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NEA COMMITTEE ON REACTOR PHYSICS

REACTOR PHYSICS ACTIVITIES IN NEA MEMBER COUNTRIES

October 1982-September 1983

NEACRP-L-265

OECD NUCLEAR ENERGY AGENCY
38 boulevard Suchet 75016 Paris

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REACTOR PHYSICS ACTIVITIES IN NEA MEMBER COUNTRIES

This document is a compilation of national activity reports presented at the Twenty-Sixth Meeting of the NEA Committee on Reactor Physics, held at Oak Ridge National Laboratory, Tennessee, U.S.A., from 17th to 21st October 1983.

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AUSTRALIA

REACTOR PHYSICS ACTIVITIES IN AUSTRALIA (October 1982 - September 1983)

by

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1. METHODS OF CALCULATION

1.1 AUS Modular Scheme

Only limited further development of AUS has been undertaken this year. The 3-dimensional neutron diffusion module POW-3D has been completed by the inclusion of perturbation routines; a 1-dimensional diffusion module AUS100, suitable for calculations of spatially dependent group-condensation spectra, has been added to the scheme to complement POW-3D; bilinear weighting has been included in the MIRANDA cross-section preparation module.

1.2 Monte Carlo Point Reaction Rate Estimation

The use of next flight estimation of point reaction rates in calculations with the Monte Carlo code MORSE has been investigated with particular reference to a typical fusion blanket experiment of a central source in a metre cube of lithium carbonate.

The method of Iida and Seki[1], which involves importance sampling to determine whether a distant collision should be used, was found to give substantially improved results. The infinite variance problem for collisions very near a detector is eliminated by analytically averaging the contribution for collisions within a certain radius of the detector. It was also found advantageous to select the group into which the neutron was scattered, rather than include the contribution of all possible groups to the next flight estimation.

2. FUSION REACTOR BLANKET NEUTRONICS

2.1 Validation of Group Cross-section Library

Fusion neutronics calculations at the AAEC are based on group cross-section data from AUS.ENDF200G which is a 200 neutron, 37 photon group library derived from ENDF/B IV. To validate this library and the changes to AUS system modules which were required to include photon and kerma factor data, a number of comparisons have been made with published calculations of conceptual fusion reactor blanket designs.

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The STARFIRE study was chosen for the major investigation. The published results include tritium production levels and energy releases for a large number of 1D scoping calculations using ENDF/B IV data. Though reasonable agreement (∿4%) with total energy release results was obtained, an unexpected difference of about 7% was obtained for tritium production.

The comparison was therefore extended to three other studies for which published tritium production results using ENDF/B IV were available, viz. the CTR standard blanket, the European Economic Community blanket concept for INTOR, and DEMO. Agreement to about 1% was obtained for these systems.

This attempt to validate AUS system results by comparison with published design studies is considered less than satisfactory, because no single self-consistent set of results for a comprehensive range of parameters appears to be available. An international benchmark calculation enabling the comparison of many aspects of fusion blanket neutronics, particularly those not amenable to direct experimental verification, would be very useful.

3. RESEARCH REACTOR (HIFAR) NEUTRONICS

The computational models of HIFAR previously developed for use with the AUS scheme have been applied to routine HIFAR reactivity and burn-up calculations in an indirect manner by extracting flux factors and reactivity coefficients for use in a simple fuel and reactivity accounting program, HIFUEL.

The HIFUEL approach is similar to the semi-empirical method used for operational purposes over many years via the HIBURN program[2], but uses calculated instead of measured parameters. This has the advantage of replacing experimentally based data (e.g. flux depression factors, reactivity coefficients) from a number of sources, which are often difficult to reconcile with confidence, with a single self-consistent data set. It also allows comparison of routine and more complex, special-purpose HIFAR calculations in a consistent manner.

HIFUEL calculations were run for a sequence of HIFAR operating programs spanning several years, and the results compared with the corresponding operational data. HIFUEL consistently predicted reactivity changes to better than 0.5% in k_{eff}, and fluxes to better than 10%.

REFERENCES

1. Iida, H. and Seki, Y. (1980). Reduction of computational time for point detector estimator in Monte Carlo transport codes. NSE 74, p213.

2. McCulloch, D.B. and Trimble G.D. (1969). A method of estimating fuel burn-up and higher isotope production in the reactor HIFAR. AAEC/TM508.

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REACTOR PHYSICS ACTIVITIES IN AUSTRIA
October 1982 - September 1983

compiled
by
F. Putz

1. THEORY

1.1 Reactor Computations

An additional shut down system consisting of small boronated shut down spheres and provided for future HTRs with pebble-bed cores has been studied at ITP/TU Graz (Institut für theoretische Physik der Technischen Universität Graz) jointly with the Institute of Reactor Development of KFA Jülich (FRG). For the neutronics analysis a problem dependent 6o group cross section library had been developed. In order to take into account the selfshielding of the shut down spheres it was necessary to develop various cell models and to investigate the influence of the geometry underlying the cell models. Criticality calculations using the code CITATION-2D are in good agreement with experimental results gained from experiments performed on multisphere configurations at the Siemens-Argonaut reactor of TU Graz.

1.2 Transport Theory

At ITP/U Graz (Institut für theoretische Physik der Universität Graz) a study is in progress of the passage of neutron pulses through stratified media using the method of multiple collision probabilities.

1.3 Reactor Kinetics

Based on the theory of the well known TWIGLE-code a reactor kinetics code is being developed at ÖFZS (Österreichisches Forschungszentrum Seibersdorf) including an improved hydraulics model. Application of a coarse mesh rebalancing method leads to a reduction of computer time. An exponential transform reduces the number of time steps, the increments of which are calculated by a predictor-corrector method. The code is provided for the simulation of reactivity transients.

1.4 Nuclear Data

The existing data bank of all available neutron-nuclei scattering lengths has been updated at AI (Atominstitut der Österreichischen Universitäten, Wien) by including the recent values from the literature. Altogether 96o values for the various elements and isotopes have been collected and referred.

2. EXPERIMENTAL STUDIES

2.1 Burn-up Determination

An important activity at AI in the preceeding year was the inter-comparison of non-destructive techniques to determine spent fuel burn-up. Application of ionization and fission chambers has been compared with the use of thermoluminescence dosimeters (TLD) and solid state nuclear track detectors (SSTD). After careful selection of optimal TLDs and SSTDs and of various types of converter foils such as depleted, natural or enriched

uranium these detectors have been exposed in the first stage to spent TRIGA fuel elements and in a second stage to spent fuel elements from the Halden boiling water reactor. It could be shown that TLDs and SSTDs offer some considerable benefits compared to ionization and fission chambers such as easy transportation and handling, simple application, parallel exposure to many fuel assemblies at the same time and low costs. In no way these two detector types may replace chambers but they can be used as additional back-up methods in case that routine measurements show deviations from the expected burn-up value.

2.2 Reactivity Measurement

Measurements of the subcritical reactivity have been carried out at the Siemens-Argonaut reactor of TU Graz using the inverse kinetics method and the source jerk method respectively. The results gained with these two methods agree satisfactorily.

REACTOR PHYSICS ACTIVITIES IN BELGIUM

September 1982 - August 1983

J. DEBRUE (S.C.K./C.E.N., MOL)

FAST REACTORS

The belgian participation (BELGONUCLEAIRE and SCK/CEN) to the RACINE programme in the French zero power reactor MASURCA at CADARACHE concerned mainly the analysis of the gamma heating experiments. The experimental data of CADARACHE, WINFRITH and MOL were normalized on the basis of a common calibration irradiation. There is general good agreement between the TLD results of the three participants and also between TLD and ionization chamber, although no firm conclusion can be drawn before the correction of TLD signals for neutron sensitivity, non-saturation of fission product activity and cavity effect are applied.
The influence of cell heterogeneity has been calculated with UK-DeBeNe methods.
The absolute normalization of the neutronic fluxes necessary to compute absolute gamma doses has been investigated, namely through a campaign of intercalibration of fission chambers at SCK/CEN.

The participation of BELGONUCLEAIRE to the CEA studies on the subcritical approach for the first loading of SUPERPHENIX began with a survey of different possible loading policies, calculated first in 1D and later in 2D geometry. The objective is to assess the reactivity, counting rates and source multiplication correction factors at various stages of the approach. A detailed programme of measurements in MASURCA was prepared (RACINE-R1S) to simulate features of the SUPERPHENIX core.

The development of the computer programme SUPERCAPHE, intended to support the management of the core, blanket and control subassemblies of SUPERPHENIX, was pursued. The tests concerning the reactivity and power distribution calculated with the 3D-version of the module GESTION were carried on.

THERMAL REACTORS

The main reactor physics activities concern the validation of the calculation methods applied in pressure vessel surveillance programmes. The achievement of this objective is based upon systematical comparisons of calculations with measurements performed in research and power reactors, in the frame of a cooperative agreement with US laboratories (HEDL, ORNL, NBS) sponsored by the NRC.
The validation work related to research reactor experiments makes use of measurement data obtained in PCA (Pool Critical Assembly, ORNL), PSF (Pool Side Facility of the Oak Ridge Research reactor ORR) and VENUS (critical facility at SCK/CEN, Mol). The investigated parameters are the neutron propagation from the reactor core up to the vessel wall, the damage exposure in the vessel and the gamma heating in the surveillance capsules and in the vessel wall.
The experiments in PCA and PSF are driven by MTR-type reactor and involve comparatively simple, finite-slab geometry. The reactor configuration built in VENUS is an engineering mock-up which simulates rather closely the main features of a power reactor : low enriched fuel pins, staircase-shaped core boundary, core baffle, barrel (see figure). In this assembly, the azimutal part of the fast neutron flux distribution outside the core is significantly sensitive to power rating details in core boundary fuel pin rows ; the peak-to-peak azimutal variation along the cylindrical barrel is about a factor 2.
The measurements performed in VENUS from January to June 1983 are summarized in Table II. The measurement points are located in a 45° sector. Detailed measurements in the steel pieces (baffle, barrel) were performed with miniaturized detectors. Reference neutron and gamma-ray fields developed at BR1 were used for validation and standardization of the techniques.
The theoretical analysis is made in two steps. Calculation of the power distribution in the core and determination of the fission sources (X-Y geometry). After conversion of the X-Y source distribution in a R-θ distribution, neutron transport calculation (S_8 - P_3) in order to determine reaction rates and neutron spectra at the measurement locations outside the core. A 17 group cross section set, derived from the ORNL 171 group set (VITAMIN-C), is used for these calculations. This work is under progress.

Fast neutron fluence measurements performed around the BR3 reactor core during previous cycles have been complemented by activation analysis of scraps taken from the thermal shield at the mid-plane level. On the other hand, material sampling in

the longitudinal weld of the vessel above the core has been performed to assess the chemical composition of this material ; a coincidence technique was applied to determine the copper content (∿ 0.2 wt %) in the samples after irradiation in BR1. This work is part of a dosimetry-metallurgy programme conducted to better define the embrittlement of the BR3 vessel steel.

FUSION

The application of a delayed neutron counting technique has been investigated and tested in order to measure the neutron flux density at several locations around the JET torus. Fissile targets irradiated at these locations will be measured with ^{3}He counters embedded in a moderator at the terminals of pneumatic rabbits. The irradiation conditions occurring in the (D, D) phase of JET operation (neutron energy of 2.45 MeV) have been simulated at the BR1 reactor ; the sensitivity and different measurement parameters have been studied.
In support to the irradiation of fusion materials in BR2, the adjustment of the (dpa/helium production) ratio in steel is looked for in order to fit better the irradiation conditions of the first wall in a fusion reactor. Irradiation strategy and spectrum tailoring in BR2 are examined, namely through the irradiation of small nickel samples.

Table I

Measured parameter	Technique used (Laboratory)	Number of points	Number of runs
Gamma spectrum	Compton spectrometer "Janus" (HEDL)	2	2
Gamma dose distribution	TLD detectors (Rolls Royce) (CEGB) (SCK/CEN)	16	7
Gamma dose perturbation induced by the Compton spectrometer "Janus"	TLD detectors (SCK/CEN)	1	2
Fast neutron spectrum	Proton recoil counters (4 different types) (SCK/CEN)	3	25
Fast neutron spectrum and perturbations induced by proton recoil counters	Nuclear emulsions (HEDL)	16	9
Axial buckling in different core and reflector regions, for different neutron energy ranges	Miniature fission chambers (NBS) (SCK/CEN)	17	5
Thermal neutron flux depression inside the fuel	Dy activation (SCK/CEN)	2	1
Thermal neutron flux distribution	Dy activation (SCK/CEN)	17	3
Fission rate distribution	^{235}U fission chambers (NBS & SCK/CEN)	43	(*)
Fast neutron flux distribution (fission chambers)	^{237}Np fission chambers ($E_n \geq 0.6$ MeV) (NBS & SCK/CEN)	60	(*)
	^{238}U fission chambers ($E_n \geq 1.5$ MeV) (NBS & SCK/CEN)	39	(*)

Table I (continued)

Measured parameter	Technique used (Laboratory)	Number of points	Number of runs
Fast neutron flux distribution (SSTR = Solid State Track Recorders)	SSTR with ^{237}Np foils $E_n \geq 0.6$ MeV (HEDL)	15	1
	SSTR with $^{238}UO_2$ foils $E_n \geq 1.5$ MeV (HEDL)	18	3
	SSTR with $^{235}UO_2$ foils for correction on $^{238}UO_2$ (HEDL)	18	3
Fast neutron flux distribution (activation)	Rh activation ($E_n \geq 0.8$ MeV) (HEDL & SCK/CEN)	13	7
	In activation ($E_n \geq 1.2$ MeV) (HEDL & SCK/CEN)	30	12
	Zn activation ($E_n \geq 2.8$ MeV) (HEDL & SCK/CEN)	16	3
	Ni activation ($E_n \geq 2.8$ MeV) (HEDL & SCK/CEN)	35	3
	S activation ($E_n \geq 2.9$ MeV) (HEDL & SCK/CEN)	16	3
	Al activation ($E_n \geq 7.2$ MeV) (HEDL & SCK/CEN)	4	1
Axial power distribution	Fuel activation (SCK/CEN)	6	1
Radial power distribution	Fuel activation (SCK/CEN)	132	1
	SSTR inside fuel (HEDL)	6	2

FORM. N° 1.15.2
PLAN N° 51.228

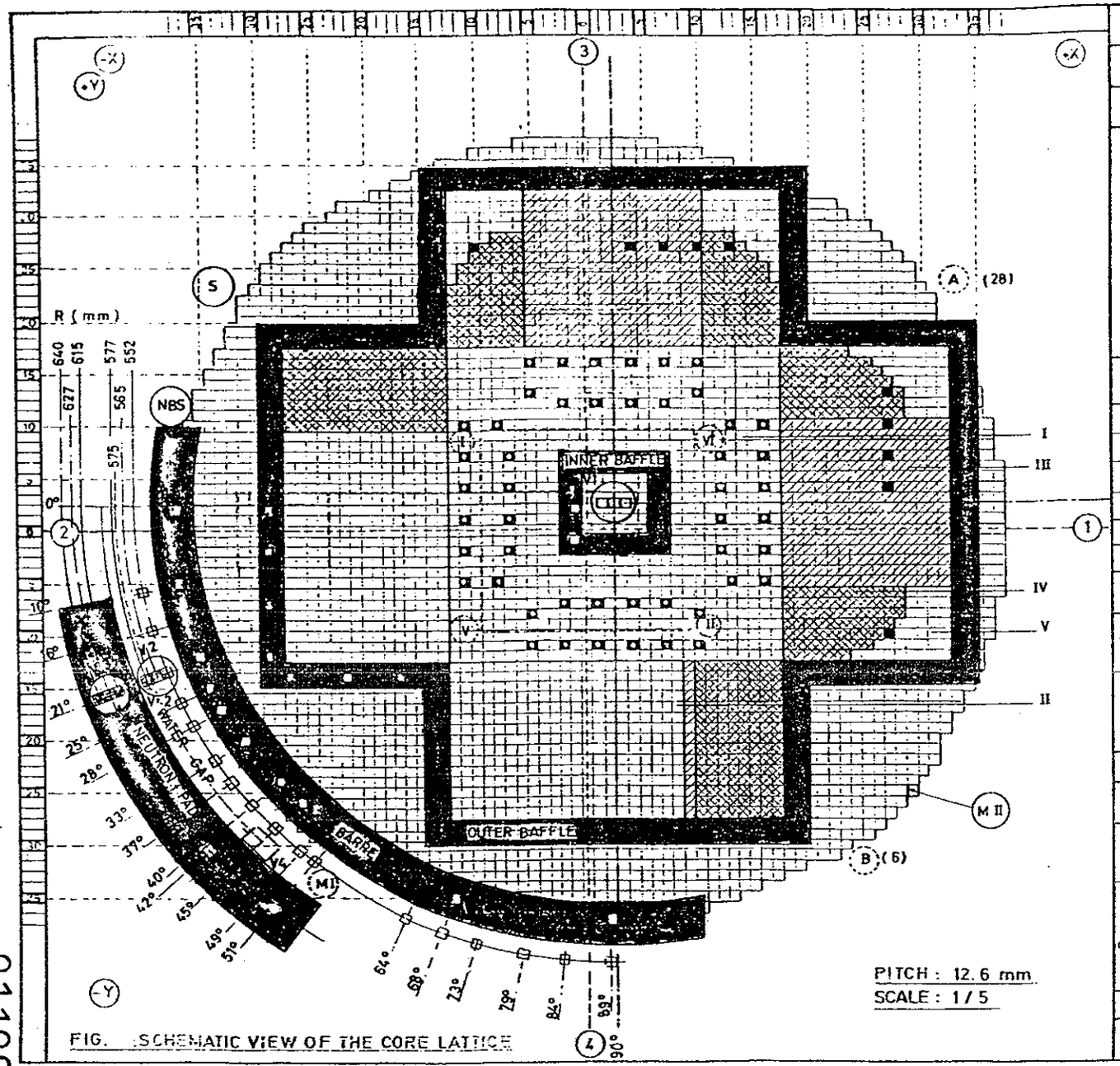

FIG. : SCHEMATIC VIEW OF THE CORE LATTICE

C.E.N-S.C.K | VENUS CRITICAL EXPERIMENT

EXPERIMENT : LWR-PVS - BENCHMARK

DATE : from 24.01.83 to 23.06.83

CONFIGURATION N° 29 / 14

DENSITY RODS : NONE

BORON CONCENTRATION (p.p.m) : NONE

START-UP SOURCE (S) : YES

NEUTRON GENERATOR (N) : NONE

ABSORBING ROD ■ : 8R + 2M

PYREX ROD ◘ : 48

FUEL REGIONS TOTAL : (2600 - 58) pins

DESIGNAT.	%U5/%Pu (1)	COLOUR OR SYMBOL	NUMBER OF FUEL PINS	EQ. RADIUS OR X . Y	INVES-TIGATED CELL
I	3/0		800-48		
II	4/0 (3d)		645		
III	4/0 (1st)		495 - 6		
IV	4/0 (2d)		504 - 2		
V	3/0 (2)		156 - 2		

MONITORING DETECTORS

DESIGNAT.	DIAMET.	SYMBOL	N° CABLE	LOCATION	LEVEL	CHANNEL
PROPORTIONAL BF3 COUNTER						
FISSION CHAMBER	1/8"	M I		R=552	Bottom	C
	1/4"	M II		X=+29 Y=-25	Bottom	D
	1"	NBS				

CONTROL DETECTORS

DESIGNAT.	DIAMET.	SYMBOL	N°	LOCATION	LEVEL	CHANNEL
PROPORTIONAL COUNTER	1"	(A)		28	mid-pl.	A
	1"	(B)		6	mid-pl.	B
ION CHAMBER (NON-COMPENSATED)		(I)		X=-11 Y=+9	under reactor	I
		(II)		X=+11 Y=-9	under reactor	II
		(V)		X=-11 Y=-9	under reactor	V
		(VI)		X=+11 Y=+9	under reactor	VI

REMARK :

VERIFIED BY: APPROVED BY:

REMARKS : (1) Supply order of 4/0 fuel.
(2) Out of specifications (0.25 < Bowing < 0.[illegible]m).

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REACTOR PHYSICS ACTIVITIES IN CANADA

(October 1982 - September 1983)

by

P.M. Garvey

Chalk River Nuclear Laboratories
Chalk River, Ontario, Canada

SUMMARY

Support of the design and operation of the current CANDU PHW reactors continued through the development, validation and application of various computer codes, both by Ontario Hydro (OH) and CANDU Operations of Atomic Energy of Canada Limited (AECL). Initial design studies for a 300 MWe CANDU PMWR were started by AECL.

The reactor physics program within the Research Company of AECL (AECL-RC) is largely associated with the development of Advanced Fuel Cycles for the CANDU PHWR. The major activities within this area were the development and validation of lattice cell codes, assessment of the characteristics of the CANDU PHWR for both the thorium fuel cycle and the LWR spent fuel (the Tandem Fuel) cycle, and system studies showing the impact of such fuel cycles on uranium requirements. An experimental lattice physics program in the ZED-2 zero energy research reactor at CRNL is in hand for $(Pu,U)O_2$ fuels, and $(Pu,Th)O_2$ fuel is being fabricated for a similar program to start in 1984.

Code development and application continued within AECL-RC in support of the research reactors NRX, NRU and WR-1. Further experiments were undertaken in ZED-2 in support of these reactors.

Further measurements were made of the neutron yield of targets irradiated with 100 MeV protons. Measurements in a graphite-thorium assembly irradiated with 14 MeV neutrons were completed and evaluated.

Development by AECL-RC of the low power heating reactor SLOWPOKE-3 continued and the program was expanded to evaluate small high temperature reactors (SHTR) for production of both heat and electricity with a thermal power less than 20 MW.

CANDU PHWR Support

Measurements were made of the reactor physics characteristics of the four 600 MWe reactors (Point Lepreau, Gentilly-2, Wolsung-1 and Cordoba) that were commissioned during this period. Good agreement with calculation was obtained.

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Improvements were made to the interpretation of the in-core self powered flux detector signals in terms of the global flux/power distribution[1,2].

An extension to the model expansion method was initiated that allows more accurate calculations especially close to absorbers[3].

The capability to study the dynamics of the primary heat transport system taking into account both the neutronic-thermohydraulic coupling and the reactor regulating system has been developed and applied to the CANDU PHW system.

Advanced Fuel Cycles

Development and assessment of Advanced Fuel Cycles for the CANDU reactor continued within AECL-RC.

An experimental program of measurements on the characteristics of $(Pu,U)O_2$ fuel was started in ZED-2. Planning is in hand for a similar series of $(Pu,Th)O_2$ lattice experiments that will start in 1984. These experiments are largely of the "substitution" type together with reaction rate measurements. Development of improved methods to interpret these substitution experiments continues.

The NJOY system is being used to prepare ENDF/B-V multigroup data for the lattice cell codes LATREP, SOLACE, WIMS-CRNL and RAHAB/OZMA. Further improvements have been made to these codes and new versions issued[4,5] Several of these codes continue to be tested against thermal test lattices and in general good agreement has been obtained with experiment[6].

The code, FISSPROD, that calculates the fission product concentration, is being updated with cross sections from ENDF/B-V. A new version was issued[7]. A new version of the monopole heterogeneous source-sink code MICRETE was issued[8].

Further development of methods to solve the neutron transport equation continued[9].

Assessment of the core characteristics of thorium fuelled CANDU reactors continued. This work was, however, temporarily suspended to evaluate the core characteristics of a CANDU reactor when operating on the Tandem Fuel Cycle (TFC). In this cycle LWR spent fuel, after removal of fission products and the higher actinides, is refabricated into CANDU fuel bundles after dilution, if required, by either natural or depleted uranium. This is a joint study with the Korean Advanced Energy Research Institute. Similar studies are in progress with EPDC (Japan) on the recycling of the separated uranium from reprocessed LWR fuel.

Research Reactors

A program to develop a suitable low enriched uranium (LEU; 20% U-235 in U) fuel for the NRU and NRX research reactors has been in hand for the past several years. The prime candidate fuel is a U-Si-Al in Al dispersion, as this would allow the current geometry and U-235 loading to be retained. Good experience has been obtained to date in the irradiation behaviour of this fuel. Detailed reactor physics calculations are now in process to identify

its impact on the reactor core characteristics. Previous but less detailed calculations indicated that due to the well thermalized spectrum in such reactors, the impact of the increased U-238 would be quite small.

The cores of these two reactors are highly heterogeneous and thus require detailed reactor physics codes to allow the core characteristics (flux distributions, reactivity worths, etc.) to be accurately calculated. Development continues of improved methods, concentrating largely on the use of a combination of fine mesh, albedos, and discontinuity factors in hexagonal geometry multi-group diffusion theory. Further experiments in ZED-2 in support of these reactors were undertaken[10,11].

Advanced Systems

The measurements of the reactor yield from thick targets when irradiated with 100 MeV protons from the McGill University Cyclotron was extended to include iron, copper and thorium. The previous results for lead and lithium were used in the evaluation of a conceptual thermal neutron source based on an intermediate energy proton accelerator[12].

A workshop was held at CRNL in September 1983 to discuss the present status of the codes, data and conceptual designs for the target/blanket of an Accelerator Breeder.

An experiment to assess the neutronics characteristics of a fusion reactor blanket assembly containing graphite and thorium was completed. The experimental results have been used to validate codes and cross section libraries used in such studies[13].

New Applications

A uranium oxide LEU core is under development for the 20 kW SLOWPOKE-2 research reactor. It had been hoped to utilize the U-Si-Al in Al dispersion fuel under development for NRX. However, as reliable manufacture of such fuel with the higher uranium concentration required for SLOWPOKE-2 could not be guaranteed, the evaluation is now focused a UO_2 pin concept. Reactor physics calculations to identify the core characteristics are currently in progress.

Reactor physics characteristics of the 2 MW SLOWPOKE-3 heating reactor were reported[14]. An experimental program to evaluate the thermolhydraulic characteristics of this concept is in progress. Calculations pertaining to its reactor physics and dynamic characteristics continue[15].

A further program is in hand to establish the technical and economic ability of higher power (20 MWth) reactors for dual heat and electricity production. Reactor physics characteristics of several concepts have been established. These were an organic pool concept and two designs in which there were either organic on pressurized light water cooled CANDU fuel bundles in tubes in a D_2O moderator.

References

1. "Rational Mapping (RAM) of In-Core Data" R.A. Bonalumi and N.P. Kherani, CNS/ANS International Conference on Numerical Methods in Nuclear Engineering, Montreal, September, 1983.

2. "Possible Use of Measurements to Amend Cross Section Tables (PUMA)", N.P. Kherani and R.A. Bonalumi, CNS/ANS International Conference on Numerical Methods in Nuclear Engineering, Montreal, September, 1983.

3. "The Use of a Finite-Difference Scheme to Improve the Accuracy of the Modal Expansion Method" A.R. Dastur and P.S.W. Chan, CNS/ANS International Conference on Numerical Methods in Nuclear Engineering, Montreal, September, 1983.

4. "LATREP", J. Griffiths, Atomic Energy of Canada Limited, Report AECL-7603 (1983).

5. "WIMS-CRNL A User's Manual for the Chalk River Version of WIMS", G.J. Phillips, Atomic Energy of Canada Limited, Report AECL-7432 (1982).

6. "Testing ENDF/B-V Data for Thermal Reactors" D.S. Craig, Atomic Energy of Canada Limited, Report AECL-7090, Addendums 1, 2 & 3 (1983).

7. "FISSPROD-3 An Expanded Fission Product Accumulation Program using ENDF/B-V Decay Data" W.H. Walker et al., Atomic Energy of Canada Limited, Report AECL-6973 (1982).

8. "Micrete Version 4.1 User's Manual and Program Description", R.A. Judd, Atomic Energy of Canada Limited, Report AECL-7679 (1982).

9. "Improved Solution of Integral Transport Equation Across a Plane Boundary" M.S. Milgram, CNS/ANS International Conference on Numerical Methods in Nuclear Engineering, Montreal, September, 1983.

10. "ZED-2 Experiments on the Effect of a Co Absorber Rod on an NRU Loop" G.M. Arbique and P.M. French, Atomic Energy of Canada Limited, Report AECL-7515 (1983).

11. "Experiments Performed in ZED-2 in Support of the Irradiation of $(Th,Pu)O_2$ Fuel (BDL-422) in NRU" R.T. Jones, Atomic Energy of Canada Limited, Report AECL-7918 (to be issued).

12. "Characteristics of a Thermal Neutron Source Based on an Intermediate Energy Proton Accelerator" M.A. Lone et al., Atomic Energy of Canada Limited, Report AECL-7839 (1983).

13. "Neutronics Evaluations of Activations in Graphite-Thorium Assemblies, 14 MeV Neutron Sources: Comparisons with Measurements" S.A. Kushneriuk and P.Y. Wong, Atomic Energy of Canada Limited, Report AECL-8060 (1983).

14. "Reactivity Calculations for the 2 MWth SLOWPOKE-3 Heating Reactor" J.D. Irish, Canadian Nuclear Society 4th Annual Conference, Montreal, June, 1983.

15. "Large-Signal, Dynamic Simulation of the SLOWPOKE-3 Nuclear Heating Reactor" C.M. Tseng and R.M. Lepp, Atomic Energy of Canada Limited, Report AECL-8107 (1983).

91100020

Risø National Laboratory

Recent Reactor Physics Activities in Denmark

by

Hans Neltrup

1. Design calculations on a silicium doping rig at DR 3

Doping silicium crystals by neutron absorption has become an activity of considerable economic significance at the DR 3 reactor.

The irradiation of the ϕ 100 mm by 400 mm silicium mono-crystals takes place in vertical experimental tubes extending into the heavy water reflector of the 10 MW Dido-type reactor. Quality requirements by the customer demands that the irradiation should have a high degree of homogeneity across the crystal. In the radial direction this is obtained satisfactorily by a device that rotates the crystal around a vertical axis during irradiation. In the vertical direction constant inflow of neutrons is aimed at by stepwise changes in the absorber thickness of a SS. absorber tube surrounding the rig.

The design of this absorber tube was supported by a combination of measurements and reactor physics computations.

In order to be able to handle the reactor physics the reactor-rig configuration was converted as shown in Fig. 1 into a cylinder symmetric one, in which a realistic representation of the rig in the centre was surrounded by an annular reflector, an annular homogenised reactor core followed by an outer reflector. The system was completed by axial reflectors in top and bottom.

The flux in this system was calculated by use of three group diffusion theory. By varying the group 3 absorption cross section in top and bottom reflector the thermal, group 3 flux along the rig axis was fitted to the corresponding flux measured in the actual rig position. The kind of fit obtained in this way is illustrated in Fig. 2.

Once a good fit is obtained the effect of varying absorber configurations can be examined. In fig. 2 the flux flattening effect of a number of absorber configurations is illustrated.

The reactor physical aspect of this problem and the participation in the RERTR project has triggered an increased interest in the DR 3 operational staff for a full 3-dimensional reactor physics model for the entire reactor. One object of such a model would be an independent evaluation of the reactivity worth of the different experiments, which is traditionally monitored semi empirically by a complicated inter-calibration system.

Test reactors have by nature small dimensions and a complicated structure, which has made reactor physicists reluctant with regard to realistic modelling. One particular characteristic of the DR 3 reactor is the "signal arm" type coarse control rods. In order to be able to model theese the 3-dimensional diffussion theory program DC4 has been modified so that triangular meshes can be introduced. The modified program is operational and has been tested, but no realistic calculations have been performed so far.

2. Neutron diffusion theory

An investigation of multigroup neutron diffusion theory with emphasis on the k_{eff} spectrum, has been initiated. The aims of this investigation are the usual ones (distribution of eigenvalues in the complex plane, completeness and regularity of generalized eigenfunctions, positivity of fundamental mode). The models considered are realistic in the sense that albedo

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boundary conditions and ragged reactor boundaries are taken into account.

3. Response matrix program for fuel boxes

The interactive effect of gadolinium poisoned pins in the benchmark formulated by NEACRP has been examined at the beginning of lilfe by use of the REPRO-FLUSO program.

A mutual flux tilt of 12% across the poison pins in the diagonal connecting direction was found. A 4-6% reduction in the total absorption in the poison pins compared with the absorption in a single poison pin in a 3 by 3 pin configuration was found depending on the kind of normalisation used.

4. Determination of pin power distribution from coarse mesh solution

A method, the superposition method, which calculates the local pin power histories for the individual assemblies was investigated . It consists in developing the solution to the heterogenous assembly problem with boundary conditions derived from the global solution after a set of precalculated solutions (base solutions) to a number of assembly problems.

Typically the base solutions would have white boundary conditions on three sides whereas $\lambda \cdot J / \phi$ could have a given shape, constant, linear or parabolic along the fourth side. Here λ is the eigenvalue of the problem. This type of base solution proved to be very efficient even when only a few were used.

In an earlier test of this method a number of representative regions of a reactor core, each surrounded by suitable driver and/or reflector zones was selected and calculations carried out separately on each of them.

With a view to certain objection to this procedure a quarter core was calculated in a subsequent investigation pin by pin as one large fuel assembly. This calculation constituted a reference solution with which the superposition method and other current methods such as the flux normalization method could be compared. Two different quarter cores shown in Fig. 3 one with strong flux gradient and one more homogeneous were investigated. The results showed that with the superposition method the error in local pin power could be kept within 5-8% using only 8 base solutions, whereas the normalization method could lead to errors as large as 40%.

A second investigation was concerned with the power distribution during burnup. The overall quarter core calculations were performed as illustrated in Fig. 4 pin by pin as one large assembly and the burnup for each assembly was carried out with the correct spectrum by use of the "Flux-luppe" method. The obtained assembly power distribution was compared with the one obtained by the following iterative scheme. Each assembly was burned in the flux from a detailed heterogeneous calculation with white boundary conditions. The assembly power level distribution was taken from a fine mesh overall solution as above. At the end of each burnup step new homogenized fine mesh cross sections were fed back into the subsequent fine mesh calculation. The scheme is illustrated in Fig. 5. The investigation showed that if the error in pin power should be kept below 5% during the lifetime which is a desired goal there has to be a coupling between the pin cell depletion and the global calculation. If up to 10% errors can be tolerated the global flux tilt effect can be neglected in the assembly depletion calculations.

Fig. 1

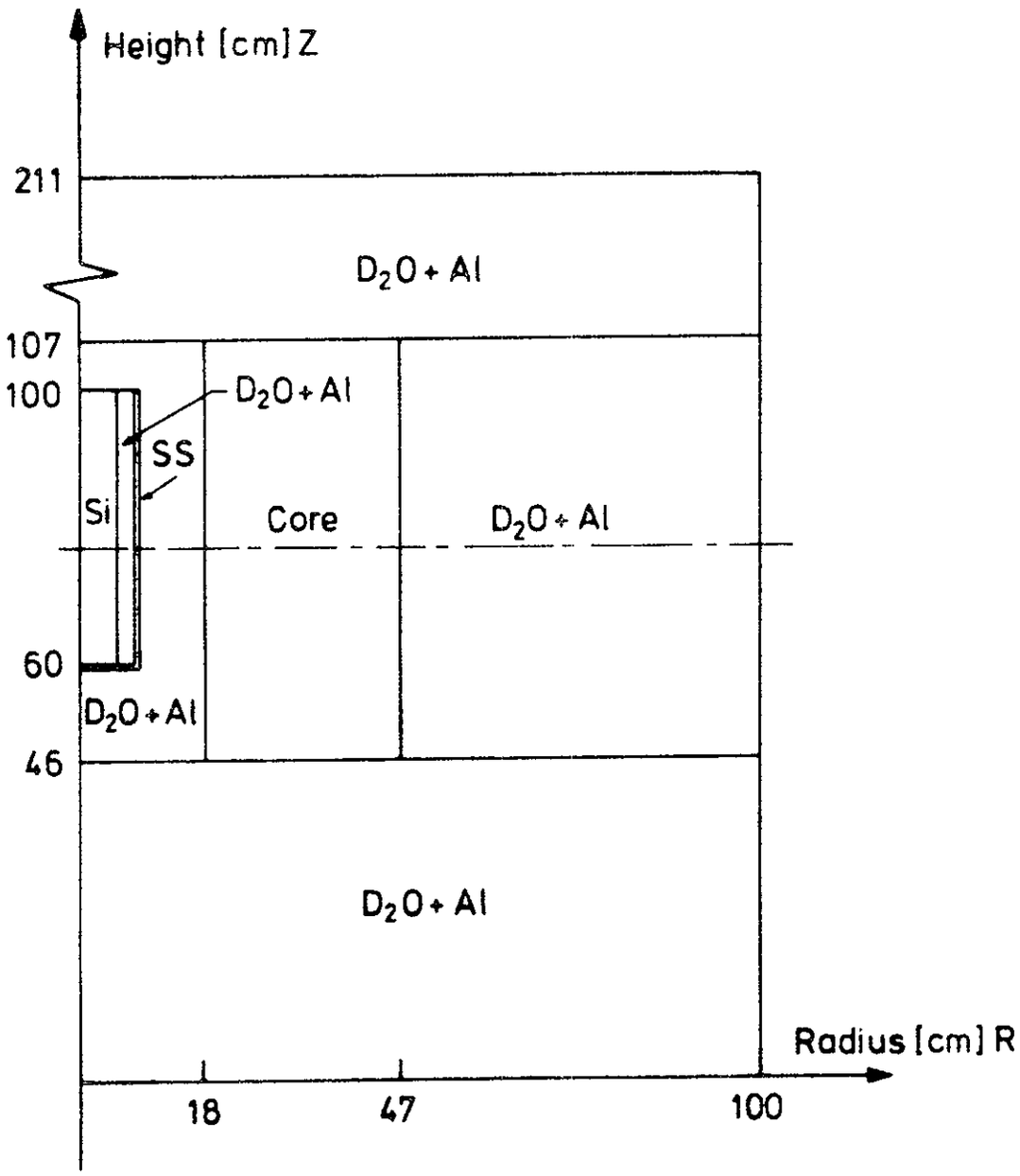
Height [cm] Z
211
$D_2O + Al$
107
100
$D_2O + Al$
SS
Si
Core
$D_2O + Al$
60
$D_2O + Al$
46
$D_2O + Al$
Radius [cm] R
18
47
100

Fig. 2

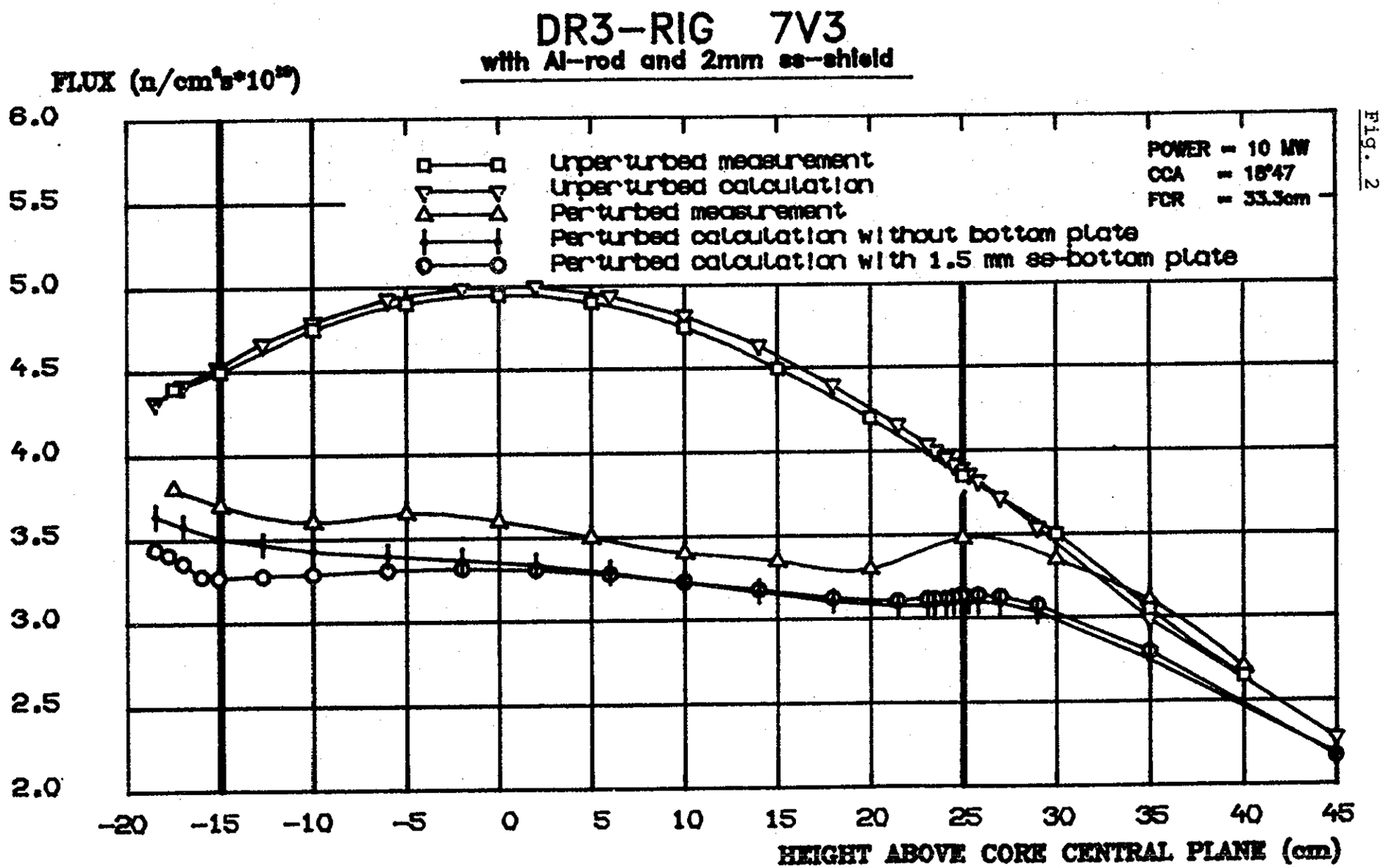

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Figure 3. Benchmark problem definition.

Test 1

A	B	B	A	A	B	B	A	A	B	B	B	B	C
B	A	A	B	B	B	B	B	B	A	A	B	B	C
B	A	A	B	B	B	B	B	B	A	A	B	B	C
A	B	B	A	A	B	B	B	B	B	B	B	B	C
A	B	B	A	A	B	B	B	B	B	B	B	B	C
B	A	A	B	B	A	A	B	B	B	B	B	C	C
B	A	A	B	B	A	A	B	B	B	B	C	C	C
A	B	B	A	A	B	B	B	B	B	C	C	C	C
A	B	B	A	A	B	B	B	B	C	C	C	C	C
B	A	A	B	B	B	B	B	C	C	C	C	C	C
B	A	A	B	B	B	B	C	C	C	C	C	C	C
B	B	B	B	B	B	C	C	C	C	C	C	C	C
B	B	B	B	B	C	C	C	C	C	C	C	C	C
C	C	C	C	C	C	C	C	C	C	C	C	C	C

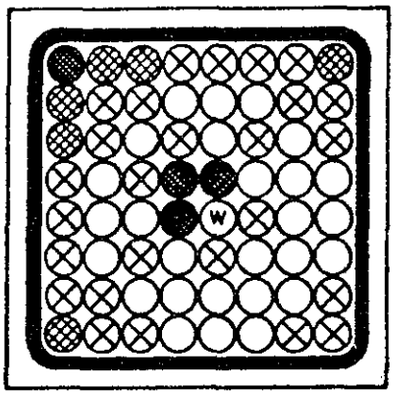
Test 2

(Rod 4)

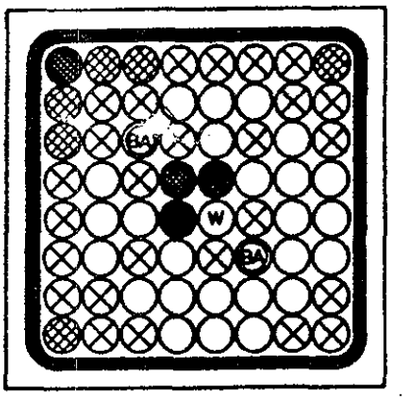
A	B	B	A	A	B	B	A	A	B	B	B	B	C
B	A	A	B	B	A	A	B	B	A	A	B	B	C
B	A	A	B	B	A	A	B	B	A	A	B	B	C
A	B	B	A	A	B	B	A	A	B	B	B	B	C
A	B	B	A	A	B	B	A	A	B	B	B	B	C
B	A	A	B	B	A	A	B	B	B	B	B	C	C
B	A	A	B	B	A	A	B	B	B	B	C	C	C
A	B	B	A	A	B	B	B	B	B	C	C	C	C
A	B	B	A	A	B	B	B	B	C	C	C	C	C
B	A	A	B	B	B	B	B	C	C	C	C	C	C
B	A	A	B	B	B	B	C	C	C	C	C	C	C
B	B	B	B	B	B	C	C	C	C	C	C	C	C
B	B	B	B	B	C	C	C	C	C	C	C	C	C
C	C	C	C	C	C	C	C	C	C	C	C	C	C

Lay-out of the assemblies.

Type A



Type B



0.97 W/O U-235

1.40 - -

1.88 - -

2.47 - -

BA 2.47 - - With BA

W Water hole

91100027

Figure 4 Core-depletion approximation.

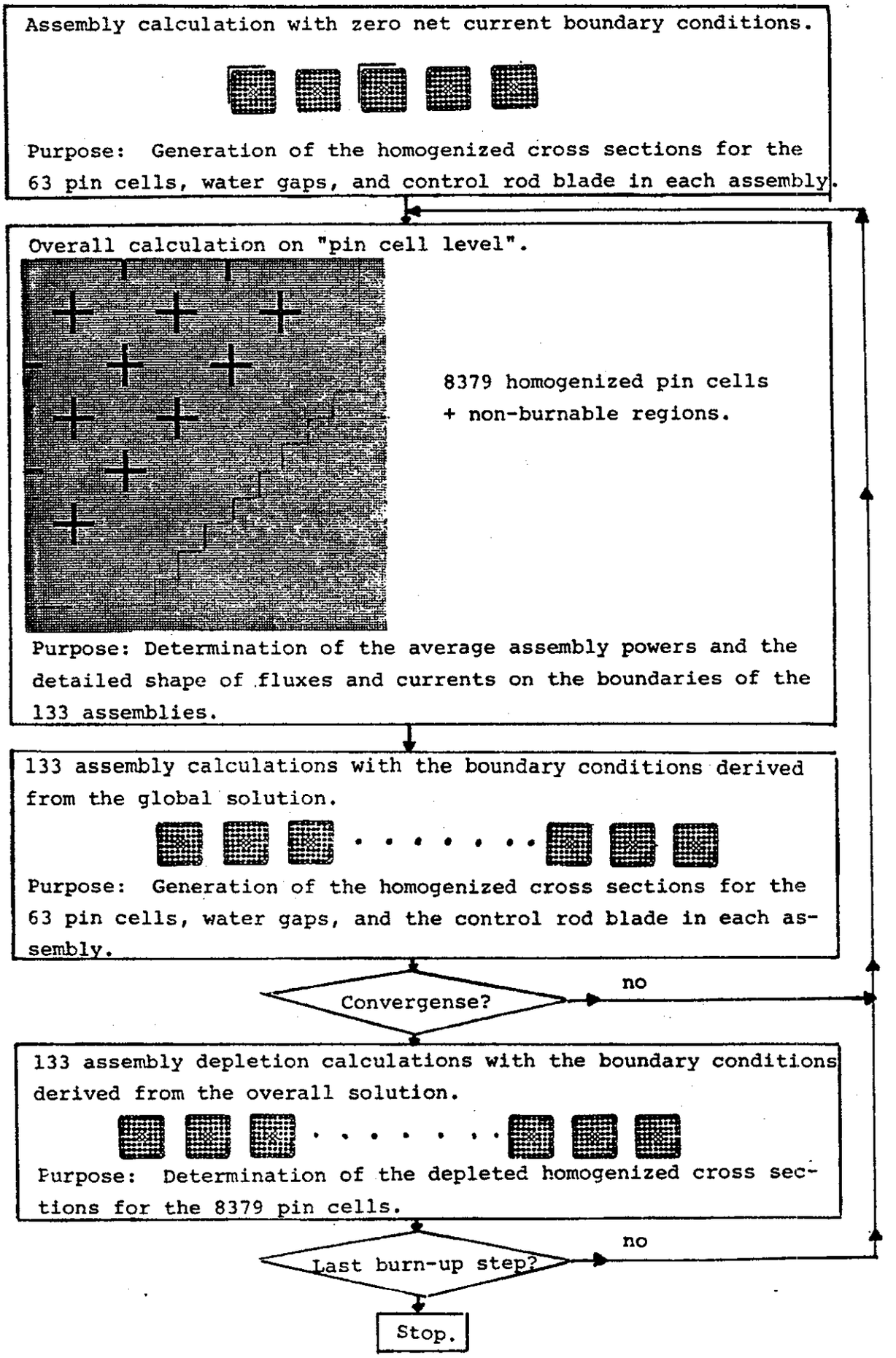
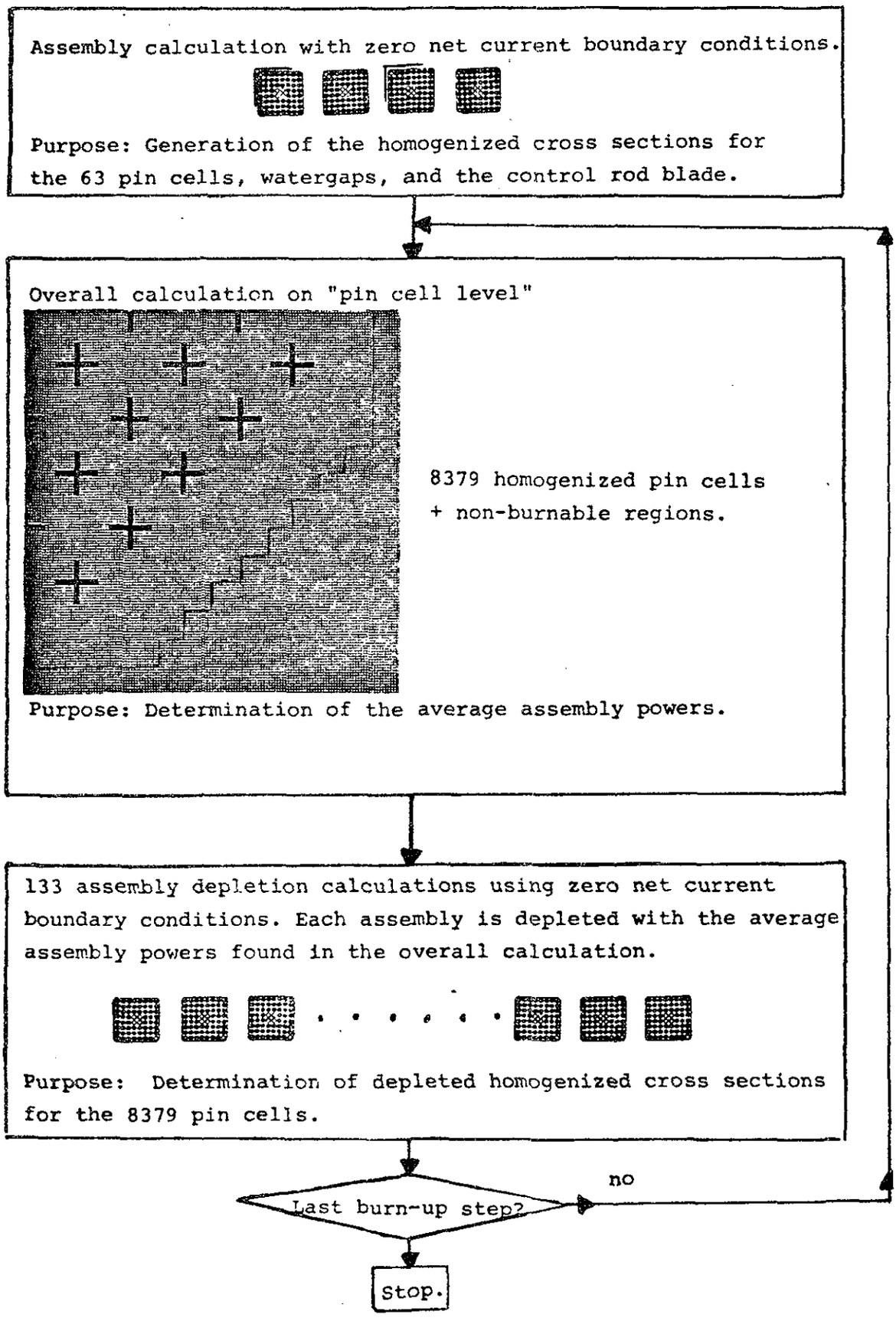


Figure 5 Assembly-depletion approximation.



TRYK: RISØ REPRO

REACTOR PHYSICS IN FINLAND
STATUS REPORT TO THE NEACRP 1983

Compiled by R.Höglund
Technical Research Centre of Finland

1. GENERAL

With 38% of the electric power used in Finland in 1982 produced by the four operating nuclear reactors, and 36% during the first half of 1983, the reactor physics activities at the Nuclear Engineering Laboratory of the Technical Research Centre of Finland (VTT) have mainly aimed at the safe and reliable operation of these reactors. The development of the computer code system has thus been restricted to minor improvements usually in connection with work carried out for the power companies and the radiation protection authorities. A license agreement with Studsvik Energiteknik AB has made the CASMO cell code available to VTT, thus resulting in a more complete code system for the BWR calculations also.

2. CELL CALCULATIONS

VTT made at the end of the year 1982 with Studsvik Energiteknik AB a license agreement, which gives it the right to use the CASMO fuel assembly burnup program and some auxiliary programs (MICBURN, CASPOL, MOVEROD and CLIB) in its own activities and for in-core fuel management and other services to Finnish customers. The programs were delivered at the beginning of February 1983 and they were installed on VTT's own CDC computer. After some successful test calculations, several sets of group constants have already been produced with CASMO for fuel assemblies of TVO's boiling water reactors.

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3. CORE CALCULATIONS

The BOREAS code system has again been used extensively to analyze the burnup cycles of the two TVO reactors and the suitability of different kinds of fuels for these reactors. According to test calculations corresponding to critical core conditions, the effective multiplication factor is getting closer and closer to 1 as the equilibrium cycle is approached, being about 0.998 during the fourth burnup cycle against 0.993 to 0.994 for the first couple of cycles. The main reason for this change might be found in the cell data on which the calculations are based. During a single burnup cycle, the variations in k_{eff} are about 0.002 to 0.004.

Furthermore, some studies have been made on the 400 MW version of the SECURE heating reactor. The calculations have, as usual, required some development work on BOREAS and its auxiliary codes. Since the CASMO burnup code is now in use at VTT, the links between CASMO and BOREAS will have to be improved in order to make the transfer of data as smooth as possible.

As a rather new application of BOREAS, shut-down margins have been calculated for a cold core with all control rods except one fully inserted. Due to the lack of sufficient test material, the reference level of k_{eff} is still somewhat uncertain, but the calculated value for a critical core seems to be slightly higher in the cold case than in the hot one.

The data condensation code BROAD that produces two-group parameters for one-dimensional dynamics calculations has been further developed and used for different applications. Essentially it creates neutron flux weighted average cross sections at a number of axial levels in the reactor core using the results of the corresponding BOREAS (steady state) calculation.

4. PRESSURE VESSEL IRRADIATION

Development of REPVICS (REactor Pressure Vessel Irradiation Calculation System) has been finished and the system has been tested by comparison of calculated and measured $^{54}Fe(n,p)^{54}Mn$ reaction rates. This comparison, which was reported at the 6th ICRS in Tokyo in May (paper 9-4), shows good agreement. The DOT (S_N) results tend to lie slightly above the measured values with the maximum difference amounting to about 25%. The HEXANN (Monte Carlo) results do not show any systematic deviation large enough to be distinguished from the fairly considerable stochastic scatter. However, since the $^{54}Fe(n,p)^{54}Mn$ reaction is insensitive to neutrons below 2.7 MeV, which account for 2/3 of the radiation damage, further verification through comparison with measured values for reactions with a lower threshold is desirable.

REPVICS has also been applied to calculations of fluxes and displacement rates in the Loviisa reactors. The results agree within 15% with the best previous estimates.

The calculation scheme is shown in figure 1.

91100032

5. DYNAMICS CALCULATIONS

Various submodels of the BWR dynamics code TRAB have been further developed and some new options included in the program which has been extensively utilized in various transient analyses.

To avoid time-consuming manual tuning of the steady state axial power distribution a procedure was developed to achieve the given power distribution. Group constant data are modified by solving the artificial control rod parameters, which result in the desired power distribution. The power fitting option can be used e.g. to obtain a definite initial state for a transient calculation or in order to get the power distribution given by a stationary calculation in spite of the differences caused by different models in stationary and dynamic programs. An example of the fitting is given in figures 2 and 3.

The steam line model of TRAB has been further developed to enable improved analysis of fast overpressurization transients (important reactivity transients). In the original model steam is supposed to be saturated everywhere in the pressure vessel, but the present version takes into account the possibility of wet or superheated steam both in the steam lines and the steam dome.

A new acceleration method for coupled neutronic and thermohydraulic iterations has been implemented in the program to improve the convergence of the steady state solution.

TRAB has also been modified to enable analysis of the SECURE district heating plant. Modelling of SECURE required a special application of the code because of the unconventional features of the plant. The TRAB-SECURE model includes many

assumptions that are related to the description of the consequences of an uncontrolled dilution of the borated water in the primary system but it might be possible to extend the model to some other cases as well.

RHOG/LL

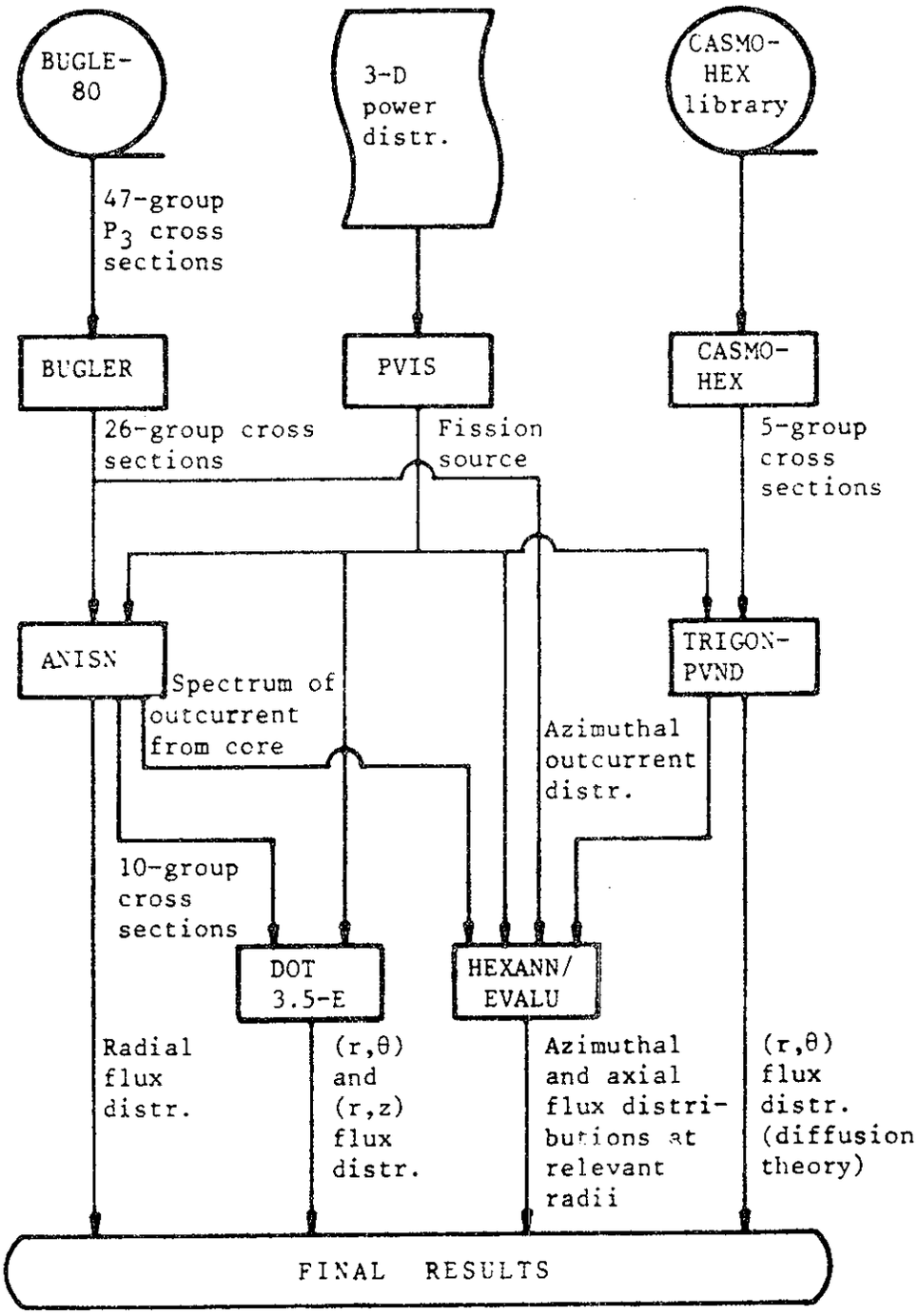

Figure 1. Pressure vessel irradiation calculation scheme.

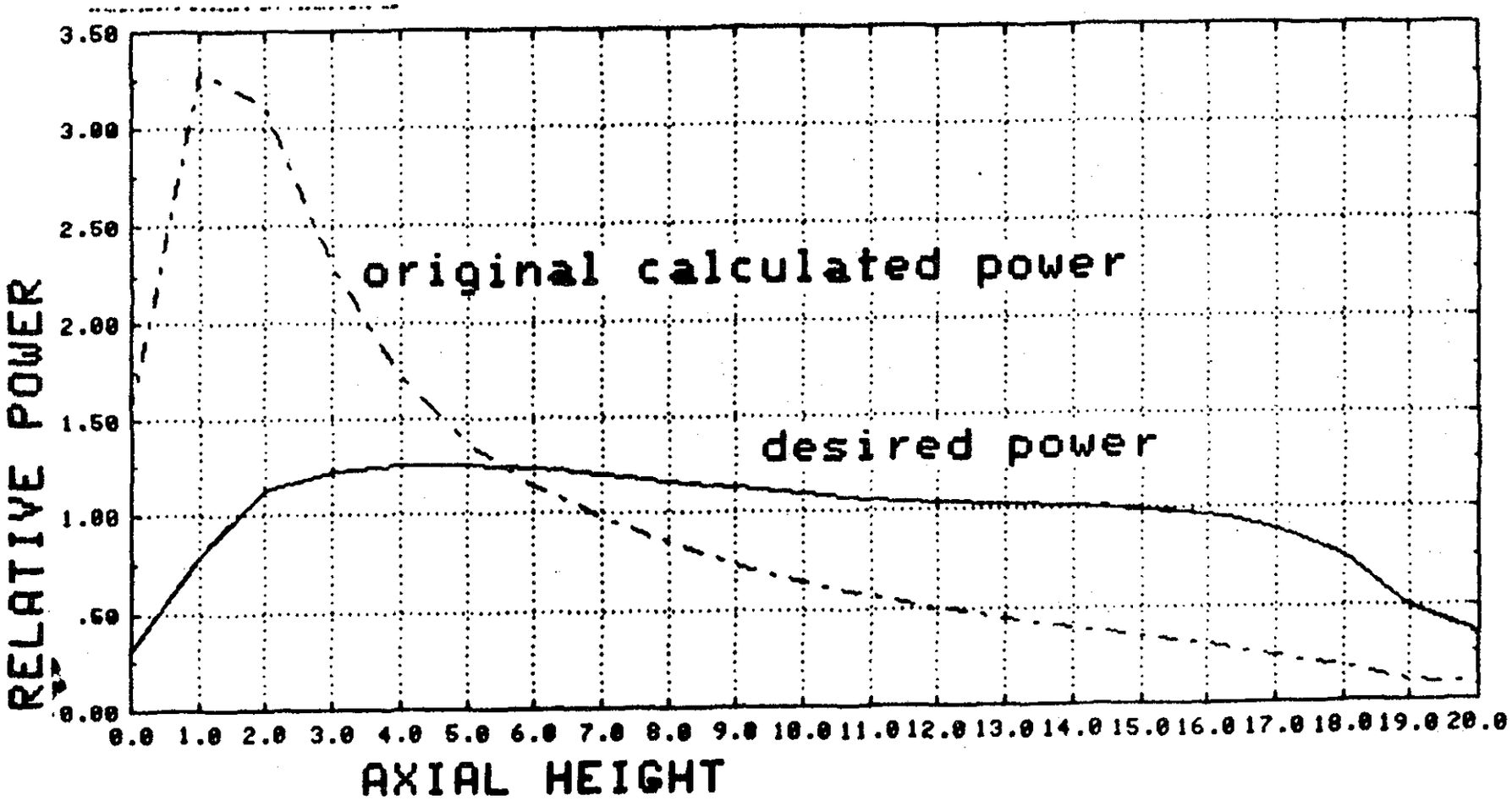

Figure 2. Power distribution obtained by using the original cross section data set and the desired TRAB distribution.

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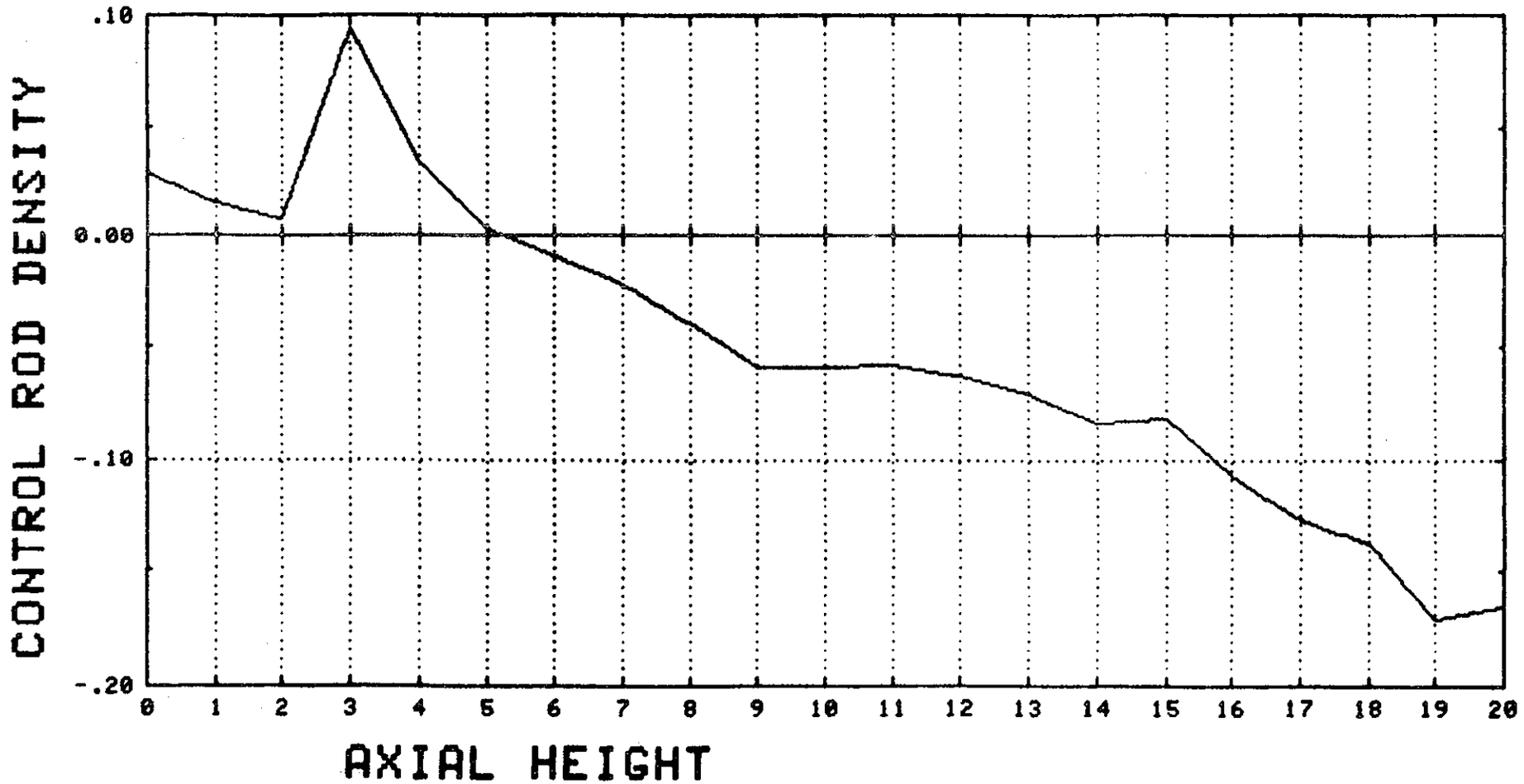


Figure 3. Control rod density needed to modify the original cross section data set in order to obtain the desired power distribution with TRAB.

REACTOR PHYSICS ACTIVITIES IN FRANCE

October 1982 - September 1983

C. GOLINELLI - M. SALVATORES
Commissariat à l'Energie Atomique

1 - GENERAL -

The effect of the economic crisis has been to reduce the electricity requirement. But, after a national debate, the French government has decided to pursue the nuclear plant implementation with a slowing down. Now, two 1400 MW reactors are ordered both in 1983 and 1984 and one in 1985 with one option for another one.

The situation is then the following :

	PWR 900	PWR 1300
Operational	23	0
Start-up (84)	5	1
Under construction	6	14

New orders : 5 PWR 1400

In 1990, the 53 units will correspond to 54 GW installed.

This very important domestic program leads to some requirements in the field of the reactor physics, mainly related to the need :

- to lenghten the duration of the batches (12 to 18 months)
- to increase the fuel exposition (33 GWD/t to 45 GWD/t).

Secondly, since the French politics aims to reprocess spent fuel, large quantities of recovered materials will be available and recyclings of plutonium and uranium are studied.

The third aspect of the national program which has bearings on the physics program is connected to the transformation of the PWRs in order to increase the conversion factor. The goal is to reduce the natural uranium consumption, in the perspective of the LMFBR's implementation after the year 2000.

For what concerns FBR's, the start-up of SUPERPHENIX is scheduled in the second semester of 1984. The order for the subsequent 1500 MWe plant is foreseen in 1986. Finally, a joint utility-industry-national laboratory task force (ECRA) has been established to study improved FBR designs, in particular from the economic poin of view.

91100038

2 - FAST REACTOR PHYSICS -

2-1 : Introduction

The major activities of the Fast Reactor Physics Program during the period October 1982 - September 1983 were devoted to the following topics :

- SUPERPHENIX start-up, related both to the actual experimental program definition, and to the mock-up in MASURCA of some configurations of the approach-to-criticality sequence.
- Control rod studies in the heteregeneous configurations of the RACINE program.
- PHENIX experiment analysis to validate SUPERPHENIX design calculations.

2-2 : Experimental studies

2-2-1 : PHENIX

The experiments aimed to study the Boron capture rate in the absorber subassembly have been analyzed, and some results are reported at this meeting /1/. The agreement of calculations (using standard design tools) and experiments are satisfactory, even if fairly significant uncertainties have to be associated to the experimental results.

Preliminary results indicate an agreement of the order of ± 30 % for the calculation/experiment comparison on the reaction rate values of the DINOSAURE experiment, devoted to the neutron dosimetry in subassemblies located far from the core center (internal fuel storage location).

As for the fuel irradiation experiments, the TRAPU experimental results related to pins with a high Pu240 and Pu241 content (45 % and 15 % respectively), are now available and the analysis is underway. The PROFIL-II experiments, related to small samples of single material irradiation, are becoming available and the experimental results related to the major actinides are being analyzed. Finally, the DOUBLON experiment for the irradiation of standard blanket subassemblies pins in the first and second blanket raws have been completed and they are also being analyzed.

For what concerns the secondary sodium activation, the standard design calculation scheme has been validated against the experimental value, and a good agreement (less than a factor of two) has been obtained for the integral value, with a good prediction of the spatial flux shape, experimentally observed on the heat-exchanger axis /2/.

2-2-2 : SUPERPHENIX

A first criticality configuration has been defined (chess-board type configuration, with a homogeneous distribution in the core of dummy subassemblies). The different steps of the approach to criticality have also been defined, with a study of the appropriate calculation methods, which should be applicable on a large range of reactivity values and flux levels on the in-core detectors.

The definition of the physics experiments planned for that phase, has been continued, with special emphasis on control-rod configurations. In connection with that program, experimental studies have been performed to define the experimental methods to be used in SUPERPHENIX. In particular, experimental studies have been performed in MASURCA to validate an absolute reactivity worth measurement technique (CARPENTIER method) and a relative reactivity measurement technique (modified source multiplication), taking into account the peculiarities of the application of these techniques to a large LMFBR.

2-2-3 : RACINE program

The RACINE 1E configuration devoted to multiple control-rod interaction experiments has been completed /3/. Calculation/experiment comparisons show a good coherence among the values related to different control rod patterns. Some of the experiments, performed in subcritical configurations, will be repeated successively in critical configuration, to allow the measurement of the perturbed reaction rate distributions and to verify the relation between control rod reactivity worth and flux distribution calculation/experiment discrepancy.

The RACINE 1S phase has been set up to provide a validation of the data and the calculational methods used for the start-up configuration assessment of SUPERPHENIX. A series of configurations has been studied starting from a critical configuration in which 40 dummy steel subassemblies have been simulated (see Fig. 1). The successive configurations were made subcritical by the insertion of further dummy subassemblies. Flux-to-reactivity relation of control rod effects and neutron flux transmission to out-of-core detectors have been studied.

91100040

2-2-4 : HARMONIE shielding experiments

The first phase of the JASON program has been completed. Different Boron-steel layers arrangements have been studied and also the use of borated steel. Calculation/experiment comparison indicate a worse agreement on integral flux responses in comparison to standard steel/Na mixtures /4/.

2-2-5 : ERMINE experiments

Heterogeneity experiments have been performed to compare pin and plate configurations. For each geometry, two different, and comparable, degrees of heterogeneity have been studied. These experiments should be completed by the end of 1983.

2-3 : Theoretical studies

2-3-1 : Control rod studies

Control rod studies have been adressed, besides the analysis of the integral experiments of the RACINE 1E configuration, to sensitivity analysis both of energy dependent and of space dependent effects /5/. The results of these studies should be used to shape future integral experiments on MASURCA.

2-3-2 : Heterogeneity

The CADENZA experiment, proposed by the UK to the NEACRP /6/ as a heterogeneity calculation method benchmark, has been analyzed 2D collision probabilities routines have been used to analyzed the pin geometry, and a simulation has been attempted of the 3D effects of the plate geometry. A discrepancy ($\sim$ 0.6 % in $\Delta K/K$) has been observed when comparing the C/E for the two geometries. Further studies to explain the discrepancy are planned, in the framework of the NEACRP exercise.

2-3-3 : Transport calculations

In order to improve the efficiency of standard two dimensional transport calculations, based on the discrete ordinate method, it has been developed a 1D transport code, which makes use of a diffusion initialisation, the synthetic diffusion acceleration method for inner iterations and the Tchebychev method for outer iterations. A gain in computer time of a factor $\sim$ 4 has been obtained, in comparison with the standard ANISM code. It is planned to extend the study to a 2D code.

2-3-4 : Multidimensional design calculation codes

The new code system for design calculation /7/ is now operational in the 3D hexagonal option. Substantial computer time gains are expected with respect to older versions.

For what concerns the inhomogeneous calculation option in 2D or 3D, significant improvements have been obtained using appropriate initialisations from the associated homogeneous problem. Moreover parametrical studies have been performed to define optimal utilisation options (number of groups, mesh size effects, axial buckling, etc...).

2-3-5 : Cross-section Evaluation

There has been active participation to the JEF project, both in terms of the definition of data needs (e.g. covariance matrices) and in terms of evaluation of data (e.g. Pu239 in the resonance region). Future work concerns the benchmark phase of the project, and in particular the associated sensitivity and uncertainty studies.

Work has been performed in the framework of the task force on U238 resonance parameter discrepancies and on a critical study of Ni, U235 and Pu239 data, in the framework of INDC.

Finally, a methodology is being developed for the simultaneous evaluation of isotopes of atomic masses A, A-1 and A-2, with application to the U and Pu isotopes.

In connection with the JEF file development, it is foreseen to improve the present point cross section processing strategies. A system, based on the NJOY code is developed at SACLAY (THEMIS system). It is foreseen to improve the present unresolved resonance data treatment and to make the system compatible with the sub-group method, used in the CARNAVAL system.

2-3-6 : Fission Product Data

In view of the results of the NEACRP benchmark on a LMFBR burn-up calculation /8/, it has been considered of interest to assess data uncertainties associated with pseudo fission product cross-section calculation. A first study, related to fission product yields, has been completed /9/. Further studies will concern the volati fission product migration, and the global data adjustement, based on power reactor fuel pin irradiation experiments.

91100042

3 - LIGHT WATER REACTOR PHYSICS -

The Light Water Reactor Physics program has found a new interest due to the extension of the French nuclear program. The three main areas for research and development are :

- improvement of the reactors in operation
- modification of the fuel cycle in the next future
- studies of new types of reactors for mid-term.

3-1 : Theoretical studies

Transport theory

A new method has been developed for fine structure burn-up calculations of very heterogeneous large size media. It is the generalization of the well-known surface-source method, allowing the coupling of actual two-dimensional heteregeneous assemblies, known as 'substructures". The method has been applied to a rectangular medium, divided into sub-structures, containing rectangular and/or cylindrical fuel, moderator and structure elements. A zone-wise flux expansion is used to formulate a direct collision probability problem within it. The coupling of the sub-structures is performed by making extra assumptions on the currents entering and leaving the interfaces.

New characteristic methods for the solution of the x, y geometry discrete ordinates neutron transport equation have recently been introduced : five polynomials, without any continuity requirement are used on each mesh cell, the first one approximates the angular flux inside the cell and the others are valid along the cell edges.

Diffusion theory

An unified formulation of non conforming finite elements with quadrature formula and simple nodal scheme has been developed. The theoretical convergence is obtained for the previous scheme when the mesh is refined. Numerical tests are provided in order to support the theoretical results.

3-2 : Experimental studies

3-2-1 : EOLE

The CAMELEON experimental program announced at the 24th meeting is being performed at the EOLE facility. It is devoted in particular to the qualification of the absorber effect calculations (Program started in October 1982).

The first studies are devoted to the gadolinium effect. The program consists in reactivity effect measurement and power or flux distribution for :

- parametric studies (Gd content, type of support) for one central rod
- interaction between absorber rods (1 to 12 Gd rods, rod distance, etc...)
- interaction between absorber and water holes
- interaction among gadolinium and silver, indium, cadmium
- simulation of a PWR assembly.

The second part is devoted to the other absorbers and materials : silver - indium - cadmium - hafnium - B_4C - borated glass, steel, zirconium and "grey rods".

The third part will be performed in 1984 with the β_{eff} measurement, the reflector effect, (edge subassemblies, baffle, ...).

A new experiment is prepared for the end of 1984. It is a tight lattice assembly with mixed oxyde fuel. The neutron parameters for this kind of lattice must be qualified. For this experiment, it is necessary to design a very large zone to obtain the asymptotic spectrum.

3-2-2 : MINERVE

The MINERVE facility is shared between the FBR program and the PWR one. The name of the experiment for the second case is MELODIE. The goal of MELODIE is mainly based on the oscillation technique. Small samples are oscillated in the selected spectra. The microscopic cross sections can then be tested. Three objectives characterized the experimental program :

- thorium cycle
- recovered uranium recycling
- gadolinia with uranium oxyde or aluminia.

3-2-3 : MELUSINE

The MELUSINE reactor is located at the GRENOBLE Center. A 13x13 UO2 rods assembly is loaded at the Center of the MTR core. The goal of the irradiation is to follow the gadolinium depletion but also the power distribution as a function of time and the isotopic composition of the various types of rods. The experiment was started in November 1982. The first poisoned rod was unloaded in March 1983. The experimental program consists in two measurements :

- map of flux distribution by γ scanning over the rods
- destructive isotopic analysis.

The results of the isotopic analysis will be obtained in September 1983. The end of the irradiation is scheduled in January 1984.

3-2-4 : Spent Fuel Analyses

The present French program in the field of the Spent Fuel Analyses is a long term one, without new specific developments.

The results from TIHANGE reactor are analyzed and the search of tendencies shows small differences between the predictions and the measurements.

The FESSENHEIM program has been started by measuring the isotopic composition of the second cycle fuel. The fourth and the fifth cycles data are expected to be available in the near future.

3-3 : Instrumentation, Core Surveillance and Simulation

In the frame of the improvement of surveillance technique, we can indicate the main actions during the last year :

- implementation of a new algorithm to use the thermocouple measurement,
- follow-up of the baffle jetting and, also, mechanical vibrations by neutron noise analysis.

A paper is presented at this session about the simulation studies. EDF is equipped with full scale simulation for the normal conditions of operation to train the operators. EDF is also building an accidental simulator for staff training. For general studies, a new simulator is designed. The primary circuit is partially simplified to realize the studies in real time.

3-4 : Fuel cycle

The industrial applications are directly connected with the reprocessing plant construction and the follow-up of the process. The isotopic correlation technique is used with sucess to measure the burn-up of the supplied spent fuel assembly at the entrance of La Hague plants. An industrial use is envisaged at the time of subassembly departure from the storage ponds.

The available recovered materials introduce a requirement for the recyclings. Two types are studied :

- recovered uranium recycling by re-use in PWRs with re-enrichment
- available plutonium recycling.

Compatibility with the other options (high burn-up, very long batches) are examined to give the necessary input to at the economics specialists. In this framework, various strategies will be compared.

- R E F E R E N C E S -

/1/ G. HUMBERT et al
Paper at this meeting

/2/ J.C. CABRILLAT et al
Paper presented at 6th ICRS
TOKYO May 1983

/3/ G. HUMBERT et al
Paper presented at the ANS Topical Meeting
KIAMESHA LAKE September 1982

/4/ J.P. TRAPP - A. DE CARLI
Paper presented at the 6th ICRS
TOKYO May 1983

/5/ G. PALMIOTTI - M. SALVATORES
Paper presented at this meeting

/6/ J.M. STEVENSON
NEACRP-A-445

/7/ C. GIACOMETTI et al
NEACRP-A-531

/8/ G. PALMIOTTI - M. SALVATORES
NEACRP-A-504

/9/ G. RIMPAULT - L. MARTIN-DEIDIER
Paper to be presented to Specialist Meeting ou Yields and decay data of Fission Product Nuclides
BROOKHAVEN - 24.27 October 1983

/10/ M. ARIES - J. BOUCHARD et al
Quatre ans d'expériences d'utilisation des correlations isotopiques calculées dans l'établissement du bilan d'entrée de la Hague.
AIEA - Colloque International sur les progrès récents en matière de garanties appliquées aux matières nucléaires.
WIEN - 8.12 November 1983

/11/ M. DARROUZET et al
Etablissement d'un bilan d'entrée de combustibles de réacteurs à eau par mesures non destructives gamma et neutron.
AIEA - WIEN 8.12 November 1983

/12/ P. BERNARD et al
Experience with utilization of neutron noise in PWRs for monitoring the fuel and international structures.
AIEA - PRAGUES 21.25 June 1982

/13/ B. PAPIN et al
Utilisation des thermocouples pour la surveillance de la distribution radiale de puissance des réacteurs à eau pressurisée.
AIEA - Colloque international sur la commande et l'instrumentation des Centrales nucléaires.
MUNCHEN - 11.15 October 1982

/14/ J. BOUCHARD et al
Besoin en données nucléaires pour les réacteurs à neutrons thermiques.
International Conférence on Nuclear data for Science and Technology.
ANTWERPEN - September 1982

/15/ B. DUCHEMIN et al
Decay heat calculations any the CEA radioactivity data bank and the PEPIN code.
ANTWERPEN - September 1982

/16/ M. DARROUZET et al
Uranium recycle in PWR's
ANS Topical Meeting.
KIAMESHA LAKE - September 1982

/17/ T. VERGNAUD
TRIPOLI 2 - Energy dependent three dimensional MONTE CARLO Code.
ISPRA Course on MONTE CARLO Methods and Their Application to Radiation Shielding
ISPRA - 25.29 October 1982

/18/ P. BERNARD et al
SALAMANDRE un simulateur d'étude de séquences incidentelles de PWR.
IASTED - Lugano (Schw.) - 21.24 June 1983

/19/ C. MESSAINGUIRAL-BRUYNOOGHE
A transport calculation estimation of the neutronical effect induced on in-core detectors signals by fuel assembly vibrations.
KFK Reactor Physics Dept. - 16th Informal Meeting on Reactor Noise.
18.20 May 1983.

/20/ J.P. HENNART - J.J. LAUTARD - A. KAVENOKY
On the relationship between some modal schemes and the finite element method in static diffusion calculations.
Mathematics and Computational Meeting SALT LAKE CITY - 28.31 Mars 1983

/21/ J.J. LAUTARD - A. KAVENOKY
State of the art in using finite element method for neutron diffusion calculation.
Mathematics and computational meeting SALT LAKE CITY - 28.31 Mars 1983

/22/ Z. STANKOVSKI - A. KAVENOKY
A Sub-critical method for integral transport calculations
Mathematics and computational meeting SALT LAKE CITY - 28.31 Mars 1983

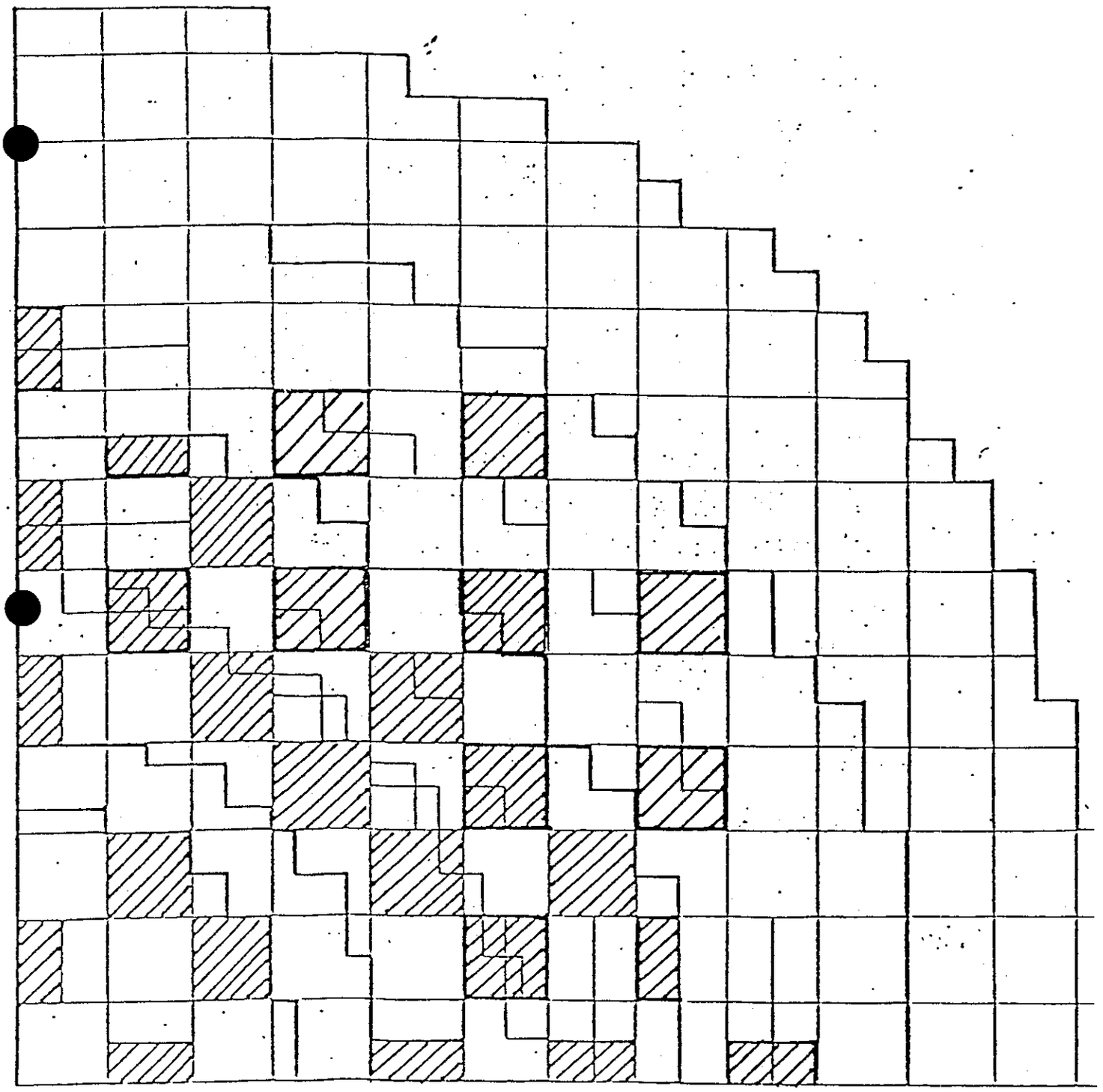
/23/ M.F. ROBEAU - J.J. LAUTARD - A. KAVENOKY
ARIANE-0 : Un système d'aide à la programmation scientifique.
INRIA - PARIS, 17.19 Mai 1983

/24/ P. RIBON
Extraction de l'information sur les densités de niveau nucléaire dans la région des énergies de résonance.
AIEA - BROOKHAVEN (USA), 11.15 Avril 1983

/25/ A. KAVENOKY - J.J. LAUTARD
The neutron kinetics and thermal-hydraulic transient computational module of the NEPTUNE system - CRONOS.
ANS - KIAMESHA LAKE, 22.24 Septembre 1982 - CEA CONF 6479

MASURCA RACINE-1S PROGRAM

CONFIG. S8 - SUBCRITICAL
(- 8450 10^{-5} $\Delta k/k$)
88 DILUENTS



REACTOR PHYSICS ACTIVITIES IN THE FEDERAL REPUBLIC OF GERMANY

Compiled by

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GENERAL

On recommendation of the Minister for Research and Technology, the Federal Government decided on 26 April, 1983 to complete the construction of the fast reactor prototype SNR-300 and of the high temperature prototype reactor THTR-300. The necessary additional financing for SNR-300 amounts to about 1.15 billion DM and will be shared by the government (about 2/3), the utilities and the manufacturers (about 1/3). For THTR-300, the Federal Government and the State of North-Rhine-Westfalia will finance about 40 % of the additional costs of about 1 billion DM, utilities and manufacturers take care of about 10 %, about 50 % of the additional costs will be covered by a collateral loan.

The fast test reactor KNK-II has become critical with the second core loading. The pin diameter of this core is 7.6 mm (the first core had pins with 6 mm diameter). The problem of gas bubbles in the sodium which often led to reactor scram in the first core, was removed by installing cyclons in the primary circuit: the gas then no longer passes through the core. Up to now, KNK-II could be operated without any perturbation.

For the construction of a reprocessing plant of 350 t throughput per year two sites are under discussion at present.

I. REACTOR PHYSICS ACTIVITIES AT THE NUCLEAR RESEARCH CENTER KARLSRUHE

1. Fast Reactor Physics

Experiments on reactivity effects of material displacements in fast reactor accident situations were completed in the single-zone uranium fuelled critical assembly SNEAK-12A. Calculations used current KfK methods and data and partially also the corresponding moduls of the SIMMER-II accident analysis code system.

For all cases investigated satisfactory agreement between theory and experiment was reached when two-dimensional transport eigen-value calculations were used. The application of perturbation theory or diffusion theory in a number of cases led to larger discrepancies, particularly when the experiments involved fuel compaction /1,2/.

SNEAK-12B contains a central Pu-zone with fuel rods instead of fuel plates which is a better simulation of a fast power reactor; the results obtained so far confirm those found in SNEAK-12A /3/.

Within the frame of the DeBeNe-British BIZET program measurements of the worths of simulated control rods for fast power reactors have been made in ZEBRA and SNEAK by the modified subcritical monitoring method (MSM). The assemblies used were the conventional and unconventional core arrangements from the BIZET programme and a compacted version of a conventional core. The control rods were mainly natural B_4C, with some study of 40 % enriched B_4C and Eu_2O_3. Correction factors for the MSM were obtained from eigen-value and source-mode diffusion-theory calculations in XY geometry.

The measured rod worths and interactions are compared with calculated values from methods and data similar to those used by the different participants in the BIZET programme to predict the corresponding parameters in fast power reactors. In general, acceptable agreement is found /4/.

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A review of the experiments and their interpretation in the DeBeNe-French RACINE-program covers especially the determination of critical mass, reaction rate distributions and spectral indices, reactivity of sodium voiding and control rod worth in heterogeneous assemblies of MASURCA.

The analysis is made independently by CEA and DeBeNe using their own calculational techniques and cross sections /5/.

Another review article on the physics of unirradiated LMFBR cores describes how problems have been identified and widely solved over the past ten years /6/. The following table summarizes the status of the prediction capability of present theoretical methods. A brief outline of remaining tasks is also given in the paper.

Conventional collapsing for group cross sections used in multigroup nuclear reactor calculations is usually performed using normal (real; direct) flux weighting. The application of more advanced collapsing procedures using in an appropriate manner real, adjoint and bilinear weighting was in the past restricted in general to fundamental mode problems. Although the principles have been published for more than ten years, there seems to exist little recent experience on the merits and possible difficulties of these improved procedures for multidimensional diffusion problems for practical purposes, e.g. in the nuclear design and analysis of large Liquid Metal Fast Breeder Reactors (LMFBRs). A recent publication explains certain somewhat unusual features of the collapsed group constants obtained by adjoint and bilinear weighting and describes the experience gained in representative 1-dim. and 2-dim. test cases /7/. It could be shown for criticality and perturbation calculations that in general it is advantageous to apply these improved collapsing methods if the necessary precautions are taken. The possible disadvantages seem to be only minor and the associated complications are considered to be tolerable. Compared to the conventional collapsing procedures these improved procedures are especially useful for multidimensional problems. It could be proven that they are favorable with respect to computer time and storage needed due to the fact that the necessary number of coarse groups can be kept fairly small without deteriorating too much the accuracy

Table Survey of the Prediction Capabilities of Present Group Sets in Fast Critical Assemblies. Figures: (C-E)/C in %; exc ≡ exception. (See Ref. 6)

Group Set	k_{eff}	c^8/f^9	f^8/f^9	f^5/f^9
	Adjusted Nuclear Data			
CARNAVAL-IV (France)	-0.5 and better	c^8/f^5: ≈ 4	f^8/f^5: ≈ 4	≈ 0
FGL-5 (UK)	<0.5	-1 to -2	≈ 3	≈ 1
JFS-2 (Japan)	-0.5 and better	only values from adjusted ratios available		
KfKINR (DeBeNe)	0.3 to 1 exc: RACINE (1.3)	3 to 5	-2 to -3 f^8/f^5: -4 to -5	2 to 3
OSCAR-76 (USSR)	±0.6 exc: BFS-3-3 (2.9)	-0.7 to 3	f^8/f^5: -0.5 to 8 (8 for BFS-35-2)	± 2
	Non-Adjusted Nuclear Data			
BNAB-78 (USSR)	-1 to -2	1.5 to 4.5 c^8/f^5: 1.4 to 2.8	f^8/f^5: -3 to +3 exc: BFS-31-4 (-7)	-2 to 3.5
JENDL-2 (Japan)	≈ -1	≈ 7	≈ -1	≈ 4
USA ENDF/B-IV	≈ -1,5	7 to 9	-3 to -9	2 to 3
USA ENDF/B-V	<1 exc: ZPR6/6A (-1)	4 to 5 c^8/f^5: -7 to 4	f^8/f^5: -4 to 8 exc: JEZEBEL (-8)	± 4

Power Profile: All sets predict the power in most parts of the core within 2%, in regions of strong flux gradients to about 3% (conventional cores), increasing up to 5% in heterogeneous cores. In using ENDF/B-IV data and KfKINR, a radial tilt in C/E is observed: overestimation of the power in the outer core regions relative to near center regions: the tilt is enlarged in heterogeneous cores up to 5%.

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and reliability of the coarse group results compared to reference results of corresponding fine group calculations with uncollapsed group constants. This is for instance very advantageous in the calculation of the Na-void-effect with 2- or 3-dimensional reactor models.

To improve the calculation of resonance self-shielding on the basis of multilevel cross sections, a critical comparison was made between D. Cullen's SIGMA1 code for Doppler broadening of tabulated, linearly interpolable cross sections /8/ and the DOBRO code developed at KfK for calculation of Doppler broadened multilevel cross sections directly from resonance parameters /9/. The test problem involved calculation of 3-channel Reich-Moore cross sections for Pu-241 broadened to 900 K. The CPU time needed was shorter and the precision higher with DOBRO, where interpolation errors do not occur, than with SIGMA1.

In the area of fast reactor safety investigations much effort was spent to determine an upper limit for the mechanical energy release in HCDAs for the prototype reactor SNR-300 together with corresponding risk analyses. These studies were performed at the request of the Committee on Future Energy Politics, set up by the Parliament of the Federal Republic.

The experimental validation of complex safety code systems like SIMMER is pursued. As an example, the experimental modelling of the movement of molten clad material under the drag forces of Na-vapour could be described theoretically satisfactorily with respect to pressure lossses and material distribution.

2. Fuel Cycle Analysis for PWRs

The methods and data validation for the analysis of the PWR fuel cycle was documented /10/. At present, the activation of the end-pieces of PWR fuel elements is under investigation; results from calculations will be compared to experiments. Studies on plutonium and uranium recycling in thermal reactors are in progress.

As a special aspect, the neutron source density in a block of vitrified high active waste (HAW) in boron-silicate glass was determined. The knowledge of the neutron source is important for shielding during vitrification and during transport of HAW in the glass product. Boron-silicate glass was selected at KfK because of its high durability and leach resistance. The neutron emission originates from spontaneous fission in heavy nuclides, and from (α,n) processes on light elements as boron, oxygen and others. The investigation was performed for vitrified PWR-HAW. The glass contains about 57 % SiO_2, 14 % Na_2O and 12 % B_2O_3 /11/. The determination of the (α,n)-neutron source was performed on the basis of recently measured neutron yields from (α,n) processes in thick targets /12/. The result is that the (α,n)-neutron source density in the glass is determined to about 80 % by (α,n) processes on B_2O_3. It should be noted that the (α,n) neutrons have energies mainly between 5 and 5.5 MeV! In addition, the contribution from (α,n) processes dominates the spontaneous fission neutrons. When Cm-242 has decayed after about 5 years, the (α,n) high energetic neutron source is still about 50 % of the total neutron source, originating from alphas of Am-241 and Cm-244.

A reduction of the neutron emission can be obtained, if the B_2O_3 concentration is reduced, but then the long-term quality of the glass product might be influenced unfavorably. This topic will be pursued.

3. Studies on Advanced PWRs (Tight Lattice Cores)

The intercomparison of homogeneous and heterogeneous tight lattice PWR-configurations have been pursued. As the nuclear data basis, KEDAK-4 has been adopted. A variety of about 60 critical assemblies (fast and epithermal systems) have been calculated. The agreement in the criticality prediction is satisfactory for the purpose of a consistent analysis of the feasibility studies envisaged. An important constraint of the investigations is to guarantee a sufficiently negative coolant void reactivity feedback for a tight lattice PWR, so that normal PWR licensing procedures can be applied. To study this effect more deeply, operational transients without scram

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(ATWS) were investigated with a varying coolant density reactivity feedback in a homogeneous tight lattice core. It was found that only in a widened lattice, i.e. enlarging the moderator to fuel volume ratio from about 0.5 to 0.6 - 0.7, a sufficiently negative reactivity feedback can be guaranteed. This has the penalty of decreasing the conversion ratio, but to a still acceptable value of 0.9. A consistent comparison of all interesting concepts is in progress; thermal hydraulic investigations concentrate on the improvement of the theoretical basis for tight lattice cores as well as for LOCA and reflood conditions.

A collection of papers on the tight lattice PWR is presented separately to this meeting /13/.

/1/ F.Helm, G.Henneges, W.Maschek,
Measurements and Computation of the Neutron Physics Effects of Accident-Caused Core Distortions in LMFBRs. Submitted for publication in Nucl. Sci. Eng. (1983); NEACRP-A-Report, 26th Meeting (1983).

/2/ H.Küsters,
Validation of Neutronic Calculations for Distorted Core Configurations Arising in Accident Situations of LMFBRs. Summary Report from the 25th Meeting of NEACRP (1982) to CSNI; NEACRP-A-Report, 26th Meeting (1983)

/3/ F.Helm, G.Henneges,
Reactivity Effects of Fuel Rearrangement in Fast Reactor Rod Bundles, ANS-Winter Meeting (1983); NEACRP-A-Report, 26th Meeting (1983)

/4/ H.Giese, S.Pilate, J.M.Stevenson,
Control Rod Worths and Interactions in Fast Reactors. Submitted for publication in Nucl. Sci. Eng. (1983); NEACRP-A-Report. 26th Meeting (1983)

/5/ G.Humbert, F.Kappler, M.Mortini, G.Norvez, G.Rimpault, B.Ruelle, W.Scholtyssek, A.Stanculescu,
Parametric Studies for the Heterogeneous Core Concept in the Framework of the PRERACINE and RACINE Programs Submitted for publication in Nucl. Sci. Eng. (1983); NEACRP-A-Report, 26th Meeting (1983)

/6/ H.Küsters, S.Pilate,
The Present Accuracy of Physics Characteristics of Unirradiated Fast Reactors.
To be published in Annals of Nucl. Energy; NEACRP-A-Report, 26th Meeting (1983)

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/7/ E.Kiefhaber,
Application of Real, Adjoint and Bilinear Weighting for Collopsing Group Constants Used in Space Dependent Neutron Diffusion Problems;
KfK-Report 3430 (1982)

/8/ D.E.Cullen and C.R.Weisbin,
Nucl. Sci. Eng. 60 (1976) 199

/9/ F.H.Fröhner,
Proc. Conf. on Nucl. Data Eval. Meth. and Procedures, BNL 1980, BNL-NCS-51363 (1981), vol. I, p. 375, also available as KfK 2388, Karlsruhe (1980)

/10/ U.Fischer, H.W.Wiese,
Verbesserte konsistente Berechnung des nuklearen Inventars abgebrannter DWR-Brennstoffe auf der Basis von Zell-Abbrand-Verfahren mit KORIGEN,
KfK-Report 3014 (1983)

/11/ H.W. Wiese,
private Communication (1983)

/12/ G.J.H. Jacobs,
Neutron Energy Spectra Produced by α-Bombardement of Light Elements in Thick Targets.
Ph.D. Thesis. University of Eindhoven (1982)

/13/ Investigations on a Tight Lattice PWR in the Federal Republic of Germany,
NEACRP-Report, 26th Meeting (1983)

II. REACTOR PHYSICS ACTIVITIES AT KRAFTWERK UNION

1. Computational Methods for LWR Analysis

Recent Developments in Nodal Reactor Analysis

The progress that has been made in the past few years in the rapidly expanding field of nodal reactor analysis methods is discussed in some detail in a review paper [1] which was presented at the 1983 ANS Topical Meeting in Salt Lake City.

Although the transverse nodal methods appear now to be well established a close examination reveals a number of weak points which require special attention. The extremely accurate results that have been obtained [2,3,4] with transverse nodal methods on an assembly size mesh for several two- and three-dimensional benchmark problems, unfortunately, are not fully representative for practical LWR applications. This high degree of accuracy is obtained for reactors with a fresh initial core, but not necessarily for depleted cores, that is, for situations in which the assumption of nodewise constant cross sections is not fulfilled. As a consequence, systematic errors can accumulate during burnup which for PWRs may become as large a 10 % in the nodewise power distribution at the beginning of a new reload cycle. It is shown that a nonlinear extension of the nodal expansion method [5] allows to nearly eliminate these errors in a relatively straight-forward way with only a small increase in computing time.

Another basic problem is the fact that the existence of nodewise constant homogenized group diffusion theory parameters, which usually is taken for granted in nodal methods, is questionable. Furthermore, the fact that important local information is lost when dealing only with node averaged quantities is a severe weakness of the nodal method. It is not surprising, therefore, that these questions currently receive

increased attention. It was the purpose of the paper [1] to review the more recent attempts to overcome these difficulties. In particular, the work on nodal methods has led to new ideas on consistent procedures of homogenization and dehomogenization [6-8]. The "equivalence theory" is described as an exact method of homogenization and as the theoretical basis for the development of new and powerful approximate methods of homogenization [7-9]. Closely related to this is the inverse problem of reconstructing detailed flux and pin power distributions for the heterogeneous assemblies throughout the core ("dehomogenization"). In a number of recent papers [1,8,10] it has been shown that this can be done with remarkable accuracy.

Although many refinements are still needed it is very likely that the advanced transverse nodal methods, combined with consistent methods of homogenization and dehomogenization, will allow order of magnitude savings without the need for making concessions in accuracy and spatial detail compared to conventional full-core fine mesh finite-difference procedures. These new methods are theoretically well-founded and have the potential for including transport theory approximations in a consistent manner.

Considerable progress has als been made in the development of nodal schemes for the solution of the two-dimensional discrete ordinate equations [11-14]. All these nodal transport methods have in common that a spatial integration over the transverse dimension of the node is performed which converts the 2-D transport equation to a coupled set of one-dimensional discrete ordinate equations. Suitable approximations are then made to represent the one-dimensional spatial dependence of the node-interior sources and the angular fluxes

along the edges of the node. The higher order nodal schemes, while computationally more costly per mesh cell, allow the use of a much coarser mesh to obtain results of comparable accuracy and, hence, yield net savings in both computer time and storage requirements. However, improved methods of convergence acceleration are needed in order to get the full benefit of these new nodal transport schemes. It may be of interest to note that the nodal equivalent acceleration method described in reference 1 was successfully applied to speed up the nodal discrete ordinate calculations with the program MULTIMEDIUM for a number of PWR assembly problems [1].

The Statistical Analysis for the Minimum critical Power Ratio

A critical power ratio of CPR > 1 protects the rods in a fuel bundle against the effect of boiling transitions. The uncertanties in the burnout-correlation (XL-correlation), in fuel characteristics and in the reactor conditions however yield a small probability for a rod with a nominal CPR > 1 to have really CPR < 1. The statistical analysis of the thermo-hydraulics of a BWR determines the mean number of rods being subject to boiling transition in dependence of the minimum critical power ratio (MCPR) in the core.

An analytical code, VASKA, has been developed to carry out these calculations [15]. Fig. 1 shows the results of a typical statistical analysis by the analytical model in comparison to corresponding results yielded by a Monte Carlo code.

Basic input for such an analysis is a "pessimistic" power distribution in the BWR core. The analysis yields a value $MCPR_{99.9}$, a minimum critical power ratio of the core at which a mean of 99.9 % of the rods in the core is protected against burnout. The operational MCPR has to be adjusted in such a way, that even during the worst transient the value of $MCPR_{99.9}$ cannot be reached. It has been shown, that, starting from a realistic power distribution, even during the worst transient none of the allowed 0.1 % rods will be subject to boiling transition.

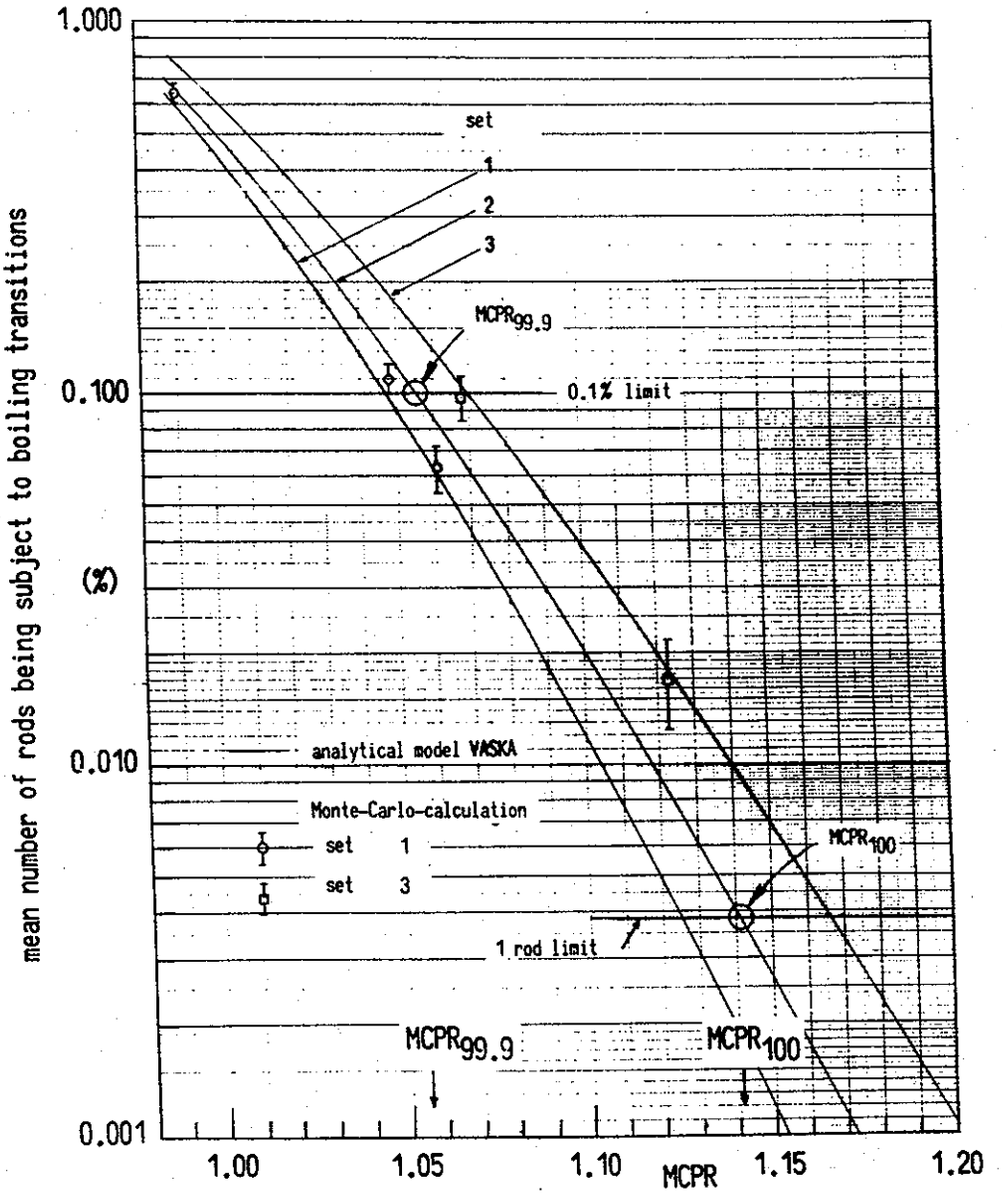


Fig. 1 Mean number of rods (%) being subject to boiling transitions in dependence of the minimum critical power ratio MCPR

Kraftwerk Union

2. Improved Determination of the Power Density Distribution by Means of Gamma-Sensitive Traversing In-core Probes in Boiling Water Reactors

Power distributions in boiling water reactors (BWR's) are calculated by the process computer using flux measurements from traversing in-core probes (TIP's). If gamma-sensitive traversing in-core probes are used instead of thermal neutron TIP's, determinations of power density distributions in a boiling water reactor are improved significantly. The reason is that signals of gamma TIP's are less dependent upon geometrical tolerances or detector position fluctuations within the water gap.

The reliability of KWU-gamma TIP's and the experimental verification of the improvements have been demonstrated at three German BWR's (KWW Cycle-6, KKB Cycle-2, KKP1 Cycle-2). Comparisons of process computer power distribution calculations performed prior to and following installation of gamma-sensitive TIP's indicate improved plant operation margins. Table 1 summarizes the changes that occured as a result of the gamma TIP installation.

	KWW		KKB		KKP1	
	Thermal TIP	Gamma TIP	Thermal TIP	Gamma TIP	Thermal TIP	Gamma TIP
Core Thermal Power	41.7%	41.5%	95.3%	93.5%	99.9%	99.6%
Core Flow	35.6%	35.9%	112.4%	112.5%	101.1%	101.8%
Radial Power Factor	2.11	1.95	1.87	1.76	2.10	1.98
Total Peaking Factor	1.53	1.39	1.43	1.34	1.39	1.30

3. Advanced BWR Core Performance Calculations

The objective of the Advanced BWR Core Performance Calculations (Fortschrittliche Nuklearrechnungen FNR) is to provide improved capacity factors and margins of safety of BWR operations. The additional capabilities are based upon accumulated experience of operation and licensing of BWR plants.

The Advanced BWR Core Performance Calculations are performed independent of the NSS process computer core performance monitoring software on a high speed minicomputer located in the BWR plant. For FNR-system the 32-bit CPU 3287 which is an integrated part of Siemens 300 systems for Process Automation is used. The minicomputer CPU 3287 is produced by Gould Inc., SEL Computer Systems Division.

The minicomputer receives on-line operating data from the process computer or redundant data acquisition system at periodical time intervals permitting real-time operating data to be combined with current state-of-the-art analysis and color graphic display capabilities.

The FNR software consists of the PREDICTOR, which is based upon the 3D-BWR simulator RS3D, and a high speed version of the process computer periodic core performance evaluation program P1. The main purpose of the PREDICTOR is to predict future operating core states, the effect of changing the values of existing parameters, and definition of core performance and characteristics during transient operations. The PREDICTOR system will be able to improve the efficiency of reactor startups maneuvers and steady state operation. It is expected that significant improvements in capacity factor are possible.

The BWR simulator module of the PREDICTOR uses diffusion theory which is valid over the range of conditions involved. Further, a diffusion theory based model is used to adapt the simulated TIP readings of 3D-BWR simulator calculation to the measured TIP readings. This optimizes the accuracy of the BWR simulator

calculations of the TIP readings without changing basic system parameters and allows the flexibility of accurately evaluating changed core conditions (e.g. exposure, core flow or pressure, control rod positions, etc.).

Communication capabilities will permit simple commands and output results. Normal terminal input is complemented by reactor and cycle specific restart disc files which are input from magnetic file. Output is normally on color graphics display terminals or line printers and permanent disc file, and magnetic tapes are used for large volume output.

References:

/¯1_7 M. R. Wagner, K. Koebke, "Progress in Nodal Reactor Analysis", Proceed. of 1983 ANS Topical Meeting, Vol. 2, p. 941, March 28-31, 1983, Salt Lake City, USA

/¯2_7 H. Finnemann, F. Bennewitz, M. R. Wagner, Atomkernenergie, 30, 123 (1977)

/¯3_7 M. R. Wagner, H. Finnemann, K. Koebke, H.-J. Winter, Atomkernenergie, 30, 129 (1977)

/¯4_7 J. J. Dorning, "Modern Coarse-Mesh Methods - A Development of the '70's", Proceed. of ANS Topical Meeting on "Computational Methods in Nuclear Engineering", Vol. 1, p. 3-1, April 23-25, 1979, Williamsburg,Virginia

/¯5_7 M. R. Wagner, K. Koebke, H.-J. Winter, "A Nonlinear Extension of the Nodal Expansion Method", Proceed. of Internatl. ANS/ENS Topical Meeting on "Advances in Mathematial Methods for Nuclear Engineering Problems", Vol. 2, p. 43, April 27-29, 1981, Munich, FRG.

/¯6_7 K. Koebke, "A New Approach to Homogenization and Group Condensation", IAEA-TECDOC 231, p. 303, IAEA Technical Comm. Mtg. Lugano, Switzerland (Nov. 1978)

/¯7_7 K. S. Smith, A. F. Henry, and R. A. Loretz, "The Determination of Homogenized Diffusion Theory Parameters for Coarse Mesh Nodal Analysis", Proceed. of ANS Topical Meeting, Sun Valley, Idaho, Sept. 14-17, 1980, p. 294

/¯8_7 K. Koebke, "Advances in Homogenization and Dehomogenization", Proceed. of Internatl. ANS/ENS Topical Meeting, Munich, April 27-29, 1981, Vol. 2, p. 59

/-9_7 A. Y. Cheng, C. L. Hoxie, A. F. Henry, "A Method for Determining Equivalent Homogeneous Parameters", Proceed. of Internatl. ANS/ENS Topical Meeting, Munich, April 27-29, 1981, Vol. 2, p. 3.

/-10_7 H. S. Khalil, P. J. Finck, A. F. Henry, "Reconstruction of Fuel Pin Powers from Nodal Results", Proceed. of 1983 ANS Topical Meeting, Vol. 1, p. 367, March 28-31, 1983, Salt Lake City, USA

/-11_7 M. R. Wagner, "A Nodal Discrete-Ordinates Method for the Numerical Solution of the Multidimensional Transport Equation", Proceed. of ANS Topical Meeting, Williamsburg, Virginia, April 23-25, 1979, Vol. 2, p. 4-117

/-12_7 R. D. Lawrence and J. J. Dorning, "A Discrete Nodal Integral Transport Theory Method for Multidimensional Reactor Physics and Shielding Calculations", Proceed. of ANS Topical Meeting, Sun Valley, Idaho, Sept. 14-17, 1980, p. 840.

/-13_7 W. F. Walters, R. D. O'Dell, "Nodal Methods for Discrete-Ordinates Transport Problems in (x,y) Geometry", Proceed. of Internatl. ANS/ENS Topical Meeting, Munich, April 27-29, 1981, Vol. 1, p. 115

/-14_7 R. E. Pevey and H. L. Dodds, Jr., Trans. Am. Nucl. Soc. 39, 751 (1981)

/-15_7 M.Schrader, Jahrestagung Kerntechnik, Berlin, June 14-16, 1983 Compacts p.121

III. REACTOR PHYSICS ACTIVITIES AT THE UNIVERSITY OF STUTTGART (IKE)

1 Three-dimensional reactor burn up calculations for PWR

(A. Wörner, D. Lutz, K. Neumann, W. Bernnat)

ENDF/B-IV/V cross section data and neutron physics methods available at IKE have been applied to calculate burn up-dependent microscopic two-group cross sections. With these two group constants also depending on moderator density, fuel temperature and boron density 3D-reactor burn up calculations for two cycles of an operating PWR have been carried out. The deviation from calculated to measured cycle length was + 2,6 % (case A) and - 0,4 % (case B). For one of these cycles (case B) also 2D-calculations were performed with several buckling concepts. But even for the best 2D-case with burn up-dependent bucklings from the previous 3D-calculation the difference was + 4,4 % between calculated and measured cycle length in comparison to - 0,4 % for the 3D-calculation.

2 2/3D cell calculation method with the p^{ik}-concept

(K. Neumann)

To consider more dimensional effects by the spectral calculation for preparation of few-group cross section datas a method is developed on basis of the first collision equation. The first collision probabilities, p^{ik}, used in the first collision equation, are statisically calculated with a Monte-Carlo method for more dimensional geometries represented by the combinatorial geometry. The geometry allows many different subzones for example radial division of pins, fine zones for absorber pins and real 3D arrangements. The calculation of the collision probabilities is divided in two parts, the analysis of geometry and the actual p^{ik}-calculation. Because of this division it is possible to use the same geometry information for different material-zone arrangements, for example burn up dependent problems or multigroup resonance calculations. The calculation time is therefore, not too high. The method has been verified by comparison with Benchmark results.

3 Radiation transport through cavities in twodimensional S_N-Method

(K. Neumann, D. Emendörfer, W. Bernnat)

The calculation of neutron- and photon transport is difficult with the S_N-method in twodimensional geometry including enlarged cavities. For long and narrow cavities it is nearly impossible. Since other methods, like Monte-Carlo, P_N - or first collision source methods are not simple in application, an analytical transport method, imbedded in a S_N-transport code, is developed to calculate radiation transport in cavities.

The directional distribution of neutron fluxes is approximated at every boundary mesh of the single cavity in legendre polynomials, orthogonally in each quadrant. This polynomial-approximation allows the analytical representation of streaming between different mesh elements. Especially, the calculation is exact in small cavities for mesh elements far away from one another. The transfer coefficients of the approximation can be calculated before S_N-calculation and they do not depend on energy group. Coupling flux moments into cavity with transfer coefficients results in flux moments out of cavity during S_N-calculation.

Therefore, no higher S_N-orders are needed. The method is realized for cylindrical and annular cavities in r,z-geometry. It is possible to define several cavities.

The method is coupled with the S_N-code DOT 4.2. The variable attributes of the code are almost conserved.

4 Theoretical Analysis of the Energy Deposition Rate in the Gamma Thermometer due to Neutron and Gamma Sources

(W. Bernnat)

A computational study was performed to investigate the total volumetric energy deposition rate in gamma thermometers as a function of the power distribution in the fuel elements. The study is divided into a radiation transport analysis for the fast neutrons and gammas and in the determination of neutron- and gamma sources in the surrounding fuel pins of the gamma thermometer.

The determination of the energy deposition rate in the gamma thermometer due to neutron and direct or (n,γ)-sources in neighbouring pins was performed by adjoint Monte-Carlo transport calculations for fast neutrons and gammas.

Knowing the adjoint neutron- or gamma flux in a pin surrounding the gamma-thermometer, the contributions of this pin to the energy deposition rate is simply the product of the volume source (neutron or gammas) in the pin and the adjoint flux (integrated over the energy).

26th NEACRP MEETING

Oak Ridge, Oct. 17-21,1983

Reactor Physics Activities in Italy

Compliled by R. Martinelli, ENEA

1. FAST REACTORS

In the framework of the Research and Development Agreement with the French CEA, fast reactor activities at ENEA have been concentrated mainly on the interpretation of integral experiments (RACINE, NEFERTITI and JASON Programmes) and on the validation of the formulaires specific to core, blanket and shielding calculations (including data evaluations and adjustments).

1.1. Core integral experiments (RACINE)

The measurements on configurations 1D and 1E (sodium void effects around simulated control rods, and reactivity and power distributions associated to off-center absorbers, respectvively) have been practically completed /1,2/.

These experiments have been designed, carried out and interpreted according to the methodology established at CEA for the extrapolation of integral parameters -and of their uncertainties- to the design of large power reactor cores /3/. New experiments are planned on that basis, aimed at reducing the uncertainties on the reactivity losses per cycle due to burnup and buildup of heavy isotopes /4/. Much attention in the RACINE programme has been placed on problems relating to the startup of SUPERPHENIX 1 (Source range instrumentation response, control rod calibration methods, etc.). The RACINE 1-S experiment is presently being run /5/, which simulates a chessboard

configuration of the SPX-1 core. The experimental program includes measurements of radial power distributions, reactivity worths of diluent subassemblies and subcriticality margins up to about 30 dollars.

1.2. Blankets

The parametric experimental study of simulated BOC blanket regions (Programme NEFERTITI in the fast source reactor TAPIRO of ENEA, Casaccia) has been completed. The analysis of this large set of experiments is underway.
A supplementary configuration, a mock-up of an EOC blanket (enriched U-235 pins simulating Pu buildup) will be studied in the next months.

1.3. Shielding

The version Nr.1 of the adjusted shielding formulaire PROPANE has been implemented and its satisfactory performance extensively tested /6,7/. A new version is already under development, supported by JASON experimental programme in HARMONIE at CEA, Cadarache.

1.4. Data evaluation, adjustment and processing

A new version of the processing code FOUR ACES has been implemented by ENEA (Bologna). Partial re-evaluations have been made of fission product and transactinide isotopes; most of the files evaluated in the framework of CEA-ENEA cooperation have been recommended for adoption in the first version of JEF.
(The Italian contributions to JEF activities include an effort of critical analysis, intercomparison and selection of available files also for absorbing nuclides such as Gd and Hf, and for structural materials such as Zr).
Finally, an adjustment of the effective cross-sections of actinides has been completed /8/, basing upon the results of irradiation experiments in PHENIX.

Other activities, not connected with CEA-ENEA agreement, are in progress at ENEA (Bologna).

1.5. Neutronic design of PEC core

The analysis of the reference configuration (78 subassemblies) has been completed. In particular, the transport, heterogeneity and mesh-size corrections have been recalculated for control rod worths, leading to a 13% reduction with respect to the worths calculated via standard design routes. The slight value (less than 5%) of rod interaction factors has been confirmed.
Problems related to the start-up of the reactor have been tackled, and the design of in-pile measurements devices is being developed.

1.6. Transport and improved diffusion methods

A generalized Fourier transport (GFT) method has been recently proposed /9/ to treat scattering anisotropy in homogeneous spherical geometry, in view of spectrum and shielding problems. A Fortran formulation and a GFT extension to heterogeneous media are in progress. Improved diffusion models based on the definition of size-dependent diffusive parameters have been proposed for optically small fast systems with different geometries /10/.

2. LIGHT WATER REACTORS

Work is still in progress at ENEL-CRTN (Milano), aiming at development and validation of synthesis and polynomial nodal method (PNM) codes for fast and slow dynamics analysis of LWRs.

2.1. Codes for 3-D Transient Analysis

In recent years, QUANDRY-EN /11/, developed at CRTN on the basic structure of EPRI's coarse-mesh code QUANDRY, has played a key role in assessing the accuracy of the home made synthesis code SYNTH-C /12/, i.e. in defining the error bounds of the final solution.

Comparison were made on the basis of node or assembly-averaged values. The most recent development of QUANDRY-EN by CRTN, including:

- the introduction of generalized equivalence theory concepts for the homogenization of large nodes (proving that the new scheme converges to the same solution as a fully heterogeneous diffusion calculation);
- the application of K-W poynomial interpolation methods /13/ to the converged solution, in order to determine detailed flux and power distributions inside homogeneous nodes;
- the adoption of a more sophisticated thermal-hydraulic module

constitute a significant improvement of the performance of this coarse-mesh code and add to its potential as a tool for SYNTH-C validation.

2.2. Depletion codes: NORMA-3D

The coarse-mesh NORMA-3D /14/ has been implemented by CRTN to simulate the long-term neutronic and thermal-hydraulic behaviour of large PWRs in a tridimensional scheme.

The code is a development of NRMS, a code based on PNM.

The polynomial nodal approach was chosen because:

- the converged solution contains the necessary information to determine directly the detailed power distribution inside each node by the polynomial interpolation method developed by Koebke and Wagner
- the PNM is able to deal easily with several neutron groups.

After extending NRMS to 3 dimensions, CRTN implemented various additions to the basic structure, namely:

- a diagonal and rotational geometry
- a new treatment of the transverse leakage and rebalance schemes
- an implicit treatment of the burnup

- a simplified thermal-hydraulic model
- the determination of the pin power distribution
- the management of the fuel assemblies.

At present, work is in progress in order to enable NORMA-3D to deal with the "equivalent parameters" derived from the "equivalence theory" and with a more complex thermal-hydraulic model for the determination of the thermal limits of the core. A detailed account of the computational accuracy and efficiency of PNM and of NORMA-3D in particular, is given in /15/.

2.3. Measurements at Caorso BWR

A series of dynamics tests during the second fuel cycle of the 900 MWe Caorso BWR power station have been planned and licensed. Such tests, conducted by ENEA and ANSALDO, are scheduled to start at the end of October 1983, and will include stability analysys studies. A measurement campaign aimed at characterizing the neutron environment inside the primary containment of the plant, was started in March 1983. This campaign is based -like a previous one made in 1981 inside the reactor drywell- on the use of the Multiple Foil Activation technique. Seven irradiation positions have been chosen for the instrumented stations, designed and installed by ENEL-CRTN in cooperation with CESNEF (Milano) /16/. The analysis of the results obtained at the end of cycle 2 will also be aimed at the validation of CRTN's shielding calculation methods, thus completed the information drawn from the calculation-experiment comparisons made at the conclusion of the first campaign /17/.

3. CRITICALITY SAFETY

A considerable calculational effort has been spent at ENEA, Fuel Cycle Dept., in producing and testing reference libraries (219 and 123 groups, AMPX-2) and in validating Montecarlo codes

(mainly KENO-IV) for calculations related to criticality safety of reprocessing plants (storage pools and final product cells) and to spent fuel transport cask safety verifications. More than sixty critical and subcritical experiments have been analyzed in the process of library and code validation : the most recent results of this work are reported in /18/.
As a contribution to the second part of CSNI Benchmark on fuel transport casks /19/, ENEA (Bologna) has accurately investigated the effects on k_{eff} values of different ways of modelling the (very) complicated array and of the spatial treatment of resonances. Further contributions are planned in this context, i.e. a study of the stability of Montecarlo solutions for large arrays and of the effect of packaging material.

4. NOISE ANALYSIS

The analysis, performed by ENEA Casaccia, of LPRM and TIP signal fluctuations in Caorso at full power, indicate the presence of a second transit time of the coolant flow in the upper half of the core. The implications in terms of separation in the distribution of coolant flow velocities are being studied. Preliminary conclusions are presented in /20/.
Acoustic and thermal noise are being analyzed in other in-pile and out-of-pile experiments, namely :

- vibrations of instrumented tubes and of recirculation pumps in Caorso (by ENEA Casaccia)
- vibrations of rotating machinery, like a SPX-1 pump shaft in a sodium loop (by ENEA Bologna)
- temperature fluctuations of conventional and intrinsic, fast response (SS-Na) thermocouples on a LMFBR fuel pin irradiated in a sodium loop at SILOE (CEA Grenoble), for early detection and follow-up of the evolution of failures in the pin (by ENEA Bologna).

REFERENCES

/1/ M.Martini, Private communication.

/2/ U.Broccoli, Private communication.

/3/ G.Palmiotti, M.Salvatores, Private communication.

/4/ G.Palmiotti, M.Salvatores, Private communication.

/5/ R.DeWouters, M.Martini, Private communication.

/6/ J.P.Trapp, A.DeCarli, 6th ICRS, Tokyo 1983

/7/ J.C. Cabrillat, V.Rado et al., 6th ICRS, Tokyo 1983

/8/ G.Oliva, Private communication.

/9/ G.Ghinassi et al., Nat.l Seminar on Reactor Physics,Bologna 1983

/10/ F.Premuda, RT/FI(83)4

/11/ E.Brega et al., CRTN-N5-21 (1979)

/12/ E.Brega et al., CRTN-N5-19 (1979)

/13/ K.Koebke, M.R.Wagner, Atomkernenrgie 30 (136-142) 1977

/14/ E.Brega et al., CRTN-N5/82/06

/15/ E.Brega et al., submitted to Annals of Nuclear Energy

/16/ E.Borioli et al., CRTN-N1/83/ (in press)

/17/ P.Barbucci et al., 6th ICRS, Tokyo 1983

/18/ P.A.Landeyro et al., Int.l Seminar on Criticality Studies, Dijon 1983

/19/ CSNI Restricted Report Nr. 78

/20/ A.Federico et al., S.M. on In-Core Instrumentation, Halden 1983

Reactor Physics Activities in Japan
(September 1982 - September 1983)

Compiled by
T. Asaoka (JAERI) and K. Shirakata (PNC)

Thermal Reactor Physics

A core performance calculation program, COMS (Core Operation and Management System), was developed for predicting present and future power distributions as a BWR on-line core management system to meet with demands on plant availability and operational flexibility. The program consists of a nodal coupling type nuclear-thermalhydraulic model and is adapted to the current core state by using measured TIP (traversing in-core probe) data. The adaption algorithm has been verified through the simulation, where root mean square (RMS) errors of calculated TIP readings to the measured ones are 5% in future prediction and 3% in present estimation.[1] In addition, the TARMS software package was developed as an effective on-line, on-site tool for BWR core management. To obtain highly accurate nodal powers and the core critical eigenvalue, newly-developed methods were implemented to automatically minimize errors included in in-core-neutron-flux-monitor readings. A sort of machine-learning method was also developed to minimize the errors caused by simplicity of the physics models adopted in TARMS. Comparisons of TARMS to experimental data, including those from gamma-scannings and TIP's, have shown RMS errors of about 3% in nodal powers (to the gamma scannings) for the core monitoring, and of less than 4% (to the TIP's) in the core performance prediction.[2]

On the other hand, a new iteration method for solving one-group diffusion equation was developed for an efficient core performance prediction on the basis of an analytic solution technique. Test calculations for LWRs have indicated that the method can achieve a significant reduction in computing time and memory in comparison with the conventional finite difference method.[3]

The development of a standard computer code system for nuclear calculations, SRAC (Standard Reactor Analysis Code), has been completed and the benchmark calculations have shown that the system can predict nicely the experimental k_{eff} values for various types of critical assemblies.[4] On the other hand, the research of adaptability of vector processing to large-scale nuclear codes has been proceeded to test vectorized versions of DOT-3.5, TWOTRAN and ANISN based on the finite difference method, PALLAS-2DCY and BERMUDA on the direct integration method, and so on. The gain obtained from the vectorization was investigated in relation to the numerical method, geometry and problem type.[5),6)]

As for the testing of the data and methods, temperature coefficients were obtained from 1- and 2-dimensional criticality calculations and 1-dimensional perturbation calculation, and the results were compared with the measured values for 3 loading patterns in a light-water cooled core of KUCA.[7] In addition, an evaluation model of fast neutron irradiation dose in JMTR was investigated in relation to the radiation damage of materials by introducing exposure parameters such as the displacement per atom and the damage fluence.[8]

Concerning the FUGEN, a heavy-water moderated, boiling lightwater cooled, pressure tube type reactor, an on-line core performance evaluation system ATROPOS was developed to carry out safe and efficient reactor operation by offering detailed useful information on such items of core performance as the thermal power, power distribution and thermal operation limits. The system has been verified by using the start-up test data from the FUGEN initial core.[9] On the other hand, a new cluster analysis code MESSIAH was applied to calculate reactor physics parameters measured in the critical facility DCA for the FUGEN. The MESSIAH code utilizes the collision probability method to solve the neutron transport equation. The calculated reactor physics parameters, especially the micro-parameters, agree fairly well with the experiment, but the calculated void reactivity in doller unit is slightly smaller than the experimental value as shown in Fig. 1, which is probably attributed to an over-prediction of the diffusion constant.[10] In addition, the reactor physics behavior of Gd burnable poison fuel pins was studied through measurements of reactivity change, coolant void reactivity, local power distribution and thermal neutron flux distribution at the DCA.[11] Furthermore, a feasibility study on the use of thorium fuels was performed for the plutonium natural uranium mixed oxide fuelled FUGEN. The study has indicated that the use of Th fuels even in a simple form of only straight Th on a once-through cycle basis, results in many improvements in reactor performance.[12]

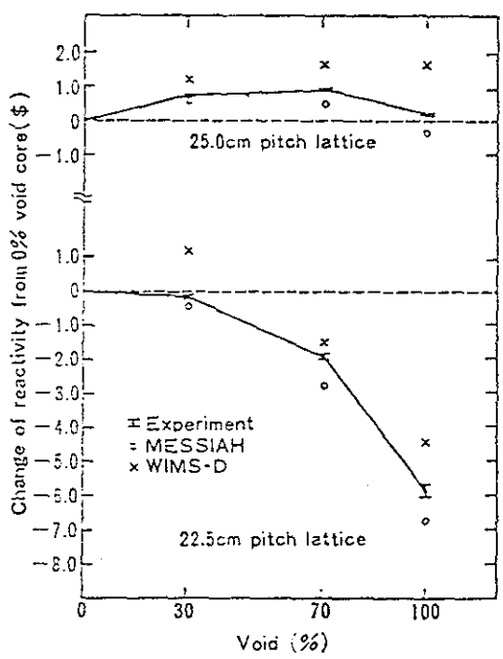


Fig. 1 Comparison of void reactivity with experiment

For the investigation of nuclear characteristics of FUGEN HWR Demonstration Plant, parameter studies on reactor physics have been continued by using DCA. The items of the recent studies are as follows;

(1) Reactivity worth of B-10 solution has been measured. Boron-10 was filled in a cylinder which simulates the guide tube of control rod for the nuclear simulation of rapid injection system of B-10 solution adopted in the design of the Demonstration Plant.
(2) Temperature coefficients of reactivity on both H_2O coolant and D_2O moderator have been measured for Pu lattices by raising the whole core temperature up to 80℃.
(3) Radial thermal neutron flux distribution has been measured on the whole Pu core.

1) Fukuzaki T., Mitsuta T. et al. :"Core Performance Calculation Program for On-Line Core Management", J. Atomic Energy Soc. Japan, 25, 639 (1983) (in Japanese)
2) Tsuiki M., Uematsu H. et al.:"TRAMS : An On-Line Boiling Water Reactor Management System Based on Core Physics Simulator", Topical Meeting on Advances in Reactor Computations, Salt Lake City (1983)
3) Itagaki M.:"Analytic Solution Technique for Solving One-Group Diffusion Equations for Core Simulations", J. Nucl. Sci. Thechnol., 20, 627 (1983)
4) Tsuchihashi K., Takano H. et al. :"SRAC: JAERI Thermal Reactor Standard Code System for Reactor Design and Analysis", JAERI 1285 (1983)
5) Ishiguro M. and Tsutsui T. :"Vector Processing of the Neutron Transport Codes", JAERI-M 82-199 (1983) (in Japanese)
6) Harada H., Higuchi K. et al. :"Vectorization of Nuclear Codes on FACOM 230-75 APU Computer", JAERI-M 83-024 (1983) (in Japanese)
7) Wakamatsu S., Hashimoto K. et al. :"Calculation of Temperature Coefficients for the Light-Water-Moderated Core of Kyoto University Critical Assembly", J. Atomic Energy Soc. Japan, 24, 963 (1982) (in Japanese)
8) Sakurai F. and Niibo T. :"Evaluation Method of Fast Neutron Irradiation Dose in JMTR", J. Atomic Energy Soc. Japan, 25, 372 (1983) (in Japanese)
9) Natori H., Kaneto K. et al. :"Development of On-Line Core Performance Evaluation System for FUGEN", J. Atomic Energy Soc. Japan, 24, 792 (1982) (in Japanese)
10) Kadotani H. and Hachiya Y. :"Analysis of Heavy-Water-Moderated, Cluster-Type Fuel Lattices by Cluster Physics Code MESSIAH", J. Nucl. Sic. Technol., 19, 689 (1982)
11) Wakabayashi T. and Minatsuki I. :"Critical Experiments on Gadolinium Poisoned Cluster-Type Fuel Assemblies in Heavy Water Lattices", Nucl. Sci. Engng., 83, 50 (1983)
12) Haga T. :"A Feasibility Study on Use of Thorium Fuel in Pu MOX Fuelled FUGEN-HWR", Japan-US Seminar on Thorium Fuel Reactor, Nara (1982)

Fusion Neutronics

At the Fusion Neutronics Source (FNS) facility, the angular distribution and the energy spectrum of the source neutrons were measured for a water cooled type tritium target shown in Fig. 1. The MORSE-GG code was used to calculate the neutron transport and the secondary gamma-ray emission and transport in the target assembly. As seen from Fig. 2, a fairly good agreement between the experimental result and the calculated value has been obtained for the neutron spectra in forward direction to the T beam line ($\theta=0^{\circ}$). For the spectra at other angles, however, some discrepancy is observed in the shape of the source peak and of the tail below 1.7 MeV, due probably to an inappropriate consideration of the detector position and the pulse shape in the accelerator operation.[1)]

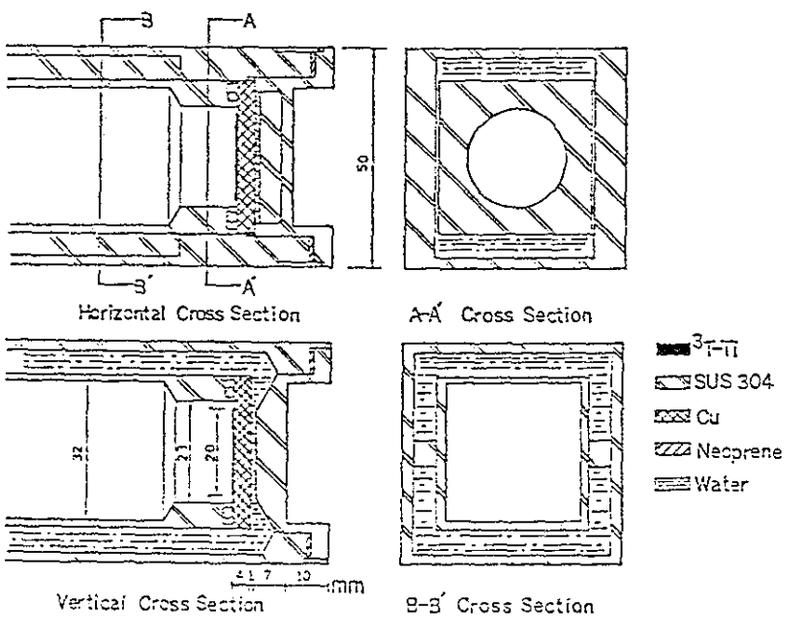
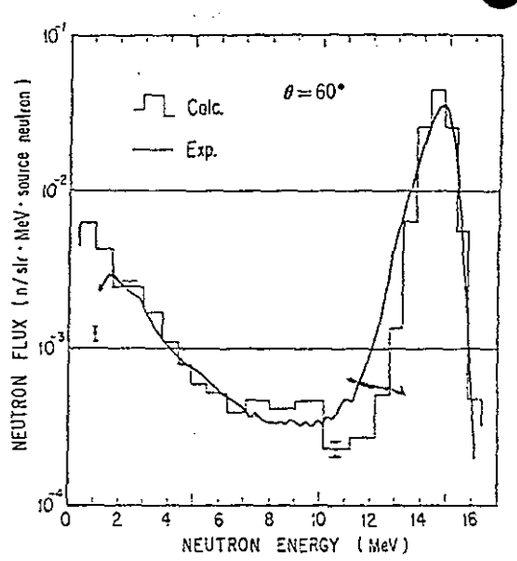
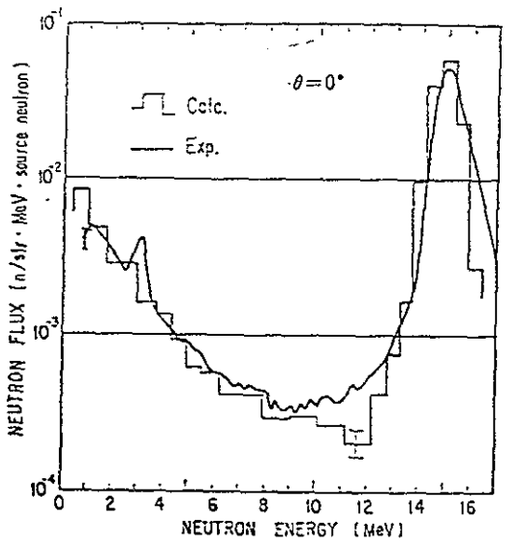


Fig. 1 Cross-sectional view of target assembly





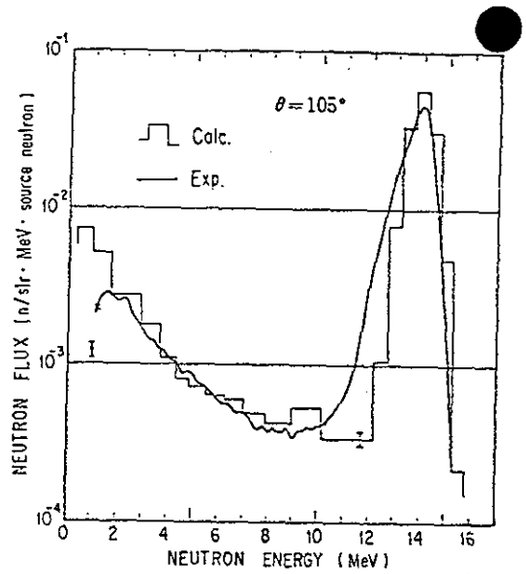


Fig. 2 Measured and calculated neutron spectra at 0°, 60° and 105°

Angular dependent neutron leakage spectra from Li_2O slab assemblies were measured also at the FNS by using the time-of-flight method[2] and the data were analyzed with a new discrete ordinates code BERMUDA-2DN based on the direct integration method in a multigroup model.[3] As seen from Figs. 3 and 4, the calculated spectra agree well with the observed values from the viewpoint of the absolute comparison. In addition, at the OKTAVIAN facility, measurements of leakage angular neutron spectra from slabs of typical shield materials were carried out by means of the time-of-flight technique.[4]

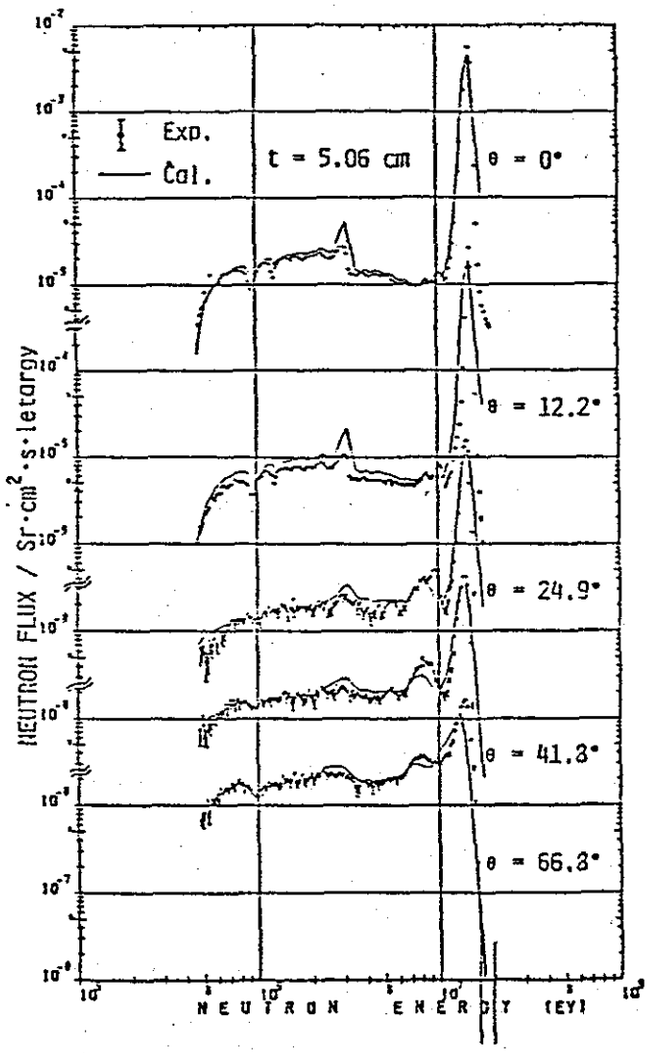


Fig. 3 Angle-dependent leakage spectra from Li_2O slab assembly (5.06 cm in thickness)

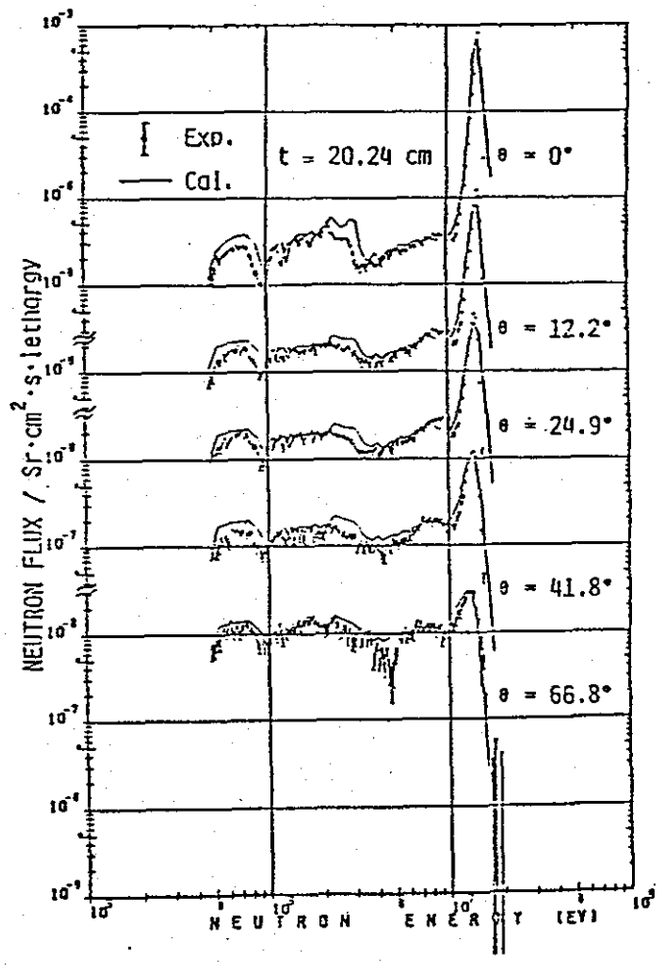


Fig. 4 Angle-dependent leakage spectra from Li_2O slab assembly (20.24 cm in thickness)

Detailed neutronics analysis was performed on a tokamak fusion experimental reactor (FER) of a swimming pool type shown in Fig. 5. The Monte Carlo code MORSE-I, a modified version of the MORSE-GG to consider toroidal geometry, asymmetric torus cross section and neutron source distribution in plasmas. As a result of the analysis, it becomes clear that a modification of the blanket structure and the material composition should be made to improve the tritium breeding performance and to reduce the nuclear heating rate of the vacuum vessel in the diverter zone.[5),6)]

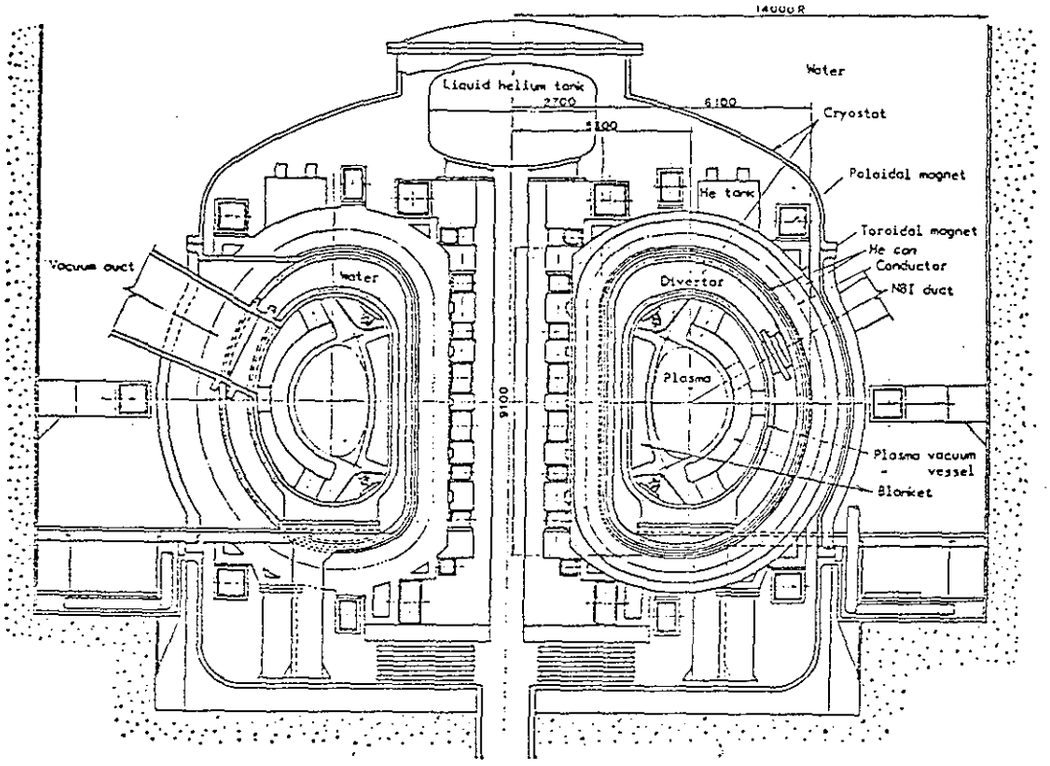


Fig. 5 Vertical cross section of SPTR

A study was made on the nuclear characteristics of the blanket/shield design of a D-D tokamak reactor. The graphite blanket of 1 m thichness has the characteristics of much smaller residual radioactivity, afterheat and biological hazard potential compared to other materials.[7)] In addtion, neutronics analysis was carried out to assess the tritium breeding capability of the heliotron-H reactor design by using the ANISN code and the MORSE-I code.[8)] Furthermore, neutronics properties of a laser fusion reactor were discussed on the basis of one-dimensional neutron transport calculations in burning D-T plasmas and blankets.[9)]

In addition, computational models for spallation and fission reactions were evaluated for developing an accelerator breeding and transmutation code system NMTC/JAERI[10)] which performs the Monte Carlo simulation of nuclear reactions in a heterogeneous target. By running the NMTC/JAERI code for thin targets of Bi, Pb, Th and U in the energy range of 50-1000 MeV, proton and neutron non-elastic and fission cross sections were

derived from the counts of real collisions and fission events in the targets, to compare the results with the experimental data.[11] Furthermore, the energy spectra of neutrons emitted by thick targets of C, Fe, Cu and Pb to the incident of 30- and 52-MeV protons were obtained by unfolding the pulse height distribution measured with an NE-213 scintillator.[12]

1) Seki Y., Oyama Y. et al. :"Monte Carlo Calculations of Source Characteristics of FNS Water Cooled Type Tritium Target", J. Nucl. Sci. Technol., 20, 686 (1983)

2) Maekawa H., Oyama Y. et al. :"Measurements of Angular Flux on Surface of Li_2O Slab Assemblies and Their Analysis by a Direct Integration Transport Code BERMUDA", ANS Fifth Topical Meeting on Technology of Fusion Energy (1983)

3) Suzuki T., Hasegawa A. et al. :"BERMUDA-2DN: A Two-Dimensional Neutron Transport Code", Sixth International Conf. Radiation Shielding (ICRS), Paper 3b-2 (1983)

4) Yamamoto J., Takahashi A. et al. :"Measurement and Analysis of Leakage Neutron Spectra from SS-316, Concrete, Water and Polyethylene Slabs with D-T Neutron Source", Sixth ICRS, Paper 4b-9 (1983)

5) Mori S., Seki Y. et al. :"Neutronics Design of Tritium Breeding Blanket for Fusion Experimental Reactor", J. Nucl. Sci. Technol., 20, 154 (1983)

6) Mori S., Mohri K. et al. :"Nuclear Analysis of Blanket and Shield Design for Tokamak Fusion Experimental Reactor", Sixth ICRS, Paper 5b-9 (1983)

7) Nakashima H., Tsukahara K. et al. :"Nuclear Analysis of Blanket/Shield Design for D-D Tokamak Fusion Reactor", J. Nucl. Sci. Technol., 19, 663 (1982)

8) Nakashima H., Ohta M. et al. :"Tritium Breeding Capability of Heliotron-H Fusion Reactor Blankets", J. Nucl. Sci. Technol., 19, 762 (1982)

9) Ido S., Nakai S. et al. :"Neutronics Calculations in Pellets and Blankets of Laser Fusion Reactor Concept SENRI-I", J. Nucl. Sci. Technol., 19, 1019 (1982)

10) Nakahara Y. and Tsutsui T. :"NMTC/JAERI, A Simulation Code system for High Energy Nuclear Reactions and Nucleon-Meson Tranport Processes", JAERI-M 82-198 (1982) (in Japanese)

11) Nakahara Y. :"Evaluation of Computational Models for Fission and Spallation Reactions Used in Accelerator Breeding and Transmutation Analysis Code", J. Nucl. Sci. Technol., 20, 511 (1983)

12) Nakamura T., Fujii M. et al. :"Neutron Production from Thick Targets of Carbon, Iron, Copper and Lead by 30- and 52-MeV Protons", Nucl. Sci. Engng., 83, 444 (1983)

Shielding

Concerning the 3-dimensional transport method, discrete ordinates codes PALLAS-XYZ and PALLAS-RTZ have recently been developed for dealing respectively with (x,y,z) and (r,θ,z) geometries on the basis of a direct integration method to the integral transport equation,[1] and the PALLAS-XYZ has been applied to analyze an experiment of fast neutron streaming through a large void duct[2] and to analyze detailed neutron fluxes in a PWR pressure vessel.[3] In addition, a double finite element method where both the space and the angle finite elements are employed, was applied to solve the multigroup neutron transport equation by using the Galerkin's weighted residual method and the variation method.[4),5]

As for streaming problems, a series of measurements of 14 MeV D-T neutrons streaming through a slit and a duct in concrete shields was carried out using a Cockcroft-Walton-type neutron generator and the results have been shown to agree well with the calculated values with the MORSE-CG code.[6),7] At the FNS facility, 2 types of streaming experiments were performed to obtain benchmark data for verifying the data and methods.[8] Furthermore, another FNS benchmark experiment on D-T neutron and secondary gamma-ray streaming through a concrete bent duct was analyzed with the MORSE-GG code to give a good agreement with each other.[9),10]

The albedo Monte Carlo code MORSE-ALB was improved to treat the deep penetration of radiation in shielding configuration of large scale geometry. For this geometry, the Monte Carlo-Monte Carlo, or discrete ordinates-Monte Carlo coupled calculation is advantageous, and the arbitrary coupling surfaces were made to be applicable to forward and adjoint calculations. Besides, newly added to the MORSE-ALB code were several functions, such as next event surface crossing flux estimator for cylindrical and spherical shell geometry, importance sampling method based on the exponential transformation, and volume sources for various configuration. The improved code was applied to the analyses of the shielding experiment on the fast neutron source reactor YAYOI and the streaming measurement on the primary coolant pipe of JOYO.[11]

Concerning the shielding measurement of JOYO, several measurements were performed to obtain the shielding characteristics data. Neutron flux distributions and energy spectrum were measured by using activation foils in sodium in the reactor vessel, in graphite shield and in reactor pit room. The measured data will be analyzed by two-dimensional discrete ordinates code and albedo Monte Carlo code.

The FFTF/JOYO Shielding Data Exchange Meeting was held in Tokyo on May 12 and 13, 1983, between USDOE and PNC. Following the meeting, an analysis has been proceeded for the shielding

characteristics of FFTF, and the calculation will be compared with the measurement to predict the reliability of shielding calculations.

As concerns the shielding of spent fuel transport casks, shielding experiments were performed using PWR spent fuel assemblies to obtain the benchmark data for evaluating a code system of shielding safety analysis. The first step analysis was carried out for examining the present status of the code system composed of ORIGEN2, ANISN and DOT-3.5.[12] In addition, integral experiments with a Cf-252 source were performed for a cask as designed and that lost its resin shield. The measured neutron and secondary gamma-ray dose rates were compared with the Monte Carlo calculation using the next-event surface crossing estimator.[13),14]

1) Takeuchi K. and Kanai Y. :"Development of a Series of PALLAS Discrete-Ordinates Direct-Integration Codes", Sixth International Conf. Radiation Shielding (ICRS), Paper 3b-1 (1983)

2) Sasamoto N., Takeuchi K. et al. :"Analysis of Neutron Streaming Through Void Duct with Three-Dimensional Transport Code PALLAS-XYZ", Sixth ICRS, Paper 6a-4 (1983)

3) Takeuchi K. and Sasamoto N. :"Analysis of Detailed Neutron Fluxes in a PWR Pressure Vessel by Two- and Three-Dimensional PALLAS Transport Codes", Nucl. Technology, 62, 207 (1983)

4) Fujimura T., Nakahara Y. et al. :"Application of Space-and-Angle Finite Element Method to the Three-Dimensional Neutron Transport Problems", Sixth ICRS, Paper 3b-8 (1983)

5) Fujimura T., Nakahara Y. et al. :"Solution of Three-Dimensional Neutron Transport Equation by Double Finite Element Method", J. Nucl. Sci. Technol., 20, 620 (1983)

6) Hashikura H., Fukumoto H. et al. :"Neutron Streaming Through a Slit and Duct in Concrete Shields and Comparison with a Monte Carlo Analysis", Nucl. Sci. Engng., 84, 337 (1983)

7) Hashikura H., Oka Y. et al. :"Fast Neutron Streaming Studies Using the Fast Neutron Source Reactor, YAYOI and a 14 MeV Neutron Generator", Sixth ICRS, Paper 6b-4 (1983)

8) Nakamura T., Oyama Y. et al. :"Radiation Streaming Studies at the Fusion Neutronics Source (FNS) Facility", Sixth ICRS (1983)

9) Tanaka S., Oyama Y. et al. :"A Benchmark Experiment on D-T Neutrons and Secondary Gamma Rays Streaming Through a Concrete Bent Duct", JAERI-M 82-130 (1982)

10) Seki Y., Tanaka S. et al. :"Monte Carlo Analysis of a Streaming Experiment of D-T Neutrons and Gamma Rays Through a Concrete Bent Duct", Sixth ICRS, Paper 6b-6 (1983)

11) Kawai M., Hayashida Y. et al.:"Application of Albedo Monte Carlo Method to FBR Neutron Streaming Analysis", Sixth ICRS, Paper 6a-2 (1983)

12) Tanaka S., Sakamoto Y. et al. :"Shielding Experiments for a Shielding Safety Evaluation Code System of Spent Fuel Transport Cask", Sixth ICRS, Paper 7-6 (1983)

13) Ueki K., Inoue M. et al. :"Validity of the Monte Carlo Method for Shielding Analysis of a Spent-Fuel Shipping Cask : Comparison with Experiment", Nucl. Sci. Engng., 84, 271 (1983)

14) Ueki K., Yamakoshi H. et al. :"Investigation of the NESX Estimation in the Monte Carlo Calculations for Shielding Analysis of a Cask", Sixth ICRS, Paper 3a-7 (1983)

15) Miura T. and Sasamoto N. :"Experimental Study of Neutron Streaming Through Steel-Walled Annular Ducts in Reactor Shields", Nucl. Sci. Engng., 83, 333 (1983)

Fast Reactor Physics

1. Experiments at FCA

According to the modification of the Fast Experimental Reactor "JOYO" to MK-II, a series of mockup experiments were carried out on FCA Assembly X from April 1982 to February 1983. The major modifications considered on the mockup experiments were: (1) increase of plutonium content in fuel material and (2) replacement of the uranium blanket by the stainless-steel reflector.

FCA Assembly X consisted of three different versions of Assembly X-1, X-2 and X-3. The first two assemblies were clean cores with no control rod positions and were constructed for a physics mockup study. To make clear the effects of the stainless-steel reflector, systematic experiments were made in the clean core with the uranium blanket (Assembly X-1) as well as that with the simulated reflector (Assembly X-2). Measurements were made for criticality, fission rate and sample worth distributions, and B_4C rod worths.

The third assembly (Assembly X-3) was an engineering mockup core of JOYO MK-II which included six sodium channels to simulate the control rod positions. In Assembly X-3, mainly measured were simulated B_4C rod worths and distortion of neutron flux distribution due to insertion of Na channels and/or B_4C control rods.

Experimental studies on fundamental physics aspects of conventional large fast reactor cores are in progress on FCA Assembly XI. The first version of the assembly (Assembly XI-1) went critical at the end of February 1983. The assembly has a central test region of 60cmϕ x 90cm height simulating the core composition of a homogeneous fast reactor and a driver region mainly fuelled with ^{235}U. Measurements are being made for criticality, reaction rate and sample worth distributions, and sodium-void and Doppler reactivity worths.

Various TLDs with different effective mass numbers are being irradiated in Assembly XI-1 to experimentally determine the gamma-ray heating of the core and blanket materials in fast reactors. ^{235}U and ^{238}U foils are being irradiated in the assembly to obtain the detailed information on breeding performances, together with on neutron flux distribution near the core-blanket boundary.

2. Two-Dimensional Benchmark Problems

Two-dimensional benchmark problems were set up for the ten fast critical assemblies with clean cores, FCA-V-1, FCA-V-2, FCA-VI-1, FCA-VI-2, ZPR-6-7(Ref), ZPR-6-7(H240), ZPR-6-6A, ZPPR-2, ZPPR-9 and SEFOR. In order to perform the benchmark calculations, the data bank which consists of the data for the geometries, the compositions and the experiments and of the correction factors were produced. Furthermore, two-dimensional benchmark calculation code system was generated. Using this system, the benchmark tests were performed for the JFS-3-J2 set.

3. Group Constant Set JFS-3-J2

The group constants for ^{181}Ta, ^{151}Eu, ^{153}Eu, ^{237}Np, many fission products and the lumped nuclides of ^{235}U, ^{238}U and ^{239}Pu were generated to advance the JFS-3-J2 set for

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application to experimental analysis and design study of large fast reactors. The SLAROM code was improved so as to be able to use the JFS-3-J2 set.

4. Development of Calculational Method

Concerning the method development, the finite Fourier transformation method was applied to solve the multigroup neutron diffusion equation for 2-dimensional triangle geometry. Sample calculations for a fast breeder reactor have shown that the method gives good results with fewer mesh points than the usual finite difference method.[1)]

A new mehtod was developed for first-order perturbation theory, in order to reduce the computing and labor cost of analysis of experiments on fast critical assmblies.[2)]

An effective homogenization method of control rods, which preserves the integrated reaction rates in a heterogeneous channel by iteratively changing the cross sections used in a homogeneous super-cell calculation, has been extended to treat off-center control rod channels in FBR. An albedo at the super-cell surface was combined with collision probabilities to treat the neutron leakage. The method has been applied to the central rod worth calculation in a typical demonstration LMFBR, and to the 1-D off-center rod worth calculation.[3)]

To treat the neutron drift in a asymmetric cell of fast critical assembly drawer, a formula has been derived for calculating the drift coefficient based on collision probability method. It was utilized to calculate homogenized diffusion parameters for asymmetric plate cells of the ZPPR-9 core. A flux distribution was calculated in a one-dimensional core model with the drift coefficient, and was compared with that obtained from a transport calculation treating plate-wise heterogeneity. When using the drift coefficient the flux depression in lower energy was well reproduced in the core center, which was caused by the presence of ^{238}U plate.

5. Reactivity Analysis of Pin and Plate Cores

The reactivities of plate and pin ZEBRA-CADENZA cores were analyzed at Osaka University and JAERI, and the results were presented at the NEACRP specialist meeting held on 21-23 June, 1983 at Winfrith.

6. Analysis of JUPITER Experiments

The gamma-dose rate distributions in ZPPR-9 and -10D assemblies, measured with the TLD detectors, were analyzed using JENDL-2 neutron data and ENDF/B-IV gamma-ray production data. The gamma-ray source distributions were calulated by 7 groups XY and RZ diffusion model, and the gamma-ray distributions were calculated by 20 groups XY and RZ S_8P_3 method. Results are as follows.

(1) Analytical results are consistent between ZPPR-9 and ZPPR-10D.

(2) Dose rates in the core region are predicted within an error of 10%. However. dose rates in CRPs tend to be underpredicted.

(3) Dose rates in the blanket region are underpredicted by about 10%.

Analyses of reaction rate distributions in ZPPR-9 and -10 were refined by considering both the shim rod and the cell asymmetry effects for all the measurements. As the results, the spatial dependence of C/E value was rather enhanced.[4)]

The reference analysis of the JUPITER-II experiments, the radially heterogeneous large LMFBR core critical experiments, was started. Cell models to be applied to heterogeneous core configurations have been studied.

7. Effect of Cell Model on Heterogeneous Core Parameter Calculation

Effect of cell model on predicting the nuclear characteristics of a heterogeneous critical assembly has been investigated by analysing the physics experiments made at ZPPR-7A. The investigation has been made using data from 3 kinds of cell models representing the core and inner blanket regions. The first cell model is similar one used for homogeneous LMFBR analysis. The other cell models are composed of core drawers followed by blanket drawers, taking account of the interaction effects between adjacent different kinds of subassemblies. The principal effects of cell model for ZPPR-7A are 0.43% $\Delta k/kk'$ for criticality, 5% for $^{238}U(n,f)$ reaction rate distribution and 15% for Na-void reactivity effect.

8. Double Heterogeneity Effect of Fuel Pin and Subassembly in a Fast Power Reactor[5)]

The double heterogeneity effect due to the fuel pin and the subassembly is estimated for neutronics parameters of a prototype fast power reactor. Both of the heterogeneity effects caused by the flux fine structure and the resonance shielding are taken into account. The model of the hexagonal unit subassembly consists of the smeared fuel, the wrapper tube, and the outer sodium regions, where the average cross sections of the smeared fuel region are obtained by a unit pin cell calculation in cylindrical geometry. The heterogeneity effect of the whole reactor model is calculated based on two-dimensional diffusion theory and perturbation theory. The double heterogeneity effect is found to be 0.5%Δk for k_{eff}, the positive sodium-void worth is reduced by 26%, and the negative Doppler reactivity increases by 7% for a prototype fast breeder reactor. These results are considerably larger than the estimates made by earlier workers.

9. Analysis of Heterogeneity Effect in FCA-VI-2 by Monte Carlo Code VIM

A continuous energy Monte Carlo Code VIM which has been developed in ANL has been converted into the FACOM M200 computer. This code can be used for verifying the accuracy of nuclear characteristics calculated with the code based on multigroup deterministic theory.

The analysis of heterogeneity effect for pin and plate cell in FCA-VI-2 was performed by using the VIM code. The results calculated with VIM were in a good agreement with those calculated with the SRAC and/or SLAROM code based on the collision probability method.

10. Burnup Analysis of JOYO MK-I Core

The burnup characteristics of JOYO MK-I core were calculated based on the actual operational data of the reactor, and were compared with the post-irradiation data. In case of a low burnup of 20,000 MWD/T, the C/E values are summarized as follows.

Effective multiplication factor	:	0.993
Control rod worth	:	1.024
Burnup reactivity loss	:	0.96 ~ 1.02
Burnup	:	0.93 ~ 0.98

11. Conceptual Design Study of He-Cooled Actinide Burning Fast Reactor.

A conceptual design study of a He-cooled actinide burning fast reactor was made as a transmutation system of waste actinides, in addition to the integral experiments to measure fission rate ratios and sample reactivity worths in 8 assemblies of FCA to evaluate the nuclear cross sections of the major actinides.[6)]

Some other reports on fast reactor physics[7),8),9),10)], published in this period, are also given in the reference.

1) Yokoo T. and Kobayashi K. :"Solution of the Multigroup Diffusion Equation for Two-Dimensional Triangular Regions by Finite Fourier Transformation", Nucl. Sci. Engng., 83, 415 (1983)
2) Azekura K., Kawashima K. et al. :"Simple Method for Evaluating Mesh Size Effects in Control Rod Worth Analysis", J. Nucl. Sci. Technol., 20, 518 (1983)
3) Ono S. and Takeda T. :"A Homogeneization Method of Control Rods with Neutron Leakage Effect", NEACRP-A- (1983)
4) Kato Y. et al. :"Radial Dependence of C/E Value in JUPITER-I Core Analysis", NEACRP-A- (1983)
5) Nakagawa M. and Inoue H. :"Double Heterogeneity Effect of Fuel Pin and Subassembly in a Fast Power Reactor", Nucl. Sci. Engng., 83, 214 (1983)
6) Murata H. and Mukaiyama T. :"Fission Reactor Studies in View of Reactor Waste Programme", Third International Conf. Emerging Nuclear Energy Systems, Helsinki (1983)
7) Takeda T. and Wachi E. :"Interference Effect of Neutron Streaming Between Different Fast Critical Assembly Cells", Nucl. Sci. Engng., 81, 551 (1982)
8) Kamei T. and Yoshida T. :"Error due to Nuclear Data Uncertainties in the Prediction of Large Liquid-Metal Fast Breeder Reactor Core Performance Parameters", Nucl. Sci. Engng., 84, 83 (1983)
9) Akiyama M., Furuta K. et al. :"Measurements of Gamma-Ray Decay Heat of Fission Producets for Fast Neutron Fissions of U-235, Pu-239 and U-233", J. Atomic Energy Soc. Japan, 24, 709 (1982) (in Japanese)
10) Akiyama M., Furuta K. et al. :"Measurements of Beta-Ray Decay Heat of Fission Products for Fast Neutron Fissions of U-235, Pu-239 and U-233", J. Atomic Energy Soc. Japan, 24, 803 (1982) (in Japanese)

National Programs

1. JOYO

At the Experimental Fast Reactor "JOYO", the MK-I core has been converted to the MK-II core since January, 1982, and its initial criticality was achieved on November 22, 1982. The MK-II core is to be used as an irradiation bed for fuel and material.

The nuclear and hydrodynamic experiments of the Mk-II core were done at low power. Results of various kinds of experiments were in good agreement with predicted values. The power ascension tests were started on February, 1983, and the Mk-II core reached its rated power, 100MWt, in March.

A series of core performance characteristics experiments, including reaction rate distribution and reactivity worths due to substitutions of core assemblies by irradiation rigs, were carried out from April to July.

The first cycle of normal operation, 45 days long, was started in August, and so far the reactor has been operated satisfactory. The reactor will be operated in the way of four cycles per year.

2. MONJU

The safety evaluation work for "MONJU" by the Nuclear Safety Commission of Japan was started in May, 1982 and completed in May, 1983. Pre-construction works in the site are now being conducted and the excavation for the base mat will be started in June, 1985. The negotiation between PNC and manufacturers for the contract of the plant is in progress and will be concluded soon. Achievement of the initial criticality of the reactor is expected on March,1991.

The Codes of Practice on design and construction of LMFBR power plant are now being studied by the regulatory body for MONJU.

3. Demonstration FBR

Design studies have been continued for the Demonstration FBR Plant, 1000 MWe class subsequent reactor to MONJU. The reactor is expected to begin construction in early 1990s, to demonstrate its performance, reliability and safety as a commercial-scale reactor and to confirm the economic prospect for future commercialization. Design criteria, systems and equipments have been reviewed mainly from the point of view of reducing capital cost.

4. FUGEN

The fourth refuelling was carried out during the scheduled shutdown in June 1982. The third annual inspection and the fifth refuelling was carried out from September 1982 to January 1983 on schedule, and FUGEN had continued stable full power operation for about 7 months until the scheduled shutdown on August 1983 for the sixth refuelling.

12 of 40 fuel assemblies discharged in this refuelling were the initial loaded ones and all the initial loaded fuel assemblies had been discharged from the core.

120 MOX and 160 UO_2 fuel assemblies, including four special fuel assemblies have been discharged for the six refuellings. The maximum burnup of MOX fuel assembly is

13,600 MWD/t, and no leaking fuel has been found for 1,014 effective full power days of operation up to the end of August 1983.

5. FUGEN-HWR Demonstration plant

Since the construction program of the 600MWe FUGEN-type HWR demonstration plant was fixed with the decision given by Japan AEC in 1982, PNC had made its design studies of the demonstration plant with cooperation of EPDC (Electric Power Development Company), the primary undertaker of the plant. These studies were completed in June 1983. The further design activities will be followed by EPDC to get the construction permit, and PNC is actively cooperating with EPDC mainly for R&D and MOX fuel fabrication of the plant.

It is expected now that the construction will start in 1988 and that the commercial operation will start in 1994.

6. Experimental Very High Temperature Gas-Cooled Reactor (VHTR)

The current use of nuclear energy is restricted in electric power generation which contributes around 30% of total primary energy supply. Therefore, an application of nuclear energy to non-electric sector will be a matter of special importance.

Since the final energy demand in the process temperature range below 900°C amounts to about 60% of the total, and new technologies such as coal gasification and thermochemical hydrogen production require also the process temperature of around 900°C, much stable supply of energy for industries will be realized provided that nuclear process heat in a temperature of about 900°C can be supplied at a reasonable price.

In 1969 JAERI started the design activity for the development of the experimental VHTR and at present is performing its detailed design compiling the results of a series of the design studies carried out thereafter.

The experimental VHTR is a helium-cooled, graphite-moderated reactor supplying the thermal output of 50MW. The functions of the experimental VHTR are as follows:

1) Demonstration test for nuclear process heat applications,
2) Irradiation test for development of fuel and material for high-temperature use,
3) Confirmation test for VHTR plant safety.

The basic specifications of a candidate core are planned as summarized in Table 1, and the concepts of the core and its fuel block are shown in Figs. 1 and 2, respectively.

As concerns R&D of the experimental VHTR, the test loop for large components, HENDEL (Helium Engineering Demonstration Loop), is now in operation. In addition, the reconstruction of the critical experimental facility, SHE, is under way to simulate the core condition of the experimental VHTR.

Table 1 Specification of the Candidate Core

Design Items	Specifications
Core dimension (Diameter/Height)	2.7/5.6 m
No. of fuel columns	54
No. of fuel blocks per column	8
No. of fuel pins per fuel block	15
Inside diameter of reactor vessel	6.2 m
Outlet coolant gas temperature	950°C

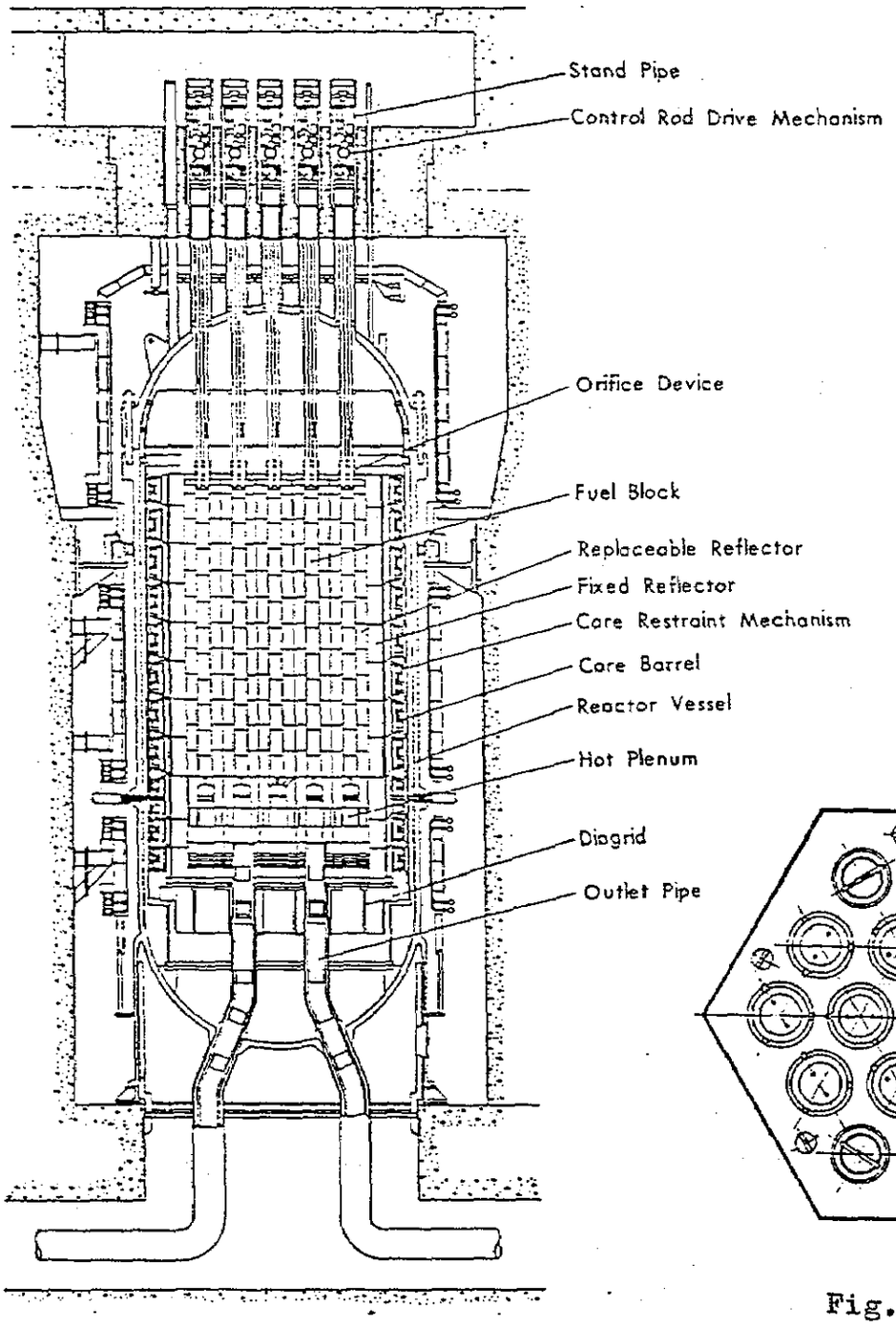
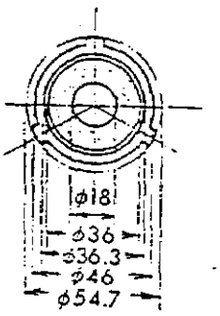


Fig. 1 General view of the candidate core



Cross Section of Fuel Pin

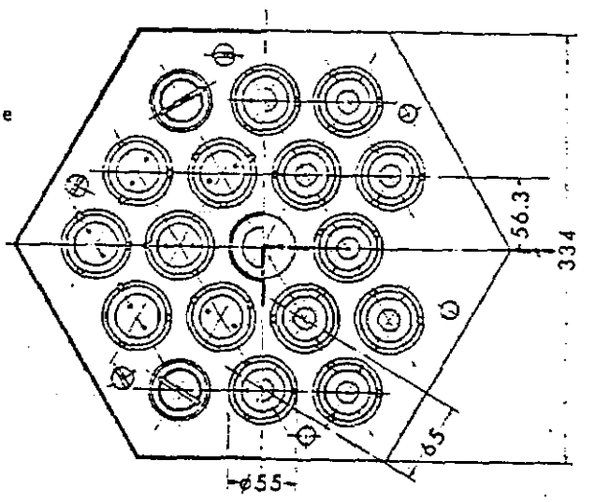


Fig. 2 Fuel block for the candidate core

Report on the Reactor Physics Activities in the Netherlands in the Period July 1982 - July 1983

compiled by M. Bustraan

1. Reactor Physics at the Energy Research Centre Netherlands at Petten

1.1. Support to fast breeder reactor development*

Fission-product data

A review of the status of fast capture cross-section evaluations for important fission-product nuclides is given in Refs. |1,2|. This work has been used as a starting point for detailed recommendations of evaluations for the Joint Evaluated data FILE (JEF-1). Recently, a common report of ECN and ENEA (Bologna) was completed, with recommendations for the 60 most important fission products to be inserted in JEF-1, also including suggestions for re-evaluations, to be made for JEF-2. Furthermore, a programme has been defined to perform integral-data tests of the JEF-1 file, in cooperation with CEA, Cadarache.

Work is in progress to redefine pseudo fission product cross sections based upon JEF-1. Recently, we have initiated a new evaluation of ^{129}I, in cooperation with R.E. Schenter, HEDL, Richland, U.S.A.

Activation of the sodium coolant

Recently, the project to evaluate activation cross sections of corrosion products, cover-gas nuclides and other nuclides in the primary cooling circuit of a fast power reactor has been completed |3|. Revisions have been made for ^{22}Na and ^{64}Zn, based upon recent measurements in the resolved resonance range. We have also compiled 26-group constants for natural Ar, that is used as a cover-gas of the coolant. Further updatings are planned for the Ni-isotopes in cooperation with CEA, Cadarache. The work will be directed to satisfy the needs for re-evaluations of the JEF data file.

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* DeBeNe cooperation on fast reactor development.

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1.2. Advanced water-cooled reactors

Upon request from KfK, Karlsruhe we have calculated group constants for fission products in the 69-group WIMS structure for application in design calculations of advanced water cooled reactors, based upon the ENDF/B-V data file. An appropriate microflux weighting spectrum was used.

1.3. Nuclear data for fusion reactors

As a contribution to the European Fusion Technology Programme (blanket technology) we have developed nuclear-model codes for the prediction of angle-energy correlated distributions of neutron emission cross sections |4-8|. In the existing evaluations these correlations are completely neglected, at least for the continuum emission. Work is in progress to improve the ENDF/B-IV evaluation of lead, that will be used in blanket configurations of future power reactors.
Further work on fusion reactor nuclear data is reported in semi-annual progress reports |9|.

References

|1| H. Gruppelaar, Status of recent fast capture cross section evaluations for important fission product isotopes, Proc. NEANDC/NEACRP Specialist's Mtg. on Fast-neutron capture cross Sections, Argonne 1982, NEANDC(US)-214/L (1983) p. 473.

|2| H. Gruppelaar et al., Report of the working group on fast-neutron capture cross sections for the most important fission-product nuclei, ibid, p. 570.

|3| H. Gruppelaar and H.A.J. van der Kamp, Evaluation of activation cross sections of corrosion products, cover-gas nuclides and other nuclides in the primary cooling circuit of a fast power reactor, Proc. Int. Conf. on Nuclear Data for Science and Technology, Antwerp, 1982, p. 643, Reidel Publ. Co., Dordrecht (1983).

|4| J.M. Akkermans, A. random walk in the land of precompound decay, ECN-121 (1982).

|5| H. Gruppelaar, C. Costa, D. Nierop and J.M. Akkermans, Calculation and processing of continuum particle-emission spectra and angular distributions, Proc. Int. Conf. on Nuclear Data for Science and Technology, Antwerp, 1982, p. 537, Reidel Publ. Co., Dordrecht (1983).

|6| C. Costa, H. Gruppelaar, and J.M. Akkermans, Energy dependence of preequilibrium angular distributions, Lett. al Nuovo Cim. 36 (1983) 431.

91100098

|7| C. Costa, H. Gruppelaar and J.M. Akkermans, Angle-energy correlated model of preequilibrium angular distributions, Phys. Rev. C28 (1983), p. 587.

|8| J.M. Akkermans, Random-walk model of precompound decay II: Stochastic uncertainties in the lifetimes and cross-sections. Z. Physik 313 (1983), p. 83.

|9| J.D. Elen (comp.), Fusion Technology Programme Semi-Annual Report January-June 1982, ECN-124 (1982), and ibid., July-December 1982, ECN-132 (1983).

1.4. Reactor Neutron Metrology

REAL-80 project

The final report on this project has been published in February 1983 |10|.

The aim of the interlaboratory REAL-80 exercise, organized by the IAEA, was to determine the state-of-the-art in 1982 of the capabilities of laboratories to adjust neutron spectrum information on the basis of a set of experimental activation rates, and to subsequently predict the number of displacements in steel, together with its uncertainty.

The input information distributed on magnetic tapes to participating laboratories comprised values, variances and covariances for a set of input fluence rates, for a set of activation and damage cross-section data, and for a set of experimentally measured reaction rates. The exercise dealt with two clearly different spectra: the thermal ORR spectrum with 19 reaction rates, and the fast YAYOI spectrum with 12 reaction rates. Cross-section data were supplied both in a 620 groups structure and in a 100 groups structure. From 30 laboratories which were asked to participate, 13 laboratories contributed 33 solutions for ORR, and 35 solutions for YAYOI.

The spectral shapes of the solution spectra showed considerable spread, both for the ORR and the YAYOI spectrum. When the series of predicted activation rates in nickel and the predicted displacement rates in steel derived for all solutions is considered, one cannot observe significant differences due to the adjustment algorithm used. The largest deviations seem to be due to effects related to group structure and/or changes in the input data.

When comparing the predicted activation rate in nickel with its available measured value, we observe that the predicted value (averaged over

all solutions) is lower than the measured value: 1 per cent lower for ORR and 7 per cent lower for YAYOI.
For the predicted displacement rate in steel we observed a coefficient of variation of 2,2 per cent for the ORR spectrum and 2,8 per cent for the YAYOI spectrum, if all of the participant responses are considered.

|10| W.L. Zijp, E.M. Zsolnay, H.J. Nolthenius, E.J. Szondi, G.C.H.M. Verhaag, D.E. Cullen, C. Ertek, "Final report on the REAL-80 exercise", Report ECN-128 (Also INDL(NED)-7; BME-TR-RES-6/82). (Netherlands Energy Research Foundation ECN, Petten, Febr. 1983).

1.5. Experience with the On-Line Power Reactor Noise Monitoring System for the Borssele (PWR) Power Reactor

For nearly two years the Borssele nuclear power plant (PWR, 450 MWe) is connected to the on-line noise monitoring system at Petten |11| by a special telephone link (200 km). By this system it was possible to follow the 9th fuel cycle (full power operation as well as shutdown) and the start-up of the 10th fuel cycle.

Noise characteristics of several primaire system sensors such as reactor safety channels, core exit thermocouples, pressure transducers and several signals of the secondary system are continuously monitored and reported. Signals from the in-core neutron detectors and core exit thermocouple are used to determine core physics parameters by the noise analysis, such as core coolant flow velocity, and reactivity effects. Local vibration of the in-core instrumentation guide tube and thermocouple response characteristics ("in-situ test") were determined on-line. Using the signals from the ex-core neutron detectors and with aid of a spectrum decomposition, the amplitudes of the core support barrel motions and the direction of the motion are continuously monitored.

In the coarse of the year, the"partial and multiple coherence "analysis and "univariate (UAR) and multivariate (MAR) autoregression" techniques were applied to the on-line measured reactor signals. Thus monitoring of physical core parameters and sensor characteristics by the current on-line noise analysis |12|, will enhance the reliability of the operation of the power plant.

The final object of this work is the development of on-line methods which permit monitoring of the mechanical status of the main components in the primary system and of the complete neutronic and thermodynamic behaviour of the system. It is intented to include and integrate into this system other noise monitoring systems such as on flow acoustics and on loose parts monitoring in the near future.

|11| E. Türkcan, Review of Borssele PWR Noise Experiments Analysis and Instrumentation. Prog. Nucl. Energy 9 (1982) 437.

|12| E. Türkcan, R. Oguma, Improved Noise Analysis Methods for On-line Determination of In-Core Instrumentations and Power Reactor Parameters. Paper to NEA-Committee on Reactor Physics Specialists Meeting on In-Core Instrumentation, OECD Halden Reactor Project, 10-13 October, 1983.

2. Reactor Physics at the Interuniversity Reactor Institute (Delft)

The first part of a reactor noise analysis programme for the Dodewaard BWR has been concluded. Results, mainly pertaining to two-phase flow velocity measurements and the determination of the at-power reactivity transfer function, were published in a D.Sc. theses |1|. A report has been issued on analysis of signals from fixed in-core SPND's in the same BWR |2|. A first series of in-core measurements with a Twin Self Powered Gamma Detector shows promising results. The signals from the TSPGD exhibit a phase relationship characteristic for a combination of a global effect and a local effect with transport time. In the near future a direct comparison with a neutron detector is expected to give further information on the usefulness of gamma noise measurements. A general paper on the interpretation of incore noise measurements in BWR's was published |3|.

The 2 MW pool type research reactor HOR has been a subject for thermal hydraulic analysis, analyzing both stationary and noise signals of thermocouples in an instrumented fuel assembly, also in connection with the possible future application of Low Enriched Uranium (LEU) in this reactor |4|.
An out-of-core boiling loop has been completed which will be used in support of a measurement programma on an in-core loop in HOR. This programme aims at the study of neutron, gamma and temperature noise signals in an electrically heated dummy fuel element with two-phase flow.

A theoretical study on a very high temperature gaseous core reactor was completed |5,6| and an improved method for handling continuum inelastic neutron scattering has been published |7|.

References

|1| Erik Kleiss, On the determination of Boiling Water Reactor Characteristics by noise analysis. D.Sc. thesis, Delft, University of Technology, Delft University Press, 1983.

|2| E.B.J. Kleiss, Investigations of the signals of Co-SPND' in the Dodewaard reactor. Report IRI-131-82-03.

|3| H. van Dam, Interpretation of In-core Noise Measurements in BWR's. Kernenergie 26 (1983) 59-63.

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|4| H. van Dam and J.W. de Vries, Research activities relevant to the application of LEU at the HOR, Report IRI-130/131-82-01.

|5| R.M. Uleman, The influence of the core temperature distribution on the reactivity of a gas core reactor. M.Sc. thesis, Delft, 1982.

|6| H. van Dam and J.E. Hoogenboom, Physics of a gaseous core reactor. To be publ. in Nuclear Technology.

|7| J.E. Hoogenboom, Consequences of inelastic discrete-level neutron-collision mechanics for inelastic continuum scattering. Ann. Nucl. En. 10 (1983) 19-29.

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3. Reactor Physics at KEMA (Arnhem)

Gadolinium

17 fuel elements, each containing 5 pins with gadolinium (out of 35 pins), have been loaded in cycle 14 of the Dodewaard reactor. The reactivity behaviour of the core indicates that the burnout rate of the gadolinium is underpredicted by about 10%. This seems to be consistent with gamma scanning results of gadolinium fuel elements that have been measured pin by pin after one cycle of burnup.

Optimalization

Our optimalization program has been expanded to three dimensional quadrant geometry. This became necessary as the reactor can be controlled by only 2 control rods, that cannot be represented in octant geometry. Running times of 8 hours on a Univac 1160H are typical to obtain the solutions. Simple calculations show that our optimization goal of maximum cycle length does not necessarily lead to maximum discharge burnup. The program is now being adapted to pressurized water reactors.

FLARE type kernels

The kernels as used in FLARE cannot easily be interpreted in terms of collision probabilities. At present we are studying alternative formulations, which maintain in one group theory the migration area for point sources.

3D transient calculations

For a PWR we have performed a 3D deboration transient study. A front with lower borium concentrations passed through the reactor in the shutdown conditions. Local supercriticality was achieved and severe damage would be the result under the assumed conditions.

Noise analysis

As a fruit of the noise work of IRI (see par. 2) noise methods are applied in the Dodewaard reactor to in-vessel and in-core velocity measurements using thermocouple and neutron signals. This enables a better characterization of the thermohydraulic condition under which the reactor operates. Attention has been paid to the measurement of thermohydraulic stability for which also calculations are under way.

91100104

STATUS REPORT TO NEACRP
(1982 - 1983)

Reactor physics activities in Norway,
September 1982 - August 1983

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1. FMS CODE SYSTEM

Activities in reactor physics is concentrated on applications and further development of FMS, the modular code system for light water reactor calculations developed originally at the Institute for Energy Technology (the previous Institute for Atomic Energy). Present and future developments, and maintenance, of the system is now the responsibility of the international consulting company Scandpower A/S, Kjeller. The main users of FMS for fuel management work and plant operation support are power utilities and other organisations in Europe, USA and Japan.

91100105

1.1 Status of RECORD

Conversion of the production version of RECORD to an IBM computer has been completed, thus both CDC and IBM code versions are available.

A major gamma physics project has been carried out with the purpose of including a model for gamma sensitive in-core detectors in RECORD. A generic data base has been produced using the AMPX-DOT code package. The model may be applied both for calculating detector response parameters and gamma heating in various regions within the fuel assembly.

Extension of the fission product model, inclusion of Hafnium as a possible control rod material, and development of a model for hybrid BWR control blades are under way.

The main features of the RECORD code for application to BWR and PWR fuel assembly calculations, and a review of the code qualification, are described in a published Kjeller Report (ref. /1/).

1.2 Status of PRESTO

BWR Code Version

A comprehensive qualification project of PRESTO-B has been carried out by Carolina Power and Light Company of Raleigh N.C. (USA) and documented in a Topical Report to the US NRC. Comparisons with reactor operating data and gamma scan data from Brunswick 1, Brunswick 2 and Quad Cities 1 plants were made. Nuclear data for PRESTO were generated with RECORD. A schematic of the RECORD/PRESTO code system is shown in Fig. 1.2.1.

The cumulative average eigenvalue for the six operating cycles analysed was

$$k_{eff} = 0.99710 \pm 0.00180$$

The average standard deviation between measured and calculated local (24 nodes) TIP ratings for 3 cycles of Brunswick 1 was 8.2%. The corresponding standard deviation for axially integrated TIP's was 4.5%.

The measured versus calculated La-140 assembly peak-to-average average difference for the Quad Cities 1, 1976 gamma scan was -2.26 ± 1.55%.

PWR Code Version

The PWR version of PRESTO has now obtained production status and has been qualified and prepared for PWR operations support.

Three operating cycles of H.B. Robinson were analysed in detail with PRESTO-P, using RECORD nuclear data.

Comparisons were made with measured radial and axial power distributions, boron letdown cycles, BOC boron endpoint data, hot, zero power critical configurations and isothermal core temperature coefficients. Good results were obtained throughout. An example of power distribution comparison is shown in Figs. 1.2.2 and 1.2.3.

2. REACTOR PHYSICS ACTIVITIES AT THE OECD HALDEN REACTOR PROJECT

A prototype of the core surveillance system SCORPIO has been developed within the Halden Project, and is today operating at the Halden computer laboratory. One module in this system, the core simulator, has been validated against power reactor (PWR) data from power/xenon transients. Another module, the strategy generator, is capable of predicting control rod positions, coolant temperature and boron concentration as input to the simulator, such that the resulting axial power offset during transients is kept close to zero. A more advanced strategy generator, based on optimal power distribution control, will be implemented during 1984.

The Halden Project has during 1983 established an experimental control room including a full scope training simulator. This laboratory will partly be used for testing and demonstration of computerized operator support systems.

REFERENCES

/1/ T. Skardhamar, H.K. Næss:
"Methods of RECORD, an LWR Fuel Assembly Burnup Code", Kjeller Report IFE/KR/E-11 V (1982).

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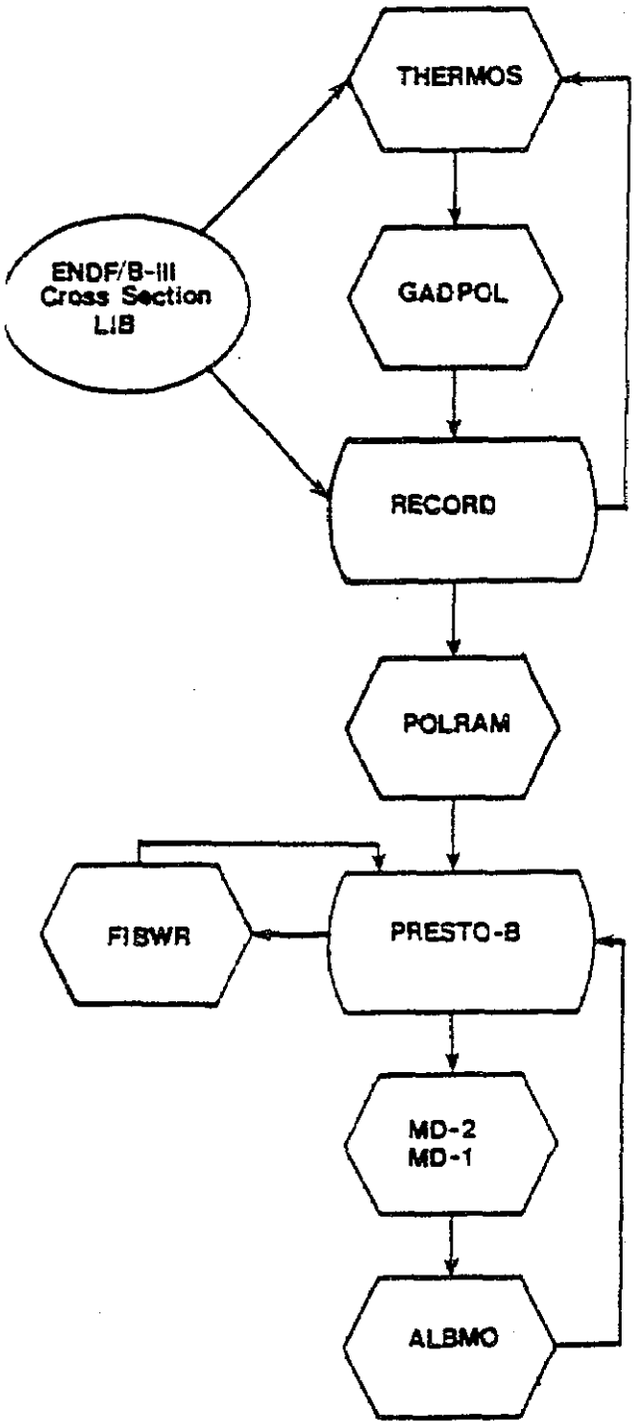
FIGURE 1.2.1

MAJOR PROGRAM MODULES USED FOR CP&L STEADY STATE BWR ANALYSIS



THERMOS: Transport theory code for burnable absorber cross-section generation

GADPOL: Interface code which provides reaction rate matching between transport and diffusion theory

RECORD: Two-dimentional multi-group code for generation of LWR lattice physics constants

POLRAM: Interface code for generation of polynomial cross-section fits of record data for use by PRESTO-B

PRESTO-B: Three-dimentional coupled neutronic and hydraulic BWR nodal simulation code

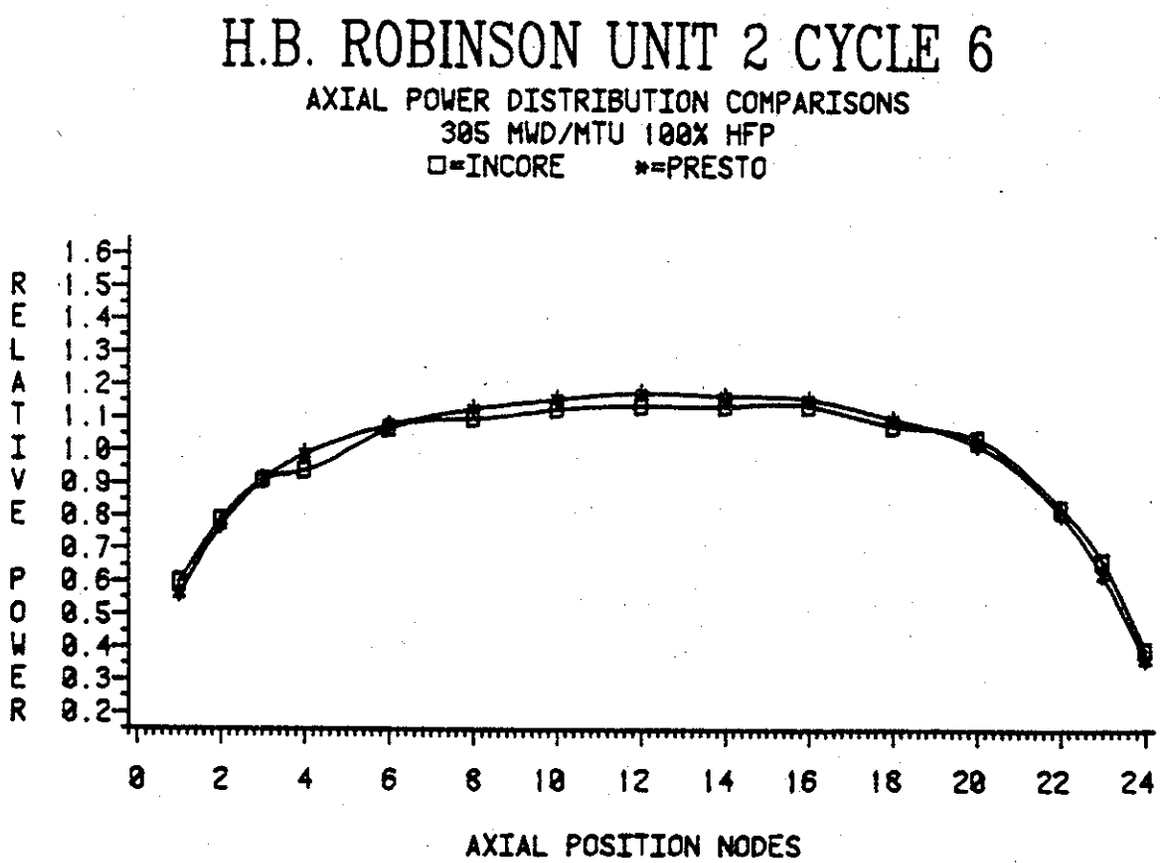
FIBWR: BWR Thermal-hydraulic code for generation of reference pressure and flow distributions

MD-2: Two-dimensional, multi-group, fine mesh diffusion theory code for evaluation of reference flux distributions

MD-1: One-dimensional analog of MD-2

ALBMO: A program module within PRESTO-B for generation of core leakage parameters

FIGURE 1.2.2


H.B. ROBINSON UNIT 2 CYCLE 6
AXIAL POWER DISTRIBUTION COMPARISONS
305 MWD/MTU 100% HFP
□=INCORE *=PRESTO
RELATIVE POWER
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0 2 4 6 8 10 12 14 16 18 20 22 24
AXIAL POSITION NODES

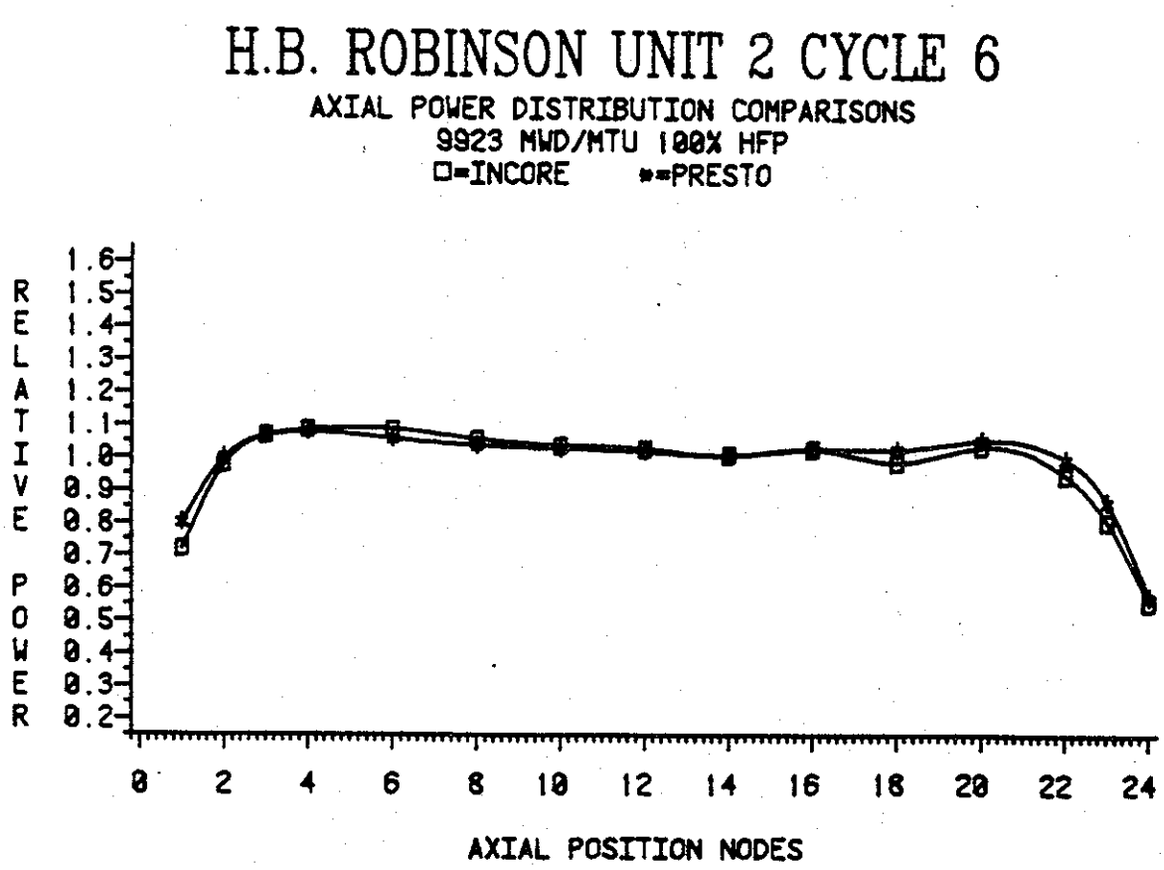
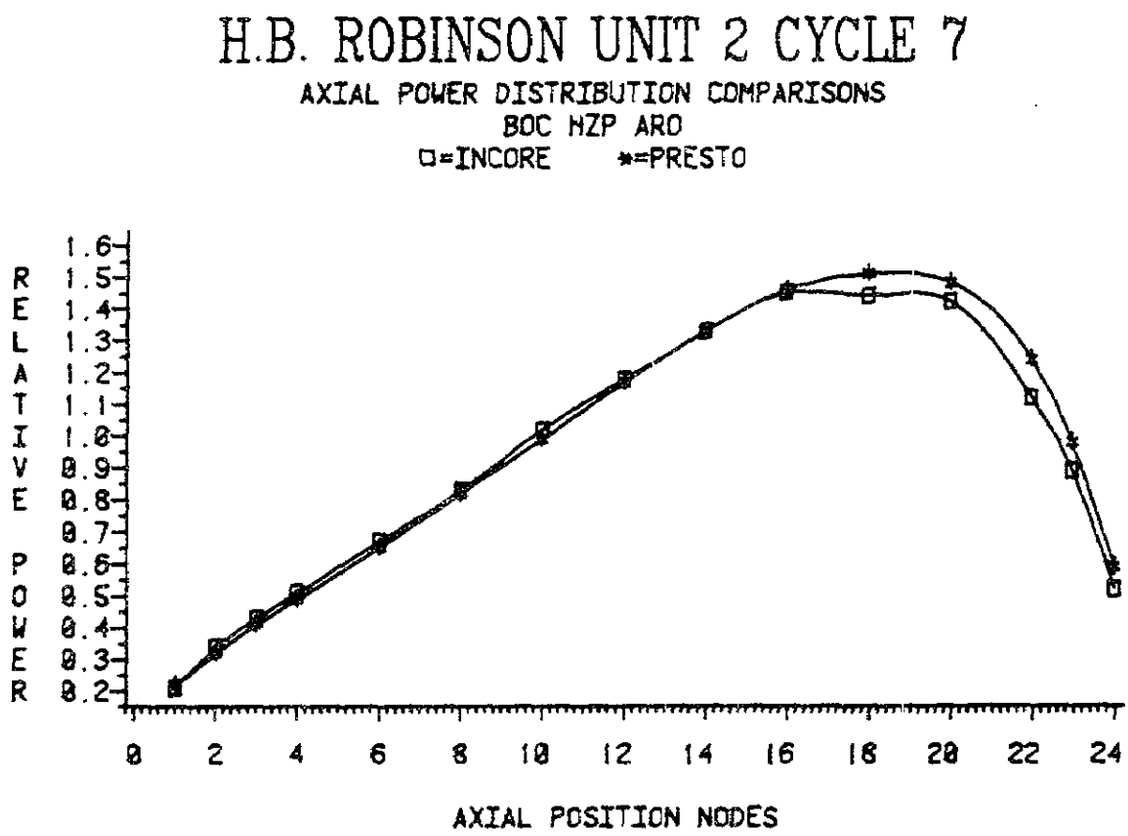
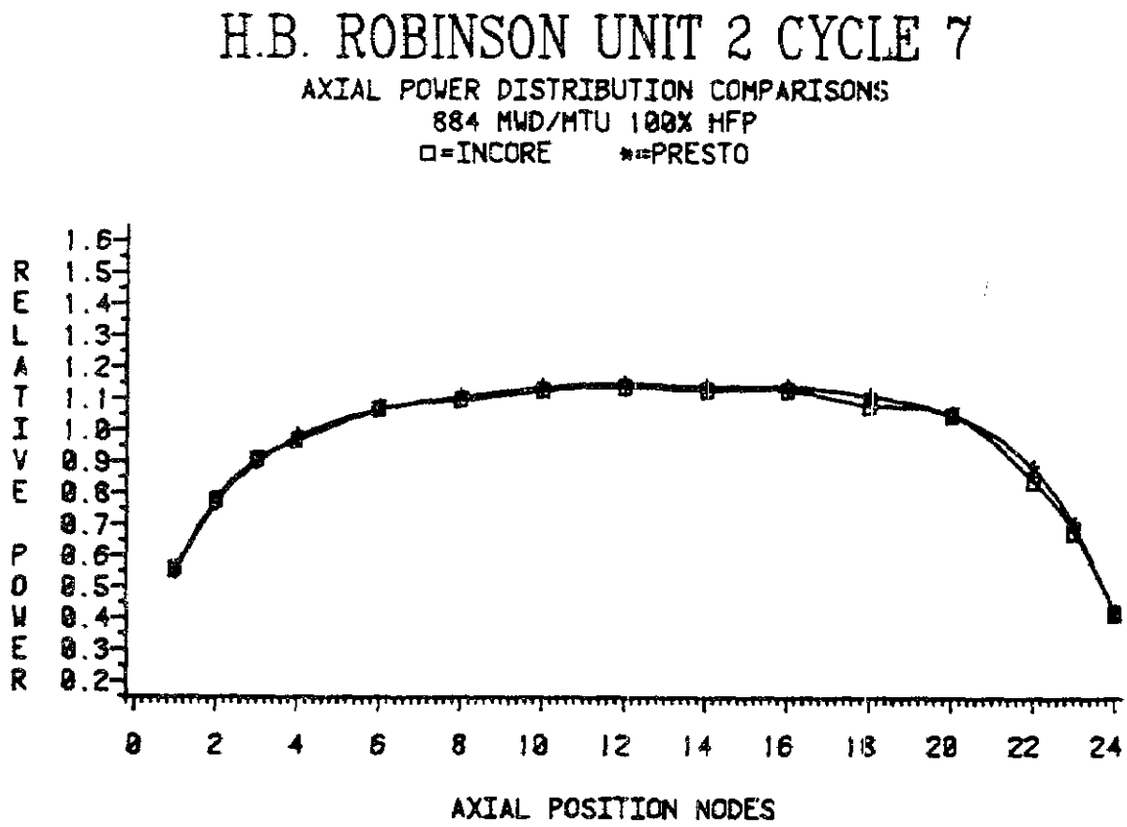

H.B. ROBINSON UNIT 2 CYCLE 6
AXIAL POWER DISTRIBUTION COMPARISONS
9923 MWD/MTU 100% HFP
□=INCORE *=PRESTO
RELATIVE POWER
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0 2 4 6 8 10 12 14 16 18 20 22 24
AXIAL POSITION NODES

FIGURE 1.2.3


H.B. ROBINSON UNIT 2 CYCLE 7
AXIAL POWER DISTRIBUTION COMPARISONS
BOC HZP ARO
□=INCORE *=PRESTO
RELATIVE POWER
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0 2 4 6 8 10 12 14 16 18 20 22 24
AXIAL POSITION NODES


H.B. ROBINSON UNIT 2 CYCLE 7
AXIAL POWER DISTRIBUTION COMPARISONS
884 MWD/MTU 100% HFP
□=INCORE *=PRESTO
RELATIVE POWER
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0 2 4 6 8 10 12 14 16 18 20 22 24
AXIAL POSITION NODES

REACTOR PHYSICS ACTIVITIES IN SPAIN

1.- PWR CORE DESIGN AND FUEL MANAGEMENT (DENIM, Department of Nuclear Energy ETSII, Universidad Politécnica de Madrid).

A first phase of development and validation of PWR core design and fuel management was completed in 1982, through a cooperative effort of an spanish utility (IBERDUERO), the nuclear research center (JEN) and the DENIM. Main results were presented at the September 82 ANS Topical Meeting on "Advances in Reactor Physics and Core Thermal Hydraulics" (Ref. 1)

Under a research contract of the International Atomic Energy Agency (CRP on codes for in-core fuel management) this computer code system has been documented, and is being made available through the NEA Data Bank.

The CARMEN system (Ref. 2) for finite difference diffusion calculation of PWR cores with space dependent nuclear and thermal hydraulic feedbacks, which was operating in the UNIVAC-1100 at JEN, was converted to FORTRAN-77 and implemented in the CYBER-835 at DENIM. Both versions have been distributed to the NEA Data Bank (Newsletter No. 3 of August 1983).

The MARIA system users manual has been published (Ref. 3) and the implementation on the CYBER-835 at DENIM was done with the collaboration of Dra. Teresa Kulikowska from the Badan Jadrowych Institute (Swierk, Poland), during her stay at DENIM. This system includes an extended version of WIMS-D-4 and auxiliar codes (PREWIM and POSWIM) for preparation and postprocessing of the WIMS calculations for PWR fuel assemblies. Every kind of PWR fuel assembly, including those with burnable absorber clusters and control rods, can be modelled by the MARIA system using a bundle description. The extensions in WIMS-D.4 include the homogeneization and group collapsing to fewgroups of the macroscopic and microscopic cross sections of the different fuel or absorber cells in the bundle for use in the CARMEN system or other core models. The MARIA system is now available, in both UNIVAC and CYBER versions, from the NEA Data Bank. The CYBER version was converted to FORTRAN-77 (FTN 5.1), which required a big effort, specially because of the non-standard procedures for calls and entries in the original and the UNIVAC version. Most of the WIMS options were tested but a few could still require some checking by future users.

The CICLON code (ref. 4) for fast survey fuel management calculations was also implemented on the CYBER-835, with the collaboration of Dr. R.P. Jain from Bhabba Atomic Research Center during his stay at DENIM in April 83. This version has been made available too to the NEA Data Bank.

Work is proceeding for documentation and further validation of our 1 group 3D PWR simulator (LOLA system), which includes new transport kernels and correction factors for the nodal K_∞ and M^2. Those transport and spectral factors, as well as the core-reflector albedoes in one-group,are explicitly calculated by an auxiliary code from fine-mesh two-group diffusion fluxes obtained from the CARMEN or the VENTURE code. Those factors and albedoes are such defined that, for quarter of assembly nodes, they do not change among the different nodes of the same kind (enrichment and number of burnable poison rods) as well as with the conditions along operation (density, Boron and burnup). Rather good accuracies have been achieved with these procedures as discussed in refs. 1 and 5. A more detailed paper will be submitted for publication in Nuclear Technology in a short term.

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Other line of development is under way for the local calculation of pin powers in full fuel assemblies or in 4 quarters of neighbour assemblies under given boundary fluxes or current-to-flux ratios. These local calculations will provide also the homogeneized cross sections and correction factors for the diffusion terms to be used in two-group coarse-mesh full core calculations. After demonstration of the accuracy and convergence of the local calculations for PWR assemblies and color sets (4 quarters of neighbour assemblies), the local-global code is being developed for iteration on the interface fluxes and currents in 2D. The final phase will be to incorporate these modules in a 3D 2-group coarse-mesh simulator for PWR operating and transient analysis, including the CARMEN modules for cross section feedback.

REFERENCES

1.- ARAGONES J.M., AHNERT C., GOMEZ-SANTAMARIA J., and OLAVARRIA I.R., "Development and Validation of Core Physics Methods for In-Core Fuel Management of PWR's", NUREG-CP-0034, pp. 315-332 (1982).

2.- AHNERT C., and ARAGONES J.M., "CARMEN System, A Code Block for Neutronic PWR Calculation by Diffusion Theory with Space-Dependent Feedback Effects", JEN-515 (1982).

3.- ARAGONES J.M. and AHNERT C., "MARIA System, A Code Block for PWR Fuel Assembly Calculations", JEN-543 (1983)

4.- ARAGONES J.M., "CICLON, A Neutronic Fuel Management Program for PWR Consecutive Cycles", JEN-336 (1977).

5.- ARAGONES J.M., AHNERT C. GOMEZ-SANTAMARIA J. and OLAVARRIA I.R., "Desarrollo y Validación de Métodos y Programas de Cálculo del Núcleo de Reactores de Agua a Presión", Nuclear España, No. 5, pp. 18-34 (1982).

2.- NEUTRONICS AND SHIELDING OF FUSION BLANKETS (DENIM)

Two mayor efforts have been done at DENIM in the development of methods and initial calculations on the neutronics of blanket in fusion reactors.

The first one is linked with the analysis of the neutronic fluxes for different classes of the first wall and blanket, to stablish finally the temporal response of the system to periodic repetition rates in inertial confinement fusion. Solid, gases and wetted walls have been initially evaluated in a very simplified way in order to contrast the typical conclusions about its performance and construction magnitudes. Other goal of this work is to be able to generate an analytical model, tested with more sophisticated calculations, which could describe the thermal characteristics of the cooling system in the blanket.

The calculations begun with a reference design of blanket from the bibliography, but getting the magnitude and spectrum of the neutron source from precedent calculations of DENIM on target analysis.

Two different libraries have been used (EURLIB-IV, MACKLIB-IV) to obtain the σ's, kerma factors for the material in the first wall and the blanket. The ANISN code is used to perform a shorter multigroup library (18 groups) to be used in the dynamic calculations. Finally, the CLARA code is computing the neutron flux and heating of the zones in the blanket in a temporal frame.

The other project is connected with the neutron damage in fusion reactors. Here, the line of work is some different of the former. The basic library ENDF/B-IV is used as data source for the NJOY code which generates the σ's for structural materials, gases (H, He) forming in the blanket, tritium reproduction, etc, Kermas and energy available (E_a) to produce atomic displacements. The pairs of atomic displacement is calculated following the expression $E_a/2E_d$, where E_d is the energy needed to remove an atom in the lattice obtained from experiments. Different materials have been just generated and added to the description of the systems in CLARA code.

Two different designs are being tested. One of them is a filled gas cavity (SOLASE). Neutronic fluxes have been calculated for it, but a major effort is being carried out to evaluate the INPORT design developped by the University of Wisconsin for the HIBALL project (Heavy Ions). This system supposes a wetted wall by Li Pb permitting a higher repetition rate. A large multigroup (18 groups) library, S_6-P_3), has been generated including materials as Li, Pb, Si, D, T, He, Fe, Cr, incorporated to the INPORT tubes and cooling reflector, first wall and blanket. The onedimensional and temporal calculations using the CLARA code are being just performed.

Other libraries including Va, Mo, Nb, and its alloys are being generated to in order to have the capability to analyse the characteristics of these refractory materials recommended for the first wall in fusion reactors.

3.- NUCLEAR DATA PROCESSING FOR FISSION AND FUSION REACTORS (DENIM)

In order to evaluate the nuclear data, our Department chose the NJOY code. The actual version, tested and optimized for our requirements, is denominated NJOY 10/81-3, updated in September 1982.

NJOY is the successor to the MINX code. It processes ENDF/B-4 and 5 files with the exception of MF8 (decay data) and MF32 (resonance covariances) Because of its great versatility, it is very efficient for using in thermal reactors, fast reactors, fusion reactors and shielding analysis.

NJOY produces neutron cross sections and group to group scattering matrices, heat and damages cross sections, activation cross sections, photon

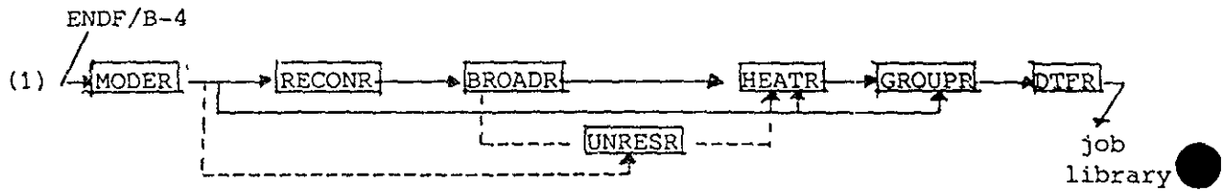
production matrices, photon interaction cross sections and group to group matrices, delayed neutron spectra, thermal scattering cross sections and matrices, and cross sections covariances).

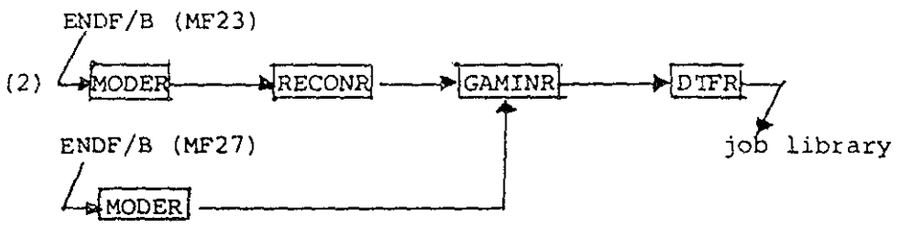
It is a modular system where each module is an essentially free-standing code devoted to one particular processing task. The communication between two modules is achieved via disk files in ENDF/B format. These files are the original ENDF/B tape, a pointwise ENDF tape (PENDF), and a groupwise ENDF tape (GENDF).

The following table is a brief description of the modules functions:

MODER	mode conversion (BCD/ internal blocked binary)
RECONR	xs pointwise reconstruction
BROADR	Doppler broadening
UNRESR	unresolved XS
HEATR	heating, damage and activation pointwise data.
HERMR	thermal pointwise data.
GROUPR	neutron and ngamma groupwise data
GAMINR	gamma interaction xs
ERRORR	xs covariances
DTFR	DTF format output
CCCCR	CCCC format output
MATXSR	MATXS format output

The modules sequence optimized by DENIM are now briefly detailed. The first provides the nuclear data for neutronic reactions, and the second for the gamma ones.





Note.- In the first sequence, the UNRESR module is only employed if the specific material has unresolved resonance parameters.

A posterior program (ANGIE) allows for the formation of the nuclear data matrix for the coupled transport of both species: neutrons and photons.

By means of this code, the nuclear data requested for our ICF systems transport calculations (NORCLA code) have been generated.

The energetic structure is 18 groups for neutrons and 12 groups for photons. The weight function for the multigroup treatment is the self flux spectrum resulting from a first interaction with a weight function (1/E + +thermal mawellian + fission spectrum).

The nuclear data obtained are essentially of two types: nuclear data for ICF target materials and nuclear data for ICF reactor structural materials (first wall, blanket, reflector and shield).

In the first group are Deuterium, Tritium, Aluminium, Gold and Lead. These two last materials, Au and Pb, required a special treatment of the Kerma factor evaluations due to a lack of consistency in the ENDF/B-4 files, revealing some negative Kerma factors in several points of the energy grid.

In the gold case, the negative Kerma factors were caused by not consistent evaluations of the reactions MT16 (n,2n) and MT17 (n,3n). In order to solve this problem, the corresponding Kerma factors for these reactions were analyticaly evaluated. Once corrected the interface tape at HEATR output, the code is rerunned at this point. The results obtained were quite satisfactory. In the case of Lead, the negative Kerma factors are resultant of the large values of the ($b\,\bar{E}_{ij}\gamma$) term in the equation

$$K_{ij}\ (E) = (E+Q_{ij} - a\,\overline{E_{ij}}_{\,n} - b\,\overline{E_{ij}}\gamma)\sigma_{ij}\ (E)$$

The solution adopted was to obtain this Kerma factor via the Kinematic limits settled in the HEATR module.

Referring to the second group of materials, the treatment was different because gamma transport was not considered. The reasons for this are that the damage produced by gamma rays in materials is low compared to the damage induced by neutrons, and also that the system is sufficiently large for considering no photon looses (photon local deposition). The materials evaluated are: Li^6, Li^7, C, Si, Pb, Fe, Cr, Ni, Mo, V, Mn, W, Ca, O and H.

Besides Kermas, group and group to group cross sections matrices, for these structural materials, the following activation cross sections have been evaluated.

- dpa $= \dfrac{E_D}{2E_d}$

- Helium production cross sections (through $n\alpha$ reactions)

- Tritium reproduction XS (only Li^6 and Li^7 through $n\alpha T$ reactions)

- Hydrogen production XS (through np reactions)

REFERENCES

(1) "The NJOY Nuclear Data Processing System, Vols. I and II". R.E. MacFarlane, D.S. Muir, R.M. Boicourt (LA-9303-M, ENDF-324) May 1982.

(2) "NJOY: A Comrehensive ENDF/B Processing System" R.E. Mac Farlane, R.J. Barrett, D.W. Muir, R.M. Boicourt (LA. NM 87545)

4.- FINITE ELEMENT-DISCRETE ORDINATES METHOD FOR THE FOKKER-PLANK CHARGED PARTICLES TRANSPORT EQUATION (DENIM)

A Finite Element procedure to find the numerical solution of Fokker-Planck equation has been developed in order to evaluate the energy deposition of charged fusion products to the thermonuclear plasma (1). This topic is relevant for ignition calculations on Inertial Confinement Systems. (2)

The basis feature of charged-particle transport is the strong space-energy coupling: the Fokker Planck representation of the collisions has continuous slowing-down and deflexion terms (3). The treatment of these terms via conventional methods (multigroup) gives poor results (2). To solve this problem a coupled discretization scheme of the variables with FEM has been stablished. Its capability, flexibility and accuracy has been fulfilled. The results are on excellent agreement with others obtained analitically (4) or by other procedures (5), and can be considered a second order approach to the solution.

Basically, the FEM treatment of the equation is based on the weak formulation and on the Galerkin procedure (6). Assuming the discontinuity of trial functions, the global problem is reduced to a set of local problems with natural boundary conditions. This representation has some advantages, direct solution, numerical stability, good treatment of flux gradients,..., and its inconvenient is the number of unknowns.

The following representations of the angular flux are available:

i) Bilinear representation r, E:

$$\Psi(r,E) = \sum_{ij} \phi_{ij}\, \xi_i(r)\, \xi_j(E) \quad (r,E) \in D_{ij}$$

ii) Linear representation r,E:

$$\Psi(r,E) = \phi_o + \xi_i(r)\phi_1 + \xi_j(E)\,\phi_2 \quad (r,E) \in D_{ij}$$

iii) Linear representation r, E, μ:

$$\Psi(r,E,\mu) = \phi_o + \xi_i(r)\,\phi_1 + \xi_j(E)\,\phi_2 + \xi_l(\mu)\phi_3 \quad (r,E,\mu) \in D_{ijl}$$

where ξ are linear functions in the finite element D.

The (iii) expansion has been developed in order to make accurate calculations when the incoming flux is highly anisotropic (confinement with unfocussed ion beams), in these cases the discrete ordinate needs a high quadrature order.

In (i) and (ii) representations the angular variable is treated via discrete ordinates, computing the Fokker - Planck deflexion term with extended transport approach (7) or with "α coefficient" method (3). The

stability of the solution is guaranteed for a mixed extrapolation - iteration procedure.

The method has been implemented on the NORCLA code (8) for ICF calculations. The coefficients of the equation are evaluated in to the code and it is possible to transport any particle in any medium without providing a cross section table.

Finally, a Lagrangian formulation is available for the efficient coupling of Nuclear and Thermo-hydrodynamics codes. (9).

REFERENCES

1.- Honrubia J.J., Aragonés, J.M. (to be published).

2.- Tran, T.M., Haldy, P.A. , Ligou, J., Lefvert, T. Atomkernergie 36,218 (1980)

3.- Mehlhorn, T.A., J. Comput. Phys. 138,1 (1980)

4.- Haldy, P.A., Ligou, J., Nucl. Fus. 17,6 (1977)

5.- Przybylski, k, Ligou, J., Nucl. Sci. Eng. 81,92 (1982)

6.- Martin W.R, et al. Annuals of Nuclear Energy 8,633 (1981)

7.- Wienke, B.R., J. Quant Spetrosc. Radiat. Transfer. 28,4 (1982).

8.- Velarde, G. et al. Atomkernenergie 32,58 (1978)

9.- Wienke, B.R., Phys. Fluids. 17,6 (1974)

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5. CORE CONVERSION STUDIES FOR RESEARCH REACTORS . JEN (JUNTA DE ENERGIA NUCLEAR).

Rather extensive work in Reactor Physics and other areas was devoted to the Lo-Aguirre swimming-pool reactor in Chile, under collaboration between JEN and CCHEN (Comisión Chilena de Energía Nuclear). The work was mainly related to "core conversion from 90 % enriched uranium (HEU) to 20 % enriched uranium (LEU), but additional effort was neccesary because reactor up-grading in several alternatives was considered, starting from a power level of 10 MW.

The kind of calculations performed were very similar to those described in past-year report, for a proposed 3-MW Ecuadorian Reactor. After an optimization study to define important characteristics of the fuel element, many analysis were done for the design of the core and related systems; specific items covered by neutronic calculations were: criticality, core fluxes, burnup, peaking factors, fluxes in experimental facilities, reactivity coefficients, kinetic parameters, reactivity defects, etc.

Besides the Lo-Aguirre case, some preliminary calculations have also be done for the JEN-1 swimming-pool reactor, related to the proposed placement of graphite for berilium in a row of reflector elements.

6. NEUTRONIC CALCULATIONS OF PWR CORES. (JEN)

The MARIA system[1] has been developed. This code package has been included in the Coordinated Research Programme (CRP) on "Codes Adaptable to Small or Medium Size Computer Available in Developing Countries for In-Core Fuel Management" of the International Atomic Energy Agency.

In generates the cross sections library for the diffusion calculations with burnup and feedback effects and the k_∞ and M^2 parameters for the nodal calculations.

The MARIA system includes three modules:

- PREWIM generates the input data for the fuel assembly calculation module for all the fuel assembly types in the core.

- WIMS-TRACA generates colapsed cross sections versus burnup.

- POSWIM makes transport corrections on the diffusion constant of the absorber materials.

References

(1) C. AHNERT, J.M. ARAGONES. MARIA System: A Code Block for PWR Fuel Assembly Calculations. JEN-543 (1983).

7, METHODOLOGY VALIDATION FOR THE DETERMINATION OF FLUENCE IN PRESSURE VESSELS OF LWR's. (JEN)

The validation of methods and analytical tools for the determination of neutron fluence in pressure vessels of comercial LWR's has been completed[5].

Calculations have been performed with the EURLIB[1] library, ANISN-JEN[2] and DOT 3.5/E-JEN[3] codes for the PCA "blind test"[4].

The method is based on a spatial flux syntesis, obtaining reaction rates for several ex-core detectors from the dosimetry file of ENDF/B-IV and calculated fluxes.

References

(1) J. PEÑA; EURLIB: Una librería acoplada de secciones eficaces para neutrones y gammas específica para cálculos de blindaje. MEMO JEN/TCR/A 01-81 (1.981).

(2) J. PEÑA; ANISN-JEN. Código unidimensional de transporte por el método de ordenadas discretas. Manual de usuario y datos de entrada. MEMO JEN/TCR/A 08-82 (1982).

(3) J. PEÑA; DOT 3.5/E-JEN: Código bidimensional de transporte por el método de ordenadas discretas. Manual de usuario y datos de entrada. MEMO JEN/TCR/A 02-83 (1983).

(4) Technical Letter for the PCA "Blind Test". L. MILLER et al. Oak Ridge National Laboratory. May 1979.

(5) J. BROS, J. PEÑA; Validación de métodos de cálculo por transporte para la determinación de la fluencia neutrónica en vasijas de reactores. To be presented at the VIII Reunión Anual de la Sociedad Nuclear Española. Dic. 1983.

8. SHIPPING CASK (JEN)

A nuclear criticality safety analysis of a shipping cask for transport and storage of spent fuel has been carried out.

The WIMS-TRACA code[1] has been used to generate zone averaged multigroup cross sections, the Keff calculation was performed with the discrete ordinates code TWOTRAN-GG[2].

In order to achieve the required subcriticality (Keff<0,95) the cask is designed to contain black absorbers.

References

(1) C. AHNERT; Programa WIMS-TRACA para el cálculo de elementos combustibles. Manual de usuario y datos de entrada. JEN-461.

(2) C. AHNERT y J.M. ARAGONES. El código de transporte bidimensional TWOTRAN-GG. JEN-512.

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9. SHIELDING (JEN)

The whole efford in this area has been directed to the implementation of the modular system NJOY[1] to the UNIVAC 1100. The aim is to make a new module which would prepare a library in few groups for the Monte Carlo code TIMOC.

Several modules have already been implementad: MODER, UNRESR, HEATR, BROADR, THERMR, RECONR, GROUPR and NJOY.

References

(1) R.E. MacFarlane, D.W. Bolcourt; The NJOY Nuclear Data Processing System, LA-9303-M (ENDF-324) May 82.

Swedish Contribution to the NEACRP activity report

POWER REACTOR LWR - Code development

(K Ekberg, Studsvik Energiteknik AB)

The 2D diffusion code MBS has been released for use by customers on CYBERNET. Verification studies on a Swedish and on a US reactor are in progress.

A new version of the MICBURN code for microscopic burnup of gadolinia loaded fuel pins, MICBURN-2, has been released.

A further developed version of the well-known cell and assembly code CASMO, denoted CASMO-2E, has been installed on CYBERNET and elsewhere. The enhancements are mostly in the area of PWR application.

A linking code MBSLINK between CASMO and MBS has been developed.

Development work on CASMO, MBS, SIMULATE and auxiliary codes continues.

91100124

Reactor physics calculations on close-packed LWR lattices

(Erik Johansson, Studsvik Energiteknik AB)

This project deals with calculations on Pu recycling in close-packed PWR lattices. The initial work, including some preliminary tests of the calculational model, i.e. the cell code CASMO and its associated libraries, has been described earlier - in STUDSVIK Report NR-82/126. Since then a study has been performed on the use of such close-packed lattices in the Swedish 12-reactor system (9 BWRs and 3 PWRs) - both in a phaseout version and in a version developing into an asymptotic steady-state operation.

We do not give any definite figures here for the savings of natural uranium and separative work because that would require a detailed description. Just as an indication we can say that the close-packed lattice in the steady-state case would require about 50 % less natural uranium and about 40 % less separative work than would once-through operation in the normal lattice. These values were obtained under the assumption of a 3 % loss of Pu on each re-processing/refabrication occasion.

At present (sept 1983) some test calculations are going on using measured results from Würenlingen on Pu-loaded close-packed lattices.

Calculations on the change from 93 % to 20 % enriched uranium in the R2 research reactor

(Erik Johansson, Studsvik Energiteknik AB)

The conversion of the R2 research reactor at STUDSVIK to the use of 20 % enriched uranium has been studied for some time. The initial reactor physics work, described in STUDSVIK Report NR-82/136 has been followed by further studies, including beam tube investigations, as accounted for in STUDSVIK Report NR-83/261. The calculational work is now continued by the RERTR group at Argonne National Laboratory.

MBS - a program for two-dimensional nuclear analysis of PWR cores

(Olov Norinder, Swedish State Power Board)

The computer program MBS has been designed specially for the calculation of the power distribution among the fuel pins of PWRs. Further MBS can determine the detector signals that should be given among the input to the computer program INCORE. This code calculates the power distribution of the reactor core from the signals of detectors that are moved through the core.

MBS has been tested against measurements and frestanding calculations for cycle 4 and 6 of Ringhals 2. MBS has shown good results in the testing.

The aim of the current phase of the project has been to further develop and test MBS so that the program can be used for ICFM work safely and rationally. In this context the program has been extensively documented. In the work the demands of the quality assurance has been particularly considered.

In Core Fuel Management at Sydkraft

(J O Gustafsson, Sydkraft AB)

Together with ASEA ATOM SYDKRAFT has the ICFM responsibility for Barsebeck Unit 1 and 2. There are a number of activities of which the most important are:

- Core follow calculations
- Refueling analysis
- Fuel design studies
- Economic analysis

The basic instrument to carry out those activities is the POLCA/CASMO system. POLCA is a 3D nodal core cimulation code.

CASMO is used to generate nodal 2D cross sections which are used as input to POLCA. We also use CASMO in our studies to optimize fuel designs. The code has a large flexibility, and it is frequently used to study the influence on k-infinity and internal power peaking factors when a pertubation is introduced (eg channel-bowing, high temperature in fuel pins, extra Zr-pin or water-hole).

Additional to the POLCA system SYDKRAFT has developed some simple codes used for administrative purpose, eg code for updating fuel records. The fuel records contain information about burnup and channel location each cycle etc.

Furthermore SYDKRAFT has developed codes for calculations of power histories, rest power, averaged k-infinity of the core. These codes are based on data from POLCA calculations and are mainly administrative routinescombined with a simple physical model. These codes are implemented on a NORD 500 computer.

Effectiveness of cylindrical control rods

(D.C. Sahni* and N.G. Sjöstrand,
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The neutron flux distribution has been calculated around a cylindrical, totally absorbing rod immersed in moderators of various size and degree of absorption. Isotropic scattering was assumed as well as a uniform source distribution. From the results obtained with a standard ANISN code the linear extrapolation distance was derived. It was found necessary to introduce a correction for the fact that ANISN does not give an exact balance between production and loss of neutrons. After such a correction the linear extrapolation distance could be obtained to an accuracy of 1-2 %. Through using two different definitions of the extrapolation distance it was possible to reproduce the values of Pellaud (1968) and Isakova (1968) and to explain the difference between their results. (Ann. Nucl. Energy 10, 351, 1983)

Complex time eigenvalues of the one speed neutron transport equation for a homogeneous sphere

(E.B. Dahl and D.C. Sahni*)

The time eigenvalue spectrum of the one-speed, isotropic scattering neutron transport equation has been studied for a homogeneous sphere with vacuum boundary conditions. There is a close relationship between the time eigenvalue problem and the criticality problem of the time independent equation for the same model. It is shown that this relation holds even when the time eigenvalues are complex. Using Carlvik's method to solve the criticality problem, it is shown that complex time eigenvalues do actually exist for this model problem. Thus, the real eigenvalues found by van Norton do not form the complete spectrum. (To be published in Transport Theory and Statistical Physics.)

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Heterogeneous treatment of water gaps and control rods in core calculations

(S.-Ö. Lindahl**)

Nodal models for calculation of core power distributions are conventionally based on assemblywise homogenized procedure may introduce substantial errors when strong heterogenities, like control rod blades in the BWR, are present. By extending the cross section homogenization over the fuel pin area only and treating the water gaps and control rods explicitely, using response matrix concepts, the error can be substantially reduced. It is shown how the explicit treatment of the gaps can be incorporated into existing nodal models. The proposed method has the side-benefit of producing cross sections for the fuel pin area that are fairly insensitive to the presence of control rods. The gap responce matrices depend only weakly on the lattice design, void content, and fuel burnup.

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REACTOR PHYSICS ACTIVITIES IN SWITZERLAND

October 1982 to September 1983

P. Wydler

1. Introduction

The reactor physics activities reviewed in this paper have been carried out at the Federal Institute for Reactor Research (EIR), except for the preparations for the fusion-fission hybrid blanket experiment LOTUS which is undertaken at the Federal Institute of Technology, Lausanne. Because of the needs of the Swiss reactor operators, Light Water Reactor orientated physics has remained an important subject, but the majority of the available funds are now concentrated on an experimental programme in which the PROTEUS reactor is used to investigate the physics properties of Light Water High Converter Reactor (LWHCR) lattices. Encouraged by the progress with a small district heating scheme called REFUNA, which will start to deliver heat from the Beznau reactors before the end of this year, a study of a new type of low power district heating reactor has been initiated and carried through its first stage. Another relatively new development is the extension of the activities in the direction of fusion problems with emphasis on blanket physics.

2. Experimental Studies on LWHCR Lattices

Phase I of the PROTEUS-LWHCR experiments was conducted during the 14-month period August 1981 to September 1982. The planning of the experimental programme, as well as the nature of the early results obtained, were discussed in previous NEACRP review papers.

Nearly 80% of the time available for the Phase I measurements was devoted to obtaining integral evidence on reactivity and reaction rate ratio variations with moderator voidage in a reference tight-pitch LWR lattice with an average Pu-fiss enrichment of about 6%. A NEACRP-A paper for the current Meeting presents the detailed experimental results and shows how they

may be useful in resolving the conflicting calculational results that have been reported in the past for the void coefficient characteristics of LWHCR's.

Other experiments carried out in the Phase I programme include measurements in a small central test zone simulating a higher (8% Pu-fiss) enrichment, as well as a limited set of small-sample reactivity measurements in the reference lattice. The final analysis of these latter experiments is currently in progress.

In spite of its usefulness in providing a first set of integral results for testing LWHCR physics calculations, the PROTEUS-LWHCR Phase I programme has had some inherent shortcomings. These have stemmed largely from the constraints imposed by the available fuel materials and include, for example, the 2-rod nature of the experimental lattice, the use of Pu of Magnox reactor origin and the relatively small test zone size. Moreover, the total time available for the experiments severely restricted the range of configurations that could be investigated.

In view of the above shortcomings it is planned to carry out a more comprehensive, second phase of PROTEUS-LWHCR experiments during 1985 - 88. New, LWHCR-representative fuel elements have been ordered for the purpose, and these will be manufactured in the F.R. of Germany during 1984. Apart from strengthening the currently established experimental base for LWHCR void coefficient calculations, the new programme of measurements will also deal with other power reactor features such as control rod worths, soluble boron effectiveness and the effects of blanket zones.

Some results for the LWHCR Phase I experiments have already been reported externally (Refs. 1 to 3).

3. Study of a Low Power District Heating Reactor

A study for a new type of low power district heating reactor has been performed. The reactor has a homogeneous core, rated at a thermal power of 10 MW, and was designed to supply a community of 2'000 to 5'000 inhabitants

with hot water at a temperature of 110 °C (Of the communities in Switzerland this size is the most numerous). A preliminary cost analysis indicates that the reactor can be expected to be competitive with conventional thermal energy sources.

To improve the safety features of the reactor an integrated design was chosen (i.e. all components of the primary circuit including the intermediate heat exchanger are enclosed in a single vessel) and the primary circuit was designed for natural circulation cooling at full power. The cylindrical liquid reactor core has a volume of about 2500 l and operates at a mean temperature of 140 °C. The fuel is a solution of enriched uranium sulfate (UO_2SO_4) in water. Although the optimum enrichment is 60%, the reactor can operate satisfactorily with 20% enriched fuel.

The physics design methods include MICROX cross section condensations, two-dimensional FINELM diffusion theory calculations, burnup calculations within the modular code system RSYST, and control rod calculations using the SURCU transport code.

Using these methods, beginning-of-life ^{235}U inventories of 48.7 and 42.5 kg were predicted respectively for the 20% and the 60% fuel enrichment. The conversion ratio is low (< 0.1), but this has the advantage of only marginal plutonium production. Over a two-year reactor cycle at a load factor of 0.5, this corresponding to a burnup of 50'000 MWd/t(HM), the reactivity loss was calculated to be 12.4 $ for the optimum enrichment. Additional reactivity margins are needed to compensate for the relatively large temperature effect and the (negative) void effect due to the dissociation of water. In total, this necessitates a relatively large reactivity compensation, which is effected using 22 control rods.

Due to the relatively high enrichment the reactor has a small (but still negative) doppler coefficient. However, a sufficiently large prompt negative reactivity feedback is ensured by the temperature-density coefficient. The dynamic properties of the reactor were studied for the cases of a (secondary circuit) loss of flow and a fast control rod withdrawal. In both cases the behaviour of the reactor is benign.

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A more detailed description of the concept is given in Refs. 4 and 5.

4. Reactor Noise Analysis

The activities in the reactor noise analysis field concentrate on measurements of two-phase flow parameters with emphasis on measurements of flow velocities in BWR fuel rod bundles near an instrument tube. The aim of the work is mainly to gain the understanding, necessary for the interpretation of fluid velocities measured by noise analysis, and to investigate whether noise analysis measurements can be used for reactor monitoring purposes and for the verification of two-phase thermohydraulics codes.

Almost all two-phase flows have a well established radial velocity and void distribution profile. From two-phase flow models it is possible to obtain the fluid velocity averaged over some cross-sectional area. It might also be expected that some average velocity could be obtained from the peak of a cross correlation function measured with two sensors. In practice more than one peak is found and no simple average velocity is obtainable. A better understanding of these signals has now been obtained from out-of-pile measurements in an air-water test loop in which both the radial velocity profile as well as the "summation velocity" obtained from cross correlations of light beams have been determined (The radial measurement involved a local light reflection probe which could be moved to different radial positions inside the flow). These studies are described in Refs. 6 and 7.

In earlier measurements in the upper part of a BWR core such multiple peaks in the cross correlation function had been observed. These can now be interpreted as being due to a pronounced radial velocity profile in the bundles around the instrument tube. Within certain limitations, by simulating the detector signals with respect to a radial velocity profile as given by an advanced subchannel analysis code (like THERMIT) and comparing the resulting noise-analytical functions obtained from this simulations with the equivalent ones inferred from reactor measurements, one is able to "recover" the radial velocity profile in the bundles and use the results for the verification of advanced subchannel analysis codes. A publication on this work is about to appear (Ref. 8).

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5. Spallation Neutron Source Studies

In connection with the proposed spallation neutron source at the 590 MeV high-current proton accelerator of the Swiss Institute for Nuclear Research (SIN) reactor physics and shielding methods are being applied for optimizing the source design. The shield of the source is composed of 4 m of cast iron and 1 m of concrete and was designed for a source strength of 10^{17} n/s. The performance of this shield has recently been reanalyzed using new cross section data and improved calculational methods.

For this purpose two high energy cross section libraries originating from the Oak Ridge and Los Alamos National Laboratories were modified and validated with the help of a theoretical benchmark problem specified jointly by EIR, SIN and KFA Jülich. Together with an experimental source spectrum (measured at a mock-up of the target) the data were used in S_N transport theory calculations of the neutron and gamma dose rates in the shield.

The results of the study are given in Ref. 9. From the spatial dependence of the dose rates it could be deduced that an increase in the steel to concrete thickness ratio would allow the outer radius of the source to be reduced by as much as 10% (This optimization did not include cost considerations). In the study the usefulness of adjoint calculations for determining surface dose rates due to monoenergetic source neutrons was demonstrated.

The application of the SIN type spallation neutron source for simulating the irradiation of the first wall in a fusion reactor has been investigated further. Neutron induced helium production and atomic displacement rates in aluminium and stainless steel samples were calculated using the neutronic data and methods described above together with damage response functions generated with the NJOY and DON codes. A particular objective was to assess reflector gains for different materials such as Fe, Ni and Pb and the influence of these reflector materials on the helium-production to displacement ratio (or so-called CTR parameter).

The results reported in Ref. 10 show that the reflectors produce a useful increase in the atomic displacement rate, but do not noticeably affect the He production rate. He production and displacements per atom were calculated to be about five times smaller than in a fusion reactor. On the other hand, the spallation source considered in this study compares favourably with currently available neutron sources, offering a ten fold increase in the He production rate. The CTR parameter was found to lie in the range 5 to 13, depending on the reflector, the latter value corresponding to an unreflected configuration and being typical for a fusion reactor.

6. Fusion Blanket Studies

Over the past two years EIR has developed methods and tested nuclear data needed for the physics analysis of the blankets of fusion and fusion-fission hybrid reactors with magnetic confinements. Although for the blanket physics the computational approaches are basically the same as those used for fission reactors, difficulties arise due to the presence of additional nuclides, new reaction types, different neutron spectra, and novel geometric configurations. A particular problem is the adequate prediction of tritium breeding, which is not only affected by the lithium cross sections but also by the n,xn cross sections of various multiplying materials.

In cooperation with General Atomic cross section sensitivity and uncertainty analysis studies were made for the European INTOR and the U.S. FED design of a fusion reactor (Refs. 11 and 12). An extension of this work included a comparison of the performance of the data libraries DLC-37, VITAMIN-C/DLC-41, VITAMIN-C/MACKLIB-IV and the Los Alamos NJOY fusion library (Ref. 13). Furthermore, three of the aforementioned libraries, together with various transport theory approximations for the blanket calculation, were tested for the hybrid design of a Tandem Mirror Reactor originating from the Lawrence Livermore Laboratory and General Atomic (Ref. 14).

The methods have been applied for the neutronic analysis of a fusion-fission hybrid reactor based on the UK design of the Reversed Field Pinch Reactor. The EIR concept of this reactor is characterized by

either an aluminium or copper first wall/shell, which acts as a neutron multiplier, and a He cooled hybrid blanket consisting of a Th metal multiplier and breeder, a LiO_2 tritium breeder and a stainless steel reflector zone. Helium is used as a coolant to minimize the non-fertile neutron captures. The study (reported in Ref. 15) showed that from the neutronics point of view the concept is feasible. In addition to being self-sufficient in the tritium fuel, the reactor breeds 0.7 kg of ^{233}U per MW(th)-year. However, an assessment of the radiation damage indicated that the lifetime of the first wall is not adequate. The possibility of using a thinner first wall is therefore being investigated.

In view of possible future design studies at EIR and planned hybrid blanket experiments at the Swiss Federal Institute of Technology at Lausanne, the EIR computational methods are being further refined. With the support of the Los Alamos National Laboratory and General Atomic a cross section generation and calculational scheme including NJOY/TRANSX-EIR/MICROX, the three-dimensional Monte Carlo codes MCNP and NMTC and the two-dimensional finite element discrete ordinates code TRIDENT for toroidal blanket geometry is being developed and tested.

7. LOTUS Fusion-Fission Hybrid Blanket Experiment

The preparations for the fusion-fission hybrid blanket experiment LOTUS at the "Institut de Génie Atomique" of the Federal Institute of Technology, Lausanne, are progressing well. The Haefely neutron generator with a source strength of $5 \cdot 10^{12}$ n/s is being installed in the test cavity and the various components of the blanket, which has the form of a 85 cm thick, 100 cm high and 100 cm wide parallelepiped, are currently being fabricated. The first experiments are expected in early 1984.

Starting at the source end, the reference blanket is composed of a 1 to 3 cm thick stainless steel sheet to simulate the first wall of a fusion reactor, a 100 mm thick neutron multiplier zone made of lead plate, a 35 mm thick spectrum adjustment zone of lithium carbonate, a 277 mm thick ^{233}U breeding zone of thorium oxide, a 150 mm thick

tritium breeding zone of lithium carbonate, a 250 mm thick graphite reflector and, finally, a 35 mm thick lithium carbonate absorber zone. The purpose of the spectrum adjustment zone is to harden the neutron spectrum in the ThO_2 breeding zone and thereby maximize the net rate of ^{233}U production. The absorber zone behind the reflector captures thermal leakage neutrons which would otherwise be lost.

The ThO_2 in the form of rods is obtained on loan basis from the Bhabha Atomic Research Centre, India. The lithium carbonate is encased in rectangular boxes made from extruded aluminium channel. Although lithium carbonate is not a viable material for fusion power reactor applications, it is an adequate substitute for the purpose of the LOTUS experiments, which are primarily intended as a neutron physics benchmark. Compared to lithium oxide it has the advantage that high purity material can be purchased at an acceptable price.

Particular emphasis is being put on adequate diagnostic techniques. These include neutron spectroscopy (NE-213 scintillators and proton recoil telescopes) and integral reaction rate measurement techniques, the latter being developed and adapted in collaboration with EIR. For Th(n,γ) , Th(n,f) and Th(n,2n) measurements, methods will be applied which had been developed earlier in connection with a zero-power reactor physics programme on thorium-bearing fast reactor lattices at the PROTEUS reactor. To measure tritium production rates two independent techniques utilizing a liquid scintillator method and the self-irradiation of TLD's are being tested.

The present status of the LOTUS experiments is described in Ref. 16. A detailed account of the diagnostic techniques and other complementary information is presented in a NEACRP-A paper for the current Meeting.

REFERENCES

1. K. Gmür, R. Chawla and R. Seiler
"A LWHCR Void Simulation Experiment using Dowtherm"
Trans. ANS, 44 , 558 (1983)

2. E. Hettergott, R. Chawla and K. Gmür
"Analysis of Test Lattice Experiments in the Light-Water-High-Conversion-Reactor PROTEUS"
Report EPRI-NP-3190 (1983)

3. R. Chawla, K. Gmür, H. Hager, E. Hettergott and R. Seiler
"Measurement of the Mean k_∞ Void Coefficient for an LWHCR Lattice between 0-100 % Void"
Jahrestagung Kerntechnik, Berlin (1983)

4. W. Seifritz et al.
"Space Heating for Small Communities with a Homogeneous Heat Producing Reactor (HHR)"
Paper to be presented at the IAEA Technical Committee Meeting and Workshop on Nuclear Heat Application, Krakow, Poland, 5 - 9 December, 1983

5. W. Seifritz et al.
"Ein homogener Heizreaktor (HHR) kleiner Leistung (10 MWth) für die nukleare Nahwärmeversorgung
Interner EIR-Bericht (1983)

6. G. Th. Analytis and D. Lübbesmeyer
"A novel cross-correlation technique for the determination of radial velocity profiles in two-phase flows"
EIR-Bericht Nr. 483 (1983)

7. G. Th. Analytis and D. Lübbesmeyer
"A study of annular flows with bubbles in the liquid ring and entrained droplets by means of stochastic analysis techniques"
EIR-Bericht Nr. 489 (1983)

8. G. Th. Analytis and D. Lübbesmeyer
"Two-Phase Flow Velocity Measurements in the Upper Part of a BWR; The Importance of Multi-Dimensional Effects for their Interpretation"
To be published in Proc. of the Thermohydraulic Division of the ANS Winter Meeting, San Francisco (1983)

9. V. Herrnberger and P. Stiller
"SINQ Bulk Shield Analysis Using the S_n-Method"
ICANS-VII-Meeting for the International Cooperation on Advanced Neutron Sources, Chalk River, Canada, 13 - 16 September, 1983

10. V. Herrnberger, P. Stiller and M. Victoria
"Some Estimates of the Fusion Radiation Damage Simulation by Spallation Neutrons"
Sixth International Conference on Radiation Shielding, Tokyo, 16 - 20 May, 1983

11. S. Pelloni and E.T. Cheng
"Cross Section Sensitivity Studies for Fusion Blankets Incorporating Lead Neutron Multiplier"
Proc. Int. Conf. on Nuclear Data for Science and Technology, Antwerp, p. 331 (1982)

12. S. Pelloni and E.T. Cheng
"Cross Section Sensitivity Study for U.S. Fusion Engineering Device"
To be published in Proc. Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, Tennessee, 26 - 28 April, 1983

13. S. Pelloni, J. Stepanek and D. Dudziak
"Intercomparison of Nuclear Data Library Sources, Group Structures and Collapsing Spectra for INTOR-EC"
To be published in Proc. Fifth Topical Meeting on the Technology of Fusion Energy, Knoxville, Tennessee, 26 - 28 April, 1983

14. D. J. Dudziak, J. Stepanek, W.T. Urban and G. Friedrich
"Comparison of Los Alamos MATXS, VITAMIN-C and DLC-37 Multigroup Libraries for a Reference Fusion Hybrid Blanket"
Proc. Int. Conf. on Nuclear Data for Science and Technology, Antwerp, p. 339 (1982)

15. J.F. Jaeger et al.
"A Hybrid Blanket for a Reversed Field Pinch Reactor"
To be published in Proc. Course and Workshop on Mirror-Based and Field-Reversed Approaches to Magnetic Fusion, Varenna, September 1983

16. P.A. Haldy et al.
"Present Status of the EPFL (Swiss) Fusion-Fission Experiment LOTUS"
Third International Conference on Emerging Nuclear Systems, Helsinki, 6 - 9 June, 1983

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Reactor Physics in the United Kingdom

J R Askew
J M Stevenson

1. GENERAL PROGRAMME

The public enquiry into the proposed PWR at Sizewell reopened during September. This followed a summer recess after a sitting of 104 days. Subject to approval of the proposal, construction would now be expected to start in 1985, a year later than originally planned.

14% of the electricity generated by the CEGB in 1982-3 was from nuclear stations. The increase compared with the previous year arose largely because of the return to service of five of the six Magnox stations which had been shut down for inspection and remedial repair of the reactor bellows. The sixth unit is due to complete its inspection and repairs this year. In addition the AGR at Hinkley Point B achieved a 9% increase on its previous best performance as a result of modifications to enable on-load refuelling to take place. Further development work will make it possible to refuel at higher loads.

In November 1982 the Government completed its review of the Fast Reactor, and gave the go-ahead to the development programme but at a rather slower rate. The ordering of the first commercial fast reactors will now not take place until early in the next century. In September 1983, the Government announced that it was opening formal negotiations to seek agreement on the joint development of fast reactors with France, Germany, Italy, Belgium and the Netherlands.

2. THERMAL REACTORS

Further validation assessment of RETRAN against LOFT experiments has been carried out. Since the last report simulation of L6-5 and the more severe transients of the L9 series have emphasised the importance of steam generator modelling.

Reactor transient studies have concentrated on the loss of off-site power anticipated transient without scram (ATWS) where all power to the main circulators is lost. The issues centre on the effect of primary voidage on natural circulation and reactivity. Work on the loss of feedwater without turbine trip ATWS was used as a basis for sensitivity and uncertainty studies. This work has been reported to the September 1983 ANS Topical Meeting on normal and abnormal transients.

LWR-WIMS/JOSHUA studies of Doppler coefficient have shown sensitivity to the way in which the fuel temperature change is obtained. It is important that the coolant enthalpy distribution changes are properly modelled.

A brief study has been made of the parameters of importance in calculating the worth of gadolinium burnable poison pins (1). These parameters included

(a) number of energy groups in the lattice calculation
(b) interval between lattice calculations
(c) interval between re-evaluation of poison pin fine structure
(d) poison pin radial subdivision
(e) effect of smearing the poison pin prior to diffusion calculations
(f) minor gadolinium isotopes.

The conclusion was that it is difficult to do better than 0.5% in reactivity at all irradiations without using an inordinate amount of computing time and that there is little point in, for example, a very fine radial pin subdivision unless all other aspects of the calculations are similarly detailed.

With the ultimate objective of developing a rapid 3D core transient code, exploratory studies have been made with a sparse matrix direct inversion technique for solving the transient neutronics equations. A typical computing speed achieved is 4 seconds IBM 3081 cpu time for a 100 second rod ejection transient without feedback in a 3D problem with 1000 mesh points. The computing time increases rapidly with number of meshes, but more seriously, the Crank-Nicholson time integration which permits long time steps is less accurate when feedback effects are introduced. Thus the implementation of a simple Doppler feedback model increased the cpu time for the rod ejection transient by an order of magnitude.

The WIMS 81 data library continues to give satisfactory results. A whole core simulation of one of the commissioning experiments on Dungeness B AGR using the MONK5W Monte Carlo code with WIMS data gave an eigenvalue of 1.0025 ± 0.3%. This was particularly reassuring as earlier deterministic calculations using the SNAP diffusion code had given k ∿1.02, thus casting doubt on the WIMS 81 library. It is now believed that black rod and streaming effects are the causes of the discrepancy, though this has not yet been demonstrated in detail.

The next step in the application of direct Monte Carlo calculations to whole reactor cores would logically be the treatment of depletion. In principle it would be expected that cycle length could be predicted to equivalent accuracy at no more cost than the calculation of reactivity at start of cycle. This has been demonstrated for greatly simplified test problems which have studied the behaviour with both negative and positive reactivity feedback effects, the latter arising for example, when burnable poisons are depleted more rapidly than the associated fuel.

Experimental studies have been carried out during the last year in collaboration with the Berkeley Nuclear Laboratories of CEGB utilising the zero-energy critical facility, BAGR, and the NESSUS irradiation thimble located within the central reflector of the NESTOR shielding reactor at Winfrith. This work has improved the accuracy with which neutron and gamma energy-deposition in moderator graphite can be determined, and consistency with ±10% has eventually been achieved. A novel feature of this work was the use of microcalorimeter which has been developed in collaboration with Imperial College to measure energy-deposition rates down to about 10 μ/Watts per gm. The programme is continuing with the aim of increasing the sensitivity of the device to cover the range from 1 to 10 μ/Watts per gm. It could then be used in penetration benchmark experiments which are conducted in the ASPIS bulk shield facility and also in zero-energy critical experiments such as those mounted in the BAGR and DIMPLE.

3. FAST REACTORS

3.1 ZEBRA and the CADENZA Programme

The ZEBRA reactor has now been shut down for about a year and the experimental team has moved to the DIMPLE reactor as discussed later in this paper. Maintenance of the ZEBRA plant is continuing to allow a return to fast reactor work when required.

A meeting was held to consider the calculations for the CADENZA benchmarks carried out in JAPAN (2), USA (2), Germany, France and the UK. As discussed in a separate paper, the discrepancies between the calculated and experimental k-values for the all-plate and extrapolated all-pin cores varied from 0.0015 dk to 0.0082 dk. Differences between the calculated reactivities for the homogeneous pin and plate cell compositions were also seen and it was agreed that all participants would calculate the perturbation worths of the various isotopic composition changes using homogeneous cell models. The results of these calculations are also compared in the same paper. It was also evident that the detailed modelling of the pin and plate cell heterogeneities was important. Two papers by Grimstone and Rowlands present the techniques used in the UK.

In the longer term, the participants have agreed to calculate the voided pin and plate cores, the worths of exchanging pin and plate elements in large and small zones, the effects of plate-cell heterogeneity on reactivity and reaction-rate distributions and reaction-rate ratios in the standard plate cell. Details of these measurements and of the experimental results have been issued to the participants.

The worths of removing or adding sodium in small zones of 9 elements were measured in all four assemblies. These worths were calculated using first order perturbation theory with the XYZ fluxes and exact/FOP corrections to the non-leakage, radial and axial leakage terms were then estimated from RZ model. The correction factors to these terms to bring them into agreement with the experiment are summarised in Table 1. Comparison of the factors for the sodium flooded cores with those from previous conventional ZEBRA assemblies shows similar values within the errors for the non-leakage term. For the leakage terms, the factors tend to be higher for the CADENZA cores, possibly due to the natural uranium breeders in these assemblies compared with the more realistic breeders in the earlier systems. For the voided cores the correction factors, particularly those for non-leakage and radial leakage, tend to be lower than those from corresponding versions with sodium present, which suggest that some caution is necessary in applying the factors in Reference 2 to power reactor situations where sodium is returned to a voided region.

The large-zone voiding measurements, which were made during the conversion to voided assemblies, have been compared with calculations using exact perturbation theory and XYZ diffusion-

theory models. The results are shown in Table 2. In general the agreement between calculation and experiment is good, in line with the k-value predictions for full-scale voiding. For the plates in fact, the agreement for the sodium worths is within the experimental errors. Applying the factors obtained from the comparisons for the 9-element voids has a relatively small influence on the agreement with experiment. It is marginally worsened for plate geometry and slightly improved for pins, while it is the additional uncertainties of about ±3% associated with the corrections which provide the agreement with experiment. Nonetheless, the overall conclusion is that, for the CADENZA cores, the calculations predict the worths of large zone voids in both fuel geometries quite well, the differences between calculation and experiment corresponding to a combination of ±5% uncertainties in the non-leakage and leakage contributions.

The effective neutron source outputs from spontaneous fission and (α,n) reactions in ZEBRA fuel have been measured in three of the CADENZA assemblies. The method used was based on the standard subcritical monitoring technique whereby in a subcritical system which is close to critical, the power level is proportional to the source strength and inversely proportional to reactivity. Calculated corrections were necessary to relate the power level, as measured by the multi-chamber scanning system, to the total neutron production rate in the assembly and to allow for the difference in adjoint weighting of the neutron and fission sources. The results, which are summarised in Table 3, supported the conclusions from recent Harwell work with individual ZEBRA components and, within an uncertainty of ±5%, confirmed the validity of current neutron source predictions for unirradiated fast reactor fuel.

Table 1 - Correction Factors for Terms of Sodium Worths Calculations to give Agreement with Small-Zone Experiments

Core	Required Factors		
	Non Leakage	Radial Leakage	Axial Leakage
CADENZA Core 22, plates with sodium	1.07	1.04	1.04
CADENZA Core 23, pins with sodium	1.02	1.10	1.00
CADENZA Core 24, plates, sodium voided	∿0.97	-	-
CADENZA Core 25, pins, sodium voided	0.95	0.97	1.04
Cores 12, 13, 15 (Flooded plates and pins)	1.04 ± 0.03	1.00 ± 0.20	0.89 ± 0.08

NOTE

Errors in the factors for the CADENZA cores arise from uncertainties in the experimental results (∿2% of the major contributions to the perturbation), least square fitting to the sets of results and the assumption that the factors are independent of position in the core zones (∿2%), the exact/FOP corrections (∿2%), and the error associated with reactivity scale in each core (2%). There is, of course, also a systematic error common to all the cores, arising from uncertainties in the delayed neutron data used to calibrate the reactivity scale.

The factors for Cores 12, 13 and 15 are taken from Reference 2.

Table 2 - Comparison of Calculated and Experimental Worths of Large-Zone Voiding (Units 10^{-2} dk/k)

Geometry	Elements Voided	Calculated Worth				Experimental Worths	C-E
		Non-Leakage	Radial Leakage	Axial Leakage	Total		
Plates	25	0.277	-0.043	-0.177	0.057 (0.068)	0.060 ± 0.004	-0.003
	69	0.680	-0.268	-0.415	-0.003 (0.018)	-0.019 ± 0.012	0.016
	129	1.094	-0.810	-0.635	-0.350 (-0.332)	-0.357 ± 0.021	0.007
	172	1.316	-1.323	-0.732	-0.740 (-0.730)	-0.733 ± 0.027	-0.007
Pins	25	0.256	-0.034	-0.196	0.026 (0.027)	0.014 ± 0.013	0.012
	69	0.639	-0.221	-0.462	-0.044 (-0.054)	-0.083 ± 0.012	0.039
	129	1.031	-0.681	-0.703	-0.353 (-0.401)	-0.388 ± 0.011	0.035
	168	1.229	-1.073	-0.802	-0.646 (-0.728)	-0.692 ± 0.009	0.046

NOTES

1. The values in brackets are after the application of the factors from Table 1 for the 9-element voiding experiments to the three calculated perturbation contributions.

2. The systematic error from the experimental reactivity scale is not included in the uncertainties shown.

3. Both cores contained ∿220 elements.

Table 3 - Comparison of Measured and Calculated Neutron Source Strengths

Assembly	Experiment	Calculation	C/E
Core 22 (metal)	$(8.07 \pm 0.51) \times 10^7$	7.90×10^7	$0.979 \pm 5.9\%$
Core 23 (mixed oxide)	$(11.12 \pm 0.70) \times 10^7$	11.57×10^7	$1.040 \pm 6.2\%$
Core 25 (mixed oxide)	$(12.16 \pm 0.79) \times 10^7$	12.23×10^7	$1.006 \pm 6.4\%$
Core 23 / Core 22	1.378 ± 0.056	1.465	$1.063 \pm 3.6\%$
Core 25 / Core 22	1.507 ± 0.056	1.549	$1.028 \pm 3.8\%$

Harwell and Winfrith are participating in the International Fission Mass and Counting Comparison which was set up following comparisons of reaction-rate ratios determined by different laboratories (3,4). Two U235 deposits weighing ∿350μg were prepared at Harwell, counted in low geometry detectors at Harwell and Winfrith and then sent to ANL East. Preliminary indications are that good agreement is found between the alpha emission rates measured at all three laboratories.

3.2 Control Rod Worths and their Effect on Power Distributions in a Power Reactor

Comparisons have been made between calculated and measured worths of boron carbide absorber rods in the Prototype Fast Reactor. The reactor has a ring of five control rods in the inner core and a ring of five shut-off rods in the outer core. The methods developed to predict absorber rod worths in fast reactors, and their validation using ZEBRA experiments and theoretical studies have been reported previously (5). These methods have been used to predict the worths of the rods in PFR. C/E was 1.07 for the control rods and 1.19 for the shut-off rods. This ratio shows agreement with the estimated uncertainties (5% for the measurements and 6% for the calculations) for the control rods, but the discrepancy for the shut-off rods is being investigated further.

Another paper considers the effects of the representation of the PFR control rods at one-third insertion (as at the beginning of a cycle) throughout a reactor run. Initial calculations have shown that the associated errors in calculated fuel compositions, when used in subsequent flux calculations produce errors in fluxes of ∿2% within the core for a single operational period. This error is dominated not by the difference in fissile content of the core but by the amounts of U238 capture producing Pu239 in the upper axial breeder.

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Calculations for a commercial fast reactor design have shown that the modelling of the movement of operating rods and their use in minimising peak subassembly power are important in obtaining the best radial form factor. Enrichments chosen to give a higher peak subassembly power in the inner core than in the outer core at the end of cycle can help improve the form factor at the worst point (start) of cycle. These calculations are discussed in a separate paper.

3.3 Distributed Heating Effects

The method for calculating distributed heating effects in TRIZ geometry using diffusion-theory methods to treat neutron and gamma-ray transport has been successfully used with an equilibrium core model of the PFR in 60° sector geometry. The gamma-ray diffusion calculation used six triangles per subassembly in the plan and 20 axial meshes, as for the neutron calculation. It was concluded that the extra complication and computing time required to use a finer mesh in the gamma-ray calculation is not worthwhile when compared with the overall accuracy of the diffusion-theory method. 37 and 13 groups were used for the neutron and gamma claculations respectively.

The method has also been used with the standard PFR 6-group structure to treat neutron diffusion and to calculate the gamma-ray source. Provided the neutron spectra used to condense the neutron cross-sections are satisfactory, this use of a six neutron-group structure introduces errors in the gamma heating distribution significantly smaller than those introduced by the use of diffusion-theory methods. The error introduced in the non-gamma heat distribution is also small (<1%).

The calculation of the heat distributions neglecting gamma transport and including the energy carried by the neutrons as being deposited at the point of neutron birth introduces significant errors in non fissile regions.

This work is described in detail in a separate paper to the NEACRP.

The above methods do not allow calculations of ratings of individual components such as samples in materials-testing sub-assemblies or absorber pins in control rods. This information is needed for experimental analysis or safety assessment. Some attempt has been made to model such assemblies using simple cylindrical models but these are of very limited application. It is planned to use the MONK6 Monte-Carlo code which has been identified as suitable for calculating these complex geometries in detail.

3.4 Bowing Predictions and Damage Gradients

The COSMOS workshop program BOWHIST has been used to compute subassembly distortions and interactions for the early stages of PFR operation. Analysis shows non-symmetric bowing when one subassembly in a cluster, having relatively little resistance to bending is forced out by the others with greater resistance to bending. It has also been found that the presence of a lowly-rated subassembly (eg. an experimental demountable subassembly) can induce thermal bowing in neighbouring standard subassemblies in the reverse direction to normal.

A new version of the CRAMP subassembly distortion code has been produced. The principal new feature is the representation of control rod distortion. The distortion of the rod and the guide tube are both computed, and the interaction between them, as a function of control-rod insertion. The program is being tested in the context of PFR operation.

The Nb93(n,n')Nb93m reaction is of interest in fast and thermal reactor investigations as a monitor of neutron damage. Its differential cross-section approximates to that for neutron damage in steel and the Nb93m half-life of 16 years ensures good integration of neutron fluence in long irradiations. Unfortunately there is a lack of experimental information about its cross-section above about 2.5 MeV.

In 1982 a collaborative experimental programme to measure the differential cross-section was undertaken using the Dynamitron accelerator at Birmingham University. The participants are; Reactor Physics Division at Winfrith, Nuclear Physics Division at Harwell, and the Radiation Centre at Birmingham. To date six energy points have been measured, spanning the range 1 MeV to 6 MeV; further measurements are planned for November 1983. It is expected that the target accuracy of ±5% will be obtained for most of the energy points. The results obtained so far are in agreement with the calculations of Strohmaier and, in the lower energy region, with the experimental results from other workers.

3.5 Gamma-Ray Energy Deposition

A detailed re-appraisal of the TLD technique used to measure gamma-ray energy deposition in the MOZART and BIZET programmes has been performed in support of the collaborative MASURCA RACINE programme. The experimental evaluation and the calculational analysis are nearing completion and comparison will be available later this year. An overall improvement of confidence in the measurements has been provided by the studies summarised below.

(a) Electron Migration

In the previous ZEBRA work the PROCEED Monte Carlo Code was used to relate the energy absorbed in the TLD to that in the surrounding material, assuming a homogeneous medium around the TLD and an isotropic photon flux. For the 1mm x 1mm x 6mm 7LiF TLD used, surrounded by iron or zirconium, the correction is about 10% in core regions, increasing to about 15% in a region where there is a greater low energy photon component. These corrections are comparable with the target accuracies of the design calculations for both thermal and fast power reactors. To represent the more complicated geometries achieved in practice and to take account of the photon source distributions, an electron transport module has been developed for the standard gamma-heating Monte Carlo Code, McBEND. Validation of this module is nearing completion.

(b) Absolute Photon Calibration

TLD do not provide an absolute measure of energy deposition and must be calibrated against a primary dosemeter. To relate the quantity measured by the primary dosemeter, to the energy absorbed by the TLD requires a calculated factor which consists of a component due to photon attenuation and a component due to electron migration. The magnitude and uncertainty of the factor is minimised by positioning the TLD within a medium immediately after its electron build-up region. A series of experiments has been performed to test the quality of calibration procedures and in particular the prediction of the calibration factor. The absolute energy fluence from a range of photon spectra obtained at the UK National Physical Laboratory (Co60, 1 MeV and 2 MeV x-ray spectra), incident on various media (LiF, Perspex, Fe), provided the input source to the calculations. For the five combinations of spectra and build-up media chosen, the range of the variation in the cavity correction (ie. electron migration effect) was about 20%. The mean corrected absolute energy deposition per unit TLD read-out light-count had a root mean square deviation of 1.5%.

A secondary standard ionisation chamber, also calibrated at NPL, was compared with chambers of the RACINE partners at Cadarache, 1m from Co60 source. The values obtained were in good agreement, ie. UK chamber 2.92 R/hr ± 1.6%; French chamber 2.94 R/hr; Belgian chamber 2.87 R/hr, providing a good common base for comparing the reactor measurements.

(c) Neutron Contribution to the TLD Signal

Two major investigations of the fast neutron response of 7LiF TLD have been carried out in the USA and Japan. A data set prepared from these results has been used in all previous ZEBRA work and typically predicts a 20% neutron contribution to the TLD signal. However, recent doubts about this work prompted a series of experiments to check the fast neutron response. Neutrons were generated at nine energy points in the range 200 keV to 6 MeV using the Dynamitron machine at the Birmingham Radiation Centre. The measured response factors (photon equivalent energy deposition per unit neutron fluence) were, in fact, in good general agreement with the previous two studies and confirmed the suitability of the existing response data set.

This work also led to the investigation of a second technique for establishing the neutron component, based on the high temperature read-out count. The crystals are normally readout using a fast ramp and plateaux heating cycle of 16s at 135°C to empty the shallow traps followed by 32s at 240°C, which provides the main integrated counts normally used in practical dosimetry, and finally 16s at 300°C to empty the residual deeper traps. The ratio of the light count from the 300°C plateaux to the 240°C plateaux for TLD-700 irradiated solely by Co60 gamma-rays is 0.05. The neutron calibrations showed this ratio ranges from about 0.3 at low energies to 0.4 at the upper calibration energies. Advantage can therefore be taken of this change in ratio to deduce directly the gamma and neutron components in a mixed-field irradiation. For the RACINE measurements comparison of the two relatively independent methods for determining the gamma component showed 70% of the results agreed to within 3% and 90% to within 6%.

4. CRITICALITY WORK

The DIMPLE water-moderated zero-power reactor at Winfrith has been restored and recommissioned. A simple 3% enriched UO_2 pin lattice, identical to an earlier DIMPLE core is presently loaded. Current experimental techniques are being tested and the results will be compared with those in the first version. This assembly will be followed by critical and subcritical arrangements of the 3% UO_2 pins in the prototype BNFL CAGR storage and transport skip which has been specially acquired.

Further validation of the MONK6 Monte-Carlo code and data has been carried out. Substantial progress has been made towards the utlimate aim of producing a code to predict keff to within 1% for all systems. Updating the uranium and plutonium data in the UKNDL is expected to bring that aim within sight. The results are described in a separate paper which also notes the value of inter-code comparisons, both within the UKAEA and through the recent international CSNI comparison exercise, and suggests some areas where good quality benchmark criticality experiments are still required.

5. SHIELDING STUDIES

The programme of data-testing benchmarks in the ASPIS shielding facility of NESTOR is continuing. Detailed studies have been carried out in graphite, iron and sodium, the last of these being irradiated with both the standard fission source plate and a simulated natural-uranium oxide breeder. Winfrith has collaborated with the ESIS Shielding Group at Ispra who have made complementary measurements in iron and sodium in their EURACOS II fission-plate facility operated at the University of Pavia.

The Winfrith programme is now turning to the materials used in transport flasks: in addition to measurements in water which have utilised a Californum-252 source to reduce the neutron background for spectrometer operation, benchmark experiments are planned in lead, concrete, polythene, boro-silicone and Jabroc. These measurements will be analysed in collaboration with BNFL using the McBEND-DUCKPOND sensitivity Monte Carlo code. The results of these programmes, which will include adjustments of the data, will be made available to the JEF evaluators for the benchmark testing of the new files.

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6. FORTRAN-77

Fortran-77 compilers are now available on all UKAEA mainframe computers and are in use. The CEGB have recently acquired such a compiler on their IBM machine in London. Many staff throughout the Authority have been on Fortran-77 conversion courses and are using Fortran-77.

The use of non-standard features has been considered at Winfrith and Risley. It has been agreed that in some circumstances, the additional facility obtained by the use of non-standard features outweighs possible reduced portability, particularly in large programs. The main area in which conformance to the standard would restrict programmers is in the simulation of dynamic run-time storage allocation.

It is intended to acquire a Fortran-77 Verifier specifically designed to flag the use of some non Fortran-77 features. However it is unlikely that the Verifier will be able to check interfaces between called and calling routines or to check violations which can only be detected during actual execution of a program.

REFERENCES

1. HALSALL M J. The Treatment of Burnable Poison Pins in LWR-WIMS. AEEW-M1999.

2. BUTLAND A T D, SIMMONS W N, STEVENSON J M. An Assessment of Methods of Calculating Sodium-Voiding Reactivity in Plutonium-Fuelled Fast Reactors. Proceeding of a Symposium on Fast Reactor Physics, Aix-en-Provence, September 1979. Volume 1, p281.

3. MADDISON D W, INGRAM G. ANL/AEEW Comparison of Reaction-Rate Ratio Techniques in ZEBRA. NEACRP-A-542.

4. BOHME R, BURBIDGE B L H. A Comparison of Central Reaction-Rate Ratio Measurement Techniques in BIZET Cores. NEACRP-A-543.

5. ROWLANDS J L, GRIMSTONE M J. Fast Reactor Control Rod Calculation Methods and Validation Experiments. NEACRP-A-514.

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Reactor Physics Activities in the United States
A Report to the NEACRP
October 17-21, 1983

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Introduction

Reactor physics activities in the U.S. have provided support necessary for the operation of FFTF and licensing reviews for CRBR. Operating measurements in FFTF have confirmed the adequacy of the methodology used for FFTF core and shielding design. Benchmark analyses of FFTF operating and burnup physics are continuing using improved data and methods.

The U.S. critical experiment program now uses only the ZPPR facility at ANL-Idaho. Using ZPPR, a series of large heterogeneous cores are being studied as part of the Jupiter II cooperative program with PNC, Japan.

Assessments of the state of the art in LMFBR physics have continued at ANL. Analyses of the sodium void reactivity worths were completed in support of CRBR licensing. Major assessments of Doppler coefficient and power distribution predictions are in progress.

Reactor design studies are continuing at industrial and national laboratory organizations to evaluate core design alternatives offering improved economics, reliability and licensability.

Critical Experiments

The ZPPR-13 series of the Jupiter II program began in June 1982 as a cooperative program with PNC, Japan. This program was designed to provide a systematic study of core neutronics for the design and analysis of large LMFBR heterogeneous cores. The ZPPR-13 series of benchmark configurations is shown in Figures 1 and 2.

The first core, designated 13A, contained a central blanket region and three alternating annular rings of fuel and blanket regions. The core volume was about 4000 liters and the critical mass was about 2500 Kgs of plutonium.

The initial measurements were of excess reactivity, gap worth, temperature coefficient, safety rod worth, shim rod worth, source term and the relative abundance of the delayed neutron groups. Next, an extensive set of U-235, Pu-239 and U-238 foils was irradiated to study the spatial distributions of the U-235 (n,f), Pu-239 (n,f), U-238 (n,f) and U-238 (n,γ) reactions. Distributions were measured radially along the x and y axes and at 30^0 and 60^0 to the axes. Axial distributions were also measured within drawers in fuel rings 1 and 3. Small sample worth measurements were made for Pu-239, U-235, U-238, B-10, graphite, iron, stainless steel and Cf-252. Following the small-sample worth measurements, the reactor was made 0.10$ subcritical and a set of control rod worth measurements was made.

The next two cores, 13B-1 and B-2, were segmented blanket ring configurations. The first measurements in ZPPR 13B-1 were operational measurements of the type carried out on the 13A core. With the reactor in its reference critical configurations, distributions of neutron reaction rates and gamma doses were measured using foils and thermoluminescent detectors respectively. With the reactor in a 0.09$ subcritical configuration, worths were measured for several control rod patterns. These included two-by-two drawer control rods, groups of rods in single rings, and interactions between groups of rods in different rings. Further control worth data were obtained for rods (two-by-three drawers in size) located in the gaps of the blanket rings. Measurements in this case included worths of individual rods, individual sodium-filled rod locations, groups of rods and groups of sodium-filled rod locations. In total, thirty separate control rod worth measurements were made in the ZPPR 13B-1 configuration.

Configuration 13B-2 provided a reference transition to hex geometry. The conversion from 13B-2 to the blanket island configuration of 13B-3 was accomplished by a series of alternating negative reactivity steps (substituting blanket drawers for fuel drawers) and positive reactivity steps (substituting fuel drawers for blanket drawers). Since flux distributions were measured after each step, the conversion served as blanket thickening and thinning experiments.

A series of pin and plate control rod worths were measured in the ZPPR 13B-4 engineering benchmark core. This configuration is similar to 13B-3 with the addition of 30 mockup control rod positions. Worths were obtained for an extensive set of rod groups with particular emphasis on the interaction between groups and the effects of a single withdrawn rod. The properties of loosely coupled cores are now being studied in core 13C. These include the decoupling of the kinetic response and the spatial sensitivity to compositional perturbations.

In the ZPPR-13 series to date, the experimental reaction rates were generally well calculated. The C/E radial bias of 2-6% in fission rates agrees closely with the radial bias in control rod worths. A small (1-3%) asymmetry was observed in azimuthal reaction rates. This was traced to a small asymmetry in blanket cell loadings and to fuel gap uncertainties (which were on the order of 1 mm) on closure of the split table assembly. The control rod position worths were not well calculated by diffusion theory. The observed variations of 10-20% indicate the need for a better treatment of neutron streaming in the sodium channels.

Computational Physics Methods

Development of an integrated design capability adequate for optimization and detailed design of LMFBR plants up to the largest practical ratings has progressed significantly. Much of 1983 effort was directed to the needs for efficient energy-dependent solutions within the range of the following variations:

- Reactor size
  Large sizes are most demanding and are receiving the most attention.

- Computer hardware
  Most implementation is on CDC, IBM, and more advanced CRAY and CYBER 205 equipment.

- o Rigor - diffusion and higher order transport approximations.
- o Methodologies - finite difference and advanced nodal.
- o Mode - forward and adjoint.
- o Mesh geometry - RZ, XY(Z), HEX(Z), etc.
- o Particle - neutron, gamma, neutron-gamma

REBUS-3 was developed at ANL to perform burnup calculations for fast reactor design. The code now runs successfully on the CRAY-1. There are diffusion theory options based on finite difference, nodal, and flux synthesis methods, and multiple-cycle, non-equilibrium calculations with fuel enrichment in selected regions at each cycle are permitted. The code has general fuel shuffling capability.

A new analytical model was developed to predict the reactivity effects of irradiation-induced subassembly bowing in an LMFBR. This model is the first which can represent heterogeneous core geometries in the necessary detail. A computer code based on this work has been written and linked with output files of neutronics, depletion, and thermal hydraulics codes.

A version of the TWODANT transport code has been developed by LANL for trial usage. This transport code solves eigenvalue and inhomogeneous source problems in (x,y) and (r,z) geometries. It employs the diffusion synthetic acceleration method with a multigrid scheme for solving the diffusion acceleration equations. Arbitrary order-of-scattering is permitted.

A companion to TWODANT for analysis of hexagonal geometries is TWOHEX. It uses an equilateral triangular mesh with spatial differencing based on the linear characteristic method. The latter is significantly more accurate than diamond-like differencing for triangles and requires no negative flux fix-up. TWOHEX uses 60° rotationally-invariant quadrature to ensure 60° symmetries and currently employs Chebyshev acceleration on both inner and outer iterations. Performance of this code is undergoing initial user tests at Los Alamos.

In other work, an advanced depletion module consisting of the LANL CINDER code and the ANL DIF3D diffusion code is being developed. CINDER performs detailed summation calculations for 877 nuclides in 40 actinide and 102 fission product chains with data for 31 ENDF/B-V yield sets. The module will be optimized for the CRAY computer.

Nuclear Data

The efforts to improve the U.S. nuclear data files for reactor design resulted in issuance of several updates and changes which were designated ENDF/B-V Mod II. These included revision of the Fe, Th-232, U-239, Li, W, Ag, Rb, Pu-239, Kr, Ag, Eu, Xe, Gd, and Zr data files. Data testing of the Mod II data indicate small but significant improvements over the ENDF/B-V data. Development of the

ENDF/B-VI data file is in progress. Initial efforts include a consistent evaluation of the standards files with the key reaction cross sections important in reactor design.

ANL scientists measured cross sections for several stable fission products near the light mass fission yield maximum (A-85-125). The new results differ from previously used data by 50-100% in some cases. The new data will be used in updates of the ENDF/B files and to improve nuclear model predictions for experimentally inaccessible radioactive fission products. Measurements of higher actinide cross sections have continued at ORELA. These include total, fission and capture measurements on Am, Th, and Pu-240 as well as the extension of U-238 resonance parameters up to 6 Kev. Measurements of $\nu(E)$ were completed for U-233, U-235, Pu-239 and Pu-241.

Shielding

Studies of the FFTF shielding have been carried out by HEDL and ORNL. Measurements of the radiation fields in FFTF were made in the reflector, the in-vessel storage locations, the reactor cavity, and the head compartment regions. The major shielding measurements were made during an 8 day FFTF run at full power; however, some measurements were made at low power and some with 36 spent fuel elements stored in-vessel. Analyses to date indicate generally good agreement between the FFTF measurements and calculations based on TSF benchmark experiments. The use of bias factors obtained from benchmark measurements was necessary to provide good predictions of the FFTF shield performance. The ringed geometry modeling which was used to calculate the spent fuel multiplication appears adequate.

ORNL and GA have investigated the neutron streaming through the exit cooling passages of the lower core reflector and structural core support region in high temperature gas cooled reactors. Mockup measurements using the Tower Shielding Facility indicate substantial streaming effects which are not well predicted by the design methods currently used. Results to date indicate the need to use measured bias factors and/or improved computational techniques to predict neutron streaming through the lower core region.

Shield design studies have been carried out on large LMFBR plant concepts by ORNL, GE, AI and ANL. These studies have indicated that substantial weight and cost savings can be obtained by the use of alternate shield materials and appropriately optimized shield configurations.

FFTF Physics

Activities included core characterization measurements, operating physics measurements and benchmark analyses. Papers on the core characterization and capture rate measurements will be presented at the November 1983 ANS meeting.

Physics measurements in cycle 2 included:

- o Individual BOC rod worths by a combination of inverse kinetics rod drop and modified source multiplication techniques.
- o Burnup reactivity by change in position of calibrated rods.
- o Single assembly substitution worth.
- o Close monitoring of the power coefficient of reactivity and stability phase margins.

Calculations of rod worths, burnup reactivity, and assembly worths agreed quite well with measurements. The measured power coefficients and stability phase margins were well within requirements. Thirty-three experiments were continued and nine new ones were loaded at the beginning of cycle 2 in January 1983. A capacity factor of 83% was achieved over the cycle.

Cycle 3 began on schedule July 4, 1983. A key goal for this cycle is to extend peak pellet exposure; about 75-80 MWd/Kg has been achieved so far. Seven new experiments were added at BOC 3.

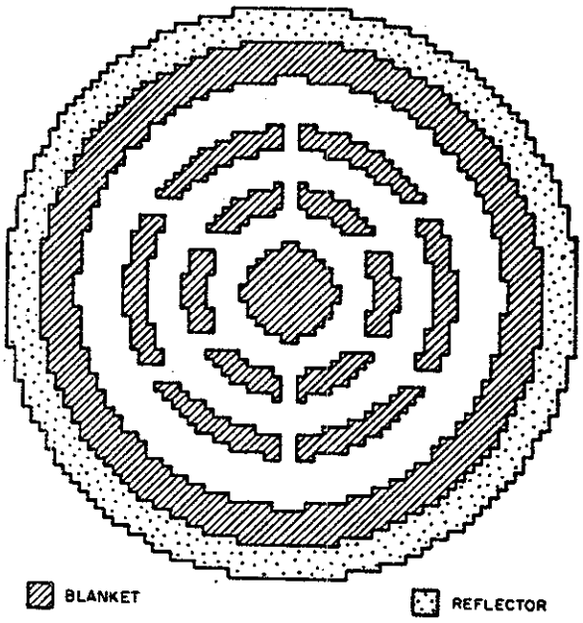
Measured FFTF data are being used as benchmarks for testing neutronic codes and data. Calculations have been performed for cold condition criticality and control worths using various models and data sets. These will be extended by near-term analyses of burnup data from the early FFTF cycles.

Core Design and Assessments

Assessments of the state-of-the-art for calculating key fast reactor physics parameters are continuing at ANL. Recent assessments address our ability to predict criticality, reaction rates, Doppler and sample worths.

Core design studies are carried out at the major industrial contractors and national laboratories. The major issues being addressed are design tradeoffs to increase inherent safety and reduce plant and fuel cycle costs.

Figure 1



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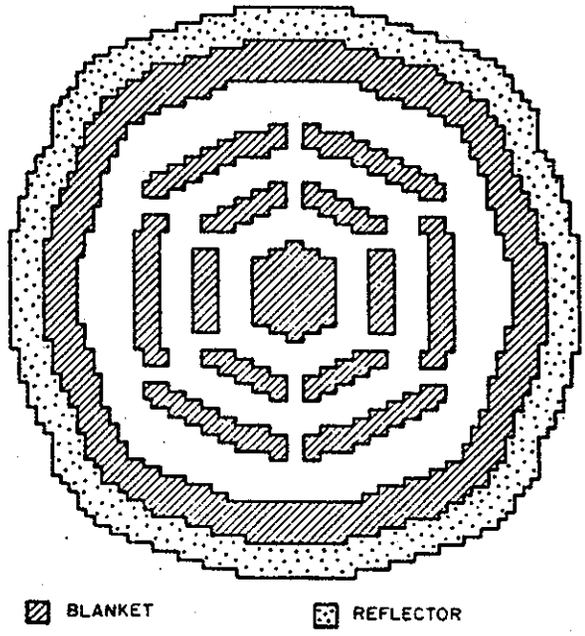

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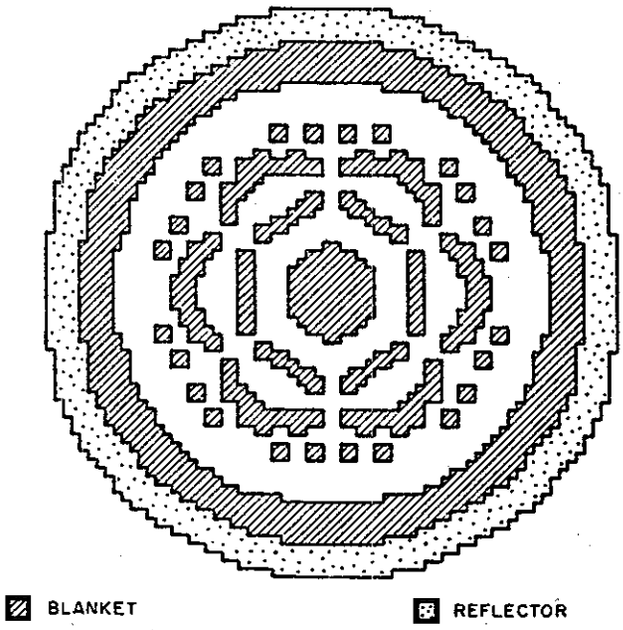

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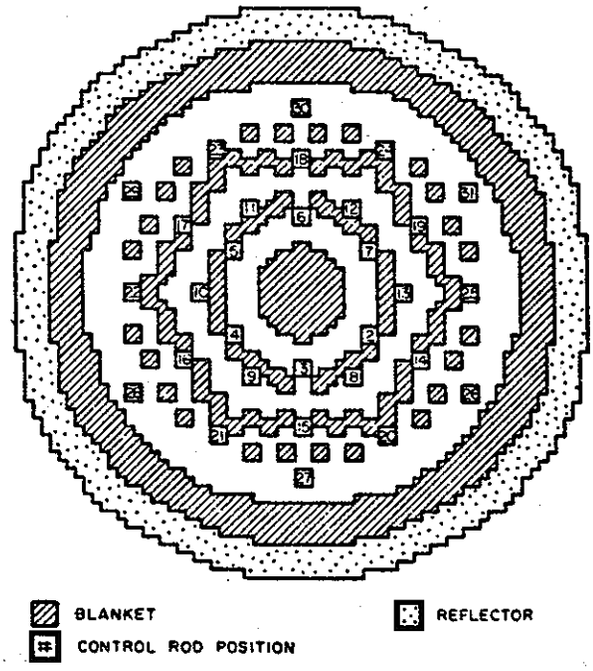

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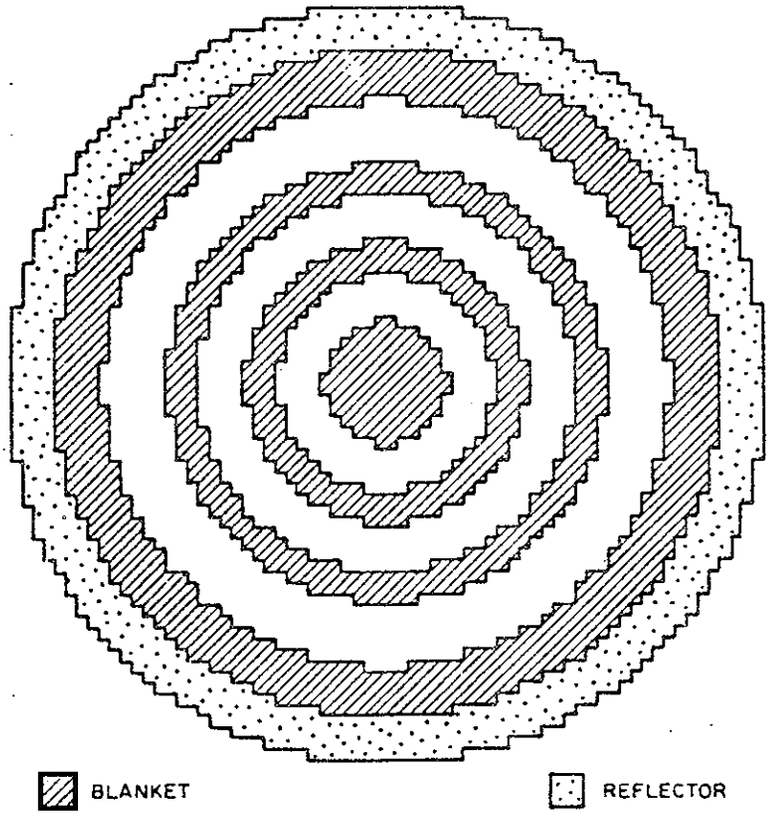
Figure 2



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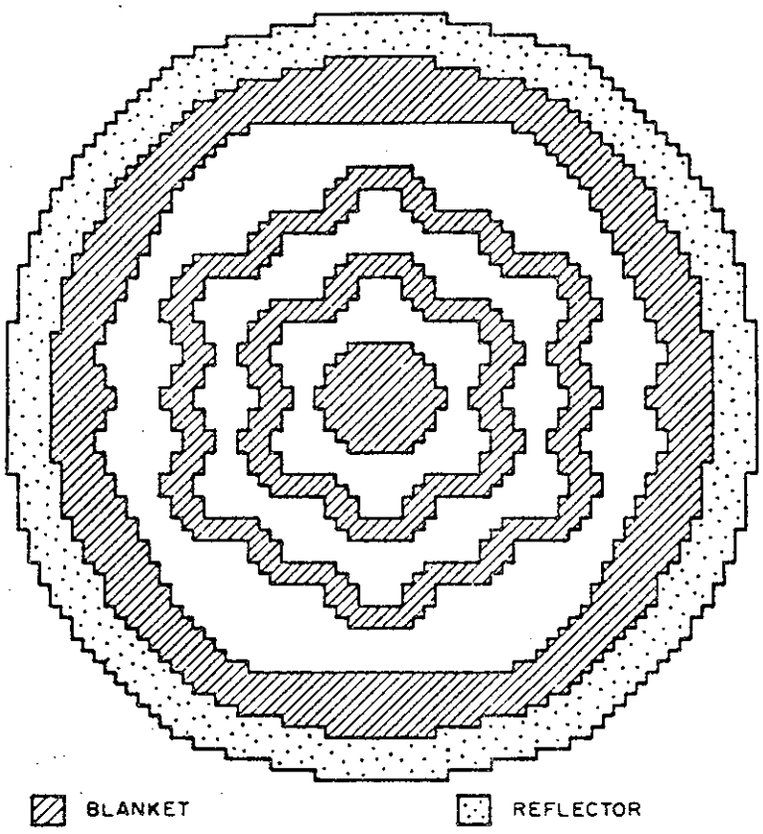

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REACTOR PHYSICS ACTIVITIES AT THE JRC ISPRA

I. CORE PHYSICS STUDIES

Planning and design of a large scale loss of coolant experiment typical of LWRs led to a considerable intensification of the reactor physics activities at the JRC-Ispra. The project under investigation consisted of a light water test loop (containing 32 fuel pins) immersed in a heavy water reactor of a high thermal flux generated by fully enriched uranium aluminum fuel elements.

The main objectives of the study which includes among its tasks a series of rather unusual problems are given in the following summary:

- To satisfy the condition of a radially flat power and decay heat distribution in the test channel an enrichment scheme depending on the fuel pin position had to be developed. The power distribution analysis was performed by DOT 3.5 and validated by KENO-IV.
  An "a posteriori" comparison with mock-up measurements fully confirmed the theoretical results. ORIGEN calculations of the decay heat sources for only 48h of irradiation time gave poor results.
- The axial power distribution of the test channel had to be determined for different control rod positions. This problem was solved by the use of 3D few group diffusion codes such as SYNTH-C and VENTURE-2.
- All parameters needed for controlling the reactor with the test loop in operation had to be verified and if necessary reanalyzed. Particular attention had to be paid to the positive void coefficient following a voiding of the test channel.

For this task DOT 3.5, KENO-IV and a newly developed Monte Carlo perturbation scheme based on correlated tracking and a second order Taylor series approach were used. With these new methods perturbation effects as small as a few pcm could be reliably estimated (e.g. temperature coefficient of the coolant).

- Finally the mock-up experiments performed for different test loop configurations were analyzed to validate the theoretical and calculational methods in use.

Reports

- "Confronto fra Calcoli di Fisica e Valori Sperimentali per il MOCK-UP di SuperSara in ESSOR"
  E. Caglioti, R. Ricchena - TN/1.05.01.83.65 (Agosto 1983)

- "Improved Neutronic Analysis of the SuperSara Experiment by the 3-D Few Group Diffusion Code Synth-C"
  E. Salina - Contact EURATOM/ARS N.1667-81-12 ED ISP I

- "Reactor Physics Analysis for the SuperSara Test Project using Monte Carlo Method"
  Tayyab Abbas, M. Aglietti-Zanon - Contract TEAM/EURATOM N. 1668-8112 ED ISPI (Feb. 83)

- "Core Physics Analysis for the SuperSara Test Project using Venture-2 Diffusion Theory Code"
  M. Aglietti-Zanon, Tayyab Abbas - Contract TEAM/EURATOM - N. 1828-82-03 ED ISP I (May 83)

- "Generalized Monte Carlo Perturbation Algorithms for Correlated Sampling and a Second Order Taylor Series Approach"
  H. Rief - submitted to Annals of Nucl. Energy (Oct. 1983)

II. RADIATION SHIELDING

During the reporting period a deep penetration sodium benchmark experiment has been carried out.The sodium column in which the measurements were performed was about 400 cm long and consisted of seven steel boxes of approximately 200 X 150 cm cross section filled with sodium. The neutron source was a highly enriched U235 converter plate. Measurements were performed along the central axis of the sodium assembly, using activation and threshold detectors for the different energy regions and proton recoil counters. In Fig. 1 activation rates for sulphur and gold are shown as functions of the source detector distance.

Interpretation of the iron and sodium benchmarks has continued simultaneously with the measurements. The first results of group cross section adjustments were presented to two international Conferences (Antwerp, Sept. 1982 and Tokyo, May 1983).

In the case of iron they indicate, for example, that the inelastic cross section of ENDF/B4 is about 5% too high near the threshold.

Publications

- W. Matthes et al.
  Adjustment of Neutron Multigroup Cross Sections with Error Covariance Matrices to Deep Penetration Integral Experiments
  Paper presented at the International Conference on "Nuclear Data for Science and Technology" - Antwerp, Sept. 1982

- "A Parametric Representation of Gamma Ray Attenuation in Two-Layer Shields"
  H. Penkuhn (EURATOM-Ispra), Schultz (U. Hannover, Germany) - 6th ICRs, Tokyo (1983)

- "Monte Carlo Shielding Analysis Using Deep Penetration Biasing Schemes Combined with Point Estimators and Algorithms for the Scoring of Sensitivity Profiles and Finite Perturbation Effects"
  H. Rief (EURATOM-Ispra), A. Fioretti (A.M.N., Genova-Italy)
  6th ICRs, Tokyo (1983)

- "Adjustment of Neutron Multigroup Cross-Sections to Integral Experiments"
  G.Hehn, R.D.Bächle, G.Pfister, M.Matthes (IKE, U.Stuttgart, Germany) , W.Matthes (EURATOM-Ispra) - 6th ICRs, Tokyo (1983)

- "On Unfolding Counting-Rate Spectra of Recoil Proton Neutron Detectors", Y. Yeivin (Hebrew Univ., Jerusalem; visiting scientist at the JRC-Ispra) - 6th ICRs, Tokyo (1983)

- "MORSE-E1, Source and Analysis Routines to be used with the MORSE Code"
  C. Ponti and R. Van Heusden - EUR 8320 EN (1983)

- ESIS Newsletter 43 (October 1982, H. Penkuhn - "A Shielding Kernel for Fission Gammas - Part I: The Unscattered Flux from a Point Source"

- ESIS Newsletter 44 (January 1983)

- ESIS Newsletter 45 (April 1983), H. Penkuhn - "A Shielding Kernel for Fission Gammas - Part II: The Scattered Energy Flux from a Point Source

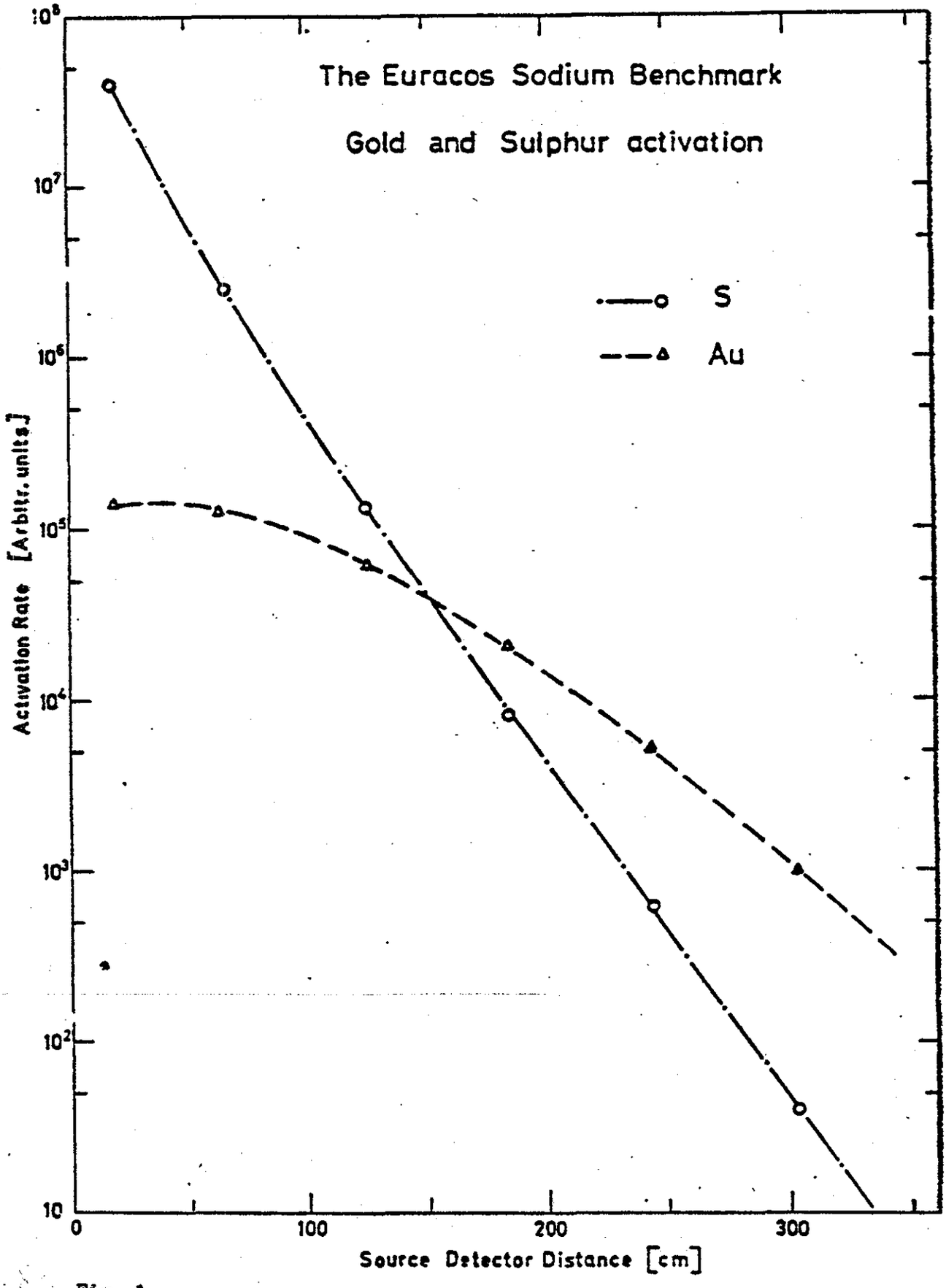

Fig. 1

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III. ACTINIDE MONITORING

In the field of nuclear safeguards and monitoring of α-contaminated waste streams special work has been carried out applying the passive neutron interrogation technique.

For the existing shift register technique measuring the doublet neutron signals of spontaneous and induced fission events corrections for induced fission counts were carried out. Investigations are continued to obtain from the analysis of multiplets of higher order additional information on the physical state of the test items.

References

"Neutron Multiplication Corrections Applying the shift Register Technique", W. Hage, L. Anselmi, K. Caruso - ESARDA, Proceedings of the 5th Annual Symposium on Safeguards and Nuclear Material Management, (Versailles, France) JRC Ispra Publication (1983)

"Neutron Signal Multiplet Analysis for the Mass Determination of Spontaneous Fission Isotopes", R. Dierckx, W. Hage - Nuclear Science and Eng., to be published