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NEA COMMITTEE ON REACTOR PHYSICS

# REACTOR PHYSICS ACTIVITIES IN NEA MEMBER COUNTRIES

October 1985 - September 1986

OECD NUCLEAR ENERGY AGENCY
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NEA COMMITTEE ON REACTOR PHYSICS

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# NEA MEMBER COUNTRIES

NEACRP-L-295

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# REACTOR PHYSICS ACTIVITIES IN NEA MEMBER COUNTRIES

This document is a compilation of national activity reports presented at the Twenty-Ninth Meeting of the NEA Committee on Reactor Physics, held at Chalk River Nuclear Laboratories, Canada, from 22nd to 26th September 1986.

# REACTOR PHYSICS ACTIVITIES IN AUSTRALIA
October 1985 - September 1986

by

D.B. McCulloch

Australian Atomic Energy Commission Research Establishment,
Lucas Heights Research Laboratories, Sutherland 2232, Australia.

## 1. METHODS OF CALCULATION

### 1.1 AUS Neutronics Code System

The only significant development of the AUS system has been an extension of the available cross-sections. That portion of the ENDF/B V data files which is unrestricted has been acquired from the IAEA Nuclear Data Section and the data processed for the AUS.ENDF2006 group library as required. In particular, a large set of fission product data has been processed, sufficient for calculations of both reactivity effects and fission product inventories.

Some limited tests of the revised fission product data have been made in applications to heat generation in nuclear fuel waste and to the AAEC's research reactor HIFAR. Only minor differences from results using ENDF/B IV data were noted.

## 2. RESEARCH REACTOR STUDIES

### 2.1 Low Enrichment Fuel in HIFAR - Conversion Strategy

A neutronics study of the conversion of HIFAR to the use of fuel with 20% $^{235}U$ enrichment (LEU) was completed. Earlier studies had established the $^{235}U$ loading required to match the fuel element consumption rate obtained with HEU fuel and the effect on performance of using this fuel. The current study was mainly concerned with the effect of varying the $^{235}U$ loading to establish the most appropriate performance/economics compromise for use of LEU $U_3Si_2$-Al fuel in HIFAR. The study established the fuel consumption rates to be expected for a range of rig burdens and the effects on radioisotope production and neutron beam facilities. The neutronics models used had been previously established in studies such as those described in NEACRP-L 273.

Fuel element power peaking and reactivity swing over a fuel cycle were also investigated. It was concluded that an increase in $^{235}U$ loading and reactor power was the preferred option. The result of these performance and fuel-cycle cost studies for a range of possible LEU conversion options and operational modes for HIFAR were used in developing recommendations on conversion strategy to low enrichment fuel.

## 2.2 Low Enrichment Fuel in HIFAR - Consequences for Irradiation Facilities

Although fuel cycle, reactivity and safety aspects of research reactor conversion to operation with LEU fuel of <20% $^{235}U$ enrichment have been very extensively studied in many centres, the influence of the conversion on the experimental facilities and services provided by the reactors has received little detailed examination.

Reactor neutronics calculations which specifically model the HIFAR production facilities for some commercially important radioisotopes were undertaken. The following three radioisotope production reactions, routinely used in HIFAR in-core hollow fuel element (HFE) irradiation facilities, were selected for study:

(i) $^{99}Mo$ production by (n,γ) reaction in natural $MoO_3$ targets;

(ii) $^{99}Mo$ production by (n,f) reaction in 1.8% $^{235}U$, $UO_2$ targets, and

(iii) $^{192}Ir$ production by (n,γ) reaction in pellets of natural Ir metal.

The target size, shape and material composition actually used in HIFAR were modelled in the calculations as well as the target cladding and carrier assembly. The five axial target positions were represented and final results obtained by combining fluxes from RZ and XY models of HIFAR.

Typical results obtained were reductions in activity of (i) 5%, (ii) 14%, and (iii) 10% for the three production reactions. It was also concluded that much less detailed models would suffice in comparing operation with LEU and HEU fuel provided that appropriate shielded cross-sections were used.

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AUSTRIA

REACTOR PHYSICS ACTIVITIES IN AUSTRIA

October 1985 - September 1986

by

F. Putz

## Reactor Calculation

The study carried out at ITP/TU Graz (Institut für Theoretische Physik der Technischen Universität Graz) jointly with the Institute of Reactor Development of the Kernforschungsanlage (KFA) Jülich (FRG) during the last years on the ingress of water into the core region of gas cooled high temperature reactors with spherical fuel elements has been completed and the results have been summarized in a detailed report. The aim of this study was to find out to what extent the code GAMTEREX, a combination of GAM-1, THERMOS-JÜL and the diffusion code EXTERMINATOR-2, and designed for the calculation of high temperature reactor cores, is applicable to the analysis of such accidents as mentioned above. To this purpose the calculated reaction rates and neutron flux distributions have been compared with measurements performed in the SIEMENS ARGONAUT research reactor, the testzone of which had been filled with AVR-fuel elements. The water ingress had been simulated by filling the interspace between the graphite spheres with a mixture of small polyethylene and polystrol particles.

One of the results of these investigations is that in huge pebble bed reactors the maximum of reactivity has been observed at much lower water concentration than in small cores. The measured increase of the effective multiplication factor was in accordance with the calculated reactivity coefficients. The highest reactivity can be expected at a water content of 35% of the interspace volume between the graphite spheres. Summarizing the results the code system GAMTEREX can be ascribed high accuracy when used for the prediction of criticality and neutron flux distributions in pebble bed arrangements.

## Assessment of mixed fuel cycles in HTRs

Within the collaboration over many years between KFA-Jülich and ÖFZS in the field of assessing the transition to low enriched fuel cycles a fresh start on new methods has been undertaken. KFA-Jülich is planning new burn up calculations for special experiments on the AVR reactor with low enriched fuel elements in a surrounding of high enriched Thorium elements.

To support the AVR crew in their calculations a method originating in Winfrith, UK, during the OECD DRAGON Project is being adapted in ÖFZS for the special needs of the pebble bed reactor concerning the resonance absorption of differently enriched fuel elements. To have a consistent set of nuclear data to go with a library condensation to 43 groups was performed on a WIMS-D library for the spectrum-code MUPO.

# REACTOR PHYSICS ACTIVITIES IN BELGIUM

September 1985 - August 1986

compiled by J. DEBRUE (SCK/CEN, MOL)

## THERMAL REACTORS

The validation of calculation methods applied in pressure vessel surveillance programmes is the objective of measurements performed in the VENUS critical facility and in the pressurized water reactors DOEL 1 and 2. This is the main reactor physics activity concerning thermal reactors.

1. The reactor configuration studied in VENUS is an engineering mock-up which simulates rather closely the main features of a PWR core : low enriched $UO_2$ fuel pins, staircase-shaped core boundary, core baffle, barrel, neutron pad (fig. 1). The investigated parameters are the pin-to-pin power distribution, the propagation of the neutrons outside the core, the damage exposure and the gamma heating in the steel structures. Results of measurements and comparison with calculations were reported earlier [1][2][3][4].

   A last campaign of measurements has been performed with the $UO_2$ loading. The primary objective was to determine the reasons and significance of large discrepancies between transport theory and experiment for the $^{237}Np(n,f)$ fission rate traverses in the water gap between the core and the simulated steel barrel along the radial direction of minimum azimuthal fast neutron flux (45° direction). The secondary objective of the campaign was to extend the range of fast neutron flux measurements by means of the $^{115}In(n,n')^{115m}In$ and $^{103}Rh(n,n')^{103m}Rh$ reactions, in particular for radii larger than the barrel; due to the low neutron flux levels, no data were available so far in the simulated neutron pad.

The $^{237}Np(n,f)$ reaction is of special interest because it fits neutron damage distributions through thick steel pieces more closely than do high threshold reactions, like $^{58}Ni(n,p)^{58}Co$ for example. On another hand, C/E for $^{237}Np(n,f)$ has often been observed to be much less than one in power reactor vessel surveillance capsules, and this requires clarification.

The fast neutron transport analyses are performed with DOT 3.5 in (R,Θ) geometry. The $S_8P_3$ computation makes use of 17-group cross-sections derived from the VITAMIN-C 171-group library by space-dependent collapsing; this collapsing is based on spectra obtained by one-dimensional 171-group ANISN calculations. Space-dependent axial bucklings introduced in the (R,Θ) calculations were measured with miniature fission chambers.

The main conclusions can be summarized as follows, keeping in mind that the fast flux variation is about two decades, from the outer baffle up to the neutron pad :

1) The C/E ratios for $^{115}In(n,n')$ decreases with radius along the 45° direction but do not display any azymuthal asymmetry, whatever the water thickness between the core and the experimental location (fig. 2).

2) The $^{237}Np(n,f)$ measurements, carried out with a fission chamber, are now in much better agreement with calculations. The previous discrepancy was due to plutonium-239 impurities (about 80 ppm as deduced from measurements in a pure thermal flux) in the neptunium of the chamber; this increased significantly the signal measured in water at the locations where the thermal-to-fast neutron flux ratio peaks. After correction for this impurity contribution, for thermal fission of $^{237}Np$ and for photofission, the signals of bare and of cadmium-covered fission chambers agree with each other.

3) The spectral indices involving $^{237}Np$, $^{103}Rh$ and $^{116}In$ measured along the 45° direction are in excellent agreement with theory. In particular, spectral softening by the steel components is well predicted in the energy range tested by these measurements, i.e. 800 keV to 3 MeV approximately.

4) At higher neutron energy, spectral inadequacies however exist in the calculations; it is due to inadequacies of the steel inelastic scattering cross-section above 3 MeV in the 171-group (and ENDF/BIV) library adopted for the calculations. This is consistent with PCA and PSF results.

The relative pin-wise power and neutron source distributions used in the fast neutron transport analysis was calculated with 2D transport codes in (X,Y) geometry and 10-group cross-sections [1][2]. Compared to experimental values obtained by gamma scans of the pins, the maximum C/E deviation was found to be about 6 % high, at pin locations adjacent to the baffle (corner pins); whereas the average deviation over the whole core was ± 2 %, which is within the experimental uncertainty. The absolute normalization was determined experimentally in several ways in order to provide the absolute source for the ex-core transport runs [1].
A major objective in the study of the pin-wise power distribution is to determine the accuracy of the calculations near the core periphery which is the region that produces most of the neutrons leaking from the core and damaging the vessel. With respect to this objective a study has been performed recently in the US [5] to test the validity of few-group diffusion calculations based on methods developed by EPRI and commonly applied by the utilities in the frame of core management studies :

- cell calculations performed with the CELL-2 code (previously EPRI-CELL) associated to the ENDF/B-V library;

- core calculation with the PDQ7 diffusion code, using two-group cross-sections derived from the cell calculations (MND model for the thermal group) for the fuel region and two-group cross-sections adjusted on fine-group transport calculations for the outer baffle [6].

This methodology leads to C/E values for the power distribution very similar to those obtained with the sophisticated transport calculation. This conclusion is important because the agreement found allows to use the results of conventional core management calculations as input for the ex-core neutron transport analysis and the neutron fluence in the reactor vessel.

The next step in the VENUS programme is to verify the ability of the methods to calculate accurately a burnt PWR low-leakage core, i.e. with fuel elements containing plutonium in the peripheral positions from the beginning of core life. This strategy has been applied at several power plants to reduce the irradiation and the embrittlement of the vessel. The VENUS fuel loading has therefore been modified and contains now mixed oxide pins in the regions adjacent to the outer baffle. Measurements and calculations will continue to be a joint effort with the participation of HEDL and ORNL in the frame of the U.S. LWR Pressure Vessel Surveillance Dosimetry Improvement Program.

2. Neutron fluence measurements related to the surveillance of PWR pressure vessels at the belgian power plants are also used to verify the validity of the fast neutron transport calculation methods in the out-of-core regions. At the DOEL 1 and 2 power plants, such measurements are performed with neutron dosimeters located

   - in surveillance capsules : five capsules, irradiated since the beginning of operation, were successively unloaded up to now

   - at four azimuthal positions in the cavity at the outer side of the vessel where dosimeters are exposed during periods of four years.

   C/E values for different threshold activation reaction rates are given in table I. The neutron transport calculations are performed with DOT and cross-section data derived from VITAMIN-C, as was made in the analysis of the VENUS measurements. The activation cross-sections for the dosimeters are taken from the ENDF/BV dosimetry file.

C/E Values

| DOSIMETER (THRESHOLD) | SURVEILLANCE CAPSULES | OUT-OF-VESSEL ANGULAR POSITIONS | |
|---|---|---|---|
| | | 39°-321° | 170°-350° |
| $^{58}$Ni (2.8 MeV) | 0.75 ± 0.05 | 0.80 ± 0.05 | 0.72 ± 0.10 |
| $^{54}$Fe (2.9 MeV) | 0.80 ± 0.05 | | |
| $^{63}$Cu (6.6 MeV) | 0.95 ± 0.04 | 0.93 ± 0.07 | 0.89 ± 0.10 |
| $^{237}$Np (0.6 MeV) | 0.77 ± 0.05 | 0.86 ± 0.10 | 0.71 ± 0.08 |
| $^{238}$U (1.5 MeV) | 0.67 ± 0.05 | 0.71 ± 0.04 | 0.64 ± 0.08 |

Preliminary conclusions can be drawn :

1) a general underestimation is observed for the calculated values, but not a significant distorsion between the capsule and the out-of-vessel positions, although the flux variation is about two decades

2) a systematical trend appears according to the type of dosimeter; approximately

| | |
|---|---|
| Cu | C/E $\simeq$ 0.9 |
| Ni, Fe, Ti | C/E $\simeq$ 0.8 |
| $^{238}$U | C/E $\simeq$ 0.7 |

Further analysis will include a better modelization of the capsules and an improved evaluation of the axial neutron leakage in the cavity.

3. An accurate knowledge of the neutron fluence in the reactor vessel is of particular interest for the BR3 case. Let us recall that a partial recovery of the damage in the vessel took place in March 1984. This annealing was performed at a temperature of 343°C; the temperature increase was obtained by circulating the primary water through the vessel, the fuel being unloaded. This operation and the analysis of its effects were described in [6].

4. General information concerning the nuclear power plants : since the start of commercial operation of DOEL-4 and TIHANGE-3 in 1985, seven power units are in operation with a total capacity of 5414 MWe (net). The average load factor was 81.8 % in 1985 and the nuclear share in overall electricity production amounted to 57.5 %.

FAST REACTORS

1. BELGONUCLEAIRE took part to the SUPERPHENIX start-up campaign at zero power and to the measurements of reactivity coefficients at different power levels. A task force has been created to analyse the experimental results.
   On another hand, BELGONUCLEAIRE is performing the analysis of experiments carried out in MASURCA (RACINE programme).

2. The analysis of the IRMA results concerning an interlaboratory comparison of reaction rate measurements in MASURCA is under completion.
   Gamma heating measurements in the frame of the BALZAC programme in MASURCA are being started. Diluant (SS) and absorber ($B_4C$) subassemblies of the superphenix type will be studied.

3. In the frame of the European PAHR program, three experiments will be performed in the BR2 reactor with the aim of studying the long-term coolability of an initially liquid sodium satured LMFBR core debris bed with internal heat dissipation. The first experiment takes place in September 1986. The $UO_2$-Na bed for this experiment is made of

natural $UO_2$ particles which occupy 65 % of the total volume (diameter of the bed : 11 cm). The maximum temperature will reach 1500°C during the irradiation. Detailed (R,Z) neutron/gamma calculations were performed to obtain the power density distribution through the bed. The maximum/average power density ratio (radial shape factor is 84 %; the gamma heating contribution amounts to 26 % on the average. An average power density of 5 $W/cm^3$ is expected, to achieve the central temperature of 1400°C.

An important characteristics of the $UO_2$-Na bed is the axial uniformity of the bed composition, defined in terms of bed porosity (fractional volume of sodium). An experimental verification of this uniformity was felt necessary by reason of the filling procedure of the device, successively with a mixture of fuel particles of different size and with sodium.

A neutron transmission has therefore been carried out by using a collimated beam issued from the BR1 reactor. The 10 m high integrated irradiation device was scanned with detail over the height of the bed, the transmitted beam being measured with a $^3$He counter. The reproducibility of the measurements amounted to 0.2 %; the sensitivity was such that 1 % variation of the porosity around the nominal value of 35 % induced a relative variation of 4 % in the measured signal.

4. The cooperative effort between ENEA-Casaccia and SCK/CEN has been continued with the objective of using TAPIRO as a reference neutron field for reactor dosimetry applications. Schematically, TAPIRO consists of a cylindrical enriched uranium core surrounded with a copper reflector. The main task has been to resolve discrepancies between calculations and measurements related to the fast neutron flux attenuation in the copper reflector. Spherical uranium-copper assemblies located in the one-meter cavity of the graphite thermal column of BR1 were therefore studied experimentally; reaction rate traverse measurements are now being compared with neutron transport calculations, using different cross-section libraries (ENDF, CARNAVAL).

## HIGH FLUX MATERIALS TESTING REACTOR BR2 : UTILIZATION OF LOW ENRICHED URANIUM (LEU) FUEL

Neutronics analyses at ANL and thermohydraulics studies at SCK/CEN are under progress while CERCA and NUKEM are trying to form circular high-density BR2 plates. At present, BR2 is loaded with fuel elements containing 400 g $^{235}U$ (90 % enriched uranium), 3.8 g boron and 1.4 g samarium; the uranium meat density is 1.3 g/cm$^3$. Preliminary fuel cycle analyses have allowed to propose following characteristics for prototype fuel elements to be tested in BR2 :

| | |
|---|---|
| primary constituent | : $U_3Si_2$ |
| total $^{235}U$ content | : 480 g (20 % enriched U) |
| uranium meat density | : 5 g/cm$^3$ |
| burnable poisons | : 0.4 mm diameter Cd wires in the webs or 2.8 g boron in the fuel (with 1.4 g samarium in both cases). |

The water channel thickness would be reduced from 3.0 mm to 2.85 mm.

The irradiation testing of such prototype fuel elements in BR2 is clearly an important step of the reduced enrichment research programme.

## FUSION

A delayed neutron counting system was developed, constructed and calibrated in the frame of the neutron diagnostics programme at JET. With this system, $^{238}U$ and $^{232}Th$ targets are exposed during a pulse in the vicinity of the torus and then transferred in a polythene assembly in order to measure the delayed neutron emission by means of $^3He$ counters. Irradiations of targets in a $^{235}U$ thermal neutron induced fission spectrum were performed at Mol for calibration purpose. From the measurements, it has been possible to derive an experimental value of the total delayed neutron yield for $^{232}Th$ with respect to $^{238}U$; this value is 1.177 $\pm$ 0.022 to be compared with the recommended values of

- Tuttle (Nucl. Sc. and Eng. 56, 37, 1975) : 1.209
- Waldo et al. (Phys. Rev. 23, n° 3, 1981) : 1.255

## NUCLEAR DATA

Spectrum integrated cross-sections for $^{6}Li(n,\alpha)$ and $^{10}B(n,\alpha)$ from five benchmark fast reactor neutron fields were compared with calculated values obtained using ENDF/B-V [7]. Two of these benchmarks are the sigma-sigma and fission cavity fields at Mol. The analyses indicate a need for revision of these cross-sections for energies above 50 keV; in particular, the $^{10}B$ appears to be underestimated in the ENDF/B-V file.

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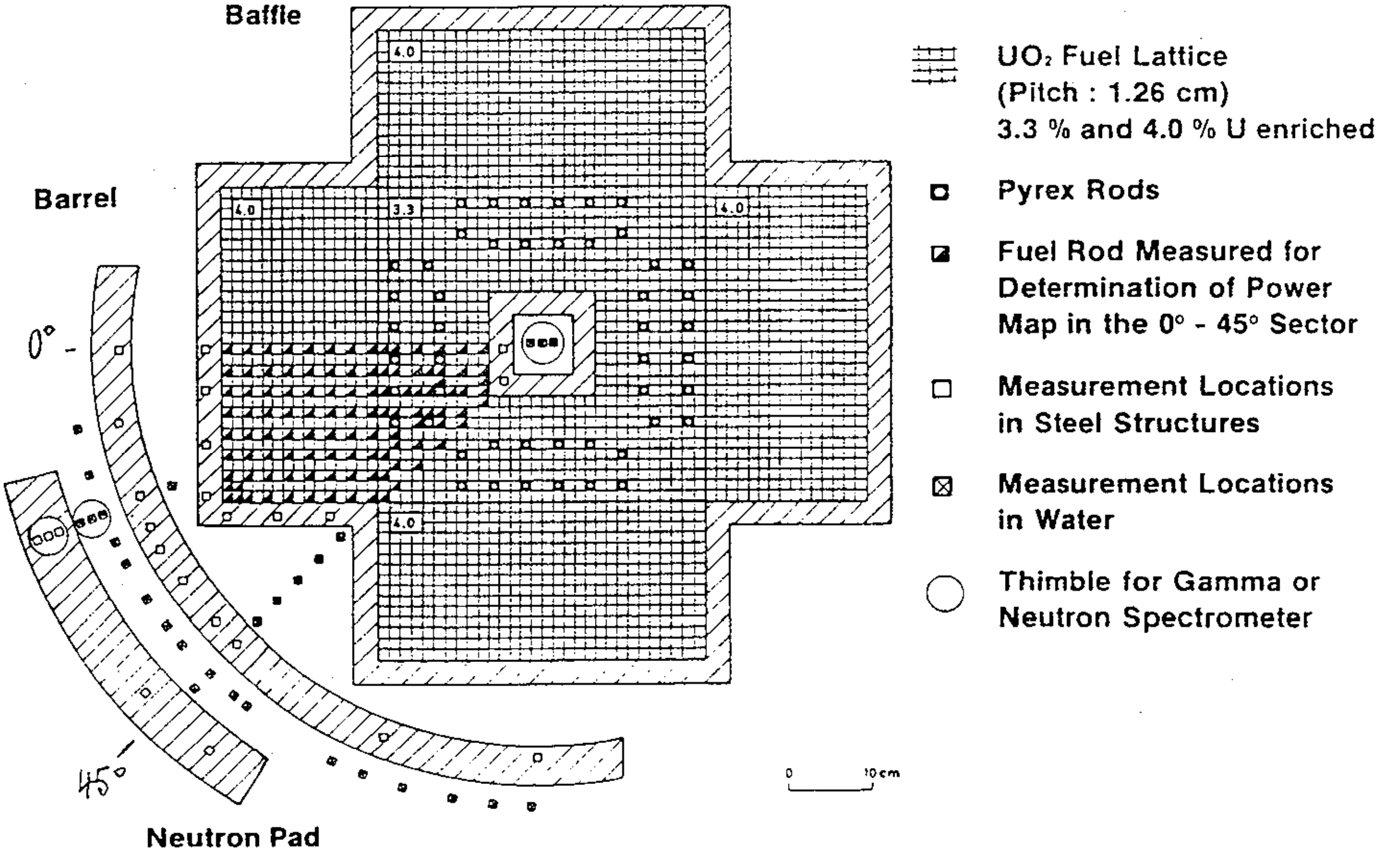
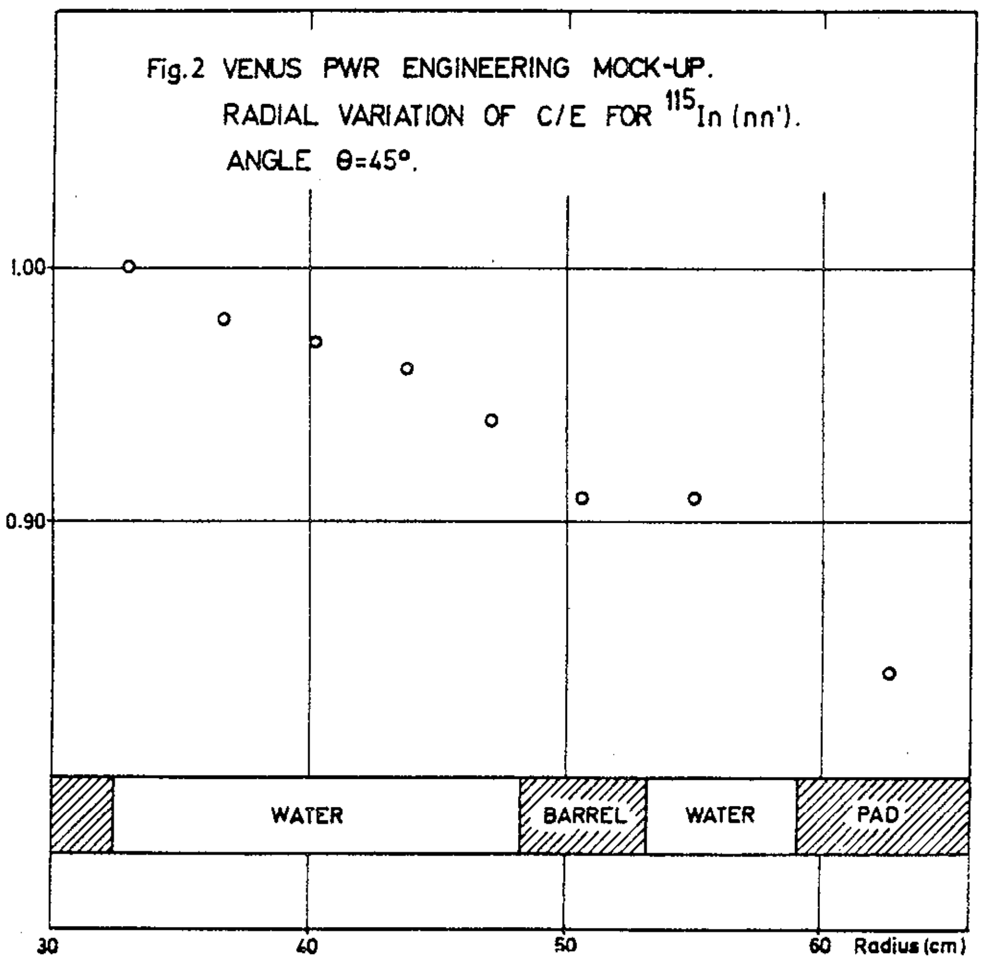



Fig. 1. GEOMETRY OF THE VENUS CORE CONFIGURATION FOR THE REACTOR PRESSURE VESSEL PROGRAM



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# Reactor Physics Activities in Canada

Compiled by Dr. F.N. McDonnell
Head, Reactor Physics Branch
Reactor Development Group
Atomic Energy of Canada Limited

## Introduction

Reactor physics activities in Canada are carried out at several sites and institutions. The majority of the work is in direct support of current CANDU reactors and future advanced CANDU concepts. There is also significant effort carried out in support of a new family of research reactors called MAPLE (Multi-purpose Applied Physics Lattice Experiments) as well as in support of the various types of SLOWPOKE reactors.

This report summarizes the activities carried out by the various organizations during the period from October 1985 to September 1986.

## 1. Utilities

There are three utilities in Canada which operate CANDU reactors to produce electricity: Ontario Hydro, Hydro Québec and the New Brunswick Power Commission.

### 1.1 Ontario Hydro - Toronto, Ontario

#### 1.1.1 Modelling of the Poison Injection Shutdown System

CANDU reactors have two independent shutdown systems. The first consists of solid absorber "shutoff" rods which are dropped into the core to effect rapid shutdown. The second consists of several perforated tubes inserted horizontally into the core connected to external tanks containing gadolinium nitrate neutron-absorber dissolved in heavy water. To shut down the reactor, fast acting valves connecting the poison tanks to a tank containing helium under pressure open, forcing the poison into the tubes and out through the holes into the heavy water moderator in the core.

Tests have demonstrated that this system is effective. However, efforts are in progress to develop a model which can be used to simulate the power run-down during a loss-of-coolant-accident for use in licensing, accident analyses, and operations support analyses in which the hypothesized conditions would not precisely duplicate those of the available tests. The aim is to improve existing modelling techniques in two areas. Ontario Hydro is developing hydraulics models which describe the transient injection of fluid from the tank into the core, and its subsequent dispersion into the bulk moderator. Physics models to describe the reactivity effect of the dispersing poison curtain are also being developed. A spatial modal kinetics code is used to compute the power transient from the reactivity as a function of time.

1.1.2 Nodal Coupling Coefficients for Training Simulators

Ontario Hydro's Pickering A full scope training simulator uses a nodal kinetics model based on the Avery formulation to treat the core spatial neutronics. Canadian Aviation Electronics (CAE) is using a similar technique in the training simulators for light water reactors which they are marketing. Ontario Hydro is providing the coupling coefficients for the nodal spatial kinetics models to be used in these simulators.

1.1.3 Commissioning Tests and Code Validations

Several Ontario Hydro reactors at both Pickering and Bruce have been started up recently. Low power physics measurements were made as part of the commissioning tests. These included critical poison concentration, reactivity device worths, flux distributions, and power rundown transients. These experiments have been simulated with Ontario Hydro's standard reactor physics computer codes. Agreement between simulation and experiments is quite satisfactory.

1.1.4 Pickering NGS-A Adjuster Redesign

Pickering NSG-A is Ontario Hydro's oldest commercial nuclear station. Its four CANDU reactors were placed in service in the early seventies. The original design had 18 adjuster (control) rods and 11 shutoff rods. Recently, it was decided to increase the reactivity depth of the shutdown system by replacing some of the adjuster rods with shutoff rods.

Ontario Hydro staff have investigated several different layouts in order to maximize the shutoff rod depth while minimizing the reduction in adjuster system effectiveness. The final design consisted of 21 shutoff rods and eight adjuster rods. This was accomplished by replacing ten adjusters with shutoff rods modified to fit the adjuster rod calandria penetrations. The remaining eight adjuster rods are to contain increased amounts of cobalt. These changes are being implemented in Units 1 and 2, which are temporarily shut down for retubing.

1.2 Hydro-Québec - Montréal, Québec

Hydro-Québec's activities in reactor physics are performed in support of the operation of the Gentilly 2 nuclear power station, a CANDU-600 that began commercial service in 1983.

On-power refuelling of the station is guided by weekly updating of calculated fuel burnups and power distributions with a flux-mapping computer program called SIMEX[1]. This code, developed at Hydro-Québec, uses time-integrated signals from in-core flux detectors in a power mapping procedure which essentially performs a least-squares fit of these signals using a set of precalculated flux distributions. Some of these distributions are the natural modes, others are perturbation modes. The originality of the method lies in the fact that the fundamental mode is recalculated weekly and accounts for most of the local effects of the flux distribution resulting from on-power refuelling. Changes in the power distribution over larger regions of the core are then given by the detector signals and the mapping procedure.

To assist the fuelling engineer in the weekly selection of fuelling sites, Hydro-Québec has developed the computer program called SRG2[2]. This code applies the various rules and guide-lines developed from past experience and based on a power shaping approach, to recommend a list of fuelling sites. The code simulates these re-fuellings with standard two-dimensional diffusion calculations and gives a prediction of their effect on reactivity and power distribution over the next few weeks. Not all sites selected by SRG2 are actually used, because the code's selection algorithm does not account for all of the factors that the fuelling engineer must consider. In particular, the code cannot simultaneously optimize the power distribution and fuel burnup. More details are given in Reference 2.

## 2. University Activity

Courses and research in reactor physics are available at several Canadian universities: University of Toronto, University of New Brunswick, Ecole Polytechnique and McMaster University.

### 2.1 Group D'Analyse Nucléaire (GAN), Ecole Polytechnique, Montréal, Québec

#### 2.1.1 Activities Related to Diffusion Theory and Applied Mathematics

A new three dimensional neutron diffusion code named TRIVAC was set up using advanced discretization algorithms and improved iteration strategies[3]. This software is characterized by a variable order discretization algorithm based on the variational or nodal collocation technique[4] and is adapted to treat CANDU or PWR reactors models. TRIVAC also has the capability to solve a direct or adjoint fixed source eigenvalue problem such as those appearing in generalized perturbation theory.

A natural extension of this type of problem is the spatial function evaluation in the quasi-static method of space-time kinetics, which is being incorporated in TRIVAC. Incorporation of a higher harmonics calculation will follow.

A new interface with TRIVAC was recently written for the OPTEX code, which will provide a 3-D capability to the optimization procedure. OPTEX uses generalized perturbation theory to evaluate the gradients of system characteristics which depend on the flux (or adjoint flux) distribution. Mathematical programming techniques are then applied to solve fuel management or core design optimization problems, while achieving the desired power shaping in the reactor[5, 6].

2.1.2 Activities Related to Transport Theory and Lattice Cell Calculations

Work in transport theory was directed toward the resolution of the two- and three-dimensional transport equations using collision probabilities and interface current methods. A two-dimensional transport code for cell calculation, compatible with WIMS, was developed using the interface current method to evaluate the collision probabilities needed for the computation of the neutron flux inside a typical CANDU rod cluster [7]. Other applications include improved techniques for computing the three-dimensional collision probabilities inside supercells for the modelling of reactivity devices in a reactor[8]. Finally, some work on resonance self-shielding calculations for fuel in exact geometry using the interface current method was also undertaken.

2.2 McMaster University - Hamilton, Ontario

2.2.1 Nuclear Energy Synergetics

The subject of interest here is the analysis of integrated and interactive nuclear energy systems. Among these are fusion-fission hybrids, spallation breeders, accelerator driven fusion, catalytic systems and others. Our emphasis is based on the underlying reaction linkages and energy balance characterization in order to obtain system indices of merit and of performance.

2.2.2 Nonlinear Reactor Dynamics

The fundamental starting point here is the concurrent retention of the time-dependence of all dominant particle densities: neutrons, fuel atoms and absorbers for fission reactors and all fuel ions and chain carrier ions for fusion systems. The lowest order description leads to Ricatti-type nonlinear descriptions and a range of solutions governed by the reactor parameters. Complex "chaotic" power variations have been identified and the dominant nuclear-thermal driving-response phenomena identified. Both the theoretical characterizations and the practical consequences for reactor design and operations are of interest.

3. Atomic Energy of Canada Limited

A broad range of reactor physics and related design and research and development activities are carried out at AECL.

3.1 CANDU Operations - Mississauga, Ontario

The design activities of AECL are performed at CANDU Operations in Mississauga near Toronto, Ontario. Some of the recent work is described below.

3.1.1 CANDU Reactor Power Uprating

Much work has been done in the past year to increase the power output of the CANDU 600 reactor. Power uprating has been considered both for existing and for future CANDU 600 mW reactors.

Contributions to the uprating from the thermalhydraulics area include an increase in coolant flow and recent results in the improving and understanding of dryout.

From the reactor physics point of view, an increase of 10% in power output for future plants is envisaged by running the CANLUB 37-element fuel bundle at the power level demonstrated in Bruce A (i.e., a maximum bundle power of 1035 kW). In addition, adopting 4-bundle-shift refuelling in the central core region to reduce power ripple, and slightly increasing the flattening of the core power distribution, can provide some 3% in power uprating, in both existing and future plants. The number of fuel channels in the design for a future CANDU 600 can also be increased by 8 to 388 channels.

The overall result of these studies is that an increase of about 10% in power can be achieved for an existing CANDU 600, while an uprating of about 17% is possible for a future CANDU 600.

### 3.1.2 Transition from a Natural Core to a Slightly Enriched Core

A detailed time-dependent fuel management study has been concluded to demonstrate that it is feasible to convert an operating CANDU reactor from a natural equilibrium core to a slightly enriched core with 1.20 wt.% U-235 without having to derate or shuffle fuel. Approximately 1000 full power days is required to displace all the natural fuel with the slightly enriched fuel. No hardware changes to the reactor or fuel handling system are envisaged.

### 3.1.3 Calculations of 3-D Flux Distributions in CANDU Reactors Using Lattice Properties Dependent on Several Local Parameters

A refinement to the current method of evaluating lattice properties for use in calculations of CANDU flux distributions has been proposed and tested. This refinement has been labelled the local-parameter calculation. The computation of lattice properties at each position takes into account locally appropriate values of coolant density, fuel temperature, and flux/power level where core-average values were previously used.

The main effect of using the local-parameter method is an overall radial and axial flattening of the calculated flux and power distributions.

Peak channel and bundle powers are typically reduced by about 2% and 5% respectively. Tests on a snapshot of a reactor history show that the local-parameter calculation improved the comparison of fuel-management program results with those of power-mapping and thermalhydraulics-based programs.

### 3.2 Whiteshell Nuclear Research Establishment - Pinawa, Manitoba

The Systems Analysis Branch at WNRE includes three groups which provide a wide range of Physics support to various AECL activities. These groups are: the MAPLE Research Reactor Technologies group, the Small Reactor Concepts group and the Shielding and Criticality group. The current scope of work for each group is described below.

#### 3.2.1 MAPLE Research Reactor Technologies Group

The purpose of this group is to provide reactor physics support for the MAPLE Research Reactor concept and for the evaluation of customer-specified variations. This has been accomplished with extensive use of the codes WIMS-CRNL, 3DDT and MCNP plus a large number of utility codes written in-house to permit effective interaction of the major codes.

The efforts of the group are currently focused on providing reactor physics support for the study of a multi-purpose research reactor concept. The work has included validation of WIMS-CRNL with both the 69-group WINFRITH library and the 89-group ENDF/B-V library, an evaluation of the various calculational options available in WIMS-CRNL (leakage, diffusion coefficients, spectrum), fuel management studies and determination of various reactivity coefficients and worths.

In addition to performing various reactor physics studies, as above, the group acquires and develops codes and modelling techniques which improve the accuracy and efficiency of its efforts.

Recent developments include:

- implementation of WIMS-CRNL on the WNRE VAX system
- preparation of a code to calculate multi-energy group, multi-half-life group neutron kinetics data from WIMS output
- development of supercell modelling capability using WIMS
- implementation of the CRNL version of 3DDT (a three-dimensional, multigroup reactor lattice code) on the WNRE VAX system
- modification of the implemented 3DDT to produce a significant reduction (approximately a factor of 10) in the time required for convergence of large meshes
- development of a code, to evaluate isotope production scenarios, which includes both thermal and resonance activation.

#### 3.2.2 Small Reactor Concepts Group

The Small Reactor Concepts Group is a new group formed to investigate th core neutronic behaviour and technical feasibility of innovative small reacto concepts for electricity production. The safety, reliability and cost of pro posed concepts are assessed using the WIMS neutron transport code, the 3DDT three-dimensional, neutron diffusion theory code and the MCNP Monte Carlo code.

The current main focus of the group is the Compact Nuclear Power Source, which will be capable of producing 20 kW(e) for 20 years. Larger versions, referred to as Nuclear Batteries, have been considered for use in remote villages or on submarines. The group has also assessed alternative materials and concepts for local (village) electricity production or space nuclear power applications.

### 3.2.3 Shielding and Criticality Safety Group

The Shielding and Criticality Safety Group is mainly involved in shielding analyses for small research and special purpose reactors. A detailed study of neutron and photon streaming through a concrete maze was also conducted for an electron accelerator using Monte Carlo techniques. Criticality safety and shielding assessments have also been performed in support of the concrete-canister-fuel-storage development program. The group contributed to a storage cask shielding benchmark program, sponsored by the NEA, with a detailed Monte Carlo analysis of a typical spent fuel shipping container.

The group acquired the full-geometry-continuous-energy Monte Carlo code MCNP and has made extensive use of its capabilities both in detailed analyses and also in comparison studies with other codes. The group is also working on a method of accurately coupling a Monte Carlo code core model to a one- or two-dimensional discrete ordinates code shield model to handle situations in which the reactor does not contain sufficient symmetry to allow accurate modelling in cylindrical or spherical geometry.

## 3.3 Chalk River Nuclear Laboratories - Chalk River, Ontario

The work performed in the Reactor Physics Branch at Chalk River is organized in four major areas:

i) Experimental activity
ii) Reactor assessment and design
iii) Code development and nuclear data, and
iv) Research reactor support.

The current scope of work for each group is described below. In addition a summary of a modest effort on fusion is also provided.

### 3.3.1 Experimental Activity

The reactor physics work of the experimental section centres around the study of potential advanced fuels for the CANDU reactor. The studies are carried out in the ZED-2 zero energy critical facility. In general not enough of the advanced experimental fuel is available to set up uniform cores, therefore methods in which a few rods (usually seven) of the experimental fuel are substituted in the central region of a natural uranium core are used. The reactivity effect of the substitution is measured by the change in critical size of the core; this may be interpreted to give the material buckling of the substituted fuel. In addition many reaction rates of interest are measured by foil activation in the fuel elements and moderator of the central substituted cell. These include fission rates and capture rates (in fertile materials) as well as several flux and spectrum indicators such as Cu-63, Au-196 and Lu-176.

In the past year a program of such experiments with $(Pu,Th)O_2$ fuel has been completed. The fuel contained 1.8% fissile Pu in heavy elements. Seven rods each containing five 36 element bundles of the fuel were available. Measurements were made in two hexagonal arrays of pitches 31 cm and 24.5 using three different coolants: $D_2O$, air and $H_2O$. In addition the effect, both on reactivity and on reaction rates, of changing the temperature of the fuel and water coolants from about 20°C to 300°C was measured. Analysis and reporting of the results is continuing.

A major series of experiments on $(U\text{-}233,Th)O_2$ fuel will commence in 1987.

Two papers on some neutron cross section measurements were published [9,10].

### 3.3.2 Reactor Assessment and Design

#### 3.3.2.1 Small Reactors

Methods of increasing the core life of the small, low power (20 kW) SLOWPOKE-2 reactors are being studied.

The physics design calculations for the prototype 2 MWt SLOWPOKE heating reactor at the Whiteshell Nuclear Research Establishment were completed. Commissioning of the reactor is scheduled to begin by the end of 1986.

Reactor physics calculations for a conceptual design of a higher power SLOWPOKE heating reactor are in progress.

The NEACRP Benchmark BWR lattice Cell Problem[11] was used to test the use of the SHETAN code for calculations of large absorbers in light water reactors. SHETAN results[12] are comparable with those that had been obtained with other codes[11].

#### 3.3.2.2 Advanced CANDU Reactor

In the previous year, a configuration of reactivity devices had been proposed for future CANDU reactors which improved the physics characteristics of enriched fuels in CANDU, while allowing the use of natural uranium. This work[13] continued with the identification of further improvements in the locations of devices. Time-dependent fuel management calculations for a variety of fuel types and enrichments, and time-dependent xenon studies were performed. For slightly enriched uranium (SEU) with a burnup of 21 MWd/kg, maximum bundle and channel powers were about 8% and 3% lower respectively, than for natural uranium in the current design (normalized to the same reactor power).

### 3.3.3 Code Development and Nuclear Data

#### 3.3.3.1 Nuclear Data

Nuclear data activities have been concerned primarily with the production of data libraries for lattice codes based on ENDF/B-V. Eighty-nine group libraries are now available for RAHAB and WIMS-CRNL. Thermal data suitable for the production of Westcott parameters have been produced for 274 materials[14]. A means of producing 89-group libraries for the Monte Carlo code MCNP based on material cross sections calculated by WIMS-CRNL has been developed. These libraries allow most neutron transport problems to be solved to a sufficient level of accuracy on relatively small computers. Nuclear data, based on ENDF/B-V, for In-115 and Lu-176 have been tested by comparing calculated and experimental In/Mn and Lu/Mn activation ratios for a few selected ZED-2 experiments[15]. The calculated and measured results for Lu are in reasonable agreement. Quite large and so far unexplicable differences exist between the In results.

Work has continued to develop a reliable method for extracting material bucklings from substitution experiments see section 3.3.1). Encouraging preliminary results have been obtained.

#### 3.3.3.2 Codes

The Monte Carlo codes REPC and MCNP have been made operational on the Chalk River computers. Methods of incorporating the resonance reaction rates from REPC into both RAHAB and WIMS-CRNL have been completed.

Most development work has been associated with WIMS-CRNL. The major change in this period has been an extension to the burnup calculation and associated library data to allow (n, 2n) reactions and branching ratios. The modifications were designed and implemented in a way that generalized the data representation and processing, so that future extensions and changes will be straightforward.

### 3.3.4 Research Reactor Support

The large NRX and NRU research reactors continued to operate, although NRX, operating since 1946, was reduced to a "hot-standby" mode, and will soon be decommissioned as part of a cost-cutting effort. NRU was shutdown for six months to repair a leaking re-entrant can, but was reloaded and recommissioned without incident. Routine calculations for NRU included flux and power distributions and fuel management. A new 3-D simulation code was developed using discontinuity factors, which gives improved results ($\pm 5\%$) for this highly heterogeneous reactor[14]. Apart from CANDU fuel and materials development, and radioisotope production, reactor physics support was provided for two commercial LOCA tests which produced fuel fragmentation and some melting. A study on the conversion of NRU to LEU was also completed[17], and the design of a high fast flux rod, intended to give a flux 1 MeV of about $2 \times 10^4$ n.cm$^2$.s, was assessed.

A major effort was provided towards the design of the new MAPLE-X reactor, a light water pool-type reactor with a $D_2O$ reflector, to be used at CRNL to meet radio-isotope requirements[16]. This reactor will be built in 1988, and will replace some of the capabilities of NRX.

Other effort was directed to commercial accelerator shielding design.

### 3.3.5 Fusion

As part of the Canadian Fusion Fuels Technology program, various branches at Chalk River are participating in studies on solid breeder blanket materials. A vented irradiation facility has been designed and tested in the NRU reactor. Results from breeder ceramic irradiations will identify the factors and mechanisms that control tritium migration and provide in-reactor measurements of heat transfer properties. Temperatures in the facility can be controlled and a temperature range of 400 - 900°C is achievable. An irradiation of $Li_2O$ is planned for 1986 November.

A comparison of the neutronics environment in fusion blankets and NRU irradiation facilities is ongoing. Of particular interest is the relative contribution to material damage from atomic displacements, helium production and nuclear transmutations.

Calculations have shown that commercial quantities of tritium can be produced economically in operating CANDU reactors. The tritium is produced by irradiation of lithium in dedicated channels previously occupied by fuel bundles. The production cost of tritium in CANDU reactors is less than the current market price of tritium.

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# RECENT REACTOR PHYSICS ACTIVITIES IN DENMARK

by Erik Nonbøl
RISØ - Department of Energy Technology

## 1. The LEWARD programme

Most of the physics progammes at Risø have been developed during a period where the preferred computer language here was ALGOL. Although a powerful language it has suffered from the drawback that very few other institutions or utilities used it. It was therefore decided to translate some of our basic RP programmes to FORTRAN77, and, a new programme has emerged, called LEWARD, which makes neutron and gamma physics calculations for LWR fuel assemblies.

Its main virtues compared to the old programme system are that it is very easy to use, and that it is written in F77, which is a widely used computer language.

The ease of input preparation, on the other hand, has made the new programme much less suited for "special purpose"-calculations than the old system, for which almost any complicated problem can be specified.

Reference is made to the enclosed report "LEWARD - A Short Description of a New LWR Assembly Neutronic Code" by C.F. Højerup

## 2. Analysis of the relationship between measured and calculated detector response for a BWR reactor

In 3-dimensional coarse-mesh solutions for power reactors there is no access to the detailed local flux at the detector position. One has to deduce the detector flux from average values of various reactor physical parameters of the surrounding fuel assemblies.

Three different calculation methods have been analvsed (Nonbøl, 1985).

1. Power Method
2. Normalization Method
3. Modulation Method

The power method simply assigns the average power of the 4 neighbour assemblies to the detector response.

The normalization method weights equally the fission rate at the detector position obtained from 4 separate 2-dimensional single-assembly calculations.

Finally, the modulation method is made by multiplying the results from the normalization method with a function taking the overall power distribution into account. This function is calculated from the average power of the 9 nearest fuel assemblies applying bivariate interpolation.

All 3 methods are compared with a detailed fine mesh calculation, and finally tested against TIP-measurements on the Brunsbüttel reactor.

The conclusion from the analyses is that none of the methods seem to give correct results. The modulation method appears to be the best, but it fails for assemblies with control rods.

One way to improve the treatment of detector response could be to represent the boundary conditions of each assembly in a better way. If the net current at each of the four boundaries of the assembly had been available from the coarse mesh calculation it would have been possible to calculate the pin power distribution inside the assembly analytically and thus also the detector response.

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NONBØL, E. (1985). Analysis of the Relationship between Measured Detector Response and Calculated Detector Response for BWR Reactors. RP-15-85.

## 3. Reactor Physics Analysis of Three Different LEU Conversion Scenarios for the Danish Research Reactor DR3

The Danish Research Reactor DR3 may have to convert its fuel from highly enriched uranium (HEU-93%) to low-enriched uranium (LEU-20%). Converting to 20% LEU fuel and sustaining the thermal power level at 10 MW leads to a 15% reduction in the thermal neutron flux. This is caused by the increased amount of $U^{235}$ necessary to compensate for the absorption in $U^{238}$.

The reduction in thermal flux is very unsatisfactory for the users of DR3 and therefore 3 different conversion scenarios have been analysed with the simulator DR3/SIM [1,2]:

1) 10 MW LEU operation
2) 10 MW LEU-RINGCORE operation
3) 12 MW LEU operation

All 3 cases have been compared with the 10 MW HEU operation as it is to-day.

Case 1, 10 MW LEU, means 20% enriched $U_3Si_2$ fuel at 10 MW. Fig. 1 shows calculations made with the 3-dimensional simulator DR3/SIM for the thermal flux in a horizontal plane through the centre of the reactor. A comparison with 10 MW HEU operation shows a decrease in the thermal flux of about 11%.

Case 2, 10 MW LEU RINGCORE, means 20% enriched $U_3Si_2$ fuel at 10 MW where 4 central fuel elements of the total 26 elements are replaced by 4 dummy elements containing heavy water. The uranium content of the remaining elements is increased by an amount corresponding to the removed uranium. The purpose of the design is to obtain an increased thermal flux in the center of the core and thus compensate for the flux reduction caused by conversion to low enriched fuel. Figure 2 shows the results for the calculation of the thermal flux compared with the old 10 MW HEU design. It is quite clear how the thermal flux in the centre of the core is increased but it happens at the expence of a corresponding decrease in thermal flux at the boundary of the core.

Finally, case 3, 13 MW LEU, means 20% enriched $U_3Si_2$ fuel at 12 MW. Figure 3 shows the results for the thermal flux compared with the old design. There seems to be good agreement for the thermal flux between the 10 MW HEU case and the 12 MW LEU case.

After having considered the 10 MW LEU-RINGCORE as an attractive alternative the board of directors at Risø resently have decided upon the 12 MW LEU case. It fulfils all the demands from the users, but it is also the most expensive solution.


[1] NONBØL, E. (1986). Flux Calculations for 3 different LEU Conversion Scenarios made with the DR3-Simulator DR3/SIM. RP-6-86.

[2] NONBØL, E. (1986). Development of a Three-Dimensional Simulator of a DIDO Type Reactor. RP-8-86.


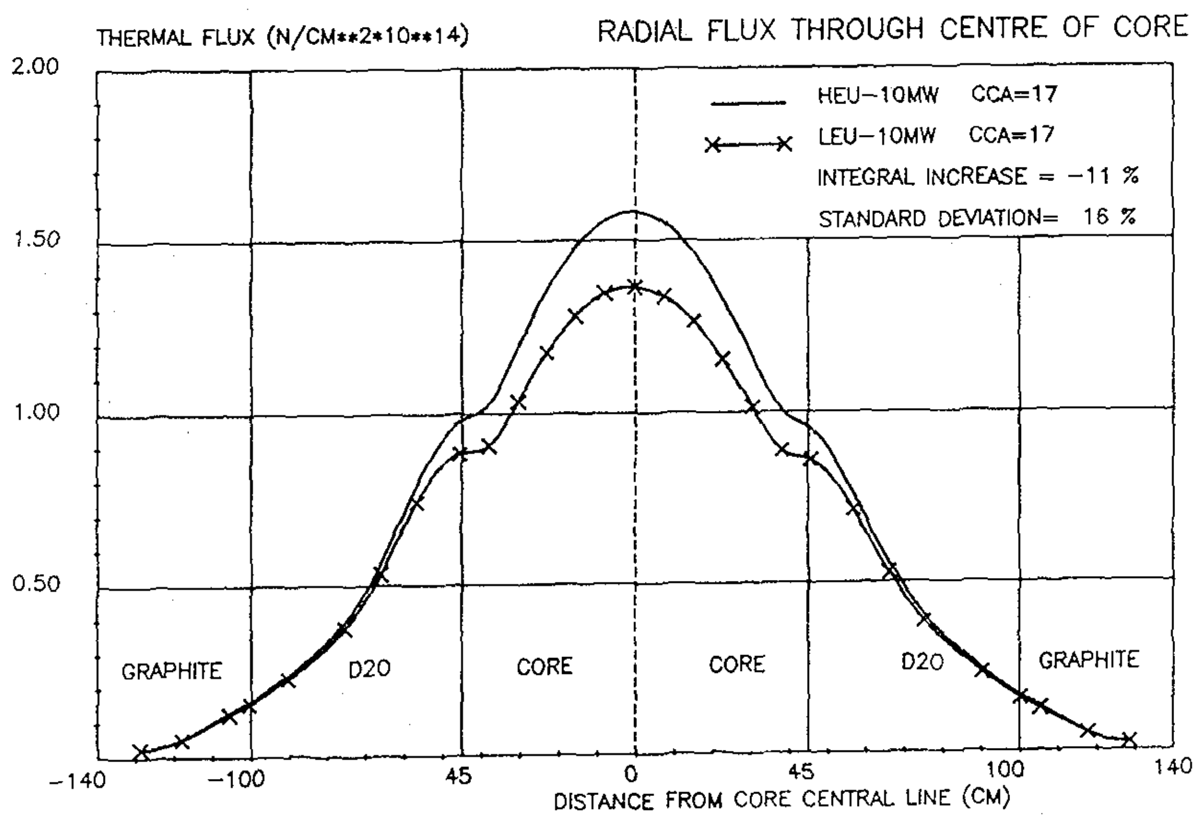


Fig.1 Radial thermal flux for LEU-10MW design compared with HEU-10MW design.

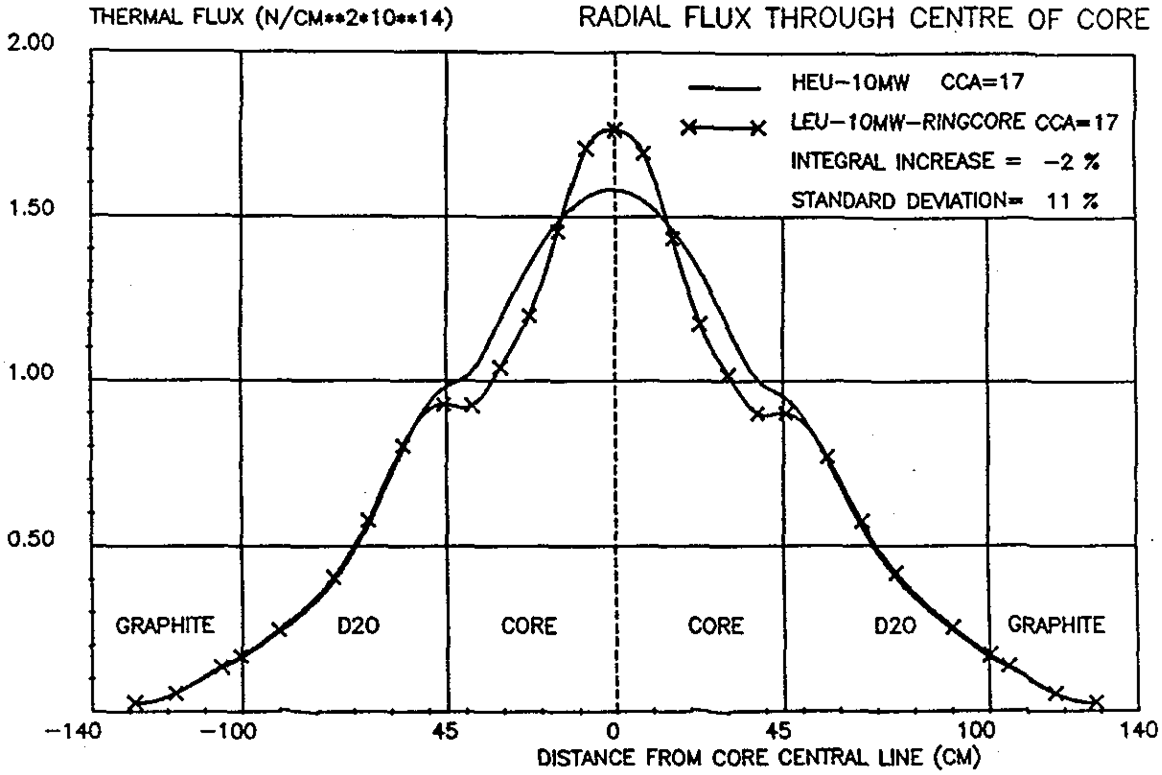


Fig.2 Radial thermal flux for LEU-10MW-RINGCORE design compared with HEU-10MW design.

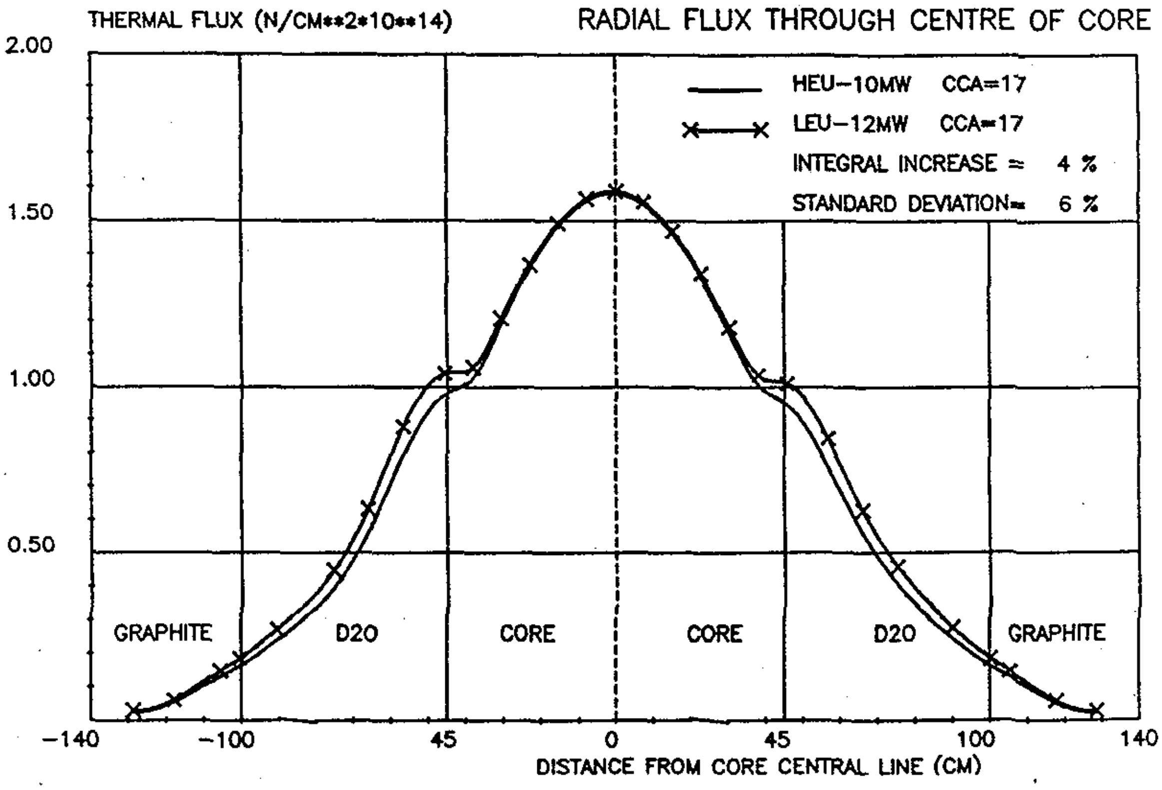


Fig.3 Radial thermal flux for LEU-12MW design compared with HEU-12MW design.

# REACTOR PHYSICS IN FINLAND

## STATUS REPORT TO THE NEACRP 1986

Compiled by M. Rajamäki
Technical Research Centre of Finland

## 1. GENERAL

The very good performance of the four nuclear power plants in Finland has continued in 1986. The major part of reactor physics at the Technical Research Centre of Finland (VTT) has, as before, been directed to serve the reliable operation of the reactors. Research and development work has aimed at verification of methods and codes, at their improvements and at acquisition of the capability to analyse the possible fifth reactor, an appropriate topic before the Chernobyl accident. Since the accident, the particular attention has been given to RBMK-reactors and their possible transient scenarios.

## 2. TRANSPORT CASK SHIELDING CALCULATIONS

Calculations were carried out for the NEACRP Intercomparison of Codes for the Shielding Assessment of Transportation Packages using the REPVICS program system composed in Finland (mostly ANISN and DOT3.5-E). The intercomparison analysis has not been completed yet, but generally speaking it appears that the Finnish results were somewhat lower than most others. It is suspected that this may be attributable to use of the BUGLE-80 cross section library, which was not optimized for this particular kind of calculations.

## 3. CELL CALCULATIONS

The newest version of the CASMO fuel burnup code, called CASMO-3G, was delivered to VTT by Studsvik Energiteknik AB in May 1986. After some test calculations quite an extensive series of production runs was carried out. The main purpose of the CASMO-3G calculations was to generate the gamma detector constants for the fuel types in the present TVO-I core.

A study concerning the so called advanced homogenization methods has been started in 1986. The aim is to introduce these methods into the CASMO-HEX/HEXBU-3D program system, which is used for analyses of the hexagonal VVER cores.

## 4. CORE CALCULATIONS

In 1985, the reloading scheme and the control rod sequences for the TVO-I reactor were for the first time planned entirely in Finland with the CASMO-POLCA code system used by VTT and TVO. The length of the seventh cycle could be predicted almost exactly and the computed value of $k_{eff}$ now remains fairly constant from one cycle to the next, being approximately 0.998 with variations of $\pm$ 130 pcm in the 1985-1986 season.

Experience with advanced fuel types has so far been good. 140 fresh fuel assemblies were loaded in TVO-I in 1986, 88 were of the 9×9 design. Almost a half of the 9×9 assemblies were equipped with natural uranium blankets. More SVEA (QUAD+) fuel will be loaded in TVO-II in 1987.

The possibilities of more efficiently computer-aided planning of reloadings and control rod patterns are being studied.

The work concerning the calculation of the moderator temperature coefficient of the VVER type (Russian PWR) reactors has been continued with purpose to obtain an additional verification and an intercomparison of the methods used in the countries utilizing VVER-reactors.

## 5. DYNAMICS CALCULATIONS

The hydraulic submodels of the PWR dynamics code TAPP has been shown to be insufficient for present needs, particularly in ATWS analyses, and therefore, an advanced thermal hydraulic model is under development. The eventual objective is to replace also the hydraulic models of other dynamics codes.

The collection of the submodels of the BWR dynamics code TRAB has been further enlarged. The newest system being modelled is the hydraulic scram of the Asea-Atom BWRs. Now also the consequences of partial scram can be analyzed.

The three-dimensional reactor dynamics code HEXTRAN is further under development. It is based on the three-dimensional stationary VVER-simulator HEXBU-3D and one-dimensional reactor dynamics code TRAWA including two-phase hydraulics. HEXTRAN is intended to the calculation of VVER-type PWRs. The two-group diffusion equations in HEXBU-3D and TRAWA are of the same form and the additional effects of time derivatives and delayed neutrons can be easily supplemented.

## 6. ACTIVITIES INDUCED BY THE CHERNOBYL ACCIDENT

Since the Chernobyl accident, a substantial amount of work has been devoted to the analysis of the physics of RBMK reactors. So far the emphasis has been mainly on establishing a set of calculation methods appropriate for this reactor type, since no codes intended for RBMK reactors are in use at the Technical Research Centre of Finland.

However, the starting point for our reactor dynamics codes was SPLOSH II, a code made for channel type reactors and its capabilities have been conserved. After some enlargements as modelling of the graphite moderator, the BWR dynamics code TRAB will be capable to computation of the RBMK-transients in which the channel flow does not reverse.

A study on power excursions stimulated by the positive void coefficient is recently going on.

FRANCE

## REACTOR PHYSICS ACTIVITIES IN FRANCE ' BY C. GOLINELLI AND M. SALVATORES, C.E.N. CADARACHE

### 1 - INTRODUCTION

After the CHERNOBYL accident and the European Nuclear Conference of GENEVA, the situation in FRANCE still appears to be relatively unaffected by the events. The slowing down of the orders is produced by the saturation of the domestic electrical requirement and the limit of exportation to foreign neighbours.

The outstanding features, this year, are the remarkable start-up of SUPER PHENIX and the behavior of all French PWRs without trouble. FRANCE is the first nation for the proportion of nuclear power in the electrical production.

| | 1982 | 1983 | 1984 | 1985 |
|---|---|---|---|---|
| Gross Production (TWh) | 266 | 284 | 310 | 329 |
| Nuclear Proportion (%) | 38.7 | 48.3 | 58.8 | 64.8 |

The theoretical limit of 80 % has been almost reached. At the beginning of 1985, the state of the operational PWR units was as follows :

- 1 unit producing 300 MWe,
- 32 units producing 900 MWe,
- 5 units producing 1 300 MWe,

and presently under construction :

- 2 units of 900 MWe,
- 14 units of 1 300 MWe
- 1 units of 1 450 MWe (N4 model).

Two units have been ordered (1 of 1 300 MWe and 1 N4) and options are ready for 2 N4s.

## 2 - FAST REACTOR PHYSICS

### 2.1 - Experimental studies

#### 2.1.1 - SUPER PHENIX

At present (July 1986) 50 % of full power has been attained and full power is expected by the end of the year, with a few months delay with respect to what was announced a year ago. The delay is due to various reasons, not connected to core performances, but essentially to the "prototype" effect of such reactor.

Concerning the neutronic experiments, the majority of them has been completed successfully. Control rod experiments have been carried out extensively. The first results, reported a year ago, indicating a C/E of the order of 1.1, has been confirmed also on the power core, both for the main (SCP) and the complementary (SAC) control rod systems. In the case of SAC, more sophisticated control rod heterogeneity calculation tools have been necessary / 1 / to have consistent results (e.g. SAC worths with different SCP insertions). Radial and axial power distributions have been measured in the core, with different control rod insertion patterns, and in the blanket/shield regions (at higher power). The first results of calculation-to-experiment comparisons, have indicated that simple, design-oriented methods can be used, which are forced to reproduce the observed experimental control rod reactivities, and which reproduce correctly reaction rate shapes, and, then, power distributions in the core.

Concerning the flux distributions in the shield, the results of C/E comparisons show an excellent performance of the shielding formulaire PROPANE, to which bias factors are associated, derived from neutron propagation experiments. An example of the C/E agreement is shown in fig. 1, relative to the Au(n,$\gamma$) radial (at mid-plane) distribution, normalized at the blanket interface. The agreement over approximately 2 meters of radial shield (SS/Na) is of the order of 10 $\div$ 20 %.

Concerning reactivity coefficients, a preliminary estimate of the E/C for the Doppler coefficient is : E/C = 1.06 $\pm$ 20 %.

### 2.1.2 - PHENIX

During this year, it has been completed a consistent analysis and re-analysis of irradiation experiments performed with pure samples (PROFIL-I and II experiments) and with fuel pins of different Pu compositions (TRAPU experiments). The analysis has shown an excellent consistency of the results and also the good performance of the data of the new JEF file for minor actinides / 2 /. Moreover, the analysis has allowed to draw conclusions on several branching ratios / 2 / and data are expected soon also for the case of Pu-236 producing route.

Concerning the definition of a volatile and gazeous fission product migration model, the Cs isotope axial distributions analysis in an irradiated pin, has given the needed complementary informations to validate the model previously proposed / 3 /.

### 2.1.3 - MASURCA

The BALZAC program / 4 / has been re-started after the completion of the neutronic experiments at SUPER PHENIX, which involved most of the personnel of the MASURCA installation. The first series of experiments were related to the study of control rod heterogeneity / 5 /. The need for these experiments is directly connected to some difficulties encountered in the analysis of some of the control rod experiments performed at the SUPER PHENIX start-up. The first results are reported in / 5 /.

In relation to these experiments, the RACINE-1E experiment analysis has been completed, including both natural and enriched boron carbide experiments / 5 /. The observed C/E for control rod worhts is slightly higher than 1 for enriched boron rod, and slightly lower than 1 for natural boron rods. Reaction rate distributies close to rod singularities are reasonable well predicted in both cases / 6 /.

.../...

### 2.1.4 - Future critical experiment programs

In the frame of the European cooperation on Fast Reactor, an agreement in principle has been reached for a common critical experiment program, involving a large quantity of Pu fuel (∿ 2 000 kg) from UK, DEBENE, ITALY and FRANCE, devoted to the study of the physics caracteristics of large LMFBRs. Among the parameters to be studied, there are rod interaction effects and in particular in safety configurations like handling error configurations, and the Neutron flux monitoring system validation in large cores. This program, called CONRAD, should last approximately three years and should be performed starting in 1988.

## 2.2 - Theoretical studies

### 2.2.1 - Design calculation methods

The design oriented code system CCRR has been up-graded with new capabilities in a new full FORTRAN-77 version. Transport based procedures are being introduced for the calculation of special subassemblies (diluent, control rods, etc...) both for neutron and for photon transport calculations.

A series of intercomparison workshops on the European design code systems (UK, DEBENE and FRANCE) has been performed in the frame of the European cooperation for fast reactors, and has resulted in recomandations for the development of a unified code system, with enphasis on portability, flexible user oriented languages and procedures, quality insurance techniques and sofware engineering practices. A final decision for such development is expected early next year.

### 2.2.2 - Cell code

The work for the development of the unified cell (ECCO) code has progressed. Studies have been performed on optimized collision probability algorithms.

.../...

Specific studies have been devoted to the problem of calculating streaming effects taking place between standard fuel assemblies, in Na-floaded or Na-voided situations, both for power reactors and for critical facilities. This type of studies can also be related to safety configurations (redistribution of gaps among subassemblies). Theoretical formulations have been proposed and practical examples have been carried out / 7 /.

Concerning the well known sub-group method for resonance absorption treatement in heterogeneous lattices, a rigorous formulation has been proposed for the statistical domain and applicable to isotope mixtures / 8 /.

2.2.3 - Perturbation methods

The flux reconstruction method based on flux harmonics development, has been extended to the multigroup case, overcoming a long standing difficulty / 9 /. Applications both to LMFBRs and to PWRs have been made. As an example, fig. 2 shows the comparison of the radial flux shape in a PWR, perturbed by the insertion of "grey" rods, as calculated with the harmonics method and directly recalculated. TPG based sensitivity methods, extended to reactivity coefficients (EGPT) / 10 /, have widely being used in connection with design studies and experiment analysis.

2.2.4 - Data assement

A preliminary assessment of the performance of the new JEF-1 files has been completed. The test has been made an a number of critical experiment results, as stored in the Integral Data Bank at Cadarache (BDI, see ref. 11). The quality of the data has been found acceptable for many major isotopes. However, a number of deficiencies have been pointed out and will be the subject of evaluation revision for the future version of the JEF-file. A complementary validation has been made an fission product data / 3 / and on minor actinide data / 2 /. Concerning new data evaluation, Pu-239 data in the resonance region have been revised in a common effort of ORNL and Cadarache / 12 /.

.../...

Efforts are also made in the Pu-241 resonance region data and Pu-239 data in energy ranges outside the resonance region.

A $^3H$ production data experiment for ternary fission of Pu-239 is also underway at the Bordeaux University.

Finally, $(\alpha,n)$ neutron production data in oxide fuel are being revised an the basis of an improved theoretical model and using fresh fuel neutron emission experimental data.

## 3 - LIGHT WATER REACTOR PHYSICS

The past year, covered by this annual report, does not show any spectacular new decisions involving new improvements in the field of Reactor Physics. The progress has been hold to satisfy the previous decisions : plutonium recycling, advanced core management (burnable poison, reloading by fourth of core, extended burn-up), advanced PWRs. Some important results have been achieved.

New facilities have gone into service but in other fields of nuclear activity :

- MEGEVE for the technological studies and the thermalhydraulic behavior of the steam generators.

- BETHSY : a simulation loop for the safety studies linked to the loss of coolant for qualifying the relevant codes (CATHARE for example).

.../...

## 3.1 - Experimental studies

### 3.1.1 - EOLE

The ERASME program / 13 /, which is devoted to the qualification of the undermoderated, mixed oxide, fuelled core comprises three configurations.

1) ERASME-S (Dec. 21$^{st}$ 1984 - Nov. 1985)

The characteristics of the test zone were as follows :

- 1 500 $UO_2$ - $PuO_2$ cells.
- Triangular pitch, moderation ratio : 0.5.
- Total plutonium content : 11 %.

This assembly is inserted into a stainless steel tank. The driver core is constituded by uranium oxide lattices. The interpretation is pratically ended and the first main results have been presented at the SARATOGA SPRINGS Conference (Sept. 86) / 14 /.

2) ERASME-R (Jan. 13$^{th}$ 1986 - Feb. 1987)

The triangular pitch of the test zone is greater. The moderation ratio reaches 0.9. It is comparable to a value of 1.1 obtained with power conditions. The experimental program is similar to the ERASME-S one :

- material buckling,
- spectrum indices (particularly U-238 capture),
- reactivity effects of various absorbants, of water holes and of fertile rods,
- power distribution pertubations near the heterogeneities,
- voidage effect.

Moreover, further measurements will be carried out on temperature coefficients, boron efficiency, effects linked to the isotopic composition of plutonium, the axial blankets... .

3) ERASME-3 is planned for the beginning of 1988. This configuration will become critical without a driver core. The expected moderation ratio will be of 1.8. This is the value of the current PWRs. The experimental program will be also similar but an emphasis is foreseen for the measurements of the absorbant capture efficiency and for the kinetic constants ($\beta_{eff}$).

### 3.1.2 - MINERVE

The running time of the MINERVE facility is shared between the FBR and the PWR programs.

A second campaign for measuring the DOPPLER effect in a fast neutron spectrum have been carried out.

In the field of the LWRs, a set of measurements is performed to qualify the fission capture by oscillating some spent fuel samples irradiated in an undermoderated assembly. This is the MORGANE experiment. The next program will be devoted to the same measurements but in a current PWR spectrum.

### 3.1.3 - MELUSINE

The burnable poison is experimentally studied with the GEDEON program. Its aim is to follow the depletion of the gadolinium isotopes along a cycle. In GEDEON II, the normal situation is represented with interferences between water holes and gadolinium rods, between two gadolinium rods / 16 /. It is a part of the general program concerning gadolinium / 17 /.

GEDEON II began the 15$^{th}$ of May 1985 and was unloaded in January 1986 with a burn-up of 3 500 MWD/t. The first rods have been removed and the isotopic analyses are in progress. A new exposure campaign will start at the end of 1987 in order to reach a burn-up of 12 000 MWD/t.

Within the framework of the RSM program, GEDEON is replaced by ICARE. The goal of this new assembly is the measurements of the capture rates of the main heavy isotopes and of several fission products located in a RSM spectrum.

.../...

The individual isotopes are introduced into $UO_2$ pellets and inserted at the axial position of the special fuel assembly where the RSM conditions are obtained with 263 $UO_2$ - $PuO_2$ rods with a moderation ratio of 0.5.

The capture rates are known through isotopic analysis of the spent pellets. The first irradiation (ICARE 1) began on April $10^{th}$ 1986 and the exposure duration is to last 6 months. The results will be issued after the unloading at the beginning of 1987. A second irradiation (ICARE 2) is planned for 1988 with a greater moderation ratio (0.9).

### 3.1.4 - Spent Fuel Analysis

The numerous French PWRs need an important analysis program. The goal is to verify the depletion calculation required by fuel management of the cores but also the accurate evaluation required by reprocessing plants.

This program consists of :

- $1^{st}$, $2^{nd}$ and $3^{rd}$ cycles from BUGEY.
- $2^{nd}$, $4^{th}$ and $5^{th}$ cycles from FESSENHEIM.
- Assembly with $UO_2$ - $Gd_2O_3$ from GRAVELINES.
- $UO_2$ - $PuO_2$ fuel from CHOOZ.

The end of this program is planned for 1988 but the first results have already been included in our calculational scheme. A paper was issued at the SARATOGA Springs Conference / 18 /.

### 3.1.5 - Instrumentation, Core Surveillance and Simulation

The neutron noise analyses are well known and CEA participated at "the Statement of the Art" published by NEACRP. A special effort is being carried out to control the mechanical vibration of the experimental thimbles by analysing the neutron noise.

.../...

An automatic processing is being studied to detect the running abnormality by reactor noise analysis. Moreover, this type of study is extended in order to diagnose automatically the incidents ~~or~~ the abnormal functionning by processing analysing and in order to determine the optimal regulation. The SALAMANDRE simulator is used for these studies.

A 3D nodal neutron code is being developped in order to compute PWR performances in real time and this is to be used in the process simulation and introduced into the core protection s· ·tem. The method is based on the resolution of the BOLTZMANN equation with enerք ꞡroups and one geometrical mesh per assembly and on an approximation ‹.alled ad Ø' → 0) of a non-variation of the neutron migration area during perturbation or modification of the core.

## 3.2 - Theoretical studies

### 3.2.1 - Transport theory

The last developments for the APOLLO II code include :

(i) A module for the solution of the multigroup transport equations has been written and it is operational for the $P_{ij}$-flux formalism. This module is an outgrowth of the techniques used in the APOLLO code and, besides using a synergetic rebalancing-rest minimization acceleration scheme for the thermal iterations, also includes a new acceleration for the external iterations. The module can solve either a critical or a source problem for the direct or for the adjoint transport equation.

(ii) A general multilevel model has been implemented. This module allows for the solution of the multigroup transport equation using a nested, group-dependent spatial representation in order to minimize calculational costs while preserving the overall accuracy. A technical description, including results for some preliminary tests will be presented next October in Montreal at the upcoming CNS/ANS International Conference on Simulation Methods in Nuclear Engineering.

.../...

(iii) The SN-BISTRO code has been adapted for use as a calculational option in APOLLO II. We are then able to define a correct set of SN transport coefficients from an equivalence ($P_{ij}$ transport to SN transport) calculation that preserves the values of $k_{eff}$ and of the reaction rates.

(iv) The calculation of $P_{ij}$'s for a geometry containing a large number of heterogeneous grains (geometry with a double heterogeneity) has been implemented for three one-dimensional geometries. Implementation for the 2D and multicell geometries is in progress.

(v) A subroutine for the computing of $P_{ij}$'s for a hollow cylinder containing a black, central region has been written. This subroutine will be used in the future to compute collision, escape and transmission probabilities for a "feeding" region in an experimental reactor set-up, thus, allowing us to compute once and for all the "response" of the feeding region under the form of a set of multigroup data that will then be used to calculate each of the possible central configuration.

### 3.2.2 - Diffusion theory

The PHAETON project aims to calculate a whole core with the thermal-hydraulic counterreactions with a very fast speed. It is a dedicated computer comprising microprocessors in parallel. The prototype ran with 4 16-bit microprocessors. Best performances are expected when 32-bit microprocessors will be available on the market. The software is the same as the CRONOS code : resolution of the diffusion equation by finite element method, two energy-groups coupled with thermalhydraulic calculation (FLICA). The counter-reactive libraries have already been qualified by analysing Xenon transients, control rod motions...

.../...

### 3.2.3 - Perturbation theory

The neutronic studies have been essentially performed in the perturbation technique field :

- to test and qualify the existing methodology,
- to improve the formalism taking into account all the results of the first step.

Studies concerning power reconstruction in a PWR have been performed using the versatile system GOLEM based on a perturbation technique called "integral conditionned method". The one-dimensional results obtained show the capability of the method for grey control rod insertion (2 % of antireactivity) and for black rod one (3 %). The discrepancies on the reactivity variation are only 100 pcm for the black rods and 20 pcm for the grey rods.

The power reconstruction is satisfactory and leads to discrepancies of less than 2 % for the hot points. The accuracy of the 1D results has led us to perform 2DXY inserted rod calculations. The major result of this step is that the localisation and the migration of the hot point after a perturbation are well calculated.

The axial (RZ) aspect has then been examined. As far as the reactivity effects and axial shapes are concerned, the results are very accurate and we therefore define a multi-channel modelisation for several control rod rings moving together.

The 3D synthesis of the XY and RZ results is performed by the TRACASYN code using basic and trial functions coming from a multi-pertubational step. The reconstructed values for the power are obtained by a new version of the VAREC code which includes a reconstruction using a graphic method (starting from the integral conditioned one) along physical axes (and not along conventional axes RZ or XY).

.../...

The results are interesting and we note that with some improvements concerning, first, the definition of the perturbations in terms of cross-sections and, second, the software of GOLEM, this code system is able to perform neutron on-line reactor calculation.

GOLEM has been used to perform burn-up 2D assembly calculations. The most difficult case was to perform a 45,000 MWD/t calculation starting from a step 0 (direct and adjoint functions). The noted discrepancies are 0.2 % for a reactivity loss of 17 % and less than 2 % as far as the power shape in concerned. These results load us to consider the capability of GOLEM to perform, with the perturbation method, on-line neutronic calculations coupled with burn-up evolution.

3.2.4 - Residual heat and source term

The independant fission yields were computed using ENDF/BV cumulative yields and new parameter values deduced from recent WAHL work. A code was developped to have these yields and their uncertainties in the ENDF/BV format.

The work to improve β and γ residual heat at short cooling time, using the statiscal model was completed. Good results have been obtained for pulse fission of U-235 and Pu-239 for a cooling time of less than 100 seconds. Improvements are still needed for cooling of terme a between 100 and 1 000 seconds.

Work to improve the decay data part of JEF1 Data Bank is in progress.

3.2.5 - JEF activities

CEA is participating at the general JEF activities in the fields of :

- Data processing code. THEMIS is now available at the NEA Data Bank.
- Benchmarking for the Thermal Reactors. A Paper has been issued at the SARATOGA SPRINGS Conference by TELLIER.
- Evaluations for the next version (JEF2).

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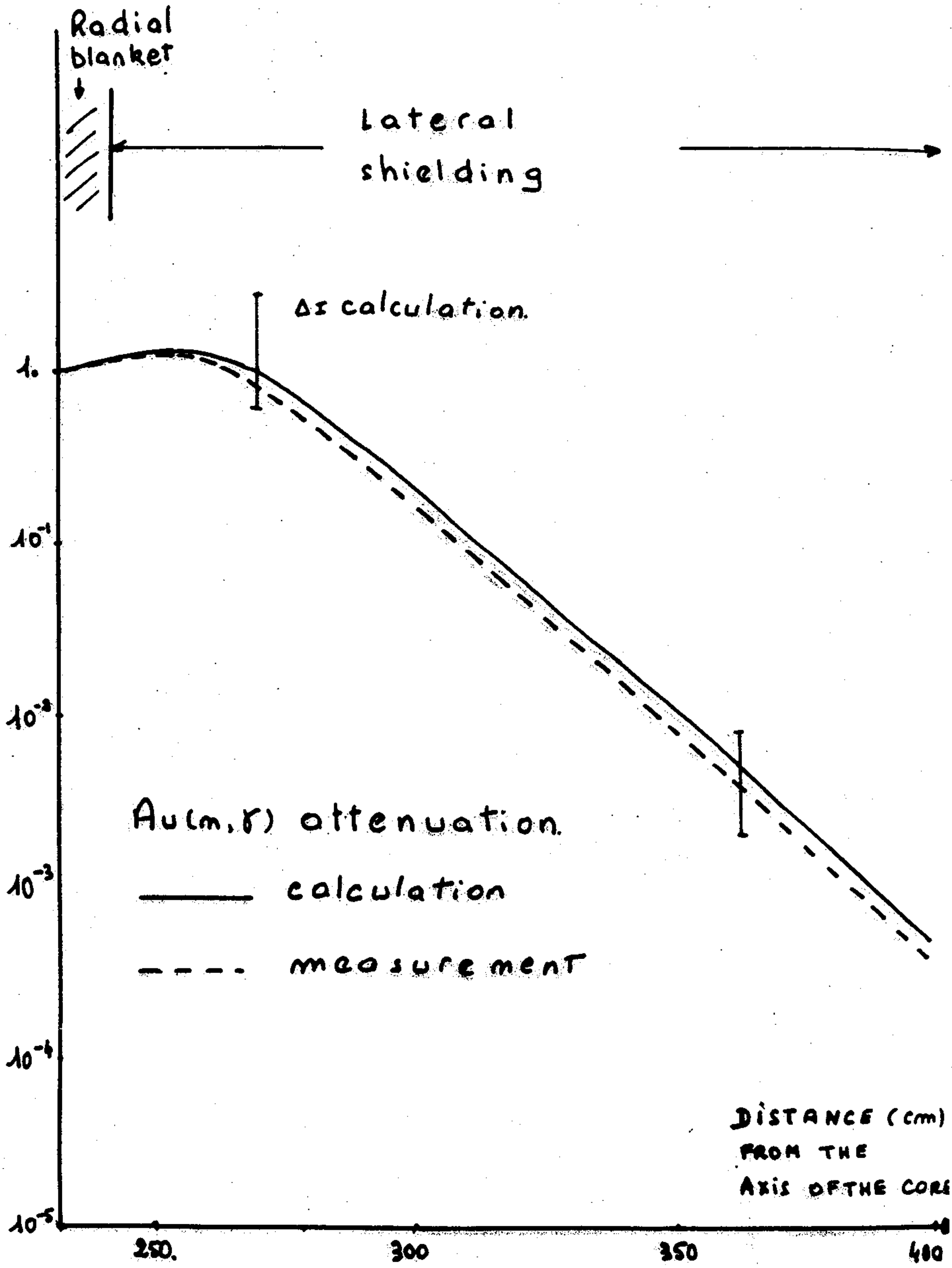
Radial blanket
lateral shielding
Δx calculation
1.
$10^{-1}$
$10^{-2}$
$10^{-3}$
$10^{-4}$
$10^{-5}$
Au(n,γ) attenuation
calculation
measurement
DISTANCE (cm) FROM THE AXIS OF THE CORE
250.
300
350
400

Fig. 1

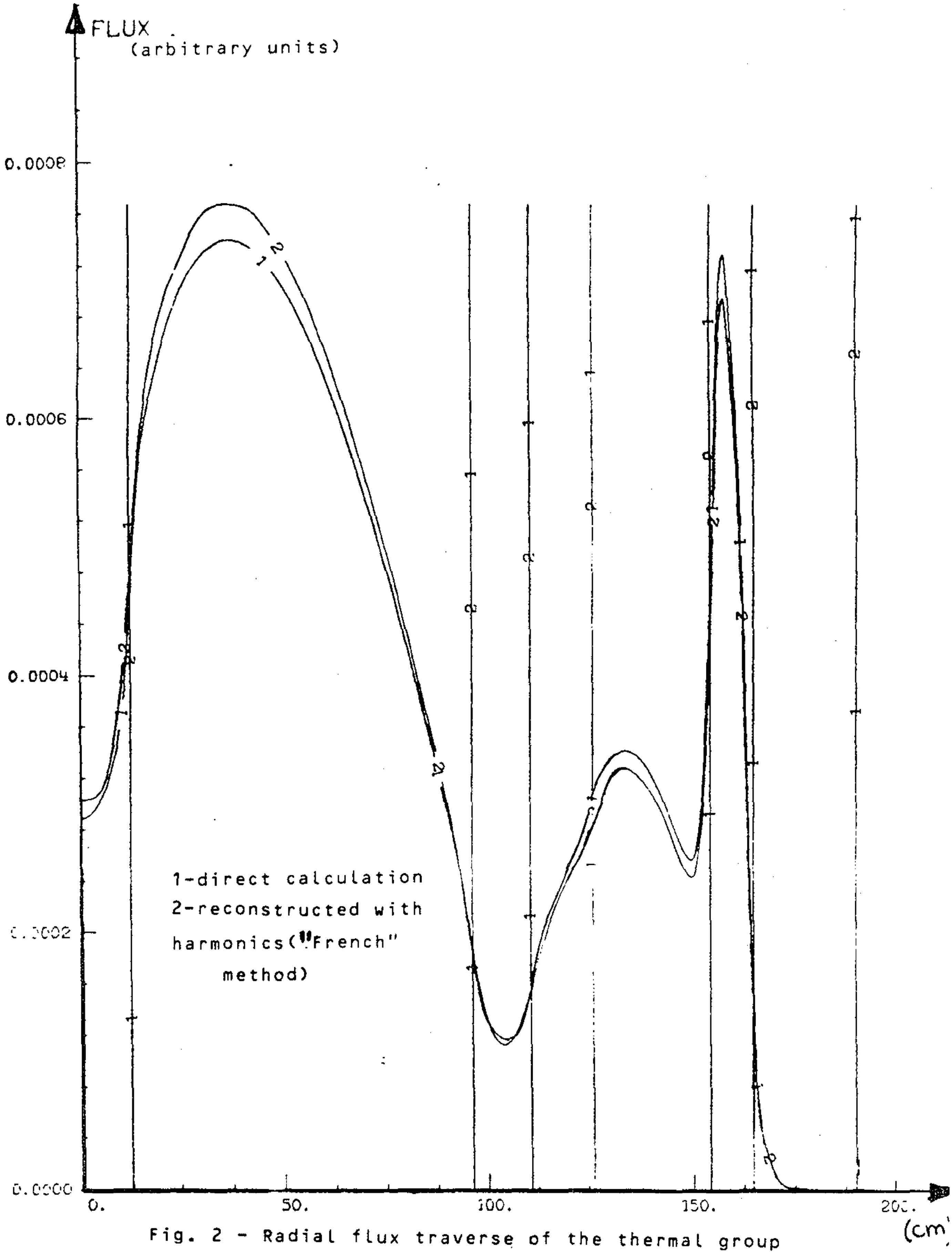


Fig. 2 - Radial flux traverse of the thermal group (in a two group schematisation) of a PWR with a black absorber inserted .

# REACTOR PHYSICS ACTIVITIES IN THE FEDERAL REPUBLIC OF GERMANY

Compiled by

H. Küsters
Nuclear Research Center Karlsruhe

## GENERAL

The gas-cooled pebble-bed reactor THTR-300 produced for the first time electricity of 110 MWe on 16 November 1985. After successful tests, it reached 60 % of the nominal power on 18 March 1986. On 4 May 1986 some absorber balls were stuck during a loading period and the reactor was shut down until 13 July 1986. On 17 September 1986 the power was increased to 285 MWe (80 % of nominal power). On 23 September 1986 the nominal power of 308 MWe was reached.

The fast prototype reactor SNR-300 has not yet obtained the licence to go into operation.

The PWR Mülheim-Kärlich (near Koblenz) with 1308 MWe reached nominal power on 10 July 1986. Commercial operation started in September. The KWU plant KBR-Brockdorf with a 1350 MWe PWR is expected to go into operation in October 1986.

# I. REACTOR PHYSICS ACTIVITIES AT THE NUCLEAR RESEARCH CENTER KARLSRUHE

## 1. Fast Reactor Physics

### 1a) Evaluation of Experiments at SNEAK

In SNEAK 12C and 12D displacements of core material from the center of the test zone to outer regions were simulated (radial displacement). These radial compactions were calculated in $S_4$-transport theory and gave satisfactory agreement with the measured reactivity effect. Using diffusion theory, large deviations were found. A few cases with small negative reactivity changes gave discrepancies up to 60 %. These cases will be considered further on /1/.

In the four central SNEAK elements various absorber configurations were simulated. Here the effect of compacting the absorber rods was studied. Calculations in (x,y) geometry, using transport theory, gave good agreement with the measured results; diffusion theory /2/ underestimates the reactivity effect of compaction.

The influence of a mixture of $B_4C$ and $ZrH_x$ rods on reactivity was investigated. For natural boron rods in $ZrH_x$ the absorption in $B_4C$ is enhanced, while for enriched boron rods the influence of $ZrH_x$ is practically negligible; only for many $ZrH_x$ rods ($\geq$ 16) the reactivity of the boron rods is decreased /2/.

In a DEBENE/French cooperation sodium loss experiments have been performed in RACINE 1D. This assembly had an internal breeder ring and a central control rod position. The experimental results could be calculated satisfactorily with exact perturbation theory to diffusion theory, but using anisotropic diffusion coefficients; large deviations occur in isotropic diffusion theory, which is caused mainly by an overestimation of the axial leakage. Transport theory gives results in good agreement with measurement.

The evaluation of the RACINE/IRMA experiments (intercomparison of reaction rate measurements) was done with standard programs of KfK. Good agreement (within the uncertainty of the measurements) was found for the absolute fission rates and for fission rate ratios; significant deviations were found for $^{238}U$ capture (+2.4% for the absolute neutron capture, +3.7% for the ratio $\sigma_{c8}/\sigma_{f9}$).

## 1b) Safety Investigations

The neutronics of the safety analysis code SIMMER was improved with respect to computing time. A more effective procedure to determine macroscopic cross-sections was implemented. In addition, the method invented by Lewis, Miller and Larsen to solve the neutron transport equation was adapted, using the multigrid method for solving the corresponding diffusion problem.

Improvements in the fluiddynamic module of SIMMER were reached by introducing more realistic exchange functions for mass, impulse and energy; three velocity fields are considered.

A new method to deal with sudden changes (especially narrowing) of the flow channel was developed and implemented in SIMMER /3/.

In the frame of the CABRI program four experiments with highly irradiated fuel rods were performed. In the pulse reactor ACRR (Sandia) important results for the fuel rod behaviour under transient conditions were found. The experimental results for the melting of the fuel clad and the movement of the molten clad material could be described satisfactorily well by theory /4/.

The experiments to simulate the movement of molten fuel through the structure of the upper axial blanket showed that in all cases a stable crust was formed on the surface. Penetration and wall temperatures could be well described by theory /5/.

## 2. Advanced Pressurized Water Reactor (APWR)

The investigation of a tight lattice PWR is performed in close cooperation with KWU-Erlangen, EIR-Würenlingen, and KfK-Karlsruhe. At present, the work is concentrating on further design studies, but especially on the experimental investigation and theoretical interpretation of the common PROTEUS experiments in Würenlingen.

## 3. Fuel Cycle Analysis for PWRs

In the area of reprocessing burnt fuel elements, a reliable determination of the activation of end-pieces (head and feet parts) of a fuel element is required for safe handling of these waste products. The calculated results were compared with experiments for the activation of the end-pieces for a BIBLIS fuel element /6/. The agreement between the experimental and theoretical activation values is vary satisfactory. Comparisons were made for $^{59,63}Ni$, $^{54}Mn$, $^{58,60}Co$, $^{55}Fe$, and $^{51}Cr$. Only for $^{55}Fe$ a larger overestimation by theory was obtained, which cannot be explained by data uncertainties or approximations in the theory. The most important activation product $^{60}Co$ could be determined theoretically with high confidence, as the comparison with experimental results showed. $^{60}Co$ is produced via $^{59}Co$, an impurity (0.3 %) in the steel composition.

It should be kept in mind that the activation of the end-pieces of a fuel element is about two orders of magnitude less than the activation of structural materials in the core region.

Recently a test of JEF data for application in PWR fuel cycle analysis was started. A group constant set for 20 heavy isotopes, 4 structural materials and about 80 fission products is being established, using the weighting spectrum of the KWO-PWR and NJOY as processing code. Many irregularities occurred, partly due to the data from JEF, partly due to the processing of these data by NJOY. Especially for high concentration of the isotopes considered, resonance self-shielding factors were physically not meaningful. The corrected group set will be applied to the irradiation behaviour of KWO-PWR fuel and to the NEACRP-$^{236}U$ benchmark. For KWO, the theoretical results will be compared with experimental results for the isotropic concentrations. Next, the NEACRP-Fast Reactor benchmark will be investigated.

## 4. Investigations on Fusion Neutronics

The transport system GANTRAS with a rigorous treatment of anisotropic scattering has been completed for plate and spherical geometry /7/. Further development is concentrating now on the treatment of 1d-cylindrical and two-dimensional geometry. A new scheme for determining the scattering matrices with full use of double-differential neutron data for application in transport calculations was developed. The scattering matrices are given in energy/angle representation. The corresponding code has been fully tested on Pb.

The data base of the Monte Carlo code MCNP is being updated at present using EFF as a data base.

KERMA factors characterize the energy deposition in matter. For Zr no data existed so far; new data for neutron scattering and capture were established. In general, the existing files often contain data of old origin. Data for Fe, Cr, Ni, Cu and Mn will be improved and will be made available for use in the European fusion blanket investigations.

The use of beryllium as neutron multiplier for both ceramic breeder and liquid metal blankets was analysed /8/. The neutron economy is best in a heterogeneous "sandwich-type" arrangement. This solution needs only a minimum of beryllium inventory. Monte Carlo calculations show that heterogeneity effects are negligible in such a blanket. The use of beryllium for liquid metal blankets makes it possible to omit the inboard breeding blanket or to reduce the blanket thickness considerably.

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## II. REACTOR PHYSICS ACTIVITIES AT KRAFTWERK UNION

### 1. Development of the Nodal Code HEXNOD

This contribution describes the work which was done during the first period of phase 2 on the development of three-dimensional nodal diffusion and transport theory methods for hexagonal-z-geometry.

Based on a 2D prototype program which had been developed and tested in phase 1 of this development project /1/ the 3D version of the nodal code HEXNOD was programmed and tested. Presently this code allows to solve the diffusion theory equations only. Work on the extension of the nodal transport method to 3D has begun and will be carried out during the next six months.

The 3D diffusion theory option of HEXNOD was tested by comparing the results for the modified 3D SNR Benchmark Problem with accurate reference solutions. For this problem excellent accuracy is obtained with very small computing times. Numerical calculations for two-dimensional neutron transport problems also indicate good accuracy of the low order nodal transport method which is presently available as an option in the 2D version of HEXNOD /2/.

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## 2. Advanced Computer Codes for In-Core Fuel Management

The KWU standard design and fuel management code system for PWRs is based on the advanced nodal reactor analysis methods developed at KWU and meets all requirements for first core and reload design. In particular, the system takes into account special features like: the accurate treatment of burnable absorbers, the three-dimensional calculation of the pinwise power distribution for the determination of DNB-ratios, a pinwise determination of the exposure history dependence of waterside corrosion of fuel rods and, finally, the capability for performing multidimensional transient calculations for load follow and safety analysis.

A key property of this code package is the fact that no reactor- or cycle-dependent adjustments are necessary. Hence, the KWU system obviates the need of any user normalization for acceptable results and, at the same time, allows much more reliable predictions from one cycle to the next or for the transition from conventional to modern reload schemes.

Modern in-core fuel management /1/ of PWRs centers on improved fuel utilization and on achieving better economics of plant operation. Respective topics are low leakage strategies, the increase of discharge burnup, flexibility in cycle length, the reduction of the neutron fluence in the pressure vessel, the use of plutonium mixed oxide fuel assemblies, or load following operation. The corresponding changes in reload design can be introduced successfully only if the related safety questions are settled accurately. Thus the basic reactor depletion code must be supplemented by codes for transient and safety analysis. Besides, pinwise DNB-ratios have to be determined which allow less conservative safety margins.

KWU's code system for in-core fuel management and safety analysis meets the above requirements, Fig. 1. It consists of the SAV-system /2/ for reload design, the PANBOX-system /3/ for transient and safety analysis and the code FDELTAH-80 /4/ for thermal-hydraulic analysis. It is a unique feature of this system that it is based on the nodal expansion method /5/. This method is used in both the stationary reactor depletion code MEDIUM-2 and the transient code PANBOX. The nodal approach combined with suitable homogenization and dehomogenization methods /6/ allows orders of magnitude savings in computing time compared to standard finite difference methods without the need for concession in accuracy and spatial detail.

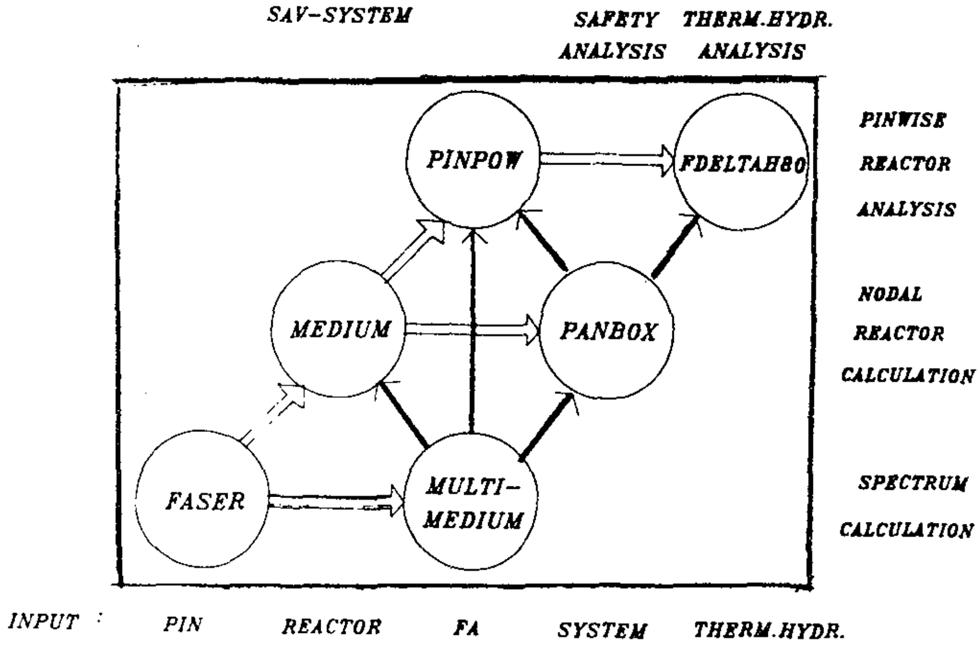



Fig. 1: KWU Advanced Computer Codes for In-Core Fuel Management

/1/ W. Böhm, H.-D. Kiehlmann, A. Neufert: "Advanced In-Core Fuel Management for KWU and other Vendor PWRs", ENC '86 Transactions, Vol. 4, p. 371

/2/ K. Koebke, L. Hetzelt, M.R. Wagner, H.-J. Winter: ATKE, 46, 224 (1985)

/3/ H. Finnemann, W. Gundlach: ATKE, 37, 176, (1981)
J. Lockau, H. Eckey, H. Finnemann, W. Gundlach: ATKE, 37, 249 (1981)

/4/ P. Suchy, G. Ulrych: "Höhere Betriebsflexibilität durch verbesserte thermohydraulische Ganzcorerechnungen in KWU-Druckwasserreaktoren und Übergang von $F_{\Delta H}$ auf DNBR", Proc. KTG Conference, Munich, (May 1985)

/5/ H. Finnemann, F. Bennewitz, M.R. Wagner: ATKE, 30, 123 (1977)
H.D. Fischer, H. Finnemann: ATKE, 39, 229 (1981)

/6/ K. Koebke: "Advances in Homogenization and Dehomogenization", Proc. ANS/ENS Intl. Topical Mtg., Munich, Vol. 2, 59 (April 1981)

## 3. Development of a BWR Training Simulator

A Training Simulator is under development which models the BWR core for normal operation and in case of transients including core stability tests.

The computer program system of the core part of the simulator mainly consists of two parts, a neutron kinetics and a thermal hydraulics model which are coupled iteratively.

Basis of the neutron kinetics model is the two group diffusion equation containing six groups of delayed neutrons. The influence of moderator density, control rod pattern, Xenon and Doppler reactivity effects in the fuel will be recognized.

In order to integrate the diffusion equation in 3D, the core will be divided into superboxes including reflector.

The thermal hydraulics part consists of cooling channels in which the equations for continuity, energy and momentum are considered for one- and two phase flow variing in time, using an empirical slip model.

The resulting moderator densities lead to cross sections for the time dependend neutron diffusion equation which is implified by partial integration over volume and surface of boxes. The neutron flux densities in the different boxes will be coupled with partial currents over the box surfaces /1/.

/1/ H. Finnemann, H. Raum:
"Nodal Expansion Method for the Analysis of Space-Time Effects in LWRs"
Proceedings of a Specialists' Meeting on the Calculation of 3-Dimensional Rating Distribution in Operating Reactors
Paris, November 26-28, 1979

## 4. Development of a Power Distribution Control (PCD) for KWU-BWR

The development of a power distribution control is a logical continuing development of the FNR (Fortschrittlicher Nuklear Rechner) /1/. The FNR is the advanced calculator for KWU-BWR core performance. The power distribution control connects the existing systems: process computer and FNW with the rod and flow control (Fig. 1). Then it will be possible to control the reactor operating under well known conditions automatically.

Fig. 2 shows the principle of the Power Distribution Control (PCD). The PDC consists of a fast calculating 3-dimensional program. Using available libraries the neutron physical and thermal hydraulic results of the FNR are homogenized and the coupling coefficients between the basic-model (RS3D) and the control-model (TRANSC) are calculated. On this basic values the power distribution (actual distribution) is determined considering the J-Xe dynamics. The calculation of the desired distribution depends on a well known control rod drive strategy and thermal hydraulic limits (e. g. MCPR).

The desired values of the control rod positioning processor and the recirculation pump controller can now be given iteratively.

The homogenization of the nuclear data from the basic model and the calculation of coupling coefficients are carried out in a stationary model (STARTC). STARTC divides the core in a number of desired superboxes corresponding to the boxes in the basic model.

Fig. 3 gives the pattern of some different assemblies to characterise the x-y superbox geometry for a 1/4 core.

In the case of a complicated control rod pattern it is significant to use a fine mesh homogenization (48 superboxes), in the case of an assembly without control rods it is adequate to use a coarse mesh homogenization (4 superboxes).

The developing work of the Power Distribution Control is in stage of various test calculations to prove the control model. Preliminary results show that - in principle - the control model works well.

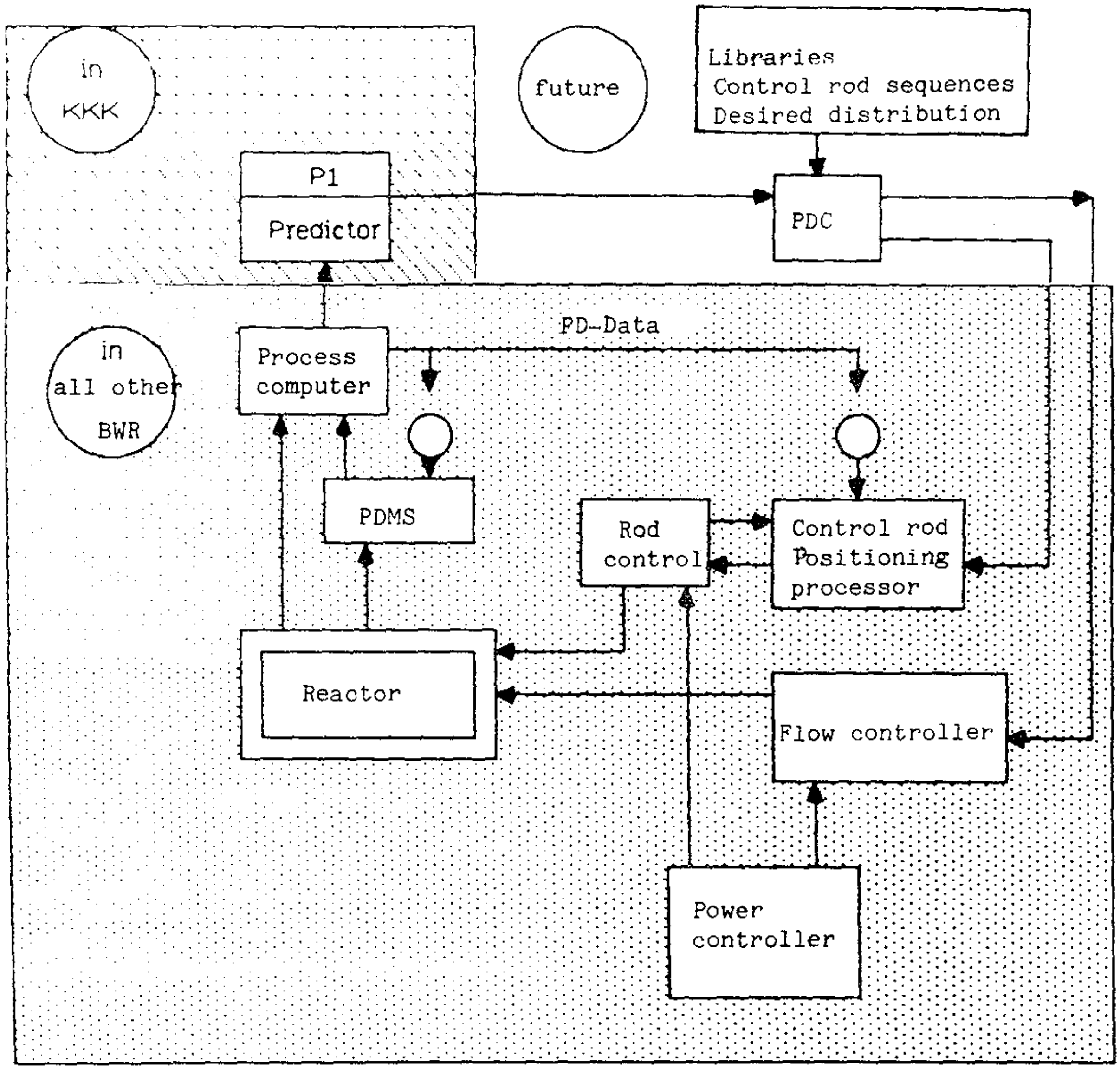
The maximum deviation of local power between the basic and the control model was in the case of

- a mass flow transient < 3 %
- a control rod transient < 7 %

by coarse mesh homogenization.

We expect an improvement of the results by fine mesh homogenization.

/1/ H.-D. Lemke, K. Hünchen,
Advanced Nuclear Predictor (FNR-System) for
On-line Operation in BWR's
ENC '86 Transactions, Geneva, June 1-6, 1986
Page 281-286




PDMS = Power Distribution Monitoring System

PDC = Power Distribution Control

Fig.1 Single - Line Diagram of the KWU- BWR Power Control

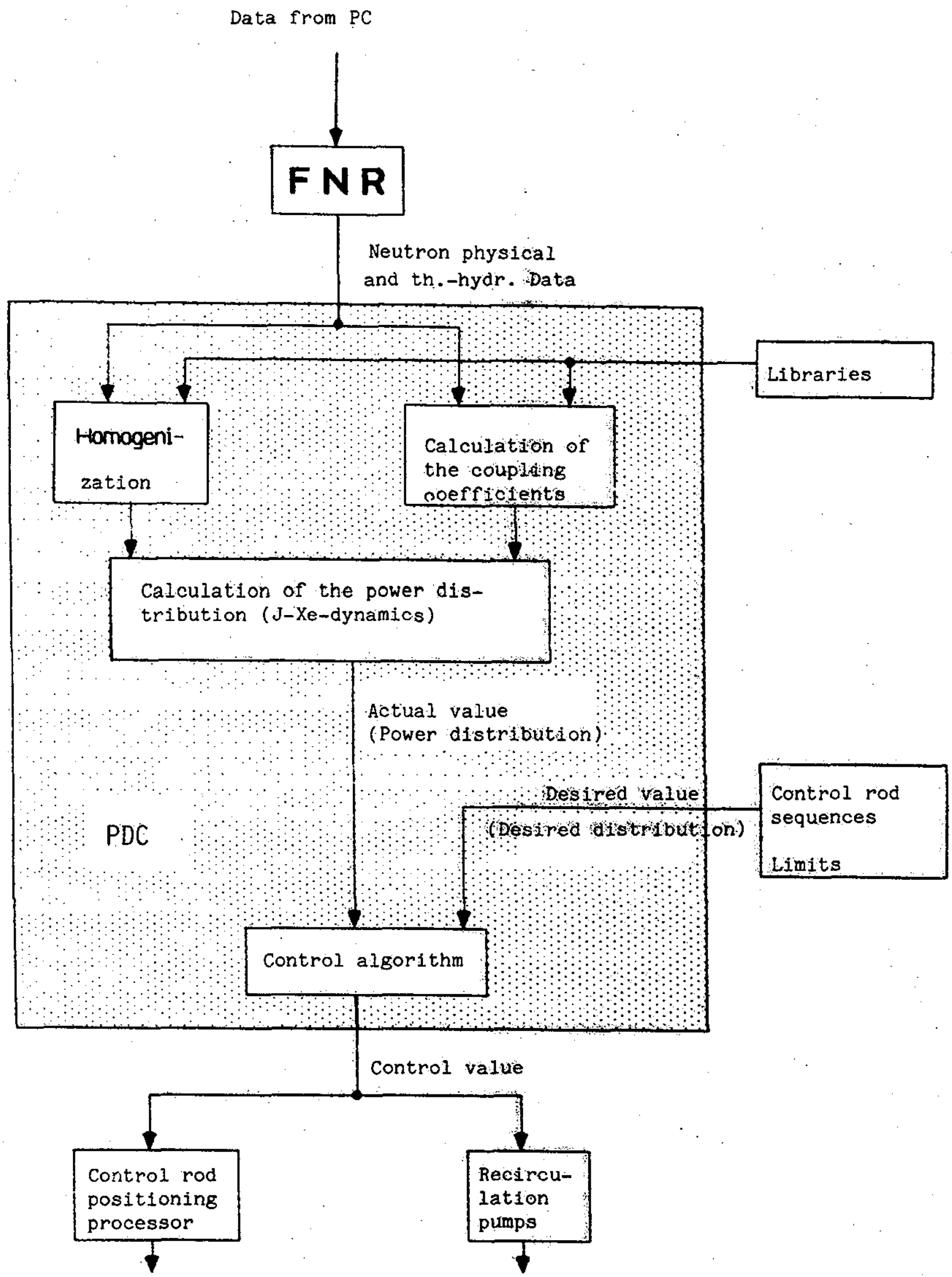



Fig.2 Single-Line Diagram of the Power Distribution Control (PDC) (KWU - BWR)

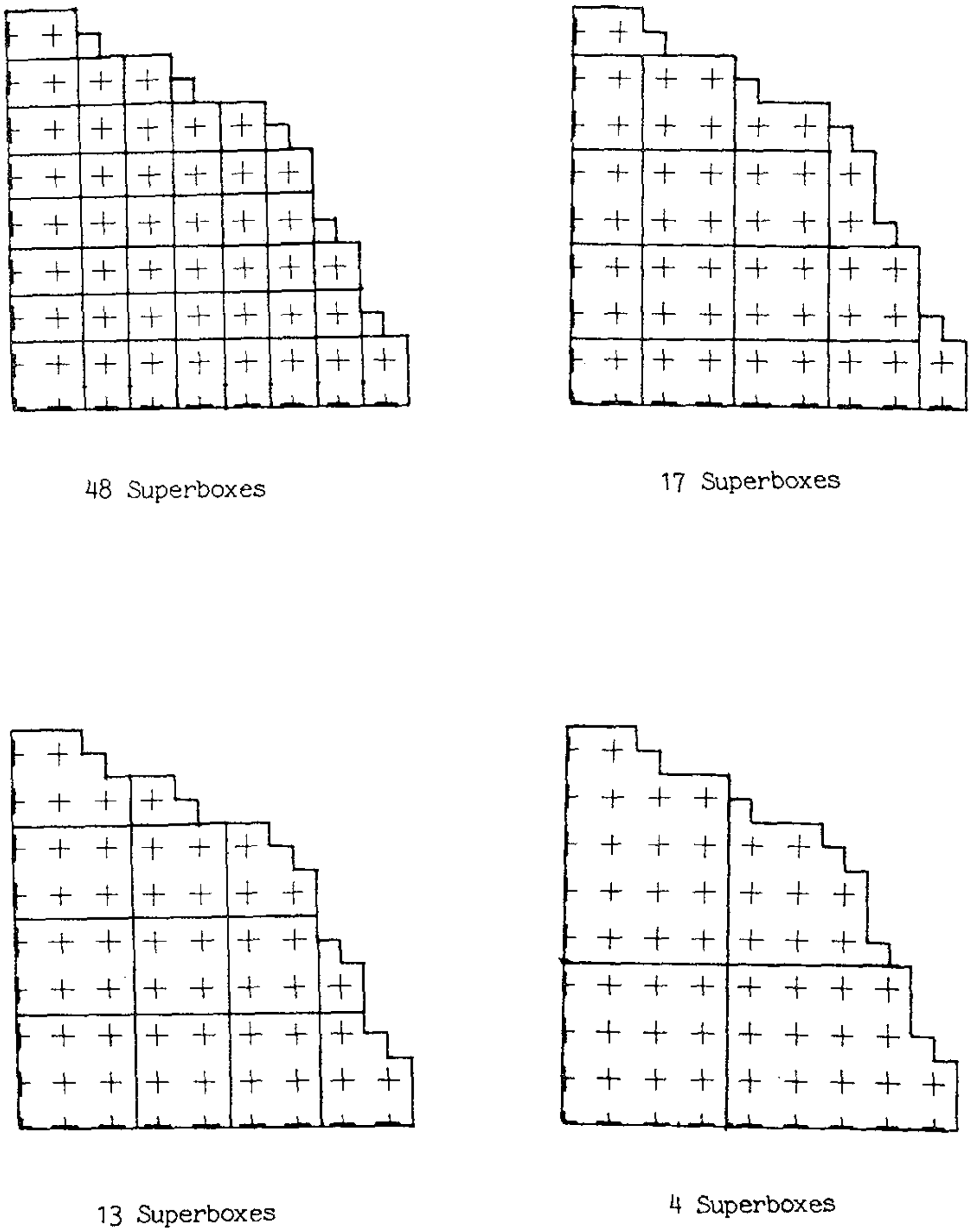



Fig.3 Pattern of Some Different Superboxes for a 1/4 Core (210 Boxes in the x-y Plane)

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

## 1. Development and Verification of a Program System for the Calculation of Neutron Spectra and Weighted Group Constants for Thermal and Epithermal Systems

(M. Arshad)

A new program system for spectrum generation and evaluation of group constants for nuclear reactors has been developed. The program is based on a combination of the first collision theory and the $B_n$-approximation to the neutron transport equation and is well suited for the thermal and epithermal lattices. The treatment of the resolved and unresolved resonances has been improved to incorporate the $\sigma_o$-dependence of the cross sections and the lattice cell heterogeneities.

The improved treatment of the resonance absorption enables to compute the assemblies containing ceramic as well as metallic fuels with a large variety of fuel to moderator ratios. Several experimental benchmarks have been computed using this program and the effective multiplication factors as well as the reaction rates lie very close to the experimental results. It has been shown that neither the $\sigma_o$-method (without one dimensional cell calculations) nor the one dimensional calculations (with cross sections independent of $\sigma_o$) alone can be used for the entire resonance energy range. Based on the results of the benchmark calculations a combination of both methods is proposed.

## 2 JEF-1 Data for Application in Thermal Reactors

(W. Bernnat, M. Mattes, D. Lutz)

During the last period a large number of different criticality calculations were carried out based on JEF-1 data:

- CSEWG benchmarks BAPL-UO2, TRX, ORNL, PNL, MIT [1]
- HIC benchmarks [2]
- SAXTON experiments [3]
- some benchmarks of the CSNI working group [4]
- analysis of criticality safety of different fuel storages
- fuel cycle calculations for a pressurized water reactor with burnable poison (boron glass)
- calculation of the pin-wise power distribution in PWR and BWR assemblies

The results of CSEWG and HIC benchmark calculations are documented in [5].

The agreement of calculations based on JEF-1 data and experimental values is very good for the most important thermal systems like homogeneous uranyl-nitrate solutions, low-enriched $U_{metal}$ and $UO_2$ lattices as well as $UO_2$-$PuO_2$ lattices with $H_2O$ moderator. For some homogeneous Pu-nitrate systems and for wide lattices of $UO_2$-$PuO_2$ systems deviations of more than 1 % from the experimental $k_{eff}$ were found.

The burnup calculations based on JEF-1 data agreed well with a reference solution [6]. Comparisons of JEF-1 results with corresponding results derived from ENDF/B-V data show a very close agreement for practical all thermal systems for which ENDF/B-V results are available.

[1] Cross Section Evaluation Working Group Benchmark Specifications. ENDF-202 (Nov. 1974)

[2] Boynton, A.R., et al.: High Conversion Critical Experiments. ANL-7203 (Jan. 1967)

[3] E.G. Taylor: Saxton Plutonium Program. WCAP-3385-54/-EURAEC-1493, 1965.

[4] Standard Problem Exercise on Criticality Codes for Spent LWR Fuel Containers. CSNI Working Group. CSNI Report, No. 71(1982)

[5] W. Bernnat, M. Mattes et al.: Analysis of Selected Thermal Reactor Benchmark Experiments Based on the JEF-1 Evaluated Nuclear Data File. IKE 6-157; JEF REPORT 7.

[6] Sizewell B PWR Pre-Construction Safety Report. April 1982

## 3 Investigation of the Neutron Scattering Dynamics in the Liquid Phase of Hydrogen ($H_2$, $D_2$) for Application of Could Neutron Source Optimization

(J. Keinert, M. Mattes, J. Sax)

An improvement on the neutron scattering dynamics model derived by KOPPEL, YOUNG /1/ is in process at IKE. The KOPPEL YOUNG theory considers exactly only the intra-molecular spin dependence and the rotational and vibrational modes of liquid hydrogen. The translational modes are treated as a motion of the free molecular unit, which is not an adequate assumption for the energy range of cold neutrons. For the translational modes we assumed both a frequency distribution as in the solid phase and a hindered translation. Therefore the neutron scattering theory for liquid hydrogen was splitted into partial scattering laws calculated with a modified Koppel-Young model (hindered translations) and with GASKET /2/ for the continuous phonon- spectrum and then linked by the convolution theoreme of Fourier transforms. The scattering laws were stored in ENDF-5 format for the ortho- and para-modifications. Scattering matrices were generated for 165 neutron energy groups up to 3 eV for neutron transport calculations with a modified version of the THERMR module of NJOY /3/.

### References

/1/ Young, J.A.; J.U. Koppel: Slow Neutron Scattering by Molecular Hydrogen and Deuterium. Phys. Rev. 135, A603 (1964)

/2/ Koppel, J.U.; Triplett, J.R.; Naliboff, Y.D.: GASKET a Unified Code for Thermal Neutron Scattering. GA-7417 (1966)

/3/ Mac Farlane, R.E.; Muir, D.W.; Boicourt, R.M.: The NJOY Nuclear Data Processing System. Vol. II: The NJOY, RECONR, BROADR, HEATR, and THERMR Modules. LA-9303-M (ENDF-324) (1982)

## 4 Analysis of LWR-Pressure Vessel Neutron Exposure

(G. Prillinger)

To determine neutron exposure of several trepans taken from the pressure vessel of the nuclear power plant Gundremmingen Block A (KRB-A) a three-dimensional neutron spectrum analysis has been performed by IKE. The work, funded by US-NRC, is part of the KRB-A program analysing through vessel wall embrittlement of a real reactor vessel at the end of operation.

The Boiling Water Reactor KRB-A has a nominal thermal power of 801 MW (250 MW electrical). The reactor was put in operation in November 1966 and, until the last shutdown on January 13th, 1977, generated a total of about 16 TWh of electrical power, with an average availability of 75 %.

After decomissioning 15 trepans have been taken at different axial and azimuthal postions within the 90 to 135 degree octant of the reactor. To get accurate assessments of the absolute magnitude of the interested exposure parameters such as fast fluence ($E > 1.0$ MeV), fluence ($E > 0.1$ MeV) or dpa (displacements per atom in iron) the three-dimensional flux distribution has been calculated using the discrete ordinate transport code DOT4.2. Combining the results of a $(R,\Theta)$- and $(R,Z)$-model of the reactor the neutron spectrum is defined with sufficient accuracy. To perform the transport calculations on an absolute scale an evaluation of the reactor power-time history and of the three-dimensional burnup-distributions at begin and end of each of the reactor cycles were necessary.

Dosimetry results from the activated trepans and from surveillance capsules, which had been irradiated during reactor operation at accelerated positions have been compared with calculated activities showing good agreement.

From the trepans material samples will be fabricated and tested to get correlations between embrittlement and neutron exposure at end of life conditions of the vessel.

# Reactor Physics Activities in Italy

R. Martinelli, ENEA

## INTRODUCTION

Thermal Reactor Physics activities in Italy have provided support to the operation of Caorso BWR/4, to the design and licensing reviews of Alto Lazio BWR/6 and of the PUN concept of PWR and to the pre-commissioning of the heavy water-moderated, boiling light water-cooled experimental reactor CIRENE.

Most activities in the LMFBR area have been carried out in the framework of the Research and Development Agreement between ENEA and the French CEA: in July, surprisingly enough, ENEA has been (provisionally) inhibited by the Parliament from assuming new committments in this area - including the development of the fast test reactor, PEC.

Actually, the whole nuclear component of the Energy Plan is heavily questioned - once again - by opposers, finding - for the first time - a strong political support.

The "National Energy Conference", scheduled for the end of the year, is expected to clarify the positions and to be the basis for unambiguous decisions.

# 1. INTEGRAL EXPERIMENTS

## 1.1. MASURCA

ENEA partecipates in the design and interpretation of "BALZAC", an experimental programme supporting the design of SUPERPHENIX 2 /1/.

The programme - based on a parametric study of simple core configurations - is aimed at:

- reducing to 5÷10% the uncertainty of the heavy isotopes component of the reactivity loss per cycle, which is estimated, for the first cycle of SPX2, at about + 30%;
- confirming the neutronic characteristics of specific design features such as the irradiated subassemblies internal core storage, the use of $B_4C$ subassemblies between core and storage, the reduction of the radial blanket and suppression of the upper axial blanket;
- reducing the uncertainties related to the gamma-ray heating and neutron energy deposition for absorber and diluent subassemblies, and validating the appropriate "formulaire";
- validating the calculational methods of the radial and axial heterogeneity of control rod subassemblies of the type envisaged for future large LMFBRs.

The status of the programme is illustrated in the French progress report to the Committee. (ENEA/VEL, Casaccia)

## 1.2. TAPIRO

### 1.2.1. Absolute Reaction Rate Measurements

Following the neutron characterization campaign/2/of this enriched-Uranium, Copper reflected cylindrical fast source reactor, the activities have been focused on the interpretation of the experimental data and on the improvement of the techniques for absolute fission rate measurements.

The analysis of the mesurements in the TAPIRO reflector and in the Copper spheres in BR1 (Mol) has shown inconsistencies in the results of the latter, thus making advisable to repeat (and increase) the BR1 experiments and to proceed to a complete re-evaluation of the Cu cross-section.

Both activities are underway.

As to the measurement techniques, present accuracies (1 ) for absolute reaction rates are 3÷4% (fission detectors) and 2% (other detectors). The results for intermediate/hard spectra are in agreement with those obtained in USA (ILRR programme); it is worth nothing that the range has been extended to much softer spectra in the case of TAPIRO experiments. (ENEA/VEL, Casaccia)

### 1.2.2. LMFBR Blanket Neutronics

The "NEFERTITI" programme/3/ has been completed at last, with the mesurements in the "End of Cycle" configuration. Due to limited availability of enriched fuel rodlets (simulating the Pu build-up) it was only possible to load blankets with reduced thicknesses.

This inconvenient should not affect the results dramatically, as the partial blanket dimensions were chosen basing on the criterion that the spectral indexes did not differ significantly from those in the full-size configuration (ENEA/VEL, Casaccia)

### 1.3. CIRENE Pre-commissioning in RB-3

A measurement campaign has been completed, aimed at the determination of the power distribution inside the fuel bundles, each containing 18 pins. These mesurements are of particular interest for the peripheral CIRENE fuel elements, wich are slightly enriched and subjected to relatively large flux gradients. The analysis of the experimental data has demonstrated the excellent accuracy of the design methods in predicting the fine structure of the power distribution/4/. (ENEA-CIRENE, Bologna)

## 2. LWR CODE SYSTEM

ENEL, with the technical support of CISE, have implemented their Integrated Code System (EICS) for LWRs, to provide an accurate and effective system for core plant design and operation. The single models, correlations, codes as well as the whole system have been checked with respect to sophisticated reference codes, uniform lattice experiments, full-scale power plants operating data and post-irradiation isotopic analyses, both for uranium and plutonium loaded lattices. This is the result of a large development programme, aimed at the optimization of the available computational methods in terms of overall performance. The main goals of the programme have been: development of simplified models and correlations and faster iteration techniques without loss of accuracy within the range of normal applications; no use of albedos or adjusting parameters; modularity in the codes structure to simplify the introduction of new materials or models; input data requirements as close as possible to the physical description of the plant structures; automatic data transfer between the codes of the system to avoid redundancy or inconsistency of the data to be supplied; automatic comparison with experimental data.

Most recently, efforts have been spent to implement and test the set of codes included in EICS to carry out interface functions between the fuel assembly library code, BWR and PWR core steady-state and dynamic simulators and full plant transient simulators (they also allow quick results comparisons and graphic representations).

The preparation of the interface codes for the 1-D core models of the plant dynamic simulators has required particular care. In fact, the interface code had to contain the neutronic and thermal-hydraulic core models, in order to preserve consistency between the 3-D and 1-D models in terms of axial power shape, control rod worths and core response to thermal-hydraulic perturbations.

EICS has been extensively tested against operation data from Caorso and Hatch 1 (BWRs) and Zion 2 (PWR).

These tests have demonstrated the effectiveness and flexibility of the code system, together with its wide range of possible applications. Moreover, the automatic transfer of information from the different codes in EICS implies that the time for input data preparation, result analysis and data handling is minimized. Also, the computing times are very short/5/. (CISE, Milano and ENEL/DCO, Roma)

# 3. CALCULATIONAL METHODS

## 3.1. Advances in Generalized Perturbation Theory

### 3.1.1. Fuel Evolution and Shuffling Studies

A new methodology based on generalized perturbation theory (GPT) has been developed which allows to make perturbative studies of burnup taking into account the coupling between the nuclide and the neutron densities. Typical quantities which can be analyzed with it are:

- the amount of a specified material in a given region at the end of the reactor life cycle;
- the d.p.a. for a specific material and position;
- the residual reactivity at the end of the reactor life cycle.

The changes of these quantities can then be estimated in terms of perturbations of the parameters related to:

- nuclide distribution at beginning of cycle;
- control material, or rod, strategy;
- shuffling strategy/6/.

A particular feature of the methodology is that it makes use only of standard GPT codes already generally available.

The calculation results so far obtained confirm the validity of the method. Parameter perturbations typical of burnup studies (i.e., of the order of 10-20%) can be reproduced, at first order expansion, to a few percent accuracy.

The proper overall control reset - to maintain criticality - is implied in the perturbed system via an intensive control variable: this gives better results than the fictitious - reset (affecting the fission production) often used in this type of studies/7/. (ENEA/VEL, Casaccia)

### 3.1.2. GPT Methods for Reliability Analysis

The potentiality of GPT methods for reliability and availability analysis has been explored in the Markov chain scheme. Formulations giving the sensitivity coefficients with respect to typical system parameters, such as the failure and repair rates, have been obtained, together with their relationship with the well known Birnbaum expressions for sensitivity coefficients/8/.

This methodology could benefit from developments of GPT methods and relevant codes in reactor physics. In fact, the evolution matrix governing the nuclide field, apart from the physical constants used, formally corresponds to the transition matrix of the Markov chain process. Codes like ORIGEN could well be used to the purpose. (ENEA/VEL, Casaccia)

3.1.3. Applications of Higher Order GPT

Completing the work carried out in the previous year, a method based on higher-order perturbations has been implemented at CEA-Cadarache, for the calculation of reactivity and power distribution variations induced by control rod movements in fast power reactors/9/. (ENEA/VEL, Bologna)

3.2. LMFBR Design Methods

A design-oriented study has been conducted/10/ to analyze heterogeneity effects in central and off-center control rods in hard-and soft- spectrum LMFBRs (typically, PEC and SUPERPHENIX). Various cross section homogeneization models are considered and suggested. The conclusion for PEC (the SPX situation is more complex) is that properly applied standard diffusion methods supply the same values of rod worths as transport methods, due to stable mesh-size dependent factors - compensating for the heterogeneity effect - which characterize diffusion calculations. (ENEA/VEL, Casaccia)

3.3. LMFBR Core Simulator Studies

Computational strategies to optimize the performance of a 3-D LMFBR core simulator - with particular reference to PEC - have been defined in a recent report/11/. It is demonstrated that the adoption of choices such as: exagonal meshes; albedo terms to eliminate part of the reflector points; three energy groups instead of six, reduces the total CPU time (CITATION code) by a factor of about 100, while keeping an accuracy quite acceptable for a simulator. (ENEA/VEL, Casaccia)

## 3.4. Heavy Isotopes in LMFBR Fuel Cycle

The analysis of new results - and the re-analysis of previous data - from TRAPU and PROFIL irradiation experiments, have been presented in a recent paper/12/ with a significant ENEA contribution.

The French experiments (TRAPU, irradiations of mixed-oxide pins containing Plutonium with different isotopic compositions but loaded with higher isotopes; PROFIL: irradiations of pure separated isotopes) have shown high accuracy and very good consistency. They have been analyzed using the CARNAVAL-IV "formulaire" and, for a large number of isotopes, also JEF-1 evaluations.

The adjusted data set CARNAVAL-IV has shown a very good performence in the analysis of these experiments for the major actinide data. The first version of the new evaluated data file JEF-1 has indicated an overall satisfactory performance, even if areas for possible improvement have also been clearly pointed out, such as: U-238 capture (underestimated), Pu-241 capture (overestimated) and Pu-239 fission (probably overestimated). (ENEA/VEL, Casaccia)

## 4. SHIELDING

### 4.1. Spent Fuel Transport Cask

The Italian contribution to the international intercomparison on shielding codes for transport casks is in progress.
Shielding calculations concerning different flask configurations were carried out via the modular code SCALE-2 in multigroup 1-dimension approach and MCNP Montecarlo code.
A complementary study in underway, aimed at developing a new pointwise cross section library based on JEF (Joint Evaluated File) to be used in MCNP shielding assessment of transportation packages. (ENEA/TIB, Bologna)

### 4.2. Neutron Propagation

A common CEA-ENEA paper /13/ ha been prepared for the NEACRP Specialists' Meeting on Shielding Benchmarks, reporting on ASPIS and HARMONIE neutron propagation experiments interpretation, performed using JEF-1 data. (ENEA/VEL, Casaccia)

## 5. DATA EVALUATION AND PROCESSING

### 5.1. Data for JEF/EFF

The new evaluation of the natural Cr and of its stable isotopes has been completed with the gamma-ray production data. The checking and testing processes are now underway and are expected to be completed soon. The re-evaluations of Al-27 and natural Si have been released for inclusion in the future JEF versions; for the present time the files are distributed within the EFF-1 tapes.

The 33 materials included in EFF-1 have been processed into the GEFF-1 group librapy in a cooperative effort of European laboratories. The italian group was in charge of the complete processing of eleven materials and of the photon interaction data for all the materials.

The french version of NJOY, THEMIS, has been adopted as the reference code. (ENEA/TIB, Bologna)

### 5.2. Delayed Neutron Data

An analysis is underway of experimental data concerning delayed neutron yields from U-233, Np-237, Pu-238, Pu-240, Pu-241. The analysis is carried out using the MINUIT code and is expected to produce coherent estimates of the errors and of their correlations. (POLITECNICO, Milano)

## 6. NOISE ANALYSIS AND RELATED METHODS

### 6.1. BWR Stability Monitor

The development of (the Caorso) BWR/4 stability monitoring system has progressed. A "Demo" system operates in a simulated on-line mode (real power plant signals are transfered to the system from a disk database) and supplies a picture of the thermal-hydraulic (power and flow) and of the dynamic (estimate of the decay ratio) states, for a selected channel or for the whole core/14/.

The complete system (also comprising a dedicated microcomputerized data acquisition unit and a specialized graphics module) will be installed at the Caorso BWR power plant in October 1987: it is designed to operate in a continuous monitoring mode for a few important signals and on request for local signals, thus becoming an operator's aid particularly useful during plant transients. (ENEA/TERM, Casaccia)

### 6.2. BWR Surveillance System

The design documentation is being completed for a BWR/4 (Caorso) surveillance system, based on Pattern Recognition techniques/15/.

Many hardware components are shared with the stability monitor (6.1.), while the software resides in the larger computer (VAX 11/780).

The operation of the surveillance system must refer to a dynamic plant model: to this purpose a global simplified model has been developed, valid for small variations of core parameters. (ENEA/TERM, Casaccia)

## 6.3. Time Series Analysis

In the framework of a research on the methodologies for time series analysis, the utilization of Kalman filters for parameters estimate in an autoregressive process, is investigated. Preliminary results show that Kalman filters can be profitably used when the process coefficients are time-varying, i.e. for transient analysis/16/. (POLITECNICO, Milano)

## 6.4. Diagnosis and Monitoring on Components

In view of the dim prospects of the nuclear industry, components common to nuclear and conventional plants become more and more appealing to noise and signal processing experts. This is, for instance, the case of turbogenerators, for which an expert system for malfunction diagnosis and monitoring equipment for fatigue damage evaluation are being developed/17/. (ENEA/TERM, Casaccia)

# 7. FUSION BLANKET NEUTRONICS

The main activity of the ENEA Fusion Technology Programme on blanket design studies has been focused on "Il Mantello" /18/, a Helium-cooled solid breeder blanket for NET (Next European Torus). This study, performed within the framework of Technological Tasks of the European Fusion Programme on components of the NET system, has been completed and, based on the relevant conclusion, a new design variant is being investigated.

The analysis /19/ has shown that the tritium breeding ratio (T.B.R.) of "Il Mantello" blanket in its reference configuration is adequate for NET application.

However, there is a typical nuclear effect of Be that occurs in all the designs where the Be multiplier is placed between the first wall and the breeding region: strong non-uniformities in nuclear heating and tritium production (and, hence, in temperature distribution and tritium inventory) come out in the form of undesirable spikings at the interface between Be and other materials (first wall structure and breeder).

This can be avoided - as in the solution adopted for "Il Mantello" - using proper Be/ceramic breeder intimate mixtures that exploit the moderating properties of Be leading, as it is well known, to a considerable increase in T.B.R. and to much more uniform distributions in nuclear heating and tritium production, compared with those obtained in the conventional multiplier/breeder configuration. Detailed 3-D neutronic analyses have been performed for the blanket designs by using the continuous Monte Carlo code M.C.N.P.. This code allows the most accurate treatment for neutron and related gamma-rays transport problems in complicated geometries.

Two 3-D complementary geometrical models have been adopted for the analysis (1-D models have also been used for parametric studies).The first is a 3-D detailed geometry of a single blanket module, with suitable boundary conditions, representing the "almost real" features of different materials (breeder, multiplier, ...), cooling system, box sides and gaps, and avoiding any material homogenization. By using this model, the fine space distribution of tritium production and energy deposition in a discrete number of cells (up to about 200) suitably shaped, and the local value of the tritium breeding ratio, have been calculated. The second model consists of a 3-D toroidal geometry representing the DN (Double Null) NET device. The geometrical description of the blanket module has been simplified by using homogenized regions (heterogeneity effects have been taken into account by using proper correction factors obtained from the first model described above). By using this model, the tritium breeding ratio from the real system, the radiation dose behind the shield and streaming effects through the major penetrations have been calculated, together with tritium production and energy deposition in the whole blanket/20/.

The same methods have been used to determine the conditions for optimum neutronic performance of the blanket. However, engineering problems are outstanding; actually, the main conclusion of the work is a strong suggestion to change the design so as to contain the coolant in tubes and to make it flow along the poloidal direction in the breeder volume. Accordingly, there are plans to study a blanket for NET where both the coolant and the breeder material (mixed breeding moderator) are contained in poloidal tubes. (ENEA/FUS, Frascati)

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# REACTOR PHYSICS ACTIVITIES IN JAPAN
# (OCTOBER 1985 - AUGUST 1986)

Compiled by

S. Matsuura, Y. Kaneko (JAERI) and K. Shirakata (PNC)

## INTRODUCTION

Analytical and experimental efforts were continued to support the developments of Liquid Metal Fast Breeder Reactor (LMFBR) and Advanced Thermal Reactor (ATR) on the national projects. Attention was drawn to some new topics in reactor physics, of which the R and D has become much active. One of these topics is concerned with High Conversion Light Water Reactor (HCLWR), which has been investigated from the viewpoint of reactor physics at Japan Atomic Energy Research Institute (JAERI), Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fabricators, and also at some universities. Moreover, improvement and assessment of HCLWR core design methods were proceeded. Another remarkable trend is an increasing attention to the criticality safety problems of reprocessing facilities. This trend has been enhanced by promoting the criticality experiment under the collaboration research program between Power Reactor and Nuclear Fuel Development Corporation (PNC) and US-DOE, and also by actually starting the nuclear fuel cycle safety experimental facility (NUCEF) program at JAERI. Concering reactor physics, much attention is paid to the evaluation accuracy of effective multification factor in complicated geometries and the measurement technique of deep subcriticality.

Much efforts have been devoted to the blanket neutronics of fusion reactor at JAERI and universities. That is, the major part of activities are the neutronics studies using the neutron source at the Osaka University based on the universities joint research program and the FNS experiments at JAERI under the collaboration research program between JAERI and US-DOE.

Related to Reduced Enrichment for Research and Test Reactors (RERTR) Program, JMTR of JAERI succeeded in the conversion from use of 93 % enriched uranium fuel to 45 % one at an on-power experiment. On the other hand, the Musashi Institute of Technology completed a series of reactor characteristic experiments to convert the U-ZrH fuel with Al cladding to a new fuel with SUS cladding.

Concerned with radiation shielding, continuous studies have been made to the radiation streaming problems at various facilities. Recently, much efforts have been paid also on benchmark problems, of which the major activity is the correspondence to the shielding benchmark conected with NEA.

## DATA AND METHOD DEVELOPMENT

Code systems SRAC and HELIOS.HX have been revised in order to improve an calculational accuracy on resonance absorption and actinide burnnp which affect significantly a design accuracy of HCLWRs. The improvements are concerning on the treatment of mutual intereference effect of resonances, updata of nuclear data for transpulutoniums and FP nuclides, and addition of new lattice geometries. It was confirmed through benchmark tests that these code systems could provide satisfactory accuracy on neutronics designes of HCLWRs.

A new formulation using a general curvilinear coordinate has been shown for neutron transport equations. This work intends to expand the conventional discrete ordinate method (Sn method) used in the Cartetian coordinate into a generaliged curvilinear coordinate which can flexibly represent actual geometries. The two dimensional transport equation was transformed into the one in a curvilinear coordinate by using the tensor transfer method and discretized by integrating over each unit cell in phase space. This method can solve transport equations in varions geometries by using an unified method. Some comparison calculations with conventional methods have shown an adequacy of the present method.

A new perturbation equation of integral transport equation has been formulated by using a generaliged surface perturbation theory, which is based on a collision probability method in order to calculate lattice characteristics. The present method was applied for calculations of $k_\infty$ dependancy on the volume ratios of three region cylindrical lattices. The differences between the present solution and the direct one is about 0.0025 $\Delta k_\infty / k_\infty$ even if the volume of outermost region is changed by 40 %.

One of the recent topics is concerning on the reevaluation of $\beta$ value. Many data obtained from the experiments at the SHE (Seme Hmogeneous Experimental Critical Assembly) show systematic descrepancies of C/E values between $\beta$ scaled parameters such $\Lambda/\beta$ and non $\beta$ scaled ones such as $k_{eff}$. To resolve such a in consistency, the value of $\beta$ has been adjusted. As a result, the most probable vlaue of $\beta$ for $^{235}U$ fission 0.00676 ± 0.00011 has been obtained. This value agree well with those of the recent measurements by Krick and Evans, and Cox et al..

## LWR VHTR AND ATR

Study on HCLWR characteristics has been extensively performed by JAERI, PWR and BWR fabricators and several universities. In JAERI, addition to the study by neutronics calculation, the critical experiments using Fast Critical Assembly (FCA) was started. The first core has a mockup test zone which consisted of a periodic array of 20 % EU, natural U, $Al_2O_3$ and polystylene plates, surrounded by a SUS buffer zone, a driver zone of Fast Breeder lattice and a blanket of depleted U. Although the core had fairly complicated configuration, it was confirmed that the prediction by the SRAC code system gave its condition with very high accuracy.

In the analytical study, effects of nuclear data and void fraction on the lattice parameters of HCLWR lattice parameters have been pursued in detail.

By the sensitivity analysis code based on a generalized perturbation theory SAINT, ambiguity due to nuclear data by nuclide and by energy affecting on neutronic characteristics has been investigated. The results show that those of resonance energy range (1 - 100 eV) contribute dominantly. A detailed analysis by the SRAC code on the component factors of void coefficient indicates that the shift of the void coefficient to the positive direction is caused by 1 eV resonance of $^{240}$PU at lower void state (0 - 10 % void fraction) and by increase of fast fission and decrease of neutron capture between 100 eV - 50 keV of $^{239}$PU at higher void state.

Moreover, study has been pursued in detail about the difference of control rod effectiveness between current PWR's and HCPWR and the availability of newly evaluated F.P. chain model (including 45 nuclides and one pseudo F.P.).

The critical experiments for the development of high temperature gas cooled reactor has been also continued using VHTRC (Very High Tempurature Reactor Critical Assembly). Following to the experiments at room temperature, those at 200 °C over the whole system has been performed. The experimental analysis by the SRAC code shows that the prediction gives the same accuracy as shown for the core at room temperature.

For investigating the nuclear characteristics of ATR (Advanced Thermal Reactor) Demonstration Power Plant and giving support to successful operation of the prototype reactor "FUGEN", studies of reactor physics parameters have been continued by using both uranium and MOX Fuels in DCA (Deuterium Critical Assembly). The measurements of 36 rod cluster carried out are as follows:

(1) Void coefficient
(2) Local power distribution in fuel cluster
(3) Power distribution in the core.

FBR

1. Preliminary Design Studies for Innovative Core Concepts

(1) Ultra Long Life Core (ULLC)
The concept has been evolved as one of the targets of the innovative approaches for the FBR cost reduction. The extension of core life time to the plant life of about 30 years, and the extension of average discharge burn-up to 200 - 250 GWD/t are desired. In the feasibility study of this concept, new cladding materials resistant to high neutron fluence, the ductless fuel sub-assembly, the coolant flow allocation accommodating power swings, and the increase of core volume with reduced power densities are investigated.

(2) Carbide Fueled Core
The carbide fueled core has the neutronics advantages of higher linear heat rate and high breeding characteristics over the oxide fueled core. Preliminary results of the feasibility study for a 1000 MWe-size carbide fueled core are as follows. These core performance parameters are obtained separately under the condition that the other parameters are the same as the comparable oxide fueled core.

a. The pressure drop of fuel bundle can be reduced to 0.9 $kg/cm^2$.
b. For the purpose of minimizing the core volume, the fuel sub-assembly length can be reduced to 325 cm by reducing the core height to 55 cm, and the core diameter can be reduced to 270 cm by keeping the core height of 100 cm.
c. The breeding ratio can be raised to 1.44, the doubling time can be shortened to 10 years and the excess reactivity can be reduced to 2.5 % $\Delta k$.

## 2. FCA Experiments and Analyses

A critical experiment programme on the axially heterogeneous LMFBR core has been performed at FCA since April 1984. This core has a disk-shaped internal blanket at the core midplane. The programme was divided into two sub-programmes due to the limitation of FCA fuel inventory. One was the FCA XII programme basing on "zone-type" assemblies for axial neutron flux tilt measurement by introducing the internal blanket. The experiments were already reported at the last NEACRP meeting. The other was the FCA XIII programme basing on "sector-type" assemblies in which the measurements were related to distortion of radial neutron flux and importance distributions especially in the vicinity of the radial edge of the internal blanket.

The FCA XIII programme has been performed from July 1985 to February 1986. Measured were radial distributions of reaction rate, power, control rod worth and sample reactivity worth. The core height including the internal blanket was 81 cm and the thickness of the internal blanket was about 30 cm.

Fission rate distributions of $^{239}Pu$, $^{235}U$, $^{238}U$ and $^{237}Np$ were measured by micro-fission counters, while $^{238}U$ capture rate distribution by depleted uranium foils. The relative power distribution was also measured by the gamma-scanning method. Sample reactivity worths of $^{239}Pu$ and $^{10}B$ were measured using a calibrated control rod by the substitution method. The natural

$B_4C$ control rod worth was measured by the modified source multiplication emthod. As a result, a flattening of radial distribution was observed for these parameters, compared with the homogeneous core.

The experiments were analyzed using an anisotropic theory code in XYZ geometry with the HENDL-2 library. Transport and mesh-size corrections were applied to the results. The C/E for $k_{eff}$ was 1.008. The calculated results of the reaction rate and power distributions agreed well with the measured ones in the core, but underestimated them in the vicinity of the internal blanket. As for the $B_4C$ control rod worth and sample reactivity worths, the calculated results significantly underestimated the measured ones by 10 - 20 %.

## 3. Physics Programs Using "JOYO"

The feedback reactivity of fast power reactor is composed of reactivities due to the coolant expansion, the axial expansion of fuel, the axial expansion of control rod drive mechanism, the radial expansion of core support plate, the bowing of subassemblies, the Doppler effect and so on. Each component of the feedback reactivity is desired to be measured separately from other components. The feedback reactivity due to the coolant expansion, for example, can be measured by changing the coolant flow rate under the condition that the power level and the inlet coolant temperature are kept constant. It is considered and planned to measure the component-wise feedback reactivities of JOYO in the near future.

It is considered to measure the decay heat of subassembly discharged from JOYO. A special equipment is being manufactured, and will be installed in the spent fuel storage pool of JOYO. The decay heat is expected to be measured as the function of cooling time.

In order to investigate the fuel irradiation performance of axial heterogeneous fuel pins, it is planned to fabricate some number of axial heterogeneous fuel pins to charge them in the test subassembly of JOYO.

## 4. Method Developments

### (1) Two-Dimensional Effect in Cell Calculations

The three 1-D models and the 2-D model were applied to the plate cell calculations of the ZPPR-9 and Zebra CADENZA cores. In the ZPPR-9 core calculation, the difference between the C/E values of $k_{eff}$ for the individual models was within 0.2 %. In the analysis of the Zebra CADENZA core, the discrepancy between the C/E values for the pin and plate cores was 0.3 ~ 0.5 % for the 1-D cell models. This discrepancy was reduced to 0.2 % by the use of the 2-D cell model, and the usefulness of the 2-D model was illustrated.

### (2) Unified Theory in Cell Homogenization

A unified theory for homogenizing a cell has been established. A general formula for cell-averaged cross sections and anisotropic diffusion coefficients is derived, which can be applied to any cell in a core. For infinite uniform lattices, the anisotropic diffusion coefficient is reduced to the Benoist's one, and the cross section is reduced to the commonly used flux-weighted one. For heterogeneous lattices, the anisotropic diffusion coefficient is reduced to a new one, and the cross section is reduced to the Rowland's one.

### (3) Transport Corrections by 3D $S_N$ Code

The transport corrections to $k_{eff}$, control rod worth, power distribution and neutron spectrum in fast critical assemblies have been evaluated using the 3D $S_N$ code TRITAC. The results were compared with those from 2D transport calculations with a RZ model and also with a combination model of XY, RZ and R. The combination model predicted the transport corrections in good accuracy.

(4) 3D Nodal Transport

Since the 3D $S_N$ code TRITAC needs a lot of time for the analysis of large fast reactor core, a nodal transport code TRITON was developed. The computing time was reduced by about 50 %, compared with TRITAC with the same accuracy. The approximation for the spatial expansion of intra-node flux and transverse leakage was investigated by comparing the calculated results with the fine mesh calculation.

## FUSION NEUTRONICS

A major half of the activities in this area is joint experimental works of university faculties by using the intense 14 MeV neutron source, Oktavian, at Osaka University.

Effects of lead neutron multiplier for tritium breeding in Li-Pb assemblies of 120 cm dia. sphere have been measured in order to present experiment, four kinds of measurements have been carrried out: neutron source characterization, tritium production rates, foil activation rates and leakage neutron spectra. Calculations using neutron transport codes such as ANISN, MCNP and NITRAN have been also performed.

The Lithium sphere of 120 cm dia. with inner cavity of 40 cm dia. was used, which is reported elsewhere. Lead shell of 40 cm o.d. and thickness 5 or 10 cm was inserted into the cavity in each run. Present experimental methos are similar to the reported previously. It is found that the tritium breeding ratio is no more than the value 1 in spherical assembly compsed of 40 cm thick natural Lithium with 5 or 10 cm thick lead neutron multiplier. Graphite reflector should be used in order to overcome the barrier.

From recent integral experimens and their analyses, it can be said the tritium breeding ratio in a pure Lithium blanket will be predicted within few % uncertainty by JENDL-3PR2 plus transport calculations. However, discrepancies between experiment and calculation are considerably large for reflected blankets and neutron multipliers. Improvements on nuclear data should be stressed for secondary neutron emission data, e.g., DDX and (n,2n) cross sections.

Another major half is the blanket integral experiments and their analyses by using FNS, the 14 MeV neutron source at JAERI with the collaboration of USDOE. Major efforts in this experimental program have been put on the tirtium production rate distributions in simulated $Li_2O$ blanket modules. The Phase I of the program was completed in the spring of 1986, in which the research was made on the effects of neutron multiplier and first wall layers on the tirtium production rates. Intercomparison studies on benchmark problems were carried out between JAERI and the U.S. to examine the data and methods used for the Phase I analyses.

As another activities at FNS, cross section measurements around 14 MeV have being carried out in systematic way on the activation of various structural materials for fusion reactors.

SHIELDING

Neutron streaming was measured through a long and narrow annular slit at a fast neutron source reactor YAYOI and then the measurements were analysed by DOT3.5 calculations. Even the calculations using a forward biased S418 did not agree the measured attenuating curve at a distance far from the entrance of the slit. So an analytic expression was presented and it was shown that the expression was in agreement with the measurement.

JAERI has already performed a program on reactor decommissioning technology development under the contract with the Science and

Technology Agnecy since 1981, and then participated in the Coordinated Research Program on Decommissioning and Decontamination under the Research Agreement between JAERI and IAEA. The contents are on the evaluation of radioactive inventory in LWR, and the research and development of decontamination of LWR.

The U.S./Japan joint program on fast reactor shielding research, designated "JASPER", are proceeding. As the first item of several experiments, attenuation experiments have been done at the TSF facility in ORNL, to investigate neutron penetration through benchmack and representative mockups of the radial shield designs for LMFBR. The neutron attenuation in various multilayer configurations, consisting of steel, sodium, graphite and boron carbide, were measured. Measurements were made using traditional TSF Bonner ball detectors, NE-213 and Benjamin neutron spectrometers and high-sensitivity $^3$He Bonner ball detectors supplied by the University of Tokyo.

In order to reduce the error and computer time of Sn calculations, an albedo-Sn method has been presented. It takes account of an albedo boundary treating the energy-and-angle dependent double-differential albedos, which are calculated using one-dimensional transport theory. A fundamental formula was given which expresses the albedo boundary source and uses the flux convergence acceleration with pointwise rebalancing technique. The test calculations were performed for the model of a sodium region surrounded with concrete shield. The results showed that the presented albedo-Sn method can calculate neutron fluxes near the albedo boundary with a good accuracy and less computer time. Discussion is also about the overlapping width required for Sn calculation using the bootstrap method.

The skyshine dose distributions of neutron and secondary gamma rays were calculated systematically using a Monte Carlo method for distances up to 2 km from the source. The energy of source neutrons ranges between thermal and 400 MeV. Their emission angle between 0 and 90 deg from the vertical was expressed using a distribution of the direction cosine with five equal intervals. The calculated dose distributions D(r) were fitted to the presented expression $D(r) = Q \exp(-r/\lambda)/r$ where the values of Q and $\lambda$ are slowly varying functions of energy. This analytic formula was applied to the benchmark problems of neutron skyshine from fission, fusion and accelerator facilities, and then it was shown that good agreements were achieved.

Evaluation of the experiments for fast neutron penetration through graphite was made with a continuous energy Monte Carlo code MCNP. Reaction rates and energy spectra obtained with the MCNP were compared with the calculated values by McBEND and the measured ones. It was shown that the penetration calculations with the MCNP is valid and that the MCNP calculations using the weight-window method is well applicable to problems of neutron penetration through graphite up to 70 cm in depth.

Neutron streaming analyses were carried out for the Heliotron-H reactor. Due to the helical geometry of the system, 14 MeV neutron streaming exists inherent in the reactor: Source neutrons generated from the plasma region leak at the coil shield surface in a direct line-of-sight manner and stream through blanket regions without collisions. After an appropriate coordinate transformation from the helical geometry, an analysis of neutron streaming succeeded using a method coupling three-dimensional Monte Carlo code and two-dimensional discrete ordinates transport code. The results showed that 40 ~ 50 % of the streaming neutrons contribute to each radiation response parameter of the helical magnet and that this fact strongly suggests the need of taking account of the contribution from streaming neutrons in order to provide an adequate shielding against the helical magnet.

Shielding calculations have been presented for design of cold and thermal neutron guides which are to be installed in a new research reactor. The calculational procedures use the Young-Koppel model for liquid hydrogen scattering kernel, the ANISN code for cold neutron source spectra and shielding of neutron guides, and the MORSE code for neutron transport through neutron guides.

## CRITICALITY SAFETY

As for the safety study for nuclear fuel cycle facilities, the yearly schedule of the government has been established and the criticality safety study is implemented in the schedule. Based on this schedule, study for the criticality problem outside reactor has been activated. The schedule mainly consists of the Japan-US joint experimental program and NUCEF program. The Japan-US joint program based on the contract between PNL and DOE is performed at PNL. NUCEF program is one of JAERI's projects, under which the safety analysis report for licensing is being prepared. Parallel to the report, the preparation has been continued for the criticality calculation code by which a number of benchmark calculations have been done for the validation.

In the experimental aspect, interest is concentrated in the measurement of highly sub-critical system. Examinations of the pulsed method and the Mihalczo method are performed at the piles of various geometries.

## NATIONAL PROGRAMS

### JOYO

Following the 5th annual inspection, the 8th duty cycle operation started on Dec. 2, 1985. Many kinds of tests utilizing the Instrumented Test Assembly (INTA) which was newly installed in the core region were carried out. At the end of the 9th duty cycle operation, the natural circulation test from the reactor power 75 MWt was conducted. The cumulative reactor power is 1,853,000 MWH at the end of the 11th duty cycle operation which was completed on Aug. 20, 1986.

Besides the operation works, the data supply to the Centralized Reliability Data Organization (CREDO) of U.S.A. based on the agreement between DOE and PNC, and an application for the licensing to extend the fuel maximum burn-up from 50,000 to 75,000 MWD/t and to increase the uranium enrichment of the MOX fuel from 12 to 18 wt%, were conducted.

MONJU

The MONJU site construction was initiated in October 1985 and is ongoing. The plant excavation was completed at the end of January 1986. In February, the preparation work for the basemat construction was started. The licensing inspection of basemat rock surface was completed in May, and the lower step mat concrete was placed in the reactor building at the end of June.

Barge shipments of the containment vessel sub-assemblies to the site were started in April. The C/V sections were then transported from the temporary unloading area to the C/V site shop and welded into C/V construction assemblies. The erection of the containment vessel was begun on July 1st. The construction work is now approximately 20 percent complete.

Demonstration Fast Breeder Reactor (DFBR)

Utilities have been carrying out design rationalization study both on loop and tank type plants. This design study has been conducted under the initiative of FBR Project Office of the Feberation of Electric Power companies, the role of which was recently succeeded to the Japan Atomic Power Company (JAPC).

Power Reactor and Nuclear Fuel Development Corporation (PNC) is playing a role of consultation and giving suggestions to the utilities' design study, especially in the areas of core, shielding, fuel, high temperature structural components and safety. PNC also started the large scale reactor design study to develop the key technologies for FBR.

The utilities, including the ten private electric power companies, had carried out the conceptual design studies by 1983. The study consisted of three phases. Regarding the study of the loop-type reactor, key concepts of the design were selected in the phase I and the design was reviewed mainly from the standpoint of operation and maintenance in the phase II. Design specifications were studied in the phase III on the basis of the further design of the total system and components. Regarding the study of the pool-type reactor, a preliminary concept was defined in the phase I studying the preceding plants' designs, and it was reviewed in the phase II mainly from the standpoint of seismic characteristics. The key subsystems of the pool-type reactor were designed in the phase III in parallel with model tests studied by the Central Research Institute of Electric Power Industries (CRIEPI).

Presently, the utilities are carrying out a design rationalization study in order to survey the feasibility of construction cost reduction and to prepare data for the determination of reactor specifications.

PNC is proceeding with a feasibility study for the scale up of main parts of a plant based on the design, construction and experiences of "JOYO" and "MONJU", aiming at the power generation cost reduction, as well as banking and arrangement of technical data and informations concerning above mentioned activities for their convenient utilization for the specification determination.

FUGEN

The 5th annual inspection and the 8th refueling were carried out from July to December 1985. At the 8th refueling, 8 $UO_2$ and 24

MOX fuel assemblies including 3 demonstration MOX fuel assemblies, were charged. The demonstration fuel assembly consists of 36 fuel pins, while the standard fuel assembly for "FUGEN" consists of 28 fuel pins, and is irradiated in "FUGEN" in order to develop the fuel of "ATR" Demonstration Plant". The 9th refueling was carried out in July 1986 and 20 $UO_2$ and 8 MOX fuel assemblies were charged. At present, "FUGEN" has continued stable full power operation.

Until now, 220 $UO_2$ and 196 MOX fuel assemblies have been discharged for refueling. The maximum burn-up is 17,900 MWD/t for $UO_2$ fuel and 18,200 MWD/t for MOX fuel, and no leaking fuel has been found for more than 1,700 effective full power days of operation up to August 1986.

ATR Demonstration Plant

The construction program of the 600 MWe ATR Demonstration Plant has started with the decision given by Japan AEC in 1982, that EPDC (Electric Power Development Company) be responsible, with cooperation of PNC and the electric power utilities, for the construction and operation of the plant, with PNC undertaking R & Ds and MOX fuel fabrication for the plant.

Since 1982 EPDC is finalizing the design of ATR Demonstration Plant and preparing for the application fot the construction permit. The capacity of the plant was already fixed to be 606 MWe. EPDC is making another efforts regarding site matters such as consensus on site acquisition. According to the current schedule for the project, the construction is expected to start in 1989, and the commercial operation in 1995.

# Report on the Reactor Physics Activities in the Netherlands in the period September 1985 - September 1986

compiled by R.J. Heijboer (ECN)

## 1. Reactor Physics at the Netherlands Energy Research Foundation ECN at Petten

The activities were mainly in the following fields:
1. Evaluation, development of methods, calculation and testing of neutron cross sections, especially for fission products, for fusion reactor materials and for use in calculations for the High Flux Reactor at Petten (HFR).
2. Neutron spectrum calculations for certain irradiation positions in the HFR.
3. Neutronics calculations for the blankets of NET and JET. 4. Development of methods and systems for noise measurements and analysis. 5. A continuous activity in the field of neutron metrology. 6. Shielding calculations for the project of storage and disposal of High Level Waste in a salt dome.

### 1.1. Fission product nuclear data (H. Gruppelaar)

An extensive report with the conclusions of integral-data tests for more than 50 nuclides of the JEF-1 fission-product data library has been issued [1]. This work was performed in collaboration with CEA Cadarache. One important observation to be found in the report is that the inelastic-scattering cross sections for many nuclides are too low in the existing evaluations, due to the neglect of direct-collective excitations [2].

Work is still in progress to correct the inelastic-scattering cross sections for the Ru and Pd isotopes.

For about 50 important fission-product isotopes the thermal and resonance regions of the JEF-1 file have been inspected. It was concluded that insertion of more recent resolved-resonance parameters would improve the thermal and resonance data of the evaluations. This work is in good progress and is performed at the NEA Data Bank in collaboration with ECN.

The new RCN-4 evaluations have now been completed for Ru-101, Ru-102, I-129, work is still in progress for Pd 107. For the derivation of lumped fission product cross sections an existing code had to be rewritten; it also can take account of leakage of gaseous fission products from the fuel towards the plenum.

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### 1.2. Activation file for fusion reactors (H. Gruppelaar)

Under contract with JRC-Ispra work was done to obtain a new data base for nuclear activation and transmutation calculations. As starting point the US data file REAC was used. This file was then extended to obtain a very complete data base for all stable (non-fissile) nuclides and nuclides with a half-life greater than 1 day. The entire data set has been checked at 14.5 MeV. For a large number of nuclides the data have been normalized, either on experimental values or on values derived from systematics. The branching ratios for isomeric-state production were corrected where necessary.

A follow-up project to further improve the quality of the file after assessment of important reactions that produce long-lived radioactivity in fusion-reactor materials is being defined.

### 1.3. The European Fusion File (EFF) (H. Gruppelaar)

The EFF is a nuclear-data file to be used in neutron transport calculations for fusion-reactor blankets. This project is part of the European Fusion Technology Programme of the European Community, to which various laboratories contribute. File management and maintenance is performed at ECN. Some features of the file were mentioned in the Activities Report of last year. The EFF-1 file has recently been distributed among the European users. Recent reports on its status are given in Refs. [1,2]. Work is in progress to obtain a second version of the file: EFF-2.

An existing evaluation for lead was revised at ECN. For this material the EFF-1 file now contains continuum double-differential cross sections in the new MF6 format of ENDF-VI. The code GROUPXS [3] for the calculation of multi-group transfer matrices in the new format was designed and developed at ECN. It includes the transformation of Legendre coefficients from the centre-of-mass to the laboratory system for continuum reactions. Extensive test calculations have been performed, to check this part of the code, by means of comparison with simple analytical cases and a recently developed fast series expansion method. In addition, comparisons were made with results obtained by Dr. O. Bersillon (CEA, Saclay). From these tests it was concluded that quite reliable results are obtained with the GROUPXS code.

For all materials in EFF-1 multi-group transfer matrices were calculated by various European laboratories with support from the NEA Data Bank. The results were stored in the so-called GEFF-1 library which is in the VITAMIN-J format, that can be condensed to VITAMIN-C and -E and many other standard group structures. The GEFF-2 file needs extensive testing before it can generally be used for routine calculations. This work is in progress.

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[3] H. Gruppelaar, D. Nierop and J.M. Akkermans, Processing of double-differential cross sections in the new ENDF-VI format - GROUPXS code description and users' manual *, ECN-182, April 1986, Netherlands Energy Research Foundation ECN, Petten.

## 1.4. Neutronics for fusion reactors (K.A. Verschuur)

The neutronic calculations for JET and for NET with the code system FURNACE [1] have been continued. Originally the activation of the foils used for the neutron diagnostics at JET (KN2) was calculated for a simplified torus model in which the TF-coils are assumed to cover the whole torus surface [2]. Now these calculations have been repeated using a 3D model of a sector of JET, so that the wedge shaped structure regions containing the view ports are taken into account [3,4]. Compared to the original calculations the neutron spectra at the irradiation positions are a bit harder, due to the presence of the structure regions, so that the calculated foil activities become somewhat higher. On the other hand the calculated thermal fluxes inside the torus are a factor of ten lower.

As mentioned in the last National Activities Report neutronic calculations were performed on the Li17Pb83 blanket in the NET-II single null torus geometry, to obtain the heat distribution and the effective tritium breeding ratio. The same calculations now have been performed for the new double null torus design NET-DN. Due to the increased neutron losses through the divertors, compared with single null torus concept NET-II (14%->20%), but also due to the changed blanket design, that contains relatively less breeder material, the calculated breeding ratio has become lower, i.e. 0.75 as compared to 1.01 for NET-II [5].

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## 2. Reactor physics at the Interuniversity Reactor Institute (Delft)

(H. van Dam)

In the report period the analysis of a BWR with natural circulation was continued with emphasis on measurement of two-phase flow velocity variations with noise methods. Both neutron and gamma detectors were applied [1,2] and a method was developed for fast measurement of incore coolant velocities [3]. Presently this method enables to determine two phase flow velocity with a statistical accuracy of about 2% within 5 seconds. This gives the possibility to measure velocity changes during operational transients (e.e. power and/or pressure changes). Experiments with the power reactor revealed a non-minimum phase behaviour in the power response to a pressure change: the short term pressure coefficient is positive like in forced circulation BWR's but the long term response (after 20 to 30 seconds) is just the opposite. This peculiar behaviour is due to the response of the natural circulation to pressure changes and improves the load following capacity of the reactor.

In the NIOBE project (Noise Investigations of Boiling Effects) much effort was devoted to the interpretation of noise signals of out-of-stream thermocouples and also to the implementation of gamma densitometry for two-phase flow. For the incore measurements a new boiling loop was designed that will be installed in the last quarter of 1986. Main features of this loop compared to the previous one are a higher power and independent flow control in the boiling channels.

Extensive research was focussed on synthesizing reactor noise signals with a digital reactor simulator [4,5]. The aim of this project is to produce artificial but realistic noise signals simulating a normally operating reactor and a reactor with anomalous behaviour. System parameters for normal and abnormal operation were selected and a set of signals was recorded for the benchmark exercise in the framework of the 5th Specialists' Meeting on Reactor Noise in 1987. This exercise aims at comparison of the available methods for noise signal analysis and of the merits of methods for anomaly detection.

Of the smaller projects two deserve to be mentioned. One is concerned with on-line subcritical reactivity determination in a research reactor; it is based on a method for determining the strength of the external neutron source by analyzing the power response to a subcritical reactivity step. The other project was devoted to an assessment, by using transport calculations, of the neutron dose received by a patient during in-vivo determination of cadmium in kidneys, performed by neutron irradiation with a Cf-252 source.


[1] H. van Dam, E.B.J. Kleiss, Measurements and analysis of neutron and gamma noise in BWR's Proceedings IAEA-NPPCI, Specialists' meeting on New instrumentation of water cooled reactors, Dresden (DDR) 23-25 April, 1985, 160-171.

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3. Reactor physics at KEMA (W.J. Oosterkamp)

A 3D BWR core simulator using transmission kernels based on age theory and 14x14x12 nodes has been validated against data from a small natural circulation BWR. The agreement with data on La-140 scanning is of the same magnitude 2-3% as the accuracy of the measurements. For the PWR in the Netherlands this approach yields the same agreement.

Noise techniques have been used to assess the stability of the BWR with natural circulation. The techniques have been extended to measure in-core and in-vessel velocities. A number of discrepancies remain, however, in these measurements.

An in-vessel waterlevel gage using heated thermocouples has been installed in the BWR. The results confirm the accuracy of the conventional instrumentation.

NORWAY

Institute for energy technology

Kjeller, Norway
September 10th 1986

***STATUS REPORT TO NEACRP***
***(1985 - 1986)***

REACTOR PHYSICS ACTIVITIES IN NORWAY
NOVEMBER 1985 - AUGUST 1986

Compiled by: T. Skarðhamar
Institute for energy technology
Box 40, N-2007 Kjeller, Norway

FMS CODE SYSTEM AND CORE SURVEILLANCE

The effort in reactor physics is continuing at the same level as reported last year. At Kjeller the activities are concentrated on the maintenance and further development of the LWR code package FMS. The responsibility for this is provided by the international consulting company Scandpower A/S. At the OECD Halden Reactor Project the activities on reactor physics are focused mainly on the continued development of the core surveillance system SCORPIO. The SCORPIO system has been supplied to the three PWR units at the Ringhals nuclear power plant in Sweden.

FMS (Fuel Management System) is a modular code system for light water reactor calculations. The basic modules of FMS are:

- RECORD, 2-dimensional lattice physics code, the data generation unit of FMS
- PRESTO, 3-dimensional core simulator
- RAMONA, codes for transient analysis in 1- to 3-dimensions.

The system is in wide use for fuel management work and plant operation support by power utilities and other organisations in Europe, USA, Mexico and the Far East.

Much effort during the past year has been made on the development of Core Master PRESTO for on-line applications at KKL, Leibstadt, Switzerland. An associated code to RECORD, called ECLIPSE, has also been developed during the reporting period. This code performs burnable absorber calculations and generates necessary effective cross sections for BA cells as input to RECORD. ECLIPSE is intended to replace the previous and more cumbersome THERMOS- GADPOL system.

See otherwise Status Report to NEACRP of last year, and also those of previous years.

SPAIN

MINISTERIO DE INDUSTRIA Y ENERGIA
JUNTA DE ENERGIA NUCLEAR

## REACTOR PHYSICS ACTIVITIES IN SPAIN

### REACTOR PHYSICS AND TRANSPORT THEORY AT JEN. 1985-1986

### NEUTRONIC CALCULATIONS OF PWR CORES (JEN)

contributed by C. AHNERT

The JEN-UPM code package (1, 2) has been applied to the core calculations of several burnup cycles (1, 2, 3 and 4) for a 900 MWe spanish nuclear power plant, and the validation of the contained methods and theories is carried out.

Comparison with both design and operating data have been done for nominal and power following conditions, obtaining a good agreement in the different parameters analized, and always with a bias within the design criterium, and of course within the safety criterium.

The parameters that have been analized were:

- Critical Boron Concentration at HZP and HFP versus burnup.

- Power densities and burnup spatial distributions in the XY transversal plane and in the axial direction.

- Reactivity coefficients at BOC and EOC; for the moderator temperature, power level, boron concentration and Doppler effect.

- Control Rods Bank Worth.

- Xenon concentration transients at BOC.

Several parameters have been determined by the both core calculations methods available:

- LOLA System, nodal one-group theory, with its analitical transport factors and albedoes calculation.

- CARMEN System, the fine-mesh two group finite-diference diffusion theory.

In the future, pin by pin core calculations by the diffusion theory method (CARMEN) will be carried out, by the implementation of this code system to an IBM-3090/150 computer.

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## COMPUTATIONAL METHODS FOR CRITICALITY AND RADIATION TRANSPORT

contributed by J. Peña (JEN)

The Neutronic and Shielding Group is considering the adoption of the code MCNP version 3 (1) as an additional tool to the conventional discrete ordinates codes for criticality and radiation transport analysis. At this respect, the validation phase of MCNP has been initiated.

The potential of this Monte Carlo code to perform both criticality and radiation transport analysis is being investigated. Our preliminary calculations have shown that the code fulfils the requirements which make it a powerful tool for radiation transport studies, nevertheless its use for criticality analysis is not yet clear. The complexity of the geometry description (pin-to-pin) and the amount of stoge and computing time required to perform subcriticality studies for typical multi-assembly systems (i.e. spent fuel storage studies) keep MCNP code in disadvantage versus other Monte Carlo codes which use space-energy collapsed cross-sections.

### Criticality

Several benchmarks involving critical fuel assembly arrays (2) are being analyzed. In order to easy the lattice geometry input the utility code GEMO (3) is being developed.

The results of these criticality studies are not yet available.

### Radiation Transport

As far as shielding is concerned the MCNP has been used to determine doses at 3 feet from cask surface of a conceptual shipping cask design (4). The results have been compared to those calculated with MORSE (5). A relative good agreement was observed.

Much more intense use of MCNP has been made in the determination of thermal neutron fluxes at different positions of the thermal column of the JEN-1 experimental reactor (6). Additionally, gamma and neutron heat depositions have been calculated at a Pb plate facing the north-side surface of the core. Thermal neutron fluxes were obtained using a directional biased fission source toward the detector positions. The results agreed fairly well with those reported by measurements.

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# CROSS SECTION PROCESSING CODE VERIFICATION PROJECT

J. Sancho, S. Mās (University of Valencia)

## Introduction

One of the objectives of the Nuclear Engineering Department of the University of Valencia, consists in arranging and evaluating a system for the management of libraries (AMPX-II) to elaborate working libraries, wich are inputs of a variety of codes, for criticity as well as for shielding.

This has been a difficult task up to now. On one hand, the AMPX-II codes system has been developed for an IBM machine so it had to be adappted to the SPERRY-UNIVAC machine available in our University (4). Furthermore, some discrepancies have been discovered between our results and the corresponding values so far admitted. Because of these reasons we have begun a validity process of the AMPX-II system.

At the same time, the International Atomic Energy Agency (IAEA) has begun a project for the intercomparison of neutron cross section processing codes in order to verify their ability to reproduce numerical results. The first round of this project is being coordinated by D.E. Cullen, Director of IAEA Nuclear Data Section. For this round of comparisons all the participants were asked to use the ENDF/B-V Dosimetry Library, to calculate flat weighted, 0 Kelvin, unshielded cross sections using the SAND-II 620 group structure.

Ph.D. J. Sancho García, member of this Department, developed this work in his Ph.D. thesis and he got these cross sections with errors smaller than 0.5% (3). 37 users have participated in this first round with 12 different systems, including AMPX.

## Objectives

Now, we are participating in the second round of the IAEA Cross Section Processing Code Verification Project, which is being coordinated by P. Vertes, member of the Hungarian Academy of Sciences.

The objectives of this project are: to test the accuracy of processing codes, to understand and eliminate the sources of discrepancies, to arrive at the point where we have a number of cross sections processing codes which can be used as safely as possible as "black boxes", with no problem.

For the second round, the Nuclear Data Section of the IAEA has made available evaluated data in the ENDF/B format for the following materials, H, C, Fe, Th-232, U-235, U-238, Np-237 and Pu-239. It has to be used the ABBN 26 group structure and flat weighting function and the following constants have to be obtained:

1. "Cold" (not Doppler broadned) self-shielded (i.e. $1/\sigma_T + \sigma_0$ weighted) group averaged cross sections and/or self-shielding factors for each of the processes: total, elastic, $(n,\gamma)$, $(n,f)$. The self-shielding conditions are shown on Table I.

TABLE I

| MATERIAL | BONDARENKO FACTOR | | | |
|---|---|---|---|---|
| Fe (MAT=1192) | 10 | 100 | 1000 | 10000 |
| Th-232 (MAT=1296) | 10 | 50 | 100 | 200 |
| U-235 (MAT=1261) | 10 | 100 | 1000 | 10000 |
| U-238 (MAT=1262) | 10 | 50 | 100 | 200 |
| Np-237 (MAT=1263) | 10 | 100 | 1000 | 10000 |
| Pu-239 (MAT=1264) | 10 | 100 | 1000 | 10000 |

2. Elastic scattering matrices with up to P3 Legendre fits without self-shielding for the materials showed on Table I plus H and C.

3. Group-averaged elastic-down constants.

4. Inelastic scattering matrices for materials in the Table I.

Working Process

The AMPX system was developed at Oak Ridge National Laboratory (Tennessee-USA) during the last years of the sixties and since then it is bound to a constant revision process. Our Department's version dates from the end of 81 and it was implanted in 83. The system consists of 7 basic modules and 23 auxiliary ones, which is adequate for the initial treatment of nuclear evaluated data in ENDF/B format and makes the use of later data easier.

One of the main codes of the system is XLACS-2 (1), an improved version of the old XLACS, which calculates weighted multigroup neutron cross sections from ENDF/B data. The module is designed to produce full-energy-range neutron cross section libraries. Energy group structure and expansion orders used to represent differential cross sections can be arbitrarily specified by the user. The multigroup data produced by XLACS-2 are produced in AMPX master library format.

XLACS-2 has been used during this second round to calculate multigroup cross sections and scattering matrices (elastic and inelastic).

All the materials supplied by the Nuclear Data Section, except H and C, are resonance nuclides and they had been processed by the auxiliar code NPTXS previously (6). This code calculates resonance cross sections going out from ENDF/B data and it produces a point-wise library which serves as input of XLACS-2.

XLACS-2 Module

There have been done the necessary modifications to enable the XLACS-2 code to read data of the ENDF/B library in its versions IV and V.

As related above in our objectives, one of them is to obtain self-shielded multigroup cross sections with $1/\sigma_T+\sigma_0$ weighted, while elastic scattering matrices have to be obtained without self-shielding. The different Bondarenko values ($\sigma_0$) to use are showed on Table I, so that for each of the six resonance nuclides it was necessary to run the program on four occasions, one for each value of $\sigma_0$.

On the other hand the scattering matrices are obtained with flat weighted function and without self-shielding - infinite dilution, $\sigma_0$ = 1.0E+8 - for each of the 8 materials focused in this econd round of the Verification Project.

The results obtained are 32 AMPX master libraries which are distributed as shown on Table II. Each of these libraries contains data related to multigroup cross sections for elastic scattering process, capture, fission and total, as well as elastic scattering matrices with up to P3 Legendre fits and inelastic scattering matrices.

TABLE II

| MATERIAL | $\sigma_0$ Infinite dilution | $\sigma_0$ $1/\sigma_T+\sigma_0$ weighted | | | | AMPX MASTER LIBRARIES |
|---|---|---|---|---|---|---|
| Fe | 1.0E+08 | 10 | 100 | 1000 | 10000 | 5 |
| Th-232 | 1.0E+08 | 10 | 50 | 100 | 200 | 5 |
| U-235 | 1.0E+08 | 10 | 100 | 1000 | 10000 | 5 |
| U-238 | 1.0E+08 | 10 | 50 | 100 | 200 | 5 |
| Np-237 | 1.0E+08 | 10 | 100 | 1000 | 10000 | 5 |
| Pu-239 | 1.0E+08 | 10 | 100 | 1000 | 10000 | 5 |
| H | 1.0E+08 | | | | | 1 |
| C | 1.0E+08 | | | | | 1 |
| | | | | | TOTAL | 32 |

DIAL Module. Modified ENDF/B format

The growing interest in the verification of nuclear data processing codes by the intercomparison of calculated group constants indicates the needs to establish an adequate computer readable exchange format for the most widely used types of group averaged nuclear parameters.

Since the ENDF/B format is presently used for the international exchange of evaluated data and has already been successfully used for the intercomparison of infinite diluted multigroup cross sections, it is somewhat natural to base a format for multigroup data on the ENDF/B format by introducing the minimum number of modifications to allow the format to accomodate a variety of types multigroup data.

This has been the criterion of IAEA about the propositon of one format for all the participants in the second round of the project to exchange the results (5).

The types of data which will be represented in the multigroup ENDF/B format includes temperature and self-shielding dependent cross sections and parameters as well as group to group transfer matrices. Because of the similarity of these multigroup data to evaluated data, the ENDF/B formats for files 3 and 5 have been modified to accommodate these data.

Compactness and efficiency of storage, while important, are considered to be less important than simplicity and convenience of use.

The DIAL program (2), belonging to the AMPX system, has been used to edit the AMPX master libraries previously obtained through the XLACS-2 code.

It has been necessary to modify thorough the DIAL code, elimanating some original subroutines, correcting others and introducing new routines, to enable the code to translate the AMPX master libraries in "formatted libraries" in modified ENDF/B format, distinguishing automatically between evaluated data in ENDF/B format and multigroup data in modified format. This has been a hara and delicated task which has taken a great deal of time and effort.

The modified data have lately been sent to PhD. Vertes for their intercomparison in the Central Research Institute for Physics of Budapest, for whose results we are now waiting. In the probable case that there are discrepancies, it will be necessary to insist on new modifications untill an acceptable accuracy for the model is obtained.

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# NUCLEAR DATA PROCESSING IN THE RESOLVED RESONANCE REGION USING REICH-MOORE PARAMETERS

G. Verdú, J.L. Muñoz-Cobo, J. Sancho. (University of Valencia)

Beside the IAEA Verification Project we have taken part in a previons intercomparison about processing of resonance data with Reich-Moore parameters.

This method is nowadays considered as more accurate than the traditional Breit-Wigner one, at one level as well as at several ones, and particularly in the reproduction of fission cross sections. Furthermore, in the future versions of ENDF/B library (IV and following ones) exists the intention of including each time a bigger number of materials with evaluations based on Reich-Moore parameters.

For the time being, the IAEA Nuclear Data Section has Reich-Moore evaluations only for two materials: Pu-241 and Ni-60. Our Departament asked for these data and they have been processed to reproduce in detail the cross sections point by point (not in groups).

The results have been organized in ENDF/B format with File 3 structure and they have been sent to Vienna for their comparison.

The AMPX-2 system, as well as NJOY, haven´t got enough capacity to work out cross sections calculations parting from Reich-Moore evaluations. The process consisted in amplifying the capacity of the NPTXS code of AMPX-II, so that it can accept and process the Reich-Moore parameters following this method, and that it produces point-wise libraries of identical organization as the ones obtained based on Breit-Wigner parameters. For this pursuit, we have started from a group of subroutines -CSRM and similars- wich used the RESENDD code (Japanese version of RESENDD, more modern, accurate and wider, available at NEA). These subroutines have been introduced and adapted to the NPTXS sequence of calculations to obtain the desired result.

Finally, for the transmission of the results, we have produced a small program (LECTOR) to move the results from the point-wise library (binary records) to an ENDF/B format organization with card image.

We have obtained some discrepancies with the IAEA evaluation we have eliminated these discrepancies by making some modifications in the RESENDD code (1).

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# NUCLEAR DATA PROCESSING IN THE UNRESOLVED RESONANCE REGION

G. Verdú, J.L. Muñoz-Cobo, J. Sancho. (University of Valencia)

The purpose of this work is to present a test of the validity of the ENDF/B treatment of the unresolved resonance region for fissile elements like Pu-239, we use ENDF/B IV libraries.

The methodology of this work, is similar to that of Reference (1), but extended to include fissions. We have already proved that the methodology of R.B. Pérez et al.(1) (2) can be extended to fissile isotopes. To this end we have developped the UNRESOLVED and BUPAR programs. BUPAR computes the average resonance parameters which fit the average capture, and average fission cross section at infinite dilution. We have obtained that there is only one unique set which make this fit.

Now we have in progress the following steps:

i) Comparison of Bondarenko self-shielding factors in the resolved region computed by the resolved resonance methodology with Bondarenko factors using unresolved methodology.

ii) Comparison of the results at different dilution, using various interpolation schemes in order to choose the interpolation scheme in the unresolved region which best fit the results.

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## MONTECARLO NEUTRON AND PHOTON CALCULATIONS

G. Verdú, J.L. Muñoz-Cobo, J. Ródenas. (University of Valencia)

During the last year the MCNP and KENO codes have been implemented and tested in our UNIVAC/1100/70.

We have compared the results of MCNP with the results of KENO and DOT 4-2 for several cases. Severe discrepancies were observed for k in some particular cases, when KENO was runned with the Hansen-Roach Libraries. But these discrepancies dissapeared when the group cross-section were generated with AMPX-Modules, obtaining in this last case good agreement among the three codes.

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# INTERPRETATION OF NEUTRON NOISE DATA AT P.W.R. BY THE PARAMETRIC EXCITATION TECHNIQUE

R. Sanchis, G. Verdú and J.L. Muñoz-Cobo. (University of Valencia)

A model have been developped, to analyze the P.S.D. of in-core and ex-core detector at P.W.R. due to neutron noise. This model is based on the parametric excitation technique (1).

In order to make practical calculatios at P.W.R., we have modified the TASK-CODE (2), which solve the Bolzmann transport equation in the frequency domain in 1D geometries, and we have added the possibility to make adjoint calculations. This make possible to calculate the detector P.S.D., due to flux perturbations produced by fuel element vibrations, or any other reactor internal vibration (3).

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STUDSVIK ENERGITEKNIK AB

# REACTOR PHYSICS ACTIVITIES IN SWEDEN
compiled by STUDSVIK ENERGITEKNIK AB

## NUCLEAR POWER IN SWEDEN 1985/86

The 12 nuclear reactors are this year all operating at full power - actually during the last year their capacity has been raised by 4 % on the average and they are now producing about 50 % of the total electricitiy production. The present government declares itself strictly loyal to the 1980 Parliament decision to close down all nuclear plants not later than 2010. The Chernobyl accident, which caused a radioactive fallout over Sweden (giving radiation levels up to 100 times normal background in some parts of Middle and North Sweden), has abruptly broken the earlier trend of more positive public attitude towards nuclear power. There are various political groups which now demand an earlier phase-out of nuclear power in Sweden. The government has after the severe accident asked for updated safety reports on the nuclear plants, especially the two Barsebäck reactors located not far from Copenhagen will thoroughly be reanalyzed.

## 1 IMPROVED NEUTRON CROSS SECTION LIBRARIES
(Klas Jirlow)

The JEF-1 neutron data file has recently been adopted at STUDSVIK for the new group cross section libraries for thermal and fast reactor calculations, respectively. The neutron data libraries are being generated with the NJOY code system, which recently has been tested on the STUDSVIK Cyber 835 computer.

## 2 REACTOR PHYSICS CALCULATIONS ON TIGHT PWR LATTICES

(Erik Johansson)

Recent studies at STUDSVIK on tight PWR lattices mainly deal with the following two items:

1) Modification and test of the nuclear data library

2) Calculations of the void reactivity for power reactors

The library used at present (in the CASMO code) is a modification of our previous version, which mainly originated from ENDF/B-III. New data for the resonance absorption in U238, based on the evaluation by Tellier, have been introduced. Furthermore. the old data for U235 and Pu239 have been replaced by data from ENDF/B-IV. New data have also been introduced for Pu241, by use of JENDL-2. This nuclide is, however, still treated as a dilute resonance absorber. The new library gave a better $k_\infty$-calculation than the original one when applied in CASMO calculations on the first three cores in phase 1 of the PROTEUS experiments. The improvement was particularly large for the reactivity change on complete voiding of the lattice.

Void reactivity values for several types of tight PWR lattices with various moderator-to-fuel volume ratios and various mixed oxide fuels have been calculated with CASMO combined with the new library. Some values came out negative, others positive. The largest positive values were obtained for water-to-fuel volume ratio of about 0.5 in a lattice fueled with normal LWR discharge plutonium mixed with depleted uranium. In spite of the good outcome of the test against PROTEUS data, there might be large uncertainties in the void reactivity calculation in application of the method on tight PWR lattices. However, even after correction for such uncertainties the void reactivity might very well remain positive for the PWR case mentioned above. In addition this reactivity would become even more positive on repeated recycling of the plutonium in the tight lattices.

## 3 IN-CORE FUEL MANAGEMENT AT STUDSVIK
(Kim Ekberg)

The new CASMO-3 has now been released to about 10 utilities and other customers. An extensive benchmark study has been completed. Calculated results from CASMO-3 have been compared with experimental results from KRITZ, B&W criticals, TRX, ESADA and others. The agreement is excellent. A report on the CASMO-3 benchmark study will be issued shortly.

An operative version of SIMULATE-3 is now being tested at STUDSVIK. Extensive core-follow calculations are in progress on both PWR and BWR. The first releases to utilities will be made before the end of 1986.

## 4 THORIUM FUEL CYCLE CALCULATIONS
(Gunnar Andersson)

The CASMO code has been extended so it can treat thorium fuel lattices properly. The old data library, mainly based on ENDF/B-IV, has been revised in important areas e.g. fission spectra, thermal data for U232 and $\nu$-values using a wide spectrum of thorium test lattices including BNL $H_2O$- and $D_2O$-criticals, ZEEP $D_2O$-lattices, ANL $D_2O$-criticals and BAPL $H_2O$-lattices. In this way new ad hoc data sets have been selected. Results from test calculations with this new library compare very well both with available integral measurements and with the best foreign calculations.

# REACTOR PHYSICS ACTIVITIES IN SWITZERLAND

November 1985 to September 1986

P. Wydler

## 1 INTRODUCTION

After two narrowly defeated antinuclear initiatives in 1979 and 1984, the Swiss nuclear energy programme just started to gather some new momentum, when the Chernobyl accident with a thunderclap shattered all hopes for a wider public acceptance of the nuclear option. Shortly after the accident, a committee of politicians from various parties launched a third initiative calling for a ten-year moratorium, and an initiative to stop nuclear energy is currently being prepared by the Social Democrats. For the first time, public pressure has succeeded in forcing the Federal authorities to consider scenarios which point towards the phasing out of nuclear energy at the end of the useful life of the existing power plants.

This development inevitably means further delays for the Kaiseraugst nuclear power plant. Nevertheless, bids for this plant by Brown Boveri/Getsco (for a Leibstadt-type BWR) and Kraftwerk Union (for a Gösgen-type PWR or a Grundremmingen-type BWR) are currently being evaluated. To compensate for the delays in the nuclear construction programme, Swiss electricity companies have increased their share in French power plants, bringing the total imported electricity to the equivalent of that from a 1300 MWe unit.

It should be emphasized that the technical success of nuclear energy in Switzerland cannot be disputed. The five operating plants, which contribute 40% to the total electricity production, have an excellent safety record and a high availability (reflected in an average load factor of 84% in 1985). In this context, power increases at Gösgen and Leibstadt are worth mentioning: Large safety margins have allowed these plants to be uprated by 7% and 4%, respectively, giving an additional power of more than 100 MWe.

In spite of the current difficulties with nuclear energy, the Swiss industry and Swiss research centres have a continued interest in small heating reactors in the 10 to 50 MWth range. Proposals for three concepts, the (water cooled) Swiss Heating Reactor (SHR), the so-called GEYSER reactor, and the Gas-Cooled Heating Reactor (GHR), have now been made. Core physics and shielding calculations in support of these concepts have been performed at the Federal Institute for Reactor Research (EIR).

Two major experimental reactor physics programmes, the PROTEUS Light Water High Converter Reactor (LWHCR) programme of EIR and the LOTUS fusion blanket programme of the Federal Institute of Technology at Lausanne, have been continued at a similar level as in the previous year.

## 2 STUDY OF SMALL DISTRICT HEATING REACTORS

The SHR, a heating reactor based on BWR technology, has been described in previous activities reports. The main features of the concept are summarised in Ref.1. The design has been optimised using standard LWR computational tools, such as the BOXER/SILVER scheme.

The GEYSER concept, proposed by the Swiss Institute for Nuclear Research (SIN), uses the hydrostatic pressure of a water column above the reactor core to produce a saturation temperature of the cooling water of about 150°C. A particular feature is the absence of control rods and hence any potential for power transients associated with an inadvertent control rod withdrawal. The reactivity is controlled using burnable poisons and self-regulating boron poisoning in the primary circuit. Two core designs, similar to either the SHR or the TRIGA design, are being evaluated. The GEYSER concept makes extensive use of passive safety features, and the proposers therefore believe that the concept represents a break-through in the endeavour for an inherently safe water reactor.

The Gas-Cooled Heating Reactor, GHR, represents a joint effort of Swiss and German industrial concerns sharing a common interest in HTR technology. The GHR has a helium-cooled, pebble-bed core and incorporates many design features of the THTR-300 as, for instance, a prestressed concrete reactor vessel. An interesting detail is that the heat from the primary circuit is removed by means of the water-cooled "liner" cooling circuits. Like the other concepts, the

GHR has a high degree of inherent safety as a result of the very conservative rating of the components.

The GHR core was laid out by Brown Boveri/Hochtemperatur Reaktorbau (HRB), but EIR has contributed some shielding calculations. A particular problem concerned the neutron and photon dose computation at the base of the motor drive of the helium circulator, since the motor/circulator assembly is integral with the prestressed concrete reactor vessel. Whereas the neutron and photon doses below the circulator could be well calculated using deterministic codes, the evaluation of the streaming through the circulator necessitated an MCNP Monte Carlo approach.

The principal core parameters of the three different heating reactors are compared in Table I. For each concept, a project and a safety report has been produced, and on the basis of these reports the most promising concept will now be evaluated.

Table I: Principal Core Parameters for Heating Reactors.

| | SHR | GEYSER | GHR |
|---|---|---|---|
| Thermal power (MW) | 10 | 10 | 10 |
| Mean power density ($MW/m^3$) | 17 | 30 27 | 2 |
| Fuel | $UO_2$ (rods) | $UO_2$ or UZrH (rods) | $UO_2$ (spheres) |
| Enrichment (% U-235) | 4.5 | 20 7 | 20 |
| Coolant | $H_2O$ | $H_2O$ | He |
| Circulation | natural | natural | forced |
| Coolant pressure (bars) | 15 | 4.7 | 15 |
| Inlet temperature (°C) | 185 | 135 | 250 |
| Outlet temperature (°C) | 198 | 149 | 450 |
| Control absorber | B-rods | boric acid | B-rods |
| Burnable poisons | $Gd_2O_3$ | Er Gd | Hf,B |
| Mean fuel burnup (GWd/tHM) | 17 | 80 40 | 62 |
| Core residence time, LF=0.5 (yr) | 11 | 27 24 | 12 |
| Temperature coefficient ($10^{-5}$/K) | -2.5 | -6.4 -2.7 | -5 |
| Void coefficient ($10^{-3}\Delta k$/% void) | -2.3 | <0 | |

## 3 LWHCR PHYSICS EXPERIMENTS AND ANALYSES

First results from the PROTEUS-LWHCR Phase II experimental programme, which is currently being conducted under a joint co-operative agreement between EIR, Kraftwerk Union and KfK Karlsruhe, have now been published (Refs.2-4). These concern measurements and calculational comparisons for the $H_2O$-moderated, reference lattice (Core 7) with a fuel-to-moderator volumetric ratio of 2.07.

The improvements in the new experiments, relative to the earlier Phase I cores - e.g. the single-rod lattice, the larger test zone diameter, as well as the longer time available for experimentation - have resulted in a significant reduction of experimental errors. Thus, accuracies ($1\sigma$ values) of between $\pm 1.5$ to 2% have been attained for central reaction rate ratios ($\pm$ 2 to 3% earlier), while $k_\infty$ could be deduced to $\pm$ 0.5% ($\pm$ 1% earlier).

Calculational comparisons for the measured neutron balance components have been made using WIMS-D with both the standard and the 1981 data libraries. Within the respective experimental errors, there is good internal consistency between the results from Core 7 and the Phase I wet-lattice measurements. It has been indicated, however, that differences between the Phase II and the Phase I lattices - due to the larger fuel diameter, single-rod nature and more characteristic Pu isotopic composition in the former case - can affect the physics more significantly than, say, fuel enrichment alone. The Phase II results are thus proving to be not only more accurate, but also more representative of an LWHCR.

As regards the performance of the two different WIMS libraries, the conclusion drawn from analyses of the Phase I lattices has been reaffirmed: It appears that the adjustments made in the 1981 version do indeed represent significant improvements for LWHCR calculations under non-voided conditions.

Nevertheless, the applicability of a data library incorporating adjustments based on non-LWHCR integral tests remains questionable. For this reason, a new WIMS library based on JEF-1 data has been generated at EIR using the NJOY-WIMSR-WILMA processing route. Testing of this library, including numerical checks on the adequacy of the WIMS resonance treatment for LWHCR applications, is currently under way.

In an investigation of control rod materials in Core 7, the relative worths of a variety of central absorber rods (from natural and enriched $B_4C$, Ag-In-Cd, hafnium, gadolinium oxide and samarium oxide) were measured and evaluated. The results confirm the choice of $B_4C$ as appropriate control rod material for LWHCR spectral conditions. Comparison of calculated and experimental reactivity worth ratios (expressed relative to a $B_4C$-pellet rod in each case) yields satisfactory agreement for $B_4C$ powder and Ag-In-Cd. The relative worth of highly enriched $B_4C$ (93% B-10), on the other hand, is overestimated in the calculations by about 15%.

Measurements of neutron balance components and control rod worths for the dry (100% void) PROTEUS-LWHCR Phase II reference lattice (Core 8) have been in progress since March 1986 and are now nearing completion. Major emphasis has been placed on achieving a reliable determination of $k_\infty$ applying alternative experimental techniques, since the void coefficient is still considered to be the most important feasibility criterion for a given LWHCR design. Apart from the material-buckling approach which results in a larger uncertainty than in the case of moderated lattices, cell worth measurements have been carried out with alternative normalisation methods (e.g. Cf-252 source) being applied for evaluating the reactivity signal obtained.

Measurements simulating partial voidage in the reference lattice are planned as the next step, with Dowtherm to be used as moderator (Core 9).

## 4 LOTUS FUSION BLANKET PROGRAMME

Over the past twelve months, the LOTUS fusion blanket programme has focused on experiments with the Lithium Blanket Module (LBM). The module consists of 923 breeder rods, each housing 20 to 22 $Li_2O$ pellets in a thin-walled stainless steel tube, and was specifically designed for fusion neutron irradiations at the Tokamak Fusion Test Reactor (TFTR), when such irradiations become feasible. The LBM programme is run by the Princeton Plasma Physics Laboratory (PPPL) and, up to now, has been sponsored by the Electric Power Research Institute (EPRI).

The current LOTUS-LBM experiments serve as a benchmark for measurement and data processing techniques which will later be used in the TFTR experiments. In order to minimize effects due to neutron back-scattering from the shielding walls of the LOTUS facility, the neutronic quantities, tritium production rates (TPR's) and neutron spectra, are measured essentially along the central axis of the module.

The TPR is evaluated using $Li_2O$ diagnostic wafers in combination with Dierckx's method of liquid scintillation counting and, independently, using the thermoluminescent lithium fluoride self-irradiation technique. A third method, based on solid state nuclear track detection, is being developed and will allow (low intensity) TPR measurements at the backend of the LBM to be performed with short irradiation times. In addition, PPPL employs a thermal extraction method to determine the tritium bred in $Li_2O$ diagnostic pellets.

Information on the neutron spectrum is obtained from dosimetry measurements. Sets of dosimeter foil packages are placed between aluminium-clad $Li_2O$ pellets inside designated removable test rods. Some fourteen suitable dosimetry materials have been selected on the basis of the threshold energy of the reaction, the half-live, the availability, and the value of the activation cross section. The required irradiation times are typically of the order of five to ten hours. During the past months, several such irradiations were carried out and it is hoped that these measurements can be complemented with neutron spectrum measurements using a miniature NE-213 spectrometer, currently under development.

Because of the complex geometry of the experimental configuration, analyses are primarily performed using the MCNP Monte Carlo code, though two-dimensional deterministic codes, such as TRISM and DOT, are used to complement the Monte Carlo calculations. Various other laboratories (LANL, PPPL and EIR) are involved in the analyses.

Preliminary results show fairly good agreement between calculation and experiment. In some cases discrepancies appear to be significant, but more thorough analyses are necessary to assess whether these discrepancies can be attributed to deficiencies in the basic nuclear data. Further analyses will include sensitivity studies (by EIR and LANL) and three-way comparisons of neutron activation data, tritium assay data, and computer predictions using a "master comparison code".

A review of the LOTUS-LBM programme was presented at the Seventh Topical Meeting on the Technology of Fusion Energy, Reno, Nevada, 15-19 June, 1986, in a session specially dedicated to this programme.

## 5 DATA AND METHODS FOR FISSION AND FUSION REACTOR CALCULATIONS

At EIR the development of an NJOY-based scheme for all types of fission reactors, for fusion reactors, and for shielding applications, has been continued. In addition to the TRANSX-CTR/ONEDANT route for resonance shielding and group collapsing, cross section preparation options using the cell codes MICROX-2 and WIMS-D are now available. The scheme was used to generate three new, mainly JEF-1 based group libraries: a 69 group library for WIMS-D, a 174 group library in NMATXS format for shielding calculations and a coupled 187/48 neutron/photon group library for fusion applications.

The new WIMS-D library contains all nuclides needed in the analysis of the PROTEUS-LWHCR experiments. The library was generated for temperatures in the range 300 to 1200 K and background cross sections in the range 1 to $10^{10}$ barns. All reaction types were processed, but scattering was restricted to $P_1$. In the thermal energy range, coherent and incoherent scattering matrices were generated for hydrogen and carbon, bound in water and graphite, respectively. In contrast to the standard WIMS-D library, all important plutonium isotopes have resonance shielding (For the PROTEUS application this is particularly important since the physics of LWHCR lattices is significantly influenced by the plutonium isotopic composition). The conversion of the data to the WIMS-D library format was accomplished using the NJOY module WIMSR and the REFORM option of the WILMA code.

The 174 group library was used in a benchmark exercise in which JEF-1 predictions were checked against results from fast critical experiments. For each experiment a one-dimensional, spherical ONEDANT calculation in $P_3S_{16}$ approximation was performed. It was found that, in general, JEF-1 gives similar results as ENDF/B-V. Eigenvalues are well predicted (within 1%), whereas calculated and measured reaction rates deviate by as much as 10% (cf. Ref.5). The results are in agreement with earlier diffusion theory results reported by Takano and Sartori from the NEA Data Bank.

The coupled 187/48 group library contains 83 nuclides mainly from JEF-1, some nuclides (Be-9 and Li-7) being taken from a preliminary version of the European Fusion File (EFF). The library includes neutron data, different kinds of gamma production matrices, excitation reaction cross sections, and heating and damage data, but data are currently restricted to a single temperature of 296 K. The most important resonances were shielded using the Bondarenko method.

Condensed versions of the library with 46 or 74 neutron groups are available for applications requiring a smaller number of groups.

A useful addition to the EIR scheme is the SENSIT-2D code. Together with the above-mentioned 74 group library and the Los Alamos covariance and uncertainty library COVFILS-2, the code will allow sensitivity calculations to be performed. Such calculations are planned, for instance, in support of the LOTUS-LBM programme (cf. Section 4).

Development of transport codes concentrated on the improvement of the Los Alamos finite element code TRISM. The code uses triagonal spatial meshes and can cope with unusual geometries, such as the toroidal geometry typical for fusion reactors. In this context, an $S_2$ synthetic acceleration method was developed and tested, with the result that the inner iterative process could be accelerated considerably.

## 6 SINQ SPALLATION NEUTRON SOURCE

The spallation neutron source, SINQ, of the Swiss Institute for Nuclear Research has entered the stage of detailed design. Some development work continues in the areas of the fluid dynamics of the liquid metal target and the target diagnostic system. Engineering work concentrates on target and heat exchanger design, the $D_2O$ moderator system, and related safety issues.

The final lay-out of the source incorporates two cold neutron sources: a liquid hydrogen source at 22 K, viewed by a pair of beam tubes providing neutrons mainly in the wave-length range 2 to 4 Å, and a 20 l liquid deuterium source. From the latter, neutrons pass through another pair of beam tubes into a system of neutron guides optimised for wave-lengths greater than 3.5 Å. For optimising the deuterium source, a mock-up experiment is planned in 1987.

Although the neutronics of the source is now considered to be well established, some activities in this field are continuing. For instance, the transport of high energy neutrons through the beam tubes is further being studied since it influences the neutron background and the design of the monochromators and the associated shielding.

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UNITED KINGDOM

# REACTOR PHYSICS IN THE UNITED KINGDOM

1985-86

J.R. ASKEW, J.M. STEVENSON - AEE WINFRITH

## 1 NEWS ITEMS

In 1985, nuclear power produced about 20% of the electricity in the United Kingdom with some of the stations recording very high load factors. The Hinkley Point A Magnox station, 430MW(e), first connected to the National Grid in 1965 recorded an annual load factor of 94%. A particularly noteworthy feature of the gas cooled reactors has been the low doses experienced by the operators.

The national perception of the acceptability of nuclear power improved after its invaluable role was demonstrated during the 1984-85 dispute in the coal industry. This has been counteracted by well-publicised incidents at Sellafield and elsewhere. The industry is now trying to improve its public image. Six of the nine nuclear power plants are to be opened to the public in September. Invitations have been given on television and in newspapers for the public to make hour-long tours of the Sellafield site including the three reactors at Calder Hall, and the Thorpe fuel handling plant. The Nuclear Industry Radioactive Waste Executive (NIREX) offers to lend a video tape explaining their plans for the safe disposal of radioactive waste for years to come. In this context a £4 million programme of exploratory drilling was due to begin this month at four land-based test sites for the storage of low level waste of which only one will be chosen. However, the people living near each site are obstructing the commencement of this work. When started, drilling is expected to take up to 7 months and monitoring and assessment of data a similar period. The promised Government enquiry will not take place until 1988. NIREX has indicated the dump will probably be a concrete-lined bunker designed to contain the radioactivity for a minimum of 300 years. The possibility of using off shore wells as nuclear waste dumps is also under consideration. The scheme is estimated to cost a third of that for the development of land dumps and involves carrying waste in sealed containers by pipeline and lowering them hydraulically down wells drilled to a maximum depth of 10,000 feet.

Following the successful experience with fuel burn-ups to 18GW days/te in AGR's, improved designs are now being loaded into the reactors at Hinkley and Hunterstone to give 21GW days/te. These incorporate gadolinia poisons in the form of toroids attached to the fuel element support grids. Construction of the two AGR stations at Heysham and Torness is on schedule. The first reactor at each station is currently undergoing hot testing and fuel loading is planned in the near future.

The report by the Inspector after the 26-month long enquiry into the plans for a PWR at Sizewell is still awaited. If the CEGB were to get the go-ahead, they would plan to build a futher three or four PWR's. However, in the present climate, and with the two opposition parties having declared that they would not support a further nuclear power programme, it seems unlikely that the Government would give its approval, even if the report were favourable, before the general election next year. Irrespective of the findings, the CEGB is likely to seek planning permission for two big coal-fired power stations. They would be sited in the south of England to ease the strain on that part of the system. As coastal or river-front stations, they would have access to cheap foreign coal if necessary.

The dismantling of the Windscale AGR is being carried out with the aim of returning the reactor to a green-field site in about 10 years.

The UKAEA and BNFL have submitted an outline planning application for the construction of a European Demonstration Reprocessing Plant for fast reactor fuels at the Dounreay Nuclear Establishment in the north of Scotland. The construction of the EDRP is planned to start towards the end of the decade and to take about 8 years. It will cost £200m and could process 60-80te of fuel per year sent from demonstration FBR's to Dounreay by sea. The public local enquiry, begun in April, adjourned for the summer break on 23 July after 59 days. The enquiry is mainly being conducted in the town hall at Thurso, the nearest town to Dounreay, but visited Orkney, an island north of Dounreay, and Invergordon, a possible port of entry for the fuel. The applicants have completed the presentation of their case and the government witnesses have also completed their presentations. The principal objectors have presented about half their evidence, together with the minor objectors. Around two-thirds of the objectors' evidence remains to be taken. The enquiry will resume on 22 September and will include medical evidence on leukaemia and cancer from a study commissioned by the Scottish Office.

A new facility for impact testing research into nuclear safety is being built at Winfrith. A compressed air gun will fire projectiles at speeds up to 100mph to simulate crashing light aircraft or plant-generated projectiles impacting onto nuclear structures. Another facility at Winfrith is a laboratory to study the quality control of packaging for radioactive wastes and the way radionuclides move around inside the waste.

## 2 THERMAL REACTOR PHYSICS

### 2.1 WIMS Codes and Data

Production of a '1986' WIMS Data Library is now well under way. Features will include a 'final' adjustment to the U238 resonance integral, a substantial reduction of ~10% to the fission product cross-sections, and revisions to much of the higher actinide data that has remained unchanged for many years. This is being processed from JEF files using NJOY.

For a variety of reasons, all the WIMS codes - WIMSD, WIMSE and LWRWIMS - are being converted to Fortran 77. The major benefit will of course be portability, and the ability to run on Unix-based microcomputer networks (see below). The available range of WIMSE modules is being examined with a view to assembling a route for performing design calculations on AGR fuel assemblies.

## 2.2 AGR Reactor Calculational Methods

Work has continued on the 3D AGR fuel management code MOPSY.T to improve its versatility for studying core follow and refuelling manoeuvres. Consideration is now being given to modularising the code and methods for inclusion in a new thermal reactor Code Package (see under PWR Methods).

The kernel method (source/sink representation) that was developed for rapid assessment of power distributions in AGR's is being extended on the SUN network to 3D and two groups, with image boundary conditions and treatment of absorbers. The colour graphics display has aroused such interest in the CEGB who are now adapting the code to run on their own database system as a tool for station operators.

## 2.3 Monte Carlo Methods

The resonance sub-group model in the Monte Carlo code MONK-5W has now been extensively tested. This means that MONK can now be run directly from a WIMSE 69 group library without the need for preliminary processing through WIMS to derive the effective broad group shielded cross sections in resonance absorbers. The new code is proving valuable for 'benchmarking' the deterministic methods used in design calculations.

A new application of Monte Carlo tenchinques is being made to study the problems of axial fine structure in AGR fuel which has axial gaps and toroidal Gd burnable poisons between elements. The use of perturbation theory together with a fast tracking procedure is showing promise as a potential design method.

## 2.4 PWR Reactor Physics Methods

A comparison of start-of-life 2D power distributions calculated by various methods was carried out. The methods included conventional lattice/reactor code synthesis tecnhiques (JOSHUA) to derive channel and pin powers, a LWRWIMS quarter core 'lattice' calculation, and a MONK-5W calculation. The JOSHUA and LWRWIMS results showed reasonable agreement. The LWRWIMS calculations were used to derive albedo boundary conditions varying aximuthally around the core which can now be input to JOSHUA. The outcome of the Monte Carlo calculations was disappointing. Although the eigenvalue converged well and agreed closely with the deterministic solutions, it has so far proved impracticable to obtain a converged power distribution - the dominance ratio of about 1.02 is believed to be the root cause of the trouble.

In cooperation with the CEGB, a new Code Package is being developed, ultimately for all thermal reactor physics calculations in the UK, ie both PWR and AGR methods will be incorporated in a modular physics code framework known as PANTHER to be driven by the CEGB Pack Conversational System (PCS). The preliminary modules of PANTHER have been specified and several of them have reached the stage of code design. For PWR work this development is seen as providing a capability independent of Westinghouse methods; for AGR work it will simplify the use of existing methods and make available relevant methods developed initially for the PWR project.

## 2.5 Microcomputer Developments

Within Thermal Reactor Physics Group at Winfrith the network of SUN microcomputers has been greatly extended in the past twelve months. Several machines are now linked via Ethernet, with links also to mainframes at Harwell. All members of the group have either a machine of their own, offering very sophisticated window management capabilities, or a simple terminal to the network. A large SUN with 4 Mbytes cpu and a 380 Mbyte disc acts as a local fileserver to discless nodes on the network. Computing times are now about a factor of 10 longer than the IBM 3084 but will be further improved in the near future by floating point accelerators.

Most of the Group's major code development on WIMS, JOSHUA, MONK and MOPSY has now been moved off the mainframes onto this network and users are generally very happy indeed with the consequent improvement in productivity and cost effectiveness they are achieving.

## 2.6 Irradiated Fuel Studies

The programme of criticality studies in Dimple - see Section 4.2 - was interrupted to allow measurements to be made on a highly-irradiated fuel sample from a thermal reactor. The results indicate that the reactivity worths of fission products are calculated too high by 10-15%. This is consistent with the analysis of earlier measurements in Hector. These results are considered in a separate presentation to this meeting.

## 3 FAST REACTORS

### 3.1 PFR

Over the last year (July 1985 - July 1986) PFR has operated with a full complement of 3 secondary sodium circuits. Maximum design power (~650MW(Th), 250MW(E)) has been achieved for prolonged periods. A total of 195 effective full power days have been accumulated during the period which has involved a total of 5 shut-downs in which fuel has been charged/discharged from the core.

Two of these were scheduled shut-downs, 2 involved discharge of fuel which had developed clad defects during operation, but with the reason for shut-down being other events and the 5th involving a shut-down specifically to remove a sub-assembly in which a clad defect had developed to the extent of producing a large delayed neutron signal.

Reactor physics monitoring of the core has continued and no significant unexplained effects observed. Some of the fuel has reached, without failure, burn-ups close to 15%. A main objective of PFR is to achieve high burn-up with substantial quantities of advanced fuel in order to demonstrate lower fuel cycle costs and improved load factors for future commercial stations.

### 3.2 Subcritical Monitoring of PFR

Considerable progress has been made in the development and validation of a calculation route for relating the count rates on the low power monitoring chambers located in the radial neutron shield to the subcritical margin in PFR during core refuelling. During the past year 2 reloads have been studied in detail; in addition the methods have been used to predict, in advance, the count rate changes during the reload currently (July 1986) in progress.

The calculations are based on individual subassembly material number densities calculated and stored as part of the routine operational support calculations. These number densitites cover isotopes in the burn-up chain up to Cm242 and Cm244 which dominate the neutron output of the irradiated subassemblies. The neutron source is calculated and used in a fixed-source neutron diffusion-theory calculation in 6 energy groups to calculate neutron fluxes at the edge of the core radial reflector adjacent to the instrument positions. Two dimensional triangular geometry is used in the calculations. In general good agreement between calculated and measured count rates are obtained, exceptions to this are mainly special PFR components known as Demountable Subassemblies (DMSA's). These DMSA's have large variations in composition at different axial positions within the core height and do not lend themselves to modelling in the 2D centre-plane model.

## 3.3 The Cadenza Benchmarks

The final items in the UK analysis of the CADENZA assemblies are the axial and radial foil scans for fission in U235, U238 and Pu239 and capture in U238. The corrected experimental results have been compared with calculated rates using standard Zebra methods, ie XYZ diffusion theory and MURAL/FGL5 cross-sections.

There are no marked differeces between the general trends for C/E for pin and plate geometry. There is generally good agreement within ~1.5% between calculation and experiment in the majority of the core in all four pin and plate geometry assemblies. There is the usual tendency for C/E ratios to be greater than unity by a few percent near the boundaries between the core and the natural uranium reflector regions with values of up to ~1.2 for U238 fission a few cm into the natural uranium.

Both the radial and axial scans in the two pin cases for U238 fission show C/E ratios which are higher in higher enrichment mixed-oxide pins. This is attributed to the uncertainties in the corrections which allow for the effects of the foil material and the end-caps of the special split pins. There is also some indication of an increase in C/E for U238 fission in the axial scans near the joins of the three mini-calandria which contain the pins and make up the core height. These joins are not represented in the calculations.

There is some evidence that the preliminary corrections for the different foil sizes, used for F8 and C8 in plate and pin environments in the pin cores, are not accurate and they are presently being re-determined in the thermal column of the Nestor source reactor.

Despite the comments at the last NEACRP meeting, no further analysis results have been received from the foreign participants.

3.4 Development of a Common European Cell Code

Work is continuing on the development of the new cell code, ECCO, for the European Fast Reactor Collaboration. A member of staff from Cadarache has been attached to Winfrith to take part in this work. A specification for the code is being drawn up, carrying out studies where necessary to decide on the methods to be used.

Proposals have been made regarding the contents and structure of the input nuclear data libraries and the output cross-sections. The input data will include a fine group ($^{1}/_{120}$ lethargy width) library and a broad group ($^{1}/_{2}$ lethargy width) library (and possibly an intermediate one). It is intended that the same form of data representation should be used in the broad and fine group libraries, and in this connection it has been verified that it would be feasible to store the inelastic scattering data in the form of fine group scattering matrices.

A specification has been drawn up for the central part of the code dealing with the flux calculations and resonance shielding by subgroups. This requires collision probabilities to be calculated for various purposes: the flux solution, the subgroup treatment, and the calculation of cell diffusion coefficients. Numerical tests of methods for calculating collision probabilities have been carried out. As well as accurate methods for specific geometries, it is proposed to provide facilities for calculating approximate collision probabilities for a wide range of configurations by combining the probabilities calculated for simple geometries. A general managing routine for the calculation of collision probabilities has been specified, together with a format for the input of the geometry description.

4 CRITICALITY

4.1 MONK 6

The superhistory-powering technique (see last year's report to the NEACRP), introduced into MONK6 to solve interaction problems in criticality work, has provided a powerful basis for doing perturbation studies. The ability to calculate the differential dKeff/dx for a variety of compositional/geometric parameters x, has now been built into MONK6. The parameters may have associated dependent variables, y. Thus

$$\frac{dk}{dx} = \frac{\partial k}{\partial x} + \sum_{l} \frac{\partial k}{\partial y_l} \frac{\partial y_l}{\partial x}$$

Small finite perturbations can also be calculated:-

$$\frac{\delta k}{\delta x} \text{ for } \frac{\delta x}{x} < \text{a few per cent}$$

The new facility is currently undergoing acceptance tests at Winfrith and is performing well. It has been shown that the standard error on $\delta k$ is approximately 100 times smaller than that produced by doing two independent Monte Carlo calculations of the same size (no. of histories) and differencing the answers.

4.2 DIMPLE

Apart from the studies with the irradiated fuel sample already mentioned in Section 2.6, the Dimple programme has consisted of further measurements in the 20-compartment, boron-steel walled transport/storage skip, described in last year's report.

The first measurements were two critical assemblies which simulated loading errors by introducing 7%-enriched $UO_2$ pins in one compartment of the skip rather than the usual 3%-enriched pins. One of the assemblies is shown in Figure 1. Here, one of the central compartments contained 248 of the smaller 7% pins whereas the remainder of the compartments contained 172 3% pins. The 172 pin clusters were obtained from the 14x14 pin clusters present in all compartments of the reference assembly by removing 6 pins from each corner. The second assembly contained the same 248-pin cluster in one of the edge compartments and 184 3% pins in the other 19. These combinations were chosen so that both the perturbed assemblies had a very similar moderator height up the fuel for critical as did the reference assembly. The 7%-enriched pins produce significant peaking in the perturbed compartments, giving asymmetric flux distributions, thereby providing a good test of calculational methods in non-uniform situations. First calculated k-values from MONK 6.3 and LWRWIMS-TWOTRAN show slightly higher k-values for the perturbed assemblies than for the reference assembly. Reaction-rate distributions from TWOTRAN appear to overpredict the peaking in the perturbed assemblies.

An extra sub-critical assembly was built with circular clusters of only 52 3%-enriched pins in each compartment. This has a lower k-value than the previous subcritical assemblies and is being used as a blind test by various organisations in the UK.

The effect of water voiding between clusters in a storage array, thereby increasing the k-value by enhanced coupling between compartments, is being studied by the introduction of blocks of aluminium between circular clusters of 112 pins and the compartment walls.

## 5 SHIELDING STUDIES

### 5.1 The Winfrith Radiation Physics Subscription Service - ANSWERS

Analytical techniques, computer programs and data sets for the shielding of nuclear plant have been under continuous development in the UKAEA during the last 25 years. Nuclear applications of the Winfrith Shielding Programs include the solution of radiation transport problems in radiological and materials dosimetry in the contexts of nuclear power stations, fuel reprocessing plant, transport of nuclear materials, particle accelerators and equipment utilising radioisotopes. In addition, the programs are used in the assessment of nuclear measurement techniques.

This expertise is now being made available to industry through the subscription service ANSWERS, which is based at Winfrith.

The facilities available to members include:-

1) Access to a fully supported set of shielding programs with unique provisions for quality assurance;

2) Full documentation on the programs;

3) The opportunity for company staff to participate in seminar/workshops on the application of the programs;

4) A "hot line" telephone service dealing with users problems encountered with the application of the programs;

5) Consultancy on design problems.

The programs which constitute the Winfrith suite of recommended methods, already widely used in the nuclear industry, are either run at commercial rates on Authority mainframes or, alternatively, leased to members for use on their in-house computers. The service is widely used in the UK, and this year, seminar/workshops have been given to prospective members in both Holland and West Germany and there is significant interest in the US.

## 5.2 Data-Testing Benchmarks

McBEND Monte Carlo calculations have been carried out for the Winfrith iron benchmark experiment using UK nuclear data (file no 908) and JEF-1 data. Reaction-rate distributions for $S^{32}(n,p)$, $In^{115}(n,n')$ and Rh103(n,n') were studied. The JEF-1 data gives better agreement with experiment - mainly as a result of improved values of the inelastic scattering cross-sections for iron in the energy region 860keV $\rightarrow$ 1.5MeV. Sensitivity studies with the perturbation code SWANLAKE suggest no significant revision is required of the JEF-1 elastic and inelastic cross-sections for iron in the energy region from 50keV to several MeV. In collaboration with the NEA, calculations are currently being carried out for the iron experiment using DOT 4.2 with JEF-1 data.

The Winfrith water benchmark experiment has been calculated using McBEND with UK data and DOT 4.2 with JEF-1 data. Agreement between calculated reaction-rate distributions and experiment is generally satisfactory. Some small modelling corrections remain to be made to the DOT calculation. The McBEND calculation with JEF-1 data will shortly be carried out.

# 6 TRANSPORT FLASK TECHNOLOGY

Winfrith is the main centre of expertise in Radioactive Materials Transport Technology in the UK. As well as the criticality work on Dimple, already discussed in Section 4.2, areas of study include:-

(a) validation of computer codes for flask shielding by experiments on the Aspis facility of the Nestor source reactor, supplemented by measurements on operational flasks.

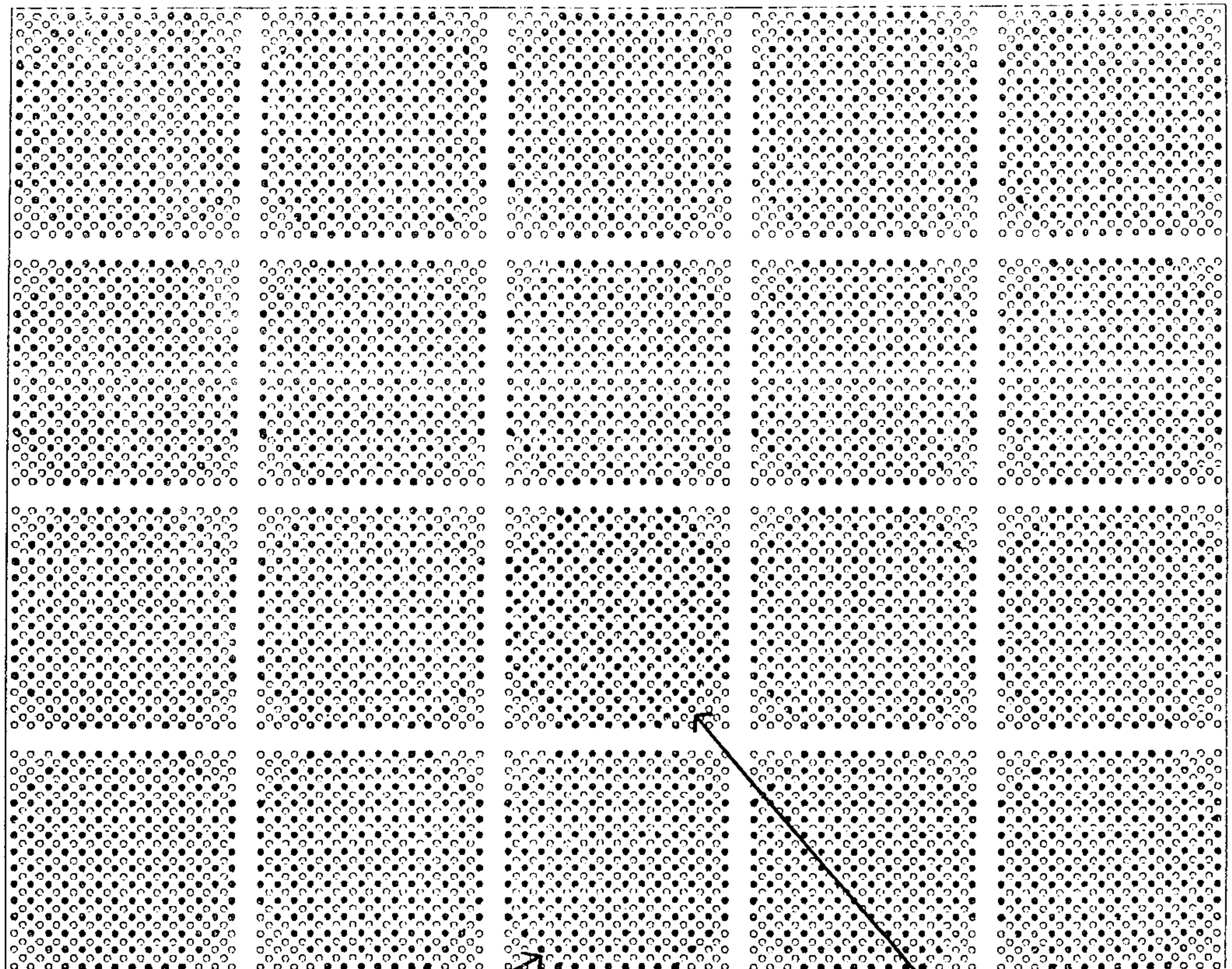
(b) the study of heat transfer by conduction, convection and radiation, often using full-size flasks. The energy generation is simulated by electrical resistance heaters. Typical of the features investigated is the performance of external cooling fins fitted to flasks.

(c) fire resistance. An improved open pool fire test facility capable of 24 hour tests is being built and will be operational later this year.

(d) impact behaviour. One facility permits the dropping of containers 9m onto a massive unyielding target. Individual components such as shock absorbers are tested using a compressed-air projectile launcher giving impact velocities of several hundred m/sec. The new Horizontal Impact Rig, mentioned in Section 1, will give larger guided capabilities, eg 1.6tonnes at 45m/sec or 250kg at 300m/sec, and will spend half of its time on flask impact behaviour.

FIGURE 1

Skip Loading for Perturbed Assembly with 7% Pins in a Central Compartment



Cluster of 172 3% enriched $UO_2$ pins

Cluster of 248 7% enriched $UO_2$ pins

# Reactor Physics Activities in the United States
# A Report to the NEACRP

P. B. Hemmig
U. S. Department of Energy
Washington, DC 20545

## Introduction

Recent reactor physics activities in the U. S. have been focused on core designs for advanced liquid metal reactors to provide improved economics, reliability, and passive safety. A wide range of core design features have been explored for both metal and oxide fuel systems and uranium fueled startup options. Tests have been carried out at EBR-II at full power to demonstrate the ability of EBR-II and similar reactors to shut down safely under loss of flow and failure to scram conditions. Tests of special gas expansion modules in FFTF have shown their ability to shut down FFTF under loss of flow without scram conditions. Further tests of reactivity feedback calculational capabilities are planned using both EBR-II and FFTF.

ZPPR critical experiments were completed for metal fueled cores which are similar in size and composition to an advanced being designed by the Rockwell International Corporation. These critical measurements confirmed key physics predictions for these cores and also tested new apparatus for measuring reactivity effects due to fuel bowing and core expansion.

An extensive program of shielding studies was initiated at the ORNL Tower Shielding Facility to improve shield designs of advanced liquid metal reactors in cooperation with PNC/Japan.

## Critical Experiments

Benchmark critical experiments were carried out in support of fast reactor designs for integral (metal) fuel cycles (IFR). The measurements were done in a fairly small (300 MWe) metal-fueled critical assembly designated ZPPR-15. The data are the first obtained for a metallic-fuel composition since the EBR-II mockup experiments were done on ZPR-III in the early 1960s. The assembly size is in the range of inherently safe LMRs being considered by DOE in its present advanced reactors program.

The ZPPR-15 program was divided into four phases which emphasized different compositions for an IFR. Each critical configuration was a clean, cylindrical, two-enrichment zone assembly. The 15A configuration was built such that the principal change relative to some past ZPPR assemblies would be simply the removal of oxygen from the fuel composition. The loading represents a Pu-equilibrium core, i.e., a near end-of-life condition for an IFR. The 15B configuration also represented a Pu-equilibrium core, but with zirconium, the third element of the ternary IFR fuel, added at 8-20 w/o in the inner core zone. The 15C composition represented a mid-life IFR core, with the fissile component consisting of 50% U-235 and 50% Pu. The 15D composition represented a near beginning-of-life IFR core, having 90% U-235 and 10% Pu. The 10% Pu was retained to provide a uniform neutron source for the measurements.

The ZPPR-15 experiments included all the standard physics parameter measurements of critical experiments: criticality, reactivity coefficients (fuel, coolant, control, structural, Doppler), fission rate distribution, gamma heating rate distribution, breeding, spectrum and kinetics. As changes were made in the core composition, measurements which were particularly sensitive to spectral changes were made. For example, the zirconium scattering cross sections were relatively uncertain, which could have a significant impact on reactivity coefficients and reaction-rate ratios.

Several new experiments were also incorporated in ZPPR-15, including in-rod boron reaction rates, $B_4C$ external shielding, bowing reactivity, axial expansion, and sodium manometer (a passive safety device between the core and blanket) worth.

Analysis of these experiments is being done with ENDF-V.2 nuclear data and 3D nodal diffusion and transport codes, so direct comparison with previous oxide-core results would require small detailed adjustments. No major surprises in physics parameter predictions have been discovered, although only the 15A and a part of the 15B experiments have been calculated so far. Predictions of the worth of voiding sodium from the manometers were poor, but this had been expected because of known approximations in the conventional analysis methods used.

In September, work started on a small, BeO-reflected assembly. After those experiments are concluded in December, a joint cooperative program with PNC of Japan is expected to start. This JUPITER-III program will include a large axially heterogeneous configuration and a very large conventional configuration.

## FFTF Physics

Beyond normal reactor operations support, considerable reactivity feedback experiments were designed and completed, and loss of flow without scram experiments were conceived, planned and successfully executed in a core containing nine gas expansion modules (GEMS). Special isotopes work was completed and test design work continued.

Cycle 7 feedback experiments were conducted as a pilot series of experiments to confirm our ability to characterize return of reactor power, flow and temperature to equilibrium following reactor state changes. A significant observation was that equilibration was difficult to achieve in times short compared to power drifts due to burnup. This indicated the need for extensive data collection and reduction for Cycle 8A feedback experiments.

Cycle 8A tests involved transitions between pairs of 198 reactor equilibrium states. Reactor conditions were varied to separately emphasize fuel temperature, coolant temperature, radial and axial expansion and bowing reactivity feedback effects. Preliminary analyses of these data suggested that analytic models of radial expansion feedback had time constants that were too short. Core component distortion tends to eliminate rapid reactivity effects due to bowing (time constant of seconds) and places bowing and radial expansion under the influence of expansion, with a time constant of about one minute, in the core restraint mechanism. These changes resulted in very significant increases in expected coolant temperatures for loss of flow without scram with GEMs (LOFWOS W/GEMs) tests.

LOFWOS W/GEMs tests were conducted successfully from up to 50% of full power to both pony motor and natural circulation flow conditions. Predicted temperatures significantly exceeded observed temperatures. Explanations of these discrepancies have not been confirmed, however, observed flow redistribution in natural circulation experiments suggests that the single channel model, in the IANUS thermalhydraulics computer program, is inadequate. SASSYS has been imported and is being used to perform multi-channel analyses of these experiments.

GEM worth measurements were made using inverse kinetics before and after each LOFWOS experiment. Values of worths for 9 GEMS varied only slightly and ranged from 136 cents to 142 cents. The time to initial appearance of negative reactivity was about 3-5 seconds following pump trip at low power and low temperature, and this time seems to decrease slightly with increasing power. The latter suggests that gas expansion at higher power moves the sodium-gas boundary closer to the core top face.

An assembly with yttrium hydride as a moderator was designed to produce cobalt-60 samples with high specific activity. Experiments conducted in ZPPR by ANL-West provided the basis for calibrating HEDL design methods for this assembly. The reduction in uncertainties of HEDL design methods after ZPPR calibrations provided adequate confidence to insert the cobalt test vehicle into FFTF in Cycle 9A. A major issue prior to the ZPPR experiments was possible excessive heating in standard fuel pins adjacent to the cobalt test assembly. Thermalization of neutrons in the test vehicle and back leakage to the core was estimated to increase linear heat rates in adjacent fuel pins by as much as 70%. Since cobalt tests are designed for Row 7 and linear heat rates are low in Row 6 fuel, these increases were found to be acceptable. Gadolinium-153 production using FFTF is also being evaluated in the cobalt test vehicle.

EBR-II Physics

The passive safety potential of metallic fuel was demonstrated for loss of flow without scram (LOFWOS) and loss of heat sink without scram events by a series of tests conducted with EBR-II in April 1986. These tests demonstrated that the unique combination of the high heat conductivity of metallic fuel and the thermal inertia of the large sodium pool can shut the reactor down during these potentially severe accident situations without depending on human intervention on operation of active engineered components. Further tests are planned of transient overpower events to further validate analytical predictions of the inherent safety potential of metallic fueled cores.

## Shielding

In September of 1985, the U.S. DOE and Japan's PNC entered into a multiyear program of experimental shielding research to support the development of advanced sodium-cooled power reactors. The program is designated JASPER and consists of a series of shielding integral experiments to be performed at the ORNL Tower Shielding Facility. Experiments will include basic studies of neutron attenuation and streaming in benchmark geometries and for representative mockups of LMR-type shields. The first experiment is nearly completed and includes measurements of several design alternatives for LMR radial shields. Fabrication is in progress for the next two experiments, which will investigate neutron streaming in the fission gas plenum and axial shield regions of an LMR.

Also, considerable shield design support analysis has been performed to support the development of U.S. advanced LMR and HTGR reactor systems. As part of the effort, a new multigroup cross section library was prepared. The library, which is based on ENDF/B-V, contains data in two different group structures and uses several different energy weighting functions appropriate for LMR shielding analyses. The library also contains a variety of commonly used response functions.

## Nuclear Data

Improvement of the ENDF/B nuclear data system is continuing in the U.S. through efforts of the Cross Section Evaluation Working Group and the BNL National Nuclear Data Center. The ENDF/B-6 standards file is nearing completion and appears internally consistent with key cross sections for reactor design. New evaluations for the resonance region cross sections for Pu-239, U-238, and U-235 are nearing completion. Other evaluations for B, Pu-240, Eu, Au, Ho, Y, Cu, Ni, Mn, Cr, Nb and Fe are in preparation.

Evaluated delayed neutron precursor and aggregate spectra are reported by T. R. England, et al, at the Birmingham specialist meeting on delayed neutrons on September 15-19, 1986. Evaluations of the delayed neutron emission probabilities for 85 precursors were also reported by F. Mann at this same meeting.

## Reactor Computations

The burnup code REBUS-3 has been converted to run on CDC/NOS and Cray/CTSS systems; the CDC implementation was on a two-level-memory machine. The DIF3D code, which performs finite-difference and nodal calculations, has been extended to Fortran 77. The utility routines used by DIF3D, REBUS-3 and most of Argonne's production reactor physics codes have also been converted to Fortran 77. Fortran 66 versions of the cross section generating codes $MC^2$-2 and SDX have been sent to the National Energy Software Center.

Because of the significance of fuel assembly motion in the understanding of inherent safety features of a reactor, a study has been initiated to investigate boundary perturbation methods for computing worths. Expressions for the worth of lateral fuel assembly motion have been derived both from the continuous form of the neutron diffusion equation and from the finite difference form. Even in the limit of small motion, conventional first-order perturbation theory turns out not to give the correct answer. Test calculations and model problems show that both forms of boundary-perturbation worths agree with exact perturbation theory. Work is underway to code a three-dimensional calculation of boundary-motion worths.

Because of the rigor of the Reich-Moore formalism in representing the energy behavior of neutron cross sections, future releases of evaluated resonance parameters for many important nuclides will use that format. In anticipation of such developments, two independent methods of Doppler broadening Reich-Moore cross sections have been developed. One direct method is to broaden the point-wise Reich-Moore cross section rigorously by solving a finite-differenced heat equation. An improved algorithm with optimal choice of mesh has been developed which provides adequate efficiency for practical applications. An analytical method has been developed which is essentially an extension of the existing POLLA approach with the explicit energy dependence of the parameters taken into account. The resulting Doppler broadened line shape functions can still be represented by the conventional chi and psi functions. Hence, the method can be readily applied in existing processing codes. A test program, WHOPPER, which incorporates both methods along with the method proposed by Frohner, has been written; test calculations are still underway.

The development of nodal transport methods has produced fast-running transport theory calculations, but the speed is achieved at the cost of restricted angular approximations. Recent work at ANL has shown that the usual isotropic transverse leakage approximations may give errors in fluxes, in fast reactor experimental configurations, of between 10 and 20 percent at blanket/reflector interfaces. Attempts to refine the treatment of the transverse leakage have, so far, led to unstable computational algorithms. At the same time an older method, the simplified spherical harmonics (SSH) method, is being reexamined with encouraging results. The SSH equations are coupled diffusion equations and can be cast in nodal form by existing, established methods. All the finite difference and nodal options in DIF3D are now available for simplified P3 (SP3) calculations. Thus, that code now has the capability to do transport-like calculations in hex-z geometry. So far tests indicate that SP3 is generally about as fast as, but often substantially more accurate than, corresponding nodal methods. SP3 is, in addition, much easier to implement in complex geometries.

In the thermal-hydraulics area, the preconditioned conjugate-gradient (CG) method has been coded into the COMMIX code as a solution method for the pressure equations. This new option is available as an alternative to the successive over relaxation (SOR) method, and tests show it to be much faster than SOR in many cases. The application of CG to COMMIX is still being monitored and debugged. Additional investigations are underway to test the potential of the multigrid method as a solution technique for the pressure equations.

Advanced LMR Concept Studies

Investigations have continued at ANL, Westinghouse (AESD), General Electric, and Rockwell International on several innovative liquid metal reactor design features. These include passive safety devices such as the Gas Expansion Modules and Self-Actuated Shutdown Systems which automatically introduce negative reactivity on loss of flow and higher than normal temperature conditions respectively. Heterogeneous cores with high internal conversion ratios and low power densities have been studied to minimize reprocessing requirements, extend core life, reduce burnup control requirements and provide attractive safety margins. Both metallic and oxide cores have been studied with both uranium and plutonium fueling.

Major analyses have continued in support of the SAFR (Sodium Advanced Fast Reactor) design by Rockwell International and PRISM (Power Reactor Inherently Safe Module) design by General Electric. Both reactors have adopted reference core designs using the U-Pu-Zr ternary alloy metallic fuel with oxide fuel designs as backup. The selection of an advanced reactor concept for more intensive design and development will be made in approximately one year upon completion of the present studies.

# REACTOR PHYSICS ACTIVITIES AT THE JRC-ISPRA
(November 85 - September 86)

## REACTOR CORE PHYSICS

### *Perturbation Analysis with KENO-EUR*

Reactor Physics activities have mainly dealt with the validation and application of KENO-EUR using new Monte Carlo perturbation schemes.

In this frame the French SCARABEE reactor containing a fast reactor test-loop composed of 37 fuel elements has been analyzed. The main interest of this study was focussed on the difference in power distribution as a function of material changes of certain structural components.

Other *"field-tests"* of this new code have been carried out in collaboration with the GRS-Garching. Besides of a series of comparisons with equivalent DOT calculations, the code has been applied to criticality studies in fuel disolvers. Again the interest centred around differential effects caused by material changes or the loss of absorbers.

Finally a hexagonal benchmark problem was analyzed and compared with other calculations, in which the central fuel rod was replaced by an absorber rod. The reactivity difference determined in a single KENO-EUR run, agreed exactly with results obtained by deterministic methods.

### *Feasibility Study on the Development of a New Neutronics Module for EAC*

In a collaboration agreement with KWU (Erlangen) a 3-D flux module for the EAC code system is developed. Based on extensive experience for LWR applications and on numerical results for LMFBR calculations in Cartesian geometry the development of a nodal method for hexagonal-z geometry is suggested which would most likely meet the specification and requirements outlined in the paper on "Requirements for a New Neutronics Code for EAC". The proposed nodal code HEXNOD would allow to solve 3-D LMFB problems in both diffusion theory and transport approximation with the same accuracy but with computing times which are smaller by at least one order of magnitude, compared to conventional FDM codes. The development of such a hexagonal nodal code would be done in close analogy to the Nodal Expansion Method (NDOM) for Cartesian geometry which have been very successfully for LWR applications in the computer codes MEDIUM-2 and MULTIMEDIUM. A very attractive feature would be that both diffusion theory and transport calculations could be done optionally in the framework of the same nodal code.

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## RADIATION SHIELDING

*Iron and Sodium Deep Penetration Benchmarks*

In completing the EURACOS shielding benchmark studies the previous 1D and 2D analysis has been complemented by 3D Monte Carlo evaluations considering carefully the geometrical configuration.

In particular it was possible to calibrate exactly the source strength of the converter plate taking into account the details of the empty irradiation channel including all kind of back-scattering effects.

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## ACTINIDE MONITORING

*The Neutron Autocorrelator*

The neutron autocorrelator is used by Supervising Authorities for the mass determination of Pu present in nuclear fuel. Determined is the spontaneous fission neutron emission rate eliminating effects of ($\alpha$-n) reactions in $PuO_2$ of neutron multiplication by primary source neutrons and of counter dead time effects.

The activity performed in this field can be subdivided into 3 main areas:

1. measurement of instrumental parameters
2. development of models
3. development of neutron transport codes to study the spatial effects of a neutron signal pulse train.

*1. Measurement of instrumental parameters*

Instrumental parameters of the auto correlator shift register and its detector are measured in support of the Nuclear Safeguard Direction of Luxemburg.

*2. Development of interpretation models*

During the past recording period an expert meeting had been arranged by the IAEA on Neutron Correlation Techniques. During the discussions the dead time correction with the shift register published by W.Matthes and R.Haas has been accepted

as a general approach to this problem. For the neutron multiplication effect the formulas found independently by Ensslin and Hage-Cifarelli were recommended for Pu assays in further use with IAEA shift registers.
At the JRC these two approaches were programmed on a PC communicating with the shift register in order to analyze measured data obtained by non specialized staff.

### 3. *Development of neutron transport codes*

The adaptation of the Monte Carlo code TIMOC for the interpretation of time and space dependent neutron signal correlation processes has been terminated.
Several test calculations showed clearly that 99% of the induced fissions in $PuO_2$ samples occur during the first 2 $\mu$sec after appearance of the primary ($\alpha$-n) or spontaneous fission events. The concept used for the neutron multiplication correction of shift register measurement data which implies that the induced fissions occur quasi contemporarily with the primary neutrons is therefore confirmed.

### 4. *Development of the time correlation analyzer*

The time correlation analyzer is an instrument permitting the measurement of signal singlets, doublets and triplets for the essay of Pu. Approximate interpretation models exist for this instrument anlyzing the frequency distribution of signals inside signal and randomly triggered observation intervals. Also exact formulations for the factorial moments of this distribution are available.

The first type of analysis method has been equipped with a subroutine permitting corrections of dead times of the He-3 neutron detector banks.
So far experiments were performed with a computer serving as a fast data acquisition system. This unit is going to be replaced by a hardware instrument permitting measurements with 16 time intervals.

### 5. *Fast time of flight multiplicity analyzer for Pu essay.*

A fast coincidence device, composed of three plastic scintillation detectors, aims at the measurement of the absolute rate of radioactive sources decaying with multiple emission of neutrons and gamma rays. The process to be investigated is the spontaneous fission, and the method is intended for determining the mass of fissile isotopes such as those of plutonium, against a background of ($\alpha$,n) reactions due to the oxygen of $PuO_2$. Moreover, a time-of-flight analysis, triggered by the coincidence signals, enables to discriminate between the detected neutrons and gamma rays.
Preliminary experiments on spontaneous fission sources such as Cm-244 and Cf-252 showed that, owing to the peculiar features of the fission process not following he pattern of a radioactive decay, the gamma ray component can come closer to the conditions required by the coincidence method for an absolute measurement.

The further application to $PuO_2$ samples and the simulation of bulk sources evidenced the difficulty to obtain an absolute measurement due to the effects of scattering and absorption of neutrons and gamma rays by the matrix material.

**SAFETY OF FUSION TECHNOLOGY**

The potential impact of a fusion reactor on the environment is essentially related to the neutron induced radioactivity of materials. As long as the radioactivity will remain within the reactor building, there will be no hazard for the environment and no concern for the public outside the plant site. These will arise as soon as any release of radioactive material will be produced as a consequence of normal operation or accidental conditions.

Most of the RAI (Radio-active Inventory) of a fusion reactor stays in the structural materials, which are quite stable and non mobile. Safety problems in the operating plant will mainly be associated with the disassembly and replacement of activated components, or with accidental situations. At the end of the life of the plant, the radioactivity of the plant materials will remain important for hundreds or thousands of years. A proper strategy shall be designed for the decommissioning of the plant, the possible recycling of materials, and the disposal of radioactive wastes.
RAI of NET has been already reported in [1] and mentioned in the program progress report of 1985.

Extrapolation to power reactors should take into account the following factors:

- scaling of the power, and hence flux and fluence
- scaling of the size and masses
- replacement of components
- selection of materials

Table I shows the main parameters required to extrapolate the RAI data from the experimental reactor NET to the power plant FCTR. The first wall of the power plant is expected to be replaced after having suffered an integrated wall loading of about 10 $MWa/m^2$, that is about three times more than the maximum values achieved in NET, and this will occur once every 7 or 8 years.

TABLE 1 - Main Parameters of NET-III-A and FCTR

| | NET-III-A | FCTR |
|---|---|---|
| Neutron wall loading ($MW/m^2$) | 0.95 | 1.78 |
| Fusion power (MW) | 740 | 3520 |
| First wall area ($m^2$) | 619 | 1582 |
| Blanket volume ($m^3$) | 288 | 1052 |
| Mass of first wall (tons) | 68 | 174 |
| Mass of breeding blanket (tons) | 620 | 2260 |

The specific activity of the FW material when dismantled for replacement will be about three times larger than the value predicted for NET, because of higher fluences; the RAI will involve a mass of material about 2.5 times higher for the FW, for each replacement. At least three replacements have to be foreseen. The total RAI of the first wall of the power plant will hence to be about 25 times larger than that of NET. This conclusion assumes that the same structural material, namely AISI 316, is used in FCTR. This will not be likely true.

Research is in progress to develop new materials to match the specific requirements of the structural materials of a TOKAMAK reactor. These efforts are intended in particular to reduce long-term radioactivity by modifying their composition.
Three main families of material are being investigated: austenitic steels, ferritic-martensitic steels, and vanadium alloys. Within each of these families of material, particular interest is presented by those alloys having lower long-term radioactivity, these are:

- austenitic steels modified to replace Ni with Mn and Mo with W and/or V
- ferritic/martensitic steels in which Mo and Nb are replaced by W, V and Ta
- vanadium alloys such as V20Ti, V-15Cr-5Ti

AMCR-33 is an example of alloy of the first family; it does not contain cobalt and molybdenum, while nickel is reduced to 0.1%. This alloy could be the structural material for the first wall, the breeding blanket and the TF coil. For the inner and outer shield one could simply use iron as a shielding material together with water., With this choice of materials, and assuming that 17Li83Pb is the breeder material, we compute the radioactive inventory shown in Table 2. This inventory is larger than that of NET during reactor operation, but close to it after long decay times. Also in this case the major contribution to the total inventory comes from the first wall and the breeding blanket.

V15Cr5Ti is an example of the third family of alloys. If we assume to use this alloy for the first wall and the breeding blanket, and keep the other materials as in the previous example, then the radioactive inventory vs decay time is that shown in Table 3. This table shows that the total inventory is reduced by about one order of magnitude as compared to the previous one. It is interesting to note that in this case the breeder material 17Li83Pb provides a fraction of the inventory which is larger than that of the first wall after decay times of about 30 years.

In these calculations we assume that the first wall is replaced after a period of time corresponding to 6 years of full power operation, and that three first walls are needed during the life of the plant. The inventory quoted in Table 2 and 3 take into account the three first walls. The other components are assumed to be irradiated for a time corresponding to 18 years of full operation.

TABLE 2 - Radioactive inventory of FCTR (Curie). Structural material AMCR-33

| Materials | Decay time (years) | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | 0 | 1 | 10 | 100 | 300 | 1000 | $10^4$ | $10^5$ |
| AMCR-33 in: FW | $1.53\cdot10^{10}$ | $4.96\cdot10^{9}$ | $3.53\cdot10^{8}$ | $2.90\cdot10^{4}$ | $1.48\cdot10^{4}$ | $1.19\cdot10^{4}$ | $4.40\cdot10^{3}$ | $4.21\cdot10^{2}$ |
| br.bl. | $7.54\cdot10^{9}$ | $1.38\cdot10^{9}$ | $1.03\cdot10^{8}$ | $4.13\cdot10^{4}$ | $2.56\cdot10^{4}$ | $2.05\cdot10^{4}$ | $7.23\cdot10^{3}$ | $3.89\cdot10^{2}$ |
| Iron in shield | $3.26\cdot10^{7}$ | $2.00\cdot10^{7}$ | $1.90\cdot10^{6}$ | 3.39 | 2.45 | 2.45 | 2.45 | 2.40 |
| AMCR-33 in TF coil | $8.7\cdot0^{3}$ | $1.8\cdot10^{2}$ | $1.4\cdot10^{1}$ | - | - | - | - | - |
| 17Li83Pb | $7.16\cdot10^{8}$ | $7.70\cdot10^{6}$ | $1.81\cdot10^{6}$ | $2.36\cdot10^{4}$ | $2.10\cdot10^{3}$ | $1.82\cdot10^{3}$ | $1.82\cdot10^{3}$ | $1.81\cdot10^{3}$ |
| Coil mix. (except AMCR) | $2.52\cdot10^{3}$ | $9.07\cdot10^{1}$ | $1.2\cdot10^{1}$ | 0.88 | - | - | - | - |
| Total | $2.35\cdot10^{10}$ | $6.37\cdot10^{9}$ | $4.60\cdot10^{8}$ | $9.39\cdot10^{4}$ | $4.25\cdot10^{4}$ | $3.42\cdot10^{4}$ | $1.35\cdot10^{4}$ | $2.72\cdot10^{3}$ |



TABLE 3 Radioactive inventory of FCTR (Curie). Structural material V15Cr5Ti

| Materials | Decay time (years) | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | 0 | 1 | 10 | 100 | 300 | 1000 | $10^4$ | $10^5$ |
| V15Cr5Ti in: FW | $3.43\cdot10^{9}$ | $6.86\cdot10^{7}$ | $6.42\cdot10^{4}$ | $1.52\cdot10^{3}$ | $1.47\cdot10^{3}$ | $1.34\cdot10^{3}$ | $4.51\cdot10^{2}$ | 1.54 |
| br.bl. | $1.96\cdot10^{9}$ | $1.03\cdot10^{7}$ | $1.15\cdot10^{4}$ | $2.59\cdot10^{3}$ | $2.52\cdot10^{3}$ | $2.32\cdot10^{3}$ | $7.80\cdot10^{2}$ | 1.33 |
| Iron in shield | $3.26\cdot10^{7}$ | $2.00\cdot10^{7}$ | $1.90\cdot10^{6}$ | 3.39 | 2.45 | 2.45 | 2.45 | 2.4 |
| AMCR-33 in TF coil | $8.7\cdot10^{3}$ | $1.8\cdot10^{2}$ | $1.4\cdot10^{1}$ | - | - | - | - | - |
| 17Li83Pb | $7.16\cdot10^{8}$ | $7.70\cdot10^{6}$ | $1.81\cdot10^{6}$ | $2.36\cdot10^{4}$ | $2.10\cdot10^{3}$ | $1.82\cdot10^{3}$ | $1.82\cdot10^{3}$ | $1.81\cdot10^{3}$ |
| Coil mix. | $2.52\cdot10^{3}$ | $9.07\cdot10^{1}$ | $1.22\cdot10^{1}$ | - | - | - | - | - |
| Total | $6.14\cdot10^{9}$ | $1.07\cdot10^{8}$ | $3.79\cdot10^{6}$ | $2.77\cdot10^{4}$ | $6.09\cdot10^{3}$ | $5.48\cdot10^{3}$ | $3.05\cdot10^{3}$ | $1.81\cdot10^{3}$ |

It is important now to mention the effects of impurities that could be present in the alloys, and introduce other long-living radioactive isotopes. Nb, Ag, Eu, Tb, Ir, Bi seem to be the most critical impurities since their presence could bring important contributions to the radioactivity of the material, already at the level 1 ppm (part per million). None of these impurities has been considered in the reported results. Their contribution is difficult to appreciate since it is affected by many uncertainties. First, it is difficult to know the concentration of these elements in the raw materials used to produce the steel and their concentration in the final alloy. These concentrations may change according to the origin of the raw materials. The library of activation cross-sections that is mainly based on model calculations and might therefore be subject to relevant uncertainties. Unfortunately the library is still incomplete.

In a collaboration agreement ECN Petten is providing a new file with activation data for all natural elements. This file called REAC-ECN[2] is now being implemented at the JRC. It is based on the REAC[3] library, in which old data have been corrected or updated and to which data for new nuclides have been added.

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## SOURCE TERM

### *Objectives*

The objectives of this Research Area were concentrated in the study of fission product and aerosol behaviour in the containment system of a PWR nuclear power plant.

The phenomena leading to the pressurization and possible rupture of the containment are in fact considered of high priority.

The resources for this activity in the Direct Action Programme are limited, however the link with the Shared Cost Actions which are being launched will increase the effectiveness of the Commission's initiatives.

The analysis of unsolved problems and of specific European needs was further prosecuted in view of the formulation of a proposal for the 1987-90 JRC programme and of the identification of priorities in the activities to be supported through the Shared Cost Action.

***Results***

*Participation of the JRC in the LACE project*

The LACE project investigates, in large scale vessel simulating a PWR containment, the thermohydraulic and aerosol behaviour under severe accident conditions in which containment bypass, failure to isolate and sudden depressurization conditions are considered.

This project is sponsored by EPRI (USA) and executed in the HEDL Laboratories in Richland, with the participation of many European and Extraeuropean countries.

The participation of the Commission, after a long negotiation, has been agreed and the Contract was signed in December 1985.

The experimental programme, which includes 6 tests, was started in October 1985 with the execution of a containment bypass test. The second test was scheduled for February 1986.

The preparatory work at Ispra to participate in the "blind" code comparison exercise organized in the frame of LACE was continued. The CONTAIN 1 code will be utilized for this purpose.

*Analytical Activity*

The analytical activity can be subdivided as follows:

1. Implementation (including analysis and testing) of computer codes.

2. Participation in code comparison exercises.

2. Analysis of experiments on aerosol behaviour.

4. Development of new aerosol models.

1. Implementation of computer codes

A first phase of analysis of the physical models utilized in the code has been completed, the available documentation is sufficient for running the code but does not give a complete information on the scientific background of model construction and validation. On some selected topics, which will be identified as part of a possible code assessment effort, a detailed analysis will be undertaken in the future.

The release by Sandia of the core-concrete interaction module CORCON has been agreed upon by NRC and therefore this module should be soon available.

The build-up of a MARCH-2/CONTAIN interface has been started, which should allow to transfer automatically the required thermohydraulic data, namely the mass and energy source for the containment building. Service modules for MARCH-2 and CONTAIN graphical outputs are in advanced stage of development.

The updating of CONTAIN from the version 1.00 to the version 1.04 utilizing the data provided by Sandia was completed.

Due to the fact that the MARCH-2 and CONTAIN codes are operational on external computer (CYBER 176 and CRAY), and the graphical outputs are elaborated, for practical reasons, on the AMDHAL computer at Ispra, some difficulties have been experienced in file transmission and format compatibility.

2. Participation in code comparison exercises

In the code comparison exercise organized by the Italian GLAST group on the Zion NPP, the analysis of the AD sequence has been completed.
The summary report has not yet been relesed by GLAST, but the single contributions have already been collected.

This collaboration has been prosecuted with the analysis of the TMLB'sequence (Figg. 1 and 2)

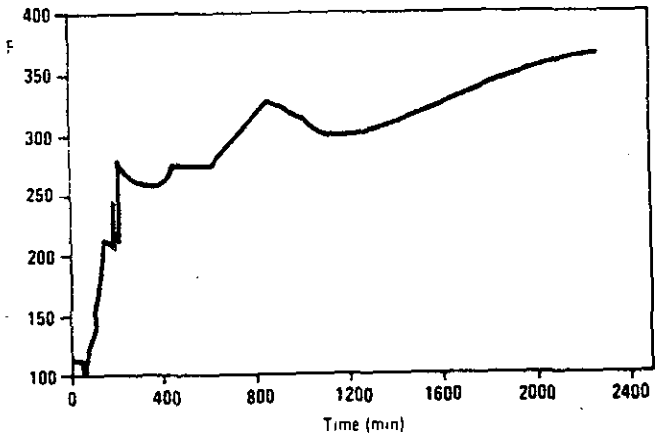


Fig. 1. ZION TMLB Sequence no hydrogen burning comprt. temperature

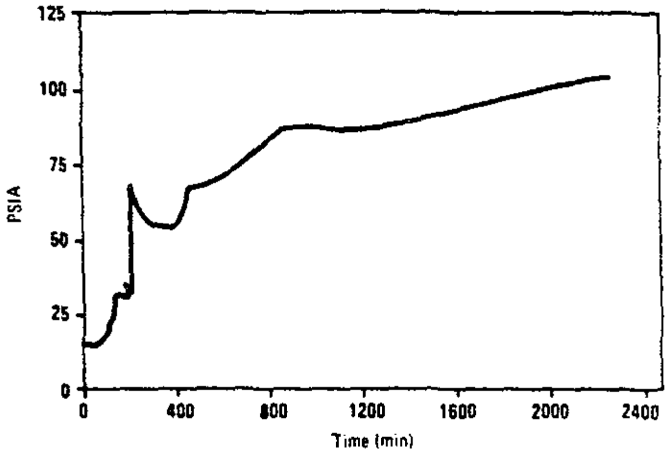


Fig. 2. ZION TMLB Sequence no hydrogen burning total compt. pressure

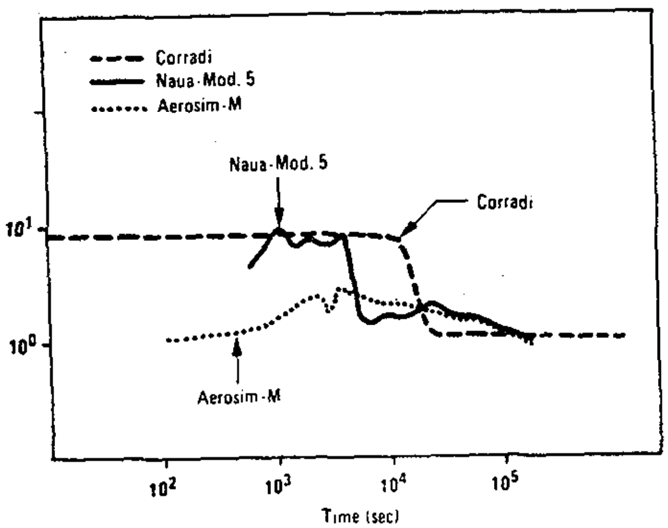


Fig. 3. Aerosol Mass Medium Radius (Total Mass) Surry $S_2CD$ Sequence

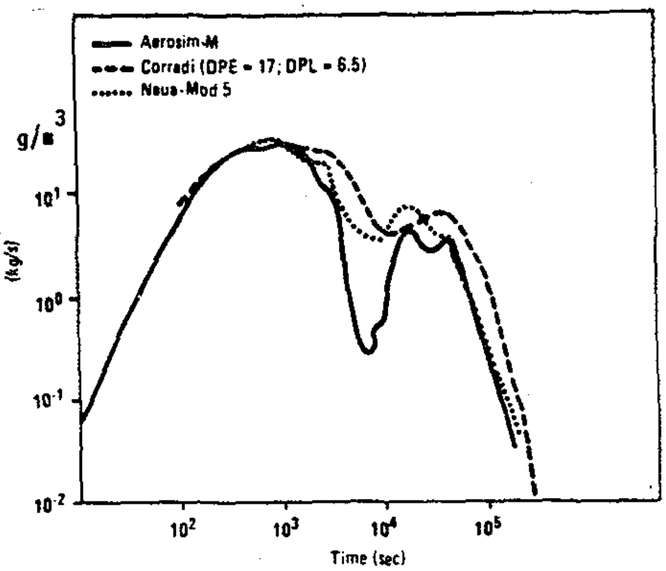


Fig. 4. Aerosol Suspended Mass Concentration (Dry Component) Surry $S_2CD$ Sequence

Because both MARCH-1 and MARCH-2 are now in operation at Ispra, a comparison of the two codes has been done for this case. This has shown that some shortcomings of MARCH-1 have been eliminated in the new code version, but also that MARCH-2 requires a much more detailed description of the plant and imposes to the user a difficult choice among alternative physical models of the same phenomena, e.g. the evolution of porous beds of core debris in the reactor cavity.
In Ref. 1 the JRC report presented at the GLAST meeting of October 1985 in Pisa is given.

The JRC partecipation in the Aerosol Code Sensitivty Study, organized by DG-XII-Brussels has been actively prosecuted. A meeting of participants was organized on 10-11 April 1986, at Ispra. The final results of the Aerosol Code Comparison, on the $S_2CD$ results of the Aerosol Code Comparison, on the $S_2CD$ sequence of the SURRY-NPP, have been discussed. Results of this study have been presented in a paper by Mr. Fermandjian at the Second International Aerosol Conference in Berlin (September 1986). In the same meeting the second phase of the collaborative activity has been defined. It is devoted to the comparison and validation of the Codes against large scale experimental data. Mr. Schoeck (KFK/FRG) was charged to prepare and present the input data necessary for the calibration of the DEMONA Test B3.

In a meeting, held at Brussels, on June 30th 1986, the preliminary results for Demona Benchmark exercise, sent by 7 working groups have been presented and discussed. This exercise will be completed in about five months.

3. Analysis of experiments on aerosol behaviour

Regarding the partecipation of JRC to the "blind" code comparison exercise organized in the frame of the LACE project a thermal hydraulic and aerosol pre-test analysis has been accomplished for tests LA2 and LA4. Calculations have been carried out with the use of CONTAIN code.
A partecipation to the "blind" post-test is also planned.

4. Development of new aerosol models

The CORRAL-2 code has been modified with the inclusion of the diffusiophoresis model, in the classical form, in order to reach a more realistic representation of aerosol retention in the containment and therefore to obtain results which are in better agreement with other codes. Already included in the code are modifications of the time evolution of particle sizes, empirically made on the basis of CSE experimental data. With these modifications the code is now called CORRADI (Ref. 3, 4).

*Analysis of unsolved problem areas*

Monitoring of the evolution of Source Term research needs and priorities has continued in the reporting period, in particular by participating in the CSNI-PWG 4 meeting in October 1985 and by attending a Workshop on Source Term organized by DG XII-D in Brussels in September 1985.

***Collaboration with external organizations***

Collaboration with GLAST (ENEA-DISP, ENEL, University of Pisa) is established for the comparative analysis of the Zion acccident sequences.

Collaboration with KFK (FRG), CEA-Fontenay-aux-Roses (F) and U.K.A.E.A. (U.K.) is carried out in the frame of the code sensitivity study organized by DG XII - Brussels.

Two study contracts with BATTELLE-Frankfurt are in the phase of execution, one dealing with state of the art on aerosol behaviour in the containment under severe accident conditions, the other with the same type of study on containment thermohydraulics.
Collaboration with EPRI is established for the JRC Participation in the LACE project.

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USSR

# A REVIEW OF EXPERIMENTAL WORK ON FAST REACTOR PHYSICS DURING 1985-1986

Zvonarev A.V., Matveenko I.P.

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## Introduction

In this paper a short review of work on fast reactor physics carried out from July 1985 to July 1986 is given.

In 1985-1986 successful operation of the BN-350 and BN-600 power reactors was continued. The BN-350 reactor is operated at 750 Mw(t) power producing 1200 kg/s of distillate and has electric power of 130 Mw. Fuel burn-up in standard subassemblies reaches 7.9% h.a. (in a high-enrichment zone). The BN-600 reactor is successfully operated at the design power level accumulating large and valuable operation experience which serves a basis for creation of more advanced and larger power units with fast reactors. In some fuel subassemblies of the BN-600 reactor a maximum burn- up of 7.55% h.a. as been achived. The BOR-60 experimental fast reactor is being successfully operated under nuclear power plant operating conditions for even more than 15 years. The reactor plant power has been uncreased up to 56 Mw. A large volume of investigations on experimental fuel elements with uranium-plutonium oxide fuel up to burn-up of 10-14% has been made, no fuel failures being detected. Operation of the BR-10 experimental reactor was continued in the research channels of which samples of fuel, absorbing and structural materials were irradiated. Nuclear-physical and medical-biological investigations are carried out on neutron beams.

Experimental studies on fast reactor physics carried out in the following traditional directions:

- studying of fast reactor mock-ups at the BFS critical facilities;
- investigation of neutron and physical characteristics at critical assemblies of simple composition and geometry to obtain nuclear data information;

- studies at power and experimental fast reactors for the purpose of refining neutron and physical characteristics of reactors being designed and to obtain data on neutron constants.

Fast Reactor Mock-Up Studies at the BFS-2 Critical facility

From 1985 at the BFS-2 facility a fast reactor mock-up with an axial interlayer of uranium metal 22 cm thick (BFS-50) is being studied. As was reported before, the first stage of these studies was begun in which the uranium core was being studied. Three versions of the assembly were successively realized: the BFS-50-1- an assembly with the unperturbed core and respectively minimum critical sizes, the BFS-50-2- with 13 dummy control rod steeves (sodium and steel) in the core; the BFS-50-3-18 dummy compensating rods half-inserted in the core were added.Efficiencies of single dummy rods and groups of them were studied, as well as fission-reaction rate distributions by small-size ionization chambers with uranium-235, uranium-238, plutonium-239 isotopes over assembly height and radius.

Preliminary comparison between experimental and calculated results has not revealed any marked discrepancy ( more than 5%) practically over the whole investigated reactivity range.

As to fission (reaction) rate distributions, for uranium-235 and plutonium-239 in the core centre an agreement is not worse than 2% and the discrepancy somewhat increases to the periphery (up to 5%). For uranium-238 some systematic underestimation (by several %) of calculated data for the core (with height) is observed, as well as discrepancies (of 7÷10%) in the central plane and near blankets.

A distortion of neutron field over the BFS-50-3 assembly height arizing due to asymmetrical insertion of absorbing sections of dummy compensating rods proved to be higher (by 4÷6%) than the calculated one.

At present there is made a transition to the BFS-50-4 assembly in which, in contrast to the previous assembly, an absorber in the dummy compensating rods is introduced for the total core height.

Experiments on Verification and Refinement of Group Constants

1. Measurements on a series of simple single-zone uranium-plutonium oxide-fuelled assemblies BFS-49-1, 49-2, 49-3 and 49-4 were carried out. Data on the $K_{eff}$ values, reaction rate ratios for main elements entering into the composition of the assemblies and on the ratios of the central coefficients of reactivity at successive addition of sodium, graphite and polyethylene were obtained. At the same assemblies a program of investigation of neutron absorption by fission products was performed. 17 of 30 most important nuclides-fission fragments were studied.

2. At the COBRA facility there were carried out investigations at the KBR-11 assembly (with the spectrum close to the BN-1600 reactor spectrum) which consisted of uranium and stainless steel.

Investigations were carried out at the KBR-12 assembly containing nickel-free steel. The results of these and earlier experiments showed that the mean neutron absorption by stainless steel was predicted in calculations using the BNAB-78 constants with an error of not more than $\pm 10\%$ both in hard and soft neutron-spectrum reactors.

The necessity of decreasing the mean capture cross-section of nickel in the BNAB-78 constants (at BN-1600 reactor spectrum) by about 15% was indicated.

3. Experiments on determination of removal cross-sections (below the uranium-238 fission threshold) for uranium, stainless steel and its main components- iron, nickel, chromium and molybdenum - were completed. Neutron transmissions of the $Cf^{252}$ point source placed inside spherical, samples of materials under investigation were measured. The spheric-electrode-type fission chamber was located close to the spherical samples outer surface.

An analysis of the data obtained has shown that the calculations by using the BNAB-78 constants predict the cross-sections of removal below the uranium-238 fission threshold for stainless steel with an error of more than experimental errors ($\sim 3\%$). For single steel components (iron, nickel) the discrepancies between calculated and experimental removal cross-sections go beyond the experimental errors.

## Studies at Power Reactors

At the BN-350 reactor a program of neutron and physical studies with a test zone of mixed uranium-plutonium fuelled subassemblies was completed.

The test zone was located in the low-enrichment zone at its boundary with the high-enrichment zone. Spatial distributions of fission and capture reaction rates for the most important fissionable isotopes were measured in the reactor before positioning a test zone into it and then the same experiments were carried out in the reactor with the test zone. In the fuel subassembly located in the centre of the test zone there were also measured the ratios of uranium-238, plutonium-239 fission and uranium-238 capture cross-sections to the uranium-235 fission cross-section. The same ratios were measured in the uranium fuel subassembly placed in the same cell of the reactor without the test zone.

Calculational analysis of the experiments was carried out using the three-dimensional code TRIGEX and the BNAB-78 data set.

The investigations carried out have shown that spatial distributions of reaction rates are predicted with the use of modern computation methods with errors of not more than 4% (an experimental error being $\pm 2\%$) even with such strong perturbations in the reactor as a test zone of mixed-fuel subassemblies. The ratios of main reaction rates determining the neutron balance in the reactor, $\sigma_c^8/\sigma_f^5$, $\sigma_f^8/\sigma_f^5$, $\sigma_f^9/\sigma_f^5$ in the core agree with the calculated values within the limits of experimental errors making up $3 \div 4\%$ both for uranium fuel subassemblies and for the central fuel subassembly of the mixed-fuel test zone.

After completion of the run the spent fuel elements of the central fuel subassembly of the test zone and the monitor-samples irradiated in it were subjected to radiochemical analyses using radiometric and mass-spectrographic methods.

From these investigations the ratios of $Pu^{239}$ capture and fission cross-sections and other breeding characteristics of the uranium-plutonium-fuelled zone were obtained, calculational analysis of data was carried out. Irradiations of structural and fissionable materials samples in the BN-350 and BN-600 reactors were continued to obtain data on their cross-sections at the reactor neutron spectra.