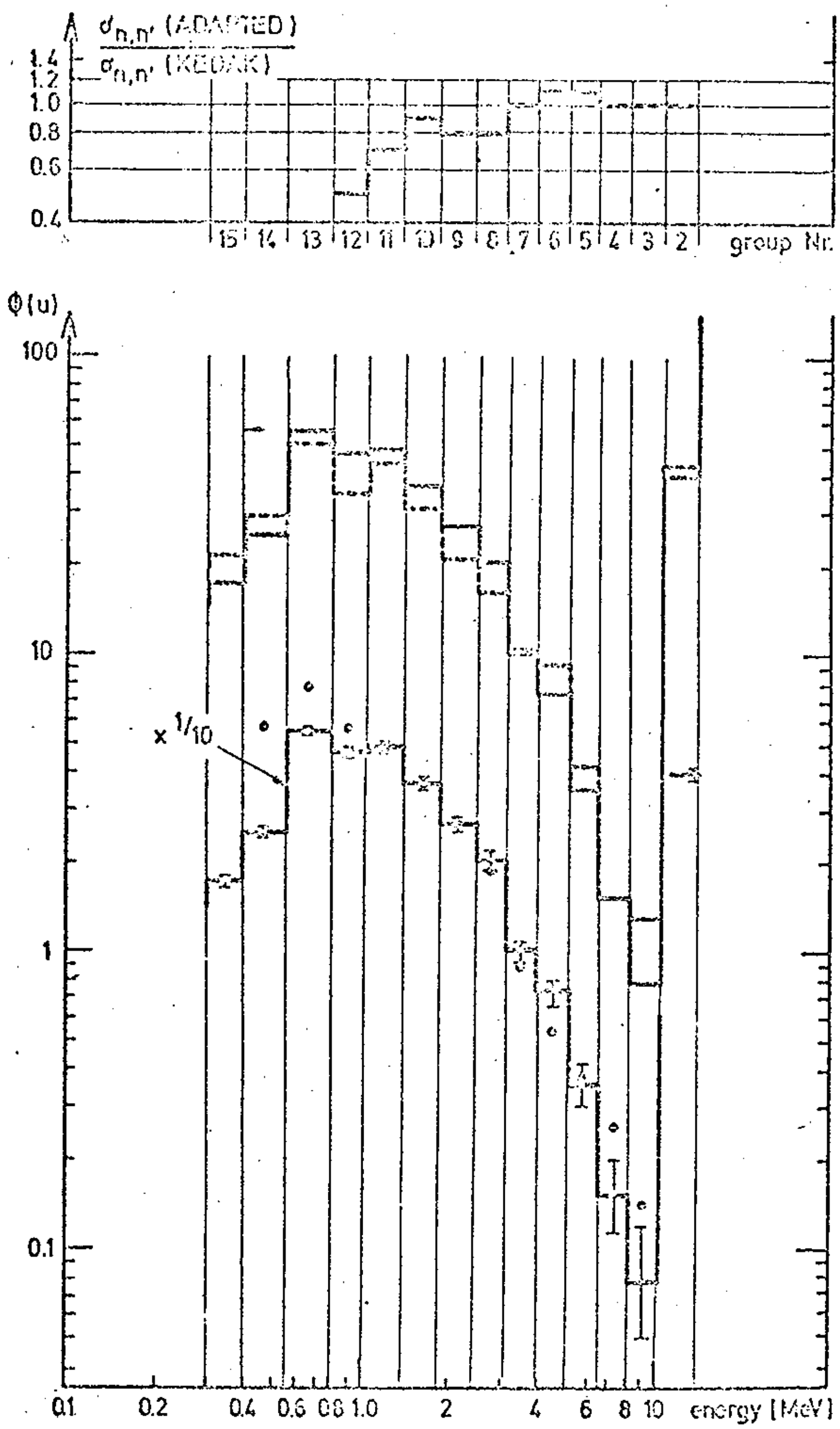
- Time-dependent spectra in the nanosecond decaying time range are sensitive to the inelastic scattering processes. Due to this fact it was possible to define recommended inelastic scattering cross section data for uranium and iron.
- The entire spectrum of small assemblies can be measured accurately by the time-of-flight method with an error margin sufficiently small to make a quantitative analysis of discrepancies between measured and calculated spectra. The interpretation of stationary spectra confirmed the recommended data from the time-dependent investigation, but secondary spectra and anisotropy of scattering neutrons must be treated more accurately to eliminate discrepancies above 4 MeV. The discrepancy at the 28 KeV iron resonance indicated the necessity of revising the corresponding cross section data.
- The combination of time-dependent and stationary spectra of small, single material assemblies, proved to be a sensitive test of calculational methods and cross section data.
- Space-dependent spectra at a sodium-iron interface are adequately calculated by two-dimensional S_N-calculation if detailed space-dependent multigroup cross section data are used.
- The theoretical analysis and the experiments performed indicated that the present method of investigation can be applied to other materials of interest in fast reactors.

The next two figures give typical examples for the effect of inelastic scattering changes on neutron spectra of an iron block.

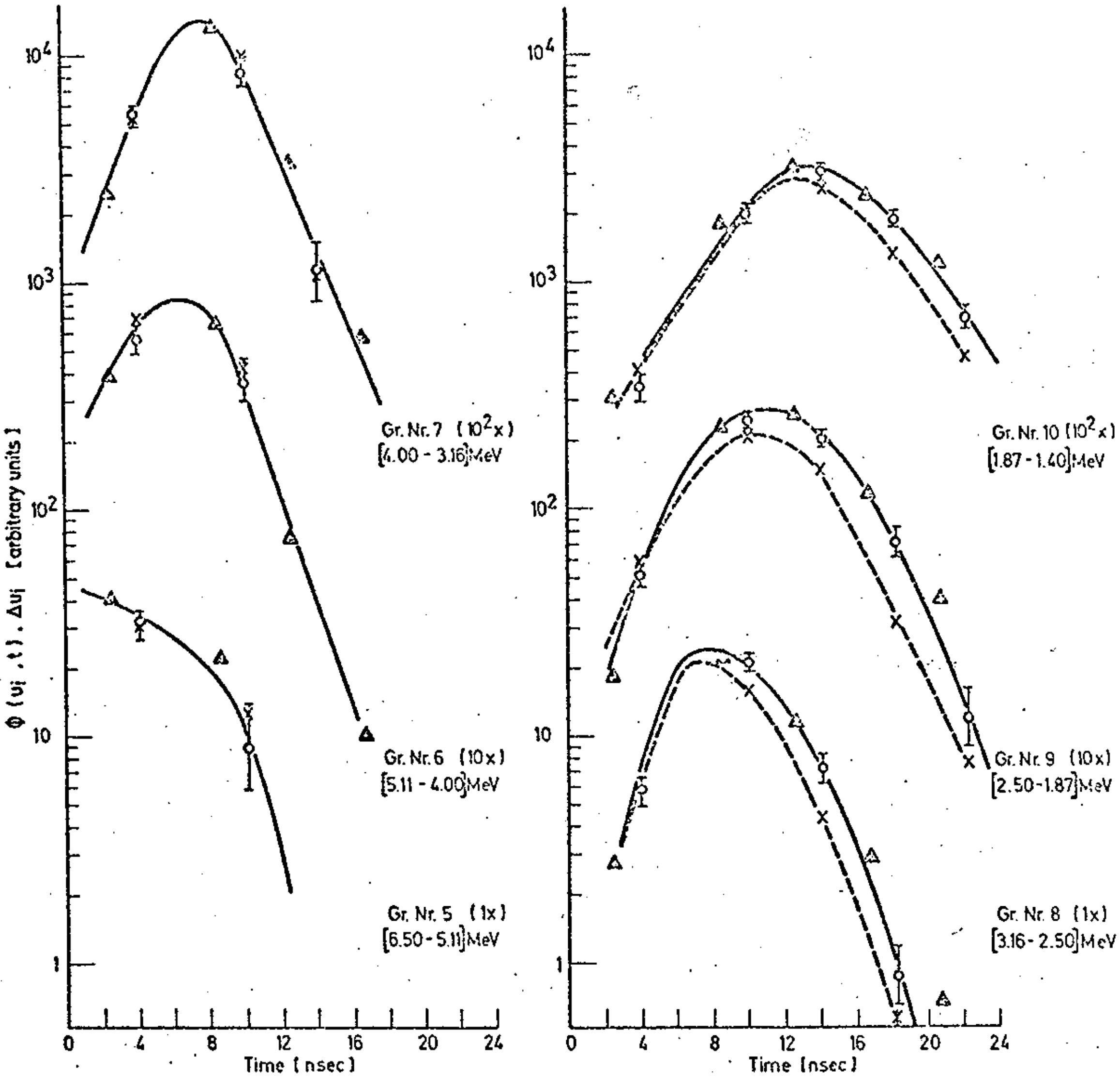
First results for neutron spectra in a <u>sphere of Li</u> have been obtained. Comparison with calculations (P_3 approximation for elastic scattering) show large discrepancies, especially between 6 and 14 MeV. This may be due to direct interaction processes, or to an incorrect representation of the elastic scattering angular distribution by P_3 with an average error of 10% above 5 MeV, which is reduced to 2% in P_5 approximation.



Measured and Calculated Leakage Spectra of the Iron Cylinder ZYLFE 3020

Measured Spectrum : • NE 213 Scintillator in pulse height mode

Monte-Carlo Calculations: { KEDAK - SET ; "ADAPTED" - SET }



Measured and Calculated Time-dependent Group Fluxes of the Iron Cylinder ZYLFE 3020

Measurement : ▲

Monte-Carlo Calculation: { x KEDAK-SET ---- ; o "ADAPTED"-SET ——

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2. Theoretical Investigations

Within the frame of effective cross section generation schemes different approximations for the background cross section, σ_o, in the energy range of resolved resonances for both Pu-239 and U-238 have been studied for a typical LMFBR composition /7/. Following models have been used:

a) averaged total cross section (model A)

b) potential cross section (for U-238 as background only) (model B)

c) effective cross section in the corresponding material composition, calculated by an iterative method (model C)

d) "exact" model within the framework of the narrow-resonance-approximation and the single-level-Breit-Wegner-formula; full energy- and temperature-dependence of the total cross sections of Pu-239 and U-238 are taken into account.

The table below shows the different values for the background cross section, σ_o, for Pu-239 as reference material, when the different approximative models A, B or C are used.

Energy Group	Model A	Model B	Model C
16	20.3	10.6	13.5
17	80.1	"	18.0
18	40.2	"	14.4
19	151.	"	18.0
20	157.	"	17.2

The next table shows the percentage deviations of the effective (n,γ)- and (n,f)-cross-sections of Pu-239 calculated with models A, B or C from those calculated with model D.

		Core regions			Blanket		
Energy group	reaction type	$\frac{A-D}{D}$ %	$\frac{B-D}{D}$ %	$\frac{C-D}{D}$ %	$\frac{A-D}{D}$ %	$\frac{B-D}{D}$ %	$\frac{C-D}{D}$ %
16 (465-215 eV)	(n,γ)	-.3	-2.2	-1.7	-1.6	-2.1	-2.0
	(n,f)	+.5	-.5	-.2	+.3	+.01	+0.1
17 (100-215 eV)	(n,γ)	+12.2	-.9	+1.0	+3.2	-.4	+.2
	(n,f)	+5.2	-2.9	-1.5	-1.0	-3.0	-2.7
18 (46.5-100 eV)	(n,γ)	+16.8	+3.0	+4.6	+11.3	+4.5	+5.2
	(n,f)	+9.0	+1.9	+2.7	+7.4	+3.8	+4.2
19 (21.5-46.5eV)	(n,γ)	+28.6	-6.7	-3.3	+3.5	-7.2	-5.7
	(n,f)	+24.9	-5.5	-2.9	+4.9	-4.7	-3.3
20 (10-21.5 eV)	(n,γ)	+20.1	-7.4	-4.8	+3.0	-7.8	-6.6
	(n,f)	+16.3	-7.3	-5.1	+2.0	-7.7	-6.7

It can be seen that models B and C are preferable to model A, and model C is slightly better than model B.

The corresponding investigation for the effective (n,γ)-cross-sections of U-238 using models A, C and D, show again that model C gives the best approximation for the "exact" model D.

If the different models are applied for the calculation of the temperature derivatives of the effective cross sections of Pu-239, there is an effect of up to 11% in energy groups 16 and 17. The total Doppler coefficient of reactor regions however shows an influence of no more than 1%.

A review article on the analysis of Fast Critical Assemblies has been completed /8/.

The code development was mainly devoted to problems related to fast reactor safety, see e.g. /5/.

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/8/ E. Kiefhaber, Evaluation of Integral Physics Experiments in Fast Zero-Power Facilities, to be published in Advances in Nuclear Science and Technology, Vol.8

INTERATOM, Bensberg

1. Analysis of SNEAK-Experiments (Wehmann)

The analysis of SNEAK experiments is partly done in cooperation GfK-IA. For instance, the reactivity coefficients in the SNR-uranium mock-up SNEAK-9A-2 were calculated in 3d synthesis with KFK-INR-set and the predecessor-group set MOXTOT. The difference to experiment is larger with the INR-set (5 ± 2)% as with the MOXTOT set (3 ± 2%). The worth of the 3 shutdown rods is being underestimated by about 5%, the shadowing effects between shim-rod- and secondary shut-down system is well represented. Changes in the power distribution by about 30%, due to control rod motion, are predicted to 2%, accuracy.

2. Work on Nuclear Data-Sets (Gebhardt, Hübner)

ENDF/B III data are being used to generate group cross sections of the ABBN type. In the unresolved resonance range for some cases irregular dependence of the resonance self shielding factors on temperature was observed. A detailed investigation showed that the use of J-Function tables together with an interpolation routine in the GRISM code yielded inaccurate values for J in extrapolation cases. This was resolved by using instead the J-table and interpolation routine from SUPERTOG III. An improvement over SUPERTOG interpolation accuracy was obtained by the normalisation $J(\xi,\beta)/J(0,\beta)$.

INSTITUT FÜR KERNENERGETIK

University of Stuttgart

1. Interpretation of Pulsed Neutron Experiments in the Critical Assembly ITR (W. Bernnat)

Pulse neutron measurements in the critical assembly ITR show a strong kinetic distortion effect due to the low absorption in the beryllium reflector /1/. We have tried to interprete this effect in terms of the difference between prompt and delayed fundamental modes. For symmetric core configerations this assumption was rather good, so that by means of calculated flux-shapes for prompt and delayed neutrons, which were calculated by two dimensional S_N-Codes, the measured values could be reasonably corrected. For asymmetric core configurations the higher harmonics cannot be neglected so that a simple correction of the measured values is not possible. Here the experimenter /1/ tries to use a detector enclosed in Cd which counts only fast and epithermal neutrons to eliminate the kinetic distortion effect of both the fundamental and higher harmonics. The interpretation of such experiments is continued.

/1/ R. Hecker et al.: Ein Beitrag zur Problematik von Reaktivitätsbestimmungen mittels gepulster Messungen an Systemen mit schwach neutronen absorbierendem Reflektor.
Reactor meeting Berlin, April (1974)

2. Estimation of the Control-Rod Efficiency in the Cavity of a Pebble Bed Reactor (W. Bernnat)

A precise estimation of the control-rod efficiency in the cavity of a pebble bed reactor has to be done by transport theory. Therefore a Monte Carlo method was used to compute this effect /1/. To get a good estimation of the small Δk-value a correlated Monte Carlo method was used, realized in the MORSE-K program /2/. With this program the control-rod efficiency was computed as a function of its position in the cavity and in the upper reflector.

/1/ W. Bernnat, A. Hübner: Monte Carlo Verfahren zur Berechnung der Regelstabwirksamkeit im oberen Hohlraum von Kugelhaufenreaktoren.
Reactor meeting Berlin, April (1974)

/2/ F. A. R. Schmidt, W. Bernnat: MORSE-K Manual.
Report Nr. 4-1 Institut für Kernenergetik Stuttgart (1972)

3. Application of the Finite Element Method to Reactor Problems

(F. A. R. Schmidt)

Theoretical investigations of the Finite Element Method were realized in a program called FEM 2D. This program solves the two dimensional multi-group diffusion equation. It uses triangular finite elements with quadratic flux approximations for the discretisation process. The resulting matrix equations are solved by a modified Cholesky procedure which avoids unnecessary operations on zeros. The number of groups as well as the number of degrees of freedoms per node are arbitrary. Simple techniques for the acceleration of the power iterations are already incorporated into the code.
The code was successfully applied to both PWR and BWR problems. Comparisons were made with the PDQ code (difference method) and the MEDIUM code (nodal method). The results were surprisingly good.
The size of the problems is considered to be a realistic one. In the PWR problem an octant of a reactor containing 103 materials was treated. In the case of the BWR calculation a five by five cell with Gd rod, water hole and absorber cross could be investigated.
Detailed results will be published as GfK-VA and IKE reports.

4. Group Constant Generation

(I. Brestrich, W. Gulden, M. Mattes, R. Rühle)

At the IKE we process ENDF/B data routinely with the programs SUPERTOG and GGC-4. In addition modular programs were developped based on RSYST /1/. They realize various approximations in space and energy by which one may obtain better group constants.
In the resolved and unresolved resonance region (below the threshold energy of the inelastic scattering) point cross sections are produced with ENDRES /2/ for futher processing in RESPU /3/ or RESAB2 /4/.
RESPU solves the slowing down Boltzmann equation by first flight collision probabilities in an absorber zone surrounded by an 1/E - moderator. The absorber zone contains a mixture of maximal three absorbing and two scattering materials. For periodic lattice structures a Dancoff-Ginsburg-correction is allowed. Group constants for this type of problems can be calculated fast and exact with RESPU. For problems with many zones and several isotopes in each zone the program RESAB2 is available. Naturally it needs much more computer time.
We have compared group constants of U-235 in a heterogeneous cell calculated by these methods and GGC-4. The results showed differences up to 100 °/o in the absorption cross section (about 1 °/o in k_{∞}).
In the keV and MeV region the programs ETORSY - STEURES - STEUMAF /5/ are producing 200 to 2000 temperature dependent group constants. Standard weighting functions and a NR-approximation are possible. The NR-weighting allows to consider the interference and overlap effects in the resolved and unresolved resonance region and simple heterogeneous effects. The scattering matrix elements are also weighted with the NR-flux. A computed B1-spectrum is used to collapse to any number of few groups.

Comparisons with GGC-4 results for Pu-239 showed differences up to 90 o/o in the unresolved resonance group cross sections and up to 50 o/o in the scattering matrix.
Benchmark calculations (GODIVA, JEZEBEL) were done with our new data (ENDF/B-III). The results are in very good agreement both for measured keff and reaction rates. More tests will be done in future.

/1/ R. Rühle: EACRP-L-91, p. 41 (June 1972 - May 1973)

/2/ I. Brestrich, B. Riik: ENDRES, ein Programm zur Berechnung von Punktquerschnitten aus ENDF/B-Daten.
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5. A Modular System for Burn-up Calculations with Hierarchical Data Structures (H. Wohland, R. Rühle)

A modular system for burn-up calculations with hierarchical structured data has been developed and partly implemented. With the syntax-printed language of RSYST-II /1/ the user may formulate and document his burn-up problems in hierarchically structured data. The data structures are transformed by modules operating on nodes of problem trees. By this method the expense of formulating burn-up problems is greatly reduced.

/1/ Rühle, R.: RSYST an Integrated Modular System for Reactor and Shielding Calculations, Proc. of the 1973 Conf. Mathematical Models and Computational Techniques for Analysis of Nuclear Systems, CONF - 730414 - P1 (1973)

6. Reactor Safety Related Activities

(H. Unger)

A new model for the thermal interaction of two molten materials at different temperatures (molten UO_2-H_2O or Na) has been developed. Experiments with H_2O-Pb, H_2O-glass and H_2O-Cu are performed presently in order to back up the theoretical model.

The fuel-pin behavior of LWR-cores during and past blowdowns is investigated and simulated by a computer program system SSYST. Emphasis is put at the ballooning phase of the pins, clad rupture and blockage of coolant channels, and their effect on emergency core cooling.
A meltdown code for LWR's is developed. It simulates the behavior of the core past a hypothetical emergency core cooling failure. The code covers heatup- and meltdown of the core structure and surroundings and is presently terminated by meltdown of the lower support plate of the core.

7. Reactor Shielding

(G. Hehn)

New studies of material damage function

The prediction of radiation damage in reactor components like pressure vessel or core grid plate has become one of the important tasks of reactor shielding. Since these reactor components are most essential for safe reactor operation, their radiation damage must be known in the design stage and has to be followed up during operation. To avoid highly conservative design with respect to radiation damage effects, the energy dependence of both neutron fluence and damage function must be taken into account. The neutron fluence is properly calculated by the S_N-programs ANISN and DOT. The uncertainties result from the damage function used. We have calculated cross sections for production of Frenkel defects from ENDF/BIII-datas with the program RICE. In producing group cross sections proper flux weighting is important. For the fast reactor SNR a 1/E-weighting in a 98 multigroup structure similar to GAM II resulted in nearly 30 °/o overestimation of Frenkel defects in the pressure vessel. This means, that a proper treatment of neutron resonances in calculating the material damage function is of great importance in the fitting procedure for instance with the program SAND II.

KRAFTWERK UNION AKTIENGESELLSCHAFT, ERLANGEN *

I. Computational Methods

1. Imbedded flux calculations.

An attempt has been made to perform imbedded two-dimensional fine mesh flux calculations in the framework of a nodal method. First the global coarse mesh problem is solved accurately using the nodal synthesis technique described previously [1]. Inner boundary conditions are then derived from the converged nodal solution that take into account approximately the spatial variation of albedos along the node interfaces. Applying these boundary conditions a conventional finite difference method is used to compute the detailed fine mesh flux distributions inside the nodes. Experience is still limited, although encouraging results have been obtained for a two-dimensional benchmark problem.

2. Xenon transients

A great fund of numerial experience has been gathered in the operating behaviour of large power PWR's in the time scale of hours. The neutron calculations can be done with a coarse mesh model (with up to some 1000 nodes) in 3 dimensions. In the area of feed back one has got to take into account the Doppler effect of the fuel, thermal-hydraulic effects for the moderator temperature and density, and the operating conditions charged for maintenance of criticality (boron concentration, rod position, power etc.)

*Provided by H. Finneman.

With a moderate amount of the nonlinearity from feed-back, uncontrolled xenon oscillations are reproduced correctly even using an ultra coarse mesh. A finer mesh proves necessary only when more detailed spatial information is needed (peaking factors, void influence); here mere interpolation is insufficient. In all feed-back considerations spectral effects must be taken into account; their neglection gives rise to an uncertainty in local and global effects of up to 50 %.

Studying radial or azimuthal xenon oscillations it is sufficient to do two-dimensional calculations with one node per coolant channel and axially constant power distribution for the moderator density and temperature evaluation. This does not hold for fuel temperature.

The results of the linearized theory are not always applicable. Even under normal operating conditions higher modes can be excited to a considerable amount. This nonlinearity implies mode coupling even at low amplitudes of 10 %.

Inspite of the neglection of the time derivatives in the neutron and feed-back equations, it is possible to expand the validity region down to some seconds. This can be achieved by a step function for all thermal-hydraulic boundary values and a pre-evaluated exponential solution for the heat conduction equation. This procedure proves helpfull in 3-dimensional reactor calculations for the case of a steam line break. Here the boundary values are calculated separately by a power plant model with neutron kinetics in a one point approximation.

3. Power distribution control

A process and filter model has been developed which describes the correlations between power density distribution and Iodine and Xenon distributions. This model should be the basis of further development for a power distribution control program of large light water reactors.

The model has been verified by using a simple test problem [2].

[1] M.R. Wagner: Synthese von mehrdimensionalen Grob- und Feinmaschenrechnungen. Tagungsbericht der Reaktortagung des DAtF/KTG, Karlsruhe, 10. - 13. April 1974

[2] H. Finnemann, H. Moldaschl: Steuerung Xenon-induzierter Flußtransienten unter Verwendung von Methoden der nichtlinearen Filtertheorie. Tagungsbericht der Reaktortagung des DAtF/KTG, Berlin, 2. - 5. April 1973

II. Plutonium Recycling

In 1973, research into three fields of plutonium recycling demonstration was made by KWU

1. Insertion of 8 all-Pu-assemblies in addition to the first Pu-prototype-assembly, which was inserted in October 1972.

2. Special measurements to compare control rod worth in the Pu prototype-assembly, burnt over one cycle, with a homologue U-assembly.

3. Measurement-program at the KRITZ facility at Studsvik

1. Plutonium recycling demonstration at the Obrigheim PWR(KWO

A first Plutonium bearing assembly was inserted in Oktober 1972. Larger-scale demonstration was started last September with the insertion of 1440 PuO_2-UO_2 fuel rods in eight reload fuel assemblies. This represents the beginning of the first normal sequence of recycle and it promises, finally, to provide the demonstration of full recycling.

All these fuel assemblies are built thus:

64 fuel rods of a Pu_{fiss}-enrichment of 2.0 $^w/o$ and
116 fuel rods of a Pu_{fiss}-enrichment of 3.2 $^w/o$

The diluent is natural uranium.

During the reload period of 1973 two fuel rods with different inital enrichments were drawn out of the prototype-Pu-assembly and substituted by dummy rods. Post irradiation examination is planned for 1974.

2. Measurements at KWO after reload of September 1973

The Pu-prototype-assembly was shuffled, for the second period of exposure, to a control rod position. Thus it was possible to measure the control rod worth as a function of moderator temperature. This single rod worth was compared with the worth of a homologue rod in a once burnt U-assembly. The experiments verify the theoretical results which indicate that there is nearly no loss of rod worth by once burnt-Pu-assemblies in comparison to U-assemblies at operation moderator temperature. The measurements in addition show a higher rod worth at reduced temperature in the Pu-assembly than in the U-assembly.

3. KRITZ-experiments

A series of KRITZ measurements were performed in summer 1973 at Studsvik as a joint KWU - C-E program. Power distribution, boron worth and control rod worth measurements were performed at a series of moderator temperatures in lattices containing central mixed-oxide and UO_2-regions surrounded by 3.1 $^w/o$ UO_2 fuel. The central region contained, in the Pu-cases, a zoned mixed-oxide KWO-Type assembly having enrichments of

3.2 and 2.0 $^{w}/o$ fissile plutonium. Both the mixed-oxide and uranium fuel rods were part of the 1973 Obrigheim reload, mentioned earlier. These experiments provide, combined with calculations, a unique basis for demonstrating the level of agreement which can now be attained in reactor design for plutonium recycling.

III. Measurements in Operating Power Reactors

In order to determine the reactivity difference between full power and zero power a series of measurements were performed last year at the Stade PWR (KKS). The evaluation of the measurements took into account transient Xenon and the decrease of the average coolant temperature by going from full to zero power. The result is a "corrected" integral power reactivity which contains Doppler reactivity and the reactivity effect appertaining to flux redistribution.

The measurements confirmed the assumptions made for the shutdown reactivity balance. The results will be published later.

PHYSICS ACTIVITIES AT KWU 1973 (BWR WORK)*

1. Test of 3-dimensional reactor simulator

Calculation of BWR reactor cores is being done by 3-dimensional models. During the last years, a considerable amount of experimental material has been accumulated - gamma scan, evaluation of incore readings, isotopic analysis of spent fuel, - and this material has been compared to the calculational results of the reactor model.

Gamma scanning in the first cycle of Lingen (end of cycle) revealed that the relative power of fuel elements was underestimated at the reactor edge by approx. 0.07 and correspondingly overestimated in the reactor center by 0.02. I.e. the calculated power distribution came out steeper than the measurements. The difference at the hot spot was 5 %.

Incore readings are evaluated during operation of BWR's as a routine procedure. A consistent evaluation of the results for KRB, Lingen, Würgassen and comparison to reactor core calculations resulted in similar conclusions as gamma scanning at end of cycle. Again, the central region of the reactor core was overestimated in power and the standard deviation was 4 %.

Isotopic analysis of some fuel pins of spent fuel of Gundremmingen power station was performed in hot cell examinations. A comparison to corresponding burn-up calculations showed that the agreement was within 10 %, taking into account local flux shape, gross power distribution and burn-up history of the fuel element.

2. Check of lattice calculations by use of critical experiments

Fuel elements for boiling water reactors have been standardized since a couple of years. New developments resulted in modifications of standards in 2 respects:

a) use of Gadolinia containing rods for burn-up compensations of reactivity, mostly for reload elements and to some degree in first core elements as well, and

b) more homogeneous fuel distribution by the use of an 8 x 8 fuel rod array instead of the earlier 7 x 7 array.

* Provided by P. Kilian

A sufficient number of such elements were available for later use in Würgassen and Brunsbüttel power stations, and in the meantime a program of physics experiments was performed in the Großwelzheim critical facility.

The experiments provided the following data:

Reactivity of different arrangements of fuel elements in an experimental reactor core.

Local power distribution in the fuel assembly and the effect of Gadolinia poisoning.

Power ratio between fuel assemblies of different enrichment, i.e. mismatch factor.

Effect of axial variation of Gadolinia content on flux and power distribution.

The experimental mismatch factor confirmed the calculated value of 1.34 very well, within experimental error. Criticality of the arrangements was in the normal range of 0.5 % when the actual configuration with local variations was calculated in detail. Approximations in the calculations by using so-called second homogenized group constants gave reactivity values which were too large by 1.5%. The local power distribution showed standard deviations of 3 %. The fuel elements which were designed for asymmetric lattice were arranged symmetrically which provided a more stringent test of the calculation results because of the purposely high peaking factors. The agreement in power of Gadolinia containing rods was very good.

3. Isotopic analysis of Gadolinia burn-up

Some Gadolinia containing rods have been irradiated in the Kahl power station in some earlier cycles and have been discharged since then, awaiting hot cell analysis. Preliminary results of out-of-pile examinations like gamma scanning have been reported earlier. An isotopic analysis, mainly for the remaining Gadolinia content, is underway presently. This will provide data for local distribution of Gadolinia in the fuel pin after burn-up, with average burn-up as a parameter because of the axial burn-up variation. Integration of the experimental differential values results in an overall check of Gadolinia exposure values.

4. Reload of Gadolinia containing fuel elements

The use of Gadolinia in reload bundles introduces a new parameter in the design of reload charges. The well-known scatter-loading can be modified in some respect to zonal-loading. Zonal-loading itself is suitable only for very small BWR cores as the experience from Kahl power station has shown. The fresh Gadolinia containing fuel assemblies with low reactivity can be used for a zonal arrangement with high reactivity at the core center and low reactivity at the edge. Several strategies were evaluated in a parametric way. The result was that there is some incentive for the modification of the scatter-loading by zonal arrangement with low reactivity at the core edge.

5. Plutonium Recycling

Present activities in Plutonium recycling are in transision from experimental irradiation of single bundles to use of complete Plutonium reload batches. The 1974 reload of Gundremmingen will contain 16 Plutonium assemblies which is 1/5 of the normal reload batch. The elements were designed as all-containing Plutonium elements, mostly by consideration of convenient fabrication as to concentrate the Plutonium in few fuel pins and assemblies. Power flattening was achieved by using three different Plutonium concentrations (0,81, 1,73 and 3,77 %). The average concentration of physical Plutonium is 1.94. The resulting power peaking factor (for zero burn-up and 60 % void content) was 1.19 - well within the design value of 1.26.

Reactivity values were checked. Because the Plutonium elements are inserted in the normal four bundle arrangements, the influence of control rod strength, e.g. is only minor. Shut-down reactivity decreases from 0.136 to 0.134 (full power, normal void content). In the years to come, 24 elements of the same design will be loaded together with the regular uranium reload.

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Reactor Noise Measurements

E. Viehl

Zero power noise measurements have been carried out with the coupled cores of the FMRB aiming at the assessemnt of reactor parameters and reactor power. Informations about the dynamic behavior as well as about component malfunctions of power reactors can be obtained by noise measurements at a higher power level. Since more or less coupled zones can exist in large reactor cores, the FMRB can be regarded as model. Further investigations shall lead to a theoretical description of the power noise at FMRB, induced by the coolant flow and temperature fluctuations of the coolant.

Burn-up Determination by Means of Reactivity Measurements

R. Hollnagel

The ^{235}U burn-up in MTR-fuel elements was determined by reactivity measurements. The reactivity changes were calibrated for nuclear fuel burn-up by observing the reactivity changes produced by cadmium absorbers. The uncertaimity in the burn-up determination turned out to be considerable. To improve this, the reactivity change produced by a known mass of nuclear ^{235}U-fuel should be measured.

Criticality of Nuclear Transports

H.-H. Schweer

The Monte Carlo Program MORSE-K used for evaluating the criticality of nuclear transports was modified. Although the general GEOM package is very versatile the description of three dimensional arrays is expensive. To facilitate the input for arrays of containers loaded with fissile material, cousequences of the periodicity of such arrays are considered. Thereby the discrepancies between calculations and experiments about the influence of materials used in nuclear transport on the reactivity of single and coupled MTR-cores (ATKE, Bd. 21/1973) should be reduced.

KERNFORSCHUNGSANLAGE JÜLICH*

THE CRITICAL PEBBLE BED ZERO ENERGY FACILITY KAHTER

An Experimental Assembly to test the Validity of Calculations for High Temperature Reactor Neutron Physics

H. Gerwin, W. Scherer

1. Motivation of Critical Experiments

Due to the fact, that economical and safety questions become more and more important in the field of nuclear reactor concepts it is necessary to ensure a high amount of precalculated information on neutronic behaviour for future projects and designs.
Tedious work has been done in many countries to establish large computer code systems for layout and design of power plants as well as for investigation of the depletion history of nuclear reactors. At KFA Jülich most of these activities concentrate on the pebble bed high-temperature reactor. There are several code systems to cover the large field from rough and global information over long time burn up calculations up to detailed studies on special effects like xenon oscillations ore 3-D-flux deformation by inserting control rods /1,2,3,4/.
Additionally different nuclear data libraries were used showing that there are still discrepancies in crossection-values, which influence the results sometimes stronger than do different methods in the calculational model.
To prove the practicability, the worth and the truth of these calculational models, codes and libraries it is necessary to perform tests using well defined experimental systems. Among these the critical assemblies give maximum information on the neutron physics of an operating multiplying system without involving the troubles of experiments with commercial power plants.
Under these aspects in 1971-1972 KFA Jülich performed a first critical experiment on pebble bed fuel elements in collaboration with CEA France using the critical facility CESAR at CEN Cadarache (France) /5/. Besides the general experimental "know-how" this experiment led to some important modifications of both calculational model and codes, especially concerning resonance absorption. As there are some restrictions in flexibility due to the experimental concept of the CESAR facility but-on the other hand- due to

*Provided by L. Wolff

the new concept of the 'OTTO'-loading scheme /6/ questions arose in connection with the control-rod efficiency of such a reactor it was necessary to built up a new critical facility for pebble bed fuel at KFA Jülich. This facility named KAHTER (Kritische Anlage HochTEmperatur Reaktor) was designed for detailed investigation of critical mass, reaction rates throughout the whole reactor and especially of reactivity effects of control rods and comparison between theory and different experimental methods.

2. Description of the Facility and Experiments done until now

KAHTER consists of a hollow graphite cylinder which forms the containment of the fuel balls /7/. It has an outer diameter of 2.96 m and a height of 3.oo m. The thickness of the radial reflector is 40 cm. The bottom reflector has an outer height of 60 cm, it falls by 10 ° to a central hole of 40 cm diameter for disloading the fuel balls. No top reflector is used for the experiments done until now.

The facility has 9 absorber rods for control and experimental equipment. One is placed in the centre, guided by an Al-tube of 10 cm outer diameter. The others operate in bare holes inside the radial reflector (r ≈ 1.15 m). The rods have an active length of 2 m and are filled with B_4C-rings of 5.6 cm outer diameter and 1 cm thickness (central rod: 0.5 cm thickness) contained in stainless steel tubes. Each rod may be moved separately but it is possible to move banks of 4 symmetrical or all 8 reflector rods simultaneously.

All the fuel handling may be done automatically by a special loading equipment.

The KAHTER facility allows a lot of various loadings. By variing the ratio of fuel balls to dummy graphite balls the moderation ratio may be altered. An inhomogeneous loading is possible and a top reflector may be placed upon the fuel. It is further possible to simulate an upper cavity by inserting Al-boxes between the fuel balls and the top reflector.

The 2nd July 1973, KAHTER became first critical. It was loaded with AVR fuel elements /8/, part of them were used in the AVR-reactor after the experiment. These fuel balls of 6 cm diameter contained 1 g U235 and 5 g Th in the form of coated particles. The first core consisted of a homogeneous mixture of 1 graphite ball and 3 fuel balls ($\varkappa = 0.25$). During the last year two further cores with $\varkappa = 0.5$ (2 graphite balls to 2 fuel balls) and with $\varkappa = 0.75$ (3 graphite balls to 1 fuel ball) were operated.
The experimental program covered the determination of

1. the critical masses of the core
   a. without rods
   b. with the central rod
   c. with the central and 4 reflector rods
   d. with 4 reflector rods (only $\varkappa = 0.5$)
2. the rod reactivities for several combinations of rods using the
   a. inverse-kinetic method
   b. doubling-time method
   c. pulsed source experiments
3. reaction rates of several detector materials

3. Present Results and Conclusions concerning the Calculation Method

The calculation method used is the following:

1. Calculation of cross sections in the resonance region for Th232 and U238 with the Nordheim method using the code ZUT-DGL / 9/.
2. Condensing of cross sections with a multigroup spectral code such as GAM-THERMOS /10/ or MUPO /11/ to few groups for various spectral zones. For the reflector zones MUPO is used because of nearly identical results and significant computer time reduction compared with GAM-THERMOS;

3. For calculating rod constants ANISN /12/ is used with a 40 fine group library made by condensing the GAM-library to 10 groups and adding the 30-group THERMOS-library /13/. Details concerning the rod calculations are given below.

4. With these cross sections and rod constants 2-D or 3-D diffusion calculations with EXTERMINATOR-2 /14/ or CITATION /15/ are performed.

5. The resulting bucklings are used to recycle the spectral calculations. It is assumed that the broad group buckling is representative for all fine groups to be condensed to it. The recycling was stopped, when two subsequent steps differed by a few m nile in k_{eff}, that is a few percent in the bucklings. This recycling procedure results in a change in k_{eff} of about one half nile between bucklings guessed as zero and the converged values. It seems to be nearly negligible for calculating critical masses, but it is very important when calculating differences in k_{eff} such as rod reactivities.

Calculations with this method result in a k_{eff} for a given configuration. By variing the height of the configuration and interpolating, critical masses could be determined. Inserting rods in a configuration gives the reactivity of the rods as

$$\rho = \frac{k_1 - k_o}{k_1 \cdot k_o}$$

in percent. As experimentalists measure ρ in \$ one has to know β_{eff} which may be found from an adjoint calculation.

The calculations are done with 4 broad groups until now. Especially for a more exact determination of β_{eff} an adjoint calculation with more broad groups will be necessary.

Before the experiment was started calculations were done for $\varkappa = 0.25$ and $\varkappa = 0.50$. For $\varkappa = 0.25$ without any rod the critical mass failed by -0.9 % only, but for $\varkappa = 0.50$ the failure was -7.4 %, indicating difficulties in the calculation model. Besides this rod reactivities were overestimated by more then 10 %. A lot of investigations and corrections were done, so that theoretical results for $\mathcal{M} = 0.75$ were not available before criticality. Correcting one point often resulted in a better agreement concerning one item but in a greater failure in another.

The most important points were:

1. Concentration of impurities is not known exactly, especially the contents of nitrogen and hydrogen in graphite is not known. Best results were obtained by using the values known from the CESAR-experiment /5/.

2. First calculations used no streaming corrections in the holes between the balls. Using the well known Behrens-correction /16/ was not very satisfying, too. After a detailed study of the Behrens paper we believe that of the 3 formula given by him, the 'general' one is wrong, one of his special cases, his second one, is the general and his first one is a special case of the second.

   A very important part of the streaming effects is due to the streaming through the guiding tubes for the rods. It caused about 0.6 nile in k_{eff}.

3. A correction had to be made in the multigroup code GAM /10/, σ_{tr} was flux-condensed like other cross sections. Correcting it into a current-condensation altered the first group coefficient by 12 %, resulting in about 0.4 nile in k_{eff}.

4. A difficulty in the calculations for KAHTER results from the fact that the "pot" formed by the graphite bottom and side reflector is not completely filled with balls. So it is not a convex system and calculation with a diffusion code can be made only with certain approximations. First we described the "chimney" by vacuum boundary conditions (not reentrant). This had to give a too low k_{eff}. We corrected it with a "reactivity" of the chimney taken from two dimensional transport calculations /17/. Unfortunately it was not possible to re-

cycle the cross sections for these calculations, so they are not extremely reliable.
Stimulated by the results of preliminary calculations involving the numerical treatment of Al-boxes for the KAHTER-OTTO configuration (see part 4 of this paper), we produced macroscopic cross-sections in the following manner: negligible moderation, absorption small ($\sim\frac{1}{10}$) against absorption + outscatter of the surrounding and a diffusion coefficient of about $\frac{1}{3}\cdot$dimension of the chimney cavity. With this cross section set calculation was convergent and transport calculation results were reproduced.

5. Calculating rod constants (D/λ-values) had to be modified by using a correction concerning the fact that the flux at the rod boundary calculated with a transport code (ANISN) is not the same as with a diffusion code. The formalism of Kushneriuk /18/ has been used. In this way good agreement in k_{eff} for all cores with and without the central rod was found. The agreement between rod reactivities remained bad. A tedious investigation of the pointkinetics theory showed, that the signal from the detector is not equal to the function p(t) in pointkinetics. The most important correcting factor is due to the fact, that the shape function of the system varies when the absorber rod is dropped into the core.

   Especially for use in CITATION, boundary conditions for a rod region are unsuited. The rod region and a part of its surroundings has to be represented by homogenised cross-sections. By doing this one has to remember that continuity in flux means continuity in flux at the boundary of the cells and this means that reaction rate in cell calculation is to be divided by the boundary flux instead of mean flux to give the same reaction rate in diffusion calculation. This should be done in all cell calculations but it is significant only if flux is deviating essentially from mean flux in the cell and cells of different type are adjacent.

With these modifications in codes and model, all KAHTER-configurations were calculated critical within 0.4 nile and, if experimental rod reactivities are modified in the manner described, rod reactivities agree within 5 %.

4. Future Planing

After the experiments with homogeneous loading have been completed it is foreseen to simulate in KAHTER a typical OTTO-core. The main OTTO-loading features are monotonic decreasing of the concentration of fissionable material from top to the bottom, a high neutron flux peak near the top, a relative high flux level in the cavitiy between core and top reflector and therefore the possibility to control the reactor by inserting absorber rods into the cavity. For a correct simulation it is necessary to poison the bottom reflector and the lower part of the core. This is done using boronated graphite balls. This poisoning simulates the increasing contents of fission products towards the bottom of the OTTO-reactor due to burn up. A graphite top reflector is still under construction and special top absorber rods will be available to investigate the cavity effect. The height of the top reflector will be variable and a special construction of Al-boxes will allow to vary the height of the cavity, too. This assembly will permit a detailed investigation of the special OTTO-problems as far as they are related to zero energy neutronics.
Theoretical work is under way concerning loading problems of that part of KAHTER experiments. Additionally control rod design is being optimized by theoretical calculations. As usual during the first part of KAHTER experiments it is foreseen to precalculate each of the scheduled loading schemes for clear comparison between experiment and unmodified theory.
The OTTO campagne will start in summer 1974 and is foreseen to cover about one year.

5. Final Remarks

After introducing the modifications described in part 3 of this paper the theoretical description of the KAHTER-experiments - as far as they have been completed until now - is very satisfying. To answer the question of the signification of KAHTER-results for large pebble bed reactors attention should be drawn to some special features of the critical assembly:

1. The KAHTER core is rather small and therefore leakage is very high (about 35 %) and much more important than in large HTR.
2. The chimney effect is an additional difficulty for the theoretical treatment.
3. For the cold KAHTER-core even small errors in the thermal scattering law have a high influence on theoretical results.
4. Errors in resonance calculation may be of other type than in hot reactors.

Exept for the last point calculations for KAHTER have to be more sophisticated than for large HTR. The modifications mentioned above will mostly be less significant for great power plants. Therefore we come to the conclusion that the modified numerical model and computer code systems, which were used for the KAHTER-calculations are applicable to the theoretical treatment of HT-power reactor neutron physics.

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REACTOR PHYSICS ACTIVITIES IN GREECE

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I. Research Reactor Program

The GRR-1 5 MW swimming-pool type Research Reactor is used, besides the extensive production of radioisotopes, for reactor dynamics and neutron physics experiments [1], (n,γ) spectroscopy, neutron diffraction by magnetic materials [2], radiation damage studies [3,4,5] activation analysis and Biological studies [6].

II. Nuclear Power Reactor Program [7,8]

The whole problem of introducing Nuclear Power into the electrical energy system of Greece, is studied by the Public Power Corporation (P.P.C.). So far investigations were conducted on the availability and utilization of the future nuclear plants, preselection of station sites, determination of the unit size and time table of introduction of various reactor units.

The GAEC, which is the authorized body for the establishment of safety criteria, safety evaluation and the licensing of the power reactors, is working on nuclear legislation as well as on preliminary studies on:

a. Collection and classification of information on computer Codes for Reactor Safety Analysis.

b. Study of computer methods for utility, Reactor Physics Analysis and

c. Tests of Computer Programs for Power Reactor Management and siting evaluation for Nuclear Power Plants.

III. Space and Time dependent Reactor Dynamics

The evaluation of the kinetics and coupling parameters of a reflected coupled-core Nuclear Research Reactor (UTR-100 at Q.M.C. of London University) through correlation and spectral-density functions measurements was completed by a member of our staff [9].

A detailed two-point critical reactor model leading to explicit formulation for the basic static and dynamic parameters of the two coupled subcritical cores has been developed and new well defined parameters have been introduced.

The concept of the density function for the time delays was reviewed and a one-parameter distribution function has been introduced unifying some continuous models studied elsewhere as separated cases. From this analysis it has been found that the basic parameters are little effected by the different limiting cases of the distribution for the same mean time delay $(\bar{\tau})$.

An electronic system has been carefully designed and constructed for these measurements and a special technique has been devised for the increase of the dynamic range of the analysed correlation function by a digital correlator.

The measured optimum values of the prompt neutron decay constant of the coupled-reactor, $b = 46.7 \pm 0.6 \text{ sec}^{-1}$, the zero crossing frequency, $\omega_0 = 756 \pm 28 \text{ sec}^{-1}$ and the coupling reactivity, $\alpha/\beta = 5.10 \pm 0.13$ \$ are in good agreement with estimates obtained by different methods. The quoted value of $\bar{\tau} = 0.55 \pm 0.04$ msec for the mean time delay seems more reasonable in comparison with recently presented values. The new parameters introduced, namely, the prompt neutron decay constant of each core, $\alpha = \beta/\Lambda = 53.5 \pm 0.8 \text{ sec}^{-1}$, the correcting factor relating the two constants, $CS = \alpha/b = 1.146 \pm 0.00$, the higher mode decay constant, $a = 736.2 \pm 13.7 \text{sec}^{-1}$ and the decay constant due to the coupling reactivity, $\alpha/\Lambda = 273.4 \pm 5.6 \text{ sec}^{-1}$ are clearly defined and evaluated.

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SUMMARY OF REACTOR PHYSICS ACTIVITIES IN ITALY IN THE PERIOD JULY 1973 - JUNE 1974

(Ugo Farinelli, Editor) (*)

(*) List of contributing Organizations:

AGIP: Agip Nucleare - Montecuccolino, Bologna
CEC: CNEN, Centro di Calcolo, Bologna
CESNEF: Centro Studi Nucleari Enrico Fermi, Politecnico Milano
CISE: Centro Informazioni Studi Esperienze, Milano
CSNC: CNEN, Centro Studi Nucleari Casaccia, Roma
ENEL: Ente Nazionale Energia Elettrica, Roma
LENA: Laboratorio Energia Nucleare Applicata, Università Pavia
NIRA: Nucleare Italiana Reattori Avanzati, Genova
PMN: Progettazioni Meccaniche Nucleari, Genova
PPU: CNEN, Programma Plutonio, Roma
PRV: CNEN, Programma Reattori Veloci, Bologna

For individual contributors please see the references.

Casaccia, June 1974

SUMMARY OF REACTOR PHYSICS ACTIVITIES IN ITALY IN THE PERIOD JULY 1973 - JUNE 1974

(by Ugo Farinelli)

1. THEORETICAL

1.1 Anisotropic diffusion method (CSNC)

A generalized formulation of the anisotropic diffusion method involves the calculation of point-dependent diffusion coefficients in each direction by means of a differential equation formally identical to a diffusion equation. Two successive runs of a standard diffusion code, with appropriate parameters, thus allow the calculation of fluxes and eigenvalues in this approximation /1/. Comparisons with the results of diffusion and transport calculations for two-dimensional problems show that the anisotropic diffusion method yields results much closer to the transport than to the diffusion solution. An automated code for these calculations will be developed; some improvement is envisaged to deal with non-uniform fission sources.
A code to calculate adjoint fluxes in the previous formulation of the anisotropic diffusion method /2/ has been developed, as a basis for a generalized perturbation code.

1.2 Neutron transport (CEC)

In the framework of the study of heterogeneity effects in plane and spherical geometries a previous research /3/ has been extended to arbitrarily inhomogeneous systems /4/ for giving rigorous foundations to subsequent numerical neutron transport methods.
As an alternative to the Fourier transformation technique, recently adopted even in connection with a multi-group neutron transport scheme /5/, a direct decomposition of the kernel of the neutron transport integral equation was proposed in /6/ for yielding practical solutions for three-dimensional subcritical systems. The extension of such a technique to criticality problems and the improvement of its convergence rate for large critical assemblies through a complex exponential transformation are now in progress.

1.3 Montecarlo methods (CEC)

1.3.1 Numerical experiments have been started for the solution by a Montecarlo method of the adjoint transport equation, for the calculation of fluxes and reaction rates in small regions. Such calculations display variances, some of them "wild" ones, and appropriate techniques for variance reduction are being tested.

1.3.2 The preparation of a thermal library for polisterene has been carried on for use in testing the Montecarlo KIM code with experimental results. To this purpose, a program that calculates the atomic vibration spectrum for high polymers of the helicoidal types has been written.

1.3.3 A Montecarlo code for time-dependent thermalization calculations (MCT) has been written. The code uses a neutron splitting technique at fixed time intervals, that allows following the neutrons for a sufficiently long time.

2. EXPERIMENTAL

2.1 Nuclear Reactor Stochastics and Noise Analysis

2.1.1 Progress in reactor-noise analysis and stochastic neutronics at CSN Casaccia has been summarized, covering the period from 1970 to 1972. Part of the theoretical activity has been realized in cooperation and under the auspices of the United States National Science Foundation, the University of California at Berkeley and the University of Wisconsin /7/ (CSNC).

2.1.2 Important recent investigations by Routti, Szeless and Ruby allowed a vast and systematic study to be performed on the nature of the PMZBB (Pal-Mogilner-Zolotukhin-Bell-Babala) profile. The first part of this work dealt essentially with the calculation of the n-th order derivative of a composite function and, in particular, the probability distribution profile from the PMZBB generating function. An extensive set of numerical results has been achieved in exploring the computational methods /8/ (CSNC).

2.1.3 In search of a new probability distribution function approximating the PMZBB profile, four generating functions were introduced, analyzed and mutually compared. They lead to: (1) the Logarithmic distribution, (2) the Radical distribution, (3) the Poisson-Logarithmic distribution and (4) the Poisson-Radical distribution. Distribution (3) is also known as Pascal's or Polya's or Negative Binomial distribution. Distributions (2) and (4) are an absolute novelty in the field of probability theory and applications. Distributions (3) and (4) belong to the class of "generalized Poisson distributions" /9/.

2.1.4 The Hinz and Gurland indicators are highly effective parameters for describing the nature of a distribution under analysis: a graphical procedure based on these parameters helps choosing the best-fitting distribution among those known in the statistical literature. An extensive set of PMZBB (Pal-Mogilner-Zolotukhin-Bell-Babala) distributions has been analyzed in correlation conditions which range from very small to very large. The latter situation is strongly outside the Negative Binomial distribution approximation but features interesting regularities. They could mean that the PMZBB distribution might be approximated by a simpler and more compact probability profile /10/ (CSNC).

2.1.5 A Specialist Meeting On Reactor Noise (SMORN-1) sponsored by EACRP is organized by NEA in cooperation with CNEN and hosted by Centro Studi Nucleari della Casaccia, the coming October 21-25, 1974. The total number of participants envisaged is about sixty of which ten from non-OECD countries (CNSC).

2.1.6 Stability problems in stochastic terms and the meaning of ergodicity have been studied. An introduction to the theory of dynamic systems is first given since for increasing complexity of models a preliminary analysis of stability becomes mandatory. The Lyapunov direct method has been used for studying nuclear reactor stability problems. The temperature feedback on power was analyzed. Finally, the Lyapunov approach and its mechanical analog has been discussed in detail and interpreted /11/ (CSNC).

2.1.7 A direct comparison has been tried among profiles of Poisson-Logarithmic, PMZBB and Poisson-Radical distributions. A likelihood test was applied upon a wide ensemble of PMZBB distributions whose correlation conditions ranged from very small to very large.
There was evidence that the PMZBB distribution may well be approximated, in not very stringent situations, by Poisson-Logaritmic and Poisson-Radical profiles. The structure of the PMZBB distribution as "generalized Poisson distribution" has then been studied: the Poisson-Logaritmic and Poisson-Radical profiles belong indeed to that class, introduced in the field of applied statistics by Gurland, some years ago /12/ (CSNC).

2.1.8 Finally, a new family of generalized Poisson distributions is presented: the Poisson-Algebraic distribution. The Poisson-Radical distribution, previously introduced, belongs to this family. This initiative offers a further step for seeking compact generating functions that might approximate the generating function of the PMZBB distribution /13/ (CSNC).

2.1.9 The activity on reactor noise at CNEN-Bologna, oriented towards the problems of correlating counts in separated time intervals has been continued. The backward Kolmogorov equation formalism in which the delayed neutrons are explicitly taken into account, was applied for the determination of the variance of the number of counts in a single time interval and for the covariance in two non intersecting intervals /14, 15/. Extension to non stationary cases is in progress (CEC).

2.2 Magnetic Scattering in Reactor (CSNC)
The investigation on reactivity effects caused by magnetic scattering of thermal neutrons by magnetized iron has been refined both calculationally and experimentally /15, 17, 18/. THERMOS and CITATION codes, respectively used for cell and reactor calculations, gave a negative effect of about $-2\times10^{-4}\ \delta K/K$. By reactivity measurements performed in reactor with two different iron yokes, negative effects of about $-1.3\times10^{-4}\ \delta K/K$ and $-0.3\times10^{-4}\ \delta K/K$

were observed. It is believed that the difference between these two values may be ascribed to different states of magnetization of the iron yokes. Magnetic measurements are under way with the purpose of interpreting observed reactivity effects.

2.3 Fast Neutron Spectrometry and Reactor Dosimetry

2.3.1 A preliminary measurement of the slowing down neutron spectrum in light water was carried out by means of the Bennett-type counters with γ-discrimination in the 20-130 KeV range. A second series of proton recoil counters, with higher gas pressure, is being calibrated with thermal neutrons of the RB1 reaction in Bologna. The experimental data are unfolded with the FERDOR code whose response functions are obtained with the Snidow-Warren method /19/. In addition a Montecarlo code is being prepared for the simulation of the cylindrical counters (CEC).

2.3.2 In the frame of a systematic investigation of activation detector cross sections, a first set of measurements has been completed on the TAPIRO reactor. Results have been presented at the Specialists' Meeting on Shielding Experiments /20/ (CSNC).

2.3.3 The problem of the required accuracies for detector cross sections in relation to the general objectives of reactor dosimetry has been investigated. The results have been presented at the IAEA Consultants' Meeting on Detector Cross Sections /21/ (CSNC).

2.3.4 Neutron spectra above 60 KeV have been analyzed at the LENA Laboratory of Pavia University by means of a collimated proton recoil spectrometer; the proton is produced in a hydrogeneous radiator, and electrons and other particles are discriminated by means of a thin semiconductor detector yielding a dE/dx signal; the energy of the proton (in coincidence with an appropriate signal from the thin detector) is measured in a "thick" semiconductor (LENA).

2.3.5 At CESNEF, Milano, a device to generate a fast-reactor type neutron reference spectrum is being built. It is based on the use of boron carbide filters installed at the periphery of the homogeneous core of the CESNEF reactor. A high density ($\sim$2.4 g/cm^3) boron carbide capsule can be obtained by fusion at a reasonable cost ($\sim$2000 \$). It is thus possible to build in an experimental channel an irradiation facility with fluxes higher than 10^{10} n/cm^{2}s. The capsule is under construction; the spectrum in the facility has been predicted by means of calculations (in cooperation with Dierckx at Euratom Ispra) using GGC, ANISN and DOT. Measuremnets should start in September (CESNEF).

2.3.6 Cross section libraries for the SAND-2 code are being updated and extended. Codes to generate on cards or tapes the 620 group cross sections starting from the most recent data from CCDN and CNEN Bologna are being implemented (CESNEF).

2.4 <u>Analysis of Fissile Materials</u> (CESNEF)
Nuclear techniques for the analysis of mixtures of fissile materials are being studied at CESNEF, Milano. Methods of analysis of U-Th mixtures and enriched uranium have been developed and tested. The techniques employed include gamma spectrometry with Ge-Li detectors and delayed neutron counting. Methods to analyze plutonium isotopic composition in spent fuel are being developed. They are summarized in Table 1.

TABLE 1

Methods for isotopic analysis of Pu in spent fuel investigated at CESNEF, Milano

Technique	Response	Status
Gamma counting by Ge-Li	All Pu isotopes except 242	Operational
Coincidence counting of neutrons from spontaneous fission	Mixtures of 238, 240 and 242	Being implemented
Counting (by long counter) of delayed neutrons from thermal or epi-cadmium fission	239+241 (thermal) 238+240+242 (epicadmium)	Being implemented
Long counter determination of neutrons from (alpha,n) reaction on oxygen (for PuO_2 fuel)	Mainly from 238	In project
Coincidence counting of prompt neutrons from resonance fission in various isotopes (using crystal monocromator)	239, 240 and 241 separately	In project

3. LIGHT WATER REACTORS

3.1 Analysis of Power Distribution Measurements in the TRINO Reactor at the End of the Third Cycle (°)

3.1.1 Cross gamma scans have been performed on forty-two Trino fuel assemblies during the shutdown at the end of the third cycle with an aim at evaluating the end-of-cycle (EOC) radial power distribution in a core containing fuel assemblies that had been irradiated to about 29.000 MWd/tonne. The total gamma activity of the fuel was measured by means of ionization chamber designed and built by Westinghouse for this purpose. By taking advantage of the symmetry of characteristics of the core quadrants it was possible to obtain sufficient information for a meaningful verification of the power distribution based on gross gamma scanning of only part of the fuel assemblies. The measurements were carried out on all the assemblies of a quadrant and on four or five assemblies in each of the other three quadrants, selected so as to reveal any power tilt in the core.

3.1.2 Reasons for selecting gross gamma scanning: after removal of the in-core instrumentation from the Trino Vercellese Reactor, power distribution measurements by means of gross gamma scanning have become of paramount importance to validate the design of the loading map of each operation cycle, and they also allow the uncertainties arising from the possible buildup of errors in successive cycles to be minimized in the calculation of the hot channel factors. The measurement of the core power density distribution based on gross gamma scanning is justified by the assumption that, shortly after shutdown, the activity of fission products whose half-time is very short as compared with the period over which the fuel was irradiated can be considered proportional to the power density. This assumption was verified experimentally at Trino at the end of the second cycle by comparing the assemblywise gross gamma activity measured immediately after the refuelling shutdown with the experimental power distributions obtained through the scanning of the 1.6 MeV activity of ^{140}La in the same assemblies /22/. The standard deviation between the two measurements was 2.8%; a higher deviation was observed in the peripheral assemblies. Gross gamma activity measurements were preferred for operational routine because of the simplicity and ruggedness of detector design, minimum electronic requirements, the few mechanical components required, the rapid collection, reduction and analysis of the test data, and for the limited interference with refuelling operations. The last of these aspects enables the nuclear designer

(°) by M.Paoletti Gualandi, S.Felici, R.Guidotti, P.Peroni (ENEL-DCO) to take timely action, if necessary, without delaying the beginning of the next cycle.

3.1.2 Measurement Method. For the gamma scans, the gross gamma activity of selected fuel assemblies is determined by four miniature, cylindrical ionization chambers located in a holding fixture such that each detector is perpendicular to the assembly axis and near the center of each of the four major assemblies faces. The ionization current in each detector is measured continuously as the assembly is inserted into and withdrawn from the detector holder. The relative sensitivity of each detector and the decay factor is determined by measuring the current from each detector when adjacent to each of the faces of the standard or reference assembly. Since it it impractical to rotate the fuel assembly, the detector positioning device is rotated 360° in 90° steps, with an insertion and withdrawal of the reference assembly being performed after each step. The detector signals of each of the selected fuel assemblies thus obtained, corrected for the relative sensitivity of each detector, is averaged over four faces and multiplied by the decay correction factor to obtain the activity normalized to the beginning of the test.

3.1.3 Calculation Method: On the basis of the information from the monthly operation reports (e.g., power, energy output, water temperature, boron concentration required to control the reactor) and by adequately representing the geometry of the square and cruciform fuel assemblies present in the core, core follow-up calculations were performed by BURSQUID code /23/ in order to take into account the impact on the EOC-3 irradiation distribution related to the real operating conditions in which the reacotr had operated during Cycles 1, 2 and 3. The BURSQUID code used for the calculations was a three-neutron-group (two thermal and one fast) version.
The geometrical model representing Cycle 1 (about 3800 mesh points, 130 compositions, three supercompositions) was limited to one quarter of the reactor since the fuel assemblies had been loaded in quadrant symmetry. Instead, for Cycles 2 and 3 (about 14.700 mesh points, 448 compositions, three supercompositions), the calculation had to be extended to the entire reactore, as the aforesaid symmetry did not exist. The effect of non-uniform axial burnup distribution in the core on neutron leakage was taken into account by using a regionwise buckling that varied with the irradiation of each region.

3.1.4 Comparison of measured and calculated values: The thermal power distribution of the forty-two scanned assemblies, calculated with the BURSQUID code, was compared with the measured value normalized to the mean value of the calculated values. The standard deviation between the experimental and calculated values was 3.1%. The scattering of the percental deviations ($\frac{\text{measured-calculated}}{\text{calculated}}\%$) revealed:

- an underestimate of the calculated values that increases with the burnup of the assemblies present in the central region of the core to a maximum of about 5%;
- an overestimate of the calculated values for the peripheral assemblies up to a maximum of about 5%.

3.1.5 Conclusions. The trend indicated above could be the result of either the uncertainties in the calculations or of biases characteristic of the experimental method. This comparison contributed towards the trimming up of the BURSQUID code on the Trino Vercellese core. The validity of the calculation method had been ascertained at the end of Cycle 1 by comparing the calculated and experimental results relating to the ^{55}Mn activity and the burnup distribution. A standard deviation of 1.4% was found for the ^{55}Mn activity; the standard deviation for the burnup distribution was 1.5% /22/.
A further confirmation was obtained at the end of Cycle 2 by comparing the results of the power distribution obtained with gross gamma scanning and the calculated values; the resulting standard deviation was 2.5%. In addition, the Keff of the core, obtained by follow-up calculations, was practically constant throughout Cycles 1, 2 and 3 (Keff = 1.00000 + 0.00500) and no bias was found in the core with the accumulated burnup.

3.2 Analysis of Rod-By-Rod Power Distributions Measured On Two Irradiated Prototype Plutonium Assemblies in the Garigliano Power Reactor (°)

3.2.0 Within ENEL's program for the recycle of the plutonium obtained from its nuclear power plants, ENEL is implementing a research program based on the performance of the prototype plutonium assemblies that were loaded into the Garigliano boiling water reactor in 1968 and 1970. This activity aimed at developing calculation methods for the nuclear design of plutonium assemblies, at experimentally verifying the capability of these methods of predicting fuel performance, and at defining the technical and economic conditions suitable for the utilization of plutonium as a recycle fissile material.

(°) by M.Paoletti Gualandi, S.Felici, R.Guidotti, P.Peroni (ENEL-DCO)

3.2.1 Rod-by-Rod Power Distribution Measurements. Within the scope of this activity, measurements of rod-by-rod ^{140}La gamma activity were taken on two irradiated prototype plutonium assemblies, respectively after one and two cycles of residence in the Garigliano power reactor, by using a Ge-Li detector and the associated equipment installed outside the spent fuel pool /24/. These measurements gave the local power distributions in the plutonium assemblies at different void contents.
The activity measured was that of 1.6 MeV gamma rays emitted in the ^{140}Ba-^{140}La decay. The experimental activity data, multiplied by the decay correction factors, were converted into relative power values by using appropriate power-to-activity conversion factors obtained for each rod from the assemblywise calculations performed with the BURSQUID code described below.

3.2.2 Calculation Method. During the reactor life, the power, burn-up and void distributions are periodically calculated for each assembly at eight axial levels by means of the ENEL version of the three-dimension diffusion code FLARE, in the framework of the fuel accountability program. With these data, a follow-up calculation was carried out for each assembly by means of the BURSQUID code so as to simulate the irradiation history at the levels at which the assemblies were scanned. For this purpose, calculations in x,y geometry were carried out by using five neutron energy groups, two of which are thermal. The neutron current around the assembly was assumed to be nil (symmetrical conditions on each side) as if the assembly were isolated from the rest of the core. Equivalent absorption cross-sections were used to simulate the insertion of a control rod. To simplify the calculations, it was also assumed that a diagonal symmetry existed during irradiation, but this meant neglecting the effect of the presence of the in-core instrumentation guide tube near one edge of the assembly.

3.2.3 Comparison of Calculated and Measured Values. The theoretical rod-by-rod power distributions of the two prototype plutonium assemblies show good agreement between the calculated and measured values of the levels at low void content, with a standard deviation between 2% and 3%. For higher void contents, the standard deviation increases to 4.4%. The power densities of the fuel rods near the assembly corners that were affected by the intermittent presence of adjacent control rods during irradiation were found to be significantly higher than those of the rods near the other corners. This effect is due to the lower burn-up accumulated in the rods near the inserted control rods, which causes the power density to increase sharply as soon as the control rods are withdrawn. This effect is offset as burn-up proceeds; however, for the assemblies scanned under this program, it was still high after about 4000 MWd/tonne. In the case of the plutonium-island assembly, a more pronounced power peaking in one corner was found because the adjacent control rod had been inserted for a long time. The calculations performed for the same two prototype assemblies revealed the same effects, but less pronounced. The axial power shapes measured on a few rods of the two plutonium assemblies appear to be more pronounced when passing from the peripheral rods to the inner ones. This phenomenon might be explained by the influence of the voids on the power sharing between rods differing in plutonium content and by the attenuation effect of the water gap on the axial perturbation caused by the voids in the outermost rods.

3.3 Processing of Experimental Data Obtained From Gamma Scanning (°)
In the field of core performance evaluation, the data measured require in general further processing in order to obtain data that are meaningful or comparable to the values obtained from the calculation codes. In the case of assemblywise gamma scanning, it is necessary to correct the gamma activity of the radioisotope being measured to account for the rod-shielding effect /25/, in addition to the other operations such as gamma-ray spectra processing, decay correction, etc.
The rod-shielding effect is due to the attenuation undergone by the gamma rays when they cross the fuel assembly regions lying on the path towards the detector. The attenuation is neglibile or almost total, depending on whether the gamma rays come from peripheral or innermost rods in the assembly. It is thus the peripheral rods that most contribute to the measurement, so that different activity distributions in an assembly give rise to different readings for the same average activity emission. Therefore, the distribution of the activity of the measured radioisotope inside the assembly must be factored in.
For this purpose, ENEL developed the ATTENUA and ATTIVA codes, which, linked with the BURSQUID code, allow the calculation of the correction factor that gives the ratio between the actual and measured activity for any isotope generated by fission. The ATTIVA code picks up the fission cross-sections of the fissile isotopes and the absolute fluxes for each fuel rod and burn-up interval from a BURSQUID calculations, and calculates the fission product build-up in each rod. The rod-by-rod activity distribution of a given fission product isotope is then fed into the ATTENUA

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code, which calculates the gamma activity for the gamma ray energy of the selected isotope, as measured by the detector. This value, divided by the total activity inside the assembly, gives the theoretical correction factor.

3.4 Isotopic Correlations (°)

Over the several years of operating experience, ENEL has collected a host of data relating to the isotopic concentratiohns present in the irradiated fuel of light water reactors. The sources of this information are essentially three:

a. Isotopic analyses of U, Pu, Am, Cm, ^{148}Nd, ^{137}Cs on specimens taken from fuel rods, within the framework of research compaigns.
b. Isotopic analyses of sample solutions taken during the reprocessing of individual batches of spent fuel. (It should be pointed out, however, that depending on the reprocessing procedure, the samples may not correctly represent the characteristics of the whole batch).
c. Gamma activity spectrum measurements on individual rods or on the whole assembly after a decay period of several months outside the core. These measurements allow the activity distribution, and thus the concentrations of some of the fission products, to be derived.

The collection of data from the analyses under (a) and (b) enabled ENEL to develop and check a number of heavy isotope correlations, some of which are very promising. This study is of particular significance because it is performed on fuel irradiated in different reactor systems (BWR and PWR), for different enrichments, void contents and power levels.

Another study under way aims at establishing correlations between heavy isotope concentrations determined by means of destructive tests and the activity of certain fission products obtained by means of non-destructive measurements. With these correlations, it would be possible to know the amount of fissile isotopes recovered from reprocessing simply by using the gamma scanning data obtained before sending the irradiated fuel to the reprocessing plant.

So far, these studies have been based on experimental data and have sometimes given rise to difficulties in interpretation because of the scanty number of measurements available. It then appeared desirable to perform qualitative and quantitative analyses of the correlations on the basis of isotopic concentrations obtained by means of the burn-up codes, in order to single out the parameters that most affect these correlations. In particular,

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ENEL developed the ATTIVA code, which provides the activities of the various fission products, to be linked to the BURSQUID code that calculated the evolution of the heavy isotope concentrations. In this manner it has been possible to single out the promising correlations between burn-up or heavy isotope concentrations and fission product activities, and to discard other that were based on experimental data and had initially appeared to be of interest.

3.5 Automatic Calculation Systems For the Assessment Of LWR Core Criticality (°)

In connection with the operation of its light water reactors, ENEL developed a method to assess core criticality, particularly for the purpose of verifying the shutdown margin. This parameter is one of the most important requirements to be satisfied in the refuelling of boiling water reactors and in the conditions following a "credible" steam pipe break in pressurized water reactors; therefore, it must be verified several times in the course of each operating cycle.

The method consists in utilizing the axial condensation of the lattice parameters relating to the individual assemblies and is thus based on the assumption that the influence of adjacent assemblies on the axial flux distribution of the assembly under consideration is very small. Thus, it allows the core reactivity to be calculated correctly only when every control rod is completely in or completely out, as is the case in most subcriticality assessments. In the representation, the effect of the control rods is taken into account by homogeneizing neutron absorption over half the area of the four adjacent assemblies.

The calculation method /26/ breaks down essentially into the following steps:

a. Determination, for each assembly or group of assemblies, of the axial distributions of the burn-up and of the void content averaged over the assembly life ("integral void content"), by means of a three-dimension core follow-up code.
b. Determination of the cross-sections of the materials other than those of the fuel cells, such as the sheath and water gap, by means of neutron spectrum calculation codes.
c. Determination of the parameters representative of the control rods, that is, the cross sections in respect of fast or epithermal neutrons, and of the extrapolation lengths for thermal

(°) by S.Felici, M.Paoletti Gualandi, P.Peroni (ENEL-DCO) neutrons, by means of a transport code.

d. Determination of a set of rod-by-rod isotopic concentrations as a function of burn-up and of the void conditions typical of the axial regions into which the core is conventionally divided, by means of a bidimensional burn-up code.
e. Calculation of the assembly lattice constants from the foregoing rod-by-rod isotopic concentrations in the conditions for which criticality is to be verified, by means of a bidimensional diffusion code in x,y geometry.
f. Grouping of the assemblies by burn-up level, axial burn-up distribution and "integral void content".
g. Axial condensation of the assembly lattice constants, for each group of assemblies, by means of a unidimensional diffusion code. For each axial region, the inputs to the computer are the assembly lattice constants (item e) referred to the related burn-up level and integral void content.
h. Determination of core criticality, by means of a bidimensional diffusion code, extended to the whole core in x,y geometry.

This method called for the utilization of a great number of codes and considerable manual work to set up, prepare and process the data for the various calculation steps. To reduce the painstaking manual work, the attendant errors, and long machine times, ENEL conceived a computer system to perform the calculations automatically, which is now being implemented by CISE. In view of the characteristics of the calculation method, it was considered desirable to divide the system into the following two subsystems.

a. Preparation of the library of the assembly lattice parameters as a function of burn-up and moderator void content, for the conditions in which the criticality is to be assessed. This subsystem contains two subroutines that respectively determine the nuclear parameters of the materials other than the fuel cells and of the control rods, by means of simplified correlations.
b. Calculation of the core reactivity for the case under consideration. This subsystem contains a subroutine that groups similar assemblies into distinct classes, a subroutine that reads and then interpolates the data contained in the library prepared by the first subsystem, the unidimensional code that performs axial condensation, and the bidimensional diffusion code that calculated the reactivity of the whole core.

3.6 BWR work at PMN
At PMN (Progettazioni Meccaniche Nucleari, Genova) efforts in this field have focused on the development of a code for calculating power distributions of a BWR, in 3-dimensional geometry and neutronics-thermohydraulics coupling. An accurate analysis of various "coarse mesh" calculational methods for neutron fluxes has been carried out for this purpose. As a result, the MARPLE code has been developed (this code is not freely available). Results from MARPLE have been compared, with very good agreement, with those from the benchmark in EACRP-U-47.

3.7 Critical experiments (CSNC)
Measurements of temperature coefficients and kinetic parameters have been performed in the critical facility ROSPO, both in small clean and large rodded cores. A few reactivity and power distribution measurements have been made in cores incorporating UO_2(depl)-Gd_2O_3 (25÷200 mg/cm^3) rods in the central element.

3.8 Burnable poisons (CSNC, PPU)
Experiments on irradiation of burnable poison samples were carried out. Reactivity measurements are in progress. A description of the samples and of the irradiation conditions is given in /27/.

3.9 Activity on code development at CNEN (PPU)
The activities in the physics area for LWR's previously performed at Laboratorio Fisica e Calcolo Reattori, have been taken up by Programma Plutonio, which aima at assessing design capabilities for LWR's. These new activities are in cooperation with Italian industry, mainly, in the physics area, with ENEL. The computational tools consist basically of a x,y code for fuel element analysis at different water densities BURNY-BEVE, and of a tridimensional core simulator named BACONE with coupled neutronics and thermohyidraulics for core analysis; a monodimensional code DANTE is used for survey calculations of axial power distribution.

3.9.1 Improvements in the codes.
An advanced version has been set-up of the BEVE code which is used in the BURNY chain to evaluate the neutronic behaviour of Gd poisoned cells, taking into account the interrelated space and energy selfshielding. The new version which is used for Pu fuel elements supplies cell constants for an energy scheme employing two groups in the thermal range; the necessary libraries have also been set up, including up and downscattering properties. The following new calculation features have been introduced into BURNY:

- description of control systems for present generation PWR's (RCC of Ag-In-Cd or B_4C);
- description by a transport method (P-1 approximation) of control rods for BWR's;
- treatment of cores with periodic lay-out of fuel elements;
- geometrical averaging of cross sections over the macro-cell to be used for tridimensional or axial codes;
- multigroup treatment of Gd absorption in the epithermal range.

3.9.2 Implementation of new codes
Using the same formalism as in the dynamic code ALADINO, an axial code named DANTE with coupled neutronics and thermohydraulics has been implemented which calculates using two energy groups reactivity and axial power distribution versus irradiation. Different compositions can be used along the axis so as to use the code to optimize the axial distribution of burnable poisons.
Both BACONE and DANTE employ the description of the nuclear behavior of the elements in terms of analytic fitting of the cross section values parametrically calculated for discrete sets of water densities and irradiation levels by BURNY-BEVE: a calculation code, named BUBA, has been developed to handle all this fitting procedures in an automatic manner.

3.10 Verification of the codes by comparison with experimental results and applications (PPU)

3.10.1 Use of burnable poisons in LWR's.
A large program is on the way to investigate in detail the behavior of Gd as a poison for both U and Pu LWR fuel. Irradiation in operating conditions (BWR capsules) are in progress in the SILOE' reactor (C.E.A. - France). A first set of experiments employing NaK cooled capsules has been recently completed: the evolution of power generated in the poisoned pin vs. irradiation was measured and the agreement with the results calculated by BURNY-BEVE is more than satisfactory.

3.10.2 Use of RCC systems in PWR's.
The results of measurements in critical experiments in PWR geometry employing either B_4C or Ag-In-Cd cluster control or spike absorbers, both in U fueled and in Zn fueled lattices have been employed to verify the reliability of the BURNY-BEVE code.
The results have been quite satisfactory /28, 29/.

3.10.3 Comparison with the results of different methods.
For particular problems, such as multizone heavily Pu loaded fuel elements, central water rods, and so on the effects connected with using more than one group in the thermal energy range and with using transport approximations (Sn method) have been evaluated.

The results by BURNY-BEVE have also been compared, with very good agreement, with the results, available in the literature or supplied by utilities and reactor vendors.
For instance a comparison is carried out by Badenwerk A.G. to which BURNY-BEVE was supplied under the provisions of a contract with CNEN.

3.10.4 Evaluation of economic incentives in replacing Inconel-Stainless-Steel spacer grids with new design Zircalloy-Inconel grids.
It was evaluated by BURNY-BEVE and DANTE, which was the reactivity and hence enrichment or lifetime gained by reducing the amount of absorbing materials in the spacer grids. Consequently the maximum additional costs was evaluated for the new grids which was compatible with an economic incentive.

3.10.5 In the frame of the bilateral agreement between CNEN and the Rumanian AEC, some of the CNEN LWR codes have been modified and applied to research reactors of the water-water type /30/.

4. HEAVY WATER REACTORS

4.1 Measurements in RB1 (CEC)

In the frame of the R.&D. programme in support of the CIRENE reactors investigations have been carried on concerning the applicability of the "poisoned" and "unpoisoned" technique for PCTR type measurements. The comparison between the results of the different techniques are under investigation.

4.2 Measurements in RB3 (CEC)

In RB3 the effect of the dumping of D_2O, in the CIRENE reactor, on the control instrumentation has been simulated. The measurements have shown that the instrumentation blinding effect depends strongly on the thickness of the D_2O shell which constitutes the dumping ring. The experimental data have been compared with the fluxes inferred by the DOT code. The agreement is quite bad. Some measurements have been done to test some interpretation methods related to strong negative reactivities in D_2O reactors.

4.3 Reactor codes

4.3.1 Heterogeneous codes (CEC)

The HEFREM code has been written which is able to combine the "heterogeneous" codes HETROIS and SOS with BUCFIT to evaluate the B^2 obtained by substitution method and flux distributions. This chain has been used for:

1) interpreting substitution experiments;
2) calculating the RB3 liquid rods efficiency in relationship to their position in the reactor and to the amount of Boron;
3) calculating the buckling of lattices in relationship to the number of fuel elements.

These studies are almost finished, and we are already gathering their results.

4.3.2 Montecarlo code (CEC)

Into the Montecarlo KIM code (k-infinite M.C.) which calculates the infinite multiplication factor and cell parameters in Canadian-type cluster cells by simulating neutron lives from fission to thermalization, a group cross section description has been introduced in the epithermal region for nuclides other than U-238, for which the resonant structure is accounted for by means of analytical formulas. The results obtained for the CIRENE reactor cell for various coolant densities are in very good agreement with experimental results. More comparisons are under way.

4.3.3 Reactor codes (CISE)

Within the CIRENE Program, development work has been continued at CISE as far as heterogeneous methods and homogenized lattice theory are concerned. In the heterogeneous methods field, Berna's formalism for the calculation of heterogeneous parameters has been fully acquired and partly simplified. Numerical studies have been carried out to spot the best way for calculating the dipole extrapolation length of a channel: the most viable procedure has been demonstrated to consist in first evaluating Berna's β_0, to be subsequently used for the calculation of the extrapolation length on the basis of the theory outlined in /31/. Such a method is simple and yields values for Berna's parameter $\tilde{\beta}$ in very close agreement with those obtained by using Berna's more complicated formalism. As far as the homogenized lattice theory goes, three main areas have been investigated.

a) Revised epithermal library for PROCELLA: the preliminary work reported on last year has been complemented by further calculations, which have shown that the very structure of the library should be modified to include resonance absorption related parameters. An investigation is now underway to supply GAM with the capability of self-generating the appropriate Dancoff correction for a fuel bundle.
b) Homogenized cell theory. The theory is developed in /31/, where some basic discrepancies between different calculations of diffusion coefficients have been resolved. Among other things, is has been proved that, in a heterogeneous lattice, the diffusion area for a given energy group is not the ratio between diffusion coefficient and homogenized absorption cross section.
c) Use of reference codes. Two codes are now being thoroughly examined, namely DOT (2-D S_N transport) and GAM (multigroup B_N slowing-down). Modifications under study for GAM are mentioned under item (a). As far as DOT is concerned, since the code is planned to be first used for determining extrapolation lengths of the fuel elements to be tested on the ESSOR reactor at CCR Ispra, and such elements cannot be properly evaluated through the PROCELLA-type design code idealizations, an investigation on the (r, θ) option of DOT has been incepted with the main purpose of generating correct dipole extrapolation lengths. As a result, the DOT boundary condition set at r = 0 has been supplemented with a new option (now operating) which allows calculating titled fluxes properly when the "point" r = 0 has no "special" (reflective or sink-type) properties.

4.4 Thorium-D_2O systems (CSNC)

The second part of the measurements on ThO_2-UO_2 heavy water lattices has been carried out in cooperation with Euratom on the ECO reactor at Ispra just prior to its decommissioning. Static reactivity measurements have been carried out on 37 element clusters with the following enrichments in U-235: 2.2%, 2.4%, 2.6% and 2.8%. Two pitches have been studied (22.3 and 28.05 cm); the coolant inside the fuel element was either air or a mixture of 70% D_2O and 30% H_2O. Results are shown in Fig. 1 where reactivity variations are expressed in terms of the critical heavy water level. The interpretation of the results is under way, in order to study the effect of enrichment on buckling. Discrepancies observed in the buckling measurements carried out on ECO in 1970 have been traced back to inhomogeneities in enrichment of the UO_2-ThO_2 pellets. A re-interpretation of the results by means of a heterogeneous method and with the actual values of enrichment is under way. Measurements of K_∞ by the PCTR method and of the cell parameters on 2.4% and 2.8% enriched 37 rod clusters are scheduled for the second half of 1974 on the RB1 reactor at CNEN Bologna.

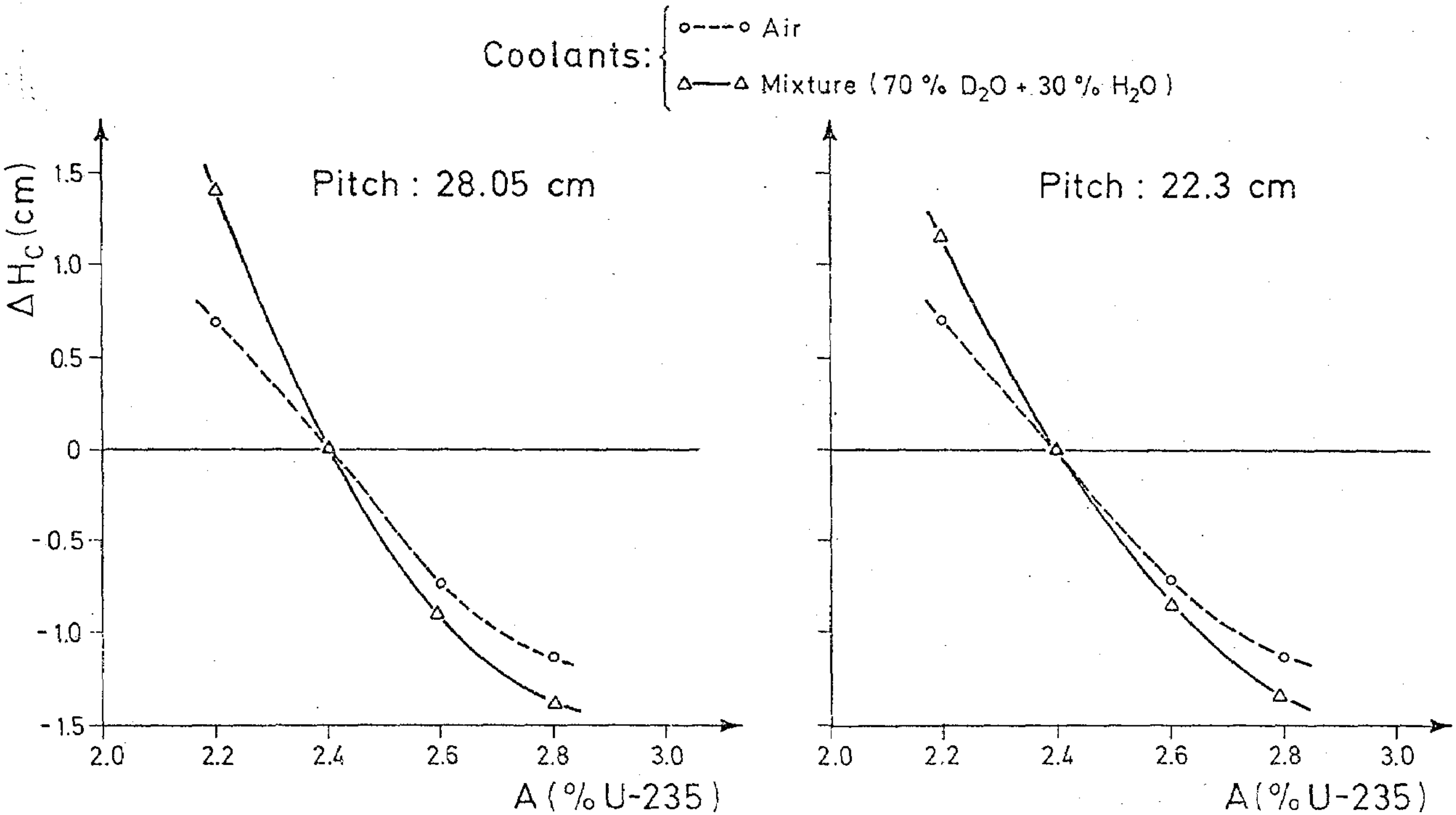

Fig. 1 - Reactivity measurements with ThO_2 - UO_2 elements in heavy water

5. FAST REACTORS

5.1 Neutronic design of the PEC Reactor (PRV, CSNC)

5.1.1 The neutronic design of the PEC reactor has been carried out on the new reference core with mixed plutonium-uranium oxide fuel /32,33/. A 27 group library has been prepared by the MC^2 code for the PEC core composition using ENDF/B-III data.

5.1.2 An extensive evaluation has been carried out for the control rod worths on the major part of the possible configurations they can assume during the rise to power, the reactor life and as a consequence of shut-down interventions. Reactivity variations of about 10% on the worth of a single compensation rod have been found when considering the rise-to-power or shut-down conditions respectively, with the corresponding conditions of extraction or insertion of the two shim rods. Similar results have been found for the safety rods. Reactivities introduced by a reduced number of safety rods have also been evaluated as a function of the insertion pattern.

5.1.3 A three-dimensional investigation in hexagonal (x,y)+axial geometry has been performed by means of the CITATION diffusion code, taking into consideration the progressive insertion of control and safety rods into the reactor. Calculation with 23x23x15 mesh points has saturated the 1 Mbyte fast storage of the IBM 360/75 computer. In the horizontal cross section one had to use only one mesh-point per subassembly. For control rods it had been previously shown that the representation of 4 mesh-points per subassembly yielded the same results as transport calculations; the boron content in the case of the 1 mesh-point hexagonal representation has been normalized so as to obtain the same rod worth as in the 4 mesh-point xy representation. With this equivalent boron content, three dimensional calculations have been carried out to obtain the reactivity versus rod position curves. Comparison of these results with those of the 2-D calculations shows that, all other conditions being the same, the reactivity worth of the rods is smaller when the third dimension is taken into account: the difference is of the order of 2-3% in the case of 3 rods and of 6-7% in the case of 8 rods. The main cause of this discrepancy is probably to be found in an axial shadow effect, the magnitude of which increases with the number of inserted rods /34,35/.

5.2 Fast reactor activities at NIRA

5.2.1 The recently formed State Industry for Advanced Reactors has taken up from IRI and ENI the activities on fast reactor design and construction, including PEC. In particular, fast reactor physics activities previously carried out at PMN of the IRI group have passed over to NIRA.

5.2.2 The work of analysis of numerical methods for the solution of space kinetic equations, previously reported and presented at the Tokyo meeting /36/ has been terminated with the conclusion that such numerical solutions cannot be appreciably accelerated. The activity in this area, in cooperation with IAC (Istituto per le Applicazioni del Calcolo, Roma) has shifted to other types of approach, and in particular to the study of asymptotic solutions and to the search of "bands" within which the solution is contained; such bands can be narrowed by iterative procedures, so as to be able to know an overestimation of power and temperatures with the desired accuracy.

5.2.3 The work on neutronic optimization by means of generalized perturbation techniques has been carried on. The relative code chain is now operational /37/.

5.2.4 An analysis of methods for the reduction of cross section groups is under way; adjoint weighting is being considered.

5.3 Experiments on TAPIRO (CSNC)

5.3.1 The results of the experiments previously carried out on the TAPIRO reactor have been further analyzed, together with those of a few new physics measurements /38/, in order to get information on the Cu cross sections and to resolve the existing discrepancies in the ENDF/B files between the capture cross section for natural copper and that for the Cu isotopes /39,40/. Further indications on the high energy fission cross section of U-235 were also obtained.

5.3.2 The space distribution of the importance function has been measured by means of a Cf-252 source in the core and the reflector of TAPIRO /41/. Several techniques for this measurement have been compared. The results are in good agreement with calculations in the core and the inner part of the reflector; some discrepancies have been found in the outer part, which are believed to derive from the simplified geometry or energy representation used in the calculations. Measurements have also been carried out for the normalization integral and for the energy dependence of the adjoint flux by means of different neutron sources; the results for this last experiment are of difficult interpretation due to uncertainties in the strength and spectrum of the sources.

5.4 Sensitivity studies and cross section adjustments

5.4.1 A method proposed by Mitani and Kuroi has been improved. Such method, by which possible systematic errors on the cross sections may be accounted for /42,43/, has been inserted in a differential data adjustment code /44/. Its use requires a small modification of already existing calculational methods, provided these allow the standard solution, i.e. that relative to normal errors, to be obtained along with the method of reduction by elements. In case the method of the Lagrange multipliers is adopted (wich results preferable in other cases) a problem of singularity of the modified cross section dispersion matrix, obtained for those cases, would be encountered.

5.4.2 A preliminary study has been performed on a method for the adjustment of the nuclear parameters themselves in place of the group cross sections /45/. This method has been called "consistent", since the consistency with the direct method adopted for the generation of the group cross sections is assumed. More precisely, the same theoretical models and the same approximation should be adopted in the (inverse) procedure of adjusting the nuclear parameters. The results of this investigation, which has been based on the algorithms of the type of those used in ETOX, seems to assess the validity of such an approach in terms of effort and time required.

5.4.3 Sensitivity studies applied to various problems have been carried out /46,47,48/.

5.5 Multigroup cross sections data

5.5.1 The CNEN revised version of the RIGEL-ETOE-MC^2 code system was used to generate cross sections for the PEC reactor neutronics design, starting from the ENDF/B version III data. An interface with neutronics codes was developed /49/. The same ENDF/B-III data were used to generate a 27 group library in the Bondarenko format. The ETOX code was used, as implemented at Casaccia. The 27 energy group structure, based on half-lethargy intervals, is the same used for all the MC^2 calculations. All the important isotopes were processed /50/.

5.5.2 An extensive analysis of methods was completed for the elastic moderation cross sections. The Stacey method, based on the continuous slowing-down theory, was found to give results in excellent agreement with the more sophisticated algorithms of MC^2-2 /51,52/. A new version of the 1DX code was developed to include the Stacey method in the Bondarenko formalism.

5.5.3 Adjustments and evaluations based on integral experiments. The ZPR-6 assembly 7, SNEAK assemblies 7A and 7B integral experiments were used within the statistical correlation procedures described in point 1, to generate adjusted infinite dilution cross sections for U-238 (σ_c, σ_{in} and σ_f), Pu-239 (σ_f and σ_c), U-235 (σ_f). The results, which were reported at the Tokyo meeting, are related to the ENDF/B-III data. Lower captures, higher fissions and inelastic cross sections for U-238 are the highlights /53/.

5.6 Dynamics (CSNC)

5.6.1 Development of fast reactor dynamics codes has beer carried on at the LFCR of CNEN, Casaccia. Codes produced up to now include:

a) NADYP /54/, a two-dimensional code in cylindrical geometry for the PEC reactor. It uses the metastatic method for the diffusion equations and the method of characteristics for the thermo-hydraulic equations. Sodium boiling will be added later.
b) NASTAP /55/ which performs the stationary calculations in the same conditions as those of NADYP.
c) DIFF-2 /56/, a diffusion sub-routine using the Ritz-Galerkin method.

5.6.2 Some EACRP Benchmark Dynamics Problems have been solved /57/; some mathematical aspects concerning non-linearities in the diffusion equations have been studied.

6. <u>HIGH TEMPERATURE GRAPHITE REACTORS</u> (AGIP)

6.1 At AGIP NUCLEARE, Bologna, the measurements programmed for the HTGR lattices using the appropriately modified RB-2 reactor have been completed; preliminary results have been presented at the 17th DCP Mēeting in Cadarache, October 3-4, 1973. In particular, the K-infinity for this lattice was measured by the null-reactivity oscillation technique, which was described in the former EACRP progress report. The overall interpretation of the set of results obtained is being completed. Techniques employed do not significanlty differ from those previously reported.

6.2 Since the beginning of 1974, in the frame of a cooperation with Euratom-Ispra, a set of measurements of Doppler effects has been carried out on HTGR lattices for temperatures up to 600°C, always using the null-reactivity oscillation technique. The oscillator employed is modified with respect to the one used for the "cold" measurements and has now a stroke of 2000 mm. The results of the measurements are being evaluated.

7. SHIELDING

7.1 Neutron propagation in iron and sodium (CSNC)

The measurements previously carried out on an iron block /58, 59/ were used to compare ENDF/B version I and III data for iron. Previous results in this field were checked /60/. A first series of measurements was carried out on a sodium block. Results are being evaluated.

7.2 Removal-Diffusion methods (CEC)

A removal-diffusion shielding code allowing one- and two-dimensional calculations in cylindrical and (x,y) geometry has been prepared; it features restart possibilities at any point of execution. This possibility is of considerable help when analyzing thick shields requiring long calculation times. Comparisons with other codes have been started on two reference problems of actual importance: one relative to a PWR shield, the other to the PEC reactor.

8. NUCLEAR DATA EVALUATION AND PROCESSING (CEC)

8.1 Isospin splitting of the ^{13}C giant dipole resonance in the continuum shell-model

An investigation on the isospin components of the ^{13}C giant dipole resonance has been carried out by means of the 1p and 2p-1h shell model including the one-particle continuum. The continuum photo-reaction cross sections and angular distributions have been compared with experiment. The T and T+1 dipole strengths with their related sum rules have been inferred from a discrete calculation based on the same assumptions as the continuum model /61/.

8.2 Direct and semi-direct radiative capture

The detailed analysis of the radiative capture cross sections for heavy nuclei in the giant resonance region by the semi-direct capture model with "volume" interaction has been continued. Shape and magnitude of the cross sections for ^{208}Pb and ^{142}Ce nuclei have been satisfactorily reproduced. A good agreement is also achieved in comparing the predicted and detected gamma-ray spectra following the capture of ^{28}Si, ^{88}Sr, ^{142}Ce and ^{208}Pb /62, 63/.

8.3 Cross section evaluation work

The following studies have been completed, concerning the evaluation of the low energy nuclear cross sections:

a) A comparison between single-level and multilevel calculations of scattering cross sections in resonance region for Cr, Ni and Fe. A paper has been presented at the meeting on structural materials held in Karlsruhe 8/9 May 1973 /64/.

b) The calculations of average radiative widths based on the detailed balance principle and the fit of the experimental photoabsorption cross sections. A code (LARA) has been prepared for such calculations. The work has been presented at the "Fission Products" Panel held in Bologna, 26/30 November 1973 /65/.

c) Evaluation of neutron cross sections of natural Cu and ^{63}Cu, ^{65}Cu isotopes in the range 0-15 MeV has been completed and the files in UKAEA format have been prepared.

d) A work has been completed on the calculation of nuclear cross sections of type $\sigma_{x,a}(E)$ where x and a may be neutron, proton and α particles. A code (CERBERO) has been prepared for such calculations.

e) Some remarks on the U-238 inelastic scattering cross sections have been prepared for INDC /66/.

9. FUSION REACTORS

Activation of copper coils, to be used in the J.E.T. (Joint European Torus) machine for confining deuterium-tritium plasma, has been evaluated by transport calculations performed by the ANISN code with 26 energy-groups in S_6/P_o approximation. Build-up of Co-60 within inner layers of copper coils, due to the (n,alpha) reactions caused in Cu-63 by the 14.1 MeV neutrons from the D-T plasma, was found to present the most severe limitation to extended operation of the machine /67/. Subsequent calculations carried out with 100 energy-groups in S_6/P_3 approximation have shown that formation of Co-60 occurs in prohibitive amounts also within outer layers of copper coils and that use of a moderator placed between plasma and copper coils does not appear to reduce as desired the number of neutrons entering copper with energies greater than the threshold (about 6 MeV) for Co-60 formation.

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Reactor Physics Activities in Japan
Period June 1973 to May 1974

by J. Hirota

1. Light water lattice experiment

Critical experiments were carried out on 3.0% PuO_2-UO_2 lattices (the atomic number density ratio of H to Pu ; 295, 402, 494, 704 and 922) using the zero power light water moderated critical assembly TCA in JAERI in cooperation with the Power Reactor and Nuclear Fuel Development Corporation (PNC). The fuel is PuO_2-UO_2 pellet of 6.1 g/cm^3 clad in Zircaloy-2 tube, and the effective size of fuel element is 10.7 mm in diameter and 706 mm in length. The plutonium contents are 63,22, 7, and 2% of ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu, respectively.

Critical sizes and power distributions were measured for the cores;

a) two region cores consisted of 3.0% PuO_2-UO_2 and 2.6% UO_2 lattices

b) cores which have water gaps

c) cores which have mock-up voided regions

In the cores a) and b), transversal gold activities and β_{eff}/ℓ were also measured. The moderator temperature coefficients were measured for the lattices of which H/Pu were 402,494,704, and 922 over the temperature range from 20°C to 80°C.

The codes used in the calculation of three group cross sections are LASER and UGMG42 (JAERI revised MUFT code) - THERMOS. Two dimensional calculations were carried out using the PDQ-5 code. The calculated K_{eff} by LASER-PDQ-5 agreed with the experimental results within the range of $-0.2 \sim 0.5\% \Delta K/K$ and $0.3 \sim 0.6\% \Delta K/K$ for the two regional and the water gap cores. The K_{eff} by UGMG42-THERMOS-PDQ-5 agreed within the range of $0.4 \sim 1.0\% \Delta K/K$ and $1.7 \sim 2.2\% \Delta K/K$ respectively.

It was experimentaly shown that the $\{\beta_{eff}/\ell\}_c$ of the two region core was synthesised as the linear combination of the $\{\beta_{eff}/\ell\}_i$ and the statistical weight W_i of each region i. For the moderator temperature coefficient of the lattices of which H/Pu was 402, for instance, the difference between the calculated and the experimental coefficient increases from -0.8 to -1.2 cent/°C with the temperature from 20°C to 80°C.

2. BWR power distribution measurement

A study has been made on the power distribution at the Fukushima-1 plant under the joint program between Tokyo Electric Power Company and Toshiba. The measurements were performed twice and fuel bundles in different burnups were used: The fuel bundles were gamma-scanned axially with a Ge (Li) detector and the relative distributions of fission product contents-La, Zr, Pr and others were obtained from the analysis of the gamma-ray spectra. The measured values were compared with the values obtained from the trace calculation by the three dimensional code which followed the reactor history from the startup to the shutdown just before the fuel discharge. The agreement between the calculation and the measurement is fairly good.

3. Non-destructive burn-up measurement

Non-destructive burn-up measurements on 2.5% PuO_2-UO_2 fuel pins have been tried by means of the gamma spectrometry at the JPDR fuel storage pool. These fuel pins were fabricated by PNC and irradiated up to about 1200 MWD/T in maximum at the Halden HBWR. The correlation between the activity ratio $^{134}Cs/^{137}Cs$ and the activity of ^{137}Cs indicates a very good linear relation at the cooling time of 45 months. The proportional constant is larger than that previously measured on the JPDR fuel pins under the same experimental condition. It is considered that the difference results from the difference in the neutron spectra.

When the fuel temperature is lower than about 1600°C the migration of Cs can be neglected, and the activity of ^{137}Cs gives a reasonable measure of integral burn-up. The above mentioned relation shows that the isotopic correlation of ^{134}Cs and ^{137}Cs also gives a good measure of burnup for the Halden HBWR. Generally, the ratio $^{134}Cs/^{137}Cs$ is affected by the shift of power distribution during irradiation and by the change of neutron spectrum in the core, because the half life of ^{134}Cs is not so long and the epithermal neutron capture of ^{133}Cs is considerably large comparing to the thermal neutron capture. In the case of JPDR fuel pins, these effects were apparently recognized. The accurate evaluation of effective neutron capture by ^{133}Cs in fuel pin is a interesting problem, because the resonance structure of ^{133}Cs has some similality to that of ^{238}U, then the ratio $^{134}Cs/^{137}Cs$ might be a good indicator of neutron spectrum in operating power reactors and also a measure of Pu/U.

4. Physics study for "FUGEN"

In order to investigate the core characteristics of the Advanced Thermal Converter Prototype Fugen (heavy water moderated light water cooled 165 MWe), a wide reactor physics parameter survey has been made by using 0.54% and 0.87% enriched PuO_2-UO_2 fuel lattices in the Deuterium Moderated Critical Assembly (DCA).

Experimental lattice conditions are 22.5 cm lattice pitch and four kinds of coolant void fractions (0, 30, 70, 100% voidage). Reactor physics parameters, which are almost the same as those measured for UO_2 lattices before, are β / ℓ measured with the pulse neutron method, material buckling for small number of fuel clusters by the substitution method, initial conversion ratio, fission and capture ratios and microscopic intra-cell thermal neutron flux distribution by the foil activation method, and power peaking factor in the cluster and macroscopic power distribution in the core by the wire activation method. These data have been compared with values calculated by the nuclear design code for Fugen (Cluster-IV and Methuselah-II).

The results on the neutron flux distribution of the two region core which consists of 37 (0.54% PuO_2-UO_2) and 84 (1.2% UO_2) fuel assemblies are summarized as follows:

(1) The local radial flux distributions in the UO_2 and PuO_2-UO_2 fuel cluster agree with the experimental data in the case of 100% coolant void fraction. In the case of 0% coolant void fraction, the calculated flux distributions in the moderator are rather low compared with the experiment.

(2) The calculated gross radial flux distribution in the position of the cluster center agrees well with the experiment within the experimental error in the case of 0% coolant void fraction. In the case of 100% coolant void fraction, the calculated radial flux distribution is rather flat compared with the experiment.

(3) Concerning the gross radial flux distributions in the position of D_2O lattices and the gross axial flux distributions, the differences between the experiment and calculation are large and it seems necessary to make the calculation model more detail.

5. Very high temperature reactor design

Since 1970, JAERI made three times of VHTR design under the cooperation of the atomic industrial groups; the first and the second preliminary designs of the experimental VHTR, and the first preliminary design of the large scale VHTR. In addition to these, the first pre-conceptual design of the experimental VHTR is now being carried out, and its results will be obtained in August, this year. After evaluating it, the first conceptual design of the experimental VHTR will be undertaken from this autumn.

To support the above designs, JAERI has made the Mark-series of the reference designs of the experimental VHTR since last August. The main parameters of the reference design are as follows: The reactor plant produces 50 Mw thermal power which is used to heat the primary helium gas to about 1000°C. This high temperature gas is then applied to the steam reformer for obtaining reduction gas. The reactor core, which is composed of 85 fuel columns, has 2.9m diameter and 3.0m height, and is surrounded by the graphite reflector. 38 control rods are inserted into the core from the top face in pairs. The coolant flows downwards and its flow is controled by orifice devices mounted on the top of the core. The hexagonal prismatic fuel elements are employed as fuel block, of which the across flatness and height are 30 cm and 50 cm, respectively, and 36 hollow type fuel pins or 18 tubular type pins are inserted into the block. The fuel is loaded uniformly in radial plane, but in four different zones in the axial direction for lowering the maximum fuel temperature, and the whole core is refueled at reactor shutdown. Average power density, average fuel burnup and fuel dwelling time are 2.5 w/cm^3, 40 GWD/T and 4 yr respectively. Design limit of the maximum fuel temperature is 1350°C.

6. A new experimental method for large fast reactors

A new experimental method, which gives us more useful informations than the usual sector-type or zone-type experiment, has been proposed.

We consider a system in a fast critical facility whose geometrical dimension is the same as that of a large fast reactor of interest. The difference between physics parameters of the system and the large fast reactor results from the difference between the nuclear characteristics of enriched uranium and plutonium or the core compositions. The basic idea and procedure of the new experimental method are as follows: The core of the system is divided into certain number of regions and measurements of physics parameters are repeated replacing the dummy composition with the real one in each region successively. From the results, the physics parameters of the large fast reactor can be estimated completely within the first order perturbation. The theoretical correction is made for the higher order which is only the interaction effect between the divided regions. The theoretical foundation is given by means of the generalized perturbation technique developed by L.N. Usachev and others, and also the higher order perturbation method developed by H. Mitani. This experimental method will be applied to the reactivity worth measurement on FCA Assembly VII-1.

7. Theoretical analysis of power reactor noise

Analytical models for power reactor noise are constructed in the framework of the linear response theory of Markoffian processes. A set of Langevin's equations for nuclear power, fuel and coolant temperature are derived on a single channel and a single phase flow model and are analytically solved. The procedure has been extensively applied to the more practical model. The calculated power spectral densities of fluctuations are compared with measured noise patterns and a reasonably well agreement is obtained. An extensive review and development article on the theory of power reactor noise is reported by K. Saito in the newly appeared journal "Annals of Nuclear Science and Engineering".

Power spectral density of neutron density is important as an aid in diagnosing faults in a power reactor. At JAERI, the point reactor kinetic equation with a Gaussian stationary and random function of time is studied without linearizing the equation. The condition for the stable mean neutron density and an integral equation for the power spectral density are derived using techniques based on the quatum field theory.

Its solution and the norm of the integral kernel for a noise with simple exponential correlation function are evaluated numerically. As the result, it is found that the correction to the usual power spectral density derived from the linearized kinetic equation increases with the correlation time of noise, and if the correlation time is less than several second, the correction can be neglected.

8. Integral check of FP group constants

Group constants of 28 important FP nuclides were produced with the data evaluated by the JNDC FP Nuclear Data Working Group. They are compared with those produced from the Cook's evaluated data at International Symposium on Physics of Fast Reactor held at Tokyo in 1973. After the symposium, the detailed information of reactivity change due to fission products on STEK cores was obtained from RCN. Hence the FP group constants have been checked using these integral data for 3 samples (HFR-101, HFR-102 and KFK-mock-up sample).

The followings are found:

1) As for HFR-101, the results are of 10% higher than the experimental data, while the RCN results are of 10% lower than the experimental ones.
2) As for HFR-102, the results agree well with the experimental values, while the RCN results are of 15% lower.
3) As for KFK-sample, the results are of 25% higher for STEK-1000, but agree well for STEK-4000, while the RCN results are of several % higher for all the cores.

Considering these tendencies, it is concluded that the JNDC group constants may over-estimate the capture. This might be caused by neglecting the Porter-Thomas fluctuation in the evaluation.

9. Energy release from decay of fission products

The energy release rates of fission products have been calculated by summation of the contributions of respective fission product nuclides for several fission nuclides and excitation energies at JAERI. An attempt is made to refine the existing values of beta or gamma energy release rates for short time after fission by including more informations of fission products, mainly short-lived ones. In the calculation 506 radioactive and 125 stable nuclides are considered. The unknown nuclear data for short-lived nuclides are estimated theoretically or statistically. The Q values are obtained by using the semi-empirical mass formula of Myers and Swiatechi. Beta decay constant λ of the nucleus is derived from its Q value by using the empirical correlation between λ and Q.

Feasibility of the method is evaluated through comparison of the calculated results with experiment for the thermal neutron fission of ^{235}U. The results are in good agreement with the experimental ones of the gamma energy release rates for short time after fission. The calculated decay powers are in good agreement with the calorimetric measurements by Lott et al.. The present results of decay power also agree well with the compilations by Shure and by Stehn and Clancy for the respective cooling time.

10. Integral experiment on a lithium metal assembly

A spherical lithium metal system was assemblied by piling up lithium blocks for investigation of neutronics of the fusion reactor blanket at JAERI. The effective radius of the system is 34.1 cm and the volume fraction of Li/sus/void is 0.74/0.19/0.07. The source neutrons are generated at the center through the D-T reaction by a 300 kv Cockcroft-Walton type accelerator. Measurements of the neutron flux distribution were carried out using micro fission chambers of ^{232}Th, ^{238}U, ^{237}Np and ^{235}U. A ^{6}LiI scintillation counter was used for the measurement of the penetration of 14 MeV neutrons.

The analysis was made using the one dimensional transport code ANISN with the approximation of P_5–S_8. The group constants of 42 group were derived from ENDF/B-III through SUPERTOG. It is found that the neutron flux distributions measured by the threshold fission chambers and the scintillation counter are well represented by the calculation. Therefore, it can be said that the spatial distribution of $^{7}Li(n, n')\alpha, T$ reaction may be properly estimated if the cross section is correctly evaluated.

Reactor Physics Activities in Norway - June 1973 - May 1974

J.O. Berg, S. Børresen

The development and evaluation of the modular code system FMS (Fuel Management System) for light water reactors was continued. The system comprises advanced computer programs for fuel assembly burnup calculations, 3-D core calculations and fuel cycle survey and optimization.

A new data library, based essentially on the ENDF/B-III file, is now being used for reactor physics calculations with the fuel assembly program RECORD. A large number of UO_2/H_2O critical experiments were analysed using the new data library. The calculated average of k_{eff} was 1.0008. Results are shown in Fig. 1. Analysis of Pu/U-criticals, involving both clean lattices and typical heterogeneities (Ref. 1) is under way.

A substantial effort has been put on the development of a reactor physics model for Gd-containing fuel. Some results have already been published (ref. 2). The method is based on a modified version of THERMOS, which produces effective cross sections for the Gd-cell and calculates the Gd depletion. The cross sections are then fed into RECORD for calculation of the fuel assembly power distribution and reactivity versus burnup.

A method for 3-D core power distribution estimation based on PRESTO (Ref. 3) is under development. Incore detector signals may thus be used to correct the calculated power distribution when the program is used as a core follow monitor.

The present version of PRESTO will handle problems of up to 13200 nodes corresponding to a full core representation of a BWR with 550 fuel assemblies (800 MWe). Previous calculations with PRESTO have given good agreement with measured power distributions in the Dodewaard reactor. Comparison with experimental results from larger power reactors are under way.

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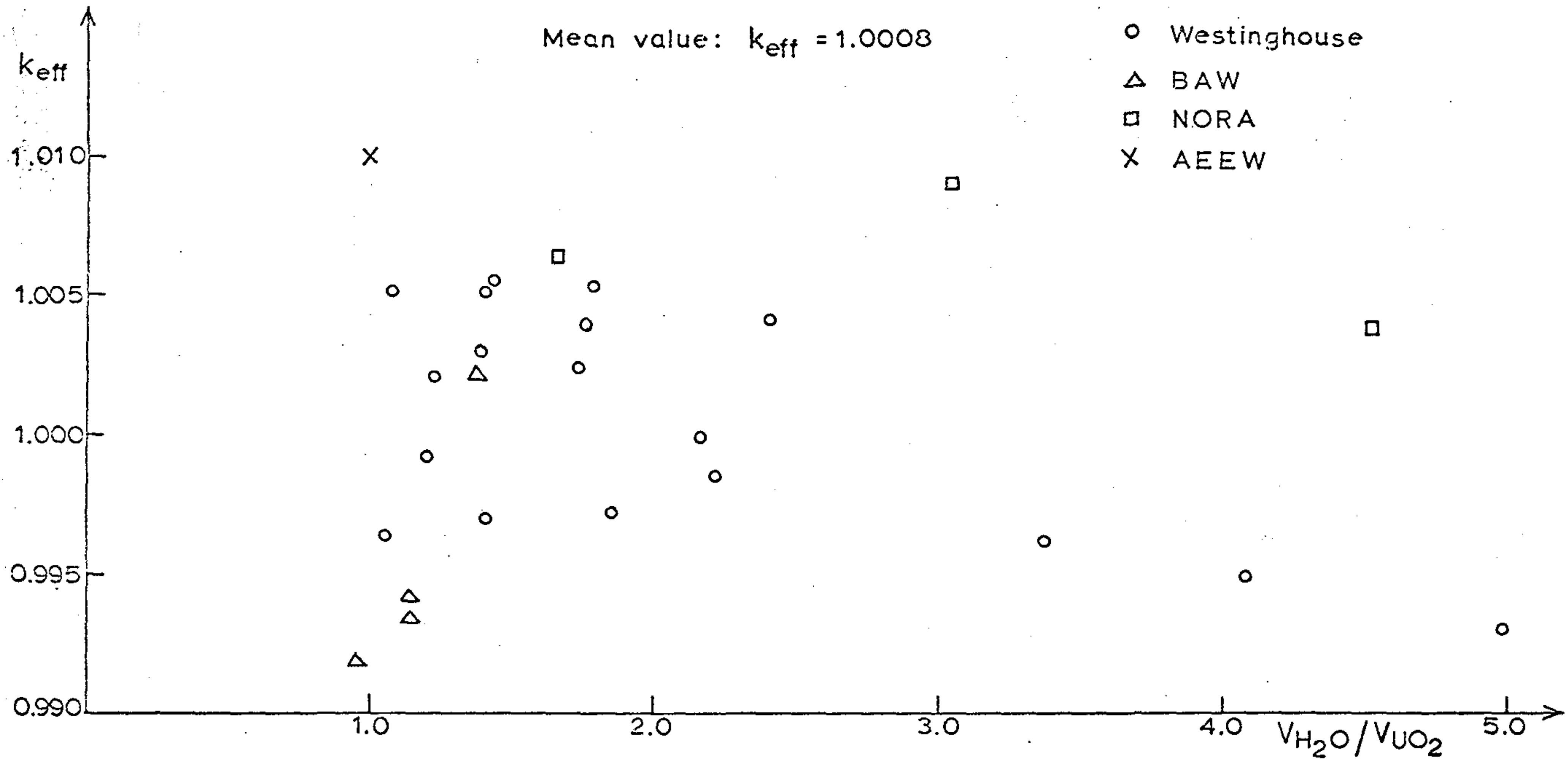

Calculated k_{eff} values for critical lattices at 20°C
MD1 results using 5-group data from RECORD Code Version 74-1

NO - 3

REACTOR PHYSICS ACTIVITIES IN PORTUGAL

J.COSTA OLIVEIRA

1. Research Reactor

A replacement core for the 1 MW swimming-pool research reactor RPI was ordered and will probably by operational by Summer 1974. The new fuel elements are MTR type with 18 straight plates and 93% enriched uranium .

From June 11, 1973 to November 15, 1973 the reactor was operated in section II of the pool in order to refund the pool impervious lining in the vicinity of one of the beam tubes (one of the square ceramic tiles was loose and had to be replaced and fixed with araldite cement) .

2. Thermal Neutron Diffusion Parameters

Thermal column could not be used during the refitting of the reactor pool lining impairing the experimental program in this field .

Some measurements of thermal neutron diffusion parameters in cadmium-poisoned aqueous solutions at room temperature have been made.

Work in progress concerns this kind of studies and also the measurement of thermal neutron diffusion parameters in moderator media at temperatures between 20°C and 100°C .

3. Reactor Noise

The auto-power spectral density function was deduced at zero-power from time-correlation analysis of the neutron population .

The reactor noise studies are now being directed to power reactor applications, aimed at the development of on-line failure detection and diagnosis methods for power reactors. Some experiences are planned now and partly being performed on neutron, temperature and coolant flow noise analysis .

4. Reactor Radiometry

Besides work performed in routine for irradiation experiments carried out in the RPI, some work was accomplished on improvement of interpretation of fast flux measurements using threshold detectors and of gamma dose measurements by calorimetric techniques .

5. Burn-up

Methods of non-destructive assay of the isotopic composition of spent fuel elements are under investigation .

Reactor Physics Activities in Spain

R. Ortiz Fornaguera

Experimental fast Reactor CORAL-I.

In the field of the fast neutron spectroscopy techniques, two proton recoil proportional counters filled with hydrogen and methane have been used to measure the scape neutron spectrum of the reactor. The results agreed with theoretical calculation performed using multigroup methods.

Using the solid state track recorder method, the absolute efficiency, ε, of the counters used in the count-to-count interval distribution technique has been measured. As we already knew the values of ε/β_{eff} through the count-to-count method, the value of β_{eff} has been evaluated. The result, $\beta_{eff} = (6.63+0.17)x10^{-3}$ agees with the calculated one using perturbation techniques, within the corresponding errors.

JEN-1 reactor.

A new logic unit for reactor safety has been developed and will be put in operation in the next months. With this unit the previous reactor instru mentation has been up to date modified.

JEN-2 reactor.

A new method has been developed to adjust the experimental probabilities $p_k(\tau)$ of obtaining k counts in a detector located in the center of the reactor. The experimental values have been adjusted by means of a generating probability function. These experimental probabilities $p_k(\tau)$ have been fitted for each one of the τ intervals of time. By this method a negative reactivity measurement can be completed in few minutes. The MINUS and MIGRAD codes have been used.

Reactor design and fuel management activities during 1973.

A fuel management study of the reload cycles for the 160 MWe Zorita PWR reactor, was conducted in assistance to the utility. Neutronic calculations using different codes and methods were performed for assesment of the design of a swimming-pool research reactor for Chile. Criticality calculations of different transportation configurations of fresh LWR's fuel were performed for safety purposes. Effects on neutronics characteristics of discharge burnup increase in the Vandellos GCR reactor were analyzed.

A number of computer codes for neutronics, thermalhydraulics and transient analysis, adquired through the CPL of NEA-OCDE at Ispra and from other sources, were implemented and tested on the UNIVAC-1106 computer of the JEN. Different codes and utility routines were developed and programmed for providing linkage between different codes and improving calculational gaps in different areas. Studies of different reactor types were performed, providing benchamark of the different codes and methods and experience in its use.

Reactor Physics Activities in Sweden, July 1973 - May 1974

By E Hellstrand

Theory

Thermal reactors

The MICBURN-BUXY code system for calculations on BA-assemblies has been verified against measurements on a fuel assembly which was irradiated in the Oskarshamn BWR. The burnup of individual fuel rods was determined by gamma-scanning and by use of the Nd148 method, and the isotopic composition of gadolinium in the BA-rods was measured. The predicted burnup distribution and gadolinium depletion agree with the measurements within the experimental uncertainties.

MICBURN has also been used for calculation of the microscopic burnup within fuel rods containing unpoisoned UO_2 or UO_2-PuO_2. This investigation shows that in ordinary fuel rods the average number densities of heavy nuclides and fission products is well predicted by use of a homogeneous treatment of the depletion.

Some new options have been added to the fuel assembly burnup code BUXY. A modification of the method for burnup calculations has made it possible to extend the burnup steps between spectrum calculations considerably, especially for fuel assemblies containing burnable absorbers. Typical step lengths are now 0.25-1 MWd/kg U for BA-assemblies.

A two-dimensional cell code, COXY, for calculation of neutron fluxes in xy-geometry has been developed. It is based upon the so-called transmission probability method. At the boundaries of the meshes, the fluxes are expressed by the double P_1-approximation. The neutron currents obtained from them are used as sources for calculating the transmission of neutron through each mesh. The angular dependence of the currents is given by three components for each of the outgoing and the incoming direction. The contribution to the outgoing current from the inner source, which is allowed to vary linearly within the mesh, is given by escape probabilities. The code has been tested against programmes based on collision probability and S_4 approximations. The calculations have been made on BWR boxes with and without a control rod, and also with rods containing uniformly distributed burnable absorbers. The size of the mesh grid can be at least a factor two larger in COXY than in the other two methods for the same accuracy. The time for computing transmission and escape probabilities is less than a tenth of the total computing time. The code is therefore an order of magnitude faster than the collision probability code used for comparisons.

A 69-group cross section library for thermal reactor calculations has been produced from ENDF/B-III. The SPENG library for fast reactor calculations has been used for obtaining fast and epithermal data. An improved version of the FLANGE-2 code has been used for obtaining thermal data. In the original FLANGE-2, the scattering matrix is calculated from point-wise data. In the Studsvik version, integration over primary and secondary groups is made using an appropriate weighting spectrum. The present library contains 57 nuclides.

Large-mesh (nodal) methods are generally used to calculate reactor core power distributions. Work has been started to create a solid theoretical foundation for large-mesh methods and concurrently to increase the accuracy of the methods. It has been possible to formulate and interprete a theory that largely preserves the relatively small amount of calculational work typical for earlier nodal schemes. The new methods will be well suited to experimental comparisons with results from small zero-power reactors as well as large reactors. The work is continuing.

Fast reactors

Integral data calculations have been made on nine uranium-loaded critical systems and on ten plutonium-loaded systems. The neutron data used have been processed from ENDF/B-III and are given in the SPENG library. During the processing, some discrepancies in group cross sections have been found when comparing with data obtained from MC^2. Some of these discrepancies have been attributed to an error in MC^2, but important discrepancies still remain unexplained. In the integral data, large deviations of k_{eff} from unity have been obtained for ZPR-III/32, ZPR-III/55 and ZEBRA/8C.

Breeding performance and fuel costs have been studied for gas-cooled and sodium-cooled breeders on a consistent basis using the fuel-cycle code EQUICYCLE. Fuel cost parameters established through this code is further used as input to long range strategy calculations regarding the role of fast breeders in the future power system in Sweden.

Experiments

Critical experiments

The analysis of the 1972-73 high temperature critical experiments in the KRITZ facility on lattices with mixed oxide fuel containing 1.5 % Pu have been completed. There are no large discrepancies between theory and experiments though the calculated values of reactivity and temperature coefficient seem to be systematically more negative than the experimental ones.

High temperature critical measurements on fuel with two different enrichments (about 2 and 3 %) fissile Pu were performed during the summar 1973 for reactor vendors outside Sweden. Analysis of these experiments have not yet been completed.

Negotiations about further KRITZ experiments on Pu-bearing fuel are going on. The measurements are planned for the autumn of this year.

Fission product activities

Residual power from fission products

The time behaviour of the residual power from fission product in reactor fuel shortly after a power shut-down is not well known. We are therefore planning to perform calorimetric measurements on small samples. The intention is to study the time period below 100 seconds after shut-down.

Calculations

The fission product activity program BEGAFIP utilizes two methods to calculate the total beta and gamma powers. At short decay times tabelled experimental values of beta and gamma production at different times after fission, integrated over the operating time, gives the desired values at a certain decay time. The second method calculates the number of atoms of each fission product by means of the Bateman equation whereupon the total beta and gamma powers are given by a simple summation. A combination of the two methods is necessary as the gamma spectra of the nuclides with short half-lives are not well known. The data library for the second method contains gamma radiation energies and intensities for each nuclide which means that in addition to the total gamma power also the fission product gamma radiation spectrum may be calculated.

BEGAFIP also yields the concentrations of various transuranium isotopes. Comparisons between results from BEGAFIP calculations and those based on the BUXY program are under way.

Reactor Physics Activities in Switzerland

June 1973 - May 1974

R. Richmond

1. Power Reactors

The present situation is summarized in Table 1.
The currently operating nuclear stations represent about 10 % of the total national generating capacity. The next four stations are expected to come into operation during 1978-79 at which stage the nuclear capacity will be about 40 % of total

Table 1. Power Reactors in Switzerland

Station	Reactor Type	Generating Capacity (MWe)	Present State
Beznau Mühleberg	PWR(2) BWR	700 300	In operation
Gösgen	PWR	900	Construction permit granted
Leibstadt Kaiseraugst Graben	BWR BWR BWR	900 900 900	Construction permits pending
Verbois	-	-	Site approved

2. Experimental Measurements

The results of measurements of neutron spectrum and reaction rates made in a typical gas-cooled fast reactor lattice in the first core of PROTEUS (1) have been used to check the predictions of a number of data sets. This was a collaborative exercise in which calculations of the PROTEUS lattice were made by several organizations. The results are summarized in Table 2.

Table 2. C/E Values for Reaction Rate Ratios in PROTEUS Core 1

Ratio	Experimental Value	C/E				
		ENDF/B-3		ASEA-ATOM	KFKINR GBRA	FGL4 EIR
		GGA	Studsvik			
σ_{c8}/σ_{f9}	0.1320±0.0022	1.042	1.067	1.028	1.016	0.975
σ_{f8}/σ_{f9}	0.0322±0.0005	0.957	0.971	0.983	1.007	0.965
σ_{f5}/σ_{f9}	1.022 ±0.015	1.014	1.026	0.989	1.005	1.015

The most accurate predictions are given by KFKINR which is the set most closely adjusted to the results of integral experiments. This, however, is not in line with the normal performance of the set which tends to overpredict the ratio σ_{c8}/σ_{f9} by about 7 %. The most significant error is the overprediction of σ_{c8}/σ_{f9} by ENDF/B-3. This appears to result from a high value of the U238 capture cross-section in this set. It is understood that this cross-section will be lowered for ENDF/B-4. The possible differences arising at the data processing stage are indicated by the 2.4 % discrepancy between the values of σ_{c8}/σ_{f9} given by GGA (GGC5) and Studsvik (SPENG) respectively.

The C/E values for the neutron spectrum are given group-wise in Figure 1 for the energy region covered by the spectrum measurements. The KFKINR values are omitted because of the different group structure. The ASEA-ATOM set gives the worst agreement with experiment but its values tend to oscillate between groups so that their effects largely cancel. The remaining three calculated values lie within 5 % of the experimental values except for the highest energy group where all three underpredict the measured value. Differences between the GGA and Studsvik values occur in the lowest energy group. They appear to be linked to the presence of the major iron resonances and therefore to relate to the resonance treatment. This is being further investigated.

In the second core of PROTEUS the fast lattice was constructed by removing one third of the PuO_2/UO_2 rods from the Core 1 lattice and replacing them by rods of 18/8 stainless steel. This raised the steel volume fraction from 11 % to 27 %. The results of the reaction rate ratio measurements and the C/E values given by FGL4 are shown in Table 3. The only significant difference from the Core 1 C/E values occurs in the case of σ_{f8}/σ_{f9} where the Core 2 calculated value shows a significantly greater error. This is consistent with the spectrum measurements which were extended in the second Core to 2 MeV and which indicated a C/E value of 0.90 in the energy region above 1.2 MeV. These results imply that the steel inelastic scattering is too high in FGL4. They are in line with the fact that in the version of FGL4 adjusted on the basis of integral measurements, the inelastic cross-section of iron is reduced by 12 %.

Table 3. C/E Values for Reaction Rate Ratios in Core 2

Ratio	Experimental Value	C/E
σ_{c8}/σ_{f9}	0.1440 ±0.0025	0.972
σ_{f8}/σ_{f9}	0.0230 ±0.0004	0.929
σ_{f5}/σ_{f9}	1.062 ±0.016	1.014

3. Calculation Methods

A method has been developed for obtaining multigroup cross-sections from the B_n approximation using a formalism which is similar to that of P_n theory, particularly from the programming point of view (2). The new method requires less memory and computing time than the normal B_n approach although it is mathematically equivalent.

Theoretical studies of streaming in GCFR lattices have now been completed (3). Comparisons will be made with the predictions of the method of Eisemann and with the results of PROTEUS experiments.

An improvement to the code DOIT involves the use of the perpendicular angular box boundary currents as unknowns in place of the angular fluxes which were used in the original version. The principle of neutron conservation is now fulfilled rigorously in each box even at low Sn approximations. The most recent version is known as DOIT-4.

A linked code system is being developed for lattice calculations on light water reactors. ENDF/B-3 data will be used in conjunction with locally-developed processing codes and the calculations will cover plutonium recycling and the use of gadolinium as a burnable poison. The EIR codes being used in the scheme are SHADOK (for the fine-group calculations for the microcell) and DOIT-4 (for the broad-group, two-dimension calculations for the boxes). A modified version of the FLARE code is being used for the whole reactor calculation.

4. Laser Induced Fission

Recently it has been suggested that it may be possible to compress a small fissionable sphere with a diameter of the order of mm into a highly supercritical assembly with the aid of multiple laser beams or intense beams of relativistic electrons. Neutronic studies at EIR have shown that, provided that laser induced pressures of 10^{12} atmospheres are achieved, the resulting atom densities (250 times normal densities) ensure super-prompt criticality with a k-eff of the order of 1.25 (4). Kinetics studies have shown that the resulting micro-explosion is much shorter than that of a conventional atomic weapon and therefore the efficiency is better. The inertial confinement time (1 ns) is considerably larger than the reciprocal initial Rossi-alpha. The use of a Li^6 D or T-D reflector leads to a large reduction in critical mass and allows fusion reactions to occur. This device constitutes a hydrogen bomb on a micro-scale.

These studies were based on unclassified information, and are entirely concerned with peaceful applications. The proposal would be to burn the pellets in a combustion chamber similar to that proposed for an electrical power plant based on laser-induced fusion.

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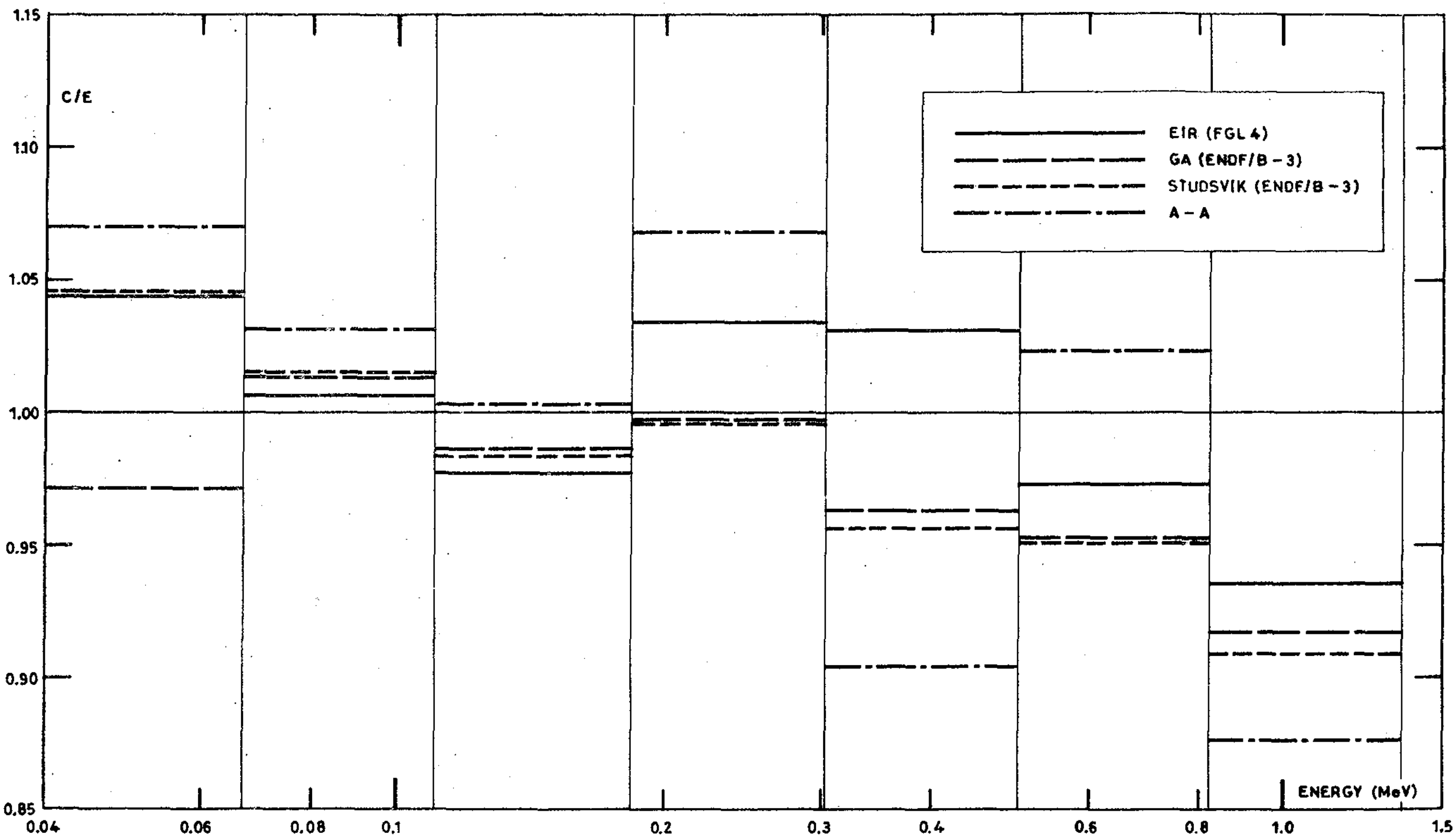


Fig. 1. CALCULATED/EXPERIMENTAL SPECTRUM.

REACTOR PHYSICS IN THE UNITED KINGDOM

C G CAMPBELL
F J FAYERS

GAS COOLED GRAPHITE REACTORS

AGR

1 Fuel loading of the Hinkley 'B' AGR power station commenced on 19 April. Combined engineering tests are in their final stages and power raising is expected in July/August.

2 The performance of the Windscale AGR itself and its fuel have continued to be encouraging. The objective of this programme is to confirm in WAGR the behaviour of fuel designs typical of the first CAGR.

3 The computational route developed by CEGB for storing and processing the data arising from the operation of AGRs has been used in the context of Windscale AGR. Results of reactivity levels, rod worths, axial and radial power distribution are in reasonable accord with experimental observation. This computation system is known as ADOS and contains a central module HET based on heterogeneous source/sink theory. Agreement with parallel computations obtained from the homogenised diffusion theory code SNAP has also been good. The testing of the system on WAGR is providing valuable experience in operation prior to the commercial AGRs going on stream.

HTR

4 Successful operation of the DRAGON reactor has continued. This has followed the replacement of the inner reflector blocks, during which all fuel was removed from the core. The reactor was reloaded and Charge IV/Core 6 operated from July 1973 to September 1973. Charge V/Cores 1 and 2 followed from November 1973 to January 1974 and January 1974 to April 1974 respectively and the third core is being irradiated currently. Charge IV was characterised by irradiation experiments of the "pin and block" design of power reactor, while Charge V sees the gradual introduction of the "integral block" designs now in favour for commercial designs. In addition to these experiments there are standard irradiations to investigate the behaviour of the intrinsic properties of fuel particles, matrix material and structural graphite. In the new inner reflector blocks the profiles between the core and outer reflector blocks have been modified to allow bowing to take place without causing structural interference. This has led to a reduction in Albedo and some significant neutron losses by axial streaming. The net result is a loss in reactivity of about 1% Δk which has to be accommodated in the future core loadings.

5 The work in the UK during the past year has concentrated on control rod aspects. Because the control rod spacing is a fundamental factor in determining core design for an HTR, a small programme of experiments has been mounted in the ZENITH II reactor to study shutdown margins in situations with rods missing in high-reactivity regions of a non-uniform core. Both symmetric and asymmetric cases are being studied, Fig 1 showing a typical configuration. (The letters indicating blocks with centrally inserted control absorbers.) In addition to reactivity measurements, spatial distributions of U-235 reaction rates and reactivity worths of small copper

samples are measured to check predictions of normal and adjoint flux. It is anticipated that these experiments will contribute to a better determination of the components of uncertainty in this configuration, which were given by Johnstone and Scott in AEEW - R 847 (presented at Julich in January 1973), as summarised in Table 1.

6 Further studies on the comparison between low enriched and Thorium cycles for the HTR have been made, including a comparison of physics calculations in collaboration with the CEA. It has been established that physics methods and data make only a small contribution to the differences between UK and US conclusions on this topic. The major difference lies in the assumed cost of fabricating the small fissile particles in the 'two particle' Thorium design, although conservative choices of load factor, in-core control requirements for Xenon override, and different values ascribed to Plutonium all make a contribution to the difference.

7 Three-dimensional fuel management studies of off-load batch refuelled reactor designs using burnable poisons have identified complicated variations of axial form factor over fuel life, especially if control absorbers are allowed to move to flatten the radial power distribution. Some simple explorations of alternative control strategies have been made using the three-dimensional code TRIBUNE in a generalised supercell mode of operation.

8 Collaboration between the Dragon Project and the UKAEA has continued during the year in analysing the zero energy experiments that were performed on the DRAGON reactor prior to Charge IV. The analysis uses UKAEA methods embodied in the WIMS-E scheme of modular codes. Final reactor calculations will be performed shortly to provide detailed power distributions for the experiment which had some control rods operating fully inserted and the remainder withdrawn. A general description of the proposed method of analysis has been given in two papers presented at the 17th Dragon Countries Physics Meeting.

WATER MODERATED REACTORS

SGHWR

9 The Winfrith SGHW Prototype Reactor has just completed its Winter period of base load generation achieving a mean load factor of 98% between November 1973 and April 1974. The fuel performance has been excellent and peak axial burnup levels of 21000MWd/teU have now been obtained. In-pile measurements of fuel pin internal pressures have been completed and good agreement with predictions obtained.

Routine monitoring of the reactivity and rating distributions has continued and comparisons with METHUSELAH-based core follow calculations made which support the conclusions from earlier work reported in AEEW - R 808. Arrangements are in progress to repeat the core follow calculations using data based on the WIMS lattice code.

10 Measurements have been made on the response of the reactor under a revised coupled control (the reactor power is coupled directly to changing grid demand). These measurements have confirmed independent predictions of void and power coefficients of reactivity and have demonstrated the feasibility of this method of control which results in greater grid stability.

11 During the second half of 1973 a short programme of experiments was undertaken in the JUNO reactor at AEE Winfrith. These experiments were intended to augment the information obtained in the comprehensive series which were

completed in the years preceeding 1967. The following topics were tackled in the JUNO programme:-

i Reactivity change associated with loss of coolant from part or all of the core. Here the objective was to provide more accurate experimental data on the change of reactivity with coolant voiding and to complement this by measurement of the change in U-238 capture rate, since this is one of the more difficult contributions to reactivity change.

ii Variation of reactivity with moderator level. In this case two extreme cases were of most interest. Firstly, a check was made on reactivity and flux profile changes associated with a variable D_2O top reflector, and secondly measurements of reactivity at low moderator levels were made to confirm the shutdown margin to be expected at dump level in a large reactor.

iii Use of burnable poisons. The use of part length burnable poison in a number of pins provides a means of increasing the shutdown margin available with a first core loading and of reducing peak linear rating. Measurements have been made to provide experimental backing for the calculational aspects of reactivity changes and power perturbations.

iv Power shaping methods. Current designs for a large reactor envisage the use of grey absorbers for power shaping. Measurements were made in JUNO of the reactivity worth and power perturbation produced by a tube filled with pressurised Krypton, which is one possible form of variable absorber under consideration.

Two experimental configurations were employed in JUNO. In the first, a core of 52 channels surrounded by a D_2O reflector of mean thickness 0.24m was fuelled on a two-zone basis with 36-pin clusters in the inner zone and 74-pin clusters in the outer zone. In the second configuration, 74-pin clusters were used in a uniform loading. Mixed enrichments were used in both cluster types, average enrichments being 2.03% and 1.86% for the two types respectively.

12 Two topics have been selected to illustrate the kind of results obtained from the JUNO experiments. These are "Loss of Coolant" and "Burnable Poison". Measurements were made on both types of JUNO core of the changes in critical moderator level and buckling produced by removal of all coolant. An H_2O/D_2O mixture was used to represent reduced density of water under normal operating conditions. In the uniform core measurements were also made with coolant removed from one half of the core. The WIMS/BASHAN eigenvalues for these cores are summarised in Table 2, which includes both XY calculations in which the measured axial bucklings are used to represent axial leakage, and an XYZ calculation in which axial leakage is calculated directly. In the burnable poison experiment one 36-pin cluster was loaded with 33 pins of the normal type and 3 pins in which a 0.73m long section contained fuel of the normal enrichment loaded with 2.33wt% of gadolinium oxide. The cluster was loaded into a central position in the inner zone of the two zone core, and measurements made of the critical height change and perturbations in the power distributions. A three-dimensional WIMS/BASHAN calculation of the poisoned case gave an eigenvalue almost identical with that of the clean core. The change in cluster power peaking factor from 1.073 to 1.188 was also well predicted by the PIJ option in WIMS. Finally, the perturbation of the macroscopic radial power shape was compared with the three-dimensional BASHAN results across the core centre plane. The largest change occurred for the poisoned channel, whose fission rate was measured to fall by 13.8%, compared with a calculated reduction of 14.8%. The results

are illustrated in Figure 2 for the perturbed core, where the rms deviation between measured and calculated channel powers was ± 1.6%.

13 One of the theoretical aspects of analysing these experiments involves evaluation of the accuracy of leakage calculations. Here a Monte Carlo program W-MONTE has been written and specifically adapted for numerical studies of leakage in SGHWR cluster geometries. This Monte Carlo method is also used as a basis for defining effective diffusion parameters in a core with a dumped moderator. The leakage studies have indicated that there are some deficiencies in the WIMS "three-region" Benoist edit, due to fine structure effects, and rather better results can be obtained from the multi-annular first order option of the code, referred to as the Ariadne method.

LWR

14 The light water reactor lattice code LWR-WIMS has had some extension of its options during the past twelve months. An improved version of the transport module TWOTRAN which uses the centred diamond difference scheme with a negative flux fixing procedure has been incorporated. A collison probability option known as CLUP which can treat explicitly the full heterogeneity of a diagonally symmetric BWR lattice has also been included. The CLUP module is a modified version of the Japanese code CLUP77. The treatment of BWR cruciform absorber rods has been improved by providing an automatic method for smearing the ends of blades which do not terminate conveniently at a calculation mesh line. A system has been established for storing on a random access device, standard input data blocks which can be modified by the user at run time. Associated with this facility is a file output system which transfers standard blocks of data to a library on a random access device for use in whole reactor calculations. Useful user experience with the code has been gained at several establishments on a wide variety of problems. In particular, a study has been made to determine the best approach to be adopted in assessment calculations using the available two-dimensional transport options. The main conclusion of this study is that a diffusion theory solution with suitable choice of pin-cell homogenisation and mesh allocation can give good accuracies for design requirements.

15 JOSHUA III, the three-dimensional fuel management code, coupled with homogenised cross-sections generated from the LWR-WIMS lattice code, has been used to simulate the performance of operating PWRs and BWRs. The prediction of reactivity is good, being within the bounds of ± 0.5% Δk. Calculations of power distribution are in general satisfactory except near the edge of a PWR core. This is believed to be due largely to the strong local absorption in the steel of the core shroud. This is difficult to represent in coarse mesh diffusion theory and alternative boundary treatments are being investigated.

The simulation studies are indicating that the modification to U-235 fission cross-section data suggested by Chawla on the basis of zero energy experiments (Journal of Nuclear Energy, Vol 27, 1973) leads to a small positive reactivity bias. Further work is needed on integral properties of nuclear data.

16 A two-dimensional neutron kinetics code (TUTANK) with two energy groups and up to six delay groups has been written for kinetics studies on water reactors. This code uses the method of alternating buckling sweeps coupled with exponential time transformations. Results have been submitted for the EACRP X-Y geometry LWR benchmark problem.

FAST REACTORS

ZEBRA

17 Summary papers covering all aspects of the MOZART programme were written jointly with PNC, Japan and presented at the Tokyo Symposium in October 1973. The main conclusions of the analysis of the MOZART results using the FGL5 data library are as follows:-

(a) The k-values of both MZA and MZB are closely predicted, the ratio of calculated to experimental values (C/E) being 1.0006 and 1.0010 respectively.

(b) Central reaction-rate ratios are well predicted apart from the ratio of U238 fission to Pu239 fission which is significantly overestimated, by about 3%, in both MZA and MZB.

(c) Calculation underestimates the central spectrum amplitude of MZA both at low energies and at high energies (above 1 MeV). The C/E value falls steadily from unity at 1 KeV to about 0.7 at 100 eV, this region being relevant to the prediction of Doppler coefficient. Coupled with the overprediction of the integral U238 fission/Pu239 fission ratio noted above, the evidence from the high energy spectrum suggests that the U238 fission cross-section data in FGL5 is too high

(d) Satisfactory agreement between calculation and experiment is found for central sample reactivity worths in MZA and MZB. In particular, the use of improved experimental techniques, the adoption of plate geometry throughout these experiments and the recent evaluations of delayed neutron data appear to have removed the 'plutonium-worth discrepancy' in these systems.

(e) The use of more refined techniques (in particular activation foils loaded into fuel plates) has shown up shortcomings in the calculation of reaction-rate distributions and the need for more detailed treatment of the cell calculations at the core/breeder interface. The use of transport theory is also indicated for such situations.

(f) The non-leakage component of Na-void reactivity is well predicted in both plate and pin geometry. The situation on the leakage component in plate geometry is less satisfactory, and the use of anisotropic diffusion coefficients appears necessary. Analysis of Na-void experiments with transport theory should also be undertaken.

(g) Comparison of calculated and experimental worths of modified fuel elements as a function of radial position reveals a spatial dependence which reflects the inadequacy of the diffusion theory prediction of the flux distributions.

(h) The analysis of the extensive control rod studies in MZC shows clearly the inadequacy of the standard diffusion theory treatment for predicting the absolute worths although interaction effects are well calculated. The application of more sophisticated methods leads to very significant improvements, but a trend of increasing C/E with enrichment for boron carbide rods remains and requires further investigation.

(i) Central cavity and fuel compaction situations are calculated to acceptable accuracy by modified diffusion theory. This method does not appear adequate for predicting such events at the core/breeder boundary and further work is needed.

18 Immediately following the completion of MOZART a short series of control rod worth experiments was carried out in the same assembly (MZC = Core 12), using up to four natural boron - carbide absorbers of the PFR/CFR reference design, which are significantly larger than the MONJU rods. The analysis of the extensive integral data on control rods obtained as a result of these Zebra studies has lead to a very significant improvement in the estimated accuracy of prediction, and using the methods now recommended the target accuracies for natural boron carbide absorbers can be achieved.

19 The Core 13 operational mock-up of PFR was made critical in Zebra in late October 1973. The objectives of this phase of the Zebra programme are twofold:-

(a) to provide physics support for PFR during work-up to full power;

(b) to serve as a more general test of calculation methods applied to the non-uniform core loadings that will arise in CFR operation.

The experimental programme includes the following topics:-

(a) reaction rates near central and off-central singularities;

(b) whole-core flux tilts caused by asymmetric loadings and compensation by control rod movement;

(c) breeder reaction rate scans, including the effects of Pu build-up;

(d) sodium void effects in a range of sub-assembly types and positions;

(e) material reactivity worths relevant to calibration of the reactivity scale (Pu239) and to special PFR sub-assemblies and experiments;

(f) control and shut-off rod interactions;

(g) worth of a mock-up Eu_2O_3 control rod (relative to B_4C);

(h) burn-up of B10 in a withdrawn shut-off rod;

(i) gamma-ray energy deposition relevant to rig heating.

20 About two-thirds of the scheduled experimental programme on Core 13 has been completed, the main items yet to be studied being energy deposition in in-core rigs, neutron streaming in control rod guide tubes, and breeder reaction rates including the influence of plutonium build-up.

21 The multi-chamber scanning system has been brought into full operation for the Core 13 programme. This comprises an installed array of inter-calibrated Pu239 fission chambers of Zebra plate geometry incorporated in fuel elements in the same way as other core components. The chamber outputs are counted sequentially under control of the ARGUS computer. At present a total of 150 chambers are in use, providing 5 axial scan points in 30 fuel element positions distributed over the inner and outer core. Adequate counting statistics are obtained in one hour's operation, with full presentation of results a few minutes later.

22 The experimental programme on Core 13 outlined above will be completed by early September. The analysis will, of course, continue beyond this date but should be essentially complete, using the standard methods, by the end of the year.

23 Following the completion of this phase of the Core 13 work, a programme of integral data studies in a simplified version of this assembly will be undertaken. Topics identified for study include absorber-moderator interactions, and measurements of fission products beta and gamma energy release.

DFR

24 The reactor continues to be the main support of the UK fast reactor fuel and materials development programme. The main emphasis now is in generating confidence in the ability to continue operation of the reactor after failures in experimental fuel pins have been detected.

25 The core centre now carries three long sub-assemblies of experimental fuel pins.

26 A series of irradiation rigs, the first of which has just been installed, are concerned with the continuous measurement of irradiation creep.

27 As well as carrying out a very full programme of experimental irradiations the reactor continues to provide electricity both for use at DERE and for export to the public electricity system.

PFR

28 As was reported at the BNES Conference the reactor began operating at low nuclear power on 3 March 1974 and a number of zero energy physics experiments were carried out.

29 Good progress is being made with the next stage of the programme leading to the generation of electricity ie the hot dynamic test of the primary circuit, leak testing of the steam generators prior to filling them with water and the reinstatement of the steam turbine which was originally commissioned last year.

TABLE 1

UNCERTAINTY IN PREDICTING 'LOCAL CRITICALITY' REACTIVITY FOR HTRs

Contribution to Uncertainty	Clean, Cold Start of Life with Burnable Poison	Clean, Cold Reactor at Fuel Cycle Equilibrium
i Worth of uniform control array	± 0.3 N	± 0.3 N
ii Cold core reactivity	± 0.9 N	± 0.7 N
iii Nuclear data	± 0.7 N	± 0.7 N
iv Calculational model	± 1.0 N*	± 1.0 N*
Total estimated uncertainty (one standard deviation)	± 1.6 N	± 1.4 N

*NOTE: In both cases it is anticipated that simulations of realistic 'local criticality' situations in later zero energy experiments will lead to significant reductions in contributions iii and iv, whilst cold experiments with burnable poisons will reduce contribution ii for the start of life situation.

TABLE 2

BASHAN EIGENVALUE CHANGES WITH REMOVAL OF COOLANT IN JUNO EXPERIMENTS

Core	BASHAN Geometry	Coolant			Change of Eigenvalue (Air-Mix)
		H_2O/D_2O Mixture	½ Mix/½ Air	Air	
Two Zone	XY	1.0014	-	1.0032	+ 0.0018
One Zone	XY	1.0012	-	1.0031	+ 0.0019
One Zone	XYZ	0.9991	1.0002	0.9970	- 0.0021

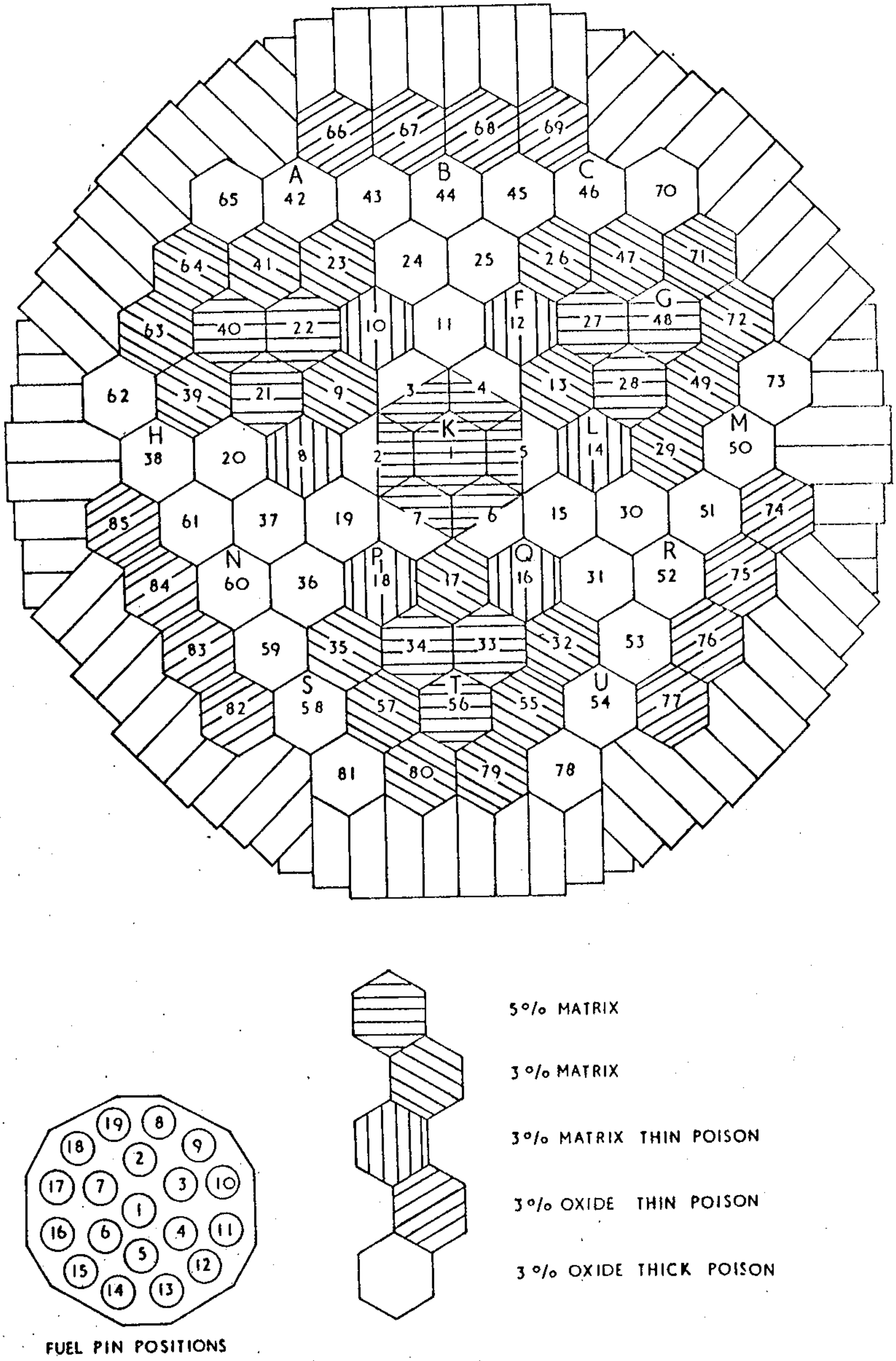
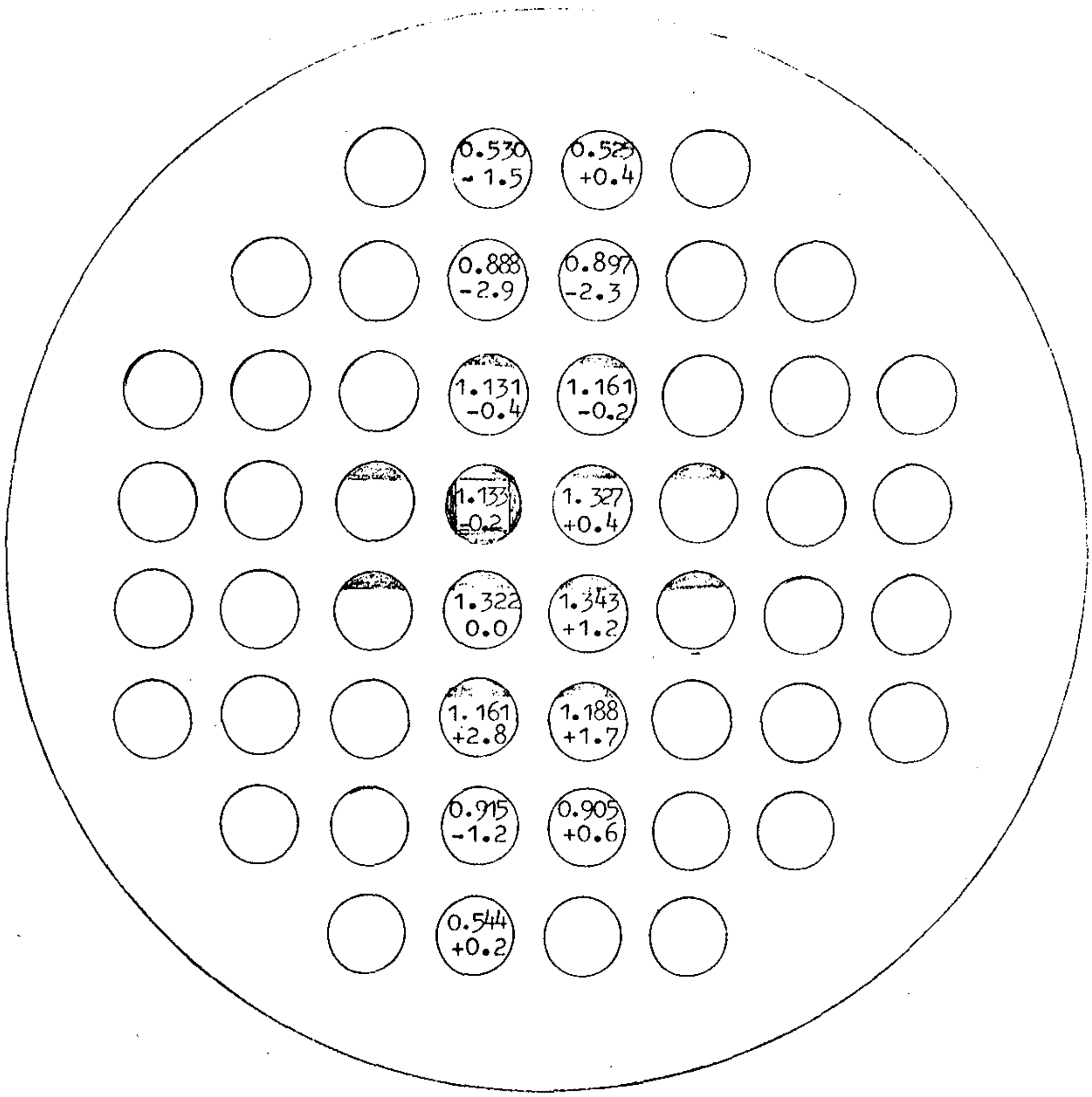


FIG. 1. CORE PLAN ZEN 4/2



74 Pin Clusters

Expt
δ

36 Pin SGHW Clusters

$$\delta = \left(\frac{\text{Calc}}{\text{Expt}} - 1\right) \times 100$$

36 Pin Cluster with 3 Gd_2O_3 Pins

FIG. 2. U_{235} FISSION POWERS IN JUNO WITH BURNABLE POISON IN ONE CHANNEL

Reactor Physics Activities in the U.S.
A Report to the NEACRP
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I. Introduction

Major activities in the preceding year have focused on design support for the Fast Test Reactor (FTR) and the Clinch River Breeder Reactor (CRBR). Of key importance are several critical experiments that have been carried out for both of these projects. Planning was initiated for an additional series of gas cooled breeder critical experiments to begin on ZPR-9 in late calendar year 1974. The Variable Temperature Rodded Zone (VTRZ) assembly was tested at design temperature in ZPR-6; however, priorities have not yet permitted its use for reactor experiments. Planning is underway to expand the ZPPR matrix so it can accommodate 14'X14'X10' assemblies, sufficient for mockups of commercial FBRs. The ENDF/B-IV nuclear data file was completed, several cross section and shielding measurements were carried out, and several new computer programs are in the process of being tested for use on various computer systems and for their performance on various benchmark problems.

II. Critical Experiments

The ZPR-9 has remained committed to the FTR Engineering Mockup Critical program. This program is scheduled for completion by December 1974, at which time, ZPR-9 will be used for a series of benchmark experiments in support of the Gas Cooled Fast Reactor (GCFR).

The VTRZ was checked out at design temperature in the ZPR-6 matrix. The VTRZ and ZPR-6 are currently in standby status. No schedule has been set for VTRZ measurements due to staff commitments on the ZPR-9 program.

The ZPPR Assembly 3 program was completed in the fall of 1973 and Assembly 4 was initiated in December of 1973. The Assembly 4 program is scheduled for completion in January 1975.

FTR Mockup

During the last year, experiments have been conducted to:

a. Determine europia and boron control rod worths

b. Characterize test loop environments, e.g., spectra, material worths

c. Measure heating rates using TLDs in europia and B_4C rod simulations

d. Measure control rod interactions

The final experiment in FTR-EMC will involve measurements in several zones constructed of high 240 content fuel. These experiments will support the design of FTR core 5 and beyond which will be fueled with light water reactor recycled fuel.

The Eu_2O_3 and B_4C comparison studies indicated that, for lightly loaded rods (e.g., composition 1 of Table I), Eu_2O_3 is worth about 10% more than natural B_4C and for a heavily loaded control rod (e.g., composition 4), Eu_2O_3 is worth about 10% less than B_4C. For moderately loaded control rods (e.g., compositions 2 and 3), the Eu_2O_3 to B_4C with ratio is approximately 1. These results may imply smaller losses of control effectiveness due to exposure of europium elements. The gamma heating measurements of europia showed that heating rates in europium across a simulated europium rod are quite flat and that heating in europium and in stainless steel is about 40% higher in the rod than in the core.

ZPPR Programs

The Assembly 3 program on ZPPR was closed out during the summer of 1973 with an extensive series of sodium void measurements.

Before loading Assembly 4, Assembly 3 was modified to more closely model the Clinch River Breeder Reactor (CRBR). Parallel voiding measurements were then performed in the modified assembly.

The current Assembly 4 program is a very close simulation of CRBR and is oriented toward blanket and blanket effects. The program consists of four phases as summarized in Table II.

During Phase 1, measurements were made to determine a breeding ratio, an extensive series of TLD heating rate measurements in core and blankets were conducted, studies of the effects of loading blankets with fissile spikes, and measurements in high 240 zone have been completed. The task of reducing these extensive data is not complete. Currently we are in phase 2 of the measurements program.

III. Shielding

The shielding program at Tower Shielding Facility included the following experiments:

a. FTR Stored Fuel Experiment - In this experiment all materials out to the stored fuel position were mocked up. Fission chamber measurements at stored fuel positions indicated fission rates higher by factors of 2-3 over calculations. The most probable explanation seems to be that flux at stored fuel position was not being properly calculated.

b. Demo Ex-Vessel LLFM Experiment - There were an extensive series of experiments performed in support of the subcritical monitoring system proposed for CRBR. The objective of the experiments was to verify that an ex-vessel system could monitor the subcritical status of CRBR core. The experiments included static attenuation measurement and rod drop measurements. The measurements at TSF indicated that the ex-vessel LLFM approach was feasible.

c. Total cross section checks were made in Pb, Fe, Ni, C, Na and SS and Cr.

d. Experiments were performed on a simulation of a lower ($\sim$2 feet) axial blanket and shield configuration of typical LMFBR. The mockup included a Na filled control rod channel simulation to test streaming calculations. Results from Bonner balls which provide gross integral measurements showed no measurable streaming. Results from Hornyak buttons show high energy streaming of about 40%.

A benchmark program using the TSF is planned to provide measurements in the following areas:

1. Radiation streaming
2. Radiation heating
3. Deep neutron penetration
4. Gamma ray production and transport

IV. <u>Code Development and Methods Activities</u>

Exportability and interchange of codes and code pieces continue to be stressed in new reactor physics code developments. ANS standards on coding practices and documentation are imposed. These have been supplemented by less widespread standardization activities. During the past year, standard interfaces between code pieces have been expanded to include γ production data, linkage between kinetics and safety codes and linkage between fuel management and fuel cost codes. A standard card format has been defined and a standard input processor prepared. These pieces, along with standard input-output and data transfer subroutines, are currently being tested. Complete updated documentation as of July 1973 on coding standards work is reported in LA-5486-MS

Multigroup neutron cross section processing code development is continuing on two levels. A first version of the MC^2-2 ($\sim$2000 groups) code is expected to be available for limited testing this year and distributed to the Argonne Code Center early next year. A second version will address problems such as selective use of integral resonance treatments, higher order scattering matrices, hydrogen treatments and spatial cross section effects. A second processing code, MINX ($\sim$250 groups) with Bondarenko type interpolation capability was completed this spring. A second improved version will be available next year. Both codes are expected to process ENDF/B IV cross sections in a recently adopted standard 240 group structure. The standard structure is designed to be compatible with widely used structures, i.e., currently used structures are subsets of the standard library. Additional groups were then superimposed where necessary to address known physics phenomena.

An interfaced version of a new multidimensional diffusion theory code, VENTURE, was completed. The code was developed principally to treat large three-dimensional problems. The code is group independent and essentially has "unlimited" mesh point capability. The first version allows directionally dependent diffusion coefficients and optionally solves the consistent P-1 equations (no transport approximation). A second version will emphasize numerical improvements to reduce running time. Trial implementation of the finite element method will be considered to further reduce running time and expand the geometrical capability.

A version of the synthesis code (SYN3D) with single channel and group collapse capability (3D, XYZ only) is presently being used by ANL to analyze ZPPR experiments. Future plans include the expansion of the geometrical limitation and the addition of multichannel synthesis capability.

A discrete ordinates two-dimensional transport code, TRIPLET, with triangular mesh capability was developed. This code employs a finite element approach in the spatial variables. A general triangular code stressing improved numerics and removal of ray effects is under development.

A new version of the DOT series of codes capable of treating large shielding problems on large or small machines is also under development.

The two-dimensional quasistatic kinetics code, FX-2, has been linked to the hydrodynamics code VENUS. The package was successfully tested for exportability and will be released to the Argonne Code Center this fall. Current efforts are directed toward trial linkage with SAS2A.

The REBUS fuel management code was expanded to include non-equilibrium capability. Subsequent efforts will stress flux and cross section interpolation and linkage to the VENTURE code (0, 1, 2 and 3D).

Future efforts will be directed toward providing a small sample capability and cross section generation.

Additional code work has been initiated this year in the following areas:

1. Decay heat calculations
2. Coupled n-γ library generation
3. Sensitivity analysis

An expanded benchmark program directed at code validation and testing is also expected to be initiated this year.

IV. Other Experiments

Blanket Studies

Fast reactor blanket investigations are continuing at the MIT Blanket Test Facility. A thermal column adjacent to a heavy water moderated test reactor is used to drive a rectangular converter zone of slightly enriched uranium fuel. Neutrons from fission in the converter are available for tests in a 6'X6'X6' cavity wiich will accept blanket-simulating assemblies.

Current topics of studies include a simulation of a demonstration reactor blanket section driven by a converter spectrum resembling that of leakage from a demonstration core, evaluating thorium blanket loadings, and graphite blanket reflection. Large discrepancies between calculated and measured ^{238}U fission rates near blanket-reflector interfaces persist. Forthcoming work will include refined energy group structures near the fission threshold, examination of possible subthreshold fission effects, and investigation of any spurious contributions from detector contamination (e.g., ^{235}U in ^{238}U).

Emphasis is being placed upon obtaining comparative measurements and forming relative assessments of TLD and miniature ion chambers for gamma heating measurements, obtaining spectra over the entire fast reactor energy range, and detailed reaction rate traverses.

Sodium Assembly Studies

Extensive time of flight measurements of position and angle dependent neutrons have been completed at RPI on a homogeneous sodium assembly. Agreement between measurements and calculations was generally good using DTF IV, 49 group ENDF/B-III derived cross sections.

V. Nuclear Data

Recent cross section measurements at RPI include capture cross sections of Pd-105, Eu-151, Eu-153 and Pu-242. ORNL has completed extensive studies of the U-238 capture and U-235 fission cross sections. Intelcom Rad Tech has completed studies of the (n α) reactions in Li^6 and B-10. The ANL FNG has provided several neutron scattering measurements up to 4 MeV for Ni, Mo, Nb, Zr and Co; fission ratios of U-238/U-235 to 2.0 MeV; delayed neutron yields of U-235, U-238 and Pu-239; and (n,p) cross sections of Co, Ni, Fe, Ti and Zr to 6 MeV.

The ENDF/B-IV data file is in the final stages of checking and processing. It includes a general purpose file of structural, control, moderator and fuel materials; a dosimetry file; and a fission product file. There are cross sections for 90 materials in the general purpose file, 36 reactions in the dosimetry file and data for 823 isotopes in the fission product file. Preliminary testing indicates that ENDF-IV will provide improved predictions over ENDF/B-III for thermal reactor, shielding, and dosimetry applications and comparable results for fast reactor applications.

TABLE I. Control Material Atom Densities (at/cc × 10^{-21})

	Eu_2O_3 Compositions				B_4C Compositions			
	#1	#2	#3	#4	#1	#2	#3	#4
^{151}Eu	0.4511	0.9022	1.7993	3.6010				
^{153}Eu	0.4862	0.9724	1.9392	3.8810				
O	1.4060	2.8119	5.6075	11.2228	0.0008	0.0015	0.0031	0.0061
Na	17.2426	15.9569	12.3632	8.2421	17.2426	15.9569	12.3632	8.2421
^{10}B					0.5786	1.1494	2.3079	4.5826
^{11}B					2.3780	4.7240	9.4852	18.8338
Fe	10.9398	11.3193	13.7504	13.5933	10.8920	11.2226	13.5550	13.2067
Ni	1.4544	1.5104	1.8690	1.8459	1.4472	1.4960	1.8400	1.7884
Cr	3.1274	3.2367	3.9367	3.8914	3.1135	3.2086	3.8800	3.7793
Mn	0.2406	0.2487	0.3010	0.2976	0.2396	0.2467	0.2968	0.2892
Mo	0.0099	0.0099	0.0099	0.0099	0.0099	0.0099	0.0099	0.0099
C	0.0316	0.0323	0.0362	0.0360	0.8063	1.5710	3.1258	6.1705
Si	0.1977	0.2045	0.2482	0.2454	0.1969	0.2028	0.2447	0.2384
Cu	0.0212	0.0218	0.0255	0.0253	0.0211	0.0216	0.0252	0.0247
S	0.0021	0.0021	0.0027	0.0027	0.0020	0.0021	0.0027	0.0026
P	0.0056	0.0057	0.0067	0.0066	0.0056	0.0057	0.0066	0.0065
Co	0.0092	0.0093	0.0104	0.0103	0.0091	0.0093	0.0103	0.0101
Aℓ	0.0062	0.0068	0.0103	0.0101	0.0061	0.0066	0.0100	0.0095
H					0.0123	0.0245	0.0491	0.0976
N					0.0027	0.0053	0.0106	0.0211

Notes: Isotopic compoisition of Eu_2O_3: ^{151}Eu (41.2805) w/o, ^{153}Eu (45.0804) w/o, O (13.6390) w/o.

Isotopic composition of B_4C: ^{10}B(13.9975)w/o, ^{11}B(63.2525)w/o, C(22.48)w/o.

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TABLE II. ZPPR-4 - Schedule of Assemblies and Program of Measurements

Phase	Step	Core	Time in Cycle	Control Rods: Type	Control Rods: Configuration[a]	Radial Blanket Loading	Special Zone	Experimental Program: f^9,f^5,f^8,c^8 Mapping	γ Heating (TLD) Mapping	Control Rod Worths	Breeding Ratio	Doppler Na Void	Leakage Spectra	Control Rod Tip Heating	Calorimetry	Blanket Spiking Feasibility	Normalization Measurements	Comments
1	a	Initial	EOC	None	-	DUO_2	No	X	X	X	X					X	X	Reference case. Experiments included detailed mappings for breeding ratio measurement.
2	a	Initial	BOC	Nat'l.	(1)	DUO_2	H240	X	X	X		X X					X	Comparison with 2b gives the effects of H240 fuel.
	b	Initial	BOC	Nat'l.	(1)	DUO_2	No	X	X	X		X X	X				X	Reference case. Comparison with 1a give changes from BOC to EOC core.
	c	Initial	BOC	50% Enr.	(2)	DUO_2	No	X	X	X		X		X	X		X	Reference case. Comparison with 2b gives the effects of CR change. Control rod tip heating and calorimetry developmental measurements included in this step.
3	a	Equil.	MOC	Nat'l.	(1)	DUO_2/PuO_2	No	X	X	X	X	X X	X				X	Comparison with 2b gives the effects of blanket buildup.
	b	Equil.	MOC	50% Enr.	(2)	DUO_2/PuO_2	No	X	X	X		X					X	Comparison with 2c gives the effects of blanket buildup.
4	a	Equil.	EOC	None	-	DUO_2/PuO_2	No	X	X	X	X						X	Comparison with 3b gives the effects of composition plus CR change. Comparison with 1a gives the effects of blanket buildup.
	b	Equil.	EOC	None	-	DUO_2/PuO_2 Nonuniform Distribution	No	X	X	X							X	Comparison with 4a gives effects of blanket fissile distribution.

[a](1) - (1 natural central plus 6 outer ring flat natural rods) - 100% inserted.
(2) - (1 natural central plus 6 outer ring flat 50% enriched rods) - 50% inserted.